UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

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--PUBLIC MEETING--

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D I S C L A I M E R

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UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

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BRIEFING ON PROPOSED FINAL RULE ON

STATION BLACKOUT

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PUBLIC MEETING

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Nuclear Regulatory Commission
Room 1130
1717 H Street, N.W.
Washington, D.C.
March 31, 1988

The Commission met in open session, pursuant to notice, at 10:02 a.m., the Honorable LANDO W. ZECH, JR., Chairman of the Commission, presiding.

Commissioners Present:

LANDO W. ZECH, Chairman
THOMAS M. ROBERTS, Commissioner
FREDERICK M. BERNTHAL, Commissioner
KENNETH ROGERS, Commissioner
KENNETH M. CARR, Commissioner

Staff and presenters seated at table:

S. CHILK - SECY
V. STELLO - EDO
W. PARLER - OGC
T. SPEIS
W. MINNERS
A. SERKIZ
A THADANI
F. ROSA

Audience Speakers:

P. BARANOWSKY
A. RUBIN
R. BAER

PROCEEDINGS

CHAIRMAN ZECH: Good morning, ladies and gentlemen. Today the Commission will be briefed by our Offices of Research and Nuclear Reactor Regulations on a proposed rule to require that light-water reactors be capable of withstanding a station blackout for a specific period of time.

A station blackout is a total loss of both off-site power and on-site emergency AC power systems.

The proposed requirement is based on information developed under the Commission's study of unresolved safety issue A-44 of station blackout.

The Commission designated station blackout an unresolved safety issue in 1980, and studies were initiated to determine whether additional safety requirements were needed.

In March, 1986 a proposed rulemaking was published in the Federal Register. Based on the Staff analysis of comments received, the proposed rule has now been finalized and is ready to be issued subject to Commission approval.

During the Staff presentation, we would be interested in hearing about the revision to the regulatory backfit analysis and the associated

regulatory guide for a review standard for the acceptance of the specific blackout duration.

This is an information briefing. The Commission will not be voting on the final rule today. I understand that copies of the slides are available in the back of the room.

Do any of my fellow Commissioners wish to make any opening comments?

[No response.]

CHAIRMAN ZECH: If not, Mr. Stello, would you proceed, please.

MR. STELLO: Thank you, Mr. Chairman. In a moment I'll turn to Dr. Speis to begin the presentation and to introduce others here at the table with us today.

I thought I'd make a few points to begin with. Station blackout clearly has been with us for some time now, and it is an issue that has been recognized as a significant contributor to core risk. In fact, those plants which are the latest plants the Commission has done a fairly thorough analysis of risk presented in NUREG 1150 that the Commission is aware of, shows that even though the core melt frequencies are getting very low, the station blackout remains a dominant contributor to risk.

With that background, we are proposing that there are two approaches one can use to deal with station blackout. One approach is to show, depending on the sources of power and site specific issues, if you can cope with a station blackout for a specific period of time that's acceptable or, on the other hand, if you add additional sources of power to the site, that's also an acceptable approach.

What the Staff will be presenting is why we believe that's a correct way to go, and we are going to suggest to the Commission that we think because of the issue that is raised that station blackout, even when malfrequencies get to be low, remain a significant contributor to risk and there are other generic issues that could be resolved, in fact, if you had an additional source of power that the Staff suggest, then the Commission ought to say in its rule that those two options are there but it prefers that the solution of station blackout be by the addition of an additional power source to the site.

And we'll identify that in the briefing and have some particular words to suggest that Commission may wish to consider adding. I think it's an important consideration, and we'll get to it in a moment.

But that -- Dr. Speis, will you continue?

CHAIRMAN ZECH: All right. You may proceed. Thank you.

MR. SPEIS: Mr. Chairman, Commissioners, to my right I have Aleck Serkiz who has been the project manager of this issue who will participate in the discussion, questions and answers.

Next, Minners, he's from the Office of Resources, the Deputy Director of the Systems Division; and from the Office of Nuclear Reactor Regulation, we have Faust Rosa, Branch Chief of the Electric Systems Branch, and Mr. Thadani, the Assistant Director for Systems who'll give you the NRR presentation.

Let me start by giving you -- can I have the next viewgraph, please.

[Slide.]

MR. SPEIS: In this viewgraph, I give you, Mr. Chairman, the briefing that will give you the dates. We'll give you the briefing as Mr. Stello said, it's in two parts. The first part will deal with the development of the rule, the second part will address the implementation of the rule. That's the part that will be provided by the Office of Nuclear Reactor Regulation.

In my presentation, I will give you a summary and background of the issue, I will summarize the safety concerns associated with the blackout issue, I will summarize the findings that we have put together. We have been working on the this issue for the last three or four issues very extensively, as you know.

We'll discuss the proposed resolution. We will discuss the benefits of the alternate AC power source that Mr. Stello just mentioned to you which is the preferred way of going, and go into some more detail of the rule itself.

May I have the next viewgraph, please.

[Slide.]

MR. SPEIS: Here we provide a summary and some background of where we are coming from. As you said, Mr. Chairman, a station blackout was designated in a recent safety issue back in 1978. The issue has been studied extensively since that time.

The issue has its origin basically in operating experience and PRA studies, not only the NUREG 1150 studies, but mostly the studies that have been done have identified station blackout as a contributor to risk.

Operational experience has been an important

factor in pushing us in this direction that we'll be recommending to you today.

From 1968 to 1985 we had a large number of, something like more than 60 total loss of off-site power events of a few minutes duration up to some hours.

From 1976 to 1985, we had made hundreds of diesel generator failures during testing as well as actual demands.

And also from 1968 to 1985, we had the number of station blackouts precursors involving total loss of off-site power, most of them for a few minutes, power was able to be restored.

So there is enough operational experience that tells us that it's an important issue in addition to the PRA studies that have been mentioned.

The other thing that tells us that this issue is important is that it has severe consequences, potentially severe consequences. If you lose power, you have limited capability of removing core decay heat. And of course in most instances you lose completely containment decay heat removal.

We have found out that severe weather conditions are a major contributor to loss of

off-site power; such weather conditions as hurricanes, ice storms, tornados.

We have done extensive studies, as I said, involving the total population of nuclear power plants. These studies have been documented in NUREG 1032. From those studies, I have -- I'm sorry. I was jumping to the next viewgraph for some strange reason.

We had extensive interactions with the ACRS, with CRGR. The rule, as you said, Mr. Chairman, has gone out for public comment, and the appropriate comments have been incorporated.

We had extensive discussions with -- interactions with industry. NUMARC sponsored the working group composed of utility people as well as technical consultants. In fact they have put together a document which has been identified as NUMARC 8700 which provides the guidelines and technical basis for addressing plant capability to withstand station blackout.

We have reviewed this document, and we have found it acceptable with some exceptions which are noted in the Reg Guide which I will discuss shortly.

NUMARC has agreed with these exceptions and has commenced to conduct the workshops for utilities

to demonstrate the use of this document.

So it is our understanding that industry now agrees with the way we are going about to resolve this issue.

MR. STELLO: In fairness, I don't believe the industry is aware that we are suggesting the Commission add its preference. I don't believe they are aware of a particular issue before this meeting, at least that's my understanding. So the agreement does not include any preference that the Commission may have. They are not aware of it.

COMMISSIONER ROGERS: Are they in agreement with the Rule?

MR. STELLO: Everything except -- yes.

COMMISSIONER ROGERS: They are in agreement with the Rule?

MR. STELLO: Yes.

MR. SPEIS: With the two options. And of course one of the options is the alternate power source, and Mr. Stello is saying that is our preferred option.

So we are here today to recommend that you gentlemen approve the issuance of the final rule which would require all LWRs to be able to withstand a station blackout for a specified duration and

maintain core cooling during that period.

I will discuss later on what are the -- what is the specified rates and where it derives from.

Again the goal of this role is to reduce the frequencies of occurrence of core damage from station blackouts. We feel that if this rule is implemented it will reduce the contribution of station blackout to core dominance by at least a factor of ten on the average.

COMMISSIONER BERNTHAL: Let me ask a question about the fundamental assumption here with respect to risk from blackout.

Is that a deterministic number in the sense that you have not tried to carry out perhaps independently an emperical study to see whether that matches what might be a normally deterministic methodology, if you understand what I'm saying?

MR. SPEIS: I understand, yes.

COMMISSIONER BERNTHAL: There are enough events --

MR. SPEIS: Yes.

COMMISSIONER BERNTHAL: -- by now that --

MR. SPEIS: I think this issue, even without rates, we have enough statistics and we know enough about the consequences of station blackout to make us

probably propose the same thing.

But I think this is an ideal issue where we are able to compare the statistics and the deterministic analogies with safety goals, and more or less they've met both of them, okay, using a safety goal type of approach and using the experience, the operational experience, the deterministic analysis, consequence analysis, we're able to reach the same conclusions. So one re-inforces the other, basically.

So it's an ideal issue where the safety goal, in fact one will come in the future to discuss implementation of the safety goal, we'll give you this as an example of how one can go about implementing the safety goal.

COMMISSIONER BERNTHAL: So you've surveyed all of the station blackout events, or perhaps precursor station blackout events, and carried out a statical analysis of that --

MR. SPEIS: Yes.

COMMISSIONER BERNTHAL: -- independently of a calculation and deterministic procedure that of course we normally do for PRA?

MR. SPEIS: All of that is discussed in this NUREG that I mentioned earlier, 1037.

COMMISSIONER BERNTHAL: And the results are rather similar --

MR. SPEIS: Very similar.

COMMISSIONER BERNTHAL: -- and they both indicate that the severe core damage frequency, well, it says here 10 to the minus 4, 10 to the minus 6.

MR. SPEIS: Let's go to the next viewgraph so we'll address a little bit those numbers.

[Slide.]

MR. SPEIS: I'm sorry, I was racing ahead earlier in the core assembly information here, but let me -- I won't repeat the first one as Mr. Chairman mentioned what station blackout is all about.

Again the genesis of this issue is both the operational experience and the extensive PRAs. The potential is severe consequences involving limited decay heat removal as well as no containment heat removal; I mentioned the severe weather conditions.

Again from all the studies that we have done involving the total population of nuclear power plants, we have estimated the range of frequency of station blackout to be somewhere to be 10 to the minus 3 to 10 to the minus 5 per reactor year. That is station blackout.

Now the contribution of station blackout to the total core damage, has been estimated to be somewhere between 10 to the minus 4 and 10 to the minus 6.

And this is what I said earlier that the goal of the resolution is to reduce the frequency of station blackout contribution and hopefully this 10 to the minus 4 number will go to 10 to the minus 5, okay, so that is the goal.

COMMISSIONER BERNTHAL: I guess the thing that's bothering me a little bit is that you have a range of frequency of station blackout: Ten to minus 3, 10 to the minus 5. That means that for a population of a hundred reactors, we should only see one every ten years, right? And that seems like that's wildly out of sync with what we are really seeing.

MR. SPEIS: This is a complete station blackout. This is not loss of off-site power. Let's make sure, you know -- loss of off-site power in the average of the United States is something like .1 per year. This is a complete station blackout.

COMMISSIONER BERNTHAL: Okay. But you still say -- you believe that -- I mean in the most favorable case, the 10 to the minus 3 means that for

a population of 100, it should happen once every ten years. And you're saying that somewhere between once every ten years and once every thousand years --

MR. STELLO: Well, are you using 10 to the minus 3 as the measure of the sample in the industry? The range is 10 to the minus 3 to 10 to the minus 4 -- I mean 10 to the minus 5.

COMMISSIONER BERNTHAL: Yes.

MR. STELLO: The average of the plant is on the order of more like 10 to the minus 4.

COMMISSIONER BERNTHAL: Fine.

MR. STELLO: Okay. And with 100 plants --

COMMISSIONER BERNTHAL: Once every 100 years.

MR. STELLO: Once every 100, and if you look --

COMMISSIONER BERNTHAL: Which bolsters my point.

MR. SPEIS: We've had three or four so far in the total operational experience.

MR. STELLO: Which is consistent with that number.

MR. SPEIS: Which is 1500 to what, 1700 years?

MR. STELLO: About 1500 reactor years of

experience that you would have expected --

COMMISSIONER BERNTHAL: Worldwide?

MR. SPEIS: No. No. United States. United States.

COMMISSIONER BERNTHAL: United States?

MR. SPEIS: Yes.

MR. STELLO: So it's inconsistent with --

MR. SPEIS: It's consistent.

MR. STELLO: -- the numbers. The frequency range that you have, the rate is reasonably consistent with the experience that you've had.

COMMISSIONER BERNTHAL: How many actual station blackout events have we had?

MR. SPEIS: Okay. We have had four. I'm familiar with four of them. They are a few minutes. The longer one was the -- the other ones were a few minutes and we were able to restore -- I mean the plants were able to restore power.

COMMISSIONER BERNTHAL: So we've had four events in 1300 odd reactor years. That doesn't comport with these numbers. You're saying we have --

MR. STELLO: Yes.

COMMISSIONER BERNTHAL: -- four station blackout events.

MR. STELLO: Right.

COMMISSIONER BERNTHAL: 1300 operating reactor years.

MR. SPEIS: It's more than 13, it's 17.

MR. STELLO: About 1500.

COMMISSIONER BERNTHAL: Well, okay. 1500.

MR. STELLO: Right.

COMMISSIONER BERNTHAL: Let's say 1600, it makes it easy here. That means one in 400.

CHAIRMAN ZECH: You've got somebody here that wants to --

MR. BARANOWSKY: Could I clarify something. My name is Pat Baranowsky and I'm the author of the report that everybody is discussing the statistics on.

The 10 to the minus 3 number is based on blackouts of about a half-hour duration or longer. So all the ones that we've had are less than a half hour.

So when Themis says there have been four precursors which is what he said is correct, they're not exactly the kind of blackouts that we're talking about in terms of high risk because they have been less than a half hour.

COMMISSIONER BERNTHAL: So these four were all less than a half hour --

MR. BARANOWSKY: That's right.

COMMISSIONER BERNTHAL: -- and the numbers, 10 to the minus 3 to 10 to the minus 5 refers to station blackout longer that one half hour?

MR. BARANOWSKY: Right.

CHAIRMAN ZECH: We haven't had any longer than a half hour; is that right?

MR. BARANOWSKY: No full blackouts that I know of longer than a half hour.

CHAIRMAN ZECH: All right.

COMMISSIONER BERNTHAL: Okay.

CHAIRMAN ZECH: All right. Thank you. Go ahead, please.

MR. SPEIS: Consistent with what Pat said, the extended duration blackouts, that is more than two hours, are the ones that we are mostly concerned, Item No. 6 on the viewgraph.

In general, this is our assessment, you know, that most plants possibly are able to cope -- could be able to cope with blackouts for around two hours, but we need to -- some evaluations to confirm this.

At present there is no regulatory requirement for plants to cope with station blackout.

May I have next viewgraph, please.

[Slide.]

MR. SPEIS: Here we summarize some of the findings. The key point here is that everything is variable. The reliablity of on-site emergency AC power systems varies considerably. The factors here are the variety of plant designs and configurations, the emergency diesel generator reliability configuration.

What I mean by configuration, Mr. Chairman, is how many diesels you have and how many you need for decay heat. For example, some plants have three and they need two, some plants have three and they only need one. So that's an important factor. Common cause failures, design errors, human errors, and things of that sort.

Frequency and duration of off-site power also varies considerably. Site is an important consideration which of course is affected by the weather, the grid design configuration, and plant specific factors associated with the CCR design, transmission lines.

I've taken all of these things into account -- then core damage frequency that can vary considerably from plant to plant. As we said earlier, that can vary anywhere from 10 to the minus

4 to 10 to the minus 6 per reactor year which is sources of magnitude.

Important factors here are the susceptability to station blackout and of course the ability to withstand the loss of all AC power.

Here we're talking about specific plant attributes. For example, the ability of the reactor coolant pump seal to withstand station blackout, the capacity of water, electrical power, air systems, all these are important attributes that tell you about the capability of a plant to withstand station blackout.

Again our proposed resolution considers plant unique characteristics and provides a cost effective way of achieving a plant specific solution.

Our approach has been to look into this in a graded way and the plants that have less reliable power sources because of location or configuration, they'll have to do more or less again depending on how good or how bad they are.

So we don't want to come up with a blank regulatory requirement that we'll treat all plants equal. Plants that are better should do less; plants that need to do more, should do more. So that has been our approach. Okay.

The next viewgraph goes into some more details of the proposed resolution itself.

[Slide.]

MR. SPEIS: Again the final solution consists of a rule and a regulatory guide. We are proposing to amend 10 CRF 50 by adding Section 50.63, Loss of All Alternating Current Power, which requires that all plants be able to cope with station blackout for specified duration.

Again this is the graded approach. This specified duration, I will discuss it later on, but I want to say right now that it will be based on plant specific characteristics which affect the reliability of both the on-site and off-site power system.

An alternative AC power source is an acceptable option. As Mr. Stello said already, we prefer the alternate AC power source due to additional safety benefits.

Also as part of the resolution, we are proposing to issue Regulatory Guide 1.155 which provides general guidance of how to comply with the rule itself. The guidance addresses such things as severe weather categories, required levels of emergency diesel generator reliability, and other related assumptions.

Also it provides guidance on the station blackout analysis, how does one go about developing -- deciding the duration which depends on these off-site and on-site factors.

Also it provides guidance on the use of the alternate AC sources.

And also it --

COMMISSIONER ROBERTS: Pardon me.

MR. SPEIS: Yes, sir.

COMMISSIONER ROBERTS: What do you mean when you say QA considerations?

MR. SPEIS: What type of quality assurance has to be considered. This issue can be --

COMMISSIONER ROBERTS: What else can this be?

MR. SPEIS: Mr. Roberts, this issue is not treated as part of -- as a design basis accident because it is there for -- the requirements that we imposed on the additions to the plant itself to meet the rule are not the same as the stringent requirements that we apply for design basis accidents. And these are described in the Regulatory Guide itself. That's what we mean by -- for example, they don't have to be seismically qualified.

We think that losing power is more frequent

from other events than from a seismic event, for example; therefore, if they add something, it does not have been to be seismically qualified and that's what we mean by that as an example.

CHAIRMAN ZECH: What you're saying is that this rule is in the category that is considered above the design basis requirements?

MR. SPEIS: Yes, if we look carefully --

CHAIRMAN ZECH: It enhances safety beyond the design basis; is that correct?

MR. SPEIS: Yes. It's like ATWS. The ATWS category basically. We looked very carefully at the quality atrributes of systems that have to be added.

CHAIRMAN ZECH: I understand that, but it is above --

MR. SPEIS: To some extent, yes. It is somewhere in between.

CHAIRMAN ZECH: Somewhere in between what?

MR. SPEIS: The design basis and the non-design basis.

COMMISSIONER BERNTHAL: It's designed to allow you to cope with the design basis accident or avoid the design --

MR. SPEIS: Avoid --

COMMISSIONER BERNTHAL: -- well, that's not

the correct terminology either, but --

MR. SPEIS: I guess I'll need the help from the lawyers when it comes to defining --

COMMISSIONER BERNTHAL: It's a preventive measure.

MR. SPEIS: It is a preventive measure, yes, but it goes beyond the regulations to some extent.

CHAIRMAN ZECH: That's what I'm trying to clarify.

MR. SPEIS: Yes. Yes. Our regulations deal with reliability, you know, there is GDC-17, general design criteria 17, that talks about -- we have to have a power source, it has to be reliable and so on and so forth. It does not address coping and we find from experience, from the extensive experience that I mentioned earlier, that we can further reduce the risk from this issue by being able to cope with station blackout. Somewhat, it is beyond the design basis. Maybe Vic can --

MR. STELLO: Clearly this is an issue which we believe merits adding the additional safety we get by doing this. It is safety beyond that now contained in our regulations which set the design basis for the plants.

CHAIRMAN ZECH: Right. That's what I wanted

to clarify.

MR. STELLO: This clearly goes beyond the design basis as set forth.

CHAIRMAN ZECH: Yes. That's what I wanted to clarify. Thank you. Let's proceed.

COMMISSIONER BERNTHAL: Let me ask another statistical nit here that's bothering me. How did we get to 1500 reactor years in this country? We haven't had an average of 50 plants per 30 years, and we're only getting 100 year right now. Are we counting subs or something?

MR. STELLO: Do you have the number, Pat?

MR. BARANOWSKY: I guess I didn't hear the question.

MR. STELLO: What's the total reactor years of operating experience for U.S. reactors? Do you have that?

MR. BARANOWSKY: It's over 1000. I don't think it's 1500. I don't know what it is right at this minute.

COMMISSIONER BERNTHAL: It is over 1000?

MR. BARANOWSKY: Yes.

COMMISSIONER BERNTHAL: Okay.

MR. STELLO: If you'd like, we'll give you --

COMMISSIONER BERNTHAL: It's not the sum. That's a small point. It can't be 1500, though.

CHAIRMAN ZECH: I'm well aware that it's over 1000. I've got it in one of my statistics books that I keep. I don't recall the exact number either, but I know it's over 1000 hours of commercial nuclear power operations -- years of nuclear power operation in our country.

COMMISSIONER BERNTHAL: I guess I can believe 1000; I can't believe 1500.

MR. SPEIS: Next viewgraph, please.

[Slide]

MR. SPEIS: We have listed here some of the additional benefits of the alternate AC power. We say it provides a means to cope with the reactor coolant seal failure which therefore this additional power source -- or this alternate AC power source can be used to power independent pump seal cooling systems.

At present, we have made the assumption that as part of the resolution of A-44 that the seal does not fail, but there is a degradation of some sort, and the maximum leakage is 25 GPM. And based on that, there is no problem.

So the assumption that is made then is that

the seal is not going to fail. But we have a separate issue dealing with the seal issue, and if from that issue we find out that the seal indeed fails, given a station blackout, then that leakage could go anywhere from 60 to 400 or so GPM. In that case, they'll have to prove that their seals -- plant specific analysis will have to be done to show that the seal does not leak or some independent system has to be provided to be able to cool the seal itself and that's why we are saying there is a benefit by going with the alternate AC sources at this point this time. It will take care of the seal issue so that we won't have to argue with that issue later on.

Also it simplifies operator actions needed to cope with station blackout. Basically you go directly into the alternate source so you don't have to undertake activities that involve the loss of power itself.

Also it alleviates environmental concerns associated with station blackout. For example, overheating of electrical equipment and control room habitability.

So we're recommending then that we add to Section 50.63, Section 2, C.2, the thing that is in the parenthesis there. If the potential for common

mode failures can be minimized, use of an alternate AC source is a preferred option since this approach will also benefit other safety concerns. The next viewgraph slide.

[Slide.]

MR. SPEIS: The rule itself again -- its licensed LWR plant must be able to withstand for a specified duration and recover from a station blackout. Item 2, which is very crucial which goes into the graded approach that I mentioned earlier, Mr. Chairman, where the duration itself will be based on plant specific characteristics as well as location of the plant itself.

These are the four important factors that the duration will be based on, the redundancy of the on-site emergency AC power resources -- it's how many you have and how many you need -- the reliability of on-site AC power sources, the expected frequency of the loss of off-site power, and the other one is the probable time to restore off-site power.

The use of alternate AC power sources is an option, we mentioned that already. And of course the Reg Guide provides the guidance for complying with the rule itself.

This brings my presentation to an end, and

Mr. Thadani now can continue from NRR to discuss the implementation and the review priorities.

MR. THADANI: Good morning.

CHAIRMAN ZECH: Good morning. Please proceed.

MR. THADANI: Thank you. If the station blackout grew, as is structured now it would require the licensee to make an information submittal in about nine months following the issuance of the rule.

[Slide.]

MR. THADANI: The content of the submittals would be expected to include the duration of that plant as well as justification of that category; proposed modifications, if any, to meet the requirements of the rule, and the schedule for implementation of the modifications.

We expect submittals covering over 100 plants, so it's clear that it's important that we prioritize our activities and focus on those plants which deserve early attention.

And our basis for looking at plants would be relative safety significance. The top priority will be given to plants which, we believe, were most susceptible to station blackout events.

We have information from Office of Research

which has identified we believe approximately 17 units which belong in this category, and we would pay early attention to those units.

We would also screen early on the licensee submittals to identify, determine if there are other plant units that need early attention.

We'd use other other factors in assigning priorities as Mr. Stello and Dr. Speis have mentioned that if proposals come in with alternate AC power source, we expect to assign higher priorities for two reasons: Number one, not only would that approach resolve the station blackout issue, but it would also likely resolve some of the other issues in some of the operating reactors.

So we would assign high priority for that reason, plus our review, I expect, would be minimal in those cases.

For the remaining plants we will include consideration of residual risk. For example, we would expect to pay higher attention to plants with Mark-1 and ice condenser containments over plants which have large dry containments.

Our focus is going to be what's most important in terms of -- again in safety we can achieve. Let's work on those plants first and go on

down.

Mr. Rosa is going to discuss what we're going to review, the content of our review as well as the schedules to these reviews, and I think it would become a little clear to you why it is important for us to prioritize our activities. And those are the kinds of thoughts we'd utilize in prioritizing our activities.

CHAIRMAN ZECH: All right.

COMMISSIONER BERNTHAL: Let's see. The NUREG 1150 results -- just refresh my memory here -- did show a station blackout to be the dominant risk for the ice condensers --

MR. STELLO: BWR.

COMMISSIONER BERNTHAL: -- and the Mark-1's.

MR. STELLO: No, the Grand Gulf and the Peach Bottom, I think we were in the 90s. One was 95 and the other 80 something.

COMMISSIONER BERNTHAL: And the ice condensers as well?

MR. STELLO: I don't remember.

MR. THADANI: The ice condenser was very significant.

COMMISSIONER BERNTHAL: That's what I thought.

MR. STELLO: That's the ignitor problem.

COMMISSIONER CARR: I have a little trouble trying to figure out why you would put a high priority on the ones who have alternative AC source proposals since it looks like they would be solving the problem and you could leave them until later and go to the guys who might not have the problem solved.

MR. THADANI: In fact, those licensees who proposed alternate AC power source, we'd like to take a quick look and make sure we're satisfied with the proposal so they can go ahead and implement and make the necessary improvements early on.

So that's really the motivation: Not to hold it back, not to delay implementation.

COMMISSIONER CARR: Okay.

CHAIRMAN ZECH: All right.

MR. THADANI: Okay. Mr. Rosa will discuss --

CHAIRMAN ZECH: Thank you very much. You may proceed.

MR. ROSA: Next slide, please.

[Slide.]

MR. ROSA: The Regulatory Guide 1.155 describes the means acceptable to the Staff for achieving conformance with the rule. And the Staff

review will simply ascertain that by review of the applicants' submittals, that the guidelines of the Reg Guide have been implemented and thereby achieving the attainment of requirements of the rules.

The review will focus on those aspects of the requirements that are deemed most important for verifying conformance with the rule, and the first is the determination of the proposed minimum acceptable station blackout duration. The review will make sure that the characteristics of the off-site and on-site power systems and diesel generator reliability have been adequately considered in arriving at the minimum acceptable station blackout duration.

The next important element in assessing conformance with the rule is the station blackout coping capability that will be described in the submittals. We expect that those plants that elect to provide coping capability analyses will do so in some detail and the review will verify that we are in agreement with the assumptions and the results of the analyses.

COMMISSIONER BERNTHAL: Are you going to make a suggestion as to what you think would be a minimum acceptable coping time under any circumstance? I mean let's just say good engineering

judgment or common sense. Would the number be four hours or two hours or one hour or what would it be?

MR. ROSA: Well, the guidance has a category of two hours for those plants that are most capable of sustaining a station blackout. They're least susceptible to a station blackout. I would expect that that would be a minimum.

MR. STELLO: I don't understand that, but let me clarify now before we get too far. I thought that if someone met all the requirements for alternate AC they did not have to show any coping capability, it was not required.

The answer is zero if you add additional power supply that eliminates the blackout as a consideration that you need not show any coping capability; am I wrong?

MR. ROSA: No, you're not wrong. An alternate AC source that can be started within an hour is acceptable, but coping capability has to be demonstrated for that one hour.

If the alternate AC source provided can be started and brought into play from a shutdown in ten minutes, then no coping analysis is required. That's what the guidance states.

COMMISSIONER BERNTHAL: I guess the thing

that bothers me a little bit about that philosophy is that -- well, let me ask a question. Have you looked at the comparison of what the world standard is these days -- compared this with the world standard to determine whether in fact you then will be going beyond the world standard in terms of alternate capability? Because -- I'm using world standard generically. The French, for example, I believe, require something like 20 hours of coping capability. The Germans I believe require something like eight hours if I remember correctly. I'm not certain about these numbers any more.

But generally the Europeans have required, all other things aside, I believe, a fairly extended period for coping, quote unquote, but I don't know, quite frankly, whether the kind of analysis that you are proposing, redundancy really, whether that sort of comparison has been carried out. Maybe you could comment on that.

In other words, have they not paid the attention that you intend to pay to the redundancy in a plant with alternate sources?

MR. ROSA: I believe that -- considering the differences in off-site power reliability and frequency of loss of off-site power that exists

between, let's say, the Europeans and ourselves, that rule is adequate for meeting the U.S. requirements in regard to station blackout.

Now the French, I believe, in their Palo Alto reactor do provide coping capability in the order of 20 hours.

MR. SPEIS: I will like to say something because basically the French approach is not different from our approach. They have provided cooling, direct cooling to the seals.

They have found from their analysis that the Achilles heel is the coping domain, okay, and therefore they provide an independent power source and that is our recommendation, too.

To compare coping times, whether they are 15 or 4 hours, you know, these are not very easily done. For example, if you don't consider equipment qualification or if you lose power for ten hours, somebody has to make sure that the equipment is operable that are needed for the duration of the coping, and I don't think those analyses have been done to really prove that one can cope for eight hours or 16 hours.

Therefore, in fact, that is the reason, even though they say that, that is the reason in their

last analysis they go and put an independent power source to cool the seals. So in that regard, you know, our proposal is not different than theirs.

CHAIRMAN ZECH: We have a comment from --

MR. RUBIN: Yes, I'd like to amplify a little bit. My name is Alan Rubin, task manager of the U.S. Site 44. Having visited the Palo Alto site and reviewed the French experience, there is some other reasons why the French and other countries have gone beyond what we are proposing in this resolution.

One important part is the stability of the grid and the frequency of initiating events of losses of off-site power.

I believe the French have about a factor of two higher frequencies of total losses of off-site power than we have here, and that's a direct factor in terms of the frequencies of core damage.

COMMISSIONER BERNTHAL: A factor of two, yes, that's not a terribly impressive difference. What about Germany where I think it's eight hours? Isn't it six or eight hours?

MR. RUBIN: I haven't seen the data for the frequencies for German losses of off-site power, but in Sweden I know they have had some large losses which have affected a significant portion of the

country, I'd say about half the country and they got particularly concerned because of the north-south transmission lines in that country and they have additional redundancy in terms of diesels and gas turbines at that site because of the transmission line situation.

COMMISSIONER BERNTHAL: And you don't know about Germany.

MR. RUBIN: I don't know about the frequencies of losses, but I know they have additional capability in terms of redundancy, diversity in power supplies.

MR. SPEIS: The Germans rely on diesels basically. They don't have any extra cooling of the seals. They don't think it's a concern because of the large number of diesels they have basically.

The English for their size, well, they rely on seal cooling just similar to our proposal, and the French, so I think in general, you know, we're not inconsistent with --

COMMISSIONER BERNTHAL: Well, the reason I make a point of it -- and we should go on here, but that's the first question that you and this Commission is going to be asked, it seems to me, because whether we're playing on the same level field

or not -- and it sounds like what you're saying is that we're not -- at least the word that's around is the French have 20 hours and the Germans have eight hours and the Swedes, I don't know how many hours they have, I don't know what the Japanese have either, and it just seems to me that we better be prepared to explain why what we are doing is the functional equivalent of that. I think that's what you're telling us.

MR. SPEIS: I think, yes. And we have looked very carefully at what the grid is, what the Germans and what the Swedes and what the French have done, and I think we're satisfied that we are, if I use your words, functional equivalent.

COMMISSIONER BERNTHAL: Okay. Just so that we've got that nailed down --

MR. SPEIS: Yes.

COMMISSIONER BERNTHAL: -- because I think it's important.

MR. MINNERS: A small clarification. We were just talking about two hour coping capability. I think most plants will probably have a four hour capability because the industry has indicated that they will make a commitment to make changes to the design and put everybody in a four hour category.

The rule allows or requires some plants -- or allows some plants to have a 16 hour coping capability depending on those -- if they are unusual plants. We don't expect to find many or any of those, but there's a possibility that some plant would have to have a 16 hour capability.

CHAIRMAN ZECH: All right. Can we proceed?

MR. ROSA: The next area of concentrated review effort will deal with the modifications, potential modifications that would be proposed in ICT submittals.

These areas include alternate AC source additions, reactor coolant pump seal failures, features to prevent reactor coolant pump seal failure, battery capacity, addition of or adequacy of existing batteries, condensate storage capacity.

Now the review will attempt to ascertain that any modification may in fact do -- provide the intended enhancement in coping capabilities and it will also verify that whatever modifications are made, do not adversely impact existing safety related systems.

An additional element of the review will address what procedures in training are being provided for station blackout.

It is expected that regional audits may be performed to look at the procedures that are in place and the training that is being conducted.

Finally, the question of operability requirements for station blackout equipment, how these are defined and implemented.

The question of possible imposition of technical specifications has been discussed. These requirements could also be contained in administrative procedures.

If technical specifications are decided on, I believe they will be minimal in addressing perhaps at best the alternate AC source operability requirements.

In any event, whatever is decided on in regard to technical specifications will conform to the Commission's interim policy statement on technical specifications. Next slide, please.

[Slide.]

MR. ROSA: The schedule shown on this slide assumes that the station blackout rule will be issued on June 1st of this year. It takes into account the 270 days allowed for industry response, and it assigns what we consider to be a reasonable allocation of Staff resources to this task given the

other workload that has to be performed and the amount of technical assistance funding that is made available for this task.

The rule states that 30 days after the notification to a licensee that their proposed fix for the station blackout issue is acceptable, that they should provide a firm schedule for implementation which should not exceed two years unless some very firm justification for extending that beyond two years is provided.

So the completion dates or implementation dates shown on there fall two years and one month following the Staff evaluation completion for the particular sites.

The reviews have been based on site reviews rather than unit reviews because in a site review, a unit -- a two unit site with essentially similar plants would not require two reviews. So it's shown there in terms of site reviews.

The first 24 highest priority sites, the review would be completed by November the 1st of '89; the next 35 sites evaluation would be completed by October 1st of '90; and the final 16 remaining sites evaluation will be completed on March 1st of '91.

If things go according to plan, all sites

should have implemented the station blackout rule by March 31st of '93.

I might say one other thing about the review process. We have interacted with the utility working group that produced NUMARC 8700, their initiatives document, and have obtained an agreement in most areas.

We believe that this will result in a standardized licensee submittal in both format and content which will ease the task of the Staff in reviewing it.

I think that's a plus for both the regulatory process and the industry.

CHAIRMAN ZECH: All right.

MR. STELLO: We're through, Mr. Chairman.

CHAIRMAN ZECH: All right. Thank you very much. Questions, my fellow Commissioners? Commissioner Roberts?

COMMISSIONER ROBERTS: March the 8th, we sent you a bunch of questions and you responded. I think I read it last night. Response to Question 4: "Plants with very short required coping times may be able to disband seal failure without core recovery." Qualify that. What is short?

MR. SPEIS: Several hours, Mr. Roberts.

COMMISSIONER ROBERTS: Several hours?

MR. SPEIS: Yes. A few hours, less than four.

CHAIRMAN ZECH: Do you want to come to the microphone, please. Identify yourself for the reporter.

MR. BAER: Yes, I'm Robert Baer. It's a complicated issue. If all the seals --

CHAIRMAN ZECH: Identify your --

MR. BAER: Oh, I'm Chief of the engineering and issues branch in Research.

CHAIRMAN ZECH: Thank you very much.

MR. BAER: This generic issue 23 is assigned to my branch. If all the seals -- if the seals fail completely on a given pump, leakage could be as great as 480 GPM.

And with four pumps, it could withstand only about an hour or so before the core is uncovered. If there was no other -- you know, if you don't cool the seals.

So it could be for plants under an hour, you could probably expect complete seal failure. But for longer duration coping times, the seals have to hold at least to some degree.

CHAIRMAN ZECH: All right. Thank you.

Anything else, Commissioner Roberts?

COMMISSIONER ROBERTS: No.

CHAIRMAN ZECH: Commissioner Carr?

COMMISSIONER CARR: No.

CHAIRMAN ZECH: Commissioner Rogers?

COMMISSIONER ROGERS: Well, yes. This timetable, the 3-31-93, full implementation date, is that set by the availability of NRC Staff resources to review licensee proposals?

MR. ROSA: I would say that to some extent it is. We have considerable other workload. Events are occurring day by day that require Staff effort in reviewing and resolving, and we have devoted what we consider to be a reasonable amount of Staff resources for this task.

COMMISSIONER ROGERS: Well, suppose a licensee wants to go faster and get this thing out of the way and behind them, are they limited by your identification of the highest priority sites?

Suppose somebody, in your opinion, has a low priority site, wants to get it behind them and get it off their books by moving more rapidly, will they be prevented from doing so because of the Staff limitations?

MR. STELLO: No. We encourage it. And if

all of them want to do that, we'll find a way to get them done.

COMMISSIONER ROGERS: All right. And it seems to me that that's a rather long time to wait to get this issue totally bundled up and put to bed, and I would think, anyway, if we could move more rapidly, it would be very desirable and I would hope that the limitation just isn't simply our own Staff's resources and inability to review proposals.

MR. STELLO: Let me say what I have said a number of times in the past, that I think the industry ought to seize on this particular issue as an opportunity to solve not only the station blackout issue, but if they go about doing this carefully, they can integrate and solve a number of other issues that -- the pump seal being one, decay heat removal being another issue. There are other issues that are out there and I think they could go a long way in getting rid of when they look at the plant from their own perspective, so I'd hope that the schedule allows sufficient time for them to be able to take a pretty good look at being able to integrate a whole base of solutions into the plant.

COMMISSIONER ROGERS: Well, I'd just like to say that I really didn't have as much time as I would

like to really study the report, but everything I've seen of it seems to indicate that it's an un usually capable and fine piece of work, and tough issue, thorny issue, and the detail and thoroughness with which the Staff approached this problem, I felt to be very impressive. To try to get your arms around it and get it in some kind of a shape for dealing with what I thought was really an impressive piece of work.

I want to compliment you on it, everyone who participated in it.

COMMISSIONER CARR: Can I follow up on that a little bit? Does the utility have to get our permission to add on-site power sources? I don't see any reason they should. They might have to get our permission to hook it into our system, but I would think they can go all out and if they want to put in a gas generator or something, they could put it on site certainly without us telling them they couldn't.

MR. STELLO: But the application of bringing that onto the site to dealing with the compliance of this rule --

COMMISSIONER CARR: Well, as I say, for that they may have to have our permission to hook it up. They could go a long ways to getting the problem

solved before 1993, I'd think.

MR. STELLO: They could do that now.

COMMISSIONER CARR: Okay.

COMMISSIONER BERNTHAL: Yes, I share the concern, though, that Commissioner Rogers raises. It is an awfully long timetable and we've had one or two cases of plants that were waiting for start up where they have taken fairly rapid steps on their own, bring in gas turbines or whatever it might be.

You know, you guys -- we have to worry about hooking it up, as Commissioner Carr says, but I would hope that we don't get distracted by global solutions here when we have one rather important item that as you point out and as that page shows represents 90 percent, 90 odd percent in -- well, one assume in many of the boilers.

So that by taking care of that single item, in effect I guess you dropped core melt probability a factor of ten. I think that's pretty important. We ought to get at it.

MR. THADANI: Yes. In fact, if I may just make a comment, if the utilities were to come and propose some alternate AC power source, I'm firmly convinced that our review process is minimal in that regard, and it gets down to the point of hooking it

up, essentially. And if that were the proposal, then I don't expect the Staff would need to do much of a review and therefore the schedule would in fact be inappropriate as you see it.

The schedule is developed on the basis of a substantial amount of analyses and reviews and back and forth, if you will, but nevertheless the focus of our review will be to address plants where we believe this issue is very important as early and as quickly as possible, and that's what we mean by dividing it up in terms of number of sites that we have -- we intend to address early on. So we are very sensitive to the point you make, Commissioner Rogers.

COMMISSIONER ROGERS: I will hope that if this whole thing is approved and goes through and goes into action, that we be would be kept informed of how this implementation schedule is actually working in practice and that I would hope to see some modification of it once the process begins, that maybe some new things would come such as initiatives from the industry itself, to move more quickly and rapidly to get this thing behind them.

I would think that everybody would like to see this thing cleaned up and out of the way because it's been hanging around for so long and it looks as

if you've got all the elements here to move fairly quickly and rapidly to clean it up.

MR. STELLO: We'll do our best to improve the schedule and we will keep the Commission informed if it goes forward with the rule on progress.

CHAIRMAN ZECH: All right. Commissioner Bernthal?

COMMISSIONER BERNTHAL: Yes, I had a couple of other things I wanted to ask about, and then a suggestion.

I'm not sure I quite understood. If we can go back to one of your slides here, I guess it's No. 5, I'm not sure. You have this additional recommendation which has, I believe, appeared between the time you sent the package and now.

If I missed the point when you explained it then you'll have to re-explain it, but I didn't quite understand why you have reached a technical judgment that an alternate AC source is to be preferred over whatever the other options might be. Why is that?

MR. SPEIS: Well, we reached the point, I guess, in the last few months or maybe few weeks that we looked at the resolution of the seal issue, and if we are not able to ascertain the integrity of the seal, then it's possible that one of the requirements

will be to put a power source to provide cooling to the seal itself. So we are bringing this to the attention -- we are bringing this right now on this issue because it's an integral part of this issue. Independent power sources provide it, it results for the cooling of the seal, the reactor coolant pump seal at the same time, so --

COMMISSIONER BERNTHAL: I'm not sure I -- you're going to have to do a little more better. I'm not sure I follow you.

MR. STELLO: Let me give you a few more reasons. Look at the 80 percent core melt frequency -- 90 percent core melt frequency. Those plants will probably meet this rule without doing anything. Okay?

COMMISSIONER BERNTHAL: Yes.

MR. STELLO: So you are going to wind up with still dominating. If you add an alternate power source, you're going to get a factor ten reduction in those plants.

Remember, they have been re-analyzing in the 10 to the minus 6 range now.

COMMISSIONER BERNTHAL: Yes.

MR. STELLO: Excuse me, down two. Second, there are the -- a lot of the intangibles that you

get as indirect benefits of having that alternate source especially if it's diverse. You don't have to worry about any common mode failures, contamination of fuel supplies, common mode failures due to maintenance or whatever, especially with the diversification.

In addition, events in the plant such as floods, fires, the kinds of things that again become intangible that can affect a lot more equipment.

You add substance -- dimention to the defense in depth that you have in the plant by having the diverse alternate power source.

We aren't saying you're required to do, but everything we know suggests that you have through it the mechanism to solve a whole host of other technical issues and further substantially improve the safety of the plant.

COMMISSIONER BERNTHAL: Okay. What, for example, qualifies, just for my information, as an alternate AC source? Does that mean, for example, that you bring in a gas turbine or --

MR. STELLO: A gas turbine, another diesel generator.

COMMISSIONER BERNTHAL: Okay.

COMMISSIONER CARR: Let me ask you if I

understood what you said. It sounds to me like what you're saying is that they may qualify adding additional power source under the blackout rule, but when we solve the pump seal problem we may require it anyway.

MR. STELLO: And that's why I said I hope to make the statement that when the industry goes out and they look at all the other things that are out there on the horizon as they understand it, things they'd like to do with this plant.

This may be a desirable thing for them to do. They wouldn't have to deal with the coping analysis at all if they did have an alternate source.

COMMISSIONER BERNTHAL: Okay. I want to get back just a little bit to the minimum, although I guess the minimum is zero hours of coping capability.

You've been giving an explanation of why we in the public should be willing to accept what on the surface at least appears to be a different -- appears to be a lesser standard for coping capability than -- it seems to be the mode now in Europe these days, and I would just suggest that there needs to be put together fairly coherent understandable comparison and explanation of your arguments for why we are the functional equivalent, if you will, if indeed we are,

to the requirements that Europe is placing on its plants, and included in that kind of argument and justification it seems to me should be the consideration of the differences of off-site power reliability, for example, as you've mentioned, but also included in it should be the fact that weather conditions in this country, particularly in the winter, very often can be much more severe than in France and Germany, for example. You don't believe that?

MR. STELLO: Down in the Alps?

COMMISSIONER BERNTHAL: Oh, come on, Vic. Most of France and Germany is not the Alps.

MR. STELLO: But you've a lot of interconnecting grid systems that go over the mountains and there are problems.

COMMISSIONER CARR: I remember last winter when they froze up the river, they lost quite a few of their nuclear plants because the intakes were frozen.

MR. STELLO: Let me -- I see where you're going and I'm a little bit troubled because I don't know that we even have enough information or if it's wise for us to try to get into studying their off-site power supplies, the weather problems, the

kinds of things that we have generated an awful lot of data and had access to a lot of data in the United States to really be able to go through that.

COMMISSIONER BERNTHAL: What I'm saying is that it seems to me that one of the principal causes of long-term loss of off-site power, at least in this country, will be weather-related. You don't have to live in the Midwest very long or the Northeast or the vast land mass of the United States of America to understand that's far more likely in this country than in the more benign climates of Europe. I mean, Europe is simply a more milder climate.

MR. MINNERS: But I don't think the statistics show that. I'm no expert on this, and correct me if I'm wrong, but I think the French have a higher rate of loss of off-site power --

COMMISSIONER BERNTHAL: But that may be for different reasons. Is that true?

MR. MINNERS: But I guess the point I'm trying to make, you have to look at the overall loss.

COMMISSIONER BERNTHAL: No, I agree. And I agree the reliabilty of the system is on the one hand, but on the other hand are these factors that are more severe in our country.

All that I'm saying is that those are the

questions that are going to be asked and we better be prepared to answer them.

MR. STELLO: Well, I could tell you this. If they're asked anywhere in the foreseeable future, we're not going to be able to answer and in order to answer them, we're going to have to initiate a very, very large program.

We do not have the kind of information -- you would need to know the stability of the grid systems in each of those countries and that's going to take a great deal of doing. We don't have that information.

COMMISSIONER BERNTHAL: We have no statistics on loss of off-site?

MR. STELLO: We could go try to get them, but we don't have them readily available, and we don't have very much information on the details of what they've done for their coping analysis. All we do know is that the French started out that way. When they did, they decided they had to go back and add additional equipment in the plant for the coping with seals, I don't know what else they did. I don't really know that we have a fairly good understanding of what the Germans did. We'll have to go back to them and collect a lot of information.

We'll do that if the Commission wants us to, but we don't have it now and it will take a substantial effort for us to do it.

MR. SPEIS: I think it is fair to say that the Germans and the Japanese -- we're not aware that they have done any coping now because they depend on this reliability.

You know, the Japanese, they overhaul their diesels and every month or every six months, whatever, and they want to make sure they operate -- I don't think they have done any --

COMMISSIONER BERNTHAL: I thought the Germans had an eight hour coping standard or objective?

COMMISSIONER CARR: I'm not sure all that data would be of any value other than just a discussion or an argument over who's doing it better than somebody else is.

What we are really saying is that we don't have a problem, we're trying to make them safer in the long run than safe and we are trying to get down another factor of ten, not in the reliability of power at the site, but we're trying to get down another factor of ten in the contribution to a core melt problem.

And I'm not sure it's worth going out on a major exercise to find out a lot of data just for argument sake.

COMMISSIONER BERNTHAL: Well, I think it's worth being able to give the public fairly convincing, straightforward, simple answers to the question of why can we permit in principle, at least zero hours when other countries are requiring rather long coping periods and -- I'm not convinced that we don't have the data at hand to make a coherent --

COMMISSIONER CARR: I would think the French citizens would be more on the other side arguing why do they need 20 hours of coping capability when we get by with only six?

COMMISSIONER BERNTHAL: It's all public perception that --

COMMISSIONER ROGERS: Mr. Chairman, if I could --

MR. STELLO: Excuse me. If I may say something. It's disturbing to me that there's a suggestion that if someone comes in and adds additional power supplies to the on-site system, a diverse turbine generator, that somehow in any way is less desirable or it's degree of goodness, if you will, is somehow less than a plant that maybe has 10

hours, 20 hours of coping capability. I'd certainly like never to have a station blackout is the better objective.

COMMISSIONER BERNTHAL: I'm not disagreeing. If you are prepared to say that, that's what I'm saying.

MR. STELLO: That's exactly what we're saying, is our preferred course is go to zero coping capability because you will not have a blackout. You will need none, zero analysis to show coping because you're not going to have a blackout.

COMMISSIONER BERNTHAL: Look, what I am saying is that I think you have the data and the arguments in hand to present to the public a coherent explanation, a simple explanation of why we don't have to do -- that's what you're telling us, we don't have to have coping time because --

MR. STELLO: Oh, that we can do because that's internal to us.

COMMISSIONER BERNTHAL: Now, you have already mentioned today the lower reliability of the European grid systems -- I don't know the statistics. I don't know whether that's true, but you've mentioned it and it seems to me that another element of consideration here on the other side of the

argument may well be the severe weather conditions that we're always subject or often subject to in this country.

We do after all deal with tornadoes. Other countries don't, for example. So all that I am saying is better put it together and be prepared to deal with it, and I think it's worth doing that so the public understands. That's all that I'm trying to say.

COMMISSIONER ROGERS: Well, if you simply present it in such a way that focuses on the advantages of the alternative power source, just what that does, that that reduces the time minimum that's required to establish a coping policy, then you can put those other numbers -- the other countries in some perspective and not have to use them as the starting point for your argument, you know, it's just background, and then you don't really need to do a detailed study because you're not going to base your reason on those data but on an alternate approach.

I'd like to just raise a little question about the wording of some of the pieces of paper here because it seems to me that it could be a little confusing in that the proposed resolution is referred to as one which requires that all plants be able to

cope with station blackout for a specified duration.

That's not what it is. It's that each plant be able to cope with station blackout for a specified duration.

And there isn't a single number that applies to all plants, and I see the public notice enclosure, one, uses those same words to require that commercial nuclear plants be capable of withstanding a total loss of AC power for a specified time.

That's not correct. That's strictly not correct. It's that each plant has its own time. And I would suggest that you carefully review the wording that goes out on this so that you're not subject to some criticism of saying one thing and doing something else that's different because it suggests that it's a single number, and there isn't a single number, there's a single approach, but there's not a single number.

MR. SPEIS: Al here brings to my attention in another paragraph it stated the way you're saying it so we'll better make sure it's consistent.

COMMISSIONER ROGERS: Yes. Right.

MR. SPEIS: Okay.

CHAIRMAN ZECH: Commissioner Bernthal, any other questions?

COMMISSIONER BERNTHAL: I had one other point I wanted to raise and I see there are industry representatives here so perhaps if we are not prepared to explain ourselves, the industry would like to explain, though, why what we are doing is good enough or is better, perhaps, than what the rest of the world is doing.

The last question on the backfit analysis. I have not had a chance to go through that element of your presentation here and it hasn't really been mentioned much here, but since this rulemaking and step here does go beyond adequacy, I take it we will have to base this rule on some sort of backfit analysis, cost benefit analysis, will we not?

The last time around on this issue, I thought it was a pretty shaky proposition, and I don't remember exactly what the numbers were. I think it came down to a factor of two and I'd hate to argue the errors on a number like that. Are we prepared to provide the arguments we need here or are we going to rely on qualitative arguments?

MR. SPEIS: Well, we have done the arithmetic to the best of our ability. I think the latest numbers that we came up with was that the value impact ratio showed numbers like two and a half

thousand person rems per million dollars compared to the thousand dollars that we use as a criteria. And that is of course with no on-site costs. If you use on-site costs, that number goes to six thousand.

Now the thing I said earlier, Dr. Bernthal, that we feel it meets the backfit rule and of course this is in the category of safety improvements and it does have to meet the criteria of the backfit rule.

COMMISSIONER BERNTHAL: What's the cost benefit ratio that you will argue now and what are your uncertainties?

MR. SPEIS: Well, I guess -- we discussed them in the report, I don't remember them, but, you know, that was our best estimate. That number can be anywhere from 500 to 5,000.

COMMISSIONER BERNTHAL: I think it was a factor of two. Is it still a factor of two or something like that?

MR. RUBIN: It's in the report. Alan Rubin again. I authored the regulatory analysis and on Table 12 of the regulatory analysis there's a summary of the cost benefit ratio. The best estimate is $2,400 -- excuse me. 2,400 person rem per million dollars. The high and low estimates, the high estimate is 5,000 person rem per million dollars, the

low estimate is 700 person rem per million dollars.

And the details of how those numbers were developed are included in NUREG 1109.

MR. SPEIS: With no on-site costs, right?

MR. RUBIN: That is without on-site costs. The number that somebody mentioned earlier of including on-site costs would double or triple the value of that ratio.

MR. MINNERS: It's 6100 with on-site.

MR. RUBIN: With on-site, that's right.

COMMISSIONER BERNTHAL: Is the Commission going to rely -- I'm asking the General Counsel now -- would the Commission rely on that numerical analysis or would it rely rather on a qualitative argument in this case?

MR. PARLER: Well, I'll answer it. I think the Commission always in matters such as this will rely on the best information that it has.

If cost benefit analysis in a situation like this were viewed as not favorable and not comparative -- not, relatively speaking, favorable with something else such as the costs standard, it is my judgment that they could and should rely on both including the qualitative judgment of the Commission that under the circumstances having gotten and

received the best analysis that they could, that either it is a right thing to do or it is not a right thing to do.

If under these circumstances they conclude that it's the right thing to do, I have a very high accomplished level, that if the backfit rule -- if the rule is challenged on the basis that it doesn't comply with the backfit rule that the rule will survive on that challenge. So it should rely on both.

COMMISSIONER BERNTHAL: Okay. Thank you.

CHAIRMAN ZECH: All right. Well, let me just make a couple of points.

[Commissioner Bernthal left the room.]

CHAIRMAN ZECH: Well, first of all, I'm informed regarding reactor years in our country that as of the end February of 1988, the 106 units that we had on line at that time have accumulated 1,076 reactor years of operation. And if you add the reactor years of operation of those plants that had been operating but are not permanently or indefinitely shutdown down, that's 94 more reactor years, so a total at the end of February cumulative in our country would be 1,170 reactor years of operation. That's just for the record.

Well, first of all, let me just say this. We are discussing a very, very important issue that's been before the Agency for a long time as we emphasized earlier this morning. I think the Staff frankly has done a very fine job in their analysis both by the research people and the NRR people and those who have supported them.

We're talking really about, as I understand it, beyond adequacy. It's an increasing -- enhancing safety really beyond a design basis which is I think very appropriate. You've told us that it does meet the backfit analysis.

[Commissioner Bernthal returned to the room.]

CHAIRMAN ZECH: I do think regarding Commissioner Bernthal's reference to other countries that it is worth looking into. I would think you'd perhaps have the information available and if you don't have, I would suggest that we could probably get sufficient information from our association with the European countries and I would suggest that you do that.

I think it is worth at least being as complete as we can to address that particular subject recognizing there are differences in the approaches

and so forth. It seems to me that's worth doing.

The alternate source approach -- I guess my question would be on that, why have you concluded that it would be acceptable for a utility to accept an alternate source, for example, an extra diesel generator or gas turbine and not perform a coping analysis so in order to demonstrate that there would be a high probability of handling a station blackout in a reasonable period of time?

Why have you -- as I understand it, that's what you've suggested. I'd just like to know the rationale for that.

MR. SERKIZ: Mr. Chairman, the reason -- the way --

CHAIRMAN ZECH: Maybe somebody ought to --

MR. STELLO: Warren.

CHAIRMAN ZECH: We appreciate you're trying, anyway. Maybe you could colleague -- and write him a note and he can try to explain it for you.

MR. MINNERS: Well, I think the basis of it is that the alternate AC source is going to be connected up -- be able to be connected up to the normal electrical distribution system. So you're not doing anything unusual and we are going to -- I guess maybe in lieu of a coping analysis, we will also

require licensees to perform a test. So once the equipment is installed, they are going to have to show that you can connect it all up and start it up and operate the equipment. So we are really having an on-site test pretty much in lieu of a coping analysis.

CHAIRMAN ZECH: All right. Go ahead.

MR. STELLO: Let me try adding a dimension to that. And if I remember right, a diesel generator, gas turbine, you can get 9.99 reliability and you'd have a potential there for a station blackout, whatever the number is, X, so you can add this independent power source. The potential of ever having that problem can then be reduced by one to two orders of magnitude.

Such you'd never have, you reduce it to a residual risk so low as the likelihood of ever having a station blackout that you don't need to do anything.

On the other hand, if you don't have that power source and you are dealing with just the overall probability of happening of X, then you need to add a dimension called coping or another way to say it is it allows you to be to deal with a station blackout for a length of time to provide you with the

time it takes to either have the off-site power restored or perhaps to repair or modify whatever you need to do to one of the diesels and get them started.

And typically the numbers, if someone would help me, if you can extend the loss or cope with a loss of off-site power for X hours, what's X, you can get about a factor of three or four improvement in having it restored.

What's the likelihood? Does nobody remember the number?

MR. BARANOWSKY: Typically if you went from say two to four hours, you get about a factor of three; in four to eight, about a factor of three reduction. Roughly, it varies.

MR. STELLO: That's the likelihood of having an off-site power restored.

CHAIRMAN ZECH: All right.

MR. STELLO: So it clearly is desirable to have that source on-site.

CHAIRMAN ZECH: All right. Commissioner Rogers, I think you --

COMMISSIONER ROGERS: Well, yes, on that, I wonder if you could tell me how it would work if a licensee installed, let's say, a diesel generator or

gas turbine or something of that sort, for emergency power, without doing a coping analysis and then for some reason after some period of time while this thing has been in place and sitting there and some routine test is performed on it, it's found that it has to be torn down and something has got to be fixed.

What condition does that put the plant in then? Is that a condition for shutting the plant down? Is it a condition for -- is it a condition -- what status is it in until that correction is made, particularly in the absence of a coping analysis? If there were a coping analysis that said well, you know, this plant is okay for eight hours or something of that sort and that's all the length of time it would take them to fix that alternative power source, then they're all right. But if it exceeds that, then what kind of a situation would they be in?

And if it wasn't a coping analysis, aren't they operating in a kind of a never never land under a circumstance such as that?

MR. STELLO: Well, if they put the alternate source of power on, I would assume include it in the technical specifications and allow some reasonable down-time and for repair.

If they exceed it, then, yes, they'd have to shut down if they are able to repair it within that period of time, just as they have for the on-site diesels today. Exactly the same philosophy.

CHAIRMAN ZECH: Let me emphasize one other point, too. If we step back and see what we're really trying to do in station blackout, of course, as we know, it's a loss of power on-site as well as off-site and the main thing that concerns all of us would be to keep that core covered. That means we need water, and in order to keep the core covered, we need power and that's really fundamentally what we're talking about and that's why it's such an important issue.

Whenever we have a real potential station blackout, the first thing I think about is how is the power situation because you got to get water to cover the core. And I know you -- this is what it's all about and so it really is an important issue and I think our efforts to enhance the safety in this regard is absolutely the right thing to do.

I, too, though, would like to make a reference to the schedule. I certainly hope we could speed up that schedule. I recognize there's many things involved that need to be analyzed and so

forth, but I would certainly encourage the utilities to take the initiative to speed up that schedule. I think it's important.

If I were a utility executive, I would certainly want to make sure that my facility would adopt these enhanced safety measures and so that in the middle of the night if I were a utility executive and they called me, I would not have to think about a shortage of power. I'd like to think I had extra power. I think if I were a utility executive, I'd put in not only an extra diesel engine but maybe two of them or maybe three. To me it's well worth the assurance that you're going to have power.

So I think we are doing the right thing and I hope the utilities will -- I think it's in their own best interest to add this enhanced safety to their plants.

So, I would also like to say to my fellow Commissioners that we have heard the briefing now. We all had a chance to think about this for a considerable period of time. We reflect on what we've heard and make any comments we may have to the Staff and perhaps we can move forward with this very important rulemaking issue.

Are there any other comments?

[No response.]

CHAIRMAN ZECH: All right. With that we'll stand adjourned. Thank you for a very fine briefing.

[Whereupon at 11:33 a.m., the meeting was adjourned.]

CERTIFICATE OF TRANSCRIBER

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

TITLE OF MEETING: Proposed Final Rule on Station Blackout

PLACE OF MEETING: Washington, D.C.

DATE OF MEETING: Thursday, March 31, 1988

were transcribed by me. I further certify that said transcription is accurate and complete, to the best of my ability, and that the transcript is a true and accurate record of the foregoing events.

MARIO A. RODRIGUEZ

Ann Riley & Associates, Ltd.

STATION BLACKOUT

(UNRESOLVED SAFETY ISSUE A-44)

COMMISSION BRIEFING
MARCH 31, 1988

THEMIS P. SPEIS
WARREN MINNERS
ALECK SERKIZ

OFFICE OF NUCLEAR REGULATORY RESEARCH

ASHOK THADANI
FAUST ROSA

OFFICE OF NUCLEAR REACTOR REGULATION

BRIEFING OUTLINE

DEVELOPMENT OF RULE:

IMPLEMENTATION OF RULE:

SUMMARY AND RECOMMENDATION

°	COMMISSION DECISION TO ISSUE PROPOSED STATION BLACKOUT RULE FOR COMMENT	MARCH 5, 1986
°	PUBLIC COMMENTS CONSIDERED AND INCORPORATED AS APPROPRIATE	
°	ACRS AND CRGR REVIEWS	JUNE 9, 1987 JUNE 23, 1987
°	DISCUSSION WITH NUMARC'S WORKING GROUP ON STATION BLACKOUT (NUGSBO). INDUSTRY NOW AGREES WITH PROPOSED RESOLUTION	SPRING - FALL 1987
°	STAFF RECOMMENDS THAT COMMISSION ISSUE FINAL RULE	

USI A-44 SAFETY CONCERNS

1. STATION BLACKOUT (SBO) MEANS COMPLETE LOSS OF AC POWER TO ESSENTIAL AND NON-ESSENTIAL BUSES (LOSS OF OFFSITE POWER AND THE UNAVAILABILITY OF THE ONSITE EMERGENCY AC POWER SYSTEM).

2. LIMITED DECAY HEAT REMOVAL (DHR) AND NO CONTAINMENT HEAT REMOVAL (CHR) W/O AC POWER.

3. SEVERE WEATHER CONDITIONS (E.G., HURRICANES, ICE STORMS, TORNADOES) ARE MAJOR CONTRIBUTORS TO LOSS-OF-OFFSITE POWER (LOOP).

4. ESTIMATED RANGE OF FREQUENCY OF SBO IS 1E-3 TO 1E-5/RX-YR.

5. ESTIMATED RANGE OF CONTRIBUTION OF SBO TO CORE DAMAGE FREQUENCY (CDF) IS 1E-4 TO 1E-6/RX-YR.

6. EXTENDED DURATION SBOS (> 2 HRS) CAN BE SIGNIFICANT CONTRIBUTORS TO RISK.

PRESENTLY THERE IS NO REGULATORY REQUIREMENT FOR PLANTS TO COPE WITH STATION BLACKOUT.

SUMMARY OF STAFF FINDINGS
(NUREGS-1032 & -1109)

RELIABILITY OF ONSITE EMERGENCY AC POWER SYSTEMS VARIES CONSIDERABLY

- PLANT DESIGN AND CONFIGURATION
- EMERGENCY DIESEL GENERATOR RELIABILITY AND CONFIGURATION
- COMMON CAUSE FAILURES

FREQUENCY AND DURATION OF OFFSITE POWER LOSS VARY CONSIDERABLY

- SITE CHARACTERISTICS (WEATHER, GRID)
- PLANT FACTORS (SWITCHYARD DESIGN, TRANSMISSION LINES)

CORE DAMAGE FREQUENCY CAN VARY CONSIDERABLY FROM PLANT TO PLANT

- SUSCEPTIBILITY TO STATION BLACKOUT (SBO)
- ABILITY TO COPE WITH LOSS OF ALL AC POWER

PROPOSED RESOLUTION CONSIDERS PLANT UNIQUE CHARACTERISTICS AND PROVIDES A COST EFFECTIVE WAY OF ACHIEVING A PLANT SPECIFIC SOLUTION

PROPOSED RESOLUTION

1) AMEND 10CFR50 BY ADDING SECTION 50.63, "LOSS OF ALL ALTERNATING CURRENT POWER," WHICH REQUIRES THAT ALL PLANTS BE ABLE TO COPE WITH STATION BLACKOUT (SBO) FOR A SPECIFIED DURATION. AN ALTERNATE AC SOURCE IS AN ACCEPTABLE OPTION.

2) THE STAFF PREFERS THE ALTERNATE AC SOURCE OPTION DUE TO ADDITIONAL SAFETY BENEFITS

3) ISSUE REG 1.155, "STATION BLACKOUT" WHICH PROVIDES:

   - GUIDANCE FOR SEVERE WEATHER CATEGORIES, REQUIRED LEVELS OF EDG RELIABILITY AND OTHER RELATED ASSUMPTIONS.

   - GUIDANCE FOR SBO ANALYSES, PROCEDURES, AND TRAINING FOR COPING WITH SBO, AND Q/A CONSIDERATIONS RELATED TO ALTERNATE AC SOURCES.

   - GUIDANCE ON THE USE OF ALTERNATE AC SOURCES.

   - GUIDANCE ON EDG RELIABILITY MONITORING AND RELIABILITY PROGRAM.

ADDITIONAL BENEFITS OF ALTERNATE AC

1) PROVIDES A MEANS TO COPE WITH RCP SEAL FAILURE (GSI-23); CAN BE USED TO POWER AN INDEPENDENT PUMP SEAL COOLING SYSTEM.

2) SIMPLIFIES OPERATOR ACTIONS NEEDED TO COPE WITH SBO.

3) ALLEVIATES ENVIRONMENTAL CONCERNS ASSOCIATED WITH SBO (E.G., OVERHEATING OF ELECTRICAL EQUIPMENT AND CONTROL ROOM HABITABILITY).

ADDITIONAL RECOMMENDATION

THE STAFF RECOMMENDS ADDING THE FOLLOWING SENTENCE TO 50.63, SECTION C.2:

> "IF THE POTENTIAL FOR COMMON MODE FAILURES CAN BE MINIMIZED, USE OF AN ALTERNATE AC SOURCE IS A PREFERRED OPTION SINCE THIS APPROACH WILL ALSO BENEFIT OTHER SAFETY CONCERNS."

50.63 REQUIREMENTS

LOSS OF ALL ALTERNATING CURRENT POWER

1. EACH LICENSED LWR PLANT MUST BE ABLE TO WITHSTAND FOR A SPECIFIED DURATION AND RECOVER FROM A STATION BLACKOUT.

2. STATION BLACKOUT DURATION SHALL BE BASED ON:

   (I) REDUNDANCY OF ONSITE EMERGENCY AC POWER SOURCES.
   (II) RELIABILITY OF ONSITE EMERGENCY AC POWER SOURCES.
   (III) EXPECTED FREQUENCY OF LOSS OF OFFSITE POWER.
   (IV) PROBABLE TIME TO RESTORE OFFSITE POWER.

3. USE OF ALTERNATE AC POWER SOURCES IS AN OPTION PROVIDED CONDITIONS STATED IN THE RULE ARE MET.

4. RG 1.155 PROVIDES GUIDANCE FOR COMPLYING WITH THE RULE.

REVIEW PRIORITIES

BASED ON RELATIVE SAFETY SIGNIFICANCE

- SUSCEPTIBILITY TO STATION BLACKOUT (~17 UNITS)

OTHER FACTORS

- ALTERNATE AC SOURCE PROPOSALS
- MARK 1 AND ICE CONDENSER CONTAINMENTS

SBO REVIEW PROCESS

THE STAFF REVIEW WILL ASCERTAIN THAT THE GUIDELINES OF RG 1.155 HAVE BEEN IMPLEMENTED. THE REVIEW WILL FOCUS ON:

- o THE DETERMINATION OF THE PROPOSED MINIMUM ACCEPTABLE SBO DURATION

  OFFSITE AC POWER CHARACTERISTICS
  ONSITE EMERGENCY AC POWER CHARACTERISTICS
  EMERGENCY DIESEL GENERATOR RELIABILITY

- o SBO COPING CAPABILITY

  DECAY HEAT REMOVAL
  EQUIPMENT ENVIRONMENT INCLUDING CONTROL ROOM

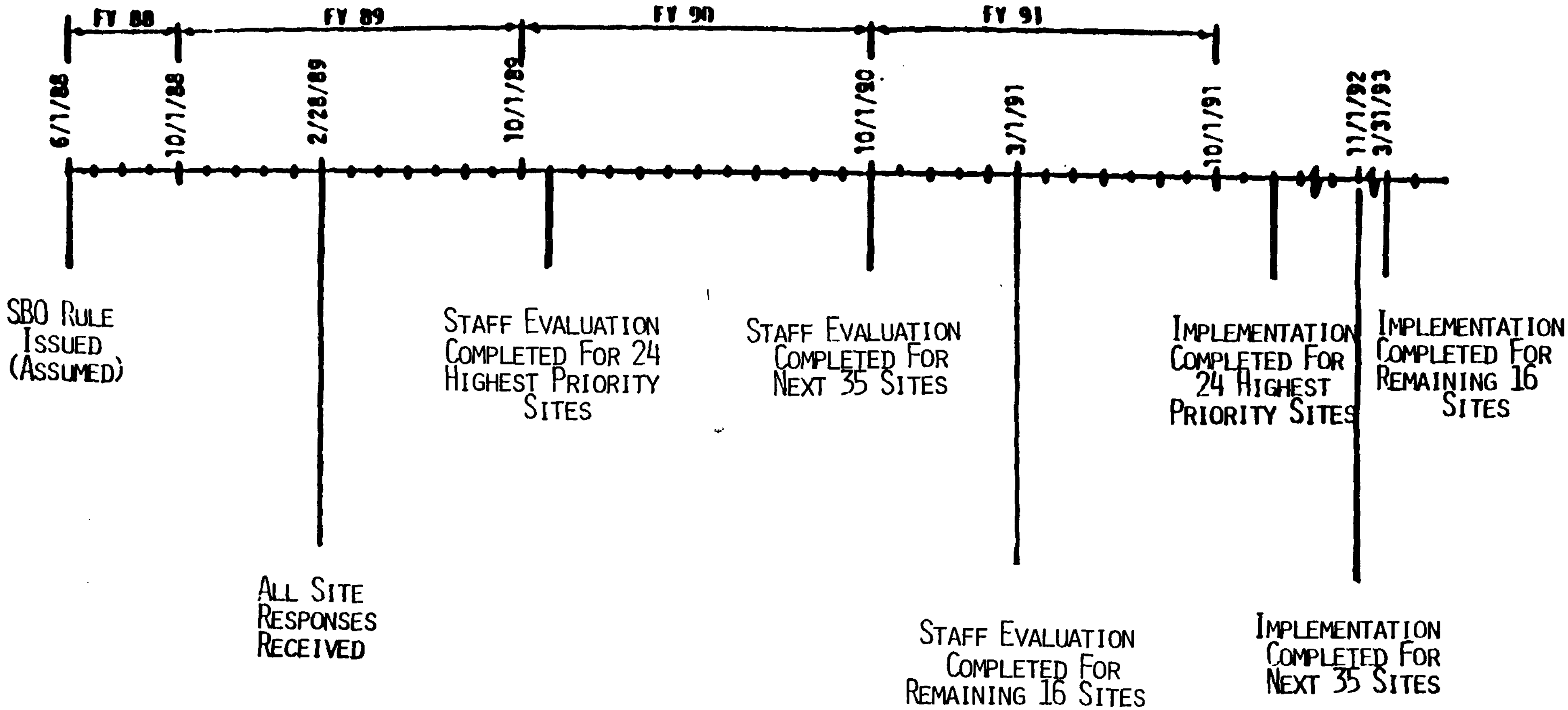
- o POTENTIAL MODIFICATIONS

  ALTERNATE AC SOURCE
  REACTOR COOLANT PUMP SEAL FAILURE
  BATTERY CAPACITY
  CONDENSATE STORAGE CAPACITY

- o PROCEDURES AND TRAINING FOR SBO

- o OPERABILITY REQUIREMENTS FOR SBO EQUIPMENT

NRR SBO REVIEW AND IMPLEMENTATION SCHEDULE



RULEMAKING ISSUE
(Affirmation)

January 21, 1988 SECY-88-22

For: The Commissioners

From: Victor Stello, Jr.
Executive Director for Operations

Subject: FINAL STATION BLACKOUT RULE, USI A-44

Purpose: To obtain Commission approval for publication of a Notice of Final Rulemaking on the subject of Station Blackout.

Background: On March 21, 1986, a Notice of Proposed Rulemaking on the subject of Station Blackout was published in the Federal Register. That Notice (Enclosure A) invited public comments regarding a proposed Station Blackout Rule (and the associated draft Regulatory Guide), which would require that light-water-cooled nuclear power plants be capable of withstanding a loss of offsite and onsite emergency ac power for a specified duration and maintaining reactor core cooling during that period.

The specified duration would be determined for each plant based on a comparison of the individual plant design with factors that have been identified as the main contributors to risk of core melt resulting from station blackout. These factors are: (1) the redundancy of onsite emergency ac power sources, (2) the reliability of the onsite emergency ac power sources, (3) the frequency of loss of offsite power and (4) the probable time needed to restore offsite power. The Notice of Proposed Rulemaking proposed amending the regulations by adding a new §50.63 and by adding a new final paragraph to General Design Criterion 17, Appendix A of 10CFR Part 50.

Discussion: In response to the Notice of Proposed Rulemaking, 53 letters commenting on the proposed rule were received. Forty-five of these were from the nuclear industry, comprised of electric utilities, consortiums of electric utilities, vendors, a trade association and an architect/engineering firm. Other letters were submitted by the Union of Concerned

CONTACT:
Warren Minners, RES, (492-3510)
Aleck Serkiz, RES, (492-3555)

Scientists (UCS), the Department of Nuclear Safety of the State of Illinois (IDNS), a representative of the Professional Reactor Operator Society, a citizens group, a consultant, and three individuals. UCS, IDNS and the citizens group supported the Commission's objective in the proposed rule, but did not believe it went far enough to reduce the possibility of a serious accident that could be initiated by a station blackout. Largely, the industry comments were opposed to generic rulemaking to resolve the station blackout issue. The Nuclear Management and Resources Council (NUMARC) with the support of 39 industry letters submitted, along with its comments on the proposed rule, a set of industry initiatives that it believed would resolve this issue without rulemaking.

The staff has considered the comments received and has prepared a draft Federal Register Notice of Final Rulemaking (Enclosure B). The Supplementary Information Section of the Notice includes discussion of the comments received and their disposition in the final rule. The significant changes from the proposed rule are: (1) the requirement for licensees to determine their plant's maximum station blackout coping capability has been deleted (coping for an acceptable duration, as specified in the final rule, would provide adequate protection), (2) the concept of an "alternate ac source" will be accepted as demonstration of adequate station blackout coping capability, and (3) the withdrawal of the modification to GDC 17; instead, the previously proposed change to GDC 17 has been incorporated into §50.63. The proposed modification to GDC 17 has been deleted so that it is clear that station blackout would not be considered with other events usually associated with design basis events, such as, single failure, QA (Appendix B), EQ (50.49), and seismic design.

As part of the public comment process, the Commissioners specifically requested comments on four issues when the proposed rule was published. They are:

1. Quality Classification of Modifications
2. Whether the Backfit Analysis Adequately Implements the Backfit Rule.
3. Cost-Benefit and Whether §50.63 Meets the "Substantial Increase in the Overall Protection of the Public Health and Safety."
4. Whether NRC Should Require Substantial Improvements in Safety that Go Beyond Those Proposed in this Rulemaking.

These issues are the first four subjects discussed under "Comments on the Proposed Rule" in Enclosure B.

The Regulatory/Backfit Analysis for the Resolution of Unresolved Safety Issue (USI) A-44, Station Blackout (NUREG-1109) has been revised somewhat to reflect the final rule and information received during the public comment period. It is provided as Enclosure C. Regulatory Guide 1.155 is provided as Enclosure D and presents a method acceptable to the staff to comply with the rule. The staff's technical findings for USI A-44, Evaluation of Station Blackout at Nuclear Power Plants (NUREG-1032) are provided as Enclosure E.

The ACRS reviewed the staff's proposed final resolution for the station blackout and issued their recommendations on June 9, 1987 (Enclosure F). The ACRS recommended that a rule not be issued, but that the staff continue to work with NUMARC on the technical aspects of industry's initiatives. ACRS further recommended that the staff proceed with issuance of the final rule if by September, 1987 the staff determines that the NUMARC initiatives will not be effective or timely in reducing the risk from station blackout. The staff met with the ACRS on November 5, 1987 and briefed the Committee on the results of working with the NUMARC/NUGSBO group, and the development of NUMARC-8700. NUMARC also briefed the Committee on their expanded initiatives and NUMARC-8700. The ACRS Chairman indicated to the staff that this resolution was acceptable and that there was no need for another ACRS letter.

The Committee to Review Generic Requirements (CRGR) considered the station blackout issue at meeting No. 115 on May 27, 1987. In the minutes of that meeting dated June 23, 1987 (Enclosure G), the CRGR recommended to the EDO that the proposed resolution for USI A-44 be approved; subject to a number of revisions recommended by the Committee. The staff is in agreement with the revisions recommended by the Committee, and has modified the station blackout package accordingly, with the exception of Discussion Item No. 4. The staff believes that this recommendation to have a review standard for the acceptance of the specified blackout duration has been suitably addressed in the standards incorporated into R.G. 1.155.

By letter dated November 23, 1987, NUMARC submitted a report (NUMARC-8700, Guidelines and Technical Bases for NUMARC Initiatives Addressing Station Blackout at Light Water Reactors, Enclosure J) which provides guidance and methodologies which can be used by the licensees for implementing the NUMARC initiatives. This report has been reviewed by the staff and discussed with representatives of

NUMARC at a number of meetings. Consequently, the staff has referenced NUMARC-8700 in Regulatory Guide 1.155 as providing guidance acceptable to the staff for meeting the requirements of the final rule. Regulatory Guide 1.155 provides a cross-reference to NUMARC-8700 and notes where the regulatory guide takes precedence.

USI A-44 is related to other generic issues. These include Generic Issue B-56, Diesel Generator Reliability; USI A-45, Shutdown Decay Heat Removal Requirements; Generic Issue 23, Reactor Coolant Pump Seal Failures; and Generic Issue 128, Electrical Power Reliability, which includes Generic Issue A-30, Adequacy of Safety-Related DC Power Supplies, Generic Issue 48, LCO for Class IE Vital Instrument Buses in Operating Reactors, and Generic Issue 49, Interlocks and LCOs for Redundant Class IE Tie Breakers. The Regulatory/Backfit Analysis (NUREG-1109) discusses these relationships further.

The resolution of Generic Issue B-56 is of particular relevance to the resolution of USI A-44 because of the importance of diesel generator reliability in station blackout. Regulatory Guide 1.155 identifies a minimum diesel generator reliability level of 0.95 and the need for implementing a diesel generator reliability program to maintain this reliability level, or any higher level that may have been used in the determination of the coping duration required by the station blackout rule. In cases where sites do not meet the minimum reliability guidance provided in Regulatory Guide 1.155, the required coping duration must be further justified or increased to the next highest duration level. Regulatory Guide 1.155 also identifies the major elements of the diesel generator reliability program which should be implemented. The resolution of Generic Issue B-56 will provide more explicit guidance for this reliability program which will be included in revisions of Regulatory Guide 1.108, "Periodic Testing of Diesel Generators Used as Onsite Electric Power Systems at Nuclear Power Plants," and the appropriate sections of the Standard Review Plan and Standard Technical Specifications. Because Regulatory Guide 1.155 already includes basic guidance regarding minimum diesel generator reliability and the measures to be taken to achieve the necessary levels, the staff has concluded that the resolution of USI A-44 need not be delayed pending final resolution of Generic Issue B-56. The final resolution of B-56, which is scheduled for FY88, will provide further assurance that the diesel generator reliability will be maintained equal to or above the levels selected for determination of the required plant coping duration.

It should be noted, based on all evidence that staff has on hand, that no undue risk exists with, or without, the promulgation of the station blackout (SBO) rule. However, SBO may still remain an important contributor to residual risk. This SBO rule will enhance safety by accident prevention and thereby reduce the likelihood of a core damage accident being caused by a station blackout occurrence. This does not mean however, that further enhancements in reducing the overall residual risk are not achievable by additional improvements in severe accident management, given the assumption that core damage occurs, whether from SBO sequences or other causes (such as small or large LOCA sequences). Initiatives that provide such safety enhancements (through improvements of core damage management procedures) are currently being pursued apart from the SBO rule. Therefore, this rule should be viewed as being in the same accident prevention context as the ATWS rule (§50.62) and the fire protection rule (§50.48). Such rules recognize multiple failure possibilities resulting from common cause effects that should be addressed. This concern has been recognized in the Introduction to Appendix A of 10CFR50.

Although USI A-44 will be resolved with the publication of the final rule, the related generic issues should not be considered resolved as a consequence of this action. These related generic issues are being coordinated with the resolution of A-44. Any additional requirements or guidance contained in the resolutions of these generic issues will be consistent with the requirements of this station blackout rule and is not expected to cause licensees to revise analyses, procedures or equipment that were changed to comply with the station blackout rule. The corresponding Standard Review Plan sections and the Temporary Instructions for inspectors will be issued prior to implementing the rule and Regulatory Guide.

Recommendation: We recommend that the Commission:

1. Approve publication of the Federal Register Notice of Final Rulemaking provided as Enclosure B;

2. Note that NUREG-1109, Regulatory/Backfit Analysis for the Resolution of Unresolved Safety Issue A-44, Station Blackout (Enclosure C) will be placed in the Public Document Room.

3. Note that the NRC will issue Regulatory Guide 1.155 on Station Blackout (Enclosure D) and NUREG-1032, Evaluation of Station Blackout Accidents at Nuclear Power Plants (Enclosure E).

4. Certify, pursuant to the requirements of the Regulatory Flexibility Act, 5 U.S.C. 605(b), that this rule will not have a significant economic impact on a substantial number of small entities;

5. Note that neither an Environmental Assessment nor an Environmental Assessment in support of a Finding of No Significant Impact are required for this rulemaking action under the categorical exclusion provisions of §51.22(c)(3)(ii and iii) involving recordkeeping and reporting requirements;

6. Note that the Office of Management and Budget has reviewed and approved the information collection requirements subject to the Paperwork Reduction Act of 1980;

7. Note that appropriate Congressional Committees will be informed by means of letters similar to that in Enclosure H;

8. Note that the Public Notice shown in Enclosure I will be released.

OGC has no legal objection to the final package.

Victor Stello, Jr.
Executive Director
for Operations

Enclosures:
See next page

Enclosures:
A Federal Register Notice of Proposed Rulemaking
B Proposed Federal Register Notice of Final Rulemaking
C NUREG-1109, Regulatory/Backfit Analysis for the Resolution of Unresolved Safety Issue A-44, Station Blackout
D Regulatory Guide on Station Blackout (RG 1.155)
E NUREG-1032, Evaluation of Station Blackout Accidents at Nuclear Power Plants, Technical Findings Related to Unresolved Safety Issue A-44
F Letter from W. Kerr to L. W. Zech, Jr. dated June 9, 1987
G Letter from E. L. Jordan to V. Stello, Jr. dated June 23, 1987
H Draft Congressional Letter
I Public Notice
J NUMARC-8700, Guidelines and Technical Bases for NUMARC Initiatives Addressing Station Blackout at Light Water Reactors, November 20, 1987

Commissioners' comments or consent should be provided directly to the Office of the Secretary by c.o.b. Friday, February 12, 1988.

Commission Staff Office comments, if any, should be submitted to the Commissioners NLT Wednesday, February 3, 1988, with an information copy to the Office of the Secretary. If the paper is of such a nature that it requires additional time for analytical review and comment, the Commissioners and the Secretariat should be apprised of when comments may be expected.

This paper is tentatively scheduled for affirmation at an Open Meeting during the Week of February 15, 1988. Please refer to the appropriate Weekly Commission Schedule, when published, for a specific date and time.

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ENCLOSURE A

FEDERAL REGISTER NOTICE

[7590-01]

Enclosure A

NUCLEAR REGULATORY COMMISSION

10 CFR Part 50

Station Blackout

AGENCY: Nuclear Regulatory Commission.

ACTION: Proposed rule.

SUMMARY: The Nuclear Regulatory Commission is proposing to amend its regulations to require that light-water-cooled nuclear power plants be capable of withstanding a total loss of alternating current (AC) electric power (called "station blackout") for a specified duration and to maintain reactor core cooling during that period. This proposed requirement is based on information developed under the Commission's study of Unresolved Safety Issue A-44, "Station Blackout." The proposed change is intended to provide further assurance that a station blackout (loss of both offsite power and onsite emergency AC power systems) will not adversely affect the public health and safety.

DATE: The comment period expires [insert a date 90 days after the publication of this Notice of Proposed Rulemaking]. Comments received after this date will be considered if it is practical to do so, but assurance of consideration cannot be given except as to comments received before this date.

ADDRESSES: Send comments to: The Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555, Attention: Docketing and Service Branch. Copies of comments received may be examined and copied for a fee at the NRC Public Document Room, 1717 H Street, NW, Washington, DC.

FOR FURTHER INFORMATION CONTACT: Alan Rubin, Division of Safety Technology, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, DC 20555, Telephone: (301) 492-8303.

SUPPLEMENTARY INFORMATION: The alternating current (AC) electric power for essential and nonessential service in a nuclear power plant is supplied primarily by offsite power. Redundant onsite emergency AC power systems are also provided in the event that all offsite power sources are lost. These systems provide power for various safety systems including reactor core decay heat removal and containment heat removal which are essential for preserving the integrity of the reactor core and the containment building, respectively. The reactor core decay heat can also be removed for a limited time period by safety systems that are independent of AC power.

The term "station blackout" means the loss of offsite AC power to the essential and nonessential electrical buses concurrent with turbine trip and the unavailability of the redundant onsite emergency AC power systems (e.g., as a result of units out of service for maintenance or repair, failure to start on demand, or failure to continue to run after start). If a station blackout persists for a sufficient time such that the capability of the AC-independent systems to remove decay heat is exceeded, core melt and containment failure could result.

The Commission's existing regulations establish requirements for the design and testing of onsite and offsite electric power systems that are intended to minimize the probability of losing all AC power. See General Design Criteria 17 and 18, 10 CFR Part 50, Appendix A. The existing regulations do not require explicitly that nuclear power plants be designed to assure that the core can be cooled and the integrity of the reactor coolant pressure boundary can be maintained for any specified period of loss of all AC power.

As operating experience has accumulated, the concern has arisen that the reliability of both the onsite and offsite emergency AC power systems might be less than originally anticipated, even for designs that meet the requirements of General Design Criteria 17 and 18. Many operating plants have experienced a total loss of offsite power, and more occurrences can be expected in the future. Also, operating experience with

onsite emergency power systems has included many instances when diesel generators failed to start. In a few cases, there has been a complete loss of both the offsite and the onsite AC power systems. During these events, AC power was restored in a short time without any serious consequences.

In 1975, the results of the Reactor Safety Study (WASH-1400) showed that station blackout could be an important contributor to the total risk from nuclear power plant accidents. Although this total risk was found to be small, the relative importance of the station blackout accident was established. Subsequently, the Commission designated the issue of station blackout as an Unresolved Safety Issue (USI); a Task Action Plan (TAP A-44) was issued in July 1980, and work was initiated to determine whether additional safety requirements were needed. Factors considered in the analysis of risk from station blackout included: (1) the likelihood and duration of the loss of offsite power, (2) the reliability of the onsite AC power system, and (3) the potential for severe accident sequences after a loss of all AC power, including consideration of the capability to remove core decay heat without AC power for a limited time period.

The technical findings of the staff's studies of the station blackout issue are presented in NUREG-1032, "Evaluation of Station Blackout Accidents at Nuclear Power Plants, Technical Findings Related to Unresolved Safety Issue A-44."[(1)] Additional information is provided in supporting contractor reports: NUREG/CR-3226, "Station Blackout Accident Analyses" published in May 1983; NUREG/CR-2989, "Reliability of Emergency AC Power Systems at Nuclear

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[(1)] NUREG-1032 was issued for public comment on . Copies of this report are available for public inspection and copying for a fee at the NRC Public Document Room at 1717 H Street, NW, Washington, DC 20555. Free single copies of NUREG-1032 may be requested by writing to the Publication Services Section, Division of Technical Information and Document Control, U.S. Nuclear Regulatory Commission, Washington, DC 20555.

Power Plants" published in July 1983; and NUREG/CR-3992, "Collection and Evaluation of Complete and Partial Losses of Offsite Power at Nuclear Power Plants" published in .[2] The major results of these studies are given below.

- Losses of offsite power can be characterized as those resulting from plant-centered faults, utility grid blackout, and severe weather-induced failures of offsite power sources. Based on operating experience, the frequency of total losses of offsite power in operating nuclear power plants was found to be about one per 10 site-years. The median restoration time was about one-half hour, and 90 percent of the offsite power losses were restored in approximately 3 hours (NUREG/CR-3992).

- The review of a number of representative designs of onsite emergency AC power systems has indicated a variety of potentially important failure causes. However, no single improvement was identified that could result in a significant improvement in overall diesel generator reliability. Data obtained from operating experience show that the typical individual emergency diesel generator failure rate is about 2.5×10^{-2} per demand, and that the emergency AC power system unavailability for a plant which has two emergency diesel generators, one of which is required for decay heat removal, is about 2×10^{-3} per demand (NUREG/CR-2989).

- Given the occurrence of a station blackout, the likelihood of resultant core damage or core melt is dependent on the reliability and capability of decay heat removal systems that are not dependent on AC power. If sufficient AC-independent capability exists, additional time will be available to restore AC power needed for long-term cooling (NUREG/CR-3226).

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[2] Copies of these documents are available for public inspection and copying for a fee at the NRC Public Document Room at 1717 H Street, NW, Washington, DC 20555. Copies may also be purchased by calling (301) 492-9530 or by writing to the Publication Services Section, Division of Technical Information and Document Control, U.S. Nuclear Regulatory Commission, Washington, DC 20555, or purchased from the National Technical Information Service, Department of Commerce, 5285 Port Royal Road, Springfield, VA 22161.

- It was determined by reviewing design, operational, and site-dependent factors that the expected frequency of core damage resulting from station blackout events could be maintained near or below 10^{-5} per reactor-year for any nuclear plant with readily achievable diesel generator reliabilities, provided that the plant is designed to cope with station blackout for a specified duration. The duration for a specific plant is based on a comparison of the plant's characteristics to those factors that have been identified as the main contributors to risk from station blackout (NUREG-1032).

As a result of the station blackout studies, improved guidance will be provided to licensees regarding maintaining minimimum emergency diesel generator reliability to minimize the probability of losing all AC power. In addition, the Commission is proposing to amend its regulations by adding a new §50.63 and by adding a new final paragraph to General Design Criterion 17, Appendix A of 10 CFR Part 50, to require that all nuclear power plants be capable of coping with a station blackout for some specified period of time. The period of time for a specific plant would be determined based on the existing capability of the plant as well as a comparison of the individual plant design with factors that have been identified as the main contributors to risk of core melt resulting from station blackout.

These factors, which vary significantly from plant to plant because of considerable differences in design of plant electric power systems as well as site-specific considerations, include: (1) redundancy of onsite emergency AC power sources (i.e., number of sources minus the number needed for decay heat removal), (2) reliability of onsite emergency AC power sources (usually diesel generators), (3) frequency of loss of offsite power, and (4) probable time to restore offsite power. The frequency of loss of, and time to restore offsite power are related to grid and switchyard reliabilities, historical weather data for severe storms, and the availability of nearby alternate power sources (e.g., gas turbines). Experience has shown that long duration offsite power outages are caused primarily by severe storms (hurricanes, ice, snow, etc.).

The objective of the proposed rule is to reduce the risk of severe accidents resulting from station blackout by maintaining highly reliable AC electric power systems and, as additional defense-in-depth, assuring that plants can cope with a station blackout for some period of time. If the proposed rule is adopted, all licensees and applicants would be required to assess the capability of their plants to cope with a station blackout (i.e., determine the amount of time the plant can maintain core cooling and containment integrity with AC power unavailable), and to have procedures and training to cope with such an event. Plants would be required to be able to cope with a specified minimum duration station blackout selected on a plant-specific basis.

On the basis of station blackout studies conducted for USI A-44, and presented in the reports referenced above, the NRC staff has developed a draft regulatory guide entitled "Station Blackout,"[(3)] which presents guidance on (1) maintaining a high level of reliability for emergency diesel generators, (2) developing procedures and training to restore offsite and onsite emergency AC power should either one or both become unavailable, and (3) selecting a plant-specific minimum duration for station blackout capability to comply with the proposed amendment to General Design Criterion 17. Application of the methods in this guide would result in selection of a 4-hour or 8-hour station blackout duration, depending on the specific plant design and site-related characteristics. However, applicants and licensees could propose alternative methods to that specified in the regulatory guide in order to justify other minimum durations for station blackout capability.

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(3) A notice of availability and request for comments on the draft regulatory guide will be published within a few days of this Notice of Proposed Rulemaking. Copies of the draft regulatory guide are available for public inspection and copying for a fee at the NRC Public Document Room at 1717 H Street, NW, Washington, DC 20555, and will be distributed to those on the automatic distribution list for draft regulatory guides. Free single copies of the draft regulatory guide may be obtained by writing to the U.S. Nuclear Regulatory Commission, Washington, DC 20555, Attention: Director, Division of Technical Information and Document Control.

If the proposed rule and regulatory guide are issued, those plants with an already low risk from station blackout would be required to withstand a station blackout for a relatively short period of time and probably would need few, if any, modifications as a result of the rule. Plants with currently higher risk from station blackout would be required to withstand somewhat longer duration blackouts. Depending on their existing capability, these plants might also need to make modifications (such as increasing station battery capacity or condensate storage tank capacity) in order to cope with the longer station blackout duration. The proposed rule would require licensees to develop, in consultation with the Office of Nuclear Reactor Regulation, proposed plant-specific schedules for implementation of any needed modifications.

FINDING OF NO SIGNIFICANT ENVIRONMENTAL IMPACT: AVAILABILITY

The Commission has determined under the National Environmental Policy Act of 1969, as amended, and the Commission's regulations in Subpart A of 10 CFR Part 51, that this rule, if adopted, would not be a major Federal action significantly affecting the quality of the human environment, and therefore an environmental impact statement is not required. There would not be any adverse environmental impacts as a result of the proposed rule for the following reasons: (1) there would be no additional radiological exposure to the general public or plant employees, and (2) plant shutdown is not required so there would be no additional environmental impacts as a result of the need for replacement power. The environmental assessment and finding of no significant impact on which this determination is based are available for inspection at the NRC Public Document Room, 1717 H Street NW, Washington, DC. Single copies of the environmental assessment and the finding of no significant impact are available from Mr. Karl Kniel, Office of Nuclear Reactor Regulation, U. S. Nuclear Regulatory Commission, Washington, DC 20555, Telephone: (301) 492-7359.

PAPERWORK REDUCTION ACT STATEMENT

This proposed rule amends information collection requirements that are subject to the Paperwork Reduction Act of 1980 (44 U.S.C. 3501 et seq.). This rule has been submitted to the Office of Management and Budget for review and approval of the paperwork requirements.

REGULATORY ANALYSIS

The Commission has prepared a regulatory analysis for this regulation. The analysis examines the costs and benefits of the rule as considered by the Commission. A copy of the regulatory analysis, NUREG-1109, For Comment, "Regulatory Analysis for the Resolution of Unresolved Safety Issue A-44, Station Blackout," is available for inspection and copying for a fee at the NRC Public Document Room, 1717 H Street, NW, Washington, DC 20555. Free single copies of NUREG-1109 may be obtained by writing to the Division of Technical Information and Document Control, U.S. Nuclear Regulatory Commission, Washington, DC 20555.

The Commission requests public comment on the regulatory analysis. Comments on the draft analysis may be submitted to the NRC as indicated under the ADDRESSES heading.

REGULATORY FLEXIBILITY CERTIFICATION

In accordance with the Regulatory Flexibility Act of 1980, 5 U.S.C. 605(b), the Commission hereby certifies that this proposed rule, if promulgated, will not have a significant economic impact on a substantial number of small entities. This proposed rule specifies that nuclear power plants be able to withstand a total loss of AC power for a specified time duration and maintain reactor core cooling during that period. These facilities are licensed under the provisions of 10 CFR 50.21(b) and 10 CFR 50.22. The companies that own these facilities do not fall within the scope of "small entities" as set forth in the Regulatory Flexibility Act or the small business size standards set forth in regulations issued by the Small Business Administration in 13 CFR Part 121.

LIST OF SUBJECTS IN 10 CFR PART 50

Antitrust, Classified information, Fire prevention, Incorporation by reference, Intergovernmental relations, Nuclear power plants and reactors, Penalty, Radiation protection, Reactor siting criteria, Reporting and recordkeeping requirements.

For the reasons set out in the preamble and under the authority of the Atomic Energy Act of 1954, as amended, the Energy Reorganization Act of 1974, as amended, and 5 U.S.C. 553, notice is hereby given that adoption of the following amendments to 10 CFR Part 50 is contemplated.

PART 50 - DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES

1. The authority citation for Part 50 continues to read as follows:

AUTHORITY: Secs. 103, 104, 161, 182, 183, 186, 189, 68 Stat. 936, 937, 948, 953, 954, 955, 956, as amended, sec. 234, 83 Stat. 1244, as amended (42 U.S.C. 2133, 2134, 2201, 2232, 2233, 2236, 2239, 2282); secs. 201, 202, 206, 88 Stat. 1242, 1244, 1246, as amended (42 U.S.C. 5841, 5842, 5846), unless otherwise noted.

Section 50.7 also issued under Pub. L. 95-601, sec. 10, 92 Stat. 2951 (42 U.S.C. 5851). Sections 50.57(d), 50.58, 50.91, and 50.92 also issued under Pub. L. 97-415, 96 Stat. 2071, 2073 (42 U.S.C. 2133, 2239). Section 50.78 also issued under sec. 122, 68 Stat. 939 (42 U.S.C. 2152). Sections 50.80-50.81 also issued under sec. 184, 68 Stat. 954, as amended (42 U.S.C. 2234). Sections 50.100-50.102 also issued under sec. 186, 68 Stat. 955 (42 U.S.C. 2236).

For the purposes of sec. 223, 68 Stat. 958, as amended (42 U.S.C. 2273), §§ 50.10(a), (b), and (c), 50.44, 50.46, 50.48, 50.54, and 50.80(a) are issued under sec. 161b, 68 Stat. 948, as amended (42 U.S.C. 2201(b));

§§ 50.10(b) and (c) and 50.54 are issued under sec. 161i, 68 Stat. 949, as amended (42 U.S.C. 2201(i)); and §§ 50.55(e), 50.59(b), 50.70, 50.71, 50.72, 50.73, and 50.78 are issued under sec. 161o, 68 Stat. 950, as amended (42 U.S.C. 2201(o)).

2. In §50.2, a definition of "station blackout" is added as a new paragraph (y) to read as follows:

§50.2 Definitions.

* * * * *

(y) "Station blackout" means the complete loss of alternating current (AC) electric power to the essential and nonessential switchgear buses in a nuclear power plant (i.e., loss of the offsite electric power system concurrent with turbine trip and unavailability of the onsite emergency AC power system).

3. A new §50.63 is added to read as follows:

§50.63 Loss of All Alternating Current Power.

(a) Requirements. Each light-water-cooled nuclear power plant licensed to operate must be able to withstand and recover from a station blackout as defined in §50.2 for a specified duration in accordance with the requirements in paragraph (e) of General Design Criterion 17 of Appendix A of this part.

(b) Limitation of Scope. Paragraphs (c) and (d) of this section do not apply to those plants licensed to operate prior to [insert the effective date of this amendment] if the capability to withstand station blackout was considered in the operating license proceeding and a specified duration was accepted as the licensing basis for the facility.

(c) Implementation - Determination of Station Blackout Duration.

(1) For each light-water-cooled nuclear power plant licensed to operate on or before [insert the effective date of this amendment], the licensee shall submit to the Director of the Office of Nuclear Reactor Regulation by [insert a date 270 days after after the effective date of this amendment]:

(i) a determination of the maximum duration for which the plant as currently designed is able to maintain core cooling and containment integrity in the event of a station blackout as defined in §50.2(y);

(ii) a description of the procedures that have been established for station blackout events for the duration determined in paragraph (c)(1)(i) of this section and for recovery therefrom;

(iii) an identification of the factor(s) that limit the capability of the plant to cope with a station blackout for a longer time than that determined in paragraph (c)(1)(i) of this section;

(iv) a proposed station blackout duration to be used in determining compliance with paragraph (e) of General Design Criterion 17 of Appendix A of this part, including a justification for the selection based on: (1) the redundancy of the onsite emergency AC power sources, (2) the reliability of the onsite emergency AC power sources, (3) the expected frequency of loss of offsite power, and (4) the probable time needed to restore offsite power; and

(v) an identification of the factors, if any, that limit the capability of the plant to meet the requirements of Criterion 17 for the specified station blackout duration proposed in the response to paragraph (c)(1)(iv) of this section.

(2) After consideration of the information submitted in accordance with paragraph (c)(1) of this section, the Commission will notify the licensee of its determination of the specified station blackout duration to be used in determining compliance with General Design Criterion 17 of Appendix A of this part.

(d) Implementation - Schedule for Implementing Equipment Modifications.

(1) For each light-water-cooled nuclear power plant licensed to operate on or before [insert the effective date of this amendment], the licensee shall, within 180 days of the notification provided in accordance with paragraph (c)(2) of this section, submit to the Director of the Office of Nuclear Reactor Regulation a schedule for implementing any equipment and procedure modifications necessary to meet the requirements of General Design Criterion 17 of Appendix A of this part. This submittal must include an explanation of the schedule and a justification if the schedule does not provide for completion of the modifications within two years of the notification provided in accordance with paragraph (c)(2) of this section.

(2) The licensee and the NRC staff shall mutually agree upon a final schedule for implementing modifications necessary to comply with the requirements of Criterion 17.

4. In Appendix A, General Design Criterion 17 is revised to read as follows:

APPENDIX A

General Design Criteria for Nuclear Power Plants

*   *   *   *   *

I. Overall Requirements * * *

Criterion 17-Electric Power Systems. (a) An onsite electric power system and an offsite electric 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.

(b) The onsite electric power supplies, including the batteries, and the onsite electric distribution system, shall have sufficient independence, redundancy, and testability to perform their safety functions assuming a single failure.

(c) Electric power from the transmission network to the onsite electric distribution system shall be supplied by two physically independent circuits (not necessarily on separate rights of way) designed and located so as to minimize to the extent practical the likelihood of their simultaneous failure under operating and postulated accident and environmental conditions. A switchyard common to both circuits is acceptable. Each of these circuits shall be designed to be available in sufficient time following a loss of all onsite alternating current power supplies and the other offsite electric

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.

(d) Provisions shall be included to minimize the probability of losing electric power from any of the remaining supplies 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 electric power supplies.

(e) The reactor core and associated coolant, control, and protection systems, including the station batteries, shall provide sufficient capacity and capability to assure that the core is cooled and containment integrity is maintained in the event of a station blackout (as defined in §50.2(y)) for a specified duration. The following factors shall be considered in specifying the station blackout duration: (1) the redundancy of the onsite emergency AC power sources, (2) the reliability of the onsite emergency AC power sources, (3) the expected frequency of loss of offsite power, and (4) the probable time needed to restore offsite power.

Dated at Washington, DC, this ____ day of __________ 1984.

For the Nuclear Regulatory Commission

Samuel J. Chilk,
Secretary of the Commission.

/ Underlined text indicates proposed additional paragraph to GDC 17.

[7590-01]

1/15/88

ENCLOSURE B

Federal Register Notice of Final Rulemaking

NUCLEAR REGULATORY COMMISSION

10 CFR Part 50

Station Blackout

AGENCY: Nuclear Regulatory Commission.

ACTION: Final rule.

SUMMARY: The Nuclear Regulatory Commission is amending its regulations to require that light-water-cooled nuclear power plants be capable of withstanding a total loss of alternating current (ac) electric power (called "station blackout") for a specified duration and maintaining reactor core cooling during that period. This requirement is based on information developed under the Commission's study of Unresolved Safety Issue A-44, "Station Blackout." The amendment is intended to provide further assurance that a loss of both offsite power and onsite emergency ac power systems will not adversely affect the public health and safety.

EFFECTIVE DATE:

FOR FURTHER INFORMATION CONTACT: Aleck Serkiz, Division of Reactor and Plant Systems, Office of Nuclear Reactor Regulatory Research, U.S. Nuclear Regulatory Commission, Washington, DC 20555, Telephone: (301) 492-3555.

SUPPLEMENTARY INFORMATION:

Background

The alternating current (ac) electric power for essential and nonessential service in a nuclear power plant is supplied primarily by offsite power. Redundant onsite emergency ac power systems are also provided in the event that all offsite power sources are lost. These systems provide power for various safety functions, including reactor core decay heat removal and containment heat removal, which are essential for preserving the integrity of the reactor core and the containment building, respectively. The reactor core decay heat can also be removed for a limited time period by safety systems that are independent of ac power.

The term "station blackout" means the loss of offsite ac power to the essential and nonessential electrical buses concurrent with turbine trip and the unavailability of the redundant onsite emergency ac power systems (e.g., as a result of units out of service for maintenance or repair, failure to start on demand, or failure to continue to run after start). If a station blackout persists for a time beyond the capability of the ac-independent systems to remove decay heat, core melt and containment failure could result.

The Commission's existing regulations establish requirements for the design and testing of onsite and offsite electric power systems that are intended to reduce the probability of losing all ac power to an acceptable level. (See General Design Criteria 17 and 18, 10 CFR Part 50, Appendix A.) The existing regulations do not require explicitly that nuclear power plants be designed to assure that core cooling can be maintained for any specified period of loss of all ac power.

As operating experience has accumulated, the concern has arisen that the reliability of both the onsite and offsite emergency ac power systems might be less than originally anticipated, even for designs that meet the requirements of General Design Criteria 17 and 18. Many operating plants have experienced a total loss of offsite power, and more occurrences can be expected in the

future. Also, operating experience with onsite emergency power systems has included many instances when diesel generators failed to start. In a few cases, there has been a complete loss of both the offsite and the onsite ac power systems. During these events, ac power was restored in a short time without any serious consequences.

In 1975, the results of the Reactor Safety Study (WASH-1400) showed that station blackout could be an important contributor to the total risk from nuclear power plant accidents. Although this total risk was found to be small, and not undue, the relative importance of the station blackout accident was established. Subsequently, the Commission designated the issue of station blackout as an Unresolved Safety Issue (USI); a Task Action plan (TAP A-44) was issued in July 1980, and studies were initiated to determine whether additional safety requirements were needed. Factors considered in the analysis of risk from station blackout included: (1) the likelihood and duration of the loss of offsite power; (2) the reliability of the onsite ac power system; and (3) the potential for severe accident sequences after a loss of all ac power, including consideration of the capability to remove core decay heat without ac power for a limited time period.

The technical findings of the staff's studies of the station blackout issue are presented in NUREG-1032, "Evaluation of Station Blackout Accidents at Nuclear Power Plants, Technical Findings Related to Unresolved Safety Issue A-44."[(1)] Additional information is provided in supporting contractor reports: NUREG/CR-3226, "Station Blackout Accident Analyses" published in May 1983; NUREG/CR-2989, Reliability of Emergency AC Power Systems at Nuclear Power Plants" published in July 1983; NUREG/CR-3992, "Collection and Evaluation of Complete and Partial Losses of Offsite Power at Nuclear Power Plants" published in February 1985; and NUREG-CR 4347, "Emergency Diesel Generator Operating

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[(1)] Draft NUREG-1032 was issued for public comment on June 15, 1985.

Experience, 1981-1983" published in December 1985.[(2)] The major results of these studies are given below.

°Losses of offsite power can be characterized as those resulting from plant-centered faults, utility grid blackout, and severe weather-induced failures of offsite power sources. Based on operating experience, the frequency of total losses of offsite power in operating nuclear power plants was found to be about one per 10 site-years. The median restoration time was about one-half hour, and 90 percent of the offsite power losses were restored within approximately 3 hours (NUREG/CR-3992).

°The review of a number of representative designs of onsite emergency ac power systems has indicated a variety of potentially important failure causes. However, no single improvement was identified that could result in a significant improvement in overall diesel generator reliability. Data obtained from operating experience in the period from 1976 to 1980 showed that the typical individual emergency diesel generator failure rate was about 2.5×10^{-2} per demand (i.e., one chance of failure in 40 demands), and that the emergency ac power system unavailability for a plant which has two emergency diesel generators, one of which was required for decay heat removal, was about 2×10^{-3} per demand (NUREG/CR-2989).

°Compared to the data in NUREG/CR-2989, updated estimates of emergency diesel generator failure rates indicated that diesel generator reliability has improved somewhat from 1976 to 1983. For the period 1981 to 1983, the mean failure rate for all demands was about 2.0×10^{-2} per demand (i.e., one chance of failure in 50 demands). However, the data

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[(2)] Copies of these NUREGS are available for public inspection and copying for a fee at the NRC Public Document Room at 1717 H Street, NW, Washington, DC 20555. Copies may also be purchased through the U.S. Government Printing Office by calling (202) 275-2060 or by writing to the Superintendent of Documents, U.S. Government Printing Office, P. O. Box 37082, Washington, DC 20013-7082.

also indicate that the probability of diesel generator failures during actual demands (i.e., during losses of offsite power) is greater than that during surveillance tests (NUREG/CR-4347).

°Given the occurrence of a station blackout, the likelihood of resultant core damage or core melt is dependent on the reliability and capability of decay heat removal systems that are not dependent on ac power. If sufficient ac-independent capability exists, additional time will be available to restore ac power needed for long-term cooling. (NUREG/CR-3226).

°It was determined by reviewing design, operational, and site-dependent factors that the expected frequency of core damage resulting from station blackout events could be maintained near 10^{-5} per reactor-year with readily achievable diesel generator reliabilities, provided that plants are designed to cope with station blackout for a specified duration. The duration for a specific plant is based on a comparison of the plant's characteristics to those factors that have been identified as the main contributors to risk from station blackout (NUREG-1032).

The staff's technical findings show that station blackout (SBO) does not pose an undue risk to public health and safety. The findings summarized above show that: recovery from loss of offsite power occurs for the most part in less than 4 hours, emergency diesel generator reliability is high (i.e._ 0.95), and that given an SBO the likelihood of core damage is more dependent on decay heat removal systems that are non ac dependent. However, plant design and operational characteristics, plus site dependent factors (such as anticipated weather conditions) introduce a level of variability which warrants a need for plant specific coping analyses to provide greater assurance that core cooling can be maintained until ac power is restored. Thus the Commission believes that 10 CFR 50.63 will bring about a significant increase in protection to the public health and safety. As a result of SBO coping analyses, improved guidance will be provided to licensees regarding maintaining minimum emergency diesel generator reliability to minimize the probability of losing all ac

power. In addition, the Commission is amending its regulations by adding a new §50.63 to require that all nuclear power plants be capable of coping with a station blackout for some specified period of time. The period of time for a specific plant will be determined based on a comparison of the individual plant's design with factors that have been identified as the main contributors to risk of core damage resulting from station blackout.

These factors, which vary significantly from plant to plant because of considerable differences in design of plant electric power systems as well as site-specific considerations, include: (1) redundancy of onsite emergency ac power sources (i.e., number of sources minus the number needed for decay heat removal), (2) reliability of onsite emergency ac power sources (usually diesel generators), (3) frequency of loss of offsite power, and (4) probable time to restore offsite power. The frequency of loss of, and time to restore, offsite power are related to grid and switchyard reliabilities, historical weather data for severe storms, and the availability of nearby alternate power sources (e.g., gas turbines). Experience has shown that long duration offsite power outages are caused primarily by severe storms (hurricanes, ice, snow, etc.).

The objective of the rule is to reduce the risk of severe accidents resulting from station blackout by maintaining highly reliable ac electric power systems and, as additional defense-in-depth, assuring that plants can cope with a station blackout for some period of time. The rule requires all plants to be able to cope with a station blackout for a specified acceptable duration selected on a plant-specific basis. All licensees and applicants are required to assess the capability of their plants to cope with a station blackout (i.e., determine that the plant can maintain core cooling with ac power unavailable for an acceptable period of time), and to have procedures and training to cope with such an event. Licensees may use an alternate ac power source, if that source meets specific criteria for independence and capacity, and can be shown to be available within one hour to cope with a station blackout. A coping analysis is not required for those plants that choose this alternate ac approach if the alternate ac can be demonstrated by test to be available to power the shutdown busses within 10 minutes of the onset of station blackout.

On the basis of station blackout studies conducted for USI A-44, and presented in the reports referenced above, the NRC staff has developed Regulatory Guide 1.155 entitled "Station Blackout,"[3] which presents guidance on (1) maintaining a high level of reliability for emergency diesel generators, (2) developing procedures and training to restore offsite and onsite emergency ac power should either one or both become unavailable, and (3) selecting a plant-specific acceptable station blackout duration which the plant would be capable of surviving without core damage. Application of the methods in this guide would result in selection of an acceptable station blackout duration (e.g., 2, 4, 8 or 16 hours) which depended on the specific plant design and site-related characteristics acceptable to the staff. However, applicants and licensees could propose alternative methods to that specified in the regulatory guide in order to justify other acceptable durations for station blackout capability. unavailable for an acceptable period of time), and to have procedures and training to cope with such an event. Licensees may use an alternate ac power source, if that source meets specific criteria for independence and capacity, and can be shown to be available within one hour to cope with a station blackout. A coping analysis is not required for those plants that choose this alternate ac approach if the alternate ac can be demonstrated by test to be available to power the shutdown busses within 10 minutes of the onset of station blackout.

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(3) A notice of availability and request for comments on the draft regulatory guide was published in the Federal Register on April 3, 1986 (51 FR 11494). Free single copies of the regulatory guide may be obtained by writing to the Distribution Section Division of Information Support Services, U.S. Nuclear Regulatory Commission, Washington, DC 20555.

acceptable station blackout duration which the plant would be capable of surviving without core damage. Application of the methods in this guide would result in selection of an acceptable station blackout duration (e.g., 2, 4, 8 or 16 hours) which depended on the specific plant design and site-related characteristics acceptable to the staff. However, applicants and licensees could propose alternative methods to that specified in the regulatory guide in order to justify other acceptable durations for station blackout capability. Additionally, the regulatory guide on station blackout presents guidance on quality assurance and specifications for alternate ac source(s) and non-safety related equipment required for coping with station blackout. The equipment installed to meet the station blackout rule must be implemented such that it does not degrade the existing safety related systems. This is to be accomplished by making the non-safety related equipment independent to the extent practicable from existing safety related systems. The guidance provided in the regulatory guide illustrates the specifications that the staff would find acceptable for non-safety systems and equipment. The quality assurance guidance for the non-safety related equipment for which there are no existing NRC quality assurance requirements (e.g., Appendix B, Appendix R) embody the following elements: (1) design control and procurement document control, (2) instructions, procedures and drawings, (3) control of purchased material, equipment and services, (4) inspection, (5) test and test control, (6) inspection, test and operating status, (7) non-conforming items, (8) corrective action, (9) Records, (10) Audits. NRC inspections will focus on the implementation and the effectiveness of these quality controls as described in the proposed regulatory guide.

Based on the rule and regulatory guide, those plants with an already low risk from station blackout would be required to withstand a station blackout for a relatively short period of time and probably would need few, if any, modifications as a result of the rule. Plants with currently higher risk from station blackout would be required to withstand somewhat longer duration blackouts. Depending on their existing capability, these plants might need to make hardware modifications (such as increasing station battery capacity or condensate storage tank capacity) in order to cope with the longer station

blackout duration. The rule requires that each ligh-water-cooled nuclear power plant licensed to operate must be able to withstand for a specified duration and recover from a station blackout. The rule requires each plant to perform a coping analysis and identify the coping duration, along with the basis therefore and a description of procedures established for coping and recovery. If modifications to equipment or plant procedures are necessary, these are to be identified and a schedule provided for implementing such changes.

It should be noted, based on all evidence that staff has on hand, that no undue risk exists with, or without, the promulgation of the station blackout (SBO) rule. However, SBO may still remain an important contributor to residual risk. This SBO rule will enhance safety by accident prevention and thereby reduce the likelihood of a core damage accident being caused by a station blackout occurrence. This does not mean however, that further enhancements in reducing the overall residual risk are not achievable by additional improvements in severe accident management, given the assumption that core damage occurs, whether from SBO sequences or other causes (such as small or large LOCA sequences). Initiatives that provide such safety enhancements (through improvements of core damage management procedures) are currently being pursued apart from the SBO rule. Therefore, this rule should be viewed as being in the same accident prevention context as the ATWS rule (§50.62) and the fire protection rule (§50.48). Such rules recognize multiple failure possibilities resulting from common cause effects that should be addressed. This concern has been recognized in the Introduction to Appendix A of 10CFR50.

Proposed Rule

On March 21, 1986, the Commission published a proposed rule in the Federal Register (51 FR 9829) that would require (1) light-water-cooled nuclear power plants to be capable of coping with a station blackout for a specified duration, and (2) licensees to determine the maximum duration for which their plants as currently designed are able to cope with a station blackout. A 90-day comment period expired on June 19, 1986.

On April 3, 1986 (13 days after the proposed rule was published), the NRC published in the Federal Register (51 FR 11494) a notice of availability and request for comments on a draft regulatory guide entitled "Station Blackout" (Task SI 501-4). This draft guide provided guidance for licensees to comply with the proposed station blackout rule. Many letters commenting on the proposed rule also included comments on the draft regulatory guide. Responses to these comments provided below address the public comments on the draft guide as well as on the proposed rule.

Comments on the Proposed Rule

The Commission received 53 letters commenting on the proposed rule.(4) Forty-five of these were from the nuclear industry, comprised of electric utilities, consortiums of electric utilities, vendors, a trade association, and an architect/engineering firm. Other letters were submitted by the Union of Concerned Scientists (UCS), the Department of Nuclear Safety of the State of Illinois (IDNS), a representative of the Professional Reactor Operator Society, a citizens group, a consultant, and three individuals. Largely, the industry comments were opposed to generic rulemaking to resolve the station blackout issue. The Nuclear Management and Resources Council (NUMARC), formerly the Nuclear Utilities Management and Resources Committee, submitted, along with its comments on the proposed rule, a set of four industry initiatives that it believes would resolve this issue without rulemaking. Thirty-nine of the industry letters supported NUMARC's submittal. NUMARC proposed a fifth initiative (see item 21) by letter dated October 5, 1987. On the other hand, UCS, IDNS, and the citizens group supported the Commission's objective in the proposed rule, but did not believe the rule and guidance associated with the rule went far enough to reduce the possibility of a serious accident that could be initiated by a total loss of alternating current (ac) power.

(4) Copies of these letters are available for public inspection and copying for a fee at the NRC Public Room at 1717 H Street, NW, Washington, DC.

Every letter was reviewed and considered by the staff in formulating the final resolution of USI A-44. Because of the large number of comments, it was not practical to prepare formal responses to each one separately. However, since many comments were on similar subjects, the discussion and response to the comments has been grouped into the following subjects:(5)

1. Quality Classification of Modifications.

2. Whether the Backfit Analysis Adequately Implements the Backfit Rule.

3. Cost-Benefit and Whether §50.63 Meets the "Substantial Increase in the Overall Protection of the Public Health and Safety."

4. Whether NRC Should Require Substantial Improvements in Safety that Go Beyond Those Proposed in this Rulemaking.

5. The Need for Generic Rulemaking.

6. Applicability of the Proposed §50.63 to Specific Plants.

7. Plant-Specific Features and Capabilities.

8. The Source Term Used to Estimate Consequences.

9. Specificity on the Extent of Required Coping Studies.

10. Acceptable Duration for Coping with a Station Blackout.

11. Credit for Alternate or Diverse AC Power Sources.

12. Trends on the Reliability of AC Power Sources.

13. Sharing of Emergency Diesel Generators Between Units at Multi-Unit Sites.

14. Clarification of the Definitions of Station Blackout and Diesel Generator Failure.

15. Specificity and Clarification of Requirements.

16. Technical Comments on NUREG-1032.

17. Relationship of USI A-44 to Other NRC Generic Issues.

18. An Alternative of Plant-Specific Probabilistic Assessments.

19. Procedures and Operator Actions During Station Blackout.

(5) The first four subjects are ones on which the Commissioners specifically requested public comments when the proposed rule was published.

20. Schedule Provisions in the Proposed §50.63.

21. Industry Initiatives

The comments and responses to each of these subjects are presented on the following pages.

1. Quality Classification of Modifications

The Commission requested comments on whether the staff should give further consideration to upgrading to safety grade the plant modifications needed (if any) to meet the proposed rule. Upgrading to safety grade would further ensure appropriate licensee attention is paid to maintaining equipment in a high state of operability and reliability.

Comments - The prevailing view by industry on this subject is represented by the following comments submitted by NUMARC:

> Quality Classification is Unnecessary - Equipment used to prevent or respond to a station blackout should be sufficiently available and operable to meet its required function. To this extent, the Commission's desire that appropriate attention be paid to maintaining a sufficiently high state of operability and reliability is appropriate. The point of departure begins with the method for achieving this objective. Specifically, by itself, a "safety grade" classification scheme does not solely equate with high states of equipment operability and reliability. Such classification systems too often can become a documentation exercise more than a process for providing the requisite level of system functionality.

Duquesne Light agreed with this view and expressed the following comments:

> Any plant modifications or additional equipment required to meet the proposed rule should not be specified safety grade. For equipment which is to be manually started and placed in service for testing or in the

> event of a loss of power condition there is no necessity for specifying safety grade since adequate reliability can be obtained through normal surveillance testing and the proper maintenance of commercial power plant equipment. The cost difference in safety grade vs. commercial grade modifications is significant and must be emphasized.

The opposite point of view was taken by the IDNS.

> No credit should be given for the capability of equipment to respond to a station blackout unless that equipment was originally designed, constructed, inspected, performance tested, qualified, certified for the intended safety-related purpose, and the equipment is maintained to the highest industry safety standards.

Gulf States Utilities commented,

> The proposed rule does not provide sufficient direction on the quality classification of plant modifications that may be required to meet the rule. ...the quality classification of plant modifications implemented to meet the proposed rule should be commensurate with classification of the system they support.

Response - The proposed §50.63 does not specifically address the topic of safety classification of plant modifications; however, detailed guidance is provided in Regulatory Guide 1.155 dealing with quality assurance and equipment specifications for non-safety related equipment. Any safety related equipment used either presently, or in modifications resulting from this rule, should meet the criteria currently applied to such equipment.

The technical analyses performed for USI A-44 (NUREG-1032) show that plant-centered events (ie. those events in which design and operational characteristics of the plant itself play a role in the likelihood of loss of offsite power), and area - or weather - related events (e.g. grid reliability or external influences on the grid) are the dominant causes of loss of offsite power. Neither seismic events nor events related to single failure causes

were found to be major contributors to loss of offsite power. Therefore, both the staff's findings and public comments received do not support an explicit need for plant modifications for coping with station blackout (SBO) to be seismically qualified.

The substantial increase in protection sought by this rule can be achieved by modifications which meet criteria somewhat less stringent than generally required by safety grade criteria. Safety related equipment modifications to meet all safety grade related criteria would be more burdensome and expensive and would likely achieve only a very small further reduction in risk. The major contributors to the residual risk of loss of offsite power are adequately dealt with by modifications which conform to the quality assurance and equipment specification guidance provided in Regulatory Guide 1.155.

2. Whether the Backfit Analysis Adequately Implements the Backfit Rule

In addition to comments on the merits of the proposed rule, the Commission specifically requested comments on whether the backfit analysis for this rule adequately implements the Backfit Rule, 10 CFR 50.109.

Comments - The Commission received two differing views in response to this request. On one hand, NUMARC expressed the view that the proposed rule does not meet the backfit rule standard because the analysis of the factors set forth in §50.109(c) were not adequately considered by the staff. Specifically, NUMARC stated --

1. Installation and Continuing Costs Associated With the Backfit Have Been Underestimated.

2. Potential Impacts on Radiological Exposure of Facility Employees Should Be Further Addressed.

3. The Relationship to Proposed and Existing Regulatory Requirements Should Be Considered Further.

4. Potential Impacts of Differences in Facility, Type, Design or Age Should Be Considered Further.

5. The Reduction in Risk from Offsite Releases to the Public Has Been Overestimated.

On the other hand, the Ohio Citizens for Responsible Energy (OCRE) and UCS commented that the backfit rule should not apply to the proposed rule. OCRE took the position that "application of the backfit rule to [NRC] rulemakings ... is plainly illegal," and the Commission is not empowered to consider costs to licensees in deciding whether to impose new requirements. UCS commented that the cost benefit analysis should not be applied in this case because safety improvements are needed to secure compliance with existing NRC regulations, specifically General Design Criterion 17, Electric Power Systems (Appendix A to 10 CFR Part 50).

Response - NUMARC's comments on the backfit analysis were taken into account by the staff in revising the draft version of NUREG-1109, and a separate appendix that addresses the factors in §50.109(c) was added to that report. All but Item 2 above are on the same subjects as letters from other commenters and are discussed in more detail under subjects 3 (Item 1), 6 (Item 4), 8 (Item 5), and 17 (Item 3) in this section. NUMARC's Item 2, the potential impact on radiological exposure of facility employees, would need to be assessed in detail only if it were a major factor in the value-impact analysis. The effect of radiological exposure on facility employees, if any, would be extremely small in comparison to the reduction in radiological exposure to the public from accident avoidance. Therefore, this factor would have no impact on the overall value-impact analysis.

Contrary to OCRE's and UCS's comments, the Commission may subject the rulemaking process to internal controls. Moreover, the Commission is empowered to consider the costs of incremental safety improvements which go beyond the level of safety necessary to ensure no undue risk to the public health and safety. See UCS, et al., v. NRC, D.C. Cir. Nos. 85-1757 and 86-1219 (August 4, 1987). The improvements embodied in sec. 50.63 go beyond the level of safety necessary to ensure no undue risk. Finally, contrary to USC's comment on GDC 17, new station blackout measures cannot be imposed on licensees as a matter of compliance with GDC 17, under the compliance exception in the backfit rule, 10 CFR 50.109(a)(4)(i). GDC 17 does not explicitly require that each plant be able to withstand station blackout for a specified time, or that each licensee perform a coping assessment and make whatever modifications may be necessary in the light of that assessment. Nor are any of these highly specific requirements logically compelled by any part of GDC 17. Moreover, GDC 17 has never been interpreted by the staff or the Commission to contain these specific requirements. Thus, to impose them under GDC 17 would amount to a backfit which resulted from a new staff and Commission interpretation of GDC 17.

The issue in this rulemaking is whether some additional protection is warranted beyond that already provided. The Commission is entitled to inquire, and seek public comment, on whether additional safety measures should be imposed where there is a substantial increase in the overall protection of public health and safety and the cost of implementation is justified in view of this increased protection.

3. Cost-Benefit Analysis and Whether §50.63 Meets the "Substantial Increase in the Overall Protection of the Public Health and Safety"

Chairman Zech and Commissioner Roberts requested comments on the analysis of cost benefit, value impact, and safety improvements and the station blackout standing on the overall risk (e.g., is the reduction of risk only a small percentage of the overall risk, or is it a major component of an already small risk?) Chairman Zech and Commissioner Roberts were particularly interested in specific comments assessing whether or not this proposal meets the "substantial

increase in the overall protection of the public health and safety..." threshold now required by the backfit rule.

Comments - (A) One of the major comments by industry on the cost-benefit analysis was that the costs of implementing the proposed requirements have been underestimated. NUMARC and the Atomic Industrial Forum (AIF) commented that the cost estimates for hardware modifications reported in NUREG/CR-3840, "Cost Analysis for Potential Modifications to Enhance the Ability of a Nuclear Plant to Endure Station Blackout," were too low. Commonwealth Edison and other utilities felt that performance of an analysis to determine the maximum duration a nuclear plant could cope with a station blackout would be substantially costlier than what is estimated in NUREG-1109. Industry also expressed concern that the interpretations associated with the proposed rule could lead to substantial costs above those addressed by the NRC staff in its backfit analysis. AIF commented that "The estimate of 120 NRC man-hours per plant [for NRC review] ... appears inadequate to account for technical review and evaluation of the determination of maximum coping capability and of the description of station blackout procedures which the rule would require each licensee to submit."

(B) Several commenters expressed the view that the NRC failed to consider all the risks associated with a station blackout in its value-impact assessment. UCS thought independent failures, in addition to failures that lead to a station blackout, should be included. One individual stated that "both NRC reports [NUREG-1109 and NUREG-1032] are completely deficient in that neither look at sabotage." OCRE commented that seismic events should also be considered.

(C) With respect to safety improvements and overall risk, different points of view were expressed. On one hand, NUMARC commented -- While the risk reduction might be large [for a] limited number of plants, the risk reduction associated with the majority of plants will be small. Thus, as a general matter, the reductions in risk offered by the proposed rule constitue a small percentage of the overall risk, a risk which is already small (and acceptable).

AIF stated that there is no standard by which to conclude that "substantial additional protection will be realized."

A different view was expressed by UCS who stated that "station blackout is clearly a major component of the total risk posed by operating nuclear plants.

The magnitude of the total risk is largely unknowable due to the enormous uncertainty which surrounds probabilistic assessments."

Response - (A) In order to adequately respond to industry's comments above, the staff and NRC contractors reviewed the cost estimates associated with implementing the station blackout rule. Based on this review, the estimated costs for hardware modifications were reviewed and are in the range of from 20 percent to almost 140 percent greater than the estimates in NUREG/CR-3840, depending on the specific modification considered. On average, the cost estimates for hardware backfit were found to be approximately 80 percent greater than estimated in NUREG/CR-3840. However, the cost estimates in NUREG/CR-3840 were not used by the staff in the value-impact analysis in the draft version of NUREG-1109 where estimates approximately 100 percent greater than the NUREG/CR-3840 estimates were used. Therefore, the revised cost estimates used in the final value-impact analysis are not significantly different from the estimates used in the draft version.

Industry's comments on the costs to assess a plant's capability to cope with a station blackout were based on the proposed rule that required an assessment of the maximum coping capability and the potentially unbounded nature of such an assessment. Based on public comments, the Commission has revised the final rule to modify the requirement for licensees to determine the maximum coping capability. (See response to public comments in subject number 9.) Instead, a coping assessment is required only for a specific duration. The cost for such a study is estimated to be from 70 to 100 percent higher than the original estimates by the staff, and these revised costs are used in the final value-impact analysis.

The staff revised its estimate of the resource burden on NRC for review from 120 to 175 person-hours per reactor. This revision was based on technical review required for other comparable NRC activities.

(B) The technical analyses performed for USI A-44 indicated that the contribution to core damage frequency from independent failures, in addition to failures that must occur to get to a station blackout, is low. Likewise, results of USI A-44 studies and other probabilistic risk assessments have shown that, for station blackout sequences, the contribution to core damage frequency from seismic events is low.

Sabotage cannot now be analyzed adequately on a probabilistic basis. Even though sabotage was not explicitly considered in the staff's value-impact analysis, it is discussed in NUREG-1109 under other considerations. These considerations support the conclusion that a station blackout rule will provide a substantial safety benefit.

(C) The revised value-impact analysis performed for the resolution of USI A-44 indicates that there are substantial benefits in terms of reduced core damage frequency and reduced risk to the public that result from the station blackout rule, and the costs are warranted in light of these benefits. The best estimate for the overall value-impact ratio is 2,400 person-rem per million dollars. Even if those plants with the highest risk (and therefore the greatest risk reduction) were not considered, the value-impact ratio for the remaining plants is still favorable (i.e., about 1,500 person-rem per million dollars).

Recent analyses performed for NUREG-1150, "Reactor Risk Reference Document," Draft for Comment, February 1987 indicate that station blackout is a dominant risk contributor to overall residual risk for most of the six plants analyzed. These results support the comment by UCS in response to the Commissioner's request for comments on this subject.

4. Whether NRC Should Require Substantial Improvements in Safety that Go Beyond Those Proposed in this Rulemaking Commissioner Asselstine requested comments on whether the NRC should require substantial improvements in safety with respect to station blackout, like those being accomplished in some other countries, which can be achieved at reasonable cost and which go beyond those proposed in this rulemaking.

Comments - NRC received eight letters that included comments on this subject. Five of these were from the nuclear industry, none of which felt that the approach to station blackout taken in European countries should be used to justify safety improvements that go beyond the proposed §50.63. The main justification for industry's argument is that foreign countries may have reasons for requiring activities that differ from, or exceed, those in the U.S. For example, Washington Public Power Supply Systems (WPPSS) commented, "It is not apparent that the details of U.S. grid stabilities and onsite power reliabilities are substantially similar enough to those found abroad to warrant a simple adoption of these [European] measures."

In another comment from industry on this subject, NUMARC stated that there are several reasons why many of the features for coping with a station blackout in new French nuclear power plants may already exist at most U.S. plants. In fact, they said, "The French approach to station blackout does not appear to depart significantly from current regulatory approaches in the U.S." Similarly, AIF stated, "The assertions of extensive station blackout coping capability at foreign (notably European) nuclear power plants are not sufficiently substantiated to serve as even part of the basis for the proposed requirements."

Three other letters (UCS, OCRE and IDNS) supported the NRC rulemaking to require all plants to be able to cope with a station blackout, but urged the Commission to go beyond the proposed rule. IDNS stated that --

> The goal of holding the expected frequency of core damage from station blackout to 10^{-5} per reactor-year is not sufficiently stringent. With relatively modest modifications to the proposed rule, a frequency of 10^{-7} appears achievable at reasonable cost. Specifically, the rule should require no less than 20 hours decay heat removal capacity instead of only four or eight hours in the proposed rule, in the event of a blackout.

Reponse - The staff agrees with industry's comments that foreign countries may have valid reasons for imposing requirements that differ from or exceed those in the U.S. For example, it appears that there is a higher frequency of losses of offsite power in France than in the U.S. This experience, along with French safety objectives, led the French to design their new standard nuclear power plants to be able to cope with a very long duration station blackout (i.e., up to three days). The French safety approach and their station blackout design features are documented in NUREG-1206, "Analysis of French (Paluel) Pressurized Water Reactor Design Differences Compared to Current U.S. PWR Designs," June 1986.

The Commission believes that the staff has adequately considered foreign approaches in preventing core melt from station blackout in developing the resolution of USI A-44. Although the rule requires plants to be able to cope with station blackout for a specific duration, that duration is not specified in the rule. Guidance to determine an acceptable duration is included in the station blackout regulatory guide. This guidance should apply to most plants, but if there were adequate justification, different requirements (either more or less stringent than the regulatory guide) could be applied to specific plants. The use of alternate AC sources provides a means to achieve further incremental decreases in core melt frequency.

5. The Need for Generic Rulemaking

Comments - Five letters from the nuclear industry commented that generic rulemaking is not necessary to resolve the station blackout issue. Their reasons for this issue were as follows:

> A generic rulemaking is inappropriate since the historic number of sites experiencing a loss of all offsite power is small. (Texas Utilities)
>
> The station blackout issue should be handled on a plant-specific basis and does not need to be resolved by generic rulemaking. Each plant has unique probability for a loss-of-power event based on transmission system, location of plant, and onsite power systems. (Duquesne Light)

> The Commission need not pursue generic rulemaking in order to resolve a non-generic issue. In the proposed station blackout rule, the number of plants of concern is acknowledged to be limited. (NUMARC)
>
> Station blackout has been found not to be a generic issue. Station blackout risk is plant specific and, according to the staff's own analyses, the proposal requirements are expected to result in modifications at no more than a few facilities, if at any. Requiring all licensees to undertake extensive analyses under the provisions of the proposed rules when only a small group of plants may have a need for remedial action is not appropriate. (AIF)

Response - The Commission believes that a rule is appropriate to ensure that station blackout is addressed at all nuclear power plants. The plant-specific features that contribute to risk for station blackout (e.g., diesel generator configuration, probability of loss of offsite power) are considered by the staff in the station blackout regulatory guide to determine an acceptable coping duration for each plant. Even though not all sites have experienced a loss of offsite power, there is not sufficient assurance that such events would not occur in the future. Since historic experience has shown that a total loss of offsite power occurs about once every 10 site-years, and many nuclear plants have operated for less than 10 years, it is not surprising that some plants have experienced a loss of offsite power while others have not.

Even though it is likely that many plants will not need hardware modifications to comply with the rule, the assessment of station blackout coping capability for a specific duration and implementation of associated procedures will effect a safety benefit for all plants. The "limited number of plants of concern" in NUMARC's letter refers to those plants having the highest risk from station blackout (i.e., those that would need hardware modifications). Without a plant specific assessment, these plants can not be identified. Even excluding these plants from consideration, the staff's analysis has shown that the improvements in safety associated with the rule are consistent with backfit considerations set forth in §50.109.

6. Applicability of the Proposed §50.63 to Specific Plants

Comments - Four letters included comments or questions regarding the applicability of the rule to specific plants. For example, does the rule apply to high temperature gas cooled reactors (i.e., Fort St. Vrain)? What about TMI-2 or plants that are near completion but will not have an operating license prior to the amendment's effective date? Houston Power and Lighting Company wrote --

> Proposed Section 50.63 provides schedular guidance for implementing station blackout-related modifications on plants that already hold operating licensees or will be licensed to operate prior to the effective date of the amendment. Plants who may be NTOL's [near-term operating license] but will not be licensed prior to the amendment's effective date should be accorded the same compliance period under parts (c) and (d) of this section. Otherwise this proposed rule could be interpreted to imply that plants not licensed prior to the effective amendment date must comply with the rule and make all necessary modifications prior to receiving an O.L. [operating license]. The rule should be amended to address plants which are scheduled to receive an O.L. within a short time following implementation of this rule.

Response - Rather than identifying specific plants for which the rule does <u>not</u> apply, §50.63(a) specifies when it does apply (i.e., "each light-water-cooled nuclear power plant licensed to operate"). Since Fort St. Vrain is an HTGR, the generic rule would not apply. Station blackout will be considered individually for that plant based on its unique design. Since TMI-2 is not licensed to operate, likewise, the rule would not apply to that plant. Any plant licensed to operate after the date the rule becomes effective will comply with the same 270 day schedule for information submittal applied to plants previously licensed. This affords NTOLs the same compliance features as plants already licensed to operate.

7. Plant-Specific Features and Capabilities

Comments - A number of utilities described plant-specific features and capabilities that reduced the risk posed by a station blackout event compared to the staff's analysis. Examples of such features are given below.

°Availability of alternate, independent ac power sources such as diesel generators, gas turbines, or nearby "black start" ac power sources.

°Extremely reliable offsite power supplies because of multiple right-of-ways or underground feeders to back up above ground transmission lines.

°Dedicated shutdown systems and associated diesel generators to meet the fire protection requirements of Appendix R to 10 CFR Part 50.

°Common or shared systems between two units at multi-unit sites such as dc power, auxiliary feedwater, or diesel generators.

Response - The analyses performed for USI A-44 clearly show that plant-specific features do affect the risk from station blackout, and the station blackout regulatory guide takes this into account in providing guidance on different acceptable coping durations depending on the most significant of these features. Those plants with extremely reliable offsite and onsite ac power supplies need only have a very short (e.g., 2-hour) coping duration to be acceptable. Plants that have a dedicated shutdown system with its own independent power supply could take credit for this system to cope with a station blackout. The final rule and regulatory guide have been clarified to give credit for alternate ac power supplies (see response to subject 11).

Therefore, the Commission believes that for almost all sites, plant-specific differences have been adequately accounted for in the resolution of USI A-44, but the door is open to licensees who believe their plants have additional capability that should be considered by the staff in demonstrating compliance with the rule.

8. The Source Term Used to Estimate Consequences

Comments - Letters from NUMARC and others in the industry commented that the consequences of offsite releases that would result from a station blackout event are overestimated, and new source term information would lead to the prediction of much lower consequences for this event. Several commenters felt that the approach taken by the staff to estimate consequences of a station blackout event -- decreasing the estimated consequences of the SST1 siting source term from NUREG-CR/2723, "Estimates of the Financial Consequences of Nuclear Power Reactor Accidents" (September 1982), by a factor of three -- was improper.

AIF felt that "implementation of any requirements resulting from the resolution of USI A-44 should be deferred until the results of the source term research can be taken into account." They based this statement on the premise that if the consequences used in the staff's value-impact analysis were reduced by a factor of 10, none of the alternatives would be feasible.

UCS expressed a different point of view in their letter which said "... available evidence indicates that the consequences of an accident involving station blackout may be even worse than those estimated either in WASH-1400 or the NRC's more recent studies."

Response - NRC has had an extensive research effort underway since about 1981 to evaluate severe accident source terms. The staff has reviewed the results of this research to take into account the public comments received on this subject. Since there is still a great deal of uncertainty regarding source terms and associated consequences, the staff revised its value-impact analysis for USI A-44 considering a range of estimates for consequences of a station blackout.

The NRC research on severe accident source terms has resulted in the development of significant new analytical tools by NRC contractors, as discussed in NUREG-0956, "Reassessment of the Technical Bases for Estimating Source Terms,"

July 1986. The analytical methods developed, generally referred to as the Source Term Code Package (STCP), have been used to analyze a number of severe accident sequences for five reference plants, namely: Peach Bottom, a BWR Mark I design; Sequoyah, a PWR ice condenser; Surry, a PWR with a sub-atmospheric containment; Grand Gulf, a BWR with a Mark III containment; and Zion, a PWR with a large dry containment (NUREG-1150, "Reactor Risk Reference Document," Draft for Comment, February 1987).

The results of these analyses show that releases from station blackout sequences can be expected to vary significantly depending upon the plant and the specific sequence. Although generalizations are difficult, it appears that calculations using the STCP yield release fractions for most of the sequences range from about one third of an SST1 release (for the case of Surry, without condensation) to roughly one order of magnitude less than this. However, the uncertainties in our present understanding also do not preclude the possibility of a large release, approaching that of the SST1 estimate.

To determine the consequences in terms of person-rem, given the above range of release fractions, data taken from NUREG/CR-2723 indicate that the variations in person-rem associated with releases of magnitude SST1, SST2 and SST3, are virtually identical to the variations in latent cancer fatalities for the same three releases. Hence, the estimated change in latent cancer fatalities with release fractions provides a reliable indication of change in person-rem as well.

Table 10 in NUREG/CR-2723 presents variations in estimated latent cancer fatalities associated with changes in SST1 release fractions (for all elements except noble gases). This table shows that a release fraction of one third of an SST1 release would yield a value of about 50 percent of the latent cancer fatalities (and person-rem) of an SST1 release. Similarly, a release fraction of one third of an SST1 release would yield an estimated person-rem of about 15 percent of that associated with an SST1 release. Consequently, for value-impact calculations, the staff estimated the range of consequences of station

blackout, in terms of person-rem, to be from 0.15 to 0.5 of the estimated person-rem of an SST1 release. As noted, the original value-impact analysis was based on 0.3 times the estimated person-rem of an SST1 release.

With regard to a possible delay in the resolution of USI A-44 until "better" source terms become available, key considerations appear to be when better source terms are likely to become available, and to what degree uncertainties in phenomenology as well as differences between investigators will be resolved. Although research on source terms is expected to continue well into the future, improvements in our knowledge are expected to be largely evolutionary beyond this point, in that the major phenomena appear to have been accounted for, at least in a first-order fashion, both in NRC as well as industry models. Resolution and narrowing of the remaining uncertainties would also benefit from improved experiments and analytical models that are likely to become available gradually. For these reasons, significantly better source terms than those presently available are likely to be forthcoming only after a number of years. Since the range of severe accident source terms and consequences suggested above from estimating station blackout sequences is sufficiently broad to cover likely improvements in source term knowledge, the resolution of USI A-44 should not be delayed.

9. Specificity on the Extent of Required Coping Studies

Comments - Several letters by industry expressed concern that the studies necessary to demonstrate that a plant can cope with a station blackout are not well defined and could potentially be unbounded. These comments focused on two main points. First, the proposed rule required plants to determine the maximum duration the plant could cope with a station blackout, yet the draft regulatory guide included specific guidance on acceptable coping durations (e.g., 4 or 8 hours). Determining the maximum duration, rather than assessing the plant's capability for a specific acceptable duration, could be an open-ended requirement. Along these lines, NUMARC stated --

Unless the required coping demonstration is specifically bounded by clearly stated definitions, assumptions, and criteria, there could conceivably be hundreds of supporting special effects analyses which licensees may have to consider as a result of the exercise of discretion by individual staff reviewers. Under the rule as proposed, licensees cannot ascertain the ultimate requirements they will be expected to meet (including the potential plant modifications they will need to make) to demonstrate compliance.

Second, industry also commented on the potential open-endedness of analyses to determine the operability of equipment in environmental conditions resulting from a station blackout (e.g., without heating, ventilation, and air conditioning). Unless these analyses were well defined, industry felt the analyses could be much more costly than estimated by the staff. However, NUMARC made the following statement relating to the need for detailed prescriptive requirements by NRC that appears to contradict their earlier statement.

The point .... is not that regulations must be prescriptive by their very nature. Prescriptive regulations, which outline in detail exactly what steps are required by licensees to satisfy a proposed regulation, are, in many instances, unnecessary and counterproductive.

Response - With regard to the proposed requirement that each plant determine its maximum duration for coping with station blackout, the staff agrees with the industry comments. First of all, it would be difficult to adequately define "maximum duration" in this sense. Second, if licensees determine that their plants can cope with a station blackout for a specified duration and restore AC power through an acceptable coping analysis, the additional safety benefit gained from simply the knowledge that a longer, or "maximum duration", coping duration exists is small. Third, the costs for assessing "maximum duration" will be higher since more extensive analyses will be required to analyze a transient which would go beyond the coping analysis for a specified duration and recovery from station blackout. Therefore, the rule and

regulatory guide have been revised accordingly to delete the requirement for licensees to determine a plant's maximum coping capability.

With regard to the comments on assessments to determine equipment operability during a station blackout, the staff feels strongly that such assessments are necessary to determine a plant's response to station blackout. By deleting the requirement to determine a plant's "maximum" coping capability, the assessment of equipment operability would not be as costly as assumed by industry. Guidance on acceptable coping assessments is provided in the station blackout regulatory guide. Also, guidelines to evaluate the effects of loss of ventilation under station blackout conditions are provided in Appendix E of NUMARC-8700, Guidelines and Technical Bases for NUMARC Initiatives Addressing Station Blackout at Light Water Reactors. These efforts provide additional definitions, criteria, and standards for licensees' assessments of equipment operability without the need for "prescriptive regulations" by NRC.

In order to evaluate further industry's comments on this subject, NRC requested Sandia National Laboratories to identify specific tasks necessary to determine operability of equipment during a station blackout, and estimate the cost to perform these tasks. Results of this study were used in the revised value-impact analysis performed for this issue ("Equipment Operability During Station Blackout Events," NUREG/CR-4942).

10. Acceptable Duration for Coping with a Station Blackout

Comments - Several comments with differing views were directed at guidance in the draft regulatory guide on acceptable station blackout coping durations in order for plants to comply with the proposed rule.

Washington Public Power Supply commented that "it should be possible for certain utilities to demonstrate [an acceptable] zero hour blackout." One individual recommended "that a 30 minute period be a margin, and that no duration under 4 hours be accepted by the staff." NucleDyne Engineering commented that "advanced reactors should require the capability to safely

withstand a station blackout of at least 8 hours," and IDNS wrote that "the rule should require no less than 20 hours decay heat removal capability instead of only 4 or 8 hours."

Response - Although a diversity of comments was received on this subject, none provided supporting analysis or information to back up the opinions expressed. However, the staff did re-analyze the estimated risk from station blackout events for different plant- and site-related characteristics and revised its guidance on acceptable coping durations accordingly based on a goal of limiting the average contribution to core damage from station blackout to about 10-_ per reactor-year. Most plants would still need a 4- or 8-hour coping capability. Those few plants with the most redundant onsite emergency ac power system, coincident with significantly lower than average expected frequency of loss of offsite power would need only a 2-hour capability to be acceptable. Any plant with minimum redundancy in the onsite emergency ac power system coincident with low reliability and a significantly higher than average expected frequency of loss of offsite power would need to substantially improve its ac power reliability or be able to cope with a station blackout for 16 hours.

11. Credit for Alternate or Diverse AC Power Sources

Comments - Ten letters from the utility industry commented that more credit should be allowed for the availability of alternate power sources such as onsite gas turbines. The two comments below represent the utilities' viewpoint.

> The station blackout rule should be clarified to allow credit for diverse and very reliable offsite power sources or diverse and very reliable onsite electrical generation. (Public Service Company of Colorado)
>
> The option of providing an additional alternate source of ac power is eliminated by [the proposed resolution]. The inconsistency in this approach can best be understood by considering an example at a generic nuclear power station.

> If the licensee were to provide an additional independent diesel generator capable of providing the necessary ac power to prevent station blackout, the licensee ... would still be required to withstand at least 4 hours without ac power. They would receive no credit for the additional diesel generator in the coping analysis. If the licensee were to use that same diesel engine to power a charging pump, even though it would be of less significance to mitigation of reactor core damage than the diesel generator, the licensee could take credit for it in coping with the blackout.
>
> Since a diesel charging pump will not provide for equipment loading flexibility, lighting, ventilation, instrumentation, etc., it is obviously of lower value than an additional source of ac power. The fixed category approach taken in [the proposed resolution], however, will not permit taking credit for the same diesel engine when used as a generator through the actual reliability for the machine is the same. (Toledo Edison)

Response - The proposed resolution did not intend to ignore the alternative of adding additional power sources or taking credit for such sources if they already exist. For example, as specified in the regulatory guide, if a licensee added an emergency diesel generator to one of its plants that had minimum redundancy in the onsite emergency ac power sytem, the acceptable station blackout coping duration could be reduced. For some plants, however, adding a diesel generator would not result in a reduction in the acceptable coping duration, and the point made by Toledo Edison is a valid one. The rule and regulatory guide have been revised to clarify that alternate ac power sources are given credit to cope with a station blackout provided that certain criteria are met (e.g., independence, redundancy, high reliability, maintenance, and testing).

12. Trends on the Reliability of AC Power Sources

Comments - Five letters included comments on the reliability of ac power sources. Four letters from industry felt that improved ac power reliability

should be factored into the staff's technical analysis. Examples of these comments include the following:

> "... the frequency of loss of offsite power activities has been decreasing..." (Washington Public Power Supply System);

> "... offsite power availability in the absence of regulation has significantly improved over the past decade." (Southern California Edison Company);

> "[NUREG/CR-4347] ... shows an improvement in diesel generator reliability over that shown in the earlier document [NUREG/CR-2989]." (General Electric); and

> "Typically the reliability of onsite power systems increases during the first few years following startup." (Gulf States Utilities)

IDNS on the other hand felt that potential vulnerabilities still exist in onsite emergency ac power systems, and licensees should demonstrate that they have taken steps to reduce the probability of loss of ac power.

Response - The staff and its contractors have extensively analyzed the industry experience and trends in ac power reliability as documented in NUREG-1032, NUREG/CR-2989, NUREG/CR-3992, and NUREG/CR-4347. Trends have shown that two aspects of ac power reliability have improved somewhat -- the reduced frequency of losses of offsite power due to plant-centered events, and a slight improvement in average diesel generator reliability from 1976 through 1983. These factors have been taken into account in the staff's analyses and the resolution of USI A-44. However, data also demonstrate that there are practical limits on ac power reliability, and the defense-in-depth approach of being able to cope with a station blackout is warranted.

13. Sharing of Emergency Diesel Generators Between Units at Multi-Unit Sites.

Comments - Several letters from industry stated that some plants with two units on a site have the capability to crosstie electrical buses between units and therefore have improved flexibility in providing ac power. Since the magnitude of the electrical loads necessary to provide core cooling during a station blackout is significantly less than that required for a design basis accident,

it could be possible to provide ac power to both units at the site using only a single diesel generator.

Response - The proposed rule and draft regulatory guide do not prohibit the approach discussed above. If licensees can demonstrate that such crosstie capability exists; procedures are in place to accomplish the crosstie and shed nonessential loads, if necessary; and no NRC regulations are violated (such as separation, minimum redundancy and independence), then credit would be given for this capability as shown in the station blackout regulatory guide (e.g., reduced acceptable station blackout coping durations for greater diesel generator redundancy).

14. Clarification of the Definitions of Station Blackout and Diesel Generator Failures.

Comments - (A) Three letters from the utility industry recommended that the definition of station blackout in §50.2 should be clarified to exclude ac power from the station batteries through inverters. This source of ac power from the station batteries would be available in the event of a loss of both the offsite and onsite emergency ac power sources (i.e., diesel generators).

(B) Several industry letters commented that the definition of diesel generator failure should be clarified, particularly with respect to the treatment of short-term failures that can be recovered quickly. A letter from Sargent and Lundy Engineers commented that --

> A definition of failure on demand for emergency diesel generators needs to be provided. Under the context of a station blackout, a diesel generator which fails to start automatically upon detection of an offsite power loss, but is successfully started manually from the main control room or from the local control panel, should not be considered a failure on demand.

Response - (A) The staff agrees with comment A and revised the definition of station blackout accordingly.

(B) Based on actual experience, failures of diesel generators to start due to failures in the auto-start system make up less than 20 percent of all diesel generator failures. Therefore, discounting these failures would not have a significant impact of overall diesel generator reliability statistics. However, the staff agrees in principle with comment B and has clarified the station blackout regulatory guide so that auto-start failures of diesel generators need not be counted in determining the failure rate if the diesel generator is capable of being started manually immediately after it does not start automatically.

15. Specificity and Clarification of Requirements

Comments - Public comments were received regarding the specificity and clarification of the proposed rule and draft regulatory guide. These ranged from general to specific comments as the following two excerpts indicate:

> We are concerned that, if the proposed rule is adopted, the staff will promulgate regulatory guidance criteria which will be unrealistic and excessive, i.e., compounding the event with other accidents, imposing passive failure criteria, applying seismic, environmental qualification and other qualifications to equipment that could otherwise be used in response to such an event, etc. (Maine Yankee Atomic Power Company)

> Definitions of P1 and P2 [in Table 3 of the draft Regulatory Guide] use frequency of extremely severe weather and severe weather interchangeably, thus creating confusion in the definition. (Washington Public Supply System)

Response - Some of the comments on this subject relate to other subjects discussed elsewhere in this section. Some comments were quite specific while others were general in nature or expressed views that were not substantiated

with backup material. The staff has taken these comments into consideration and revised and clarified the rule and regulatory guide accordingly. Additional guidance is provided in NUMARC-8700 which has been reviewed by the staff and referenced in the regulatory guide as providing a method the staff finds acceptable for meeting the rule.

16. Technical Comments on NUREG-1032

Comments - In addition to comments on the proposed rule and draft regulatory guide, several letters contained comments on the staff's draft technical report, NUREG-1032, "Evaluation of Station Blackout Accidents at Nuclear Power Plants."

Response - NUREG-1032 was issued in draft form for public comment in May 1985 (50 FR 24332). The comments received were reviewed and considered by the staff and resulted in a re-evaluation of the technical analysis. Details of the specific comments and responses are not presented here. Rather, NUREG-1032 was revised extensively over the past year to address the public comments. In general, the overall conclusions on the risk from station blackout events did not change significantly as a result of the re-analysis. One of the major changes resulting from the re-analysis was a revision to the definitions of plant characteristics, especially the clustering of plants into site and weather-related groups (Appendix A in NUREG-1032). These changes are reflected in revisions to the guidance in the station blackout regulatory guide to determine plant-specific acceptable station blackout coping durations.

17. Relationship of USI A-44 to Other NRC Generic Issues

Comments - The major public comment regarding the relationship of USI A-44 to other NRC generic safety issues was that the proposed rule may not be necessary or should be postponed because of ongoing work to resolve related generic issues. Some comments were general in nature such as the following one from Southern California Edison Company:

> Promulgation of a final station blackout rulemaking at this time will unnecessarily complicate the final resolution of related generic technical issues... The NRC must develop and implement a program to coordinate the resolution of all power-related generic issues prior to finalizing any individual proposed rule.

> AIF suggested that the implementation of any requirements for station blackout be deferred until the requirements from USI A-45, Shutdown Decay Heat Removal Requirements, are known and until the effect of source term changes can be evaluated.

NUMARC mentioned specific proposed and existing regulatory requirements that should be considered because they could reduce the need for a station blackout rule (e.g., B-56, Diesel Generator Reliability and GI 23, Reactor Coolant Pump Seal Failures). Other related issues mentioned in the public comments were A-30, Adequacy of Safety Related DC Power Supplies, and implementation of safe shutdown facilities to meet the fire protection requirements of Appendix R.

Response - The question that needs to be addressed is "should a requirement be imposed now to reduce risk, or should it be postponed until related issues are resolved sometime in the future?" Potentially, this could result in substantial delays and thereby not resolving generic safety issues in a timely manner. The staff has considered the resolution of USI A-44 in light of the related issues mentioned in the comments. Although these issues are identified as separate tasks within NRC, they are all managed in a well established program that coordinates all related issues. A brief discussion of the most relevant issues is presented below. (Additional information is provided in NUREG-1109, "Regulatory Analysis for the Resolution of Unresolved Safety Issue A-44, Station Blackout.")

Resolution of USI A-45 will occur at some time following issuance of the station blackout (SBO) rule (§50.63) and after plant specific SBO coping evaluations have been performed by licensees per NUMARC/NUGSBO Initiative 5, utilizing guidelines provided in NUMARC-8700. Further, the resolution of USI A-45 is expected to be highly plant specific and focused on loss of decay heat removal considerations from other causes beyond SBO. Utilization will be made of A-44 evaluations (as applicable) and any plant equipment modification needs

identified from A-45 will be carefully evaluated to maximize effective use of previously identified A-44 equipment needs.

Maintaining emergency diesel generator reliability, the purpose of B-56, is an integral part of the resolution of USI A-44. However, the Commission believes that additional defense-in-depth will achieve a substantial increase in protection to public health and safety.

The resolution of GI 23 (reactor coolant pump seal leakage) deals with loss of RCS inventory and associated degraded core conditions. USI A-44 deals with SBO induced effects, which result in loss of AC power, thereby impacting a broader spectrum of plant equipment and safety related functions. Although the resolution of GI 23 will contribute to establishing a higher level of assurance that seal leakage will be minimized (thereby minimizing the need for power to replace water inventory losses over the SBO duration and recovery phase), resolution of GI 23 by itself will not address the broader scope of USI A-44 safety concerns.

Some licensees have implemented dedicated shutdown systems that are independent of normal and emergency ac power to meet Appendix R requirements. If applicable, these features would be credited in the resolution of USI A-44 by providing the capability to cope with a station blackout.

Thus, the resolution of USI A-44 is coordinated with related generic issues, and implementation of a final resolution should not be delayed further. (Response to comments on the effect of source term changes is included in subject number 8.)

18. An Alternative of Plant-Specific Probabilistic Assessments

Comments - Several utilities suggested that, in lieu of the requirements in the rule, licensees should be permitted to submit plant-specific evaluations to demonstrate that the frequency of core damage from station blackout events is 10^{-5} per reactor-year or less. In a similar vein, the suggestion was made that

NRC should specify a target level of reliability for ac power systems in order to satisfy NRC's criteria for core damage frequency. A few licensees submitted limited probabilistic assessments to show that for some plants station blackout could have a very small probability of severe consequences.

Response - The Commission does not preclude licensees from submitting plant specific probabilistic assessments to support a determination that station blackout would have a very small probability for causing core damage. However, the requirements of the rule must be met. The Commission would observe that the use of probabilistic assessments was important as input to the regulatory decisionmaking that culminated in the station blackout rule, and related guidance. As expressed in the Commission's Safety Goal policy statement of August 1986, the Commission has acquired a reasonable degree of confidence about the usefulness and value of probabilistic assessments in assisting regulatory decisionmaking on complex safety issues. In short, such assessments are of value in complementing and focusing the more traditional and deterministic defense-in-depth approaches. On the other hand, any licensee must decide whether or not his plant specific AC power configuration and other related equipment is sufficiently unique to merit the conduct and submittal of a probabilistic assessment as part of achieving compliance to §50.63. The Commission's experience also indicates that probabilistic assessments are resource intensive and these can be of marginal utility if their only end result is to delay rule compliance.

19. Procedures and Operator Actions During Station Blackout

Comments - (A) Several letters from industry commented that, in response to Generic Letter 81-04, "Emergency Procedures and Training for Station Blackout Events," dated February 21, 1981, utilities already have procedures in place to prepare plant operations for station blackout events. Owners groups have established generic guidance for station blackout operating procedures for licensees to use in developing plant-specific procedures. A representative of the Professional Reactor Operator Society, commented that --

> Generic procedures are used by most operating facilities. These procedures are not carried into adequate depth of specific power plant operations. The industry has relied too heavily on generic procedures and has not given a real look at what specific steps must be taken. Extrapolation of these procedures must be required. Specific maintenance procedures must be established and followed.

(B) Other comments on procedures related to the timeliness of operator actions, both inside and outside the control room. Houston Lighting and Power suggested that --

> In Section 3.1 (Part 6), [of the regulatory guide] the first sentence should be revised to read, 'Consideration should be given to timely operator actions <u>both inside and outside of the control room</u> that ...', so that credit can be taken for existing equipment that may not have actuation and control from the control room.

Illinois Power Company recommended that --

> ... Section C.3.3, Item 3.a, of the proposed regulatory guide should be modified to read:
>
> a. The system should be capable of being actuated and controlled from the control room, or if other means of control are required (e.g., manual jumping of control logics or manual operation of valves), it should be demonstrated that these steps can be carried out in a timely fashion.

Response - (A) Licensees may take credit for station blackout procedures already in place to comply with the station blackout rule. However, for the most part, these procedures were developed without having the benefit of a plant-specific assessment to determine whether a plant could withstand a station blackout for a specific duration. Therefore, these procedures may need to be modified after licensees have determined an acceptable station blackout coping duration and evaluated their plant's response to a station blackout of this duration.

(B) The staff agrees with the comments related to operator actions outside the control room, and the regulatory guide was revised accordingly.

20. Schedule Provisions in the Proposed §50.63

Comments - Two letters contained comments on the proposed schedule in §50.63. OCRE felt the scheduling provisions in the proposed rule were far too generous. One individual recommended that the schedule be modified to require licensees to submit, within 9 months of the date of the amendment, a list of modifications along with a proposed schedule to implement those modifications. (According to the proposed rule, licensees would not have to submit a schedule for implementing equipment modifications until after the staff received and reviewed licensees' submittals on their plant's acceptable station blackout duration.)

Response - The staff agreed in part with these comments, and the schedule was revised accordingly. §50.63(c)(1)(iv) now requires that licensees submit within 9 months after the rule is issued a list of equipment modifications and a proposed schedule for implementing them. A final schedule would be developed after NRC has reviewed the licensees' submittal of their plant's acceptable station blackout duration.

21. Industry Initiatives

Comments - In addition to comments on the proposed rule, NUMARC endorsed the following five initiatives* to address the more important contributors to station blackout:

1. Each utility will review their site(s) against the criteria specified in NUREG-1109, and if the site(s) fall into the category of an eight-hour site after utilizing all power sources available, the utility will take actions to reduce the site(s) contribution to the overall risk of station blackout. Non-hardware changes will be made within one year. Hardware changes will be made within a reasonable time thereafter.

2. Each utility will implement procedures at each of its site(s) for:

   a. coping with a station blackout event

   b. restoration of ac power following a station blackout event, and

   c. preparing the plant for severe weather conditions (e.g., hurricanes and tornados) to reduce the likelihood and consequences of a loss of offsite power and to reduce the overall risk of a station blackout event.

3. Each utility will, if applicable, reduce or eliminate cold fast-starts of emergency diesel generators for testing through changes to technical specifications or other appropriate means.

4. Each utility will monitor emergency ac power unavailability utilizing data utilities provide to INPO on a regular basis.

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* NUMARC initially proposed a set of four initiatives. The fifth initiative regarding the performance of a coping assessment was provided in NUMARC-8700, which was submitted by letter from J. Opeka (NUMARC) to T. Speis (RES) dated November 23, 1987.

5. Each utility will assess the ability of its plant(s) to cope with a station blackout. Plants utilizing alternate AC power for station blackout response which can be shown by test to be available to power the shutdown busses within 10 minutes of the onset of station blackout do not need to perform any coping assessment. Remaining alternate AC plants will assess their ability to cope for 1-hour. Plants not utilizing an alternate AC source will assess their ability to cope for 4-hours. Factors identified which prevent demonstrating the capability to cope for the appropriate duration will be addressed through hardware and/or procedural changes so that successful demonstration is possible.

NUMARC previously opposed generic rulemaking and felt that the first four initiatives would resolve the station blackout issue.

Response - These five initiatives now include many of the elements that are included in the NRC resolution of USI A-44. The staff has followed up on the NUMARC initiatives through a series of meetings in 1986 through 1987. The result has been the development of NUMARC-8700 which provides guidelines and criteria acceptable to the staff. The procedures in NUMARC-8700 has been referenced in Regulatory Guide 1.155 as providing guidance acceptable to the staff for meeting the rule. Table 1 in Regulatory Guide 1.155 provides a cross-reference to NUMARC-8700 and notes where the regulatory guide takes precedence. NUMARC's previous concerns have been addressed in the development of RG 1.155 and NUMARC-8700.

Finding of no Significant Environmental Impact Availability

The Commission has determined under the National Environmental Policy Act of 1969, as amended, and the Commission's rules in Subpart A of CFR Part 51, that this rule is not a major Federal action significantly affecting the quality of the human environment, and therefore, an environmental impact statement is not required. There are not any adverse environmental impacts as a result of the rule because there is no additional radiological exposure to the general public or plant employees, and plant shutdown is not required so there are no additional environmental impacts as a result of the need for replacement power. The environmental assessment and finding of no significant impact on which this determination is based are available for inspection and copying for a fee at the NRC Public Document Room 1717 H Street, NW, Washington, DC. Single copies of the environmental assessment and the finding of no significant impact are

available from Mr. Warren Minners, Office of Nuclear Regulatory Research, U. S. Nuclear Regulatory Commission, Washington, DC 20555, Telephone: (301) 492-7827.

Paperwork Reduction Act Statement

This final rule amends information collection requirements that are subject to the Paperwork Reduction Act of 1980 (44 U.S.C. 3501 et seq.). These requirements were approved by the Office of Management and Budget approval number 3150-0011.

Regulatory Analysis

The Commission has prepared a regulatory analysis on this final regulation. The analysis examines the costs and benefits of the alternatives considered by the Commission. A copy of the regulatory analysis, NUREG-1109, "Regulatory/ Backfit Analysis for the Resolution of Unresolved Safety Issue A-44, Station Blackout is available for inspection and copying for a fee at the NRC Public Document Room, 1717 H Street, NW, Washington, DC 20555. Copies of NUREG-1109 may be obtained by writing the Distribution Section, Room P-1304, Division of Information Support Services, U. S. Nuclear Regulatory Commission, Washington, DC 20555.

Regulatory Flexibility Certification

As required by the Regulatory Flexibility Act (5 U.S.C. 605(b)), the Commission certifies that this rule does not have a significant economic impact on a substantial number of small entities. The rule requires that nuclear power plants be able to withstand a total loss of ac power for a specified time duration and maintain reactor core cooling during that period. These facilities are licensed under the provisions of 10 CFR 50.21(b) and 10 CFR 50.22. The companies that own these facilities do not fall within the scope of "small entities" as set forth in the Regulatory Flexibility Act or the small business size standards set forth in regulations issued by the Small Business Administration in 13 CFR Part 121.

List of Subjects in 10 CFR Part 50

Antitrust, Classified information, Fire prevention, Incorporation by reference, Integovernmental relations, Nuclear power plants and reactors, Penalty, Radiation protection, Reactor siting criteria, Reporting and recordkeeping requirements.

For the reasons set out in the preamble and under the authority of the Atomic Energy Act of 1954, as amended, the Energy Reorganization Act of 1974, as amended, and 5 U.S.C. 553, the NRC is adopting the following amendments to 10 CFR Part 50.

Part 50 - Domestic Licensing of Production and Utilization Facilities

1. The authority citation for Part 50 is revised to read as follows:
AUTHORITY: Secs. 102, 103, 104, 105, 161, 182, 183, 186, 189, 68 Stat. 936, 937, 938, 948, 953, 954, 955, 956, as amended, sec. 234, 83 Stat. 1244, as amended (42 U.S.C. 2132, 2133, 2134, 2135, 2201, 2232, 2233, 2236, 2239, 2282); secs. 201, as amended, 202, 206, 88 Stat. 1242, as amended, 1244, 1246, (42 U.S.C. 5841, 5842, 5846).

Section 50.7 also issued under Pub. L. 95-601, sec. 10, 92 Stat. 2951 (42 U.S.C. 5851). Sections 50.10 also issued under secs. 101, 185, 68 Stat. 936, 955, as amended (42 U.S.C. 2131, 2235); sec 102, Pub. L. 91-190, 83 Stat. 853 (42 U.S.C. 4332). Sections 50.23, 50.35, 50.55, 50.56 also issued under sec. 185, 68 Stat. 955 (42 U.S.C. 2235). Sections 50.33a, 50.55a and Appendix Q also issued under sec. 102, Pub. L. 91-190, 83 Stat. 853 (42 U.S.C. 4332) Sections 50.34 and 50.54 also issued under sec. 204, 88 Stat. 1245 (42 U.S.C. 5844). Sections 50.58, 50.91, and 50.92 also issued under Pub. L. 97-415, 96 Stat. 2073 (42 U.S.C. 2239). Section 50.78 also issued under sec. 122,68 Stat. 939 (42 U.S.C. 2152). Sections 50.80-50.81 also issued under sec. 184, 68 Stat. 954, as amended (42 U.S.C. 2234). Section 50.103 also issued under sec. 108, 68 Stat. 955 (42 U.S.C. 2237).

For the purposes of sec. 223, 68 Stat. 958, as amended (42 U.S.C. 2273); 50.10(a), (b), and (c) and 50.44, 50.46, 50.48, 50.54, and 50.80(a) are issued under sec. 161b, 68 Stat. 948, as amended (42 U.S.C. 2201(b)); 50.10(b) and (c), and 50.54 are issued under sec. 161i, 68 Stat. 949, as amended (42 U.S.C. 2201(i)); and 50.55(e), 50.59(b), 50.70, 50.71, 50.72, 50.73, and 50.78 are issued under sec. 161o, 68 Stat. 950, as amended (42 U.S.C. 2201(o)).

2. In §50.2, definitions of "alternate ac source" and "station blackout" are added in the alphabetical sequence to read as follows:

§50.2 Definitions

"Alternate ac source" means an alternating current (ac) power source that is available to and located at or nearby a nuclear power plant and meets the following requirements: (i) is connectable to but not normally connected to the offsite or onsite emergency ac power systems, (ii) has minimum potential for common mode failure with offsite power or the onsite emergency ac power sources, (iii) is available in a timely manner after the onset of station blackout, (iv) has sufficient capacity and reliability for operation of all systems required for coping with station blackout and for the time required to bring and maintain the plant in safe shutdown (non-DBA).

"Safe shutdown (non-DBA(design basis accident))" for station blackout means bringing the plant to those shutdown conditions specified in plant technical specifications as Hot Standby or Hot Shutdown, as appropriate (plants have the option of maintaining the RCS at normal operating temperatures or at reduced temperatures).

"Station blackout" means the complete loss of alternating current (ac) electric power to the essential and nonessential switchgear buses in a nuclear power plant (i.e., loss of offsite electric power system concurrent with turbine trip and unavailability of the onsite emergency ac power system). Station blackout does not include the loss of available ac power to buses fed by station batteries through inverters or by alternate ac sources as defined in this section, nor does it assume a concurrent single failure or design basis accident. At single unit sites, any emergency ac power source(s) in excess of the number required to meet minimum redundancy requirements (i.e., single failure) for safe shutdown (non-DBA) is assumed to be available and may be designated as an alternate power source(s) provided the applicable requirements are met. At multi-unit sites, where the combination of emergency ac power

sources exceeds the minimum redundancy requirements for safe shutdown (non-DBA) of all units, the remaining emergency ac power sources may be used as alternate ac power sources provided they meet the applicable requirements. If these criteria are not met, station blackout must be assumed on all the units.

3. A new §50.63 is added to read as follows:

§50.63 Loss of all alternating current power.

(a) Requirements. Each light-water-cooled nuclear power plant licensed to operate must be able to withstand for a specified duration and recover from a station blackout as defined in §50.2. The specified station blackout duration shall be based on the following factors: (1) the redundancy of the onsite emergency ac power sources, (2) the reliability of the onsite emergency ac power sources, (3) the expected frequency of loss of offsite power, and (4) the probable time needed to restore offsite power. The reactor core and associated coolant, control, and protection systems, including station batteries, and any other necessary support systems, shall provide sufficient capacity and capability to assure that the core is cooled and appropriate containment integrity is maintained in the event of a station blackout for the specified duration. The capability for coping with a station blackout of specified duration shall be determined by an appropriate coping analysis. Utilities are expected to have the baseline assumptions, analyses and related information used in their coping evaluations available for NRC review.

(b) Limitation of Scope. Paragraphs (c) and (d) of this section do not apply to those plants licensed to operate prior to [insert the effective date of this amendment], if the capability to withstand station blackout was specifically addressed in the operating license proceeding and was explicitly approved by the NRC.

(c) Implementation

(1) Information Submittal: For each light-water-cooled nuclear power plant licensed to operate on or before [insert the effective date of this amendment], the licensee shall submit the information defined below to the Director of the Office of Nuclear Reactor Regulation by [insert a date 270 days after the effective date of this amendment]: For each light-water-cooled nuclear power plant licensed to operate after the date of this amendment, the same 270 day schedule for information submittal applies after the date of license issuance.

(i) A proposed station blackout duration to be used in determining compliance with paragraph (a) of this section, including a justification for the selection based on the four factors identified in paragraph (a) of this section.

(ii) A description of the procedures that have been established for station blackout events for the duration determined in paragraph (c)(1)(i) of this section and for recovery therefrom; and

(iii) A list of modifications to equipment and associated procedures necessary, if any, to meet the requirements of paragraph (a) of this section, for the specified station blackout duration determined in paragraph (c)(1)(i) of this section, and a proposed schedule for implementing the stated modifications.

(2) Alternate ac source: An alternate ac power source(s) as defined in §50.2 will constitute acceptable capability to withstand station blackout provided an analysis is performed which demonstrates that the plant has this capability from onset of the station blackout until the alternate ac source(s) and required shutdown equipment are started and lined up to operate. The time required for startup and alignment of the alternate ac power source(s) and this equipment shall be demonstrated by test. An alternate ac source(s) serving a multiple unit site where onsite emergency

ac sources are not shared between units shall have, as a minimum, the capacity and capability for coping with a station blackout in any of the units; at sites where onsite emergency ac sources are shared between units, the alternate ac source(s) shall have the capacity and capability as required to assure that all units can be brought to and maintained in safe shutdown (non-DBA) as defined in §50.2. If the alternate ac source(s) meets the above requirements and can be demonstrated by test to be available to power the shutdown busses within 10 minutes of the onset of station blackout, then no coping analysis is required.

(3) Regulatory Assessment: After consideration of the information submitted in accordance with paragraph (c) (1) of this section, the Director, Office of Nuclear Reactor Regulation, will notify the licensee of the Director's conclusions regarding the adequacy of the proposed specified station blackout duration, the proposed equipment modifications and procedures and the proposed schedule for implementing the procedures and modifications for compliance with paragraph (a) this section.

(4) Implementation Schedule: For each light-water-cooled nuclear power plant licensed to operate on or before [insert the effective date of this amendment], the licensee shall, within 30 days of the notification provided in accordance with paragraph (c) (3) of this section, submit to the Director of the Office of Nuclear Reactor Regulation a schedule commitment for implementing any equipment and associated procedure modifications necessary to meet the requirements of paragraph (a) of this section. This submittal must include an explanation of the schedule and a justification if the schedule does not provide for completion of the modifications within two years of the notification provided in accordance with paragraph (c)(3) of this section. A final schedule for implementing modifications necessary to comply with the requirements of paragraph (a) of this section shall be established by the NRC staff in consultation and coordination with the licensee.

Dated at Washington, DC, this ____ day of ________ 1988.

For the Nuclear Regulatory Commission.

Samuel J. Chilk
Secretary of the Commission.

BACKFIT ANALYSIS

Analysis and Determination That The Rulemaking to Amend 10 CFR 50. Concerning Station Blackout Complies With The Backfit Rule 10 CFR 50.109

The Commission's existing regulations establish requirements for the design and testing of onsite and offsite electrical power systems (10 CFR Part 50, Appendix A, General Design Criteria 17 and 18). However, as operating experience has accumulated, the concern has arisen regarding the reliability of both the offsite and onsite emergency ac power systems. These systems provide power for various safety systems including reactor core decay heat removal and containment heat removal which are essential for preserving the integrity of the reactor core and the containment building, respectively. In numerous instances emergency diesel generators have failed to start and run during tests conducted at operating plants. In addition, a number of operating plants have experienced a total loss of offsite electric power, and more such occurrences are expected. Existing regulations do not require explicitly that nuclear power plants be designed to withstand the loss of all ac power for any specified period.

This issue has been studied by the staff as part of Unresolved Safety Issue (USI) A-44, "Station Blackout." Both deterministic and probabilistic analyses were performed to determine the timing and consequences of various accident sequences and to identify the dominant factors affecting the likelihood of core melt accidents from station blackout. Although operational experience shows that the risk to public health and safety is not undue, these studies which have evaluated plant design features and site dependant factors in detail show that blackout can be a significant contributor to the overall residual risk. Consequently, the Commission is amending its regulations to require that plants be capable of withstanding a total loss of ac power for a specified duration and to maintain reactor core cooling during that period.

An analysis of the benefits and costs of implementing the station blackout rule is presented NUREG-1109, "Regulatory/Backfit Analysis for the Resolution

of Unresolved Safety Issue A-44, Station Blackout."_ The estimated benefit from implementing the station blackout rule is a reduction in the frequency of core damage per reactor-year due to station blackout and the associated risk of offsite radioactive releases. The risk reduction for 100 operating reactors is estimated to be 145,000 person-rems and supports the Commission's conclusion that §50.63 provides a substantial improvement in the level of public health and safety protection.

The cost for licensees to comply with the rule would vary depending on the existing capability of each plant to cope with a station blackout, as well as the specified station blackout duration for that plant. The costs would be primarily for licensees to assess the plant's capability to cope with a station blackout, (2) to develop procedures, (3) to improve diesel generator reliability if the reliability falls below certain levels, and (4) to retrofit plants with additional components or systems, as necessary, to meet the requirements.

The estimated total cost for 100 operating reactors to comply with the resolution of USI A-44 is about $60 million. The average cost per reactor would be around $600,000, ranging from $350,000, if only a station blackout assessment and procedures and training are necessary, to a maximum of about $4 million if substantial modifications are needed, including requalification of a diesel generator.

The overall value-impact ratio, not including accident avoidance costs, is about 2,400 person-rems averted per million dollars. If the net cost, which includes the cost savings from accident avoidance (i.e., cleanup and repair of onsite damages and replacement power following an accident) were used, the

_Draft NUREG-1109 was issued for public comment in January 1986. Copies of this report are available for inspection and copying for a fee at the NRC Public Document Room, 1717 H Street, NW, Washington, DC 20555.

overall value-impact ratio would improve significantly to about 6,100 person-rems averted per million dollars. These values, which exceed the $1000/person-rem interim guidance provided by the Commission, support proceeding with the implementation of §50.63.

The preceding quantitative value-impact analysis was one of the factors considered in evaluating the rule, but other factors also played a part in the decision-making process. Probabilistic risk assessment (PRA) studies performed for this USI, as well as some plant-specific PRAs, have shown that station blackout can be a significant contributor to core melt frequency, and, with consideration of containment failure, station blackout events can represent an important contributor to reactor risk. In general, active systems required for containment heat removal are unavailable during station blackout. Therefore, the offsite risk is higher from a core melt resulting from a station blackout than it is from many other accident scenarios.

Although there are licensing requirements and guidance directed at providing reliable offsite and onsite ac power, experience has shown that there are practical limitations in ensuring the reliability of offsite and onsite emergency ac power systems. Potential vulnerabilities to common cause failures associated with design, operational, and environmental factors can affect ac power system reliability. For example, if potential common cause failures of emergency diesel generators exist (e.g., in service-water or dc power support systems), then the estimated core damage frequency from station blackout events can increase significantly. Also, even though recent data indicate that the average emergency diesel generator reliability has improved slightly since 1976, these data also show that diesel generator failure rates during unplanned demand (e.g., following a loss of offsite power) were higher than that during surveillance tests.

The estimated frequency of core damage from station blackout events is directly proportional to the frequency of the initiating event. Estimates of station blackout frequencies for this USI were based on actual operational experience with credit given for trends showing a reduction in the frequency of losses of offsite power resulting from plant-centered events. This is assumed to be a

realistic indicator of future performance. An argument can be made that the future performance will be better than the past. For example, when problems with the offsite power grid arise, they are fixed and, therefore, grid reliability should improve. On the other hand, grid power failures may become more frequent because fewer plants are being built, and more power is being transmitted among regions, thus placing greater stress on transmission lines.

A number of foreign countries, including France, Britain, Sweden, Germany and Belgium, have taken steps to reduce the risk from station blackout events. These steps include adding design features to enhance the capability of the plant to cope with a station blackout for a substantial period of time and/or adding redundant and diverse emergency ac power sources.

The factors discussed above support the determination that additional defense in-depth provided by the ability of a plant to cope with station blackout for a specific duration would provide substantial increase in the overall protection of the public health and safety, and the direct and indirect costs of implementation are justified in view of this increased protection. The Commission has considered how this backfit should be prioritized and scheduled in light of other regulatory activities ongoing at operating nuclear power plants. Station blackout warrants a high priority ranking based on both its status as an "unresolved safety issue" and the results and conclusions reached in resolving this issue. As noted in the implementation section of the rule (§50.63(c)(4)), the schedule for equipment modification (if needed to meet the requirements of the rule) shall be established by the NRC staff in consultation and coordination with the licensee. Modifications that cannot be scheduled for completion within two years after NRC accepts the licensee's specified station blackout duration must be justified by the licensee. The NRC retains the authority to determine the schedules for modifications.

Analysis of 50.109(c) Factors

1. Statement of the specific objectives that the backfit is designed to achieve

   The NRC staff has completed a review and evaluation of information developed since 1980 on Unresolved Safety Issue (USI) A-44, Station Blackout. As a result of these efforts, the NRC is amending 10 CFR Part 50 by adding a new § 50.63, "Station Blackout".

   The objective of the station blackout rule is to reduce the risk of severe accidents associated with station blackout by making station blackout a relatively small contributor to total core damage frequency. Specifically, the rule requires all light-water-cooled nuclear power plants to be able to cope with a station blackout for a specified duration and to have procedures and training for such an event. A regulatory guide, to be issued along with the rule, provides an acceptable method to determine the station blackout duration for each plant. The duration is to be determined for each plant based on a comparison of the individual plant design with factors that have been identified as the main contributors to risk of core melt resulting from station blackout. These factors are (1) the redundancy of onsite emergency ac power sources, (2) the reliability of onsite emergency ac power sources, (3) the frequency of loss of offsite power, and (4) the probable time needed to restore offsite power.

2. General description of the activity required by the licensee or applicant in order to complete the backfit

   In order to comply with the resolution of USI A-44, licensees will be required to --

   ° Maintain the reliability of onsite emergency ac power sources at or above specified acceptable reliability levels.

° Develop procedures and training to restore ac power using nearby power sources if the emergency ac power system and the normal offsite power sources are unavailable.

° Determine the duration that the plant should be able to withstand a station blackout based on the factors specified in §50.63, "Station Blackout" and Regulatory Guide 1.155, "Station Blackout."

° If available, an alternate ac power source, that meets specific criteria for independence and capacity, can be used to cope with a station blackout.

° Evaluate the plant's actual capability to withstand and recover from a station blackout. This evaluation includes:

- Verifying the adequacy of station battery power, condensate storage tank capacity, and plant/instrument air for the station blackout duration.

- Verifying adequate reactor coolant pump seal integrity for the station blackout duration so that seal leakage due to lack of seal cooling would not result in a sufficient primary system coolant inventory reduction to lose the ability to cool the core.

- Verifying the operability of equipment needed to operate during a station blackout and the recovery from the blackout for environmental conditions associated with total loss of ac power (i.e., loss of heating, ventilation and air conditioning).

° Depending on the plant's existing capability to cope with a station blackout, licensees may or may not need to backfit hardware modifications (e.g., adding battery capacity) to comply with the rule. (See item 8 of this analysis for additional discussion.) Licensees will be required to have procedures and training to cope with and recover from a station blackout.

3. Potential change in the risk to the public from the accidental offsite release of radioactive material.

   Implementation of the station blackout rule will result in an estimated total risk reduction to the public ranging from 65,000 to 215,000 person-rems with a best estimate of about 145,000 person-rem.

4. Potential impact on radiological exposure of facility employees

   For 100 operating reactors, the estimated total reduction in occupational exposure resulting from reduced core damage frequencies and associated post-accident cleanup and repair activities is 1,500 person-rem. No significant increase in occupational exposure is expected from operation and maintenance activities associated with the rule. Equipment additions and modifications contemplated do not require work in and around the reactor coolant system and therefore are not expected to result in significant radiation exposure.

5. Installation and continuing costs associated with the backfit, including the cost of facility downtime or the cost of construction delay

   For 100 operating reactors, the total estimated cost associated with the station blackout rule ranges from $42 to $94 million with a best estimate of $60 million. This estimate breaks down as follows:

Activity	Estimated number of reactors	Estimated total cost (million dollars)		
		Best	High	Low
Assess plant's capability to cope with station blackout	100	25	40	20
Develop procedures and training	100	10	15	5
Improve diesel generator reliability	10	2.5	4	1.5
Requalify diesel generator	2	5.5	11	2.5
Install hardware to increase plant capability to cope with station blackout	27	17	24	13
Totals		60	94	42

6. The potential safety impact of changes in plant or operational complexity, *including the relationship to proposed and existing regulatory requirements*

The rule requiring plants to be able to cope with a station blackout should not add to plant or operational complexity. The station blackout rule is closely related to several NRC generic programs and proposed and existing regulatory requirements as the following discussion indicates.

Generic Issue B-56, Diesel Generator Reliability

The resolution of USI A-44 includes a regulatory guide on station blackout that specifies the following guidance on diesel generator reliability (RG 1.155, Sections C1.1. and 2):

> The reliable operation of the onsite emergency ac power sources should be ensured by a reliability program designed to monitor and maintain the reliability of each power source over time at a specified acceptable level and to improve the reliability if that level is not achieved. The reliability program should include surveillance testing, target values for maximum failure rate, and a maintenance program. Surveillance testing should monitor performance so that if the actual failure rate exceeds the target level, corrective actions can be taken.
>
> The maximum emergency diesel generator failure rate for each diesel generator should be maintained at 0.05 failure per demand. However, for plants having an emergency ac power system [configuration requiring two-out-of-three diesel generators or having a total of two diesel generators shared between two units at a site], the emergency diesel generator failure rate for each diesel generator should be maintained at 0.025 failure per demand or less.

The resolution of B-56 will provide specific guidance for use by the staff or industry to review the adequacy of diesel generator reliability programs consistent with the resolution of USI A-44.

Generic Issue 23, Reactor Coolant Pump Seal Failures

Reactor coolant pump (RCP) seal integrity is necessary for maintaining primary system inventory during station blackout conditions. The estimates of core damage frequency for station blackout events for

USI A-44 assumed that RCP seals would leak at a rate of 20 gallons per minute. Results of analyses performed for GI 23 will provide the information necessary to estimate RCP seal behavior during a station blackout. The industry coping analysis guidelines (NUMARC-8700) recognize the possibility of leakages exceeding on assumed 25 gpm per pump and incorporate the need to reevaluate the plant specific coping analysis if the resolution of GI 23 identifies higher levels.

USI A-45, Shutdown Decay Heat Removal Requirements

The overall objective of USI A-45 is to evaluate the adequacy of current licensing design requirements to ensure that the nuclear power plants do not pose an unacceptable risk as a result of failure to remove shutdown decay heat. The study includes an assessment of alternative means of shutdown decay heat removal and of diverse "dedicated" systems for this purpose. Results will include proposed recommendations regarding the desirability of, and possible design requirements for, improvements in existing systems or an alternative dedicated decay heat removal method.

The USI A-44 concern for maintaining adequate core cooling under station blackout conditions can be considered a subset of the overall A-45 issue. However, there are significant differences in scope between these two issues. USI A-44 deals with the probability of loss of ac power, the capability to remove decay heat using systems that do not require ac power, and the ability to restore ac power in a timely manner. USI A-45 deals with the overall reliability of the decay heat removal function in terms of response to transients, small break loss-of-coolant accidents, and special emergencies such as fires, floods, seismic events, and sabotage.

Although the recommendations that might result from the resolution of USI A-45 are not yet final, some could affect the station blackout capability, while others would not. Recommendations that involve a new or improved decay heat removal system that is ac power dependent but that

does not include its own dedicated ac power supply would have no effect on USI A-44. Recommendations that involve an additional ac-independent decay heat removal system would have a very modest effect of USI A-44. Recommendations that involve an additional decay heat removal system with its own ac power supply would have a significant effect on USI A-44. Such a new additional system would receive the appropriate credit within the USI A-44 resolution by either changing the emergency ac power configuration group or providing the ability to cope with a station blackout for an extended period of time. Well before plant modifications, if any, will be implemented to comply with the station blackout rule, it is anticipated that the proposed technical resolution of USI A-45 will be published for public comment. Those plants needing hardware modifications for station blackout could be reevaluated before any actual modifications are made so that any contemplated design changes resulting from the resolution of USI A-45 can be considered at the same time.

Generic Issue A-30, Adequacy of Safety-Related DC Power Supply

The analysis performed for USI A-44 assumed that a high level of dc power system reliability would be maintained so that (1) dc power system failures would not be a significant contributor to losses of all ac power and (2) should a station blackout occur, the probability of immediate dc power system failure would be low. Whereas Generic Issue A-30 focuses on enhancing battery reliability, the resolution of USI A-44 is aimed at assuring adequate station battery capacity in the event of a station blackout of a specified duration. Therefore, these two issues are consistent and compatible.

Fire Protection Program

10 CFR 50.48 states that each operating nuclear power plant shall have a fire protection plan that satisfies GDC 3. The fire protection features required to satisfy GDC 3 are specified in Appendix R to 10 CFR 50. They include certain provisions regarding alternative and dedicated shutdown

capability. To meet these provisions, some licensees have added, or plan to add, improved capability to restore power from offsite sources or onsite diesels for the shutdown system. A few plants have installed a safe shutdown facility for fire protection that includes a charging pump powered by its own independent ac power source. In the event of a station blackout, this system can provide makeup capability to the primary coolant system as well as reactor coolant pump seal cooling. This could be a significant benefit in terms of enhancing the ability of a plant to cope with a station blackout. Plants that have added equipment to achieve alternate safe shutdown in order to meet Appendix R requirements could take credit for that equipment, if available, for coping with a station blackout event.

7. The estimated resource burden on the NRC associated with the backfit and the availability of such resources

   The estimated total cost for NRC review of industry submittals required by the station blackout rule is $1.5 million based on submittals for 100 reactors and an estimated average of 175 person-hours per reactor.

8. The potential impact of differences in facility type, design, or age on the relevancy and practicality of the backfit

   The station blackout rule applies to all pressurized water reactors and boiling water reactors. However, in determining an acceptable station blackout coping capability for each plant, differences in plant characteristics relating to ac power reliability (e.g., number of emergency diesel generators, the reliability of the offsite and onsite emergency ac power systems) could result in different acceptable coping capabilities. For example, plants with an already low risk from station blackout because of multiple, highly reliable ac power sources are required to withstand a station blackout for a relatively short period of time; and few, if any, hardware backfits would be required as a result of the rule. Plants with currently higher risk from station blackout are required to withstand somewhat longer duration blackouts; and, depending on their existing

capability, may need some modifications to achieve the longer station blackout capability.

9. Whether the backfit is interim or final and, if interim, the justification for imposing the backfit on an interim basis

   The station blackout rule is the final resolution of USI A-44; it is not an interim measure.

ENCLOSURE C

USI A-44 EDO PKG

11-12-87

NUREG-1109

Regulatory/Backfit Analysis for the Resolution of Unresolved Safety Issue A-44, Station Blackout

U.S. Nuclear Regulatory Commission

Office of Nuclear Regulatory Research
Office of Nuclear Reactor Regulation

A. M. Rubin

UNITED STATES NUCLEAR REGULATORY COMMISSION

ABSTRACT

Station blackout is the complete loss of alternating current (ac) electric power to the essential and nonessential buses in a nuclear power plant; it results when both offsite power and the onsite emergency ac power systems are unavailable. Because many safety systems required for reactor core decay heat removal and containment heat removal depend on ac power, the consequences of a station blackout could be severe. Because of the concern about the frequency of loss of offsite power, the number of failures of emergency diesel generators, and the potentially severe consequences of a loss of all ac power, "Station Blackout" was designated as Unresolved Safety Issue (USI) A-44.

This report presents the regulatory/backfit analysis for USI A-44. It includes (1) a summary of the issue, (2) the recommended technical resolution, (3) alternative resolutions considered by the Nuclear Regulatory Commission (NRC) staff, (4) an assessment of the benefits and costs of the recommended resolution, (5) the decision rationale, (6) the relationship between USI A-44 and other NRC programs and requirements, and (7) a backfit analysis demonstrating that the resolution of USI A-44 complies with the backfit rule (10 CFR 50.109).

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PREFACE

This report presents the supporting value-impact analysis, backfit analysis, and decision rationale for the resolution of USI A-44. The resolution itself consists of a rule that requires nuclear power plants to be able to cope with a station blackout for a specified period, and an associated regulatory guide that provides guidance on an acceptable means to comply with the rule. The NRC staff report that provides data and technical analyses supporting the resolution of this issue is published separately as NUREG-1032. NRC contractor NUREG/CR reports published under this task are listed in Section 5.2 of this report.

The Commission published a proposed station blackout rule in the Federal Register on March 21, 1986 (51 FR 9829) for public comment. In April 1986, the NRC published a regulatory guide on station blackout for comment (Regulatory Guide 1.155). Previously, in January 1986, NRC published a draft version of this report (NUREG-1109) for comment. All public comments on this issue were reviewed and considered by the staff in formulating the final resolution of USI A-44 and this final version of NUREG-1109. Responses to the public comments are discussed in the supplementary information section of the Notice of Final Rulemaking for the Station Blackout Rule, which is to be published in the Federal Register.

Alan M. Rubin

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EXECUTIVE SUMMARY

This report provides supporting information, including a cost-benefit analysis and a backfit analysis, for the Nuclear Regulatory Commission's (NRC's) resolution of Unresolved Safety Issue (USI) A-44, "Station Blackout." The term "station blackout" refers to the complete loss of alternating current (ac) electric power to the essential and nonessential switchgear buses in a nuclear power plant. Station blackout involves the loss of offsite power concurrent with turbine trip and the unavailability of the onsite emergency ac power system. Because many safety systems required for reactor core decay heat removal and containment heat removal depend on ac power, the consequences of station blackout could be severe.

The NRC's concern about station blackout arose because of the accumulated experience regarding the reliability of ac power supplies. In numerous instances emergency diesel generators have failed to start and run during tests conducted at operating plants. In addition, a number of operating plants have experienced a total loss of offsite electric power, and more such occurrences are expected. In almost every one of these loss-of-offsite-power events, the onsite emergency ac power supplies were available immediately to supply the power needed by vital safety equipment. However, in some instances, one of the redundant emergency power supplies has been unavailable. In a few cases, there has been a complete loss of ac power, but during these events, ac power was restored in a short time without any serious consequences.

The issue of station blackout involves the likelihood and duration of the loss of offsite power, the redundancy and reliability of onsite emergency ac power systems, and the potential for severe accident sequences after a loss of all ac power. These topics were investigated under USI Task Action Plan A-44.* In addition to identifying important factors and sequences that could lead to station blackout, the results indicated that actions could be taken to reduce the risk from station blackout events. The issue is of concern for both boiling water reactors and pressurized water reactors.

The evaluation to resolve USI A-44 included deterministic and probabilistic analyses. Calculations to determine the timing and consequences of various accident sequences were performed, and the dominant factors affecting station blackout likelihood were identified. Using this information, simplified probabilistic accident sequence correlations were calculated to estimate the likelihood of core melt accidents resulting from station blackout for different plant design, operational, and location factors. These quantitative estimates were used to give insights on the relative importance of various factors, and those insights, along with engineering judgment, were used to develop the resolution. Thus, the effects of variations in design, operations, and plant location on risk from station blackout events were used to reach a reasonably consistent level of risk in the recommendations developed.

*The technical findings of these investigations are detailed in NUREG/CR-2989, NUREG/CR-3226, NUREG/CR-3992, NUREG/CR-4347, and NUREG-1032.

Although there are licensing requirements and guidance directed at providing reliable offsite and onsite ac power, experience has shown that there are practical limitations in ensuring the reliability of offsite and onsite emergency ac power systems. Analyses have shown that core damage frequency can be significantly reduced if a plant can withstand a total loss of ac power until either offsite or onsite emergency ac power can be restored.

Because there is no requirement that plants be able to withstand a loss of both the offsite and onsite emergency ac power systems, the resolution calls for rulemaking to require all plants to be able to cope with a station blackout for a specified duration. Regulatory Guide 1.155 on station blackout describes a method acceptable to the NRC staff for complying with the rule, and specifies guidance on providing reliable ac electric power supplies. Plants with an already low risk from station blackout are required to withstand a station blackout for a relatively short period of time. These plants probably need few, if any, modifications as a result of the rule. Plants with a currently higher risk from station blackout are required to withstand blackouts of a somewhat longer duration, and, depending on their existing capability, might require modifications (such as increased station battery capacity or condensate storage tank capacity) to meet this requirement. The staff has determined that these modifications are cost-effective in terms of reducing risk to the public.

The general objective of the resolution of USI A-44 is to reduce the risk of severe accidents associated with station blackout by making station blackout a relatively small contributor to total core damage frequency. Specific actions called for in the resolution include (1) maintaining highly reliable ac electric power systems; (2) developing procedures and training to restore offsite and onsite emergency ac power should either one or both become unavailable; and (3) as additional defense-in-depth, ensuring that plants can cope with a station blackout for some period of time, based on the probability of occurrence of a station blackout at the site, as well as on the capability for restoring ac power for that site.

The method to determine an acceptable station blackout duration capability is presented in the regulatory guide. Applications of this guide result in determinations that plants be able to withstand station blackouts from 2 to 16 hours, depending on the plant's specific design and site-related characteristics. Licensees may propose durations different from those specified in the regulatory guide, based on plant-specific factors relating to the reliability of ac power systems.

The benefit from implementing the rule and the regulatory guide is a reduction in the frequency of core damage per reactor-year due to station blackout and the associated risk of offsite radioactive releases. The risk reduction for 100 operating reactors is estimated to be 145,000 person-rems.

The cost for licensees to comply with the requirements varies depending on the existing capability of each plant to cope with a station blackout, as well as the plant-specific station blackout duration determined. The costs are primarily to industry to assess the plant's capability to cope with a station blackout, to develop procedures, to improve diesel generator reliability if the reliability falls below certain levels, and to retrofit plants with additional components or systems, as necessary, to meet the requirements.

The estimated total cost for 100 operating reactors to comply with the resolution of USI A-44 is about $60 million. The average cost per reactor is estimated to be $600,000, ranging from $350,000 if only a station blackout assessment and procedures and training are necessary to a maximum of about $4 million if substantial modifications are needed, including requalification of a diesel generator.

The overall value-impact ratio, not including accident avoidance costs, is about 2,400 person-rems averted per million dollars. If cost savings from accident avoidance (cleanup and repair of onsite damages and replacement power) were included, the overall value-impact ratio would improve significantly to about 6,100 person-rems averted per million dollars.

Several NRC programs are related to USI A-44, including Diesel Generator Reliability (Generic Issue B-56), Reactor Coolant Pump Seal Failures (Generic Issue B-23), Safety-Related DC Power Supplies (Generic Issue A-30), and Shutdown Decay Heat Removal Requirements (USI A-45). These programs are closely coordinated within NRC and are compatible with the resolution of USI A-44.

REGULATORY/BACKFIT ANALYSIS FOR THE RESOLUTION OF UNRESOLVED SAFETY ISSUE A-44, STATION BLACKOUT

1 STATEMENT OF THE PROBLEM

"Station blackout" refers to the complete loss of alternating current (ac) electric power to the essential and nonessential switchgear buses in a nuclear power plant. Station blackout involves the loss of offsite power concurrent with turbine trip and the unavailability of the onsite emergency ac power system. Because many safety systems required for reactor core decay heat removal and containment heat removal depend on ac power, the consequences of station blackout could be severe.

The concern of the Nuclear Regulatory Commission (NRC) about station blackout arose because of the accumulated experience regarding the reliability of ac power supplies. In numerous instances emergency diesel generators have failed to start and run during tests conducted at operating plants. In addition, a number of operating plants have experienced a total loss of offsite electric power, and more occurrences are expected. In almost every one of these loss-of-offsite-power events, the onsite emergency ac power supplies were available immediately to supply the power needed by vital safety equipment. However, in some instances, one of the redundant emergency power supplies has been unavailable. In a few cases, there has been a complete loss of ac power, but during these events, ac power was restored in a short time without any serious consequences.

The results of the Reactor Safety Study (NUREG-75/014) showed that for one of the two plants evaluated, a station blackout accident could be an important contributor to the total risk from nuclear power plant accidents. Although this total risk was found to be small, the relative importance of the station blackout accident was established. This finding and the accumulated diesel generator failure experience increased the concern about station blackout.

The issue of station blackout involves the likelihood and duration of losses of offsite power, the redundancy and reliability of onsite emergency ac power systems, and the potential for severe accident sequences after a loss of all ac power. These topics were investigated under Unresolved Safety Issue (USI) Task Action Plan A-44, and the technical findings are reported in detail in NUREG/CR-2989, NUREG/CR-3226, NUREG/CR-3992, NUREG/CR-4347, and NUREG-1032. In addition to identifying important factors and sequences that could lead to station blackout, the results indicated that estimated core damage* frequencies from

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*Analysis has shown that for postulated station blackout events, the difference between the estimated frequency of core damage and core melt is small because of the relatively low probability of recovering ac power and terminating an accident sequence after initial core damage, but before full core melt (NUREG-1032).

station blackout vary significantly for different plants but could be on the order of 10^{-4} per reactor-year for some plants. To reduce this risk, action should be taken to resolve the safety concern stemming from station blackout. The issue is of concern for both pressurized water reactors (PWRs) and boiling water reactors (BWRs).

There is no requirement currently for plants to be able to cope with a station blackout. Existing requirements for offsite and onsite ac power systems are in General Design Criterion (GDC) 17, "Electric Power Systems," of Appendix A to Part 50 of Title 10 of the Code of Federal Regulations (10 CFR 50). They are discussed in Sections 8.2, "Offsite Power Systems," and 8.3.1, "AC Power Systems (Onsite)," of the NRC's "Standard Review Plan for the Safety Review of Nuclear Power Reactors" (SRP, NUREG-0800). Testing of emergency diesel generators is discussed in Regulatory Guide (RG) 1.108, "Periodic Testing of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants." Separation and independence of electric power systems are discussed in RG 1.6, "Independence Between Redundant Standby (Onsite) Power Sources and Between Their Distribution Systems," and RG 1.75, "Physical Independence of Electric Systems." SRP Sections 8.3.1 and 9.5.4 through 9.5.8 discuss maintenance and design provisions for the onsite emergency diesels. These licensing requirements and guidance are directed at providing reliable offsite and onsite ac power.

Experience has shown that there are practical limits in ensuring the reliability of offsite and onsite emergency ac power systems. Analyses show that core damage frequency can be significantly reduced if a plant can withstand a total loss of ac power until either offsite or onsite emergency ac power can be restored.

2 OBJECTIVES

The general objective of the requirements to resolve USI A-44 is to reduce the risk of severe accidents associated with station blackout by making station blackout a relatively small contributor to the average frequency of core damage for the total population of plants. Specific actions called for in the resolution include (1) maintaining highly reliable ac electric power systems; (2) developing procedures and training to restore offsite and onsite emergency ac power should either one or both become unavailable; and (3) as additional defense-in-depth, ensuring that plants can cope with a station blackout for some period of time based on the probability of occurrence of a station blackout at the site as well as on the capability for restoring power for that site.

3 ALTERNATIVE RESOLUTIONS

In developing the resolution of USI A-44, the staff considered four specific alternative courses of action. These are discussed below.

3.1 Alternative (i)

To achieve the objectives stated in Section 2 above, the resolution of USI A-44 calls for specific guidance relating to the reliability of offsite and onsite emergency ac power systems, as well as a requirement that plants be able to cope with a station blackout for a specific duration. The recommendations to resolve this issue are summarized as follows:

(1) The reliability of the onsite emergency ac power sources should be maintained at or above specified acceptable reliability levels.

(2) Procedures and training should be developed to restore emergency ac power and offsite power using nearby power sources if the emergency ac power system and the normal offsite power systems are unavailable.

(3) Each nuclear power plant should be able to withstand and recover from a station blackout lasting a specified minimum duration. Regulatory Guide 1.155 entitled "Station Blackout"* provides a method for determining an acceptable plant-specific station blackout duration based on a comparison of a plant's characteristics to those factors that have been identified as the main contributors to risk from station blackout. These factors include: (a) the redundancy of onsite emergency ac power sources (number of sources available for decay heat removal minus the number needed for decay heat removal), (b) the reliability of onsite emergency ac power sources (usually diesel generators), (c) the frequency of loss of offsite power, and (d) the probable time to restore offsite power. The frequency and duration of loss of offsite power are related to grid and switchyard reliability, historical weather data for severe storms, and the availability of nearby alternate power sources (e.g., gas turbines). The staff has concluded (NUREG-1032) that long-duration offsite power outages are caused primarily by severe storms (e.g., hurricanes, ice).

(4) Each nuclear power plant should be evaluated to determine its capability to withstand and recover from a station blackout of a duration as determined in (3) above. This evaluation should include such considerations as:

- Verifying the adequacy of station battery power, condensate storage tank capacity, and plant/instrument air for the duration of a station blackout.
- Verifying the adequacy of reactor coolant pump seal integrity for the duration of a station blackout. This should be done by demonstrating, via experiment and/or analysis, that seal leakage due to a lack of seal cooling will not reduce the primary system coolant inventory to the degree that the ability to cool the core during station blackout is lost.
- Verifying that the equipment needed to operate during a station blackout and the recovery from the blackout will be able to operate under the environmental conditions associated with a total loss of ac power (i.e., loss of heating, ventilation, and air conditioning).

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*Single copies of this guide may be obtained by writing to the Distribution Services, Division of Information Support Services, U.S. Nuclear Regulatory Commission, Washington, DC 20555.

(5) If the plant's station blackout capability (as determined in (4)) is significantly less than the minimum acceptable plant-specific station blackout duration determined in (3), modifications to the plant may be necessary to increase the time the plant is able to cope with a station blackout. The regulatory guide identifies specific factors to be considered if such modifications are necessary.

(6) Each nuclear power plant should have procedures and training to cope with a station blackout and to restore normal long-term decay heat removal once ac power is restored.

Because there is no requirement for plants to be able to withstand a loss of both the offsite and onsite emergency ac power systems, the resolution calls for rulemaking to require that all plants be able to cope with a station blackout for a specified duration. The regulatory guide describes a method acceptable to the NRC staff for complying with the rule, and specifies guidance on providing reliable ac electric power supplies. Plants with an already low risk from station blackout are required to withstand a station blackout for a relatively short period of time. These plants probably need few, if any, modifications as a result of the rule. Plants with currently higher risk from station blackout are required to withstand blackouts of somewhat longer duration, and, depending on their existing capability, may require modifications (such as increasing station battery capacity or condensate storage tank capacity). The staff has determined that these modifications are cost-effective in terms of reducing risk to the public.

The method to determine an acceptable station blackout duration capability, as presented in the regulatory guide, is summarized below. The guide specifies minimum acceptable blackout durations that a plant should be capable of surviving. The minimum duration is from 2 to 16 hours (see Table 1) depending on a plant's design and site-related characteristics. Most plants would fall in either the 4- or 8-hour group. Licensees may propose durations different from those specified in Table 1. Such proposals should be based on plant-specific factors relating to the reliability of ac power systems, such as those discussed in NUREG-1032, and would be reviewed by the NRC staff.

Tables 2 through 7 provide the necessary detailed descriptions and definitions of the various factors used in Table 1. Table 2 identifies different levels of redundancy of the onsite emergency ac power system used to define the emergency ac power configuration groups in Table 1. Table 3 provides definitions of the three offsite power design characteristic groups used in Table 1. The groups are defined according to various combinations of the following factors: (1) independence of offsite power (I), (2) severe weather (SW), (3) severe weather recovery (SWR), and (4) extremely severe weather (ESW). The definitions of the factors I, SW, SWR, and ESW are provided in Tables 4 through 7, respectively. After identifying the appropriate groups from Tables 2 and 3 and the reliability level of the onsite emergency ac power sources, Table 1 can be used to determine the minimum acceptable station blackout duration capability (e.g, 4 or 8 hours) for each plant. The reliable operation of the onsite emergency ac power sources should be ensured by a reliability program designed to monitor and maintain reliability over time at a specified acceptable level and to improve the reliability if that level is not achieved.

One example of an application of this method considers a nuclear power plant that has (1) two diesel generators, one of which is required for ac power for decay heat removal systems; (2) one switchyard and one alternate offsite power circuit, in addition to the normally energized offsite circuit to the Class 1E buses; (3) an estimated frequency of loss of offsite power due to severe weather of 0.005 per site-year; and (4) an annual expectation of storms at the site with winds greater than 125 miles per hour of 0.002 per year. On the basis of this information, this plant is in independence of offsite power group I3 (see Table 4), severe weather group SW2 (see Table 5), severe weather recovery group SWR2 (no enhanced recovery for severe weather, Table 6), and extremely severe weather group ESW3 (see Table 7). This combination of factors places the plant in offsite power design characteristic group P2 (see Table 3). Based on the number of diesel generators, the plant is in emergency ac power configuration group C. As indicated on Table 1, if the failure rate of each emergency diesel generator is maintained at 0.025 failure per demand or less, this plant should have the capability to withstand and recover from a station blackout lasting 4 hours or more. If the failure rate of each emergency diesel generator were between 0.025 and 0.05, the acceptable station blackout duration would increase to 8 hours. If the emergency diesel generator failure rate were greater than 0.05, then steps should be taken to improve the diesel generator reliability.

3.2 Alternative (ii)

Alternative (ii) would treat plants uniformly by requiring all plants to be able to cope with station blackout of the same duration.

3.3 Alternative (iii)

Alternative (iii) would require plants with the highest potential risk from station blackout to add either an additional emergency diesel generator or another ac-independent decay heat removal system.

3.4 Alternative (iv)

The Nuclear Utility Management and Resources Committee (NUMARC) endorsed the following industry initiatives to resolve the station blackout issue (letter from J. Miller to N. Palladino, 1986):

1. Each utility will review their site(s) against the criteria specified in NUREG-1109, and if the site(s) fall into the category of an eight-hour site after utilizing all power sources available, the utility will take actions to reduce the site(s) contribution to the overall risk of station blackout. Non-hardware changes will be made within one year. Hardware changes will be made within a reasonable time thereafter.

2. Each utility will implement procedures at each of its site(s) for:

   a. coping with a station blackout event,

   b. restoration of AC power following a station blackout event, and.

Table 1 Acceptable station blackout duration capability (hours)[1]

Maximum emergency diesel generator failure rate per demand	Offsite power design characteristic group[2]		
	P1	P2	P3
Emergency ac (EAC) power configuration group A[3]			
0.025	2	4	4
0.05	2	4	8
EAC power configuration group B			
0.025	4	4	4
0.05	4	4	8
EAC power configuration group C			
0.025	4	4	8
0.05	4	8	16
EAC power configuration group D			
0.025	4	8	8

[1]The staff will consider variations from these times if justification, including a cost-benefit analysis, is provided by the licensee. The methodology and sensitivity studies in NUREG-1032 are acceptable for this justification.

[2]See Table 3 to determine groups P1, P2, and P3.

[3]See Table 2 to determine emergency ac power configuration group.

Source: Regulatory Guide 1.155.

Table 2 Emergency ac power configuration groups[1]

Emergency ac (EAC) power configuration group	No. of EAC power sources[2]	No. of EAC power sources required to operate ac-powered decay heat removal systems[3]
A	3[4]	1
	4	1
B	4	2
	5	2
C	2[4]	1
	3[5]	1
D	2[6]	1
	3	2
	4	3
	5	3

[1]Special-purpose dedicated diesel generators, such as those associated with high pressure core spray systems at some BWRs, are not counted in the determination of EAC power configuration groups.

[2]If any of the EAC power sources are shared among units at a multiunit site, this is the total number of shared and dedicated sources for those units at the site.

[3]This number is based on all the ac loads required to remove decay heat (including ac-powered decay heat removal systems) to achieve and maintain hot shutdown at all units at the site with offsite power unavailable.

[4]For EAC power sources not shared with other units.

[5]For EAC power sources shared with another unit at a multiunit site.

[6]For shared EAC power sources in which each diesel generator is capable of providing ac power to more than one unit at a site concurrently.

Source: Regulatory Guide 1.155.

Table 3 Offsite power design characteristic groups

Group	Offsite power design characteristics			
P1	Sites that have any combination of the following factors:			
	I[1]	SW[2]	SWR[3]	ESW[4]
	1 or 2	1 or 2	1 or 2	1 or 2
	1 or 2	1	1 or 2	3
	1 or 2	3	1	1 or 2
P2	All other sites not in group P1 or P3			
P3	Sites that expect to experience a total loss of offsite power caused by grid failures at a frequency equal to or greater than once in 20 site-years, unless the site has procedures to recover ac power from reliable alternate (nonemergency) ac power sources within approximately 1/2 hour following a grid failure.			
	or			
	Sites that have any combination of the following factors:			
	I	SW	SWR	ESW
	Any I	5	2	Any ESW
	Any I	1,2,3, or 4	1 or 2	5
	Any I	5	1	Any ESW
	Any I	4	2	1,2,3, or 4
	1 or 2	3	2	4
	3	3	2	3 or 4

[1]See Table 4 for definitions of independence of offsite power (I) groups.

[2]See Table 5 for definitions of severe weather (SW) groups.

[3]See Table 6 for definitions of severe weather recovery (SWR) groups.

[4]See Table 7 for definitions of extremely severe weather (ESW) groups.

Source: Regulatory Guide 1.155.

Table 4 Definitions of independence of offsite power (I) groups

<table>
<tr><th rowspan="2">Category</th><th colspan="3">I</th></tr>
<tr><th>1</th><th>2</th><th>3</th></tr>
<tr><td>1. Independence of offsite power sources</td><td>1. All offsite power sources are connected to the plant through two or more switchyards or separate incoming transmission lines, but at least one of the ac sources is electrically independent of the others. (The independent 69-kV line in Figure 1 is representative of this design feature.)

<u>or</u></td><td colspan="2">1.a. All offsite power sources are connected to the plant through one switchyard.

<u>or</u>

1.b. All offsite power sources are connected to the plant through two or more switchyards, and the switchyards are electrically connecte (The 345- and 138-kV switchyards in Figures 2 and 3 represent this design feature.)

<u>and</u> (I = 2)   <u>and</u> (I = 3)</td></tr>
<tr><td>2. Automatic and manual transfer schemes for the Class 1E buses when the normal source of ac power fails and when the backup sources of offsite power fail.

a. The normal source of ac power is assumed to be the unit main generator.</td><td>2.a. After loss of the normal ac source,

(1) There is an automatic transfer of all safe-shutdown buses to a separate preferred alternate power source.

(2) There is an automatic transfer of all safe-shutdown buses to one preferred power source. If this preferred power source fails, there is another automatic transfer to the remaining preferred power sources or to alternate offsite power source.

<u>or</u></td><td>2.a. After loss of the normal ac power source, there is an automatic transfer of all safe-shutdown buses to one preferred alternate power source. If this source fails, there may be one or more manual transfers of power source to the remaining preferred or alternate offsite power sources.

<u>or</u></td><td>2.a. If the normal source of ac power fails, there are no automatic transfers and one or more manual transfers of all safe-shutdown buses to preferred or alternate offsite power sources.

<u>or</u>

There is one automatic transfer and no manual transfer of all safe shutdown buses to one preferred or one alternate</td></tr>
<tr><td>b. If the Class 1E buses are normally designed, the preferred alternate power sources.</td><td>2.b. Each safe-shutdown bus is normally connected to a separate preferred alternate power source with automatic or manual transfer capability between the preferred alternate sources</td><td>2.b. The safe-shutdown buses are normally aligned to the same preferred power source with either an automatic or manual transfer to the remaining preferred alternate ac power source.</td><td></td></tr>
</table>

Source: Regulatory Guide 1.155

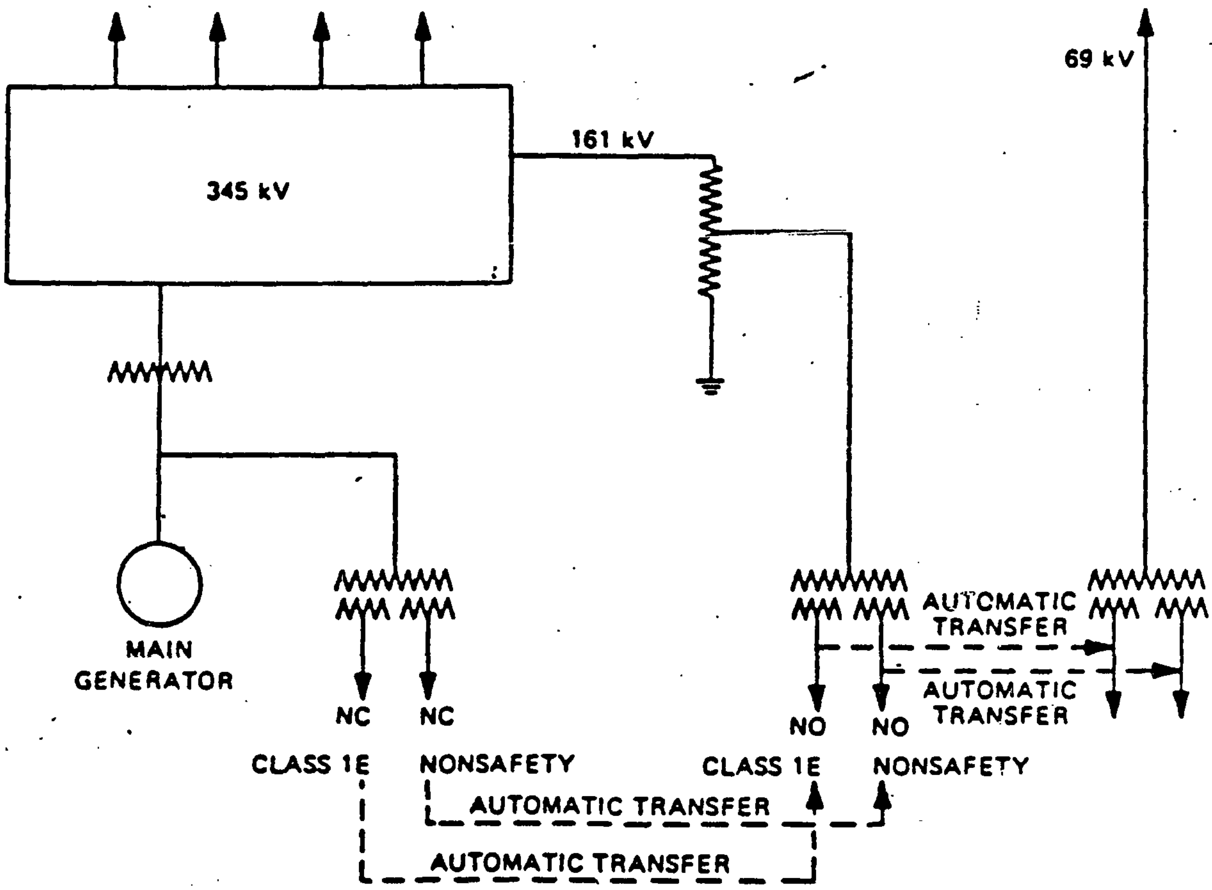

Figure 1 Schematic of electrically independent transmission line

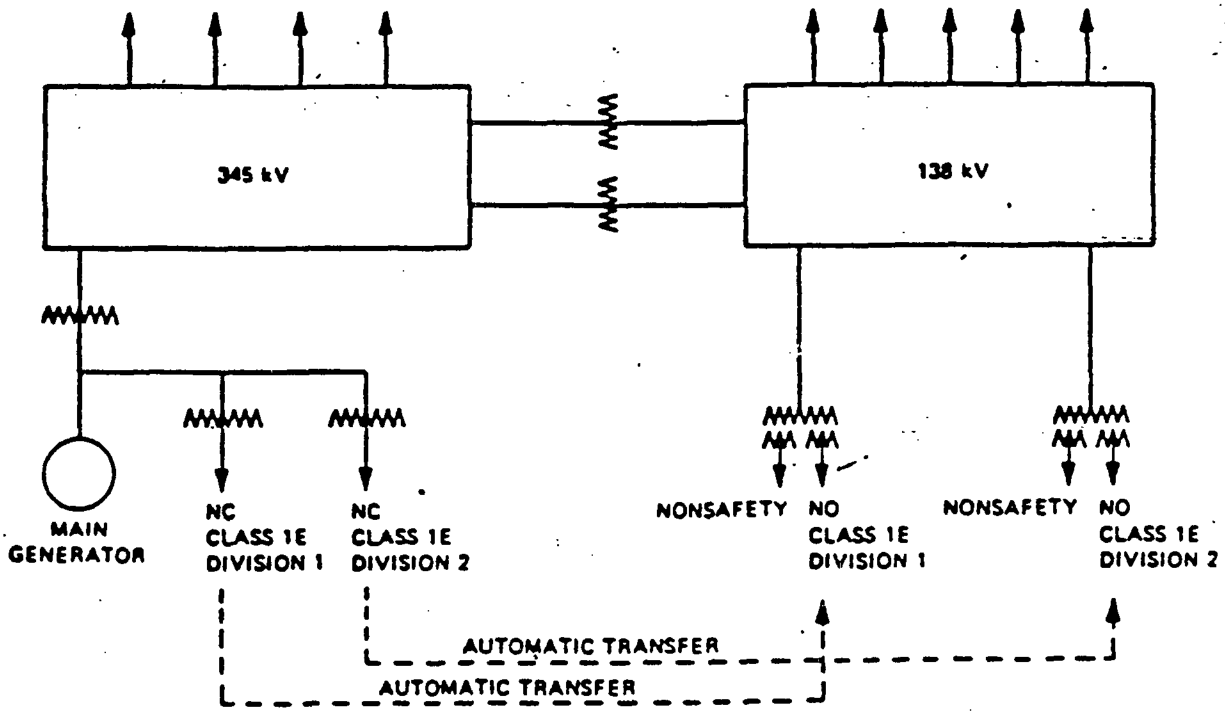

Figure 2 Schematic of two switchyards electrically connected (one-unit site)

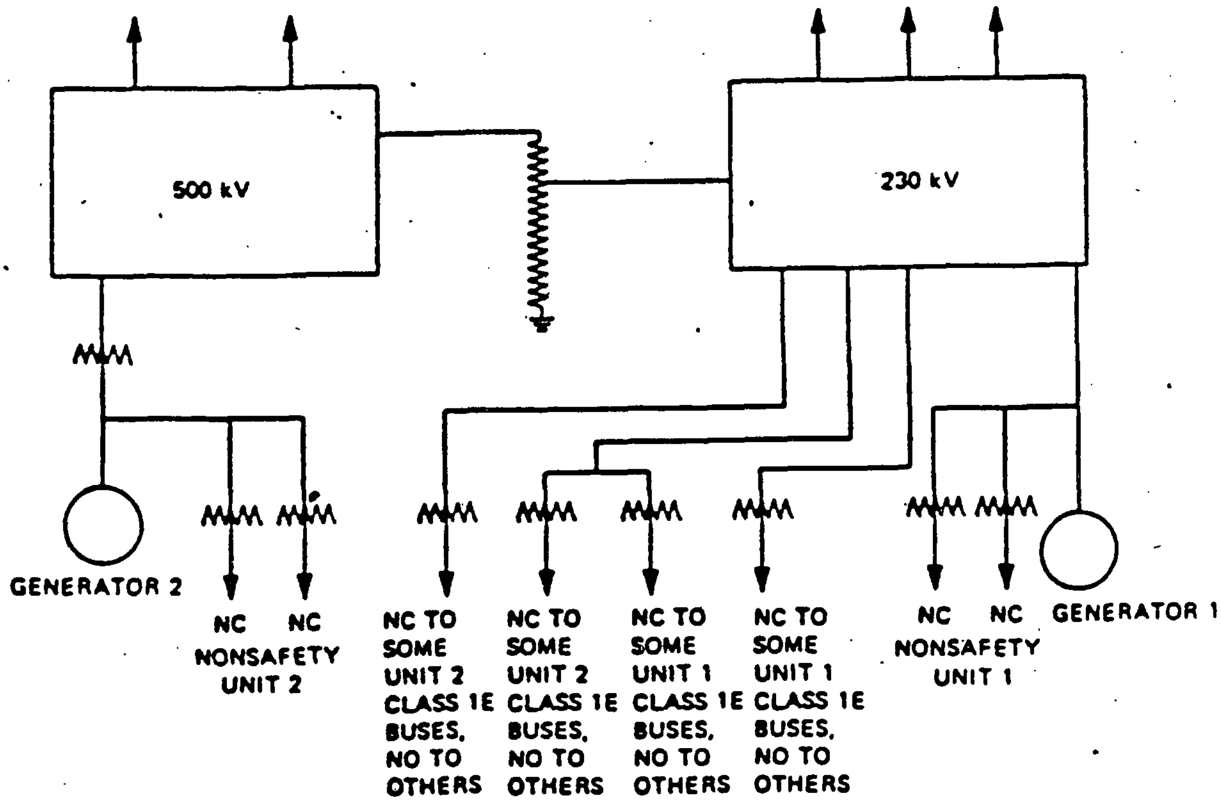

Figure 3 Schematic of two switchyards electrically connected (two-unit site)

Table 5 Definitions of severe weather (SW) groups

SW group	Estimated frequency of loss of offsite power due to severe weather, f* (per site-year)
1	$f < 0.0033$
2	$0.0033 \leq f < 0.010$
3	$0.010 \leq f < 0.033$
4	$0.033 \leq f < 0.10$
5	$0.10 \leq f$

*The estimated frequency of loss of offsite power due to severe weather, f, is determined by the following equation:

$$f = (1.3 \times 10^{-4})h_1 + (b)h_2 + (0.012)h_3 + (c)h_4$$

where

h_1 = annual expectation of snowfall for the site, in inches

h_2 = annual expectation of tornadoes (with wind speeds greater than or equal to 113 miles per hour (mph)) per square mile at the site

b = 12.5 for sites with transmission lines on two or more rights-of-way spreading out in different directions from the switchyard, or

b = 72.3 for sites with transmission lines on one right-of-way

h_3 = annual expectation of storms at the site with wind velocities between 75 and 124 mph

h_4 = annual expectation of hurricanes at the site

c = 0 if switchyard is not vulnerable to salt spray
c = 0.78 if switchyard is vulnerable to salt spray

The annual expectation of snowfall, tornadoes, and storms may be obtained from National Weather Service data from the weather station nearest the plant or by interpolation, if appropriate, between nearby weather stations. The basis for the empirical equation for the frequency of loss of offsite power due to severe weather, f, is given in NUREG-1032, Appendix A.

Source: Regulatory Guide 1.155.

Table 6 Definitions of severe weather recovery (SWR) groups

SWR group	Definition
1	Sites with enhanced recovery (i.e., sites that have the capability and procedures for restoring offsite (nonemergency) ac power to the site within 2 hours following a loss of offsite power due to severe weather).
2	Sites without enhanced recovery.

Source: Regulatory Guide 1.155

Table 7 Definitions of extremely severe weather (ESW) groups

ESW group	Annual expectation of storms at a site with wind velocities equal to or greater than 125 miles per hour (e)*
1	$e < 3.3 \times 10^{-4}$
2	$3.3 \times 10^{-4} \leq e < 1 \times 10^{-3}$
3	$1 \times 10^{-3} \leq e < 3.3 \times 10^{-3}$
4	$3.3 \times 10^{-3} \leq e < 1 \times 10^{-2}$
5	$1 \times 10^{-2} \leq e$

*The annual expectation of storms may be obtained from National Weather Service data from the weather station nearest the plant or by interpolation, if appropriate, between nearby weather stations.

Source: Regulatory Guide 1.155.

c. preparing the plant for severe weather conditions, such as hurricanes and tornados to reduce the likelihood and consequences of a loss of offsite power and to reduce the overall risk of a station blackout event.

3. Each utility will, if applicable, reduce or eliminate cold fast-starts of emergency diesel generators for testing through changes to technical specifications or other appropriate means.

4. Each utility will monitor emergency AC power unavailability utilizing data utilities provided to INPO [Institute of Nuclear Power Operations] on a regular basis.

5. Each utility will assess the ability of its plant(s) to cope with a station blackout. Plants utilizing alternate ac power for station blackout response which can be shown by test to be available to power the shutdown buses within 10 minutes of the onset of station blackout do not need to perform any coping assessment. Remaining alternate ac plants will assess their ability to cope for 1 hour. Plants not utilizing an alternate ac source will assess their ability to cope for 4 hours. Factors identified that prevent demonstrating the capability to cope for the appropriate duration will be addressed through hardware and/or procedural changes so that successful demonstration is possible. (Added in NUMARC-8700, transmitted by letter from J. Opeka to T. P. Speis, September 19, 1987.)

The industry's initial four initiatives included some of the same elements that are included in the staff's resolution discussed in Section 3.1. However, the industry initiatives (1) did not include rulemaking, (2) did not require plants to be able to withstand a station blackout for a specified period of time, and (3) did not require any specific assessment of a plant's station blackout coping capability. With the addition of Initiative 5, industry has proposed to perform coping assessments for a specified period of time on a plant specific basis.

3.5 Alternative (v)

Under this alternative no action would be taken.

4 CONSEQUENCES

4.1 Costs and Benefits of Alternative Resolutions

4.1.1 Alternative (i)

The benefit from implementing the station blackout rule and regulatory guide is a reduction in the frequency of core damage due to station blackout and the associated risk of offsite radioactive releases. The costs are primarily those incurred by industry (1) to assess the plant's capability to cope with a station blackout, (2) to develop procedures, (3) to improve diesel generator reliability if the reliability falls below certain levels, and (4) to retrofit plants with additional components or system, as necessary, to meet the requirements. These are discussed in the following paragraphs.

(1) Value: Risk Reduction Estimates

To estimate the change in expected risk that the resolution of USI A-44 could effect, both the postulated radioactive exposure (in person-rems) that would result in the event of an accident and the reduction in frequency of core damage have been estimated. A simplified method to estimate public dose for value-impact analysis would use an "average" plant to estimate the consequences of station blackout and subsequent core damage for all plants. However, using a single value does not account for the differences in offsite consequences associated with differences in the sizes of reactors and with differences in the population densities around different sites.

Because of the differences between sites and plant designs, it was not realistic to select a "typical" plant for analysis (using the value and impacts for that plant and then multiplying them by the total number of plants) to obtain an overall value-impact ratio. Instead, the staff used the method described below to estimate offsite consequences for use in this value-impact analysis. Results indicate that consequences range from 0.5 to 9 million person-rems per plant, with an average of about 2 million person-rems per plant.

NUREG/CR-2723 gives estimates of offsite consequences of potential accidents at nuclear power plants. That report includes results of calculations for 91 sites in the United States that had reactors with operating licenses or construction permits. The actual distributions of population around the sites were used in calculating estimated total population doses (in person-rems) for various fission product releases. The results include a scaling factor to account for different reactor power levels at the various sites.

The scaled results (from NUREG/CR-2723) for release category SST1* (siting source term) were used to develop estimates of site-specific consequences for station blackout events. However, these results were not used directly in the value-impact analysis for several reasons. First, SST1 overestimates the fission product release for station blackout events. Second, the consequences given in NUREG/CR-2723 include the entire population around the plant (i.e., an infinite radius), whereas Enclosure 1 of NRR Office Letter No. 16 (NRC, 1986) specifies that a 50-mile radius around the plant is to be used to calculate risk reduction estimates for value-impact analyses.

Extensive research efforts by NRC and industry have been under way since about 1981 to evaluate severe accident source terms and are reported in NUREG-0956, NUREG-1150, NUREG/CR-4624, and Industry Degraded Core Rulemaking (IDCOR) technical reports. Based on NRC's source term research, it appears that, for station blackout events, the release fractions for most plants would be roughly 1/3 to 1/30 of the releases from the SST1 estimate. One reason for this reduction is that SST1 is an estimated upper bound assuming prompt containment failure;

*Five release categories, denoted as SST1-SST5, have been defined by NRC to represent a spectrum of five accident groups. Each category represents a different degree of core degradation and failure of containment safety features. Group 1, SST1, is the most severe and involves a loss of all installed safety features and direct breach of containment.

whereas if a core melt resulted from station blackout, containment failure would be delayed for a number of hours. Results of a sensitivity study in which the consequences of a severe accident were estimated for reduced source terms indicate that if the SST1 release fraction were reduced by a factor of 3 (i.e., 66 percent reduction in SST1 releases), the consequences in terms of person-rem would be reduced by about 50 percent (NUREG/CR-2723, Table 10). Likewise, if the SST1 releases were reduced by a factor of 30 (i.e., 97 percent reduction in SST1 releases), the estimated person-rem would be reduced by about 85 percent. Therefore, the high and low estimates for person-rem consequences for station blackout accidents used in this value-impact analysis are 0.5 and 0.15 of the person-rem associated with SST1 releases, respectively. (These values correspond to reductions in SST1 release fractions by factors of 3 and 30, respectively.) A value of 0.33 of the SST1 person-rem was used as a best estimate for purposes of this analysis.

Scaling factors comparing offsite exposures within a 50-mile radius of a plant to that for an infinite radius are included in Table 3 of Sandia (1983). The total person-rem exposure within a 50-mile radius is approximately 1/4 the person-rem exposure for an infinite radius. This factor, in addition to the factor discussed above associated with reduced source terms, was used to scale the site-specific results from NUREG/CR-2723.

To clarify the discussion above, an example calculation is given for an 845-MWe PWR (Calvert Cliffs). From Appendix A of NUREG/CR-2723, the mean offsite effect conditional on release for the SST1 category is 3.61×10^7 person-rems. This number is multiplied by 0.33 to account for the smaller releases for station blackout events compared to SST1 releases and by 0.25 to account for the 50-mile radius (Sandia, 1983). The resulting offsite exposure from a station blackout event and subsequent core melt within a 50-mile radius of the plant is estimated to be about 3 million person-rems.

The reduction in frequency of core damage resulting from the resolution of USI A-44 was estimated for each plant. Plant- and site-specific characteristics for a total of 100 reactors (which represent almost all of the currently operating nuclear power plants) were used to develop these estimates. Table 8 presents an estimate of the number of reactors having the emergency ac power configurations and offsite power design characteristics identified in Tables 2 and 3, respectively. The estimate of core damage frequency for each plant was based on a function of the plant's ability to cope with a station blackout (NUREG-1032). The staff assumed that all plants, as currently designed, can cope with a station blackout for 2 hours. The reduction in core damage frequency per reactor-year for each plant then was estimated based on the plant meeting the acceptable 2-, 4-, 8-, or 16-hour station blackout duration depending on the plant's offsite power design group and its emergency ac power configuration (given in Table 1).

Examples of the reduction in frequency of core damage per reactor-year for three cases are presented in Table 9. Each of these examples is for a plant located in an area with average loss of offsite power duration and frequency. The first example is typical of a plant with one redundant emergency ac power system (e.g., one out of two diesel generators required for emergency ac power), and a failure rate of 0.025 failure per demand for each diesel generator. The second case,

which is typical of a plant with less desirable characteristics from a station blackout perspective (e.g., a minimum redundant emergency ac power system and below-average diesel generator reliability), has a reduction in frequency of core damage that is significantly larger than the first example. The third case is for plants with more favorable characteristics than in the first case and, therefore, a correspondingly lower reduction in core damage frequency.

A summary of the results of the analysis for station blackout core damage frequency estimates is presented in Figure 4. This figure presents a comparison of the estimated number of reactors versus various levels of core damage frequency before and after implementation of the station blackout rule. The histogram that represents estimates before the rule is implemented is based on the assumption that all plants have the capability to cope with station blackout for only 2 hours. The estimated mean core damage frequency for this case is 4.2×10^{-5} per reactor-year, with a range of from about 0.4×10^{-5} to 30×10^{-5} per reactor-year. The mean core damage frequency for all plants after the rule is implemented is estimated to be 1.6×10^{-5} per reactor-year with a range of 0.3×10^{-6} to 7×10^{-5} per reactor-year. Therefore, on an industry-wide basis, the estimated mean core damage frequency would be reduced by 2.6×10^{-5} per reactor-year.

For each plant the estimated risk reduction from the resolution of USI A-44 was calculated by multiplying the reduction in core damage frequency per reactor-year by two factors: (1) the remaining life of the plant (assumed to be 25 years) and (2) the estimated public dose (in person-rems) that would result in the event of an accident. The reduction in person-rems for each plant was then summed to calculate the total estimated risk reduction. The high estimate of total dose reduction (on SST1 releases divided by 3) is 215,000 person-rems, the low estimate (based on SST1 releases divided by 30) is 65,000 person-rems, and the best estimate is 143,000 person-rems (based on SST1 releases divided by 10).

Table 8 Estimated number of reactors having similar characteristics

Emergency ac power configuration group*					
Group	A	B	C	D	Total
Estimated number of reactors	12	25	47	16	100
Offsite power design characteristics**					
Characteristic	P1	P2		P3	Total
Estimated number of reactors	30	60		10	100

*See Table 2 for definition of emergency ac power configuration groups.
**See Table 3 to determine offsite power design characteristics.

Table 9 Examples of reduction in frequency of core damage per reactor-year

Plant characteristics	Estimated core damage frequency per reactor-year	Estimated reduction in core damage frequency per reactor-year
Plant with one of two emergency diesel generators (EDGs); EDG failure rate of 0.025 failure per demand; and loss of offsite power design characteristic group P2.	3.9×10^{-5} with 2-hour station blackout capability 1.8×10^{-5} with 4-hour* station blackout capability	2.1×10^{-5}
Plant with two out of three EDGs; EDG failure rate of 0.05 failure per demand; and loss of offsite power design characteristic group P2.	9.0×10^{-5} with 2-hour station blackout capability 0.6×10^{-5} with 8-hour* station blackout capability	8.4×10^{-5}
Plant with one out of three EDGs; EDG failure rate of 0.025 failure per demand; and loss of offsite power design characteristic group P2.	1.0×10^{-5} with 2-hour station blackout capability 0.4×10^{-5} with 4-hour* station blackout capability	0.6×10^{-5}

*These times are the acceptable station blackout durations from Table 1 for these example cases.

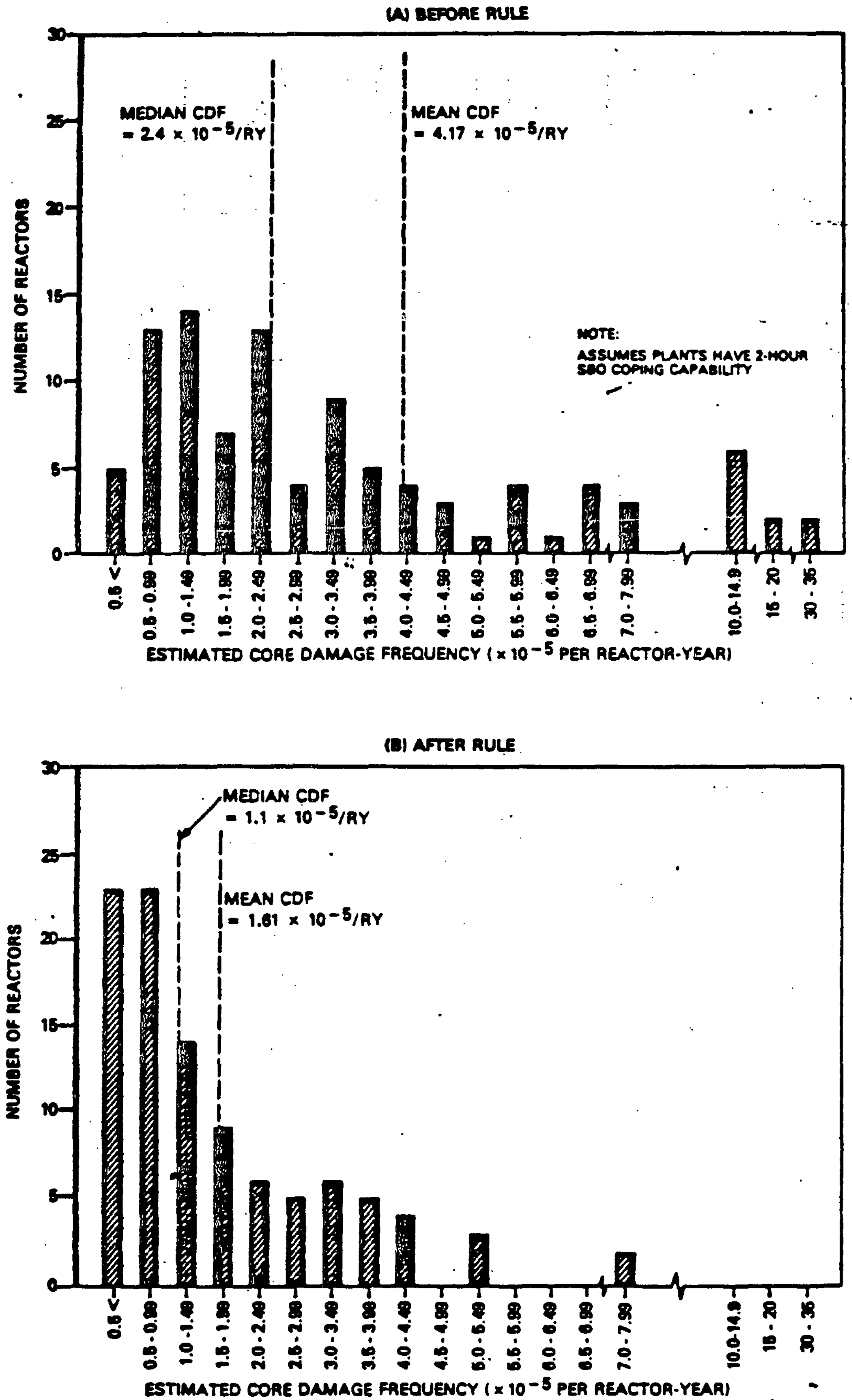

Figure 4 Comparison of estimated station blackout (SBO) core damage frequency (CDF) before and after rule

(2) Impacts: Cost Estimates

The cost for licensees to comply with the requirements to resolve USI A-44 will vary depending on (1) the existing capability of each plant to cope with a station blackout and (2) the plant-specific acceptable minimum station blackout coping duration as determined from Table 1. The staff anticipates that the majority of plants would be able to meet a 4-hour duration guideline without major hardware modifications. In addition to being able to withstand a 4-hour blackout, some plants may be capable of coping for longer periods without major modifications. To meet an 8-hour guideline, licensees of some plants may have to increase the capacity of one or more of the following systems: station batteries, condensate storage tank, and instrument or compressed air. Shedding nonessential loads from the station batteries could be considered as an option to extend the time until battery depletion. Corresponding procedures for load shedding would need to be incorporated in the plant-specific technical guidelines and emergency operating procedures for station blackout.

If equipment needed to function during a station blackout or the recovery from a blackout would not be expected to be operable because of environmental conditions associated with the station blackout (i.e., without heating, ventilating, and air conditioning systems operating), then some modifications might be necessary. These could be (1) opening room or cabinet doors to increase natural circulation, (2) installing fans that can operate with available power supplies to increase forced circulation, or (3) relocating or replacing equipment. If option 2 or 3 above were necessary, then corresponding procedures would need to be incorporated in the plant-specific technical guidelines and emergency operating procedures for station blackout.

Those plants that cannot verify adequate reactor coolant pump seal integrity for the station blackout duration may have to provide a method of reactor coolant pump seal cooling that is independent of the offsite and emergency onsite ac power supplies to maintain seal integrity and adequate reactor coolant inventory. For example, the addition of an ac-independent charging pump or a steam-driven generator to power an existing charging pump could provide seal cooling during a station blackout.

Table 10 presents cost estimates of possible hardware modifications and procedures that could result from implementation of the station blackout rule. Because the duration guidelines in the station blackout regulatory guide are based on plant-specific features, and the capability of systems and components needed during a station blackout varies from plant to plant, the modifications in Table 10 may be needed at some but not all nuclear power plants. For each modification, the table identifies an estimated range of costs per plant, the estimated number of plants needing that modification, and the estimated total cost.

The estimated total cost for industry to comply with the resolution of USI A-44 is about $60 million. The estimated average cost per reactor is $600,000. Best estimates of costs could range from $350,000, if only a station blackout assessment and procedures and training were necessary, to a maximum of about $4 million, if modifications 1 through 4 were needed (including requalification of a diesel generator).

Table 10 Estimated costs for industry to comply with the resolution of USI A-44[1]

Potential modifications	Est. no. of reactors needing modifications	Est. cost per reactor ($1000) Best est.	High est.	Low est.	Est. total cost ($1000) Best est.	High est.	Low est.
1. Assess plant's capability to cope with station blackout	100	250	400	200	25,000	40,000	20,000
2. Develop procedures and training	100	100	150	50	10,000	15,000	5,000
3. (a) Improve diesel generator reliability	10	250	400	150	2,500	4,000	1,500
(b) Requalify a diesel generator	2	2,800	5,500	1,250	5,600	11,000	2,500
4. Increase capability to cope with station blackout[2]							
(a) 4-hour plants add battery capacity	10	500	650	400	5,000	6,500	4,000
(b) 8-hour plants	17						
(1) Add compressed air		40	60	30	680	1,020	510
(2) Add condensate storage tank capacity		80	150	40	1,360	2,550	680
(3) Add battery capacity		500	650	400	8,500	11,050	6,800
(4) Replace equipment or add fans		80	140	30	1,360	2,380	510
Subtotal (8-hour plants)		700	1,000	500	11,900	17,000	8,500
5. Add an ac-independent charging pump (non-seismic) capable of delivering 50 to 100 gpm to reactor coolant pump seals[3]	--	1,500	2,500[4]	1,200	--	--	--
TOTAL COSTS					60,000	93,500	41,500

[1]Based on 100 reactors. See Appendix B for worksheets that provide the basis for the cost estimates on this table.
[2]Detailed cost estimates for these modifications are presented in NUREG/CR-3840 and revised estimates to that report (Science and Engineering Associates, 1986).
[3]It is assumed that reactor coolant pump seal integrity is sufficient to ensure core cooling for 8 hours or more; therefore, the charging pump would not be necessary. The results of Generic Issue B-23 will provide detailed information on expected pump seal behavior without seal cooling. (See Section 4.2 for further discussion.) Estimated costs are provided here for perspective should such a system be considered necessary after Generic Issue B-23 results are available.
[4]A seismically qualified and safety-grade ac-independent charging pump would be much more expensive and would not reduce the risk substantially more than a non-seismic pump.

Including costs of averted plant damage can significantly affect the overall cost-benefit evaluation. To estimate the costs of averting plant damage and cleanup, the reduction in accident frequency was multiplied by the discounted onsite property costs. The following equations from NUREG/CR-3568 were used to make this calculation:

$$V_{op} = N \Delta F U$$

$$U = C/m\ [(e^{-rt_i})/r^2]\ [1 - e^{-r(t_f - t_i)}](1 - e^{-rm})$$

where

V_{op} = value of avoided onsite property damage

N = number of affected facilities = 100

ΔF = reduction in accident frequency = 2.6×10^{-5}/reactor-year

U = present value of onsite property damage

C = cleanup and repair costs = $1.2 billion

t_f = years remaining until end of plant life = 25

t_i = years before reactor begins operation = 0

r = discount rate = 5% and 10%

m = period of time over which damage costs are paid out (recovery period in years) = 10

Using the above values, the present value of avoided onsite property damage is estimated to be $19 million. If avoided costs for replacement power are included (estimated in NUREG/CR-3568 to be $1.2 billion over 10 years), the estimated present value is $38 million. Table 11 summarizes the discounted present value of avoided onsite property damage for 10% and 5% discount rates.

Table 11 Discounted present value of avoided onsite property damage for 100 reactors

	Discounted present value	
Avoided damage	10% discount rate	5% discount rate
Cleanup and repair only	$\$19 \times 10^6$	$\$40 \times 10^6$
Cleanup, repair, and replacement power	$\$38 \times 10^6$	$\$80 \times 10^6$

(3) Value-Impact Ratio

Table 12 summarizes the total benefits and costs associated with the resolution of USI A-44. These include (1) public risk reduction due to avoided offsite releases associated with reduced accident frequencies; (2) increased occupational dose from implementation, and operation and maintenance activities, as well as reduced occupational exposure from cleanup and repair because of lower accident frequency; (3) industry costs for implementation of modifications, operation

Table 12 Value-impact summary for resolution of USI A-44

Parameter	Dose reduction (person-rems) Best est.	High est.	Low est.	Cost ($1,000) Best est.	High est.	Low est.
Public health	143,000	215,000	65,000			
Occupational exposure (accidental)[1]	1,500	1,500	1,500			
Occupational exposure (routine)[2]	NA					
Industry implementation				60,000	93,500	44,500
NRC implementation[3]				1,500	1,500	1,500
Total	144,500	216,500	66,500	61,500	95,000	43,000
Value-impact ratio[4] (Public dose reduction divided by sum of NRC and industry costs (person-rems/$\$10^6$))				2,400	5,000	700

[1]Based on an estimated occupational radiation dose of 20,000 person-rems for post-accident cleanup and repair activities (NUREG/CR-3568).

[2]No significant increase in occupational exposure is expected from operation and maintenance or implementing the recommendations proposed in this resolution. Equipment additions and modifications contemplated do not require significant work in and around the reactor coolant system and therefore would not be expected to result in significant radiation exposure. NA = not affected.

[3]Based on an estimated 175 person-hours per reactor for NRC review (NUREG/CR-3568).

[4]This does not take into account the additional benefit associated with avoided plant damage costs or replacement power costs resulting from reduced frequency of core damage. The cost for plant cleanup following a core damage accident is estimated to be $1.2 billion, and replacement power is estimated to cost about $500,000 per day (NRC, 1986). The estimated discounted present value of these avoided onsite costs is given in Table 11.

and maintenance, and increased reporting requirements; and (4) NRC costs for review of industry submittals.

The estimated total cost for industry to comply with the proposed rule is $60 million. The total public risk reduction for 100 reactors over the remaining life of the plants is about 145,000 person-rems. The overall value-impact ratio, not including onsite accident avoidance costs, is about 2,400 person-rems averted per million dollars. If cost savings to industry from accident avoidance (cleanup and repair of onsite damages and replacement power) were included, the overall value-impact ratio would improve significantly. At a 10% discount rate, the present value of avoided cleanup, repair, and replacement power is approximately $38 million. If this benefit were taken into account, the overall value-impact ratio would be about 6,100 person-rems averted per million dollars.

For any particular plant, the value-impact ratio could vary significantly (either higher or lower) than the ratio given above. However, even for plants that will not require equipment modifications to comply with the station blackout rule, the assessment of plant capability to cope with a station blackout is almost certain to result in improvements in training and procedures to handle such an event. At a ratio of $1,000 per person-rem, a decrease in core damage frequency of only about 0.5×10^{-6} per reactor-year is sufficient to justify a cost of $350,000 for the station blackout assessment and procedures and training. Improvements to enhance the capability of a plant to cope with a station blackout from 2 to 4 hours would effect such a reduction in core damage frequency for virtually all plants.

(4) Special Considerations

The quantitative value-impact analysis discussed above used estimates for benefits (risk reduction) and costs associated with the resolution of USI A-44. While this is a useful approach to evaluate the resolution, other factors can and should play a part in the decision-making process. Although they are not quantified, other considerations that bear on the overall conclusions and recommendations to resolve USI A-44 are discussed below. Overall, these considerations support the conclusion that additional defense in depth provided by the ability of a plant to cope with a station blackout for a specified duration is strongly recommended.

- Relative Importance of Potential Station Blackout Events

Probabilistic risk assessment (PRA) studies performed for this USI, as well as a number of plant-specific PRAs, have shown that station blackout can be a significant contributor to core damage frequency, and, with the consideration of containment failure, station blackout events can represent an important contributor to reactor risk. In general, active containment systems required for heat removal, pressure suppression, and radioactivity removal from the containment atmosphere following an accident are unavailable during a station blackout. Therefore, the offsite risk is higher from a core melt resulting from station blackout than it is from many other accident scenarios.

- Source Term Re-Evaluation

The consequence estimates for station blackout used in this value-impact analysis are consistent with the latest research by NRC on source term re-evaluation.

The release fractions used in this analysis are significantly lower than earlier estimates of source terms. Nevertheless, there is still considerable uncertainty, and source term research is expected to continue in the future to improve our knowledge of major phenomena and refine analytical models. Given the range of release fractions used in this analysis, it is unlikely that significantly better estimates agreed to by the staff and industry would be available for a number of years. In any event, the ability to cope with a station blackout for some period of time would make station blackout a small contributor to core damage frequency and would significantly reduce the risk associated with such events.

- Future Trends in Loss of Offsite Power Frequency

The estimated frequency of core damage from station blackout events is directly proportional to the frequency of the initiating event. Estimates of station blackout frequencies for this USI were based on actual operating experience with credit given in the analysis for trends that show a reduction in the frequency of losses of offsite power resulting from plant-centered events (NUREG-1032). This is assumed to be a realistic indicator of future performance. An argument can be made that the future performance will be better than the past. For example, when problems with the offsite power grid arise, they are fixed, and therefore, grid reliability should improve. On the other hand, grid power failures may become more frequent because fewer plants are being built, and more power is being transmitted between regions, thus placing greater stress on transmission lines.

- Trends in Emergency Diesel Generator Performance

Recent data indicate that average emergency diesel generator reliability on an industry-wide basis has been improving slightly since 1976 (NUREG/CR-4347, NSAC/108). These data are based on total valid failures and total valid starts including surveillance testing and unplanned demands (e.g., following a loss of offsite power). There are an insufficient number of unplanned demands at any one nuclear plant to determine diesel generator reliability with high statistical confidence. Therefore, target diesel generator performance levels for USI A-44 are based primarily on surveillance tests. However, data show that the industry average diesel generator failure rate during unplanned demands was higher than that during surveillance tests (0.014 failure per demand for surveillance tests compared to 0.022 failure per demand during unplanned demands (NSAC/108)). Using diesel generator reliability based only on unplanned demands would lead to slightly higher estimates of core damage frequency than was used in this regulatory analysis and, therefore, a correspondingly larger estimated benefit resulting from the resolution of USI A-44.

- Common Cause Failures

One factor that affects ac power system reliability is the vulnerability to common cause failures associated with design, operational, and environmental factors. Existing industry and NRC standards and regulatory guides include specific design criteria and guidance on the independence of offsite power circuits and the independence of, and limiting interactions between, diesel generator units at a nuclear station. In developing the resolution of USI A-44, the NRC staff assumed that, by adhering to such standards, licensees have minimized, to the extent practical, single-point vulnerabilities in design and operation that could result

in a loss of all offsite power or all onsite emergency ac power. Results of sensitivity studies presented in NUREG-1032 indicate that if potential common cause failures of redundant emergency diesel generators exist (e.g., in service water or dc power support systems), then estimated core damage frequencies can increase significantly.

- Sabotage

There have not been any total losses of offsite power or diesel generator failures attributed to sabotage. Therefore, sabotage was not considered explicitly in the risk analysis for USI A-44. However, there was a sabotage event in 1986 that caused three out of four 500-kV transmission lines at one site to be out of service for several hours. Thus sabotage could increase the probability of loss of offsite power. If saboteurs managed to simultaneously take out all offsite power and/or emergency diesel generators, the resolution of USI A-44 would provide additional defense-in-depth for a period of time to cope with such an event.

4.1.2 Alternative (ii)

The alternative of treating plants uniformly by requiring all plants to be able to cope with the same station blackout duration has been considered. This simplified approach has the advantage of being potentially easier to implement, but it also has two major drawbacks. First, operating nuclear power plants have significant differences in plant- and site-specific factors that contribute to risk from station blackout. This alternative would not take these known factors into account. For example, plants that have a more redundant emergency ac power system than other plants would not be given any credit for such features. Second, requiring all plants to be able to cope with the same blackout duration would result in one of two undesirable alternatives: (1) If a uniform duration of 4 hours or less were recommended, station blackout could still be a significant contributor to total core damage frequency for some plants and, therefore, the objective of the requirements would not be met; and (2) if a uniform 8-hour requirement were imposed, it would necessitate expenditures at some plants that would not be considered cost-effective in reducing the risk from station blackout events. Therefore, this alternative was not recommended.

4.1.3 Alternative (iii)

Another possible alternative to the recommended action is to require plants to install either an additional emergency diesel generator or another ac-independent decay heat removal system. This alternative was not recommended for several reasons. First, the cost for either of these additions (from $10 to $30 million per plant) is much higher than the estimated cost for the recommended resolution. The recommended approach is more cost-effective and meets the objective stated in Section 2. Second, the adequacy of present requirements for decay heat removal systems is being studied under USI A-45, and any major hardware changes or additions to these systems should await the technical resolution of USI A-45. Third, experience indicates that there are practical limits to diesel generator reliability, including common cause failures of redundant divisions, and the recommended resolution provides greater diversity and additional defense-in-depth.

4.1.4 Alternative (iv)

The five initiatives proposed by NUMARC are very similar to the elements proposed by the staff for resolution of this issue. NUMARC's guidelines and technical bases for addressing the station blackout issue are presented in NUMARC-8700. The procedure in NUMARC-8700 has been referenced in Regulatory Guide 1.155 as providing an acceptable method to meet the requirements of the rule. Industry now finds the proposed rule and regulatory guide acceptable.

4.1.5 Alternative (v)

This alternative would be to take no actions beyond those resulting from the NUMARC initiatives endorsed by industry and the resolution of Generic Issue B-56 (see discussions in Sections 3.4, 4.1.4, and 4.2.1). Operating experience with diesel generator failures and losses of offsite power has raised a significant concern regarding the potential risk from a station blackout event. The use of this data base with relatively straightforward application of PRA techniques indicates that station blackout events could be a significant contributor to risk for many plants. The additional actions recommended for USI A-44 would significantly reduce the estimated frequency of core damage associated with severe accidents from station blackout. Because the value-impact analysis has shown that it would be beneficial to implement these recommendations, the no-action alternative is not recommended.

4.2 Impacts on Other Requirements

Several ongoing NRC generic programs and requirements that are related to the resolution of USI A-44 are discussed below.

4.2.1 Generic Issue B-56, Diesel Generator Reliability

The resolution of USI A-44 includes a regulatory guide on station blackout that specifies the following guidance on diesel generator reliability (Regulatory Guide 1.155, Sections C.1.1 and 2):

> The reliable operation of the onsite emergency AC power sources should be ensured by a reliability program designed to monitor and maintain the reliability of each power source over time at a specified acceptable level and to improve the reliability if that level is not achieved. The reliability program should include surveillance testing, target values for maximum failure rate, and a maintenance program. Surveillance testing should monitor performance so that if the actual failure rate exceeds the target level, corrective actions can be taken.
>
> The maximum emergency diesel generator failure rate for each diesel generator should be maintained at or below 0.05 failure per demand. For plants having an emergency AC power system [configuration requiring two-out-of-three diesel generators or having a total of two diesel generators shared between two units at a site], the emergency diesel generator failure rate for each diesel generator should be maintained at 0.025 failure per demand or less.

In Generic Letter 84-15, dated July 2, 1984, the staff requested information from licensees regarding proposed actions to improve and maintain diesel generator reliability. The letter requested specific information on three areas

(1) reduction of cold fast-start surveillance tests for diesel generators
(2) diesel generator reliability
(3) the licensee's diesel generator reliability program, if any, and comments on the staff's example performance technical specifications for diesel generator reliability

A summary of the data and recommendations in response to Generic Letter 84-15 was published in NUREG/CR-4557. This information, along with other input, will be used in the resolution of Generic Issue B-56 to provide specific guidance for diesel generator reliability programs consistent with the resolution of USI A-44.

4.2.2 USI A-45, Shutdown Decay Heat Removal Requirements

The overall objective of USI A-45 is to evaluate the adequacy of current licensing requirements to ensure that nuclear power plants do not pose an unacceptable risk as a result of failure to remove shutdown decay heat following transients or small break loss-of-coolant accidents. The study includes an assessment of alternative means of improving shutdown decay heat removal and of an additional "dedicated" system for this purpose. Results will include proposed recommendations regarding the desirability of, and possible design requirements for, improvements in existing systems or an additional dedicated decay heat removal system.

The USI A-44 concern for maintaining adequate core cooling under station blackout conditions can be considered a subset of the overall USI A-45 issue. However, there are significant differences in scope between these two issues. USI A-44 deals with the probability of loss of ac power, the capability to remove decay heat using systems that do not require ac power, and the ability to restore ac power in a timely manner. USI A-45 deals with the overall reliability of the decay heat removal function in terms of response to transients, small break loss-of-coolant accidents, and special emergencies such as fires, floods, seismic events, and sabotage.

Although the recommendations that might result from the resolution of USI A-45 are not yet final, some could affect the station blackout capability, while others would not. Recommendations that involve a new or improved decay heat removal system that is ac power dependent but that does not include its own dedicated ac power supply would have no effect on USI A-44. Recommendations that involve an additional ac-independent decay heat removal system would have a very modest effect on USI A-44. Recommendations that involve an additional decay heat removal system that include its own ac power supply would have a significant effect on USI A-44. Such a new additional system would receive the appropriate credit within the USI A-44 resolution by either changing the emergency ac power configuration group or providing the ability to cope with a station blackout for an extended period of time.

The resolution of USI A-44 would necessitate average expenditures of about $600,000 per plant, with a range estimated to be from about $350,000 to a maximum of around $4 million. A resolution for USI A-45 involving the addition of a dedicated and independent system, such as an additional shutdown cooling system with its own dedicated diesel generator, would be much more expensive, with an expenditure on the order of $50 to $100 million. However, such expenditures would resolve other concerns with respect to the decay heat removal function which will be delineated in a future regulatory analysis for USI A-45.

The resolution of these two issues is coordinated along two main lines. First, technical information resulting from both studies is shared among the major participants including NRC staff and contractors. In this way, the resolution of USI A-45 will take into account any modifications resulting from the resolution of USI A-44 that are applicable to the decay heat removal function. Second, the schedules are coordinated so that by the time a final rule on USI A-44 is published--and well before plant modifications, if any, would be implemented--it is anticipated that the proposed technical resolution of USI A-45 will be published for public comment.

The technical summary findings report and the regulatory analysis for the proposed resolution of USI A-45 are targeted to be issued for public comment in mid-1988. For plants needing hardware modifications to comply with the USI A-44 resolution, this schedule would permit a re-evaluation before any actual modifications are made so that any contemplated design changes following from the resolution of USI A-45 can be considered at the same time.

4.2.3 Generic Issue B-23, Reactor Coolant Pump Seal Failures

The Task Action Plan for Generic Issue B-23 includes three tasks: (1) a review of seal failure operating experience, (2) an assessment of the effects of loss of seal cooling on reactor coolant pump (RCP) seal behavior, and (3) an evaluation of other causes of RCP seal failure such as mechanical and maintenance-induced failures. Only task 2 is closely related to USI A-44 because during a station blackout, systems that normally provide RCP seal cooling are unavailable, and RCP seal integrity is necessary for maintaining primary system inventory under station blackout conditions.

NRC and industry analyses of seal performance with loss of seal cooling are proceeding, but at this time the staff has not completed its recommendations to resolve Generic Issue B-23. The estimates of core damage frequency for station blackout events in NUREG/CR-3226 assumed that the RCP seals would leak at a rate of 20 gallons per minute (gpm) per pump. Results of the analysis for Generic Issue B-23 will provide the information necessary to determine seal behavior and, likewise, a plant's ability to cope with a station blackout for a specified time. Should this analysis conclude that there is a significant probability that RCP seals can leak at rates substantially higher than 20 gpm, then modifications such as an ac-independent RCP seal cooling system may be necessary to resolve Generic Issue B-23. If there is high probability that the RCP seals would not leak excessively during a station blackout, then no modifications would be required. A cost-benefit analysis associated with the need for an ac-independent seal cooling system would be included in the regulatory analysis for Generic Issue B-23.

4.2.4 Generic Issue A-30, Adequacy of Safety-Related DC Power Supply*

The analysis performed for USI A-44 (NUREG-1032) assumed that a high level of dc power system reliability would be maintained so that (1) dc power system failures would not be a significant contributor to losses of all ac power and (2) should a station blackout occur, the probability of immediate dc power system failure would be low. Whereas Generic Issue A-30 focuses on enchancing battery reliability (e.g., restricting interconnections between redundant dc divisions, monitoring the readiness of the dc power system, specifying administrative procedures and technical specifications for surveillance testing and maintenance activities), the resolution of USI A-44 is aimed at ensuring adequate station battery capacity in the event of a station blackout of a specified duration. Generic Issue A-30 would provide additional assurance that station battery reliability is adequate and consistent with the assumptions on which USI A-44 is based. Therefore, these two issues are consistent and compatible.

4.2.5 Regulatory Guide 1.108, Periodic Testing of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants

Regulatory Guide 1.108 describes the currently acceptable method for complying with the Commission's regulations with regard to periodic testing of diesel generators to ensure that they will meet their availability requirements. This guide may need to be modified to be consistent with the proposed actions described in Section 4.2.1 above (Generic Issue B-56). Regulatory Guide 1.108 will be revised to be consistent with the resolutions of USI A-44 and Generic Issue B-56.

4.2.6 Fire Protection Program for Nuclear Power Facilities

10 CFR 50.48 states that each operating nuclear power plant shall have a fire protection plan that satisfies GDC 3. The fire protection features required to satisfy GDC 3 are specified in Appendix R to 10 CFR 50 and in Branch Technical Position CMEB 9.5.1 (NUREG-0800). They include certain provisions regarding alternative and dedicated shutdown capability. To meet these provisions, some licensees have added, or plan to add, improved capability to restore power from offsite sources or onsite diesels for the shutdown system. A few plants have installed a safe shutdown facility for fire protection that includes a charging pump powered by its own independent ac power source. In the event of a station blackout, this system can provide makeup capability to the primary coolant system as well as reactor coolant pump seal cooling. This could be a significant benefit in terms of enhancing the ability of a plant to cope with a station blackout.

Because the plant modifications required for fire protection have already been specified, it would not be feasible to consider these modifications together with the requirements of USI A-44. However, credit would be given for improve-

*Generic Issue A-30 is being resolved as part of Generic Issue B-128, Electrical Power Issues. Generic Issue A-30 is the only part of Generic Issue B-128 that is closely related to USI A-44.

ments made for the fire protection program in meeting the station blackout rule. For example, plants that have added equipment to achieve alternate safe shutdown in order to meet Appendix R requirements could take credit for the equipment (if available) for coping with a station blackout event.

4.2.7 Generic Issue B-124, Auxiliary Feedwater System Reliability

This issue has focused on the reliability of seven older PWRs that have two-train auxiliary feedwater systems. The staff has established a review team which will perform reviews (including plant audits and walkdowns) to assess each of these plants on a case-by-case basis. Other relevant information such as auxiliary feedwater system reliability analyses will be considered in the staff reviews, as available. The staff may allow credit for compensating factors, such as feed and bleed capability, to justify acceptance of the two-pump auxiliary feedwater systems, or may decide that hardware, procedural, and/or training modifications are necessary.

If the proposed resolution of Generic Issue B-124 requires the auxiliary feedwater system in several PWRs to be upgraded, this would most likely result in the addition of an auxiliary feedwater pump. The installation of a pump that is independent of ac power would be beneficial in handling station blackout accident sequences by providing additional reliability in the ac-independent decay heat removal system. Because all PWRs now have an auxiliary feedwater train that is independent of ac power, the requirement could be met by adding a motor-driven pump. Consequently, the auxiliary feedwater system upgrades could have no effect on the station blackout issue.

4.2.8 Multiplant Action Items B-23 and B-48, Degraded Grid Voltage and Adequacy of Station Electric Distribution Voltage

These two multiplant action items have been under consideration by both the staff and licensees for several years. They relate to (1) sustained degraded voltage conditions at the offsite power sources, (2) interaction between the offsite and onsite emergency power systems, and (3) the acceptability of the voltage conditions on the station electric distribution systems with regard to potential overloading and starting transient problems. Licensees' responses to these concerns have consisted of verifying the adequacy of existing power systems or of upgrading the power systems. The modifications are designed to ensure that the power systems can perform their intended function and consequently would enhance their dependability. If additional power sources have been added to address these concerns, the plant would be placed in an improved category and may be required to withstand a blackout of lesser duration. In the resolution of USI A-44, the staff is not recommending that work that has been done on these two action items be repeated.

4.2.9 Severe Accident Program

Brookhaven National Laboratory (BNL) has proposed a set of preliminary guidelines and criteria that could be used to assess the capability of nuclear power plants to cope with severe accidents (for example, see BNL Technical Report A-3825R). This work was performed in support of the Implementation Plan for

the Commission's Severe Accident Policy Statement. The proposed guidelines cover a large number of potentially severe accident sequences. For station blackout events, the guidelines assume that plants will comply with the requirements in the station blackout rule. Therefore, the severe accident program and the resolution of USI A-44 are consistent and compatible. Requirements for operating plants to comply with additional criteria beyond those in the station blackout rule would need to be justified in accordance with the backfit rule (10 CFR 50.109).

4.3 Constraints

The staff has reviewed current Commission regulations to determine if they provide a basis for implementation of the USI A-44 requirements. This review included (1) the Atomic Safety and Licensing Appeal Board Hearing (ALAB-603) on station blackout for St. Lucie Unit 2; (2) the Commission review of that hearing; (3) GDC 17, "Electric Power Systems"; and (4) the backfit rule (10 CFR 50.109).

- St. Lucie Unit 2 Atomic Safety and Licensing Appeal Board Hearing

In ALAB-603, the board took the position that station blackout should be considered a design-basis event for St. Lucie Unit 2 because of the high frequency of such an event (10^{-4} to 10^{-5} per year at that site). As a result, the Appeal Board required St. Lucie Unit 2 to be capable of withstanding a total loss of ac power and to implement training and procedures to recover from station blackout. The Appeal Board went as far as to say,

> Our findings that station blackout should be considered as a design basis event for St. Lucie Unit 2 manifestly could be applied equally to Unit 1, already in operation at that site. By a parity of reasoning, this result may well also obtain at other nuclear plants on applicant's system, if not at most power reactors. Our jurisdiction, however, is limited to the matter before us, licensing construction of St. Lucie 2. Beyond that, we can only alert the Commission to our concerns.

The Commission upheld the Board's action on St. Lucie Unit 2. However, the Commission determined that ALAB-603 did not establish station blackout generically as a design-basis event.

- General Design Criterion 17

GDC 17 states, in part,

> Provisions shall be included to minimize the probability of losing electric power from any of the remaining supplies 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 electric power supplies.

The intent of GDC 17 is to require reliable offsite and onsite ac power systems. The ability to cope with the coincident loss of both of these systems is not addressed explicitly.

As a result of this review, the staff has concluded that there is a basis in the regulations for the recommendations to improve the reliability of the offsite and onsite ac power systems. However, because the coincident loss of both systems is not addressed explicity, a rule to require plants to be able to withstand a total loss of ac power for a specified duration will provide further assurance that station blackout will not adversely affect the public health and safety.

- Backfit Rule

On September 20, 1985, the Commission published the backfit rule (10 CFR 50.109). This rule sets forth restrictions on imposing new requirements on currently licensed nuclear power plants and specifies standard procedures that must be applied to backfitting decisions. The backfit rule states,

> The Commission shall require a systematic and documented analysis pursuant to paragraph (c) of this section for backfits which it seeks to impose....(10 CFR 50.109(a)(2))
>
> The Commission shall require the backfitting of a facility only when it determines, based on the analysis described in paragraph (c) of this section, that there is a substantial increase in the overall protection of the public health and safety or the common defense and security to be derived from the backfit and that the direct and indirect costs of implementation for that facility are justified in view of this increased protection. (10 CFR 50.109(a)(3))

In order to reach this determination, 10 CFR 50.109(c) sets forth nine specific factors which are to be considered in the analysis for the backfits it seeks to impose. These nine factors are among those discussed in the main body of this report. Appendix A provides a discussion summarizing each of these factors. The Commission also states in the backfit rule that "any other information relevant and material to the proposed backfit" will be considered. This report provides additional relevant information concerning the station blackout rulemaking. This analysis supports a determination that a substantial increase in the protection of the public health and safety will be derived from backfitting the requirements in the station blackout rule, and that the backfit is justified in view of the direct and indirect costs of implementing the rule.

No other constraints have been identified that affect the resolution of USI A-44.

5 DECISION RATIONALE

The evaluation to resolve USI A-44 included deterministic and probabilistic analyses. Calculations to determine the timing and consequences of various accident sequences were performed, and the dominant factors affecting station blackout likelihood were identified (NUREG-1032 and NUREG/CR-2989, -3992, -3226, and -4347). Using this information, simplified probabilistic accident sequence correlations were calculated to estimate the frequency of core damage resulting from station blackout events for different plant design, operational, and location factors. These quantitative estimates were used to give insights into the relative importance of various factors, and those insights, along with engineering judgment, were used to develop the resolution of USI A-44. By analyzing the effect of variations in design, operations, and plant location on risk from

station blackout accidents, an attempt was made to approach a reasonably consistent level of risk in the recommendations developed.

A survey of probabilistic risk assessment studies showed that total core damage frequency from all dominant accident sequences ranged from 2 x 10^{-5} to 1 x 10^{-3} per reactor-year, with a typical frequency of about 6 to 8 x 10^{-5} per reactor-year (NUREG/CR-3226). For those plants currently in operation or under construction, a value-impact analysis was performed to determine that the resolution of USI A-44 is cost-effective. Implementation of the resolution will result in station blackout being a relatively small contributor to total core damage frequency. (NUREG-1032 provides a more detailed discussion of the analysis of station blackout accident likelihood performed for this regulatory analysis.)

5.1 Commission's Safety Goals

On August 4, 1986, the Commission published in the Federal Register a policy statement on "Safety Goals for the Operations of Nuclear Power Plants" (51 FR 28044). This policy statement focuses on the risks to the public from nuclear power plant operation and establishes goals that broadly define an acceptable level of radiological risk. The discussion below addresses the resolution of USI A-44 in light of these goals.

- The two qualitative safety goals are:
  - Individual members of the public should be provided a level of protection from the consequences of nuclear power plant operation such that individuals bear no significant additional risk to life and health.
  - Societal risks in life and health from nuclear power plant operation should be comparable to or less than the risks of generating electricity by viable competing technologies and should not be a significant addition to other societal risk.
- The following quantitative objectives are used in determining achievement of the above safety goals:
  - The risk to an average individual in the vicinity of a nuclear power plant of prompt fatalities that might result from reactor accidents should not exceed one-tenth of one percent (0.1%) of the sum of prompt fatality risks resulting from other accidents to which members of the U.S. population are generally exposed.
  - The risk to the population in the area near a nuclear power plant of cancer fatalities that might result from nuclear power plant operation should not exceed one-tenth of one percent (0.1%) of the sum of cancer fatality risks resulting from all other causes.

Results of analyses published in NUREG-1150 for five plants (Surry, Zion, Sequoyah, Peach Bottom, and Grand Gulf) indicate that all five plants meet the risk criteria for prompt fatalities and latent cancer fatalities stated above, even considering the large uncertainties involved. Implementation of the station blackout rule will result in the average core damage frequency from station

blackout events being in approximately the range of frequencies estimated for station blackout for the five NUREG-1150 plants. Therefore, the station blackout rule meets both of the Commission's qualitative safety goals.

The Commission also stated the following regulatory objective relating to the frequency of core damage accidents at nuclear power plants.

> Severe core damage accidents can lead to more serious accidents with the potential for life-threatening offsite releases of radiation, for evacuation of members of the public, and for contamination of public property. Apart from their health and safety consequences, such accidents can erode public confidence in the safety of nuclear power and can lead to further instability and unpredictability for the industry. In order to avoid these adverse consequences, the Commission intends to continue to pursue a regulatory program that has as its objective providing reasonable assurance, giving appropriate consideration to the uncertainties involved, that a severe core damage accident will not occur at a U.S. nuclear power plant.

An estimate of the total probability of core damage for the nuclear industry is beyond the scope of this regulatory analysis, but some perspectives on station blackout are presented here. The mean core damage frequency from station blackout events before implementation of the station blackout rule is estimated to be 4.2×10^{-5} per reactor-year. Thus, the probability of core damage from station blackout is about 0.12 (i.e., about 1 chance in 8 that station blackout would result in severe core damage at one of 125 reactors over an assumed remaining 25-year life expectancy of these plants). Implementation of the station blackout rule would reduce the estimated mean core damage frequency to 1.6×10^{-5} per reactor-year, and therefore, the estimated probability of a severe core damage accident from station blackout would be 0.05 (i.e., about 1 chance in 20 of severe core damage). Therefore, implementing the resolution of USI A-44 provides reasonable assurance that a severe core damage accident from station blackout will not occur at a U.S. nuclear power plant.

The Commission also proposed the following guideline for further staff evaluation:

> Consistent with the traditional defense-in-depth approach and the accident mitigation philosophy requiring reliable performance of containment systems, the overall mean frequency of a large release of radioactive materials to the environment from a reactor accident should be less than 1 in 1,000,000 per year of reactor operation.

Given the current state of knowledge regarding containment performance and the large uncertainties with respect to the probability of containment failure following severe accident sequences, it is not possible to conclude that the safety performance guideline on the frequency of a large release would be met. This conclusion is based on the estimated mean core damage frequency for station blackout events of 1.6×10^{-5} per reactor-year coupled with the uncertainty band for the probability of early containment failure ranging from about 0.05 to 0.90 as reported in NUREG-1150. Since the potential for a high likelihood of containment failure cannot be eliminated, the overall mean frequency of a large release of radioactivity of 10^{-6} per reactor-year cannot be ensured.

Additional rationale for implementing the station blackout rule and the regulatory guide over other alternatives is discussed in the value-impact analysis (Section 4.1). This action represents the staff's position based on a comprehensive analysis of the station blackout issue. This position includes all the requirements and guidance to resolve the station blackout issue.

5.2 Station Blackout Reports

The studies and data on which this resolution is based are documented in NUREG-1032 and NUREG/CR-2989, -3226, -3992, and -4347. Summaries of these reports follow.

5.2.1 NUREG-1032, Evaluation of Station Blackout Accidents at Nuclear Power Plants, Technical Findings Related to Unresolved Safety Issue A-44

This report summarizes the results of technical studies performed in support of USI A-44 and identifies the dominant factors affecting the likelihood of station blackout accidents at nuclear power plants. These results are based on operating experience data; analysis of several plant-specific probabilistic safety studies; and reliability, accident sequence, and consequence analyses performed in support of this unresolved safety issue.

In summary the results show the following important characteristics of station blackout accidents.

(1) The likelihood of station blackout varies between plants with an estimated frequency ranging from approximately 10^{-5} to 10^{-3} per reactor-year. A "typical" estimated frequency is on the order of 10^{-4} per reactor-year.

(2) The capability of restoring offsite power in a timely manner can have a significant effect on accident consequences.

(3) Onsite ac power system redundancy and individual power supply reliability have the largest influence on station blackout accident frequency.

(4) The capability of the decay heat removal system to cope with long duration blackouts can be a dominant factor influencing the likelihood of core damage or core melt.

(5) The estimated frequency of station blackout events resulting in core damage or core melt can range from approximately 10^{-6} to greater than 10^{-4} per reactor-year. A "typical" core damage frequency estimate is 2 to 4 x 10^{-5} per reactor-year.

(6) The best information available indicates that containment failure by overpressure may follow a station-blackout-induced core melt with smaller, low design pressure containments most susceptible to early failure. Some large, high design pressure containments may not fail by overpressure, or the failure time could be on the order of a day or more.

Losses of offsite power could be characterized as those resulting from plant-centered faults, utility grid blackout, or severe weather-induced failures of offsite power sources. The industry average frequency of total losses of offsite power was determined to be about 1 in 10 site-years. The median restoration time was about 1/2 hour, and 90 percent of the losses were restored in

3 hours or less. The factors that were identified as affecting the frequency and duration of offsite power losses are

(1) design of preferred power distribution system, particularly the number and independence of offsite power circuits from the point where they enter the site up to the safety buses

(2) operations that can compromise redundancy or independence of multiple offsite power sources, including human error

(3) grid stability and security, and the ability to restore power to a nuclear plant site with a grid blackout

(4) the hazard from, and susceptibility to, severe weather conditions that can cause loss of offsite power for extended periods

A design and operating experience review, combined with a reliability analysis of the onsite, emergency ac power system, has shown that there are a variety of potentially important failure causes. The typical unavailability of a two-division emergency ac power system is about 10^{-3} per demand, and the typical individual emergency diesel generator failure rate is about 2×10^{-2} per demand. The factors that were identified as affecting the emergency ac power system reliability during a loss of offsite power are

(1) power supply configuration redundancy

(2) reliability of each power supply

(3) dependence of the emergency ac power system on support of auxiliary cooling systems and control systems and the reliability of those support systems

(4) vulnerability to common cause failures associated with design, operational, and environmental factors

The likelihood of a station blackout progressing to core damage or core melt is dependent on the reliability and capability of decay heat removal systems that are not dependent on ac power. If sufficient capability exists, additional time will be available to permit an adequate opportunity to restore ac power to the many systems normally used to cool the core and remove decay heat. The most important factors involving decay heat removal during a station blackout are

(1) the starting reliability of systems required to remove decay heat and maintain reactor coolant inventory

(2) the capacity and functionability of decay heat removal systems and auxiliary or support systems that must remain functional during a station blackout (e.g., dc power, condensate storage)

(3) for PWRs, and BWRs without reactor coolant makeup capability during a station blackout, the magnitude of reactor coolant pump seal leakage

(4) for BWRs that remove decay heat to the suppression pool, the ability to maintain suppression pool integrity and operate heat removal systems at high pool temperatures during recirculation

It was determined by reviewing design, operational, and location factors, that the expected core damage frequency from station blackout could be maintained around 10^{-5} per reactor-year or lower for almost all plants. The ability to cope with station blackout durations of 4 to 8 hours and emergency diesel generator reliabilities of 0.95 per demand or better would be necessary to reach this core damage frequency level.

5.2.2 NUREG/CR-3226, Station Blackout Accident Analyses

This report analyzes accident sequences following a postulated total loss of ac power to (1) determine the core damage frequencies from station blackout, (2) provide insights through sensitivity studies of important factors to consider for lowering the core melt frequency, and (3) provide perspectives on the risks from such an event. Probabilistic safety analyses were done on four generic "base" plant configurations. Fault trees of different systems and event trees of possible station blackout accident sequences were constructed for these plants. These event trees modeled three time periods including an initial time period for sequences resulting from unavailabilities on demand and longer time intervals in which other failures can occur such as depletion of dc power, degradation of reactor coolant pump seals, or depletion of condensate storage tank supply. Data from the offsite and onsite power studies (NUREG/CR-2989 and -3992) as well as from licensee event reports and PRAs were used to quantify the accident sequences. Lastly, containment failure modes and timing were reviewed to calculate the risk to the public from station blackout.

For the "base" cases, the total core damage frequencies from station blackout resulting from the dominant accident sequences were estimated to be in the range of 10^{-5} per reactor-year. Plants with features different from the base case designs have different core damage frequencies, so sensitivity analyses were conducted. For example, the reliability and recovery of ac power from both the offsite and emergency onsite power systems have a direct impact on core damage frequencies. Depending on the expected frequency of station blackout at a plant and other factors, the frequency of core damage associated with loss of all ac power ranged from about 2×10^{-6} to greater than 10^{-4} per reactor-year.

In summary, results of the accident sequence analyses indicate that the following plant factors are important when considering station blackout:

(1) the effectiveness of actions to restore offsite power once it is lost

(2) the degree of redundancy and reliability of the emergency onsite ac power system

(3) the reliability of decay heat removal systems following loss of ac power

(4) dc power reliability and battery capacity including the availability of instrumentation and control for decay heat removal without ac power

(5) common service water dependencies between the emergency ac power source and the decay heat removal systems

(6) the magnitude of reactor coolant pump seal leakage and the likelihood of a stuck-open relief valve during a station blackout

(7) containment size and design pressure

(8) operator training and available procedures

5.2.3 NUREG/CR-2989, Reliability of Emergency AC Power Systems at Nuclear Power Plants

The purpose of this study was to estimate the reliabilities of representative onsite ac power systems and to estimate the costs of fixes to improve the reliabilities of these systems. For this analysis, an initial design review of onsite ac power systems was done using Final Safety Analysis Reports (FSARs) for plants, plant schematics, and plant-specific procedures. The study included examining the following areas: switchyards, distribution systems, dc power systems, diesel generators, support systems, and procedures. Historical data on diesel generator operating experience for the 5-year period from 1976 through 1980 were collected from licensee event reports and responses to questionnaires sent to licensees.

Eighteen different configurations were identified, and representative plants were selected for a more detailed reliability analysis. This analysis involved constructing fault tree models for the onsite power systems and quantifying these fault trees with the data gathered on operating experience. The onsite system undependability (the probability that it will fail to start or fail to continue to run for the duration of an offsite power outage) was calculated for ac power outages up to 30 hours after a loss of offsite power. Results of a sensitivity study were used to identify potentially important contributors to unreliability, and costs of improvements were estimated.

Results showed that important contributors to onsite power undependability were independent diesel generator failure, common cause failure due to hardware failure or human error, unavailability because of scheduled maintenance, and cooling subsystem undependability. Reliability of onsite ac power systems varies from plant to plant. Depending on diesel generator configuration, the system unavailability ranged from 1.4×10^{-4} to 4.8×10^{-2} per demand. Significant variability exists so that any reliability improvements and the associated costs must be evaluated on a plant-specific basis.

5.2.4 NUREG/CR-4347, Emergency Diesel Generator Operating Experience, 1981-1983

This report is an update of operating experience of emergency diesel generators reported in NUREG/CR-2989. Estimates of diesel generator failure rate during surveillance testing and during actual demands (e.g., unplanned demands following losses of offsite power or safety injection actuation signals) are presented. The data indicate that overall diesel generator performance has improved since 1976 with an overall median failure rate estimated to be 0.019 failure per demand. However, for the 1981 to 1983 period, the diesel generator failure rate during actual demands was 0.025 failure per demand--a rate higher than that for all demands (i.e., including surveillance tests). Data from NUREG/CR-2989 and -4347, along with results of an industry survey conducted by the Electric Power Research Institute (NSAC/108), were used in the staff's evaluation of risk from station blackout events (NUREG-1032).

5.2.5 NUREG/CR-3992, Collection and Evaluation of Complete and Partial Losses of Offsite Power at Nuclear Power Plants

This report describes and categorizes events involving complete or significant partial losses of offsite power that have occurred at nuclear power plants through 1983. The purposes of this study were to provide an accurate data base to estimate frequencies and durations of losses of offsite power and to understand how offsite power design features may affect these losses as well as the ability to restore offsite power. A parallel study documenting loss of offsite power experience through 1985 was published by the Nuclear Safety Analysis Center of the Electric Power Research Institute (NSAC/103). Data from both NUREG/CR-3992 and NSAC/103 were used in the loss of offsite power analysis in NUREG-1032.

Based on industry-wide data for the years 1959 through 1983, the frequency of loss of offsite power is about once every 10 site-years. A total of 46 complete loss-of-offsite-power events were documented, ranging in duration from a few minutes up to a maximum of almost 9 hours. In approximately half of these events, offsite power was restored in 1/2 hour or less. Information for this study was collected from licensee event reports, responses to an NRC questionnaire, and various reports prepared by the utilities. Most of the event descriptions in the licensee event reports and other documentation within the NRC files did not contain sufficiently detailed information for the purposes discussed above. For example, in one case a licensee reported offsite power restoration time to be 6 hours, but actually one offsite power source was restored in 8 minutes, and all offsite power was restored in 6 hours. Because restoration of one source of offsite power terminates a loss of offsite power, the documented description was not accurate enough. In some other cases, offsite power was available to be reconnected, but the plant operators did not reconnect it for some time after it was available. The time power was reconnected was usually reported; however, the data that were actually needed were the times that power was available for reconnection. Because of the need for more accurate data, additional information was obtained by contacting utility engineers for better descriptions of the causes, sequences of events, and the times and methods of restoring offsite power.

Once these data were collected, the offsite power failures were identified as plant-centered or grid failures. In addition, the causes of the failures were attributed to weather, human error, design error, or hardware failure. The plant-centered failures were usually of shorter duration than the grid failures caused by severe weather. For this reason, the weather-related events were reviewed in detail.

Offsite power design features were tabulated for most of the operating nuclear power plants to determine which ones significantly affect offsite power system reliability. The frequency and duration of losses of offsite power caused by severe weather are affected by the number of transmission lines and rights-of-way and the availability of alternate power sources (such as hydro, gas turbines, or fossil units near the nuclear plant). Design features that may be important for plant-centered losses of offsite power are the number of offsite power sources, the electrical independence of those sources, and the relay scheme for transferring power between offsite sources.

6 IMPLEMENTATION

6.1 Schedule for Implementing the Final Station Blackout Rule

The steps and schedule listed in Table 13 summarize the implementation schedule in the station blackout rule (10 CFR 50.63(c) and (d)). Within 9 months after promulgation of the rule, licensees will submit to NRC (1) the duration for which the plant should be able to cope with a station blackout, (2) a justification for the duration, (3) a description of the procedures to cope with a station blackout for that duration, and (4) a list of equipment modifications necessary, if any, to meet the specified station blackout duration. The staff will review the licensees' submittals and inform them of its conclusions regarding the requirements of 10 CFR 50.63. Within 30 days of the staff's notification, the licensees will submit a schedule for implementing any necessary equipment modifications to comply with the rule.

The factors that must be considered to determine the minimum acceptable station blackout duration, as specified in 10 CFR 50.63(a), are relatively straightforward. In fact, licensees have reviewed their plants against these factors as part of an industry initiative supported by NUMARC. Thus, this acceptable duration can be determined in approximately 1 or 2 months. Licensees will be required to perform plant-specific analyses to determine if the plant, as designed, can cope with a station blackout for the acceptable duration, and to determine what modifications, if any, are needed to meet the acceptable duration. These analyses could require 6 to 9 months to perform. Thus, it seems reasonable to require that the information be submitted to the NRC within 9 months after the date the final rule is issued.

The implementation of procedural changes to cope with a station blackout and diesel generator reliability improvements, if necessary, will be accomplished early in the schedule. Hardware backfits, if necessary, should be implemented as soon as practical, based on scheduled plant shutdown, but no later than 2 years after the staff reviews a licensee's station blackout duration submittal. A final schedule for implementation of design and associated procedural modifications will be mutually agreed upon by the licensee and the NRC staff.

Other schedules were considered; however, the staff believes the implementation schedule in Table 13 is achievable without unnecessary financial burden on licensees for plant shutdown. The schedule allows reasonable time for the implementation of necessary hardware items to achieve a reduction in the risk of severe accidents associated with station blackout, yet achieves significant benefits early on by requiring an assessment of a plant's station blackout capability and procedures and training to cope with such an event. Shorter or less flexible schedules would be unnecessarily burdensome; longer schedules would delay necessary plant improvements.

6.2 Relationship to Other Existing or Proposed Requirements

Several NRC programs are related to USI A-44; these are discussed in Section 4.2. These programs are compatible with the resolution of USI A-44.

Table 13 Implementation schedule for final station blackout rule

Activity	Time after Commission decision to issue final rule (months)
Issuance of final rule	0
Licensees' submittal of acceptable station blackout durations to NRC, including description of procedures and list of modifications	9
Staff notification of licensees of conclusions regarding requirements	15-33*
Licensee's submittal of schedule for implementing hardware modifications	16-34
Completion of licensees' hardware modifications	**

*High priority plants - 15 months; last remaining plant - 33 months.

**Schedule to be agreed upon with NRC, but within 2 years of NRC review of submittal, unless justification is submitted by the licensee for a later date and the staff agrees.

7 REFERENCES

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APPENDIX A

BACKFIT ANALYSIS

APPENDIX A

BACKFIT ANALYSIS*

Analysis and Determination That the Rulemaking To Amend 10 CFR 50 Concerning Station Blackout Complies With the Backfit Rule 10 CFR 50.109

The Commission's existing regulations establish requirements for the design and testing of onsite and offsite electrical power systems (10 CFR 50, Appendix A, General Design Criteria 17 and 18). However, as operating experience has accumulated, the concern has arisen regarding the reliability of both the offsite and onsite emergency ac power systems. These systems provide power for various safety systems including reactor core decay heat removal and containment heat removal which are essential for preserving the integrity of the reactor core and the containment building, respectively. In numerous instances emergency diesel generators have failed to start and run during tests conducted at operating plants. In addition, a number of operating plants have experienced a total loss of offsite electric power, and more such occurrences are expected. Existing regulations do not require explicitly that nuclear power plants be designed to withstand the loss of all ac power for any specified period.

This issue has been studied by the staff as part of Unresolved Safety Issue (USI) A-44, "Station Blackout." Both deterministic and probabilistic analyses were performed to determine the timing and consequences of various accident sequences and to identify the dominant factors affecting the likelihood of core melt accidents from station blackout. Although operational experience shows that the risk to public health and safety is not undue, these studies, which have evaluated plant design features and site-dependent factors in detail, show that station blackout can be a significant contributor to the overall plant risk. Consequently, the Commission is amending its regulations to require that plants be capable of withstanding a total loss of ac power for a specified duration and to maintain reactor core cooling during that period.

The estimated benefit from implementing the station blackout rule is a reduction in the frequency of core damage per reactor-year due to station blackout and the associated risk of offsite radioactive releases. The risk reduction for 100 operating reactors is estimated to be 145,000 person-rems and supports the Commission's conclusion that 10 CFR 50.63 provides a substantial improvement in the level protection of public health and safety.

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*The backfit analysis is included as an appendix to this report. It is intended to be a standalone document that minimizes the need to refer to additional documents by including sufficient detail to assess each consideration in the backfit rule (10 CFR 50.109). Therefore, the backfit analysis repeats much of what is already included in the main body of the report.

The cost for licensees to comply with the rule would vary depending on the existing capability of each plant to cope with a station blackout, as well as the specified station blackout duration for that plant. The costs would be primarily for licensees (1) to assess the plant's capability to cope with a station blackout, (2) to develop procedures, (3) to improve diesel generator reliability if the reliability falls below certain levels, and (4) to retrofit plants with additional components or systems, as necessary, to meet the requirements.

The estimated total cost for 100 operating reactors to comply with the resolution of USI A-44 is about $60 million. The average cost per reactor would be around $600,000, ranging from $350,000 if only a station blackout assessment and procedures and training were necessary, to a maximum of about $4 million if substantial modifications were needed, including requalification of a diesel generator.

The overall value-impact ratio, not including accident avoidance costs, is about 2,400 person-rems averted per million dollars. If the net cost, which includes the cost savings from accident avoidance (i.e., cleanup and repair of onsite damages and replacement power following an accident), were used, the overall value-impact ratio would improve significantly to about 6,100 person-rems averted per million dollars. These values, which exceed the $1,000/person-rem guidance provided by the Commission, support proceeding with the implementation of 10 CFR 50.63.

The preceding quantitative value-impact analysis was one of the factors considered in evaluating the rule, but other factors also played a part in the decision-making process. Probabilistic risk assessment (PRA) studies performed for this USI, as well as some plant-specific PRAs, have shown that station blackout can be a significant contributor to core melt frequency, and, with consideration of containment failure, station blackout events can represent an important contributor to reactor risk. In general, active systems required for containment heat removal are unavailable during station blackout. Therefore, the offsite risk is higher from a core melt resulting from a station blackout than it is from many other accident scenarios.

Although there are licensing requirements and guidance directed at providing reliable offsite and onsite ac power, experience has shown that there are practical limitations in ensuring the reliability of offsite and onsite emergency ac power systems. Potential vulnerabilities to common cause failures associated with design, operational, and environmental factors can affect ac power system reliability. For example, if potential common cause failures of emergency diesel generators exist (e.g., in service-water or dc power support systems), then the estimated core damage frequency from station blackout events can increase significantly. Also, even though recent data indicate that the average emergency diesel generator reliability has improved slightly since 1976, these data also show that diesel generator failure rates during unplanned demand (e.g., following a loss of offsite power) were higher than those during surveillance tests.

The estimated frequency of core damage from station blackout events is directly proportional to the frequency of the initiating event. Estimates of station blackout frequencies for this USI were based on actual operational experience with credit given for trends showing a reduction in the frequency of losses of offsite power resulting from plant-centered events. This is assumed to be a

realistic indicator of future performance. An argument can be made that the future performance will be better than the past. For example, when problems with the offsite power grid arise, they are fixed and, therefore, grid reliability should improve. On the other hand, grid power failures may become more frequent because fewer plants are being built, and more power is being transmitted among regions, thus placing greater stress on transmission lines.

A number of foreign countries, including France, Britain, Sweden, Germany, and Belgium, have taken steps to reduce the risk from station blackout events. These steps include adding design features to enhance the capability of the plant to cope with a station blackout for a substantial period of time and/or adding redundant and diverse emergency ac power sources.

The factors discussed above support the determination that additional defense in-depth provided by the ability of a plant to cope with station blackout for a specific duration would provide a substantial increase in the overall protection of the public health and safety, and the direct and indirect costs of implementation are justified in view of this increased protection. The Commission has considered how this backfit should be prioritized and scheduled in light of other regulatory activities ongoing at operating nuclear power plants. Station blackout warrants a high priority ranking based on both its status as an "unresolved safety issue" and the results and conclusions reached in resolving this issue. As noted in the implementation section of the rule (10 CFR 50.63(c)(4)), the schedule for equipment modification (if needed to meet the requirements of the rule) shall be mutually agreed upon by the licensee and NRC. Modifications that cannot be scheduled for completion within 2 years after NRC accepts the licensee's specified station blackout duration must be justified by the licensee.

Analysis of 10 CFR 50.109(c) Factors

(1) Statement of the specific objectives that the backfit is designed to achieve

The NRC staff has completed a review and evaluation of information developed since 1980 on USI A-44, "Station Blackout." As a result of these efforts, the NRC is amending 10 CFR 50 by adding a new § 50.63, "Station Blackout."

The objective of the station blackout rule is to reduce the risk of severe accidents associated with station blackout by making station blackout a relatively small contributor to total core damage frequency. Specifically, the rule requires all light-water-cooled nuclear power plants to be able to cope with a station blackout for a specified duration and to have procedures and training for such an event. A regulatory guide (Regulatory Guide 1.155), to be issued along with the rule, provides an acceptable method to determine the station blackout duration for each plant. The duration is to be determined for each plant based on a comparison of the individual plant design with factors that have been identified as the main contributors to risk of core melt resulting from station blackout. These factors are (1) the redundancy of onsite emergency ac power sources, (2) the reliability of onsite emergency ac power sources, (3) the frequency of loss of offsite power, and (4) the probable time needed to restore offsite power.

(2) General description of the activity required by the licensee or applicant in order to complete the backfit

In order to comply with the resolution of USI A-44, licensees will be required to

- Maintain the reliability of onsite emergency ac power sources at or above specified acceptable reliability levels.

- Develop procedures and training to restore ac power using nearby power sources if the emergency ac power system and the normal offsite power sources are unavailable.

- Determine the duration that the plant should be able to withstand a station blackout based on the factors specified in 10 CFR 50.63, "Station Blackout," and Regulatory Guide 1.155, "Station Blackout."

- If available, an alternate ac power source, which meets specific criteria for independence and capacity, can be used to cope with a station blackout.

- Evaluate the plant's actual capability to withstand and recover from a station blackout. This evaluation will include

  - verifying the adequacy of station battery power, condensate storage tank capacity, and plant/instrument air for the station blackout duration

  - verifying adequate reactor coolant pump seal integrity for the station blackout duration so that seal leakage due to lack of seal cooling would not result in a sufficient primary system coolant inventory reduction to lose the ability to cool the core

  - verifying operability of equipment needed to operate during a station blackout for environmental conditions associated with total loss of ac power (i.e., loss of heating, ventilation, and air conditioning)

Depending on the plant's existing capability to cope with a station blackout, licensees may or may not need to backfit hardware modifications (e.g., adding battery capacity) to comply with the rule. (See item 8 of this analysis for additional discussion.) Licensees will be required to have procedures and training to cope with and recover from a station blackout.

(3) Potential change in the risk to the public from the accidental offsite release of radioactive material

Implementation of the station blackout rule will result in an estimated total risk reduction to the public ranging from 65,000 to 215,000 person-rems with a best estimate of about 145,000 person-rems.

(4) Potential impact on radiological exposure of facility employees

For 100 operating reactors, the estimated total reduction in occupational exposure resulting from reduced core damage frequencies and associated post-accident cleanup and repair activities is 1,500 person-rems. No increase in occupational exposure is expected from operation and maintenance activities associated with the rule. Equipment additions and modifications contemplated do not require work in and around the reactor coolant system and therefore are not expected to result in significant radiation exposure.

(5) Installation and continuing costs associated with the backfit, including the cost of facility downtime or the cost of construction delay

For 100 operating reactors, the total estimated cost associated with the station blackout rule ranges from $42 to $94 million with a best estimate of $60 million. This estimate breaks down as follows:

Activity	Estimated number of reactors	Estimated total cost (million dollars) Best est.	High est.	Low est.
Assess plant's capability to cope with station blackout	100	25	40	20
Develop procedures and training	100	10	15	5
Improve diesel generator reliability	10	2.5	4	1.5
Requalify diesel generator	2	5.5	11	2.5
Install hardware to increase plant's capability to cope with station blackout	27	17	24	13
Totals		60	94	42

(6) The potential safety impact of changes in plant or operational complexity, including the relationship to proposed and existing regulatory requirements

The rule requiring plants to be able to cope with a station blackout should not add to plant or operational complexity. The station blackout rule is closely related to several NRC generic programs and proposed and existing regulatory requirements as the following discussion indicates.

- Generic Issue B-56, Diesel Generator Reliability

The resolution of USI A-44 includes a regulatory guide on station blackout that specifies the following guidance on diesel generator reliability (Regulatory Guide 1.155, Sections C.1.1 and 2):

> The reliable operation of the onsite emergency ac power sources should be ensured by a reliability program designed to monitor and maintain the reliability of each power source over time at a specified acceptable level and to improve the reliability if

that level is not achieved. The reliability program should include surveillance testing, target values for maximum failure rate, and a maintenance program. Surveillance testing should monitor performance so that if the actual failure rate exceeds the target level, corrective actions can be taken.

The maximum emergency diesel generator failure rate for each diesel generator should be maintained at 0.05 failure per demand. However, for plants having an emergency ac power system [configuration requiring two out-of-three diesel generators or having a total of two diesel generators shared between two units at a site], the emergency diesel generator failure rate for each diesel generator should be maintained at 0.025 failure per demand or less.

The resolution of B-56 will provide specific guidance for use by the staff or industry to review the adequacy of diesel generator reliability programs consistent with the resolution of USI A-44.

- Generic Issue B-23, Reactor Coolant Pump Seal Failures

Reactor coolant pump (RCP) seal integrity is necessary for maintaining primary system inventory during station blackout conditions. The estimates of core damage frequency for station blackout events for USI A-44 assumed that RCP seals would leak at a rate of 20 gallons per minute. Results of analyses performed for Generic Issue B-23 will provide the information necessary to determine RCP seal behavior during a station blackout. Should this analysis conclude that there is a high probability that the RCP seals would not leak excessively during a station blackout, then no modifications would be required. If there is a significant probability that RCP seals can leak at rates substantially higher than 20 gallons per minute, then modifications such as an ac-independent RCP seal cooling system may be necessary to resolve Generic Issue B-23. Any proposed backfit resulting from the resolution of Generic Issue B-23 would need to comply with the backfit rule.

- USI A-45, Shutdown Decay Heat Removal Requirements

The overall objective of USI A-45 is to evaluate the adequacy of current licensing design requirements to ensure that the nuclear power plants do not pose an unacceptable risk as a result of failure to remove shutdown decay heat. The study includes an assessment of alternative means of shutdown decay heat removal and of diverse "dedicated" systems for this purpose. Results will include proposed recommendations regarding the desirability of, and possible design requirements for, improvements in existing systems or an alternative dedicated decay heat removal method.

The USI A-44 concern for maintaining adequate core cooling under station blackout conditions can be considered a subset of the overall USI A-45 issue. However, there are significant differences in scope between these two issues. USI A-44 deals with the probability of loss of ac power, the capability to remove decay heat using systems that do not require ac power, and the ability to restore ac power in a timely manner. USI A-45

deals with the overall reliability of the decay heat removal function in terms of response to transients, small break loss-of-coolant accidents, and special emergencies such as fires, floods, seismic events, and sabotage.

Although the recommendations that might result from the resolution of USI A-45 are not yet final, some could affect the station blackout capability, while others would not. Recommendations that involve a new or improved decay heat removal system that is ac power dependent but that does not include its own dedicated ac power supply would have no effect on USI A-44. Recommendations that involve an additional ac-independent decay heat removal system would have a very modest effect on USI A-44. Recommendations that involve an additional decay heat removal system with its own ac power supply would have a significant effect on USI A-44. Such a new additional system would receive the appropriate credit within the USI A-44 resolution by either changing the emergency ac power configuration group or providing the ability to cope with a station blackout for an extended period of time. Well before plant modifications, if any, will be implemented to comply with the station blackout rule, it is anticipated that the proposed technical resolution of USI A-45 will be published for public comment. Those plants needing hardware modifications for station blackout could be re-evaluated before any actual modifications are made so that any contemplated design changes resulting from the resolution of USI A-45 can be considered at the same time.

- Generic Issue A-30, Adequacy of Safety-Related DC Power Supply

The analysis performed for USI A-44 assumed that a high level of dc power system reliability would be maintained so that (1) dc power system failures would not be a significant contributor to losses of all ac power and (2) should a station blackout occur, the probability of immediate dc power system failure would be low. Whereas Generic Issue A-30 focuses on enhancing battery reliability, the resolution of USI A-44 is aimed at ensuring adequate station battery capacity in the event of a station blackout of a specified duration. Therefore, these two issues are consistent and compatible.

- Fire Protection Program

10 CFR 50.48 states that each operating nuclear power plant shall have a fire protection plan that satisfies GDC 3. The fire protection features required to satisfy GDC 3 are specified in Appendix R to 10 CFR 50. They include certain provisions regarding alternative and dedicated shutdown capability. To meet these provisions, some licensees have added, or plan to add, improved capability to restore power from offsite sources or onsite diesels for the shutdown system. A few plants have installed a safe shutdown facility for fire protection that includes a charging pump powered by its own independent ac power source. In the event of a station blackout, this system can provide makeup capability to the primary coolant system as well as reactor coolant pump seal cooling. This could be a significant benefit in terms of enhancing the ability of a plant to cope with a station blackout. Plants that have added equipment to achieve alternate safe shutdown in order to meet Appendix R requirements could take credit for that equipment, if available, for coping with a station blackout event.

(7) <u>The estimated resource burden on the NRC associated with the backfit and the availability of such resources</u>

The estimated total cost for NRC review of industry submittals required by the station blackout rule is $1.5 million based on submittals for 100 reactors and an estimated average of 175 person-hours per reactor.

(8) <u>The potential impact of differences in facility type, design, or age on the relevancy and practicality of the backfit</u>

The station blackout rule applies to all pressurized water reactors and boiling water reactors. However, in determining an acceptable station blackout coping capability for each plant, differences in plant characteristics relating to ac power reliability (e.g., number of emergency diesel generators, the reliability of the offsite and onsite emergency ac power systems) could result in different acceptable coping capabilities. For example, plants with an already low risk from station blackout because of multiple, highly reliable ac power sources are required to withstand a station blackout for a relatively short period of time, and few, if any, hardware backfits would be required as a result of the rule. Plants with currently higher risk from station blackout are required to withstand somewhat longer duration blackouts, and, depending on their existing capability, may need some modifications to achieve the longer station blackout capability.

(9) <u>Whether the backfit is interim or final and, if interim, the justification for imposing the backfit on an interim basis</u>

The station blackout rule is the final resolution of USI A-44; it is not an interim measure.

APPENDIX B

WORKSHEETS FOR COST ESTIMATES

APPENDIX B

WORKSHEETS FOR COST ESTIMATES

Section 4.1 of this report provides a summary of the estimated costs to industry and NRC associated with the resolution of USI A-44. This appendix provides supplementary information to support these cost estimates. The estimates in the following worksheets are based on information from the following references: EG&G (1983), Science and Engineering Associates (1986), NRC (1986), and NUREG/CR-3568, -3840, -4568, -4627, and -4942. The utility personnel cost used in these estimates is $100,000 per person-year, including overhead and general and administrative expenses.

<u>References</u>

EG&G, "Cost Analysis for Enhancement of DC Systems Reliability and Adequacy of Safety-Related DC Power Systems," EG&G Report RE&ET-6151, January 1983.

Science and Engineering Associates, Inc., "Response to Industry Comments on Station Blackout Cost Estimates (NUREG/CR-3840)," letter report to NRC, November 12, 1986.

U.S. Nuclear Regulatory Commission, "Regulatory Analysis Guidelines," NRR Office Letter No. 16, Revision 3, May 13, 1986.

---, NUREG/CR-3568, "A Handbook for Value-Impact Assessment," December 1983.

---, NUREG/CR-3840, "Cost Analysis for Potential Modifications To Enhance the Ability of a Nuclear Power Plant To Endure Station Blackout," July 1984.

---, NUREG/CR-4568, "A Handbook for Quick Cost Estimates," April 1986.

---, NUREG/CR-4627, "Generic Cost Estimates," June 1986.

---, NUREG/CR-4942, "Equipment Operability During Station Blackout Events," SAND87-0750, Sandia National Laboratory, to be published.

Worksheet 1 Estimated cost to assess plant's capability to cope with station blackout (SBO)

Activity	Estimated resources	
	Person-months	Dollars
Determine system capabilities (e.g., batteries, instrument air, condensate storage tank, reactor coolant pump seals)	12	-
Evaluate equipment operability		
Determine equipment/components necessary during SBO	2	-
Determine heat loads for rooms/compartments	6	-
Calculate environmental conditions during SBO	4	-
Compare equipment design/operational capability with predicted environmental conditions	2	-
Quality assurance	4	-
Total	30	$250,000

Worksheet 2 Estimated cost to develop procedures and training for station blackout

Activity	Estimated resources	
	Person-months	Dollars
Develop procedures (includes writing, review, and approval)	3	$25,000
Training		
Initial training	3	$25,000
Annual update training	0.5/yr	$5,000/yr

Total training costs are calculated by the following equation which sums the initial training costs and the present value of the annual training costs over the remaining plant lifetime.

$$C_{TL} = C_{IT} + C_{AT} \left[\frac{(1 + D)^{L} - 1}{D\,(1 + D^{L})} \right] = \$70,000$$

where C_{TL} = total training costs

C_{IT} = initial training costs

C_{AT} = annual training costs

D = discount rate (10%)

L = remaining plant lifetime (25 years)

Therefore, adding the cost to develop procedures, the total cost for procedures and training is estimated to be $100,000.

Worksheet 3 Estimated cost to improve diesel generator reliability

Activity	Estimated cost
Reliability investigation	\$100,000
Equipment modifications	\$150,000
	\$250,000

Worksheet 4 Estimated cost to requalify a diesel generator

Assuming that a plant would shut down for 5 days to requalify a diesel generator, the replacement energy cost (C_R) is the dominant cost associated with this activity. C_R can be calculated using the following equation:

$$C_R = E \times P \times R$$

where E = net electrical output (kWe)
P = shutdown period (hours)
R = replacement energy cost (\$/kWh)

The table below presents the data used to calculate the best, high, and low estimates to requalify a diesel generator.

	Value		
Paramater	Best	High	Low
Net plant electrical outpost (kWe)	900,000	1,150,000	500,000
Shutdown period (hours)	120	120	120
Replacement energy cost (\$/kWe)*	0.026	0.040	0.020
Total cost (million dollars)	2.8	5.5	1.2

*Costs from NUREG/CR-4568.

REGULATORY GUIDE 1.155

STATION BLACKOUT
(TASK SI 501-4)

A. INTRODUCTION

Criterion 17, "Electric Power Systems," of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," includes a requirement that an onsite electric power system and an offsite electric power system be provided to permit functioning of structures, systems, and components important to safety.

The Commission has amended its regulations in 10 CFR Part 50. Paragraph (a), "Requirements," of § 50.63, "Loss of All Alternating Current Power," requires that each light-water-cooled nuclear power plant be able to withstand and recover from a station blackout (i.e., loss of the offsite electric power system concurrent with reactor trip and unavailability of the onsite emergency ac electric power system) of a specified duration. Section 50.63 requires that, for the station blackout duration, the plant be capable of maintaining core cooling and appropriate containment integrity. It also identifies the factors that must be considered in specifying the station blackout duration.

Criterion 1, "Quality Standards and Records," of Appendix A to 10 CFR Part 50 includes a requirement for a quality assurance program to provide adequate assurance that structures, systems, and components important to safety will perform their safety functions.

Criterion 18, "Inspection and Testing of Electric Power Systems," of Appendix A to 10 CFR Part 50 includes a requirement for appropriate periodic testing and inspection of electric power systems important to safety.

The Advisory Committee on Reactor Safeguards has been consulted concerning this guide and has concurred in the regulatory position.

This guide describes a method acceptable to the NRC staff for complying with the Commission regulation that requires nuclear power plants to be capable of coping with a station blackout for a specified duration. This guide applies to all light-water-cooled nuclear power plants.

Any information collection activities related to this regulatory guide are contained as requirements in the revision of 10 CFR Part 50 that provides the regulatory basis for this guide. The information collection requirements in Part 50 have been cleared under the Office of Management and Budget Clearance No. 3150-0011.

B. DISCUSSION

The term "station blackout" refers to the complete loss of alternating current electric power to the essential and nonessential switchgear buses in a nuclear power plant. Station blackout therefore involves the loss of offsite power concurrent with turbine trip and failure of the onsite emergency ac power system, but not the loss of available ac power to buses fed by station batteries through inverters or the loss of power from "alternate ac sources." Station blackout and alternate ac source are defined in § 50.2. Because many safety systems required for reactor core decay heat removal and containment heat removal are dependent on ac power, the consequences of station blackout could be severe. In the event of a station blackout, the capability to cool the reactor core would be dependent on the availability of systems that do not require ac power from the essential and nonessential switchgear buses and on the ability to restore ac power in a timely manner.

The concern about station blackout arose because of the accumulated experience regarding the reliability of ac power supplies. Many operating plants have experienced a total loss of offsite electric power, and more occurrences are expected in the future. In almost every one of these loss-of-offsite-power events, the onsite emergency ac power supplies have been available immediately to supply the power needed by vital safety equipment. However, in some instances, one of the redundant emergency ac power supplies has been unavailable. In a few cases there has been a complete loss of ac power, but during these events, ac power was restored in a short time without

any serious consequences. In addition, there have been numerous instances when emergency diesel generators have failed to start and run in response to tests conducted at operating plants.

The results of the Reactor Safety Study (Ref. 1) showed that, for one of the two plants evaluated, a station blackout event could be an important contributor to the total risk from nuclear power plant accidents. Although this total risk was found to be small, the relative importance of station blackout events was established. This finding and the accumulated diesel generator failure experience increased the concern about station blackout.

In a Commission proceeding addressing station blackout, it was determined that the issue should be analyzed to identify preventive or mitigative measures that can or should be taken. (See Florida Power & Light Company (St Lucie Nuclear Power Plant, Unit No. 2) ALAB-603, 12 NRC 30 (1980); modified CLI-81-12, 13 NRC 838 (1981).)

The issue of station blackout involves the likelihood and duration of the loss of offsite power, the redundancy and reliability of onsite emergency ac power systems, and the potential for severe accident sequences after a loss of all ac power. References 2 through 7 provide detailed analyses of these topics. Based on risk studies performed to date, the results indicate that estimated core melt frequencies from station blackout vary considerably for different plants and could be a significant risk contributor for some plants. In order to reduce this risk, action should be taken to resolve the safety concern stemming from station blackout. The issue is of concern for both PWRs and BWRs.

This guide primarily addresses the following three areas: (1) maintaining highly reliable ac electric power systems, (2) developing procedures and training to restore offsite and onsite emergency ac power should either one or both become unavailable, and (3) ensuring that plants can cope with a station blackout for some period of time based on the probability of occurrence of a station blackout at a site as well as the capability for restoring ac power in a timely fashion for that site.

One factor that affects ac power system reliability is the vulnerability to common cause failures associated with design, operational, and environmental factors. Existing standards and regulatory guides include specific design criteria and guidance on the independence of preferred (offsite) power circuits (see General Design Criterion 17, "Electric Power Systems," and Section 5.1.3 of Reference 8) and the independence of and limiting interactions between diesel generator units at a nuclear station (see General Design Criterion 17, Regulatory Guide 1.6, "Independence Between Redundant Standby (Onsite) Power Sources and Between Their Distribution Systems," Regulatory Guide 1.75, "Physical Independence of Electric Systems," and Reference 9). In developing the recommendations in this guide, the staff has assumed that, by adhering to such standards, licensees have minimized, to the extent practical, single-point vulnerabilities in design and operation that could result in a loss of all offsite power or all onsite emergency ac power.

Onsite emergency ac power system unavailability can be affected by outages resulting from testing and maintenance. Typically, this unavailability is about 0.007 (Reference 5), which is small compared to the minimum emergency diesel generator reliability specified in Regulatory Position 1.1 of this regulatory guide (i.e., 0.95 or 0.975 reliability per demand). However, in some cases outages due to maintenance can be a significant contributor to emergency diesel generator unavailability. This contribution can be kept low by having high-quality test and maintenance procedures and by scheduling regular diesel generator maintenance at times when the reactor is shut down. Also, limiting conditions for operation in the technical specifications are designed to limit the diesel generator unavailability when the plant is operating. As long as the unavailability due to testing and maintenance is not excessive, the maximum emergency diesel generator failure rates for each diesel generator specified in Regulatory Position 1.1 would result in acceptable overall reliability for the emergency ac power system.

Based on § 50.63, all licensees and applicants are required to assess the capability of their plants to maintain adequate core cooling and appropriate containment integrity during a station blackout and to have procedures to cope with such an event. This guide presents a method acceptable to the NRC staff for determining the specified duration for which a plant should be able to

withstand a station blackout in accordance with these requirements. The application of this method results in selecting a minimum acceptable station blackout duration capability from 2 to 16 hours, depending on a comparison of the plant's characteristics with those factors that have been identified as significantly affecting the risk from station blackout. These factors include redundancy of the onsite emergency ac power system (i.e., the number of diesel generators available for decay heat removal minus the number needed for decay heat removal), the reliability of onsite emergency ac power sources (e.g., diesel generators), the frequency of loss of offsite power, and the probable time to restore offsite power.

Licensees may propose durations different from those specified in this guide. The basis for alternative durations would be predicated on plant-specific factors relating to the reliability of ac power systems such as those discussed in Reference 2.

The information submitted to comply with § 50.63 is also required to be incorporated in an update to the FSAR in accordance with paragraph 50.71(e)(4). It is expected that the applicant or licensee will have available for review, as required, the analyses and related information supporting the submittal.

Concurrent with the development of this regulatory guide, and consistent with discussions with the NRC staff, the Nuclear Management and Resource Council (NUMARC) has developed guidelines and procedures for assessing station blackout coping capability and duration for light water reactors (NUMARC-8700, Ref. 10). The NRC staff has reviewed these guidelines and analysis methods and concludes that NUMARC-8700 provides guidance for conformance to § 50.63 that is in large part identical to the guidance provided in this regulatory guide. Table 1 of this regulatory guide provides a section-by-section comparison between Regulatory Guide 1.155 and NUMARC-8700. The use of NUMARC-8700 is further discussed in Section C, Regulatory Position, of this guide.

C. REGULATORY POSITION

This regulatory guide describes a means acceptable to the NRC staff for meeting the requirements of § 50.63 of 10 CFR Part 50. NUMARC-8700 (Ref. 10)

also provides guidance acceptable to the staff for meeting these requirements. Table 1 provides a cross-reference to NUMARC-8700 and notes where the regulatory guide takes precedence.

1. ONSITE EMERGENCY AC POWER SOURCES (EMERGENCY DIESEL GENERATORS)

1.1 Emergency Diesel Generator Target Reliability Levels

The minimum emergency diesel generator (EDG) reliability should be targeted at 0.95 per demand for each EDG for plants in emergency ac (EAC) Groups A, B, and C and at 0.975 per demand for each EDG for plants in EAC Group D (see Table 2). These reliability levels will be considered minimum target reliabilities and each plant should have an EDG reliability program containing the principal elements, or their equivalent, outlined in Regulatory Position 1.2. Plants that select a target EDG reliability of 0.975 will use the higher level as the target in their EDG reliability programs.

The EDG reliability for determining the coping duration for a station blackout will be determined as follows:

1. Calculate the most recent EDG reliability for each EDG based on the last 20, 50, and 100 demands using definitions and methodology in Section 2 of NSAC-108, "Reliability of Emergency Diesel Generators at U.S. Nuclear Power Plants" (Ref. 11), or equivalent.*

2. Calculate the nuclear unit "average" EDG reliability for the last 20, 50, and 100 demands by averaging the results from step 1 above.

*This EDG reliability is not suitable for probabilistic risk analyses for design basis accidents because of the differing EDG start-reliability requirement that would be applicable for such probabilistic risk analyses.

3. Compare the calculated "average" nuclear unit EDG reliability from step 2 above against the following criteria:

   Last 20 demands > 0.90 reliability
   Last 50 demands > 0.94 reliability
   Last 100 demands > 0.95 reliability

4. If the EAC group is A, B, or C AND any of the three evaluation criteria in step 3 are met, the nuclear unit may select an EDG reliability target of either 0.95 or 0.975 for determining the applicable coping duration from Table 2.

   If the EAC group is D and any of the three evaluation criteria in step 3 are met, the allowed EDG reliability target is 0.975.

5. If the EAC group is A, B, or C and NONE of the selection criteria in step 3 are met, an EDG reliability level of 0.95 must be used for determining the applicable coping duration from Table 2. Additionally, if the "averaged" nuclear unit EDG reliability is less than 0.90 based on the last 20 demands, the acceptability of a coping duration based on an EDG reliability of 0.95 from Table 2 must be further justified.

   If the EAC group is D and NONE of the three evaluation criteria in step 3 are met, the required coping duration (derived by using Table 2) should be increased to the next highest coping level (i.e., 4 hours to 8 hours, 8 hours to 16 hours).

1.2 Reliability Program

The reliable operation of onsite emergency ac power sources should be ensured by a reliability program designed to maintain and monitor the reliability level of each power source over time for assurance that the selected reliability levels are being achieved. An EDG reliability program would typically be composed of the following elements or activities (or their equivalent):

1. Individual EDG reliability target levels consistent with the plant category and coping duration selected from Table 2.

2. Surveillance testing and reliability monitoring programs designed to track EDG performance and to support maintenance activities.

3. A maintenance program that ensures that the target EDG reliability is being achieved and that provides a capability for failure analysis and root-cause investigations.

4. An information and data collection system that services the elements of the reliability program and that monitors achieved EDG reliability levels against target values.

5. Identified responsibilities for the major program elements and a management oversight program for reviewing reliability levels being achieved and ensuring that the program is functioning properly.

1.3 Procedures for Restoring Emergency AC Power

Guidelines and procedures for actions to restore emergency ac power when the emergency ac power system is unavailable should be integrated with plant-specific technical guidelines and emergency operating procedures developed using the emergency operating procedure upgrade program established in response to Supplement 1, "Requirements for Emergency Response Capability" (Generic Letter No. 82-33),[1] to NUREG-0737, "Clarification of TMI Action Plan Requirements" (Ref. 12).

[1]Modifications or additions to generic technical guidelines that are necessary to deal with a station blackout for the specific plant design should be identified as deviations in the plant-specific technical guidelines as required by Supplement 1 to NUREG-0737 (Ref. 12) and outlined in NUREG-0899, "Guidelines for the Preparation of Emergency Operating Procedures" (Ref. 13).

2. OFFSITE POWER

Procedures should include the actions necessary to restore offsite power and use nearby power sources[2] when offsite power is unavailable. As a minimum, the following potential causes for loss of offsite power should be considered:

- Grid undervoltage and collapse
- Weather-induced power loss
- Preferred power distribution system faults[3] that could result in the loss of normal power to essential switchgear buses

3. ABILITY TO COPE WITH A STATION BLACKOUT

The ability to cope with a station blackout for a certain time provides additional defense-in-depth should both offsite and onsite emergency ac power systems fail concurrently. Regulatory Position 3.1 provides a method to determine an acceptable minimum time that a plant should be able to cope with a station blackout based on the probability of a station blackout at the site as well as the capability for restoring ac power for that site. Each nuclear power plant has the capability to remove decay heat and maintain appropriate containment integrity without ac power for a limited period of time. Regulatory Position 3.2 provides guidance for determining the length of time that a plant is actually able to cope with a station blackout. If the plant's actual station blackout capability is significantly less than the acceptable minimum duration, modifications may be necessary to extend the plant's ability to cope with a station blackout. Should plant modifications be necessary, Regulatory Position 3.3 provides guidance on making such modifications. Whether or not modifications are necessary, procedures and training for station blackout events should be provided according to the guidance in Regulatory Position 3.4.

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[2]This includes such items as nearby or onsite gas turbine generators, portable generators, hydro generators, and black-start fossil power plants.

[3]Includes such failures as the distribution system hardware, switching and maintenance errors, and lightning-induced faults.

3.1 Minimum Acceptable Station Blackout Duration Capability

Each nuclear power plant should be able to withstand and recover from a station blackout lasting a specified minimum duration. The specified duration of station blackout should be based on the following factors:

1. The redundancy of the onsite emergency ac power system (i.e., the number of power sources available minus the number needed for decay heat removal),

2. The reliability of each of the onsite emergency ac power sources (e.g., diesel generator),

3. The expected frequency of loss of offsite power, and

4. The probable time needed to restore offsite power.

A method for determining an acceptable minimum station blackout duration capability as a function of the above site- and plant-related characteristics is given in Table 2. Tables 3 through 8 provide the necessary detailed descriptions and definitions of the various factors used in Table 2. Table 3 identifies different levels of redundancy of the onsite emergency ac power system used to define the emergency ac power configuration groups in Table 2. Table 4 provides definitions of the three offsite power design characteristic groups used in Table 2. The groups are defined according to various combinations of the following factors: (1) independence of offsite power (I), (2) severe weather (SW), (3) severe weather recovery (SWR), and (4) extremely severe weather (ESW). The definitions of the factors I, SW, SWR, and ESW are provided in Tables 5 through 8, respectively. After identifying the appropriate groups from Tables 3 and 4 and the reliability level of the onsite emergency ac power sources (determined in accordance with Regulatory Position 1.1), Table 2 can be used to determine the acceptable minimum station blackout duration capability for each plant.

3.2 Evaluation of Plant-Specific Station Blackout Capability

Each nuclear power plant should be evaluated to determine its capability to withstand and recover from a station blackout of the acceptable duration determined for that plant (see Regulatory Position 3.1). The following considerations should be included when determining the plant's capability to cope with a station blackout.

3.2.1. The evaluation should be performed assuming that the station blackout event occurs while the reactor is operating at 100% rated thermal power and has been at this power level for at least 100 days.

3.2.2. The capability of all systems and components necessary to provide core cooling and decay heat removal following a station blackout should be determined, including station battery capacity, condensate storage tank capacity, compressed air capacity, and instrumentation and control requirements.

3.2.3. The ability to maintain adequate reactor coolant system inventory to ensure that the core is cooled should be evaluated, taking into consideration shrinkage, leakage from pump seals, and inventory loss from letdown or other normally open lines dependent on ac power for isolation.

3.2.4. The design adequacy and capability of equipment needed to cope with a station blackout for the required duration and recovery period should be addressed and evaluated as appropriate for the associated environmental conditions. This should include consideration as appropriate of the following:

1. Potential failures of equipment necessary to cope with the station blackout,

2. Potential environmental effects on the operability and reliability of equipment necessary to cope with the station blackout, including possible effects of fire protection systems,

3. Potential effects of other hazards, such as weather, on station blackout response equipment (e.g., auxiliary equipment to operate onsite buses or to recover EDGs and other equipment as needed),

4. Potential habitability concerns for those areas that would require operator access during the station blackout and recovery period.

Evaluations that have already been performed need not be duplicated. For example, if safety-related equipment required during a total loss of ac power has been qualified to operate under environmental conditions exceeding those expected under a station blackout (e.g., loss of heating, ventilation, and air conditioning), additional analyses need not be performed. Equipment will be considered acceptable for station blackout temperature environments if an assessment has been performed that provides reasonable assurance that the required equipment will remain operable.

3.2.5. Consideration should be given to using available non-safety-related equipment, as well as safety-related equipment, to cope with a station blackout provided such equipment meets the recommendations of Regulatory Positions 3.3.3 and 3.3.4. Onsite or nearby alternate ac (AAC) power sources that are independent and diverse from the normal Class 1E emergency ac power sources (e.g., gas turbine, separate diesel engine, steam supplies) will constitute an acceptable station blackout coping capability provided an analysis is performed that demonstrates the plant has this capability from the onset of station blackout until the AAC power source or sources are started and lined up to operate all equipment necessary to cope with station blackout for the required duration.

In general, equipment required to cope with a station blackout during the first 8 hours should be available on the site. For equipment not located on the site, consideration should be given to its availability and accessibility in the time required, including consideration of weather conditions likely to prevail during a loss of offsite power.

If the AAC source or sources meet the recommendations of Section 3.3.5 and can be demonstrated by test to be available to power the shutdown busses within 10 minutes of the onset of station blackout, no coping analysis is required.

3.2.6. Consideration should be given to timely operator actions inside or outside the control room that would increase the length of time that the

plant can cope with a station blackout provided it can be demonstrated that these actions can be carried out in a timely fashion. For example, if station battery capacity is a limiting factor in coping with a station blackout, shedding non-essential loads on the batteries could extend the time until the battery is depleted. If load shedding or other operator actions are considered, corresponding procedures should be incorporated into the plant-specific technical guidelines and emergency operating procedures.

3.2.7. The ability to maintain "appropriate containment integrity" should be addressed. "Appropriate containment integrity" for station blackout means that adequate containment integrity is ensured by providing the capability, independent of the preferred and blacked-out unit's onsite emergency ac power supplies, for valve position indication and closure for containment isolation valves that may be in the open position at the onset of a station blackout. The following valves are excluded from consideration:

1. Valves normally locked closed during operation
2. Valves that fail closed on a loss of power
3. Check valves
4. Valves in nonradioactive closed-loop systems not expected to be breached in a station blackout (this does not include lines that communicate directly with containment atmosphere) and
5. Valves of less than 3" nominal diameter.

3.3 Modifications To Cope with Station Blackout

If the plant's station blackout capability, as determined according to the guidance in Regulatory Position 3.2, is significantly less than the minimum acceptable plant-specific station blackout duration (as developed according to Regulatory Position 3.1 or as justified by the licensee or applicant on some other basis and accepted by the staff), modifications to the plant may be necessary to extend the time the plant is able to cope with a station blackout. If modifications are needed, the following items should be considered:

3.3.1. If, after considering load shedding to extend the time until battery depletion, battery capacity must be extended further to meet the station blackout duration recommended in Regulatory Position 3.1, it is considered acceptable either to add batteries or to add a charging system for the existing batteries that is independent of both the offsite and the blacked-out unit's onsite emergency ac power systems, such as a dedicated diesel generator.

3.3.2. If the capacity of the condensate storage tank is not sufficient to remove decay heat for the station blackout duration recommended in Regulatory Position 3.1, a system meeting the requirements of Regulatory Position 3.5 to resupply the tank from an alternative water source is an acceptable means to increase its capacity provided any power source necessary to provide additional water is independent of both the offsite and the blacked-out unit's onsite emergency ac power systems.

3.3.3. If the compressed air capacity is not sufficient to remove decay heat and to maintain appropriate containment integrity for the station blackout duration recommended in Regulatory Position 3.1, a system to provide sufficient capacity from an alternative source that meets Regulatory Position 3.5 is an acceptable means to increase the air capacity provided any power source necessary to provide additional air is independent of both the offsite and the blacked-out unit's onsite emergency ac power systems.

3.3.4. If a system is required for primary coolant charging and makeup, reactor coolant pump seal cooling or injection, decay heat removal, or maintaining appropriate containment integrity specifically to meet the station blackout duration recommended in Regulatory Position 3.1, the following criteria should be met:

1. The system should be capable of being actuated and controlled from the control room, or if other means of control are required, it should be demonstrated that these steps can be carried out in a timely fashion, and

2. If the system must operate within 10 minutes of a loss of all ac power, it should be capable of being actuated from the control room.

3.3.5. If an AAC power source is selected specifically for satisfying the requirements for station blackout, the design should meet the following criteria:

1. The AAC power source should not normally be directly connected to the preferred or the blacked-out unit's onsite emergency ac power system.

2. There should be a minimum potential for common cause failure with the preferred or the blacked-out unit's onsite emergency ac power sources. No single-point vulnerability should exist whereby a weather-related event or single active failure could disable any portion of the blacked-out unit's onsite emergency ac power sources or the preferred power sources and simultaneously fail the AAC power source.

3. The AAC power source should be available in a timely manner after the onset of station blackout and have provisions to be manually connected to one or all of the redundant safety buses as required. The time required for making this equipment available should not be more than 1 hour as demonstrated by test. If the AAC power source can be demonstrated by test to be available to power the shutdown busses within 10 minutes of the onset of station blackout, no coping analysis is required.

4. The AAC power source should have sufficient capacity to operate the systems necessary for coping with a station blackout for the time required to bring and maintain the plant in safe shutdown.

5. The AAC power system should be inspected, maintained, and tested periodically to demonstrate operability and reliability. The reliability of the AAC power system should meet or exceed 95 percent as determined in accordance with NSAC-108 (Ref. 11) or equivalent methodology.

An AAC power source serving a multiple-unit site where onsite emergency ac sources are not shared between units should have, as a minimum, the capacity and capability for coping with station blackout in any of the units.

At sites where onsite emergency sources are shared between units the AAC power sources should have the capacity and capability to ensure that all units can be brought to and maintained in safe shutdown (i.e, those plant conditions defined in plant technical specifications as Hot Standby or Hot Shutdown, as appropriate). Plants have the option of maintaining the RCS at normal operating temperatures or at reduced temperatures.

Plants that have more than the required redundancy of emergency ac sources for loss-of-offsite-power conditions, on a per nuclear unit basis, may use one of the existing emergency sources as an AAC power source provided it meets the applicable criteria for an AAC source. Additionally, emergency diesel generators with 1-out-of-2-shared and 2-out-of-3-shared ac power configurations may not be used as AAC power sources.

3.3.6. If a system or component is added specifically to meet the recommendations on station blackout duration in Regulatory Position 3.1, system walk downs and initial tests of new or modified systems or critical components should be performed to verify that the modifications were performed properly. Failures of added components that may be vulnerable to internal or external hazards within the design basis (e.g., seismic events) should not affect the operation of systems required for the design basis accident.

3.3.7. A system or component added specifically to meet the recommendations on station blackout duration in Regulatory Position 3.1 should be inspected, maintained, and tested periodically to demonstrate equipment operability and reliability.

3.4 Procedures and Training To Cope with Station Blackout

Procedures[4] and training should include all operator actions necessary to cope with a station blackout for at least the duration determined according to Regulatory Position 3.1 and to restore normal long-term core cooling/decay heat removal once ac power is restored.

[4]Procedures should be integrated with plant-specific technical guidelines and emergency operating procedures developed using the emergency operating procedure upgrade program established in response to Supplement 1 of NUREG-0737 (Ref. 12). The task analysis portion of the emergency operating procedure upgrade program should include an analysis of instrumentation adequacy during a station blackout.

3.5 Quality Assurance and Specification Guidance for Station Blackout Equipment That Is Not Safety-Related

Appendices A and B provide guidance on quality assurance (QA) activities and specifications respectively for non-safety-related equipment used to meet the requirements of § 50.63 and not already covered by existing QA requirements in Appendix B or R of Part 50. Appropriate activities should be implemented from among those listed in these appendices depending on whether the non-safety equipment is being added (new) or is existing. This QA guidance is applicable to non-safety systems and equipment for meeting the requirements of § 50.63 of 10 CFR Part 50. The guidance on QA and specifications incorporates a lesser degree of stringency by eliminating requirements for involvement of parties outside the normal line organization. NRC inspections will focus on the implementation and effectiveness of the quality controls described in Appendices A and B. Additionally, the equipment installed to meet the station blackout rule must be implemented such that it does not degrade the existing safety-related systems. This is to be accomplished by making the non-safety-related equipment as independent as practicable from existing safety-related systems. The non-safety systems identified in Appendix B are acceptable to the NRC staff for responding to a station blackout.

D. IMPLEMENTATION

The purpose of this section is to provide information to applicants and licensees regarding the NRC staff's plans for using this regulatory guide. Except in those cases in which the applicant or licensee proposes an acceptable alternative method for complying with specified portions of the Commission's regulations, the method described in this guide may be used in the evaluation of submittals by applicants for construction permits and operating licenses (as appropriate) and will be used to evaluate licensees who are required to comply with § 50.63, "Loss of All Alternating Current Power," of 10 CFR Part 50.

Table 1

Cross-Reference Between Regulatory Guide 1.155 and NUMARC-8700

Regulatory Position in R.G. 1.155	Section in NUMARC-8700
1.1	3.2.3, 3.2.4
1.2	Appendix D
1.3	4.2.1, 4.3.1
2	4.2.2, 4.3.2
3.1	3
3.2.1	2.2.1, 2.2.2
3.2.2	2.9, 7.2.1, 7.2.2, 7.2.3
3.2.3	2.5
3.2.4	2.7, 4.2.1, 4.2.2, 7.2.4, Appendices E and F
3.2.5	7.1.1, 7.1.2, Appendices B and C
3.2.6	4.2.1, 4.3.1, 7.2.1, 7.2.2, 7.2.3
3.2.7	2.10, 7.2.5
3.3.1	7.2.2
3.3.2	7.2.1
3.3.3	None (Use Regulatory Guide 1.155)
3.3.4	2.3.1, Appendices A, B, and C
3.3.5	None (Use Regulatory Guide 1.155)
3.3.6	4.2.1(12), 4.3.1(12), Appendices A and B
3.4	4
3.5	None (Use Regulatory Guide 1.155)
Appendix A	None (Use Regulatory Guide 1.155)
Appendix B	None (Use Regulatory Guide 1.155)

Table 2

Acceptable Station Blackout Duration Capability (hours)[a]

Offsite Power Design Characteristic Group[d]	Emergency AC Power Configuration Group[b]						
	A		B		C		D
	Unit "Average" EDG Reliability[c]						
	0.975	0.95	0.975	0.95	0.975	0.95	0.975
P1	2	2	4	4	4	4	4
P2	4	4	4	4	4	8	8
P3	4	8	4	8	8	16	8

[a]Variations from these times will be considered by the staff if justification, including a cost-benefit analysis, is provided by the licensee. The methodology and sensitivity studies presented in NUREG-1032 (Ref. 2) are acceptable for use in this justification.

[b]See Table 3 to determine emergency ac power configuration group.

[c]See Regulatory Position 1.1.

[d]See Table 4 to determine groups P1, P2, and P3.

Table 3

Emergency AC Power Configuration Groups[a]

EAC Power Configuration Group	Number of EAC Power Sources[b]	Number of EAC Power Sources Required To Operate AC-Powered Decay Heat Removal Systems[c]
A	3[d]	1
	4	1
B	4	2
	5	2
C	2[d]	1
	3[e]	1
D	2[f]	1
	3	2
	4	3
	5	3

[a]Special-purpose dedicated diesel generators, such as those associated with high-pressure core spray systems at some BWRs, are not counted in the determination of EAC power configuration groups.

[b]If any of the EAC power sources are shared among units at a multi-unit site, this is the total number of shared and dedicated sources for those units at the site.

[c]This number is based on all the ac loads required to remove decay heat (including ac powered decay heat removal systems) to achieve and maintain safe shutdown at all units at the site with offsite power unavailable.

[d]For EAC power sources not shared with other units.

[e]For EAC power sources shared with another unit at a multi-unit site.

[f]For shared EAC power sources in which each diesel generator is capable of providing ac power to more than one unit at a site concurrently.

Table 4

Offsite Power Design Characteristic Groups

Group	Offsite Power Design Characteristics			
	Sites that have any combination of the following factors:			
	I[a]	SW[b]	SWR[c]	ESW[d]
P1	1 or 2	1 or 2	1 or 2	1 or 2
	1 or 2	1	1 or 2	3
	1 or 2	3	1	1 or 2
P2	All other sites not in P1 or P3.			
	Sites that expect to experience a total loss of offsite power caused by grid failures at a frequency equal to or greater than once in 20 site-years, unless the site has procedures to recover ac power from reliable alternative (nonemergency) ac power sources within approximately one-half hour following a grid failure.			
	or			
	Sites that have any combination of the following factors:			
P3	I	SW	SWR	ESW
	Any I	5	2	Any ESW
	Any I	1,2,3, or 4	1 or 2	5
	Any I	5	1	Any ESW
	Any I	4	2	1, 2, 3, or 4
	1 or 2	3	2	4
	3	3	2	3 or 4

[a]See Table 5 for definitions of independence of offsite power groups (I).

[b]See Table 6 for definitions of severe weather groups (SW).

[c]See Table 7 for definitions of severe weather recovery groups (SWR).

[d]See Table 8 for definitions of extremely severe weather groups (ESW).

Table 5

Definitions of Independence of Offsite Power Groups

Category	I		
	1	2	3
1. Independence of offsite power sources	1. All offsite power sources are connected to the plant through two or more switchyards or separate incoming transmission lines, but at least one of the ac sources is electrically independent of the others. (The independent 69-kV line in Figure 1 is representative of this design feature.) or	1.a. All offsite power sources are connected to the plant through one switchyard. or 1.b. All offsite power sources are connected to the plant through two or more switchyards, and the switchyards are electrically connected. (The 345- and 138-kV switchyards in Figures 2 and 3 represent this design feature.) and	and
2. Automatic and manual transfer schemes for the Class 1E buses when the normal source of ac power fails and when the backup sources of offsite power fail. a. The normal source of ac power is assumed to be the unit main generator.	2.a. After loss of the normal ac source, (1) There is an automatic transfer of all safe-shutdown buses to a separate preferred alternate power source. (2) There is an automatic transfer of all safe-shutdown buses to one preferred power source. If this preferred power source fails, there is another automatic transfer to the remaining preferred power sources or to alternate offsite power source. or	2.a. After loss of the normal ac power source, there is an automatic transfer of all safe-shutdown buses to one preferred alternate power source. If this source fails, there may be one or more manual transfers of power source to the remaining preferred or alternate offsite power sources. or	2.a. If the normal source of ac power fails, there are no automatic transfers and one or more manual transfers of all safe-shutdown buses to preferred or alternate offsite power sources. or There is one automatic transfer and no manual transfer of all safe-shutdown buses to one preferred or one alternate
b. If the Class 1E buses are normally designed to be connected to the preferred alternate power sources.	2.b. Each safe-shutdown bus is normally connected to a separate preferred alternate power source with automatic or manual transfer capability between the preferred alternate sources	2.b. The safe-shutdown buses are normally aligned to the same preferred power source with either an automatic or manual transfer to the remaining preferred alternate ac power source.	

Table 6

Definitions of Severe Weather Groups (SW)

SW Group	Estimated frequency of loss of offsite power due severe weather, f^a (per site-year)
1	$f < 0.0033$
2	$0.0033 \le f < 0.010$
3	$0.010 \le f < 0.033$
4	$0.033 \le f < 0.10$
5	$0.10 \le f$

[a]The estimated frequency of loss of offsite power due to severe weather, f, is determined by the following equation:

$$f = (1.3 \times 10^{-4})h_1 + (b)h_2 + (0.012)h_3 + (c)h_4$$

where h_1 = annual expectation of snowfall for the site, in inches

h_2 = annual expectation of tornadoes (with wind speeds greater than or equal to 113 miles per hour) per square mile at the site

b = 12.5 for sites with transmission lines on two or more rights-of-way spreading out in different directions from the switchyard, or

b = 72.3 for sites with transmission lines on one right-of-way

h_3 = annual expectation of storms at the site with wind velocities between 75 and 124 mph

h_4 = annual expectation of hurricanes at the site

c = 0 if switchyard is not vulnerable to the effects of salt spray

c = 0.78 if switchyard is vulnerable to the effects of salt spray

The annual expectation of snowfall, tornadoes, and storms may be obtained from National Weather Service data from the weather station nearest to the plant or by interpolation, if appropriate, between nearby weather stations. The basis for the empirical equation for the frequency of loss of offsite power due to severe weather, f, is given in Appendix A to Reference 2.

Table 7

Definitions of Severe Weather Recovery Groups (SWR)

SWR Group	Definition
1	Sites with enhanced recovery (i.e., sites that have the capability and procedures for restoring offsite (nonemergency) ac power to the site within 2 hours following a loss of offsite power due to severe weather).
2	Sites without enhanced recovery.

Table 8

Definitions of Extremely Severe Weather Groups (ESW)

ESW Group	Annual expectation of storms at a site with wind velocities equal to or greater than 125 miles per hour (e)*
1	$e < 3.3 \times 10^{-4}$
2	$3.3 \times 10^{-4} \leq e < 1 \times 10^{-3}$
3	$1 \times 10^{-3} \leq e < 3.3 \times 10^{-3}$
4	$3.3 \times 10^{-3} \leq e < 1 \times 10^{-2}$
5	$1 \times 10^{-2} \leq e$

*The annual expectation of storms may be obtained from National Weather Service data from the weather station nearest to the plant, or by interpolation, if appropriate, between nearby weather stations.

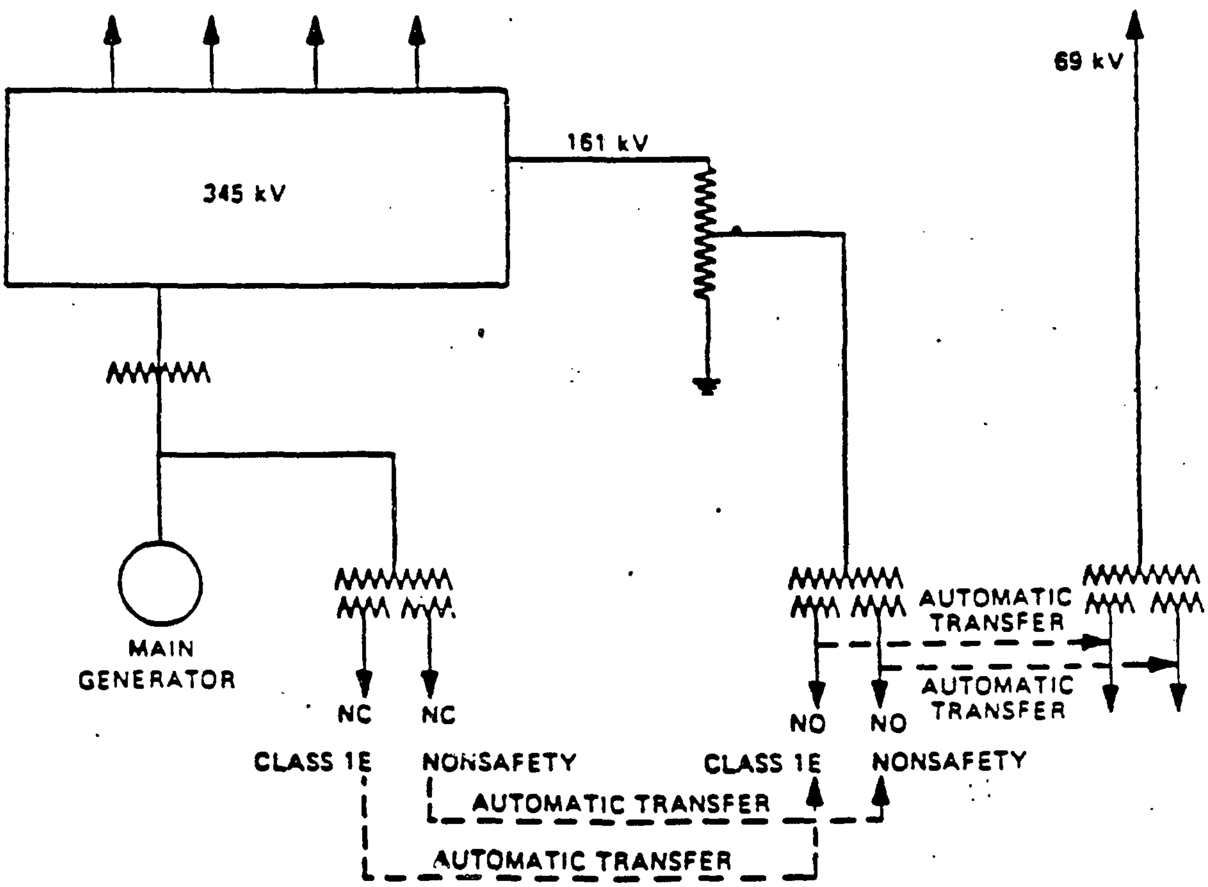

Figure 1. Schematic diagram of electrically independent transmission line

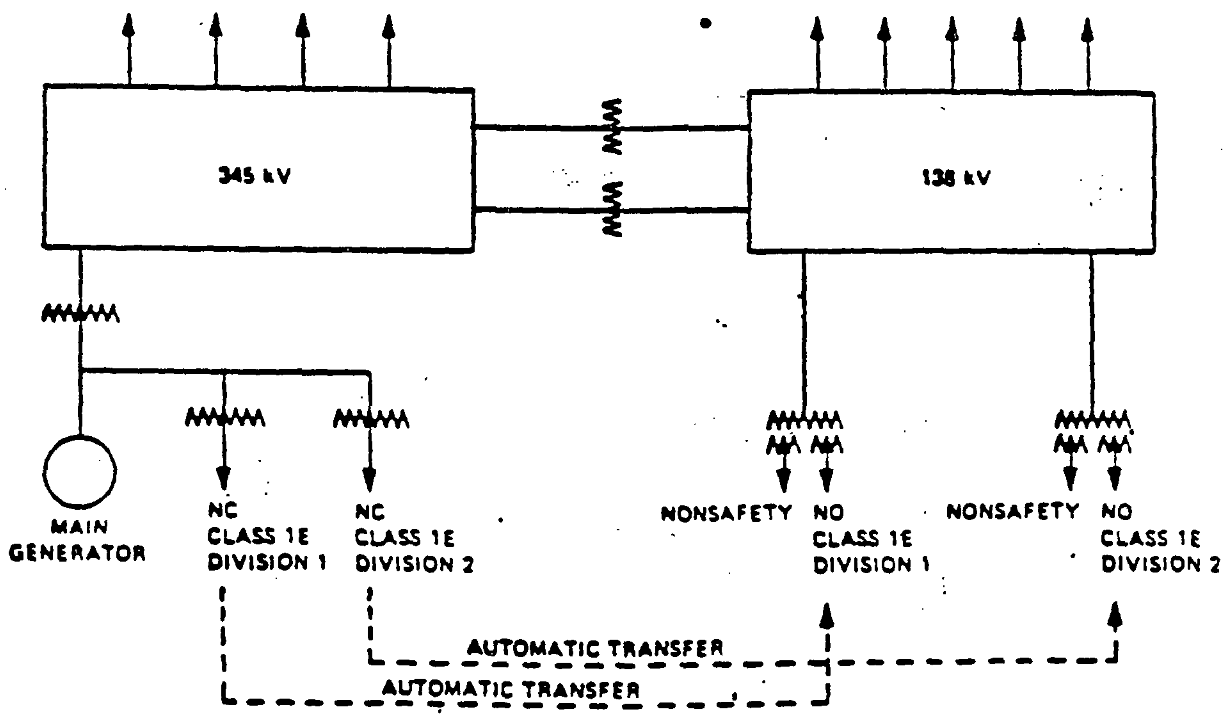

Figure 2. Schematic diagram of two switchyards electrically connected (one-unit site)

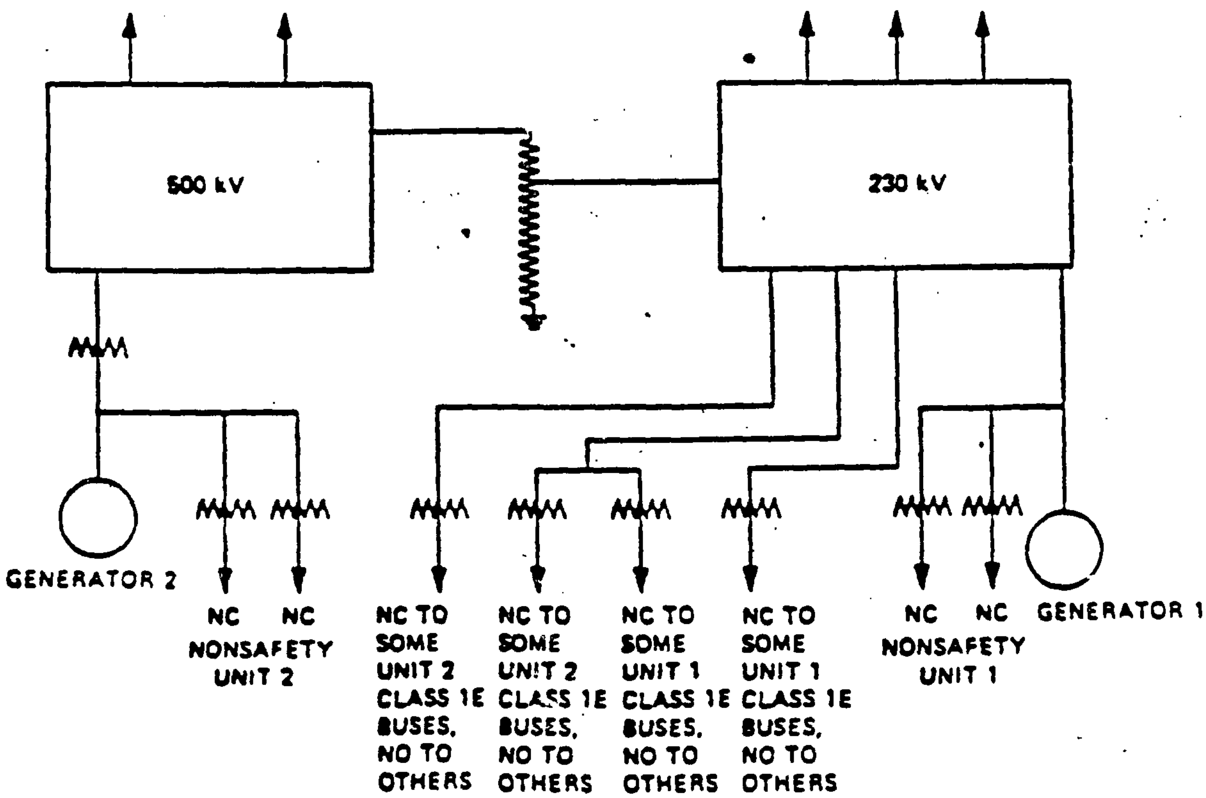

Figure 3. Schematic diagram of two switchyards electrically connected (two-unit site)

REFERENCES

1. U.S. Nuclear Regulatory Commission, "Reactor Safety Study," WASH-1400, October 1975.[1]

2. U.S. Nuclear Regulatory Commission, "Evaluation of Station Blackout Accidents at Nuclear Power Plants, Technical Findings Related to Unresolved Safety Issue A-44," NUREG-1032, publication expected November 1987.[1]

3. A. M. Rubin, "Regulatory Analysis for the Resolution of Unresolved Safety Issue A-44, Station Blackout," U.S. Nuclear Regulatory Commission, NUREG-1109, publication expected November 1987.[1]

4. U.S. Nuclear Regulatory Commission, "Collection and Evaluation of Complete and Partial Losses of Offsite Power at Nuclear Power Plants," NUREG/CR-3992 (ORNL/TM-9384), February 1985.[1]

5. U.S. Nuclear Regulatory Commission, "Reliability of Emergency AC Power System at Nuclear Power Plants," NUREG/CR-2989 (ORNL/TM-8545), July 1983.[1]

6. U.S. Nuclear Regulatory Commission, "Emergency Diesel Generator Operating Experience, 1981-1983," NUREG/CR-4347 (ORNL/TM-9739), December 1985.[1]

7. U.S. Nuclear Regulatory Commission, "Station Blackout Accident Analyses (Part of NRC Task Action Plan A-44)," NUREG/CR-3226 (SAND82-2450), May 1983.[1]

8. Institute of Electrical and Electronics Engineers, "IEEE Standard for Preferred Power Supply for Nuclear Power Generating Stations," IEEE Std 765-1983.[2]

[1]NRC publications may be obtained from the Superintendent of Documents, U.S. Government Printing Office, Post Office Box 37082, Washington, DC 20013-7082; or from the National Technical Information Service, Springfield, VA 22161.

[2]Copies may be obtained from the Institute of Electrical and Electronics Engineers Service Center, 445 Hoes Lane, P.O. Box 1331, Piscataway, NJ 08855.

9. Institute of Electrical and Electronics Engineers, "IEEE Standard Criteria for Diesel-Generator Units Applied as Standby Power Supplies for Nuclear Power Generating Stations," IEEE Std 387-1984.[2]

10. Nuclear Management and Resources Council, "Guidelines and Technical Bases for NUMARC Initiatives Addressing Station Blackout at Light Water Reactors," NUMARC-8700, November 1987.[3]

11. Electric Power Research Institute, "Reliability of Emergency Diesel Generators at U.S. Nuclear Power Plants," NSAC-108, September 1986.[4]

12. U.S. Nuclear Regulatory Commission, "Clarification of TMI Action Plan Requirements: Requirements for Emergency Response Capability" (Generic Letter 82-33), Supplement 1 to NUREG-0737, January 1983.[1]

13. U.S. Nuclear Regulatory Commission, "Guidelines for the Preparation of Emergency Operating Procedures," NUREG-0899, August 1982.[1]

[3]Copies may be obtained from the Nuclear Management and Resources Council, 1726 M Street NW., Washington, DC 20036.

[4]Copies may be obtained from the Electric Power Research Institute, Research Reports Center, P.O. Box 50490, Palo Alto, CA 94303.

Appendix A

QUALITY ASSURANCE GUIDANCE FOR NON-SAFETY SYSTEMS AND EQUIPMENT

The QA guidance provided here is applicable to non-safety systems and equipment used to meet the requirements of § 50.63 and not already explicitly covered by existing QA requirements in 10 CFR Part 50 in Appendix B or R. Additionally, non-safety equipment installed to meet the station blackout rule must be implemented so that it does not degrade the existing safety-related systems. This is accomplished by making the non-safety equipment as independent as practicable from existing safety-related systems. The guidance provided in this section outlines an acceptable QA program for non-safety equipment used for meeting the station blackout rule and not already covered by existing QA requirements. Activities should be implemented from this section as appropriate, depending on whether the equipment is being added (new) or is existing.

1. Design Control and Procurement Document Control

Measures should be established to ensure that all design-related guidelines used in complying with § 50.63 are included in design and procurement documents, and that deviations therefrom are controlled.

2. Instructions, Procedures, and Drawings

Inspections, tests, administrative controls, and training necessary for compliance with § 50.63 should be prescribed by documented instructions, procedures, and drawings and should be accomplished in accordance with these documents.

3. Control of Purchased Material, Equipment, and Services

Measures should be established to ensure that purchased material, equipment, and services conform to the procurement documents.

4. Inspection

A program for independent inspection of activities required to comply with § 50.63 should be established and executed by (or for) the organization performing the activity to verify conformance with documented installation drawings and test procedures for accomplishing the activities.

5. Testing and Test Control

A test program should be established and implemented to ensure that testing is performed and verified by inspection and audit to demonstrate conformance with design and system readiness requirements. The tests should be performed in accordance with written test procedures; test results should be properly evaluated and acted on.

6. Inspection, Test, and Operating Status

Measures should be established to identify items that have satisfactorily passed required tests and inspections.

7. Nonconforming Items

Measures should be established to control items that do not conform to specified requirements to prevent inadvertent use or installation.

8. Corrective Action

Measures should be established to ensure that failures, malfunctions, deficiencies, deviations, defective components, and nonconformances are promptly identified, reported, and corrected.

9. Records

Records should be prepared and maintained to furnish evidence that the criteria enumerated above are being met for activities required to comply with § 50.63.

10. Audits

Audits should be conducted and documented to verify compliance with design and procurement documents, instructions, procedures, drawings, and inspection and test activities developed to comply with § 50.63.

Appendix B

Guidance Regarding System and Station Equipment Specifications

	Alternate AC Sources	Alternate Battery Systems
Safety-Related Equipment (Compliance with IEEE-279)	Not required, but the existing Class 1E electrical systems must continue to meet all applicable safety-related criteria.	Not required, but the existing Class 1E battery systems must continue to meet all applicable safety-related criteria.
Redundancy	Not required.	Not required.
Diversity from existing EDGs	See Regulatory Position 3.3.4 of this guide.	Not required.
Independence from existing safety-related systems	Required if connected to Class 1E buses. Separation to be provided by 2 circuit breakers in series (1 Class 1E at the Class 1E bus and 1 non-Class 1E).	Required if connected to Class 1E battery systems. Separation to be provided by 2 circuit breakers in series (1 Class 1E at the Class 1E bus and 1 non-Class 1E).
Seismic Qualification	Not required.	Not required.
Environmental Consideration	If normal cooling is lost, needed for station blackout event only and not for DBA conditions. Procedures should be in place to effect the the actions necessary to maintain acceptable environmental conditions for the required equipment. See Regulatory Position 3.2.4.	If normal cooling is lost, needed for station blackout event only and not for accident conditions. Procedures should be in place to effect the actions necessary to maintain acceptable environmental conditions for the required equipment. See Regulatory Position 3.2.4.

Appendix B (Continued)

	Alternate AC Sources	Alternate Battery Systems
Capacity	Specified in § 50.63 and Regulatory Position 3.3.4.	Specified in § 50.63 and Regulatory Position 3.3.1.
Quality Assurance	Indicated in Regulatory Position 3.5.	Indicated in Regulatory Position 3.5.
Technical Specification for Maintenance, Limiting Condition, FSAR, etc.	Should be consistent with the Interim Commission Policy Statement on Technical Specifications (Federal Register Notice 52 FR 3789) as applicable.	Should be consistent with the Interim Commission Policy Statement on Technical Specifications (Federal Register Notice 52 FR 3789) as applicable.
Instrumentation and monitoring	Must meet system functional requirements.	Must meet system functional requirements.
Single Failure	Not required.	Not required.
Common Cause Failure (CCF)	Design should, to the extent practicable, minimize CCF between safety-related and non-safety-related systems.	Design should, to the extent practicable, minimize CCF between safety-related and non-safety-related systems.

Appendix B (Continued)

	Water Source (Existing Condensate Storage Tank or Alternative)	Instrument Air (Compressed Air System)	Water Delivery System (Alternative to Auxiliary Feedwater System, RCIC System, or Isolation Condenser Makeup)
Safety-Related Equipment (Compliance with IEEE-279)	Not required, but the existing Class 1E systems must continue to meet all applicable safety-related criteria.	Not required, but the existing Class 1E systems must continue to meet all applicable safety-related criteria.	Not required, but the existing Class 1E systems must continue to meet all applicable safety-related criteria.
Redundancy	Not required.	Not required.	Not required.
Diversity	Not required.	Not required.	Not required.
Independence from Safety-Related Systems	Ensure that the existing safety functions are not compromised, including the capability to isolate components, subsystems, or piping, if necessary.	Ensure that the existing safety functions are not compromised, including the capability to isolate components, subsystems, or piping, if necessary.	Ensure that the existing safety functions are not compromised, including the capability to isolate components, subsystems, or piping, if necessary.
Seismic Qualification	Not required.	Not required.	Not required.

Appendix B (Continued)

	Water Source (Existing Condensate Storage Tank or Alternative)	Instrument Air (Compressed Air System)	Water Delivery System (Alternative to Auxiliary Feedwater System, RCIC System, or Isolation Condenser Makeup)
Environmental Consideration	Needed for station blackout event only and not for DBA conditions. See Regulatory Position 3.2.4. Procedures should be in place to effect the actions necessary to maintain acceptable environmental conditions for required equipment.	Needed for station blackout event only and not for DBA conditions. See Regulatory Position 3.2.4. Procedures should be in place to effect the actions necessary to maintain acceptable environmental conditions for required equipment.	Needed for station blackout event only and not for DBA conditions. See Regulatory Position 3.2.4. Procedures should be in place to effect the actions necessary to maintain acceptable environmental conditions for required equipment.
Capacity	Capability to provide sufficient water for core cooling in the event of a station blackout for the specified duration to meet § 50.63 and this regulatory guide.	Sufficient compressed air to components, as necessary, to ensure that the core is cooled and appropriate containment integrity is maintained for the specified duration of station blackout to meet § 50.63 and this regulatory tory guide.	The capacity to provide sufficient cooling water flow to ensure that the core is cooled in the event of a station blackout for the specified duration to meet § 50.63 and this regulatory guide
Quality Assurance	As indicated in Regulatory Position 3.5.	As indicated in Regulatory Position 3.5.	As indicated in Regulatory Position 3.5.

Appendix B (Continued)

	Water Source (Existing Condensate Storage Tank or Alternative)	Instrument Air (Compressed Air System)	Water Delivery System (Alternative to Auxiliary Feedwater System, RCIC System, or Isolation Condenser Makeup)
Technical Specifications for Maintenance, Surveillance, Limiting Condition, FSAR, etc.	Should be consistent with the Interim Commission Policy Statement on Technical Specifications (Federal Register Notice 52 FR 3789) as applicable.	Should be consistent with the Interim Commission Policy Statement on Technical Specifications (Federal Register Notice 52 FR 4789) as applicable.	Should be consistent with the Interim Commission Policy Statement on Technical Specifications (Federal Register Notice 52 FR 3789) as applicable.
Instrumentation and Monitoring	Must meet system functional requirements.	Must meet system functional requirements.	Must meet system functional requirements.
Single Failure	Not required.	Not required.	Not required.
Common Cause Failure (CCF)	Design should, to the extent practicable, minimize CCF between safety-related and non-safety-related systems.	Design should, to the extent practicable, minimize CCF between safety-related and non-safety-related systems.	Design should, to the extent practicable, minimize CCF between safety-related and non-safety-related systems.

Appendix B (Continued)

	RCS Makeup System (PWRs and BWRs Without RCIC)	Isolation Condenser (BWRs Without RCIC)	Instrumentation and Control Room Indications for Verification of RCS Natural Circulation (PWRs and BWRs Without RCIC)
Safety-Related Equipment (Compliance with IEEE-279)	Not required, but the existing Class 1E systems must continue to meet all applicable safety-related criteria.	Not required, but the existing Class 1E systems must continue to meet all applicable safety-related criteria.	Not required, but the existing Class 1E systems must continue to meet all applicable safety-related criteria.
Redundancy	Not required.	Not required.	Not required.
Diversity	Not required.	Not required.	Not required.
Independence from Safety-Related Systems	1. Safety-grade isolation devices required between this RCS makeup system and existing safety-related makeup water systems. 2. A malfunction of this non-safety-grade makeup system should not affect the design safety function of any safety-related systems.	1. Safety-grade isolation devices between this system and existing safety-related systems. 2. A malfunction of this non-safety-related system should not affect the design safety function of any safety-related systems.	A malfunction of this instrumentation and monitoring system should not affect the design safety function of any safety-related instrumentation and monitoring systems powered by onsite or offsite ac power buses.
Seismic Qualification	Not required.	Not required.	Not required.

Appendix B (Continued)

	RCS Makeup System (PWRs and BWRs Without RCIC)	Isolation Condenser (BWRs Without RCIC)	Instrumentation and Control Room Indications for Verification of RCS Natural Circulation (PWRs and BWRs Without RCIC)
Environmental Consideration	Needed for station blackout event only and not for DBA conditions if normal cooling is lost. See Regulatory Position 3.2.4. Procedures should be in place to effect the actions necessary to maintain acceptable environmental conditions for the required equipment.	Needed for station blackout event only and not for DBA conditions if normal cooling is lost. See Regulatory Position 3.2.4. Procedures should be in place to effect the actions necessary to maintain acceptable environmental conditions for the required equipment.	Needed for station blackout event only and not for DBA conditions if normal cooling is lost. See Regulatory Position 3.2.4. Procedures should be in place to effect the actions necessary to maintain acceptable environmental conditions for the required equipment.
Capacity	Sufficient RCS makeup so that core temperatures are maintained at acceptably low values considering a loss of RCP water inventory through a postulated RCP seal failure during the specified duration of station blackout, with a minimum assumed RCP seal leakage of 20 gpm per RCP, unless a lower value is justified.	Provide sufficient capacity for decay heat removal. During the specified duration of station blackout, the isolation condenser pool side requires a water makeup system powered by sources independent from onsite and offsite ac buses.	Provide sufficient instrumentation and control room indications for parameters required for verification of RCS natural circulation during the specified duration of station blackout.
Quality Assurance	As indicated in Regulatory Position 3.5.	As indicated in Regulatory Position 3.5.	As indicated in Regulatory Position 3.5.

Appendix B (Continued)

	RCS Makeup System (PWRs and BWRs Without RCIC)	Isolation Condenser (BWRs Without RCIC)	Instrumentation and Control Room Indications for Verification of RCS Natural Circulation (PWRs and BWRs Without RCIC)
Technical Specifications for Maintenance, Surveillance, Limiting Condition, FSAR, etc.	Should be consistent with the Interim Commission Policy Statement on Technical Specifications (Federal Register Notice 52 FR 4789) as applicable.	Should be consistent with the Interim Commission Policy Statement on Technical Specifications (Federal Register Notice 52 FR 4789) as applicable.	Should be consistent with the Interim Commission Policy Statement on Technical Specifications (Federal Register Notice 52 FR 3789) as applicable.
Instrumentation and Monitoring	Must meet system functional requirements.	Must meet system functional requirements.	--------
Single Failure	Not required.	Not required.	Not required.
Common Cause Failure (CCF)	Design should, to the extent practicable, minimize CCF between safety-related and non-safety-related systems.	Design should, to the extent practicable, minimize CCF between safety-related and non-safety-related systems.	Design should, to the extent practicable, minimize CCF between safety-related and non-safety-related systems.

VALUE/IMPACT STATEMENT

A separate value/impact statement was not prepared for this regulatory guide. The regulatory analysis prepared for the station blackout rule (NUREG-1109) provides the regulatory basis for this guide and examines the costs and benefits of the rule as implemented by the guide. A copy of NUREG-1109 is available for inspection and copying for a fee at the NRC Public Document Room, 1717 H Street NW., Washington, DC 20555. Free single copies may be obtained upon written request to the Distribution Section, Room P-034, Division of Information Support Services, U.S. Nuclear Regulatory Commission, Washington, DC 20555.

ENCLOSURE E

USI A-44 EDO PKG

11-12-87

NUREG-103

Evaluation of Station Blackout Accidents at Nuclear Power Plants

Technical Findings Related to Unresolved Safety Issue A-44

Final Report

U.S. Nuclear Regulatory Commission

Office of Nuclear Regulatory Research
Office of Nuclear Reactor Regulation

P. W. Baranowsky

EVALUATION OF STATION BLACKOUT ACCIDENTS AT NUCLEAR POWER PLANTS
FINAL REPORT

NUREG-1032

ABSTRACT

"Station Blackout," which is the complete loss of alternating current (AC) electrical power in a nuclear power plant, has been designated as Unresolved Safety Issue A-44. Because many safety systems required for reactor core decay heat removal and containment heat removal depend on AC power, the consequences of a station blackout could be severe. This report documents the findings of technical studies performed as part of the program to resolve this issue. The important factors analyzed include: the frequency of loss of offsite power; the probability that emergency or onsite AC power supplies would be unavailable; the capability and reliability of decay heat removal systems independent of AC power; and the likelihood that offsite power would be restored before systems that cannot operate for extended periods without AC power fail, thus resulting in core damage. This report also addresses effects of different designs, locations, and operational features on the estimated frequency of core damage resulting from station blackout events.

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PREFACE

This report represents the culmination of several technical studies undertaken by Nuclear Regulatory Commission (NRC) staff and contractors to place a reliability and risk perspective on Unresolved Safety Issue A-44, "Station Blackout." The technical findings published in this report are intended to document the basis for future NRC regulatory activities that will be the resolution of this safety issue.

The analyses, evaluations, and results presented are meant to provide a "best estimate" assessment of the major contributors to the frequency of station blackout and the probability of subsequent core damage. Most results are presented as point estimates and are intended for use in the quantitative regulatory analyses that will be used to support a proposed resolution of this issue. The uncertainties in the quantitative analyses are large enough that rigorous application of these results should be made with caution. However, the staff believes that the qualitative insights and conclusions are correct and useful as guidance in determining what constitutes resolution of this issue.

P.W. Baranowsky

ACKNOWLEDGMENTS

The preparation of this report involved the technical contribution, review, and comment of several individuals in addition to the principal author. The contributions of the following NRC staff members are hereby acknowledged and appreciation given:

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1 EXECUTIVE SUMMARY

Station blackout is the complete loss of alternating current (AC) electrical power to the essential and nonessential switchgear buses in a nuclear power plant. Because many safety systems required for reactor core cooling and containment heat removal depend on AC power, the consequences of a station blackout could be severe. Existing regulations do not require explicitly that nuclear power plants be capable of withstanding a station blackout.

In 1975, the "Reactor Safety Study" (NUREG-75/140) showed that station blackout could be an important contributor to the total risk from nuclear power plant accidents. In addition, as operating experience accumulated, the concern arose that the reliability of both the onsite and offsite emergency AC power systems might be less than originally anticipated. Thus, in 1979 the Nuclear Regulatory Commission (NRC) designated station blackout as an unresolved safety issue (USI); a task action plan for its resolution (TAP A-44) was issued in July 1980, and work was begun to determine whether additional safety requirements were needed.

Technical studies performed to resolve this safety issue have identified the dominant factors affecting the likelihood of station blackout accidents at nuclear power plants. A summary of the principal probabilistic results is in Table 1.1. These results are based on operating experience; the results of several plant-specific probabilistic safety studies; and reliability, accident sequence, and consequence analyses performed as part of TAP A-44.

The results show the following important characteristics of station blackout accidents:

(1) The variability of estimated station blackout likelihood is potentially large, ranging from approximately 10^{-5} to 10^{-3} per reactor-year. A "typical" estimated frequency is on the order of 10^{-4} per reactor-year.

(2) The capability to restore offsite power in a timely manner (less than 8 hours) can have a significant effect on accident consequences.

(3) The redundancy of onsite AC power systems and the reliability of individual power supplies have a large influence on the likelihood of station blackout events.

(4) The capability of the decay heat removal system to cope with long duration blackouts (greater than 2 hours) can be a dominant factor influencing the likelihood of core damage or core melt for the accident sequence.

(5) The estimated frequency of station blackout events that result in core damage or core melt can range from approximately 10^{-6} to greater than 10^{-4} per reactor-year. A "typical" core damage frequency estimate is on the order of 10^{-5} per reactor-year.

Table 1.1 Summary of station blackout program technical results

Parameter	Value
<u>Operational Experience</u>	
Loss of offsite power (occurrence per year)	
Average	0.1
Range	0 to 0.4
Time to restore offsite power (hours)	
Median	0.6
90% restored	3.0
Emergency diesel generator reliability (per demand)	
Average	0.98
Range	0.9 to 1.0
Median emergency diesel generator repair time (hours)	8
<u>Analytical Results</u>	
Estimated range of unavailability of emergency AC power systems (per demand)	10^{-4} to 10^{-2}
Estimated range of frequency of station blackout (per year)	10^{-5} - 10^{-3}
Estimated range of frequency of core damage as a result of station blackout (per year)	10^{-6} - 10^{-4}

(6) Information currently available indicates that containment failure as a result of overpressure may follow a station-blackout-induced core melt. Smaller, low-design-pressure containments are most susceptible to early failure (possibly in less than 8 hours). Some large, high-design-pressure containments may not fail as a result of overpressure, or if they do fail, the failure time could be on the order of a day or more.

The losses of offsite power can be categorized as those resulting from (1) plant-centered faults, (2) utility grid blackouts, and (3) failures of offsite power sources induced by severe weather. The industry average frequency of total losses of offsite power was determined to be about 0.1 per site/year, and the median restoration time was about one-half hour. The factors identified as affecting the frequency and duration of offsite power losses are

(1) the design of preferred power distribution system, particularly the number and independence of offsite power circuits from the point where they enter the site up to the safety buses

(2) operations that can compromise redundancy or independence of multiple offsite power sources, including human error

(3) the reliability and security of the power grid, and the ability to restore power to a nuclear plant site with a grid blackout

(4) the hazard from, and susceptibility to, severe weather conditions that can cause loss of offsite power for extended periods

A review of the design and operating experience, combined with a reliability analysis of the onsite emergency AC power system, has shown that there are a variety of potentially important causes of failure. The typical unavailability of a two-division emergency AC power system is about 10^{-3} per demand, and the typical failure rate of individual emergency diesel generators is about 2×10^{-2} per demand. The factors identified as affecting emergency AC power system reliability during a loss of offsite power are

(1) power supply configuration redundancy

(2) reliability of each power supply

(3) dependence of the emergency AC power system on support or auxiliary cooling systems and control systems, and the reliability of those support systems

(4) vulnerability to common cause failures associated with design, operational, and environmental factors

The likelihood that a station blackout will progress to core damage or core melt is dependent on the reliability and capability of decay heat removal systems that are not dependent on AC power. If the capability is sufficient, additional time will be available to restore AC power to the many systems normally used to cool the core and remove decay heat. The most important factors relating to decay heat removal during a station blackout are

(1) the starting reliability of systems required to remove decay heat and maintain reactor coolant inventory

(2) the capacity and ability to function of decay heat removal systems and auxiliary or support systems that must remain functional during a station blackout (e.g., direct current (DC) electrical power, condensate storage), including effects of inoperable heating, ventilation, and air conditioning (HVAC) systems

(3) for pressurized water reactors (PWRs) and for boiling water reactors (BWRs) without reactor coolant makeup capability during a station blackout, the magnitude of reactor coolant pump seal leakage

(4) for BWRs that remove decay heat to the suppression pool, the ability to maintain suppression pool integrity and operate heat removal systems at high pool temperatures during recirculation

(5) recovery of AC power including availability of alternate AC power sources

On the basis of reviews of design, operation, and location factors, the staff determined that the expected core melt frequency from station blackout could be maintained around 10^{-5} per reactor-year or lower for all plants. To reach this level of core melt frequency, a plant would have to be able to cope with station blackouts on the order of 2 to 4 and perhaps 8 hours long and have emergency diesel generator reliabilities of 0.95 per demand or better, with relatively low susceptibility to common cause failures.

2 INTRODUCTION AND TECHNICAL APPROACH

Station blackout refers to the complete loss of AC electrical power to the essential and nonessential switchgear buses in a nuclear power plant. Station blackout involves the loss of offsite power concurrent with the failure of the onsite emergency AC power system. It does not include the loss of available AC power to buses fed by station batteries through inverters. Because many safety systems required for reactor core cooling, decay heat removal, and containment heat removal depend on AC power, the consequences of station blackout could be severe.

The concern about station blackout is based on accumulated operating experience regarding the reliability of AC power supplies. A number of operating plants have experienced a total loss of offsite electrical power, and more such occurrences are expected. During these loss-of-offsite-power events, onsite emergency AC power sources were available to supply the power needed by vital safety equipment. However, in some instances one of the redundant emergency power supplies was unavailable, and in a few cases there was a complete loss of AC power. (During these events AC power was restored in a short time without any serious consequences.) In addition, there have been numerous instances at operating plants in which emergency diesel generators failed to start and run during surveillance tests.

For one of two plants evaluated, the Reactor Safety Study (NUREG-75/014) showed that station blackout could be an important contributor to the total risk from nuclear power plant accidents. Although this total risk was found to be small, the relative importance of the station blackout event was established. This finding, with the accumulated data on diesel generator failures, increased the concern about station blackout.

An analysis of the risk from station blackout involves an assessment of (1) the likelihood and duration of the loss of offsite power, (2) the reliability of onsite AC power systems, and (3) the potential for severe accident sequences after a loss of all AC power. These topics were investigated under USI TAP A-44. This plan included the following major tasks:

(1) Estimating the frequency of station blackout at operating U. S. nuclear power plants. This analysis consisted of two parts

- estimating the frequency of loss of offsite power for various plant locations

- estimating the probability that the onsite AC power system will fail to supply AC power for core cooling

(2) Determining plant responses to station blackout and the risk associated with station-blackout-initiated accident sequences. The scope of this investigation included

- reviewing shutdown cooling systems design and assessing their capability and reliability during a prolonged station blackout

- reviewing containment designs and their ability to withstand temperature and pressure buildup during a prolonged loss of AC power

- estimating the probability of station blackout accident sequences for a spectrum of nuclear power plant designs

The principal focus of TAP A-44 was the reliability of emergency AC power supplies. This approach was taken for several reasons. First, station blackout was identified as a USI primarily on the basis of the questions raised about the reliability of onsite emergency power supplies. Second, if safety improvements are required, it is easier to analyze, identify, and implement them for the onsite AC power system than for the offsite AC power supplies or for the AC-independent decay heat removal system. For example, offsite power reliability is dependent on a number of factors--such as regional electrical grid stability, weather phenomena, and repair and restoration capability--that are difficult to analyze and to control. Also, the capability of a plant to withstand a station blackout depends on those decay heat removal systems, components, instruments, and controls that are independent of AC power. These features vary from plant to plant; thus considerable effort is required to analyze all of them or to ensure that the plants indeed have that capability. Third, significant progress has been made on improving operating PWRs by backfitting the auxiliary feedwater system to make it independent of AC power. In addition, under the TAP for USI A-45, "Shutdown Decay Heat Removal Requirements," the adequacy of shutdown decay heat removal systems for nuclear power plants is being reviewed. Thus, the reliability of emergency AC power supplies is of principal importance to USI A-44.

A preliminary screening analysis was done to identify plants most likely to suffer core damage as a result of a loss of all AC power. The intent was to survey the frequency and implication of station blackout events in operating plants and identify any plants with especially high risk that might require further analysis or action on an urgent basis. The initial results showed no such plants.

Following this initial analysis, station blackout events were evaluated in more detail. Because the station blackout issue centers on concern about the reliability of AC power supplies, typical offsite and emergency AC power supplies were evaluated and operating (failure) experience reviewed. This effort was limited to power supply availability and did not include an evaluation of the adequacy of power distribution design or power capacity requirements.

Information on loss of offsite power was collected from licensee event reports (LERs), responses to an NRC questionnaire, and various reports prepared by industry sources. Most of the event descriptions in the LERs and in other documentation in the NRC files did not contain sufficient information to provide an accurate data base for estimating frequencies and durations of losses of offsite power. For example, in one case a licensee reported that offsite power was restored in 6 hours; in fact, <u>one</u> offsite power source was restored in 8 minutes and <u>all</u> offsite power was restored in 6 hours. Because restoration of <u>one</u> source of offsite power terminates a loss of offsite power, the licensee's description was not accurate enough. In some other cases, although

offsite power was available to be reconnected, the plant operators did not reconnect it for some time after it was available because onsite power was available. To obtain more accurate data, the NRC and Oak Ridge National Laboratory staff members worked closely with the Institute of Electrical and Electronics Engineers (IEEE) and the Electric Power Research Institute (EPRI). These groups contacted utility engineers to get better descriptions of the causes and sequences of events, and the times and methods of restoring offsite power (Wykcoff, May and September 1986).

To gain a perspective on consequences, station blackout event sequences and associated plant responses were analyzed. The Interim Reliability Evaluation Program (IREP) was one source of information for developing the shutdown cooling reliability models and accident scenarios needed for this evaluation. The Reactor Risk Reference Document (NUREG-1150) and supporting studies were a source of information for developing an updated perspective on containment failure and consequences associated with a station blackout accident.

The following sections of this report summarize the results of the technical evaluations discussed above. Details of the technical assessments performed as part of USI TAP A-44 are reported in NUREG/CR-2989, -3226, and -3992. Significant use was also made of NSAC/103 (Wyckoff, May 1986) and NSAC/108 (Wyckoff, September 1986) as well as other documents produced to assess various station blackout concerns which are appropriately referenced throughout this report. Technical evaluations in this report were derived from these references to coalesce that material and extend the analysis to obtain the broader insights and bases necessary to resolve the station blackout issue in an integral manner, considering plant differences. These supplemental analyses are described in Appendices A, B, and C of this report.

3 LOSS OF OFFSITE POWER FREQUENCY AND DURATION

The offsite or preferred power system at nuclear power plants consists of the following major components:

- two or more incoming power supplies from the grid
- one or more switchyards to allow routing and distribution of power within the plant
- one or more transformers to allow the reduction of voltage to levels needed for safety and non-safety systems within the plant
- distribution systems from the transformers to the switchgear buses

Figure 3.1 provides an example of an offsite power system design used for nuclear power plants. During normal operation, AC power is typically provided to the safety and non-safety buses from the main generator through the auxiliary transformer; it may also be supplied directly through a startup transformer. A minimum of two preferred power supply circuits must be provided. Sources of offsite power other than the grid may also be provided as alternate or backup sources of power. These may include nearby (or onsite) gas turbine generators, fossil power plants, and hydroelectric power facilities. A loss of offsite power is said to occur when all sources of offsite power become unavailable, causing safety buses to become deenergized and initiating an undervoltage signal. Some loss-of-offsite-power transients will be very short--just long enough to allow switching from one failed source to another available source. Because of the short duration of this type of loss-of-offsite-power transient, it is not of concern relative to station blackout. This type of loss-of-offsite-power transient is better described as an interruption. However, if switching errors or failures of alternate sources of power compound the situation and longer term repair, restoration, or actuation of alternate power sources is required, the loss-of-offsite-power transient can be significant. This type of loss-of-offsite-power event is referred to as a total loss of offsite power.

Although total loss of offsite power is relatively infrequent at nuclear power plants, it has happened a number of times and a data base of information has been compiled (Wyckoff, May 1986; NUREG/CR-3992). Historically, a loss of offsite power occurs about once per 10 site-years. The typical duration of these events is on the order of one-half hour. However, at some power plants the frequency of offsite power loss has been substantially greater than the average, and at other plants the duration of offsite power outages has greatly exceeded the norm. Table 3.1 provides a summary of the data on total-loss-of-offsite-power events through 1985.

Because design characteristics, operational features, and the location of nuclear power plants within different grids and meteorological areas can have a significant effect on the likelihood and duration of loss-of-offsite-power events, it was necessary to analyze the generic data in more detail. The data

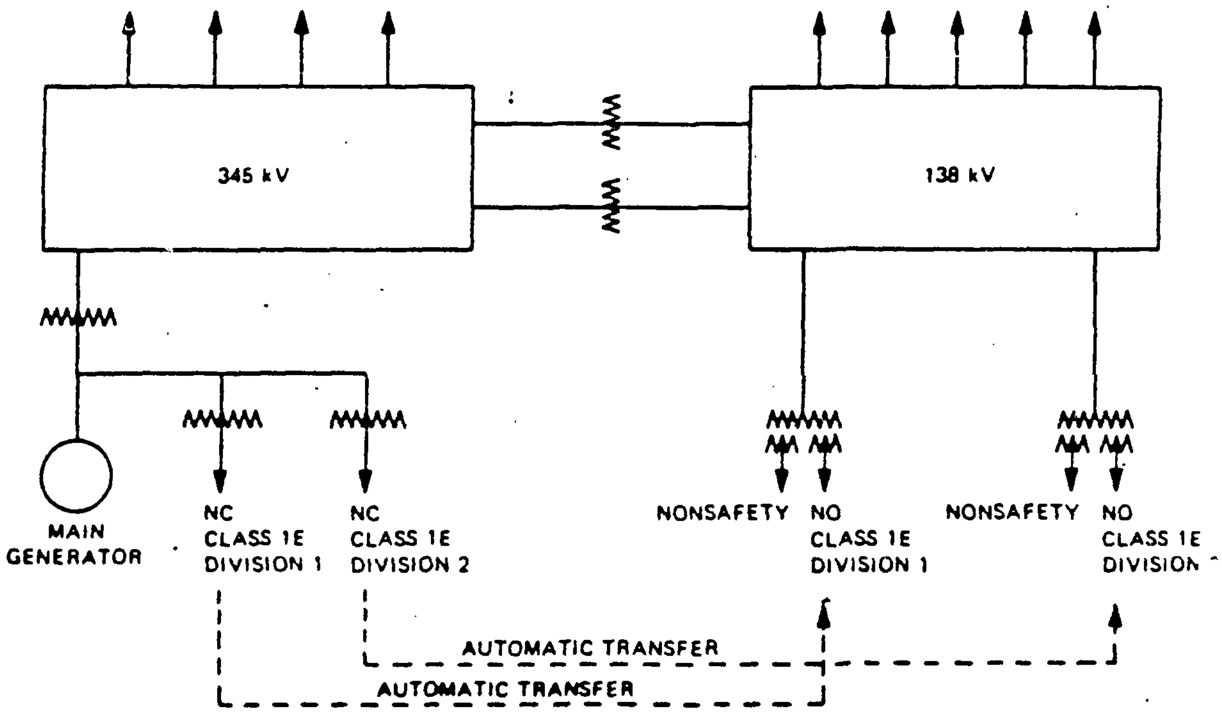


Figure 3.1 Diagram of offsite power system used in nuclear power plants

Table 3.1 Total losses of offsite power at U.S. nuclear power plant sites, 1968 through 1985

Type of event	Number	Frequency of occurrence (yr^{-1})*	Median duration (hours)
Plant-centered	46	0.087	0.3
Grid	12	0.018	0.6
Weather	6	0.009	3.5**
Total	64	0.114	0.6

*Through December 1985, 664 site-years were used to compute the frequency of grid and weather events. Reactor critical site-years totaling 527 for the same period were used to compute the frequency of plant-centered events due to data screening. (See Appendix A.)

**The median value of 3.5 hours was obtained from a two-parameter Weibull curve fit of the data. The actual median is 4.5 hours.

have been categorized into plant-centered events and area- or weather-related events. Plant-centered events are those in which the design and operational characteristics of the plant itself play a role in the likelihood of the loss of offsite power. Area- or weather-related events include those on which the reliability of the grid or external influences on the grid have an effect on the likelihood and duration of the loss of offsite power. The data show that plant-centered events account for the majority of the loss-of-offsite-power events. The area- or weather-related events, although of lesser frequency, typically account for the longer duration outages with storms being the major factor. Figure 3.2 provides a plot of the frequency and duration of loss-of-offsite-power events resulting from plant-centered faults, grid blackout, and severe weather based on past experience at nuclear plant sites.

Appendix A to this report provides a more thorough discussion of the technical bases for the loss-of-offsite power frequency and duration characteristics discussed in the remainder of this section.

Plant-centered failures typically involve hardware failures, design deficiencies, human errors (maintenance and switching), and localized weather-induced faults (lightning and ice), or combinations of these types of failure. No strong correlation was found between the frequency of plant-centered loss-of-offsite-power events and any particular design factor. However, a modest correlation was observed between the duration of plant-centered loss-of-offsite-power events and the independence and redundancy of offsite power circuits at a site. In this regard, it has been observed that a site with several immediate and delayed access circuits will generally recover offsite power more promptly

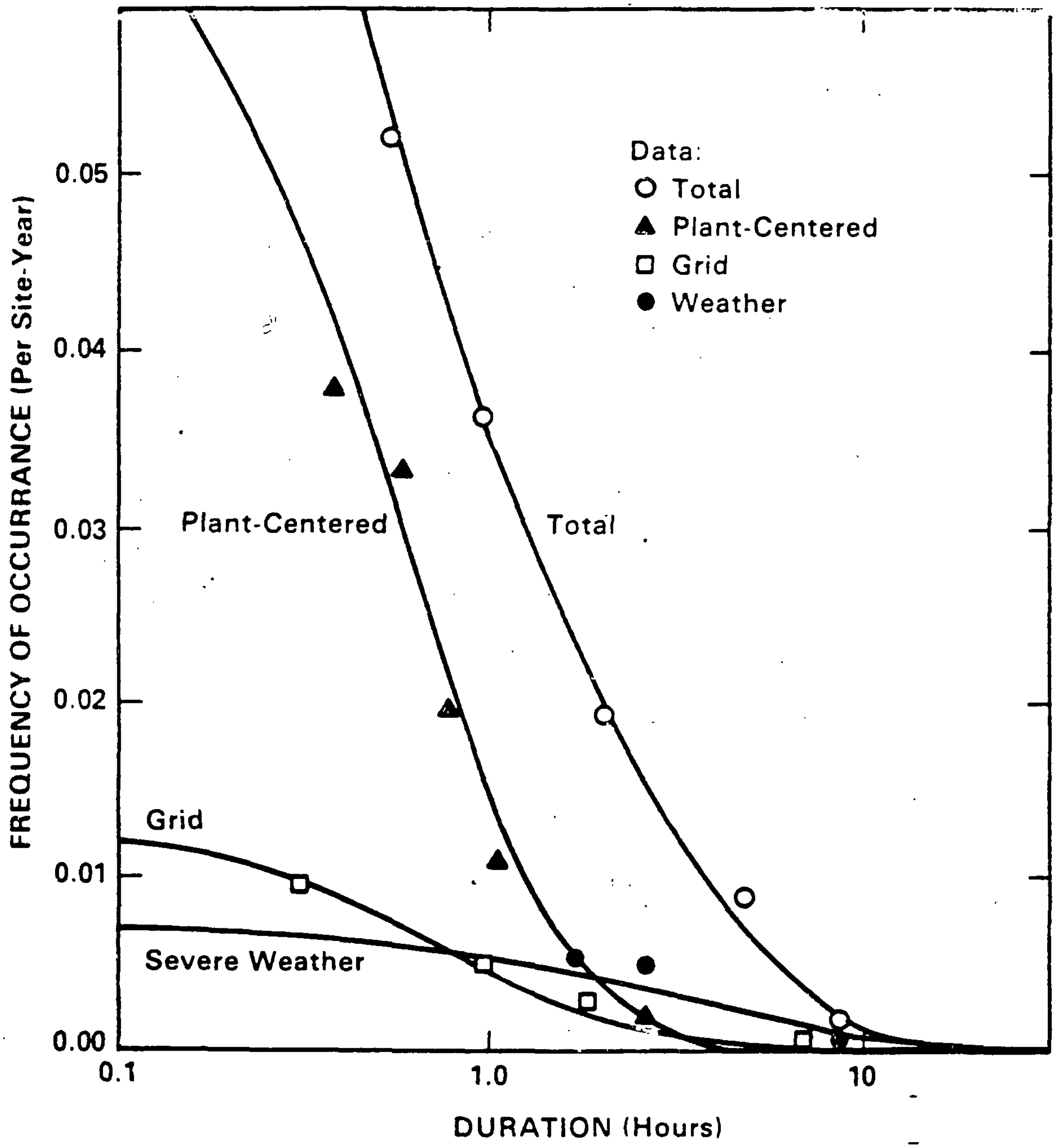


Figure 3.2 Frequency of loss-of-offsite-power events exceeding specified durations

than a site with only the minimum requirements. However, recovery from the relatively high frequency plant-centered faults can be accomplished within a few hours.

Plant location plays an important role in loss-of-offsite-power events. Factors shown to be significant were (1) the reliability of the grid from which the nuclear power plant draws its preferred power supply and (2) the likelihood of severe weather that can cause damage to the grid distribution system and hence a loss of power to the plant. Traditionally, analyses have focused on grid reliability as a dominant factor in estimating loss of offsite power at a plant site. However, a review of the historical data shows that approximately 19% of all loss-of-offsite-power events have been caused by grid problems; in fact, a large percentage of grid-related loss-of-offsite-power events can be traced to one utility's system. The grid reliability of that system dominates the data, distorting the perspective on the contribution of grid failure to loss-of-offsite-power frequency. This finding of overall grid reliability should not be unexpected when one recognizes that current distribution and dispatch systems are well coordinated. Utilities shed loads when possible and generally protect their grid from overloads and faults that could cause grid loss in the various day-to-day operations. Moreover, when there is a loss of power on the grid, the first activity that is usually undertaken is the restoration of power to the electric generation plants so that the grid may be restored to customers with appropriate power supplies. In fact, during the Northeast blackout of 1965, power was restored to a nuclear power plant in New England within about one-half an hour of the grid collapse, while power was not restored to the entire grid for 24 hours or more.

With the exception of a few utility systems, large grid disturbances are relatively infrequent, and, again with few exceptions, the duration of power outages at power plants as a result of grid disturbances is relatively short. An identified weakness in a system is usually corrected as soon as practical; it is the unidentified weaknesses that result in grid failures. In the absence of a historical trend, operating experience related to grid reliability is not necessarily an indication of future problems unless a known weakness has not been corrected. Because grids in the United States are generally very stable and system planning is directed at maintaining and improving that stability, grid reliability is usually not the principal indicator of the likelihood of loss of offsite power.

Severe weather, such as local or area-wide storms, can disrupt incoming power supplies to the plant. In fact, a number of loss-of-offsite-power events at nuclear power plants were weather related. These can be divided into two failure groups:

(1) those in which the weather caused the event but did not affect the time to restore power

(2) those in which the weather initiated the event and caused adverse conditions over a sufficiently broad area such that power was not or could not be restored for a long time

The first group includes lightning and most other weather events that are not too severe. They can cause a loss of offsite power, but their severity generally

does not contribute in any significant way to long-duration losses of offsite power. These types of weather-related losses of offsite power have been treated as either plant-centered or grid-related losses of offsite power. The second group includes losses of offsite power as a result of severe weather such as hurricanes, high winds, snow and ice storms, and tornadoes. The expected loss-of-offsite-power frequency of this group is relatively small. On the other hand, the likelihood of restoring offsite power quickly for this group is also relatively small. Although it is expected that the actions of dispatch and plant personnel can influence substantially the duration of area-wide grid disturbances that cause a loss of offsite power, severe weather conditions--and the expected duration of the resulting loss-of-offsite-power events--cannot be influenced in the same way. Therefore, one would expect severe weather to dominate the restoration characteristics for long-duration outages. The redundancy, separation, and independence of the offsite power system may affect the likelihood of some weather-related losses such as those induced by tornado strikes. The depth of this study has not been sufficient to show the effectiveness of these design considerations on reducing the likelihood of other types of weather-related outages.

There is a potentially large variation in the annual expected frequency of loss-of-offsite-power events at different nuclear power plants, depending on their design and location. A large variation also has been observed in the duration of loss-of-offsite-power events at different nuclear power plants. The expection of long-duration outages is dominated by the likelihood of severe storms and, to a lesser extent, by the likelihood of grid blackout and the ability to restore power to the site during grid loss. Grid-related losses are important only when the frequency of occurrence greatly exceeds the national average.

Appendix A describes the modeling and analyses performed by NRC staff to determine the relationship between design and location and the frequency of and duration of loss-of-offsite-power events representative of most U.S. nuclear power plant sites. Figure 3.3 provides a plot of the expected frequency and duration for loss of offsite power for site, design, grid, and weather characteristics that have been found to "cluster" reasonably well. The factor that most predominantly affects the characteristic groupings is severe weather. Table 3.2 provides a definition of the site characteristics that make up the loss-of-offsite-power clusters shown. Appendix A includes additional discussion of the characteristics of these clusters.

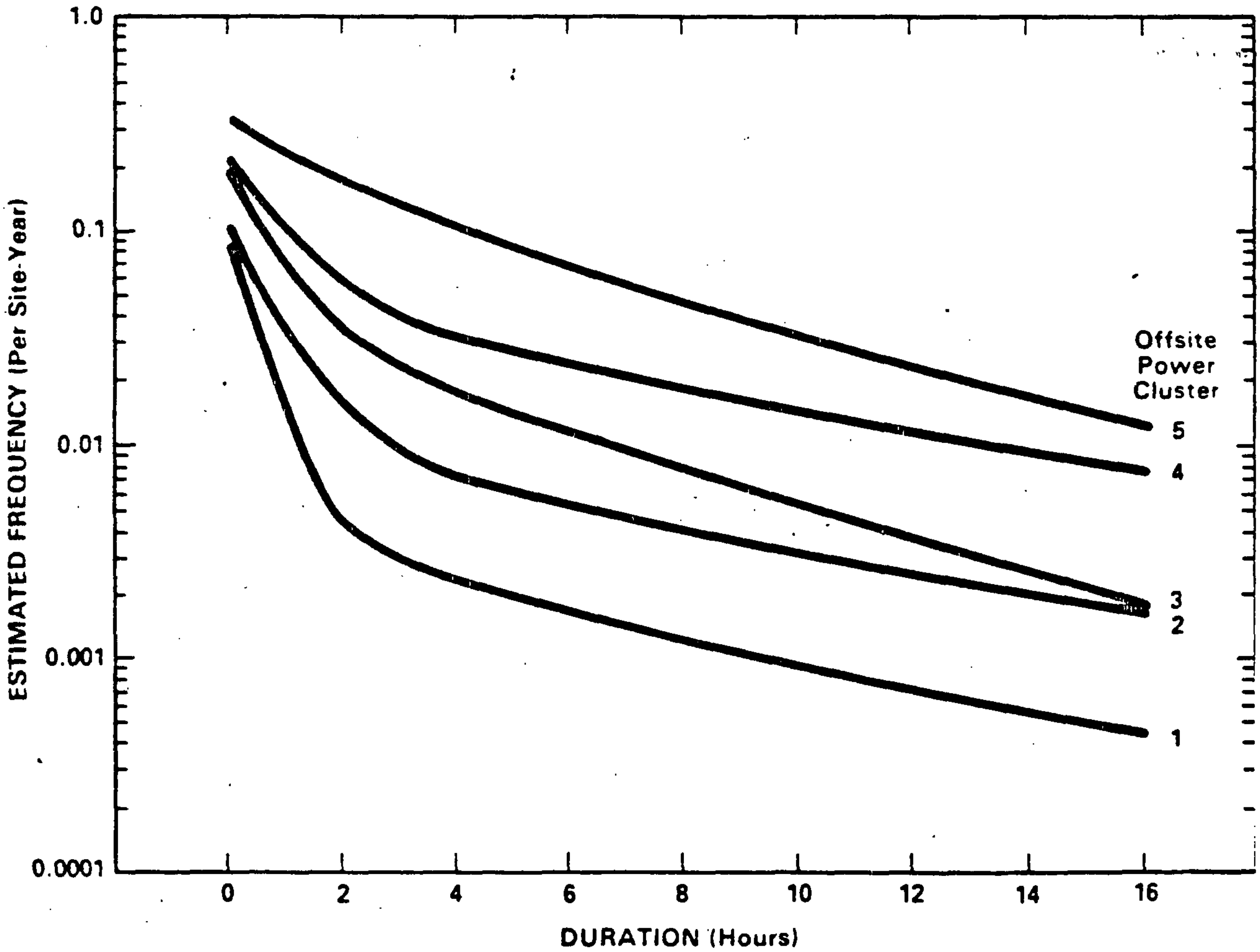


Figure 3.3 Estimated frequency of loss-of-offsite-power events exceeding specified durations for representative clusters

Table 3.2 Characteristics of some loss-of-offsite-power-event clusters that affect longer duration outages

Cluster	Characteristics
1	Sites with demonstrated high grid reliability and multiple sources of offsite power available through independent switchyard circuits and low severe-weather hazards or design features to limit loss of offsite power or hasten recovery from severe-weather events.
2	Sites with demonstrated high grid reliability and low severe-weather hazards or moderate severe-weather hazards with design features to limit loss of offsite power or hasten recovery from severe-weather events.
3	Sites located in moderate to high severe-weather hazard areas and with limited design features to preclude loss of offsite power or hasten recovery from severe-weather events.
4	Sites with known grid reliability problems and low to moderate severe-weather hazards or design features to limit loss of offsite power or hasten recovery from severe-weather events.
5	Sites located in a high severe-weather hazard area and without design features to preclude loss of offsite power or hasten recovery from severe-weather events.

4 RELIABILITY OF EMERGENCY AC POWER SUPPLIES

The emergency AC power system provides an alternate or backup power supply to the offsite power sources. Figure 4.1 is a simplified one line diagram of a typical emergency AC power system. If the offsite power system is lost, an undervoltage condition will exist on the safety buses, causing actuation of the emergency AC power system. The emergency AC power system provides sufficient functional capability and redundancy of the power requirements for the systems needed to mitigate the consequences of a design-basis accident. This typically includes a requirement to actuate emergency AC power supplies and make them available for loading within about 10 seconds after receiving an actuation signal. The emergency AC power system also meets the single-failure criterion when applied to design-basis accidents.

Emergency AC power is generally provided by diesel generator systems, although other sources such as gas turbine generators or hydroelectric power are used at some plants. Because of the preponderance of diesel generator usage, that power supply type will be the principal focus of emergency AC power system discussions in this report. Figure 4.2 identifies the typical subsystems and support systems that are needed for successful operation of the emergency diesel generator.

Emergency AC power systems typically consist of two diesel generators, either one of which is sufficient to meet AC power load requirements for a design-basis accident. This configuration has been designated by its success criterion: one out of two or more simply 1/2. In some cases, three or four or more diesel generators are used at single-unit sites, and in others, diesel generators are shared at multi-unit sites. These systems also can be described by their success criteria, or number of diesel generators required per number provided. However, for evaluating the station blackout issue, the success criterion will be defined as the number of diesel generators required to maintain a stable core cooling and decay heat removal condition with all offsite power sources unavailable.

The emergency AC power configurations that exist in the United States have been identified as follows:

(1) Emergency AC power supplies dedicated to one unit

1/2
1/3
1/4
2/4

(2) Emergency AC power supplies shared between two units

1/2
2/3
2/4
2/5
3/5

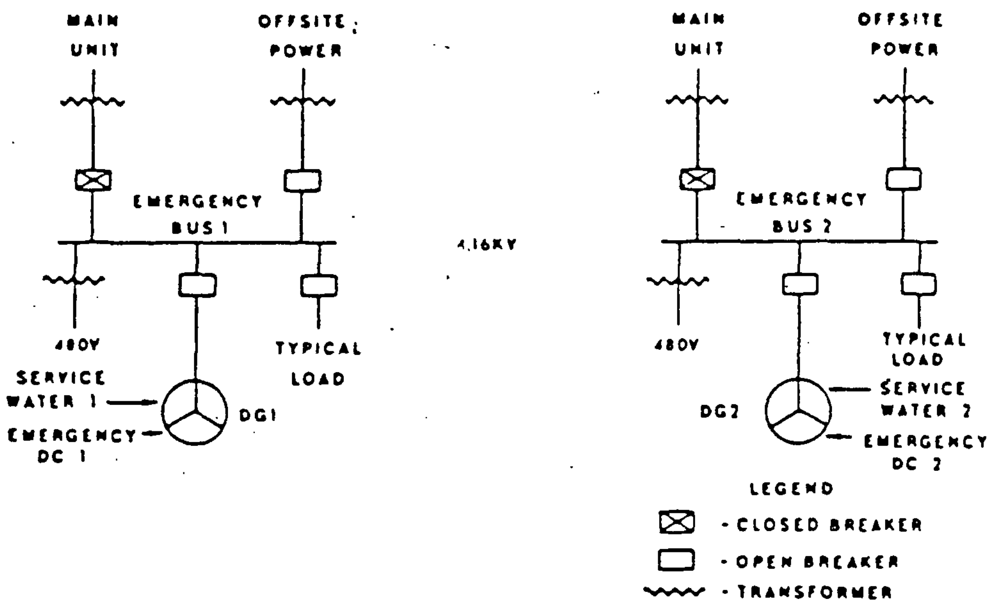


Figure 4.1 Simplified 1-of-2 onsite AC power distribution system

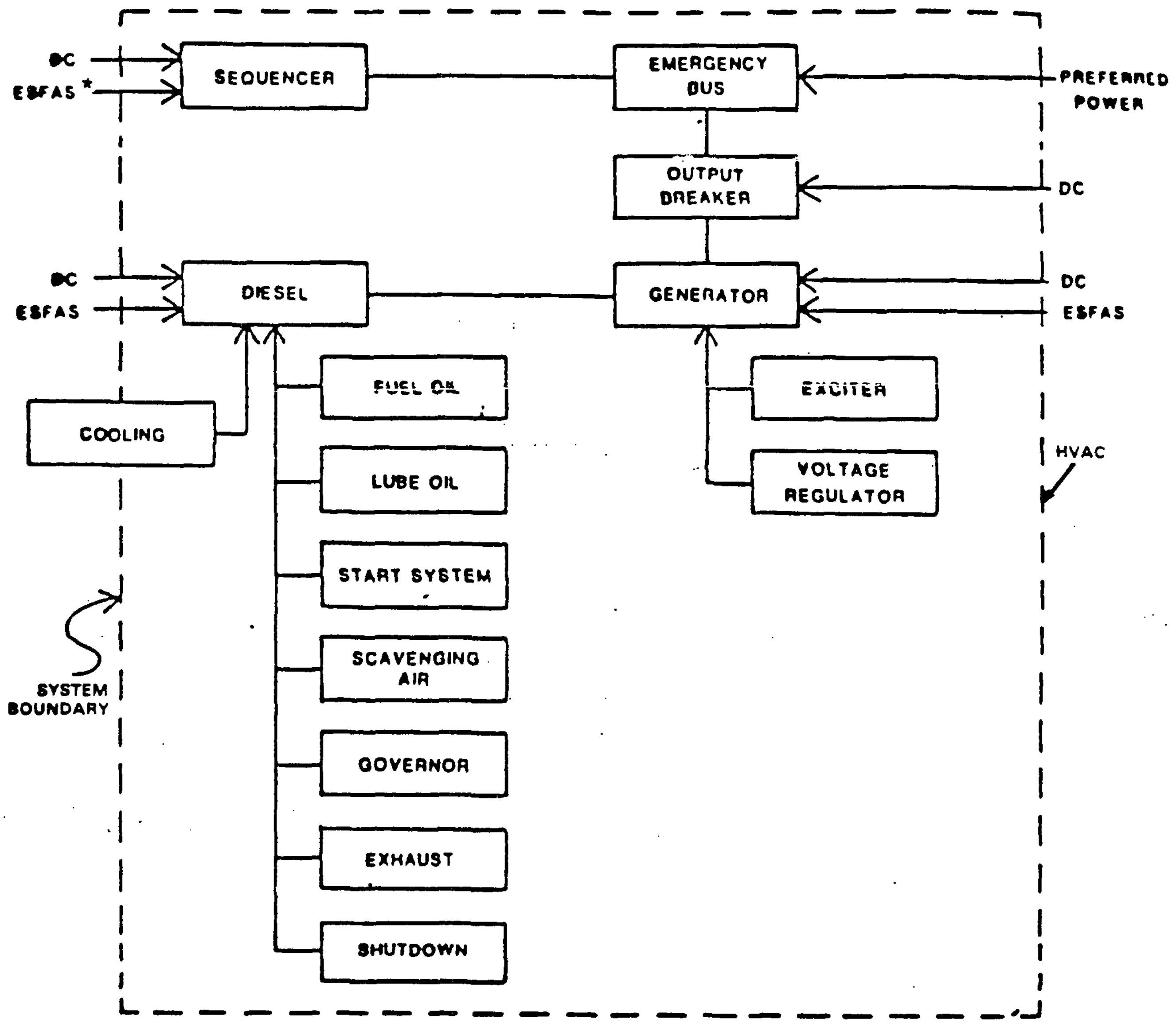


Figure 4.2 Onsite power system functional block diagram

*ESFAS = engineered safety feature actuation system

(3) Emergency AC power supplies shared between three units

3/8 [1/4 at one unit and 2/4 at 2 units with cross ties between 1 and 2 unit systems]

Although a closer review of emergency AC power supply requirements may produce some variations on these configurations, they represent a wide variety in system success criteria for reliability evaluations.

The design variability of emergency AC power systems is further complicated by dependencies on certain support systems that, by themselves, have a multitude of designs. These support systems include cooling systems (air or water), DC power, and heating, ventilation, and air conditioning (HVAC) systems. Moreover, maintenance and testing activities vary considerably, which can affect the reliability of the emergency AC power system.

Emergency AC power systems can be considered in two separate parts: power supplies and the power distribution system. In general it has been found that the individual components of the emergency AC power distribution system from the safety (switchgear) buses to the safety components are not significant contributors to the unavailability of AC power in regard to the station blackout issue. This statement is true because many independent, separate, and diverse distribution system components must fail to cause loss of all AC power to the safety systems. Although fires and earthquakes have the potential to cause such distribution system failures, these hazards have been studied as separate safety issues, and were not systematically assessed as part of the station blackout issue.

Substantial operating experience data were investigated to identify and estimate important reliability characteristics of emergency diesel generators. Initially, diesel generator reliability performance information was collected from 45 nuclear power plants with 86 diesel generators (NUREG/CR-2989). A summary of the emergency diesel generator statistical data collected is provided in Table 4.1a. In addition, information regarding diesel generator outages and downtime was obtained from responses to TMI Action Plan (NUREG-0737) items from licensees of plants with 58 diesel generators, and more than 1500 licensee event reports (LERs) covering 5 years from 1976 through 1980 were reviewed for failure information. Analysis of this operating experience showed that, on the average, diesel generators failed to start, load, or continue running approximately 2 times out of every 100 demands. It was also observed that, during the actual loss-of-offsite-power events through 1983, there were 19 instances in which one or more diesel generators failed, operated in a degraded condition, or were otherwise unavailable. During most of these events, the degraded diesel generators were able to meet minimum performance requirements and failed units were promptly restored to an operable condition. This information was supplemented with data collected from licensee responses to Generic Letter 84-15 (NUREG/CR-4347) for the years 1981 and 1982. A more recent EPRI study (Wyckoff, September 1986) has provided emergency diesel generator failure-rate data for the years 1983 through 1985. Emergency diesel generator failure statistics derived from the EPRI data are shown in Table 4.1b.

Table 4.1a Diesel generator start attempts and failures for tests and actual demands* from NUREG/CR-2989

Start attempt category	No. of demands	No. of failures	Failures per demand	No. of auto-start failures	Auto-start failures per demand	Unavailable	Unavailability
Test	13,665	253	0.019	55	0.004	---	0.006
Loss of offsite power**	100	5	0.05	3	0.03	3	0.03
All emergency demands	539	14	0.026	5	0.009	3	0.006

Failure to run: 2.4×10^{-3}/hr***

*Summarizing the responses to diesel generator reliability questionnaires based on 45 nuclear power plants, with 86 diesel generators, for operating years 1976 through 1980.

**Updated from data reported in NUREG/CR-2989.

***Based on 314 attempts at scheduled run time of 6 hours or more with 9 failures to run during these attempts.

Table 4.1b Diesel generator start attempts and failures for tests and actual demands from EPRI study (Wyckoff, Sept. 1986)

Start attempt category	No. of demands	No. of failures	Failure per demand	Unavailable
All	22,180	260*	0.012	---
Emergency	424	3	0.0071	
Loss of offsite power	41	1	0.024	1

Failure to run: 3.2×10^{-3}

*Includes 39 failures identified from LERs and/or categorized as non-failures in EPRI study (Wyckoff, Sept. 1986).

Figures 4.3a and 4.3b provide histograms of emergency diesel generator failures on demand for 1976 through 1982 and 1983 through 1985, respectively. Although the average failure on demand observed is about 2×10^{-2}, there is a significant spread from the highest to the lowest demand failure rate. The average failure rate and range have not changed substantially during this period. However, EPRI data show an average failure rate of 1.2×10^{-2} per demand. A review of the data has not identified any particular type of failure as the most dominant. At least in part, the reasons for this are (1) that there are several different types of diesel generators with different support and auxiliary system designs operating at nuclear power plants, and (2) that maintenance and test activities are not standardized within the nuclear industry. Figure 4.4 shows the percentage contribution of failure by subsystem. In general, sufficient information was not available to add high confidence to the correlation of root failure causes with specific design and operational factors. The data indicate that approximately 80% of the failures are the result of hardware-related problems and 20% are the result of human error.

These statements are not meant to imply that any one particular diesel generator is susceptible to all possible failure modes with equal importance. It is more likely that a few specific defects may exist, and if these are not discovered and corrected, failures may occur.

The failures observed can be classified into three general types:

(1) design and hardware failures related to mechanical integrity or various failure modes in the diesel generator subsystems, such as fuel, cooling, starting, and actuation

(2) operation and maintenance errors related to the correctness and adequacy of procedures or training, and human factors including the potential for errors of commission and omission

(3) failures that occur in support systems, or at interfaces with support systems and other systems, that can involve DC control power, service (or raw) water cooling, environmental control (air temperature and quality), and interface with the normal AC power system (undervoltage relays)

From 1976 through 1985 there were 145 instances in which multiple diesel generators were simultaneously failed, unavailable, or showed some degradation. There were 22 instances classified as common cause failures of two or more diesel generators (see Appendix B).

Multiple diesel generator failures can occur when a fault or degradation exists involving a common factor or dependency for two or more diesel generators. Multiple failures may also occur as a result of design and operating deficiencies similar to those previously mentioned; but in this case degradation or failure occurs concurrently in multiple diesel units. For instance, a defective crankshaft design may be such that mechanical failure is highly likely to occur after a certain amount of usage. If two or more diesel generators reach that usage level at nearly the same time, concurrent failures may result. As another example, defective maintenance procedures and training could result in human errors causing failure or simultaneous outages of two or more diesel units.

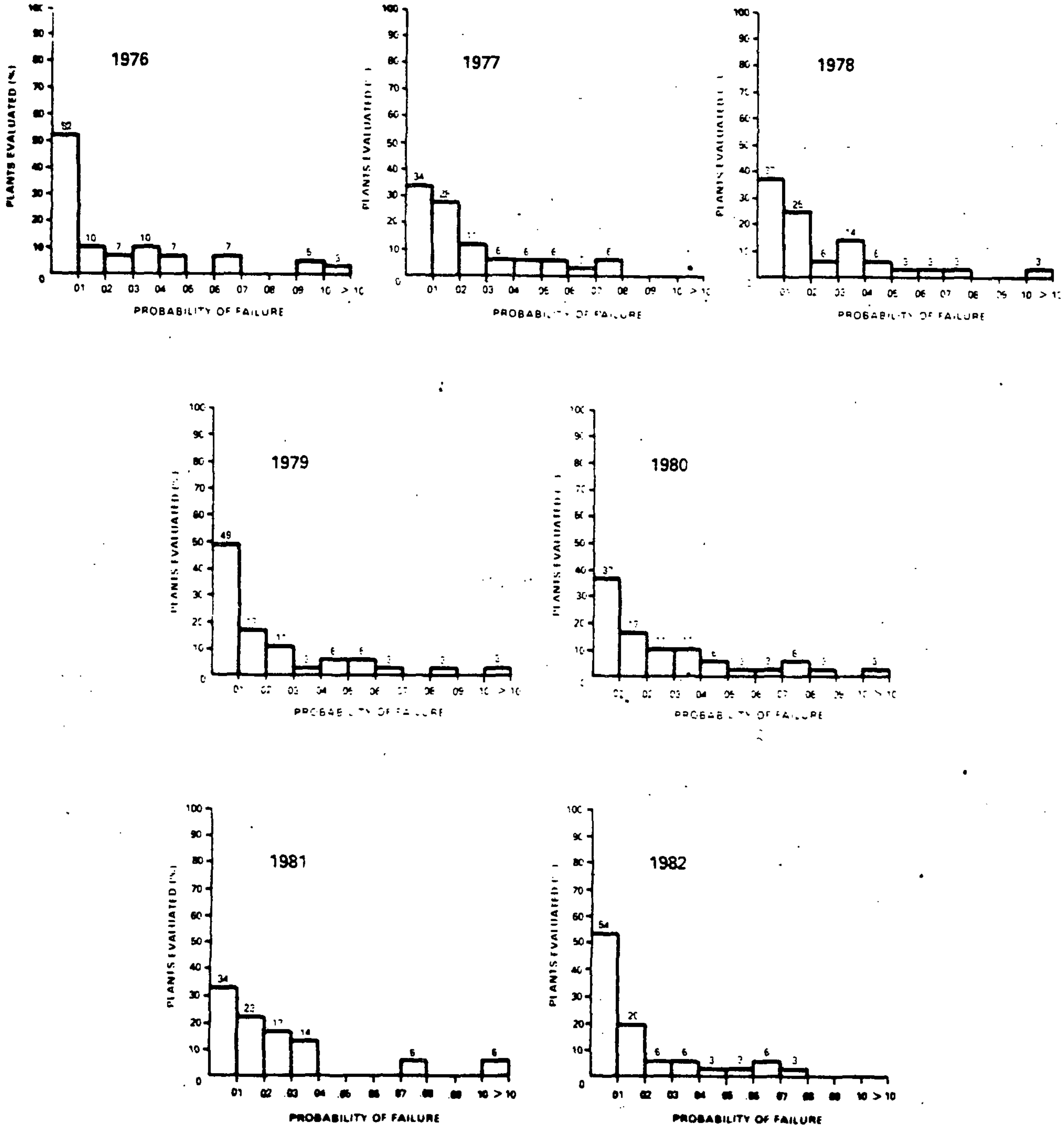


Figure 4.3a Histograms showing emergency diesel generator failure on demand for 1976 through 1982

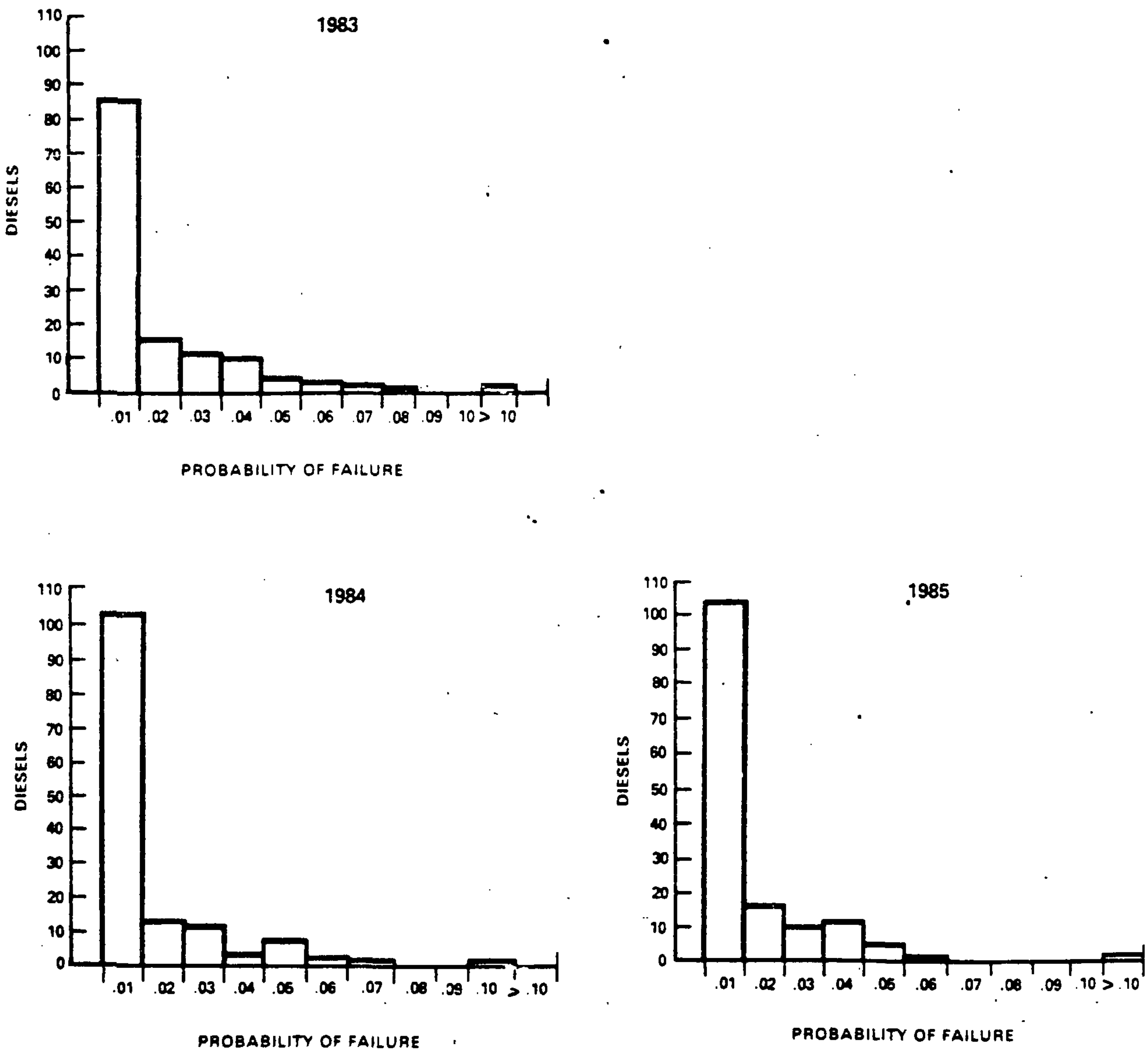


Figure 4.3b Histograms showing emergency diesel generator failure on demand for 1983 through 1985

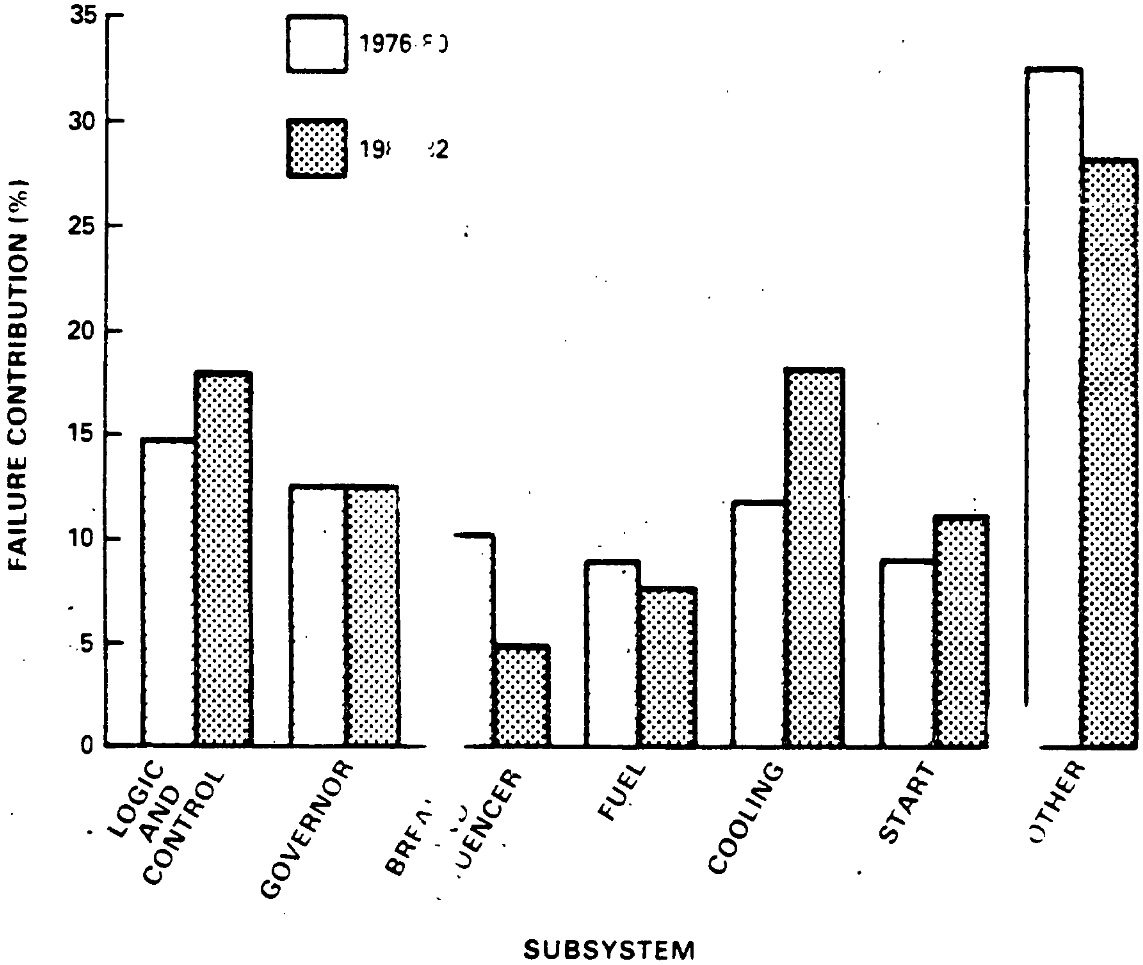


Figure 4.4 Failure contribution by diesel generator subsystem

Another type of common cause failure is related to the existence of single point vulnerabilities. Examples include a check valve in a header of a cooling water supply, the unrecognized dependence on an obscure single control circuit or element, and the use of common fuel supplies and containers.

Finally, common cause failures can be related to commonality of location with regard to environmental conditions for which adequate protection is not provided. These conditions can include fire, flood, dust, corrosive elements in the air, or temperature and humidity extremes.

In assessing the reliability of emergency AC power systems, consideration was given to the failure modes, causes, and failure rates derived from the operational data. Reliability analyses performed by Oak Ridge National Laboratory (ORNL) (NUREG/CR-2989) for 18 nuclear power plant AC power configurations and the plant-specific failure data were applied to derive typical system unavailability estimates. Figure 4.5 shows a histogram of the onsite AC power results for the 18 plants studied. The results of this work, summarized in Table 4.2, show the diesel generator configuration studied, the calculated range of unavailability on demand, and the dominant failure causes for each group analyzed. Not surprisingly, for the least redundant system configuration, the independent diesel generator failure likelihood is the most dominant failure factor. As system redundancy is increased, common cause failures become more important. Common cause failures involving hardware failure, human error, and dependent system failures were found to be important.

Although, for the most part, power supply outages resulting from testing and maintenance were not found to be large contributors to system unavailability, a few cases were identified in which extensive maintenance outages could cause significant system unavailability. The quality of test and maintenance procedures, however, can be an important factor affecting system reliability. Lower than average human-error-related diesel generator failures were observed when procedures were clearly written and had a sufficient level of detail, including complete check lists so operations personnel could verify that normal values were properly indicated after maintenance.

The impact of dependent systems (such as service water cooling and DC power) on the reliability of the emergency AC power system varies from plant to plant. The ORNL analyses did not go into detail on the reliability of those support systems. However, failures of dependent systems that affect the emergency AC power system seem to be dominated by single-point passive failures or human error. An unreliable support system can cause an unreliable AC power system. Because these support and auxiliary systems also tend to be important for the operation of decay heat removal systems--and to some extent for the supply of normal AC power from the offsite power sources--single-point vulnerabilities and human error failures in these systems have added importance.

Another potentially important reliability parameter involves the likelihood of a failed power supply (diesel) being restored to an operable state during a loss-of-AC-power transient. A histogram based on emergency diesel generator repair times following a failure is provided in Figure 4.6. The median repair time is approximately 8 hours. These data represent an aggragate for all types of failure modes, and, for the most part, they represent repair times during non-emergencies. Primarily these failures occurred during plant operation, but some occurred during plant shutdown. They do not include autostart failures.

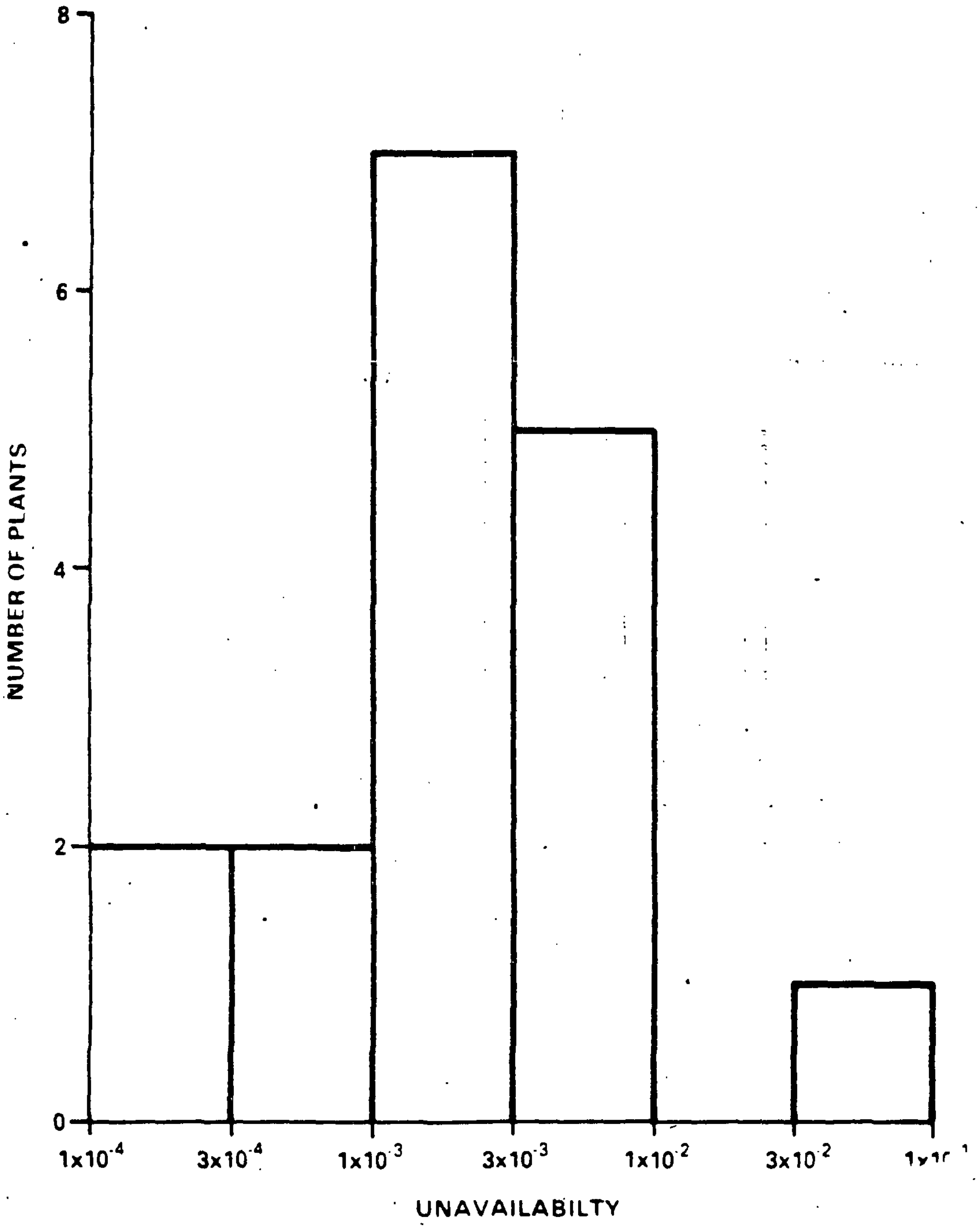


Figure 4.5 Onsite AC system unavailability for 18 plants studied in NUREG/CR-2989

Table 4.2 Results of onsite power system reliability analysis reported in NUREG/CR-2989

Diesel generator configuration	Range of system unavailability per demand	Dominant failure causes
2 of 3	4.2×10^{-3} to 4.8×10^{-2}	Independent diesel failure; human error CCF*.
1 of 2	1.1×10^{-3} to 6.8×10^{-3}	Independent diesel failure; human error CCF. T&M** outages.
2 of 4	3.7×10^{-4} to 1.7×10^{-3}	Human error and hardware CCF.
1 of 3	1.8×10^{-4} to 7.2×10^{-4}	Human error, hardware, and service water CCF, independent diesel failure; DC power CCF.
2 of 5	1.4×10^{-4} to 2.5×10^{-3}	Human error, hardware, service water, and DC power CCF.

*CCF = common cause failures
**T&M = test and maintenance

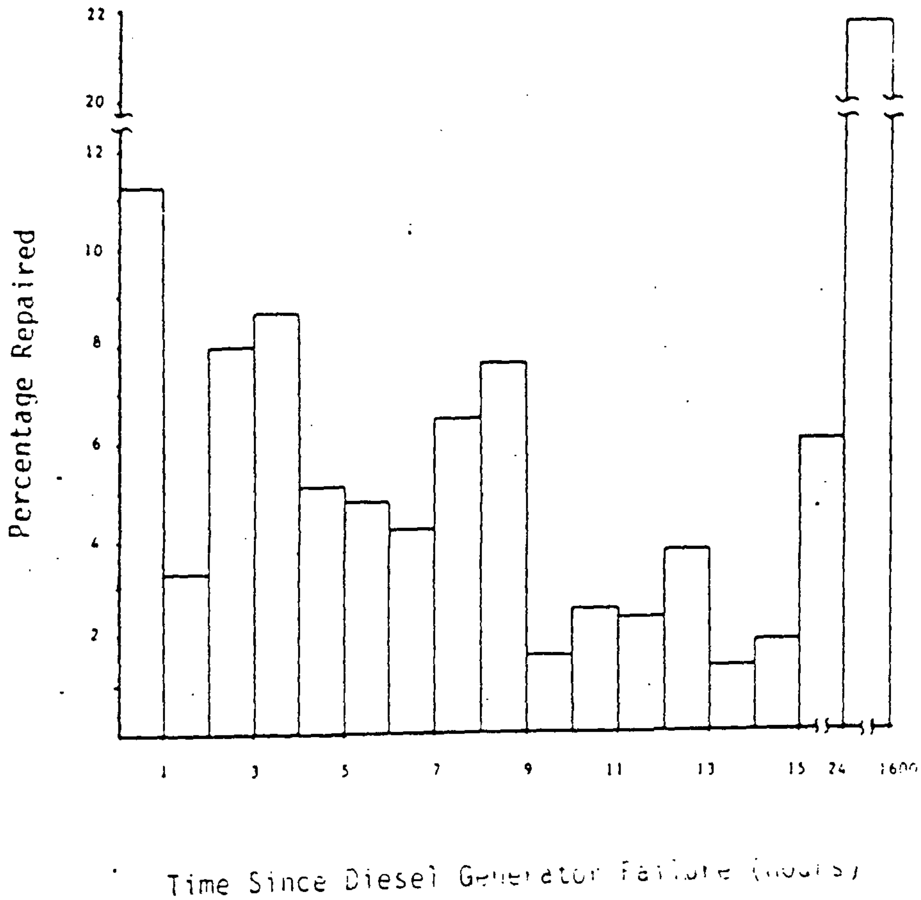


Figure 4.6 Percentage of emergency diesel generator failures repaired vs. time since failure

Source: NUREG/CR-2989

It is difficult to determine whether these data overestimate or underestimate the diesel generator repair time anticipated during an emergency. There are reasons to believe that these data overestimate the time required to repair a failed diesel generator during a station blackout. Because the typical limiting condition for operation (LCO) for a single diesel generator out of service is 72 hours or more, there is no urgency to restore a failed diesel generator as quickly as would be the case during a loss of all AC power. In addition, the LCO may not have been in force if the plant were shut down when a test failure occurred, which also would have lessened the urgency for repair. Moreover, if a failure did occur when alternate AC power sources were available, it might be seen as an opportune time to perform other routine maintenance on the failed diesel generator.

Conversely, the repair time could be underestimated by virtue of the confusion that could occur during a station blackout event. Under stressful conditions, human error is usually higher than it is under normal conditions. The diesel failure problem would have to be diagnosed, needed equipment would have to be obtained, and correct repair procedures would have to be followed; all this would have to be done under time constraints and pressure, without AC power available. Also, maintenance and operations personnel resources would be divided between activities for restoring both offsite and emergency power supplies.

In addition to conducting the plant-specific analyses, ORNL constructed generic models for different emergency AC power configurations. These generic models were used to estimate system reliability as a function of the important characteristics identified in the plant-specific analyses. Typical system dependencies and nominal values for common cause failures and procedural errors were assumed in the models, and sensitivity analyses were performed to determine the importance of all the factors considered. Overall, the most important factors tended to be system redundancy and the reliability of emergency diesel generators on demand. Not surprisingly, it was found that common cause failure is most important in highly redundant system configurations with highly reliable (for independent failure causes) diesel generators.

Based on these considerations, the NRC staff performed additional analyses of emergency AC power system reliability to extend the quantitative results and further explore the sensitivities. Figure 4.7 shows the effect of varying emergency diesel generator reliability on emergency AC power system reliability for several configurations, both with and without common cause failure. The sensitivities of system reliability estimates on variations in diesel generator running reliability are shown in Figure 4.8. Additional results, parametric analyses, and details of the analytical model are provided in Appendix B.

Thus, on the basis of a review of operating experience and reliability analyses, the following factors have been identified as being the largest contributors to AC power system unavailability:

(1) the configuration of the diesel generators in terms of the number available and the number required for shutdown cooling

(2) the reliability of diesel generators or other power sources used in the emergency AC power system

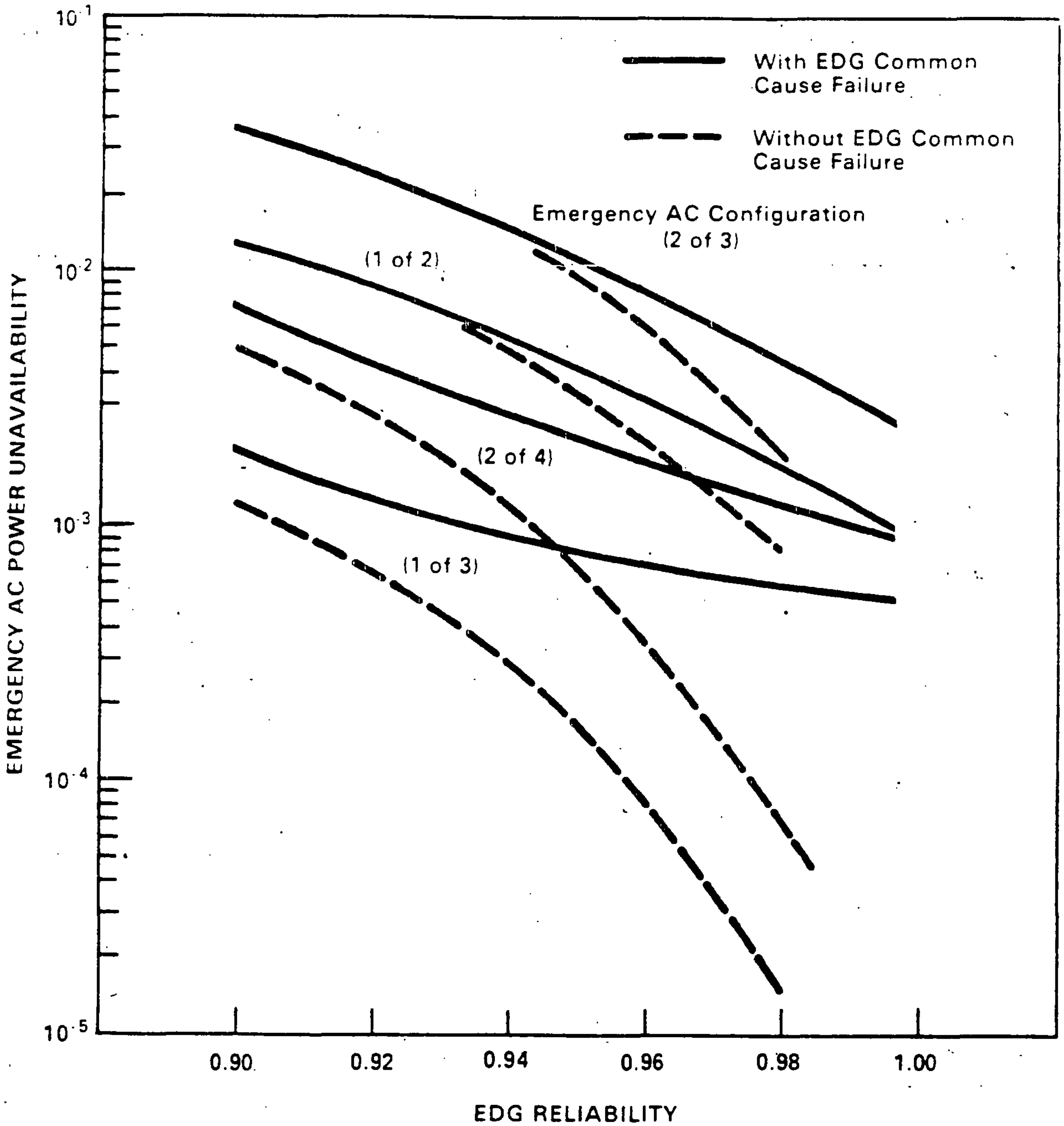


Figure 4.7 Generic emergency AC power unavailability as a function of emergency diesel generator (EDG) reliability

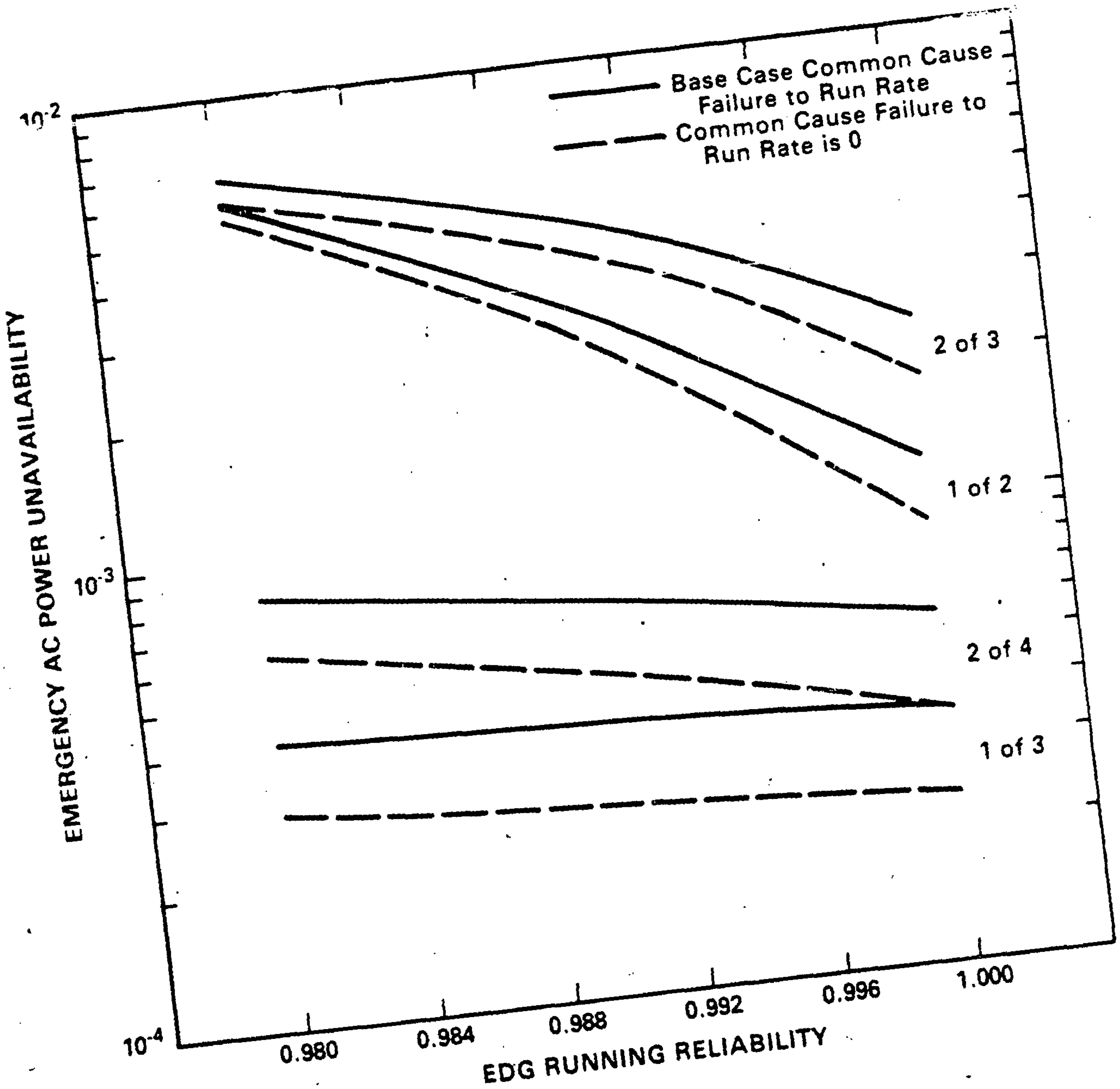


Figure 4.8 Generic emergency AC power unavailability as a function of individual diesel generator running reliability

(3) the dependence of the AC power system on support or auxiliary systems used for actuation, control, or cooling

(4) the vulnerability of the AC power system to common cause failure as a result of various design, human error, and internal or external environmental hazards

In general, it has been observed that problems with onsite emergency AC power systems are very plant-specific, and improvement in system reliability would have to be developed on a plant-by-plant basis.

5 STATION BLACKOUT FREQUENCY AND DURATION

There have been several incidents at nuclear power plants that could be classified as precursors to station blackout. In fact, there have been a few cases in which loss of offsite and emergency AC power supplies occurred simultaneously. However, none of these events progressed to be a significant safety concern. Many of these incidents occurred when plants were shut down or during refueling, when station blackout concerns are much reduced and the LCO--in terms of numbers of offsite and emergency AC power supplies available--are reduced.

The lack of a significant number of station blackout events is not surprising when one considers past frequency of loss-of-offsite-power events and the reliability record of emergency AC power systems. As a result, it has been necessary to estimate station blackout frequency by combining loss-of-offsite-power-event frequency and duration correlations with the emergency AC power reliability models. (Appendix B describes the methods used to derive station blackout frequency and duration estimates.)

Figures 5.1 through 5.3 give the results of sensitivity analyses performed to determine the effect of design, location, and emergency AC power supplies reliability. Specifically, Figure 5.1 shows the effect of site location and offsite power system design as represented by offsite power clusters 1, 2, 3, and 4. (These clusters are defined in Section 3 and Appendix A.) These clusters were combined with a typical, two-diesel-generator, emergency AC power system with a diesel generator reliability of 0.975. Cluster 2 is a close representation of the average of nuclear operating experience with regard to the frequency and duration of loss-of-offsite-power events. Cluster 4 represents sites with relatively high severe-weather hazards and susceptibility to failure from those hazards. Cluster 3 has slightly lower severe-weather hazards than cluster 4.

Cluster 1 represents the combination of the more reliable offsite power design features and sites with low severe-weather hazards or low susceptibility to severe-weather hazards. The estimated frequency of longer duration station blackouts is dependent on the likelihood of the more damaging and extensive losses of offsite power for which severe-weather hazards have been identified as a principal contributor. (Note: Seismically induced loss of offsite power has not been included, but could be accounted for through a hazard evaluation and fragility analysis; this consideration is discussed in Section 9.)

Figure 5.2 shows the effect of variations in emergency diesel generator reliability for the typical offsite system (cluster 2) and emergency AC power system (1/2 configuration). The largest change in frequency per percentile change in diesel generator reliability is obtained when reliability levels are lowest (0.9). This is somewhat of an artifact of the model in which common cause failure rates are kept constant. If there were no common cause failure contributions, or if common cause failure were correlated with the independent failure rate of diesel generators (and it may be), the frequency reduction could be proportional to the square of the percentile change in diesel reliability for the configuration analyzed.

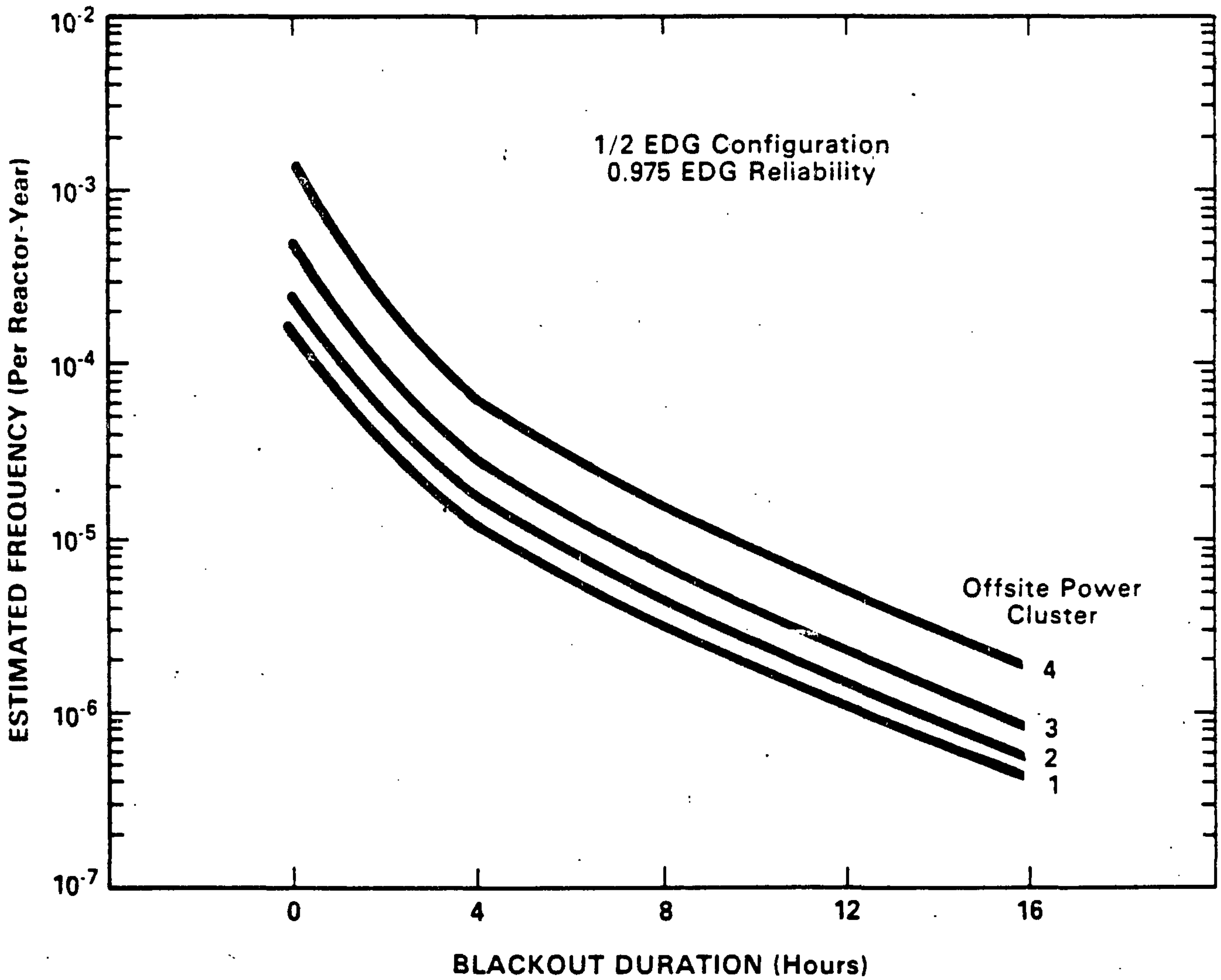


Figure 5.1 Estimated frequency of station blackout exceeding specified durations for several representative offsite power clusters

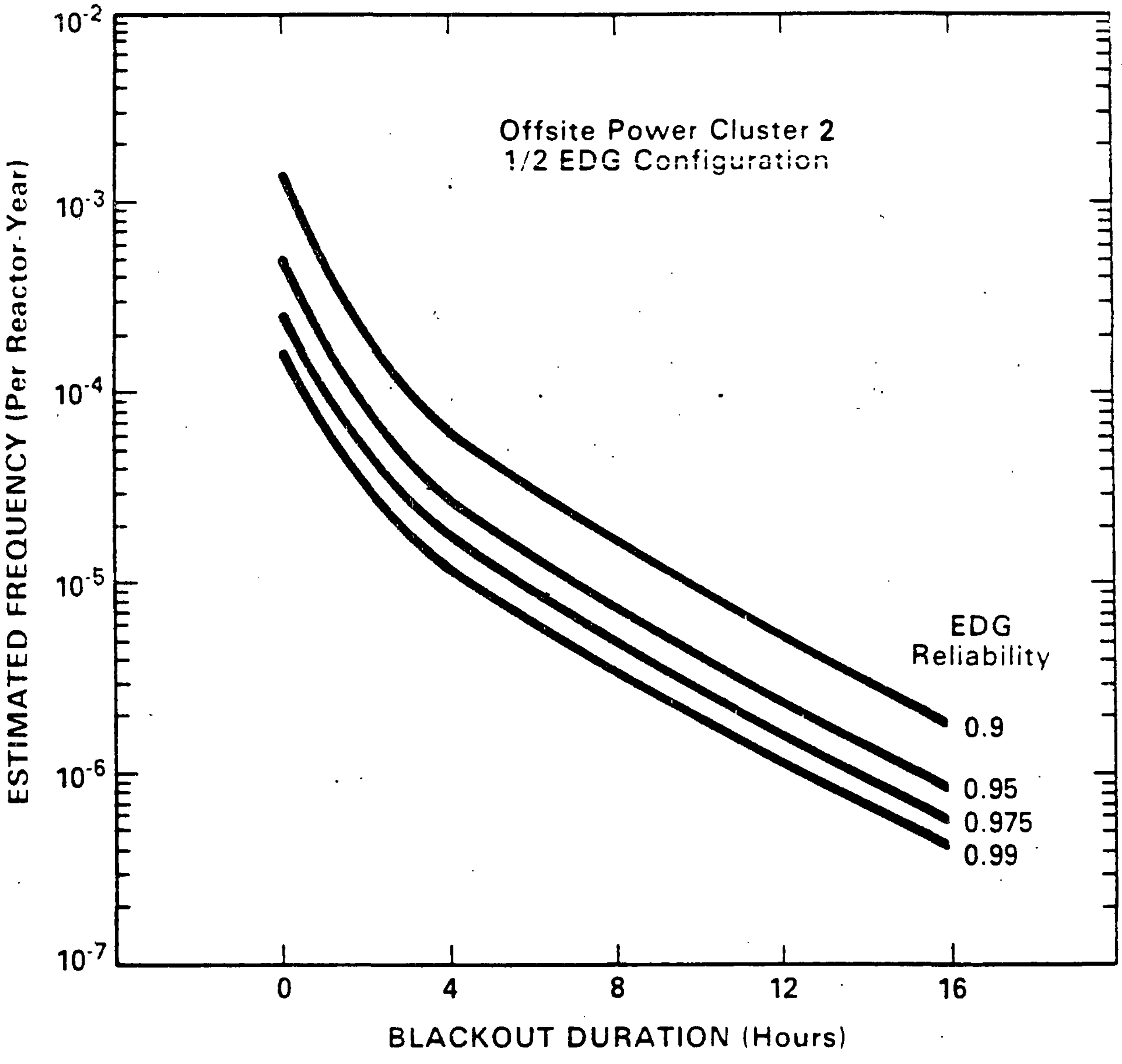


Figure 5.2 Estimated frequency of station blackout exceeding specified durations for several EDG reliability levels

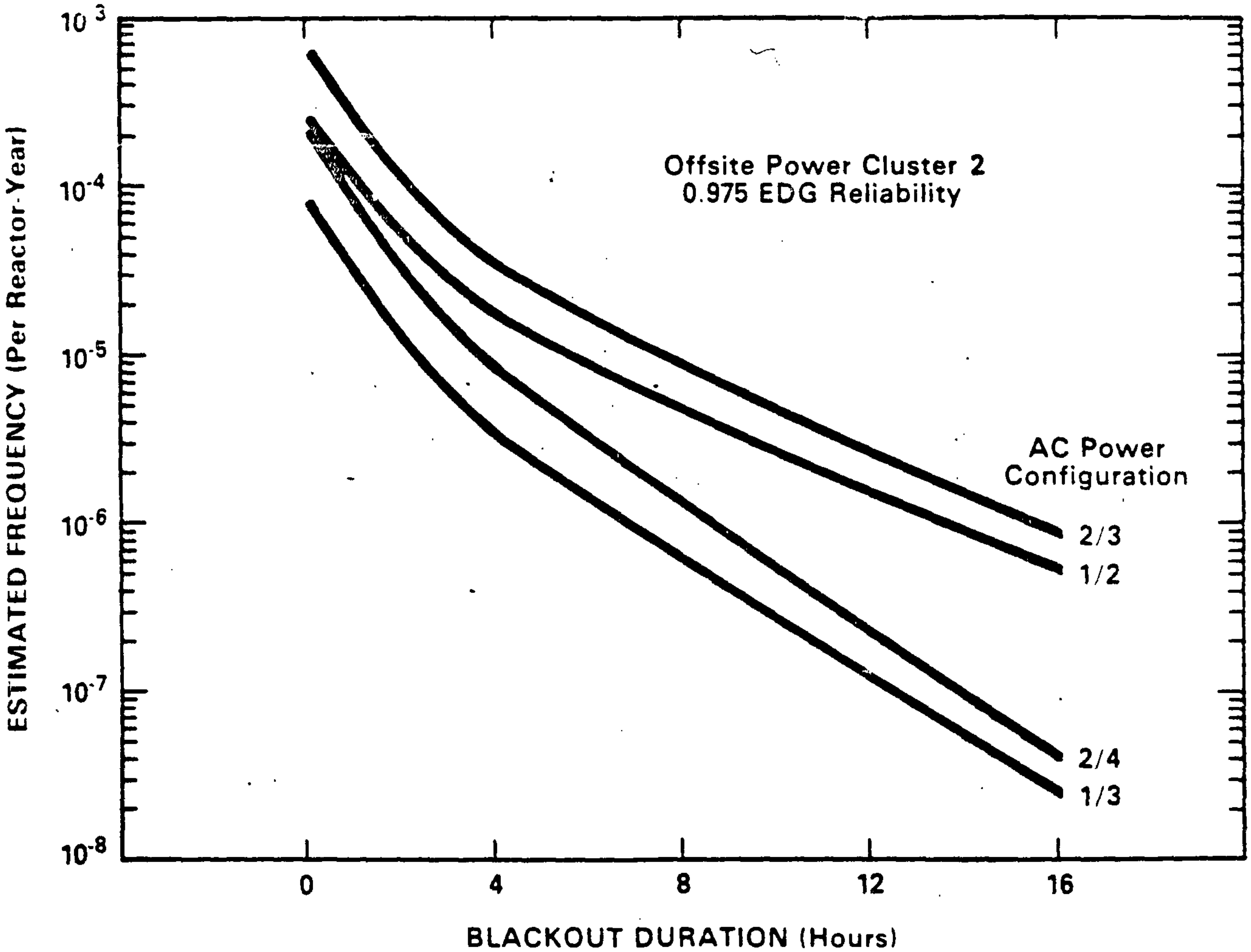


Figure 5.3 Estimated frequency of station blackout exceeding specified durations for several emergency AC power configurations

Figure 5.3 shows the effect of emergency AC power configuration and success criteria on station blackout frequency, using a diesel generator reliability of 0.975 and a generic common cause failure rate. Again the effect of common cause failures on system reliability is to reduce the difference between the four configurations that would be expected from simple redundancy considerations.

The results of the station blackout sensitivity analyses show that there is a potential for wide variation in frequency and duration, depending on location, design, and reliability. (Additional results are in Appendix B.)

6 ABILITY TO COPE WITH A STATION BLACKOUT

Station blackout is a serious concern because it has a large effect on the availability of systems for removing decay heat. In both PWRs and BWRs, a substantial number of systems normally used to cool the reactor are lost when AC power is not available. A loss of offsite power will usually result in the unavailability of the power conversion system and, in particular, an inability to operate the main feedwater system. Power to reactor coolant system recirculation pumps will also be lost, requiring that natural circulation be used for cooling to shutdown conditions. When the loss of offsite power is compounded by a loss of the emergency AC power supplies, reactor core cooling and decay heat removal must be accomplished by a limited set of systems that are steam driven, passive, or have other dedicated (or alternate) sources of power. Unless special provisions are made, the plant will have to be maintained in a "hot" mode (hot shutdown or possibly hot standby) until AC power is restored. Table 6.1 lists which functions and systems for PWRs and BWRs would be lost and which would remain available during a station blackout event. Decay heat can be removed successfully, using the AC-independent systems identified, for a limited time, depending on functional capabilities, capacities, and procedural adequacy.

For PWRs, decay heat can be removed by use of a steam-driven or dedicated diesel-driven train of the auxiliary feedwater system (AFWS). Decay heat would be rejected to the environment by the atmospheric dump valves (ADVs) or, if necessary, by the steam generator relief valves. Because residual heat removal systems, reactor coolant makeup systems, and systems to control reactivity through boration would be inoperable, the plant must be maintained in a hot condition. The plant's operating state (primary coolant pressure and temperature) would be maintained by manual operation of the AFWS and atmospheric steam dump valves. With primary coolant pumps unavailable, reactor core cooling would be achieved through natural circulation.

If the AFWS can remain operable, and if primary coolant inventory can be maintained at a level adequate to maintain the core cooling/heat transport loop to the steam generators, a PWR should be able to stay in this mode of decay heat removal for a substantial period of time. The amount of time that decay heat removal can be maintained in a PWR is generally limited by primary pressure boundary leakage and the capacity of certain support or auxiliary systems. The sources of potential leakage include reactor coolant pump seals, unisolated letdown lines, and a stuck-open pilot-operated relief valve (PORV). With provisions for manual isolation of letdown lines and reduced frequency of PORV demands, the reactor coolant pump seal leakage rate is considered to be a potentially limiting factor for some designs. If the leakage rate is low (on the order of several gallons per minute) this concern is negligible. However, if seal leakage is on the order of 100 gpm or more, reactor coolant system inventory depletion will be a factor limiting decay heat removal for an extended period of time.

Table 6.1 Effects of station blackout on plant decay heat removal functions

Plant Type	Functions (systems) remaining	Functions (systems) lost
PWR	Shutdown heat removal [steam-driven auxiliary feedwater system (AFWS), atmospheric dump valves]	Shutdown heat removal (motor-driven AFWS) Long-term heat removal [residual heat removal (RHR)]
	Instrumentation and control (DC power/converted AC power, compressed air reservoir)	Reactivity control (chemical) volume and control system) Reactor coolant system (RCS) makeup [high-pressure injection system] Pressure and temperature control (pressurizer heaters/spray and pilot-operated relief valves) Support systems [service/component cooling water systems; heating, ventilation, and air conditioning (HVAC); station air compressors]
BWR, 2/3	Shutdown heat removal (isolation condenser, fire water system)	Long-term heat removal (RHR) RCS makeup (low-pressure core spray system, feedwater coolant injection system)
	Instrumentation and control (DC power/converted AC power, compressed air reservoirs)	Support systems (service/component cooling water systems, HVAC, station air compressors)
BWR, 4-6	Shutdown heat removal and RCS makeup (high-pressure coolant injection or high-pressure core spray/reactor core isolation cooling systems)	Long-term heat removal (shutdown cooling system, low-pressure coolant recirculation system, suppression pool cooling system)
	Instrumentation and control (DC power/converted AC power, compressed air reservoirs)	Support systems (service/component cooling water systems, HVAC, station air compressors)

Natural circulation cooldown in PWRs has been successfully demonstrated by actual operating experience. The process becomes more difficult with AC power unavailable because reactor coolant makeup systems, to accommodate system shrinkage and pressurizer heaters or sprays to help control primary system coolant conditions, are inoperable. Nevertheless, analytical evaluations (Fletcher, 1981) and experimental observations (Adams, et al. 1983) show that decay heat removal can be achieved with the operational limitations associated with a station blackout. In fact, core cooling is expected to preclude core melting even with significant voiding in the primary coolant system if the steam generator is maintained as a heat sink.

To assess station blackout, BWRs have been divided into two functionally different classes: (1) those that use an isolation condenser cooling system for decay heat removal and do not have a makeup capability independent of AC power (BWR-2 and -3 designs), and (2) those with a reactor core isolation cooling (RCIC) system and either a steam-turbine-driven high-pressure coolant injection (HPCI) system or high-pressure core spray (HPCS) system with a dedicated diesel, any of which is adequate to remove decay heat from the core and control water inventory conditions in the reactor vessel (BWR-4, -5, and -6 designs). Because BWRs are designed as natural circulation reactors, at least at reduced power levels, the loss of reactor coolant recirculation poses no special consideration. Moreover, reactivity control during cooldown is adequately maintained by control rod insertion, an action that would occur automatically on loss of all AC power.

The isolation condenser BWR has functional characteristics somewhat like that of a PWR during a station blackout in that normal makeup to the reactor coolant system is lost along with the residual heat removal (RHR) system. The isolation condenser is essentially a passive system that is actuated by opening a condensate return valve; it transfers decay heat by natural circulation. The shell side of the condenser is supplied with water from a diesel-driven pump. However, replenishment of the existing reservoir of water in the isolation condenser is not required until 1 or 2 hours after actuation. It may also be possible to remove decay heat from this class of BWRs by depressurizing the primary system and using a special connection for a fire water pump to provide reactor coolant makeup. This alternative would require much greater operator involvement. Some BWR-3 designs have added an RCIC system, giving makeup capability to the AC-power-independent decay heat removal capability of the isolation condenser cooling system.

A large source of uncontrolled primary coolant leakage will limit the time the isolation condenser cooling system can be effective. If no source of makeup is provided, eventually enough inventory will be lost to uncover the core. A stuck-open relief valve or the reactor coolant recirculation pump seal are potential sources of such leakage. When isolation condenser cooling has been established, the need to maintain the operability of such auxiliary and support systems as DC power and compressed air is less for this type of BWR than it is for the PWR. However, these systems would eventually be needed to recover from the transient.

BWRs with RCIC and HPCI or HPCS can establish decay heat removal by discharging steam to the suppression pool through relief valves and by making up lost coolant to the reactor vessel. In these BWR designs, decay heat is not removed to the environment, but is stored in the suppression pool. For this type of BWR design, long-term heat removal in the form of suppression pool cooling or residual heat removal, using low-pressure coolant injection and recirculation heat transport

loops, is lost during a station blackout. The time that the plant can be maintained in a safe condition without AC power recovery is determined, in part, by the maximum suppression pool temperature for which successful operation of decay heat removal systems can be ensured both during a station blackout event and when AC power is recovered. At high suppression pool temperatures (around 200°F), unstable condensation loads may cause loss of containment suppression pool integrity. Another suppression pool temperature limitation to be considered is the qualification temperature on the RCIC or HPCI pumps to be used during recirculation. Suppression pool temperatures may also be limited by net positive suction head (NPSH) requirements for pumps in systems required to effect recovery once AC power is restored.

In general, all light-water reactor (LWR) designs include the ability to remove decay heat for some period of time. The time depends on the capabilities and capacities of support systems, such as the quantity and availability of water required for decay heat rejection, the capacity of DC power supplies and compressed air reservoirs, and the potential degradation of components as a result of environmental conditions that arise when heating, ventilation, and air conditioning (HVAC) systems are not operating. System capabilities and capacities are normally set so the system can provide its safety function during the spectrum of design-basis accidents and anticipated operational transients, which does not include station blackout.

Perhaps the most important support system for both PWRs and BWRs is the DC power supply. During a station blackout, unless special emergency systems are provided, battery charging capability is lost. Therefore, the capability of the DC system to provide power needed for instrumentation and control can be a significant time constraint on the ability of a plant to cope with a station blackout. DC power systems are generally designed for a certain capacity in the event of a design-basis accident with battery charging unavailable. However, the system loads required for decay heat removal during a total loss of AC power are somewhat less than the expected design-basis accident loads on the DC power system. Therefore, most DC power systems in operation today have the capacity to last longer during a station blackout than they would be expected to last during a design-basis accident.

Another important factor in regard to decay heat removal during station blackout is the capacity of the condensate storage tank. Normally, this tank contains a sufficient amount of water to cool the reactor until the RHR system can be placed in operation. Because the RHR system is not available when all AC power is lost, the ability to cope with station blackout is a function of the condensate storage tank capacity. The ability to provide makeup to the condensate storage tank with systems and/or components that are independent of station AC power would extend this potentially limiting factor.

Also, during a station blackout, there may be need to operate some pneumatic valves, such as a steam dump valve. Because AC power is not available, the station air compressors will be lost. For this reason, local air reservoirs are normally provided to permit the valves to be operated for a limited number of cycles. After the air supply is exhausted, these valves may have to be operated manually by the operations staff, or additional portable air tanks would have to be connected.

During a station blackout, normal plant HVAC would be unavailable. The equipment needed to operate during a station blackout and that required for recovery from a station blackout would have to operate in environmental conditions (e.g., temperature, pressure, humidity) that could occur as a result of the blackout. Otherwise, failures of necessary equipment could lead to loss of core cooling and decay heat removal during the blackout or failure to recover from the event when AC power is restored. The instrumentation and control elements of components required during station blackout are the most likely to be impacted by adverse environments. However, only limited equipment in the control room would have to be operable, thus limiting equipment-generated heat loads in that location. The same would be true for equipment in auxiliary buildings and inside containment, although sensible heat from pre-existing sources could be considerable. For control rooms and auxiliary buildings, opening doors should allow enough heat to escape to maintain equipment in an acceptable operating environment. Temperature-sensitive equipment located in normally enclosed cabinets that rely on HVAC systems to remove heat generated during normal operation could be subject to failure or degradation unless ventilation is provided. Most equipment in containment is designed to function in the more limiting environment associated with a design-basis loss-of-coolant accident, and therefore, could be expected to function during a station blackout.

Table 6.2 summarizes the design-related factors that have been identified as potentially limiting the capability of LWRs to cope with a station blackout.

Actions necessary to operate systems that are needed to establish and maintain decay heat removal and fully recover from a station blackout would not be routine. The operator would have somewhat less information and operational flexibility than is normally available during most other transients requiring reactor cooldown. On the other hand, the loss of all AC power is an easily diagnosed occurrence, although it is not always easily corrected.

Operational staff activities would have to be directed at both reactor decay heat removal requirements and the restoration of AC power. These activities would include manual operations within the control room to control the rate of core decay heat removal and special operations outside the control room. The latter would include repairing failed components, isolating sources of reactor coolant leakage, conserving DC power through load stripping, making available alternate makeup water supplies, hooking up compressed air bottles, and possibly starting local manual operation of some components. The success of these activities would require preplanning, training, and procedures. In addition, adequate lighting and communication would be required. Where local access is necessary, security and working environment (pressure, temperature, humidity, and radiation) could be limiting factors.

In PWRs, operators must control the rate at which the AFWS removes heat from the steam generators to maintain the proper pressure and temperature balance within the primary coolant system. This balance then allows adequate natural circulation and the maintenance of adequate water level in the pressurizer. Although analytical and experimental evidence suggests that natural circulation and adequate decay heat removal can be maintained when pressurizer level is lost (and, in fact, when a two-phase flow mixture exists in the reactor coolant system up to the point the reactor core is uncovered), these conditions would complicate the recovery process and add to the difficulty of operator recovery actions.

Table 6.2 Possible factors limiting the ability to cope with a station blackout event

Limiting factor	Type of plant		
	PWR	BWR 2/3	BWR 4/5/6
RCS[1] pump seal leakage	X	X	
RCS letdown/makeup and water chemistry control lines	X	X	
Stuck-open relief valve	X	X	
DC battery capacity (instrumentation and control)	X	X	X
Compressed air (valve control)	X	X	X
Decay heat removal water supply (condensate, firewater)	X	X	X
Operating environment (temperature)			
Control room (instrumentation and control)	X	X	X
Containment			X (suppression pool, wetwell, drywell)
Auxiliary building	X (AFWS[2]/room)		X (HPCI[3]/RCIC[4] room)

[1]RCS = reactor coolant system.
[2]AFWS = auxiliary feedwater system.
[3]HPCI = high-pressure coolant injection.
[4]RCIC = reactor core isolation cooling.

In BWRs, the isolation condenser appears to need less operator attention. However, operators would have to ensure that automatic depressurization does not occur and that the makeup system to the isolation condenser is operating properly within approximately 2 hours of the loss of AC power. In BWRs with HPCI or HPCS and RCIC, the operator must control pressure and the level of reactor coolant in the vessel. This requires actuation of makeup and relief systems.

In all LWRs, operators would have to be prepared to deal with the effects of the loss and restoration of AC power on plant control and safety system set points to limit additional transient complications and ensure operability of AC-powered cooling systems.

7 ACCIDENT SEQUENCE ANALYSES

Accident sequence analyses have been performed to determine the accident progression characteristics (Fletcher, 1981; NUREG/CR-1988; Schultz and Wagoner, 1982; and NUREG/CR-2182) and likelihood (NUREG/CR-3226) of a station blackout. Using fault trees and event trees, these analyses have identified functional and system failure characteristics of accident sequences. Reactor coolant system transient response analyses were used (1) to determine the capability of a plant to cope with station blackout and (2) for potentially important functional failures during a station blackout, to estimate how much time would be available for AC power recovery before core damage and core melt.

Considering the decay heat removal system capability requirements and the associated systems' reliability, failure modes, and failure causes, three phases of a station blackout transient were identified. The first phase includes the need for promptly actuating decay heat removal systems and the potential for a station blackout induced loss-of-coolant accident (LOCA), either of which can result in a loss of core cooling within 1½ to 2 hours. The second phase lasts up to approximately 8 to 12 hours and includes operational limitations in the capability of continued decay heat removal considering limited capacities (such as DC power, condensate storage tank) or interactive failure [for example, high temperature effects due to loss of heating, ventilation, and air conditioning (HVAC)], and the potential for reactor coolant loss (such as, through pump seal leakage). During this period, the running reliability of the system is less important than the successful initial actuation of the AC-independent decay heat removal systems. The third phase involves the need to eventually recover AC power and establish a stable, controllable mode of decay heat removal.

As discussed above, considering the systems and functions available for the different PWR and BWR designs resulted in the development of three event trees for the identification of station blackout accident sequences. Figure 7.1 shows the event tree for PWRs; Figure 7.2 shows it for BWRs that use an isolation condenser; and Figure 7.3 for BWRs that have AC-independent makeup systems [reactor or core isolation cooling (RCIC), high-pressure core spray (HPCS), and high-pressure coolant injection (HPCI)]. The event trees are characterized not only by the systemic and functional considerations important to station blackout accident sequences, but also by the phases of the transient that would affect the plant response and system operability for station blackouts of various durations. The event trees show the loss of all AC power as the initiating event and proceed through decay heat removal, reactor coolant inventory (integrity), and restoration of AC power to enable operation of the normal decay heat removal and makeup systems. The accident sequence logic is similar for PWRs and those isolation-condenser BWRs that do not have the capability to make up lost reactor coolant during a station blackout. These plants are susceptible to degraded core cooling as a result of relatively small losses of reactor coolant. The accident sequence logic is somewhat different for BWRs with reactor coolant makeup available during a station blackout. Most losses of reactor coolant caused by station blackout can be accommodated by the available reactor

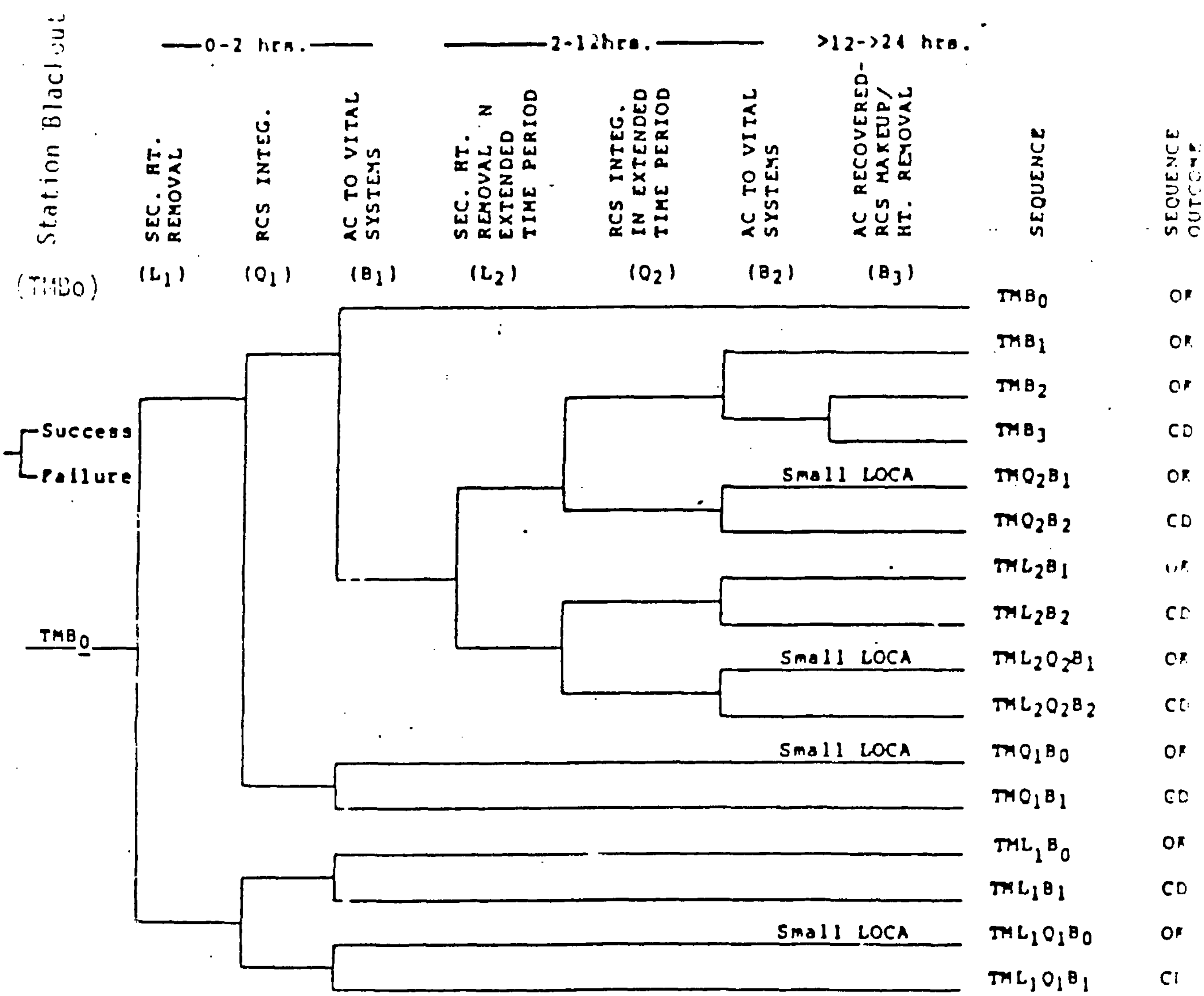


Figure 7.1 Generic PWR event tree for station blackout

Source: NUREG/CR-3226

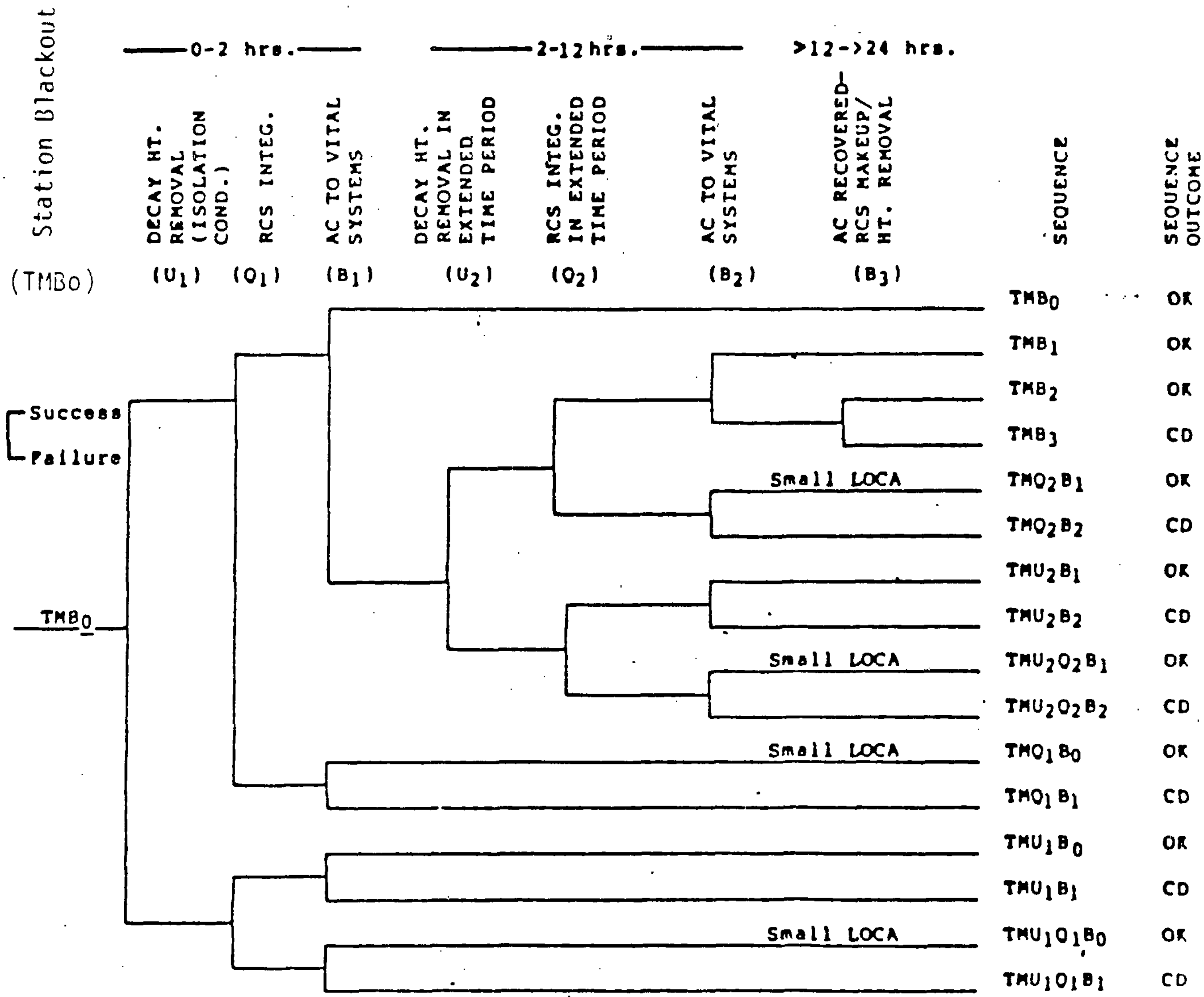


Figure 7.2 Generic BWR event tree for station blackout (BWR-2 or 3)

Source: NUREG/CR-3226

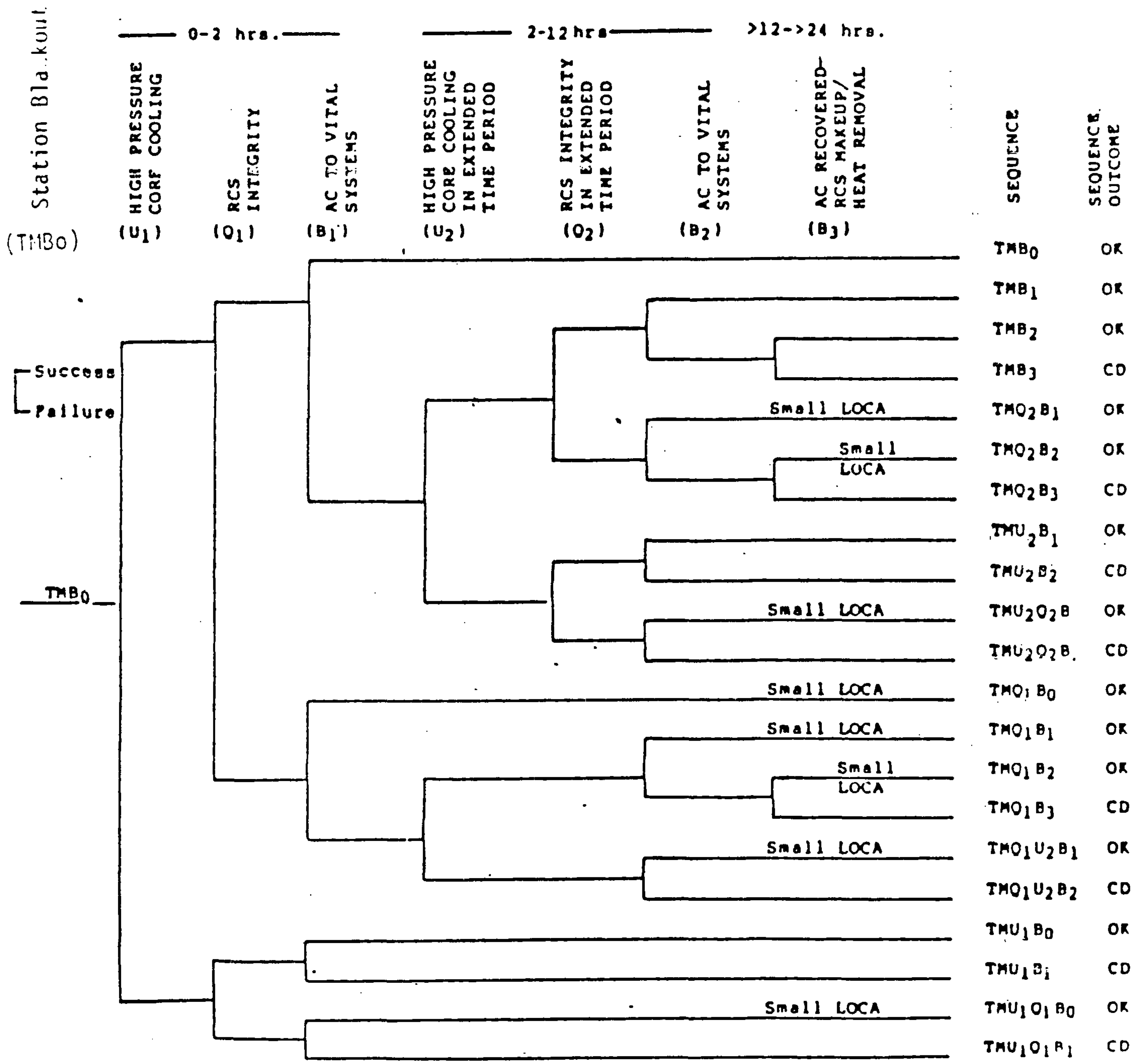


Figure 7.3 Generic BWR event tree for station blackout (BWR-4, 5, or 6)

Source: NUREG/CR-3226

coolant injection systems. Reactor coolant loss equivalent to that lost because of a stuck-open relief valve can be accommodated by the RCIC systems. The HPCI or HPCS system can provide adequate makeup to cope with larger leaks. All of the LWRs encompassed by the accident logic models are subject to the operational limitations for the longer duration blackouts as described previously in Section 6.

The event trees end with a sequence outcome state designated as "OK," meaning that stable, long-term core cooling is achieved or achievable, or "CD," meaning that an inadequate core cooling state is reached and some reactor core damage can be expected. For the latter case, core damage can be expected to proceed to core melt if effective and timely measures to restore AC power and core cooling are not taken or available. The potential difference between an accident sequence that ends in core damage and one that leads to core melt is determined by evaluating the likelihood of restoring core cooling and the cooling effectiveness from the onset of core damage to the time when irrevocable core melting has begun. This latter time in the accident sequence progression is not well known because there are significant uncertainties in the modeling of core melt phenomena. It has been estimated that the time between the onset of core damage and time that a core melt would penetrate the reactor vessel is on the order of 1 to 3 hours (NUREG/CR-1988, -2128). Considering the low probability that AC power would be restored during this time period and the uncertainty in modeling this accident process, including the ability to terminate a core melt in progress, it has been assumed that core melt would be the likely final outcome in accident sequences that progress to core damage.

Detailed plant transient response analyses were performed to cover the spectrum of sequences identified in the event trees (NUREG/CR-2181). The purposes of this work were (1) to better understand accident progression characteristics related to the timing of events and physical parameter values during the transient. and (2) to determine success states for systems, trains, components, and operator actions during station blackout sequences. The sequences were divided into three groups:

(1) failure of AC-independent decay heat removal with reactor coolant leakage less than Technical Specification upper limits

(2) failure of reactor coolant system integrity (liquid or steam leaks) with AC-independent decay heat removal systems operable

(3) failure of AC-independent decay heat removal systems with loss of reactor coolant system integrity

Variations in system failure and actuation time, reactor coolant leak rate, and operator actions were analyzed to determine both the potential for sequence outcomes with adequate (or inadequate) core cooling and the time in which AC power must be recovered to avoid core damage.

Table 7.1 shows the estimated time of core uncovery for station blackout sequences with AC-independent decay heat removal systems not available. Plants with Babcock and Wilcox (B&W)-type nuclear steam supply systems (NSSS), which have a small steam generator secondary water inventory and, thus, the smallest heat capacity, would require the most prompt recovery to avoid core damage for this particular sequence. For these plants, core uncovery was estimated to

Table 7.1 Estimated time to uncover core for station blackout sequences with initial failure of AC-independent decay heat removal systems and/or reactor coolant leaks

Sequence	Core uncovery time (seconds)		
	PWRs		
	B&W	CE	W
AFW failure	2715	6200	5800
Stuck-open PORV	3190	-	5040
100-gpm total leak rate from reactor coolant pump seals	21070	-	27950
AFW failure and stuck-open PORV	2480	-	4800
		BWRs	
		GE	
HPCI/RCIC failure		2300	
HPCI/RCIC failure and stuck open SRV		1680	

Source: Fletcher, 1981

occur within 1 hour. For plants with Westinghouse or Combustion-Engineering NSSS designs, core uncovery would take about 2 hours, as it would for a BWR-4 plant. Figure 7.4 shows how the core uncovery time is extended for sequences in which decay heat removal is initially successful but fails later during the accident. Estimates of the time core uncovery would take with a stuck-open relief valve and other types of reactor coolant leakage are also provided in Table 7.1. For BWRs with RCIC available (or HPCI or HPCS), adequate reactor coolant makeup is provided to maintain core cooling even with a stuck-open relief valve. The core uncovery time for PWRs would not be significantly shortened if a relief valve sticks open coincident with the loss of the steam turbine-driven train of the auxiliary feedwater system (AFWS). This is because loss of the AFWS for decay heat removal usually results in primary system pressure relief, which removes decay heat almost equivalent to the energy loss of a stuck-open relief valve with AC-independent decay heat removal available. If a relief valve sticks open in a BWR without RCIC or in cases when the AC-independent decay heat removal systems are unavailable, the core uncovery time would be somewhat shortened.

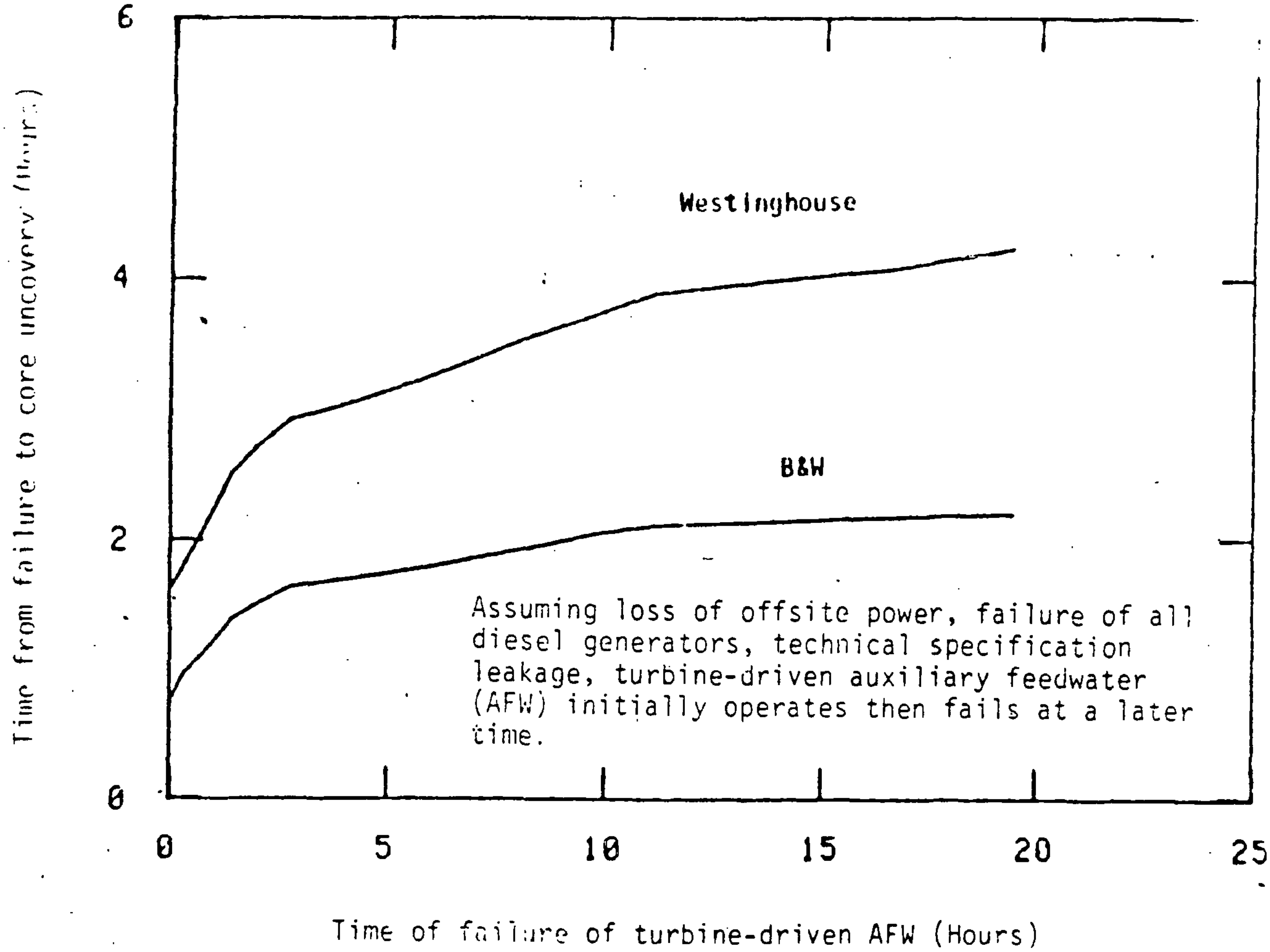
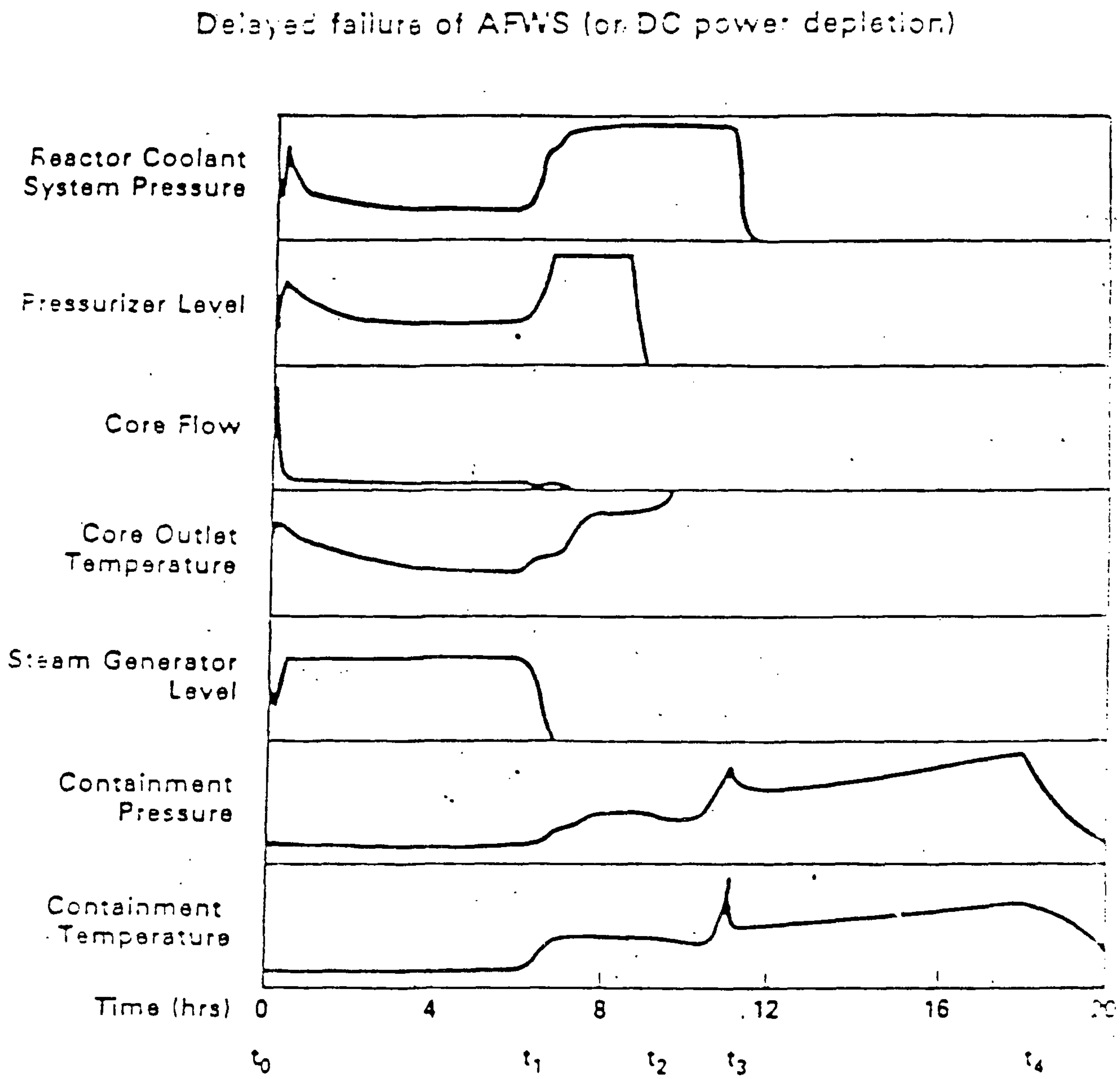


Figure 7.4 Time to core uncovery as a function of time at which turbine-driven auxiliary feedwater train fails

Source: Fletcher, 1981.

Complete accident progression analyses have been performed for several key station blackout sequences starting with the loss of offsite power through to core melt and containment failure. A time line presentation of a PWR sequence in which AFWS operation is initially successful but fails several hours into the transient is provided in Figure 7.5. Station blackout occurs at zero hours (t_0). After the initial fluctuations in reactor coolant system pressure, core outlet temperature, pressurizer level, core flow, and steam generator level, a relatively stable period of decay heat removal with primary coolant natural circulation follows. When AFW makeup to the steam generator becomes unavailable in about 6 hours (t_1), the steam generator level begins to drop, causing decreased heat transport from the primary coolant system. As the steam generator dries out and heat transfer to the secondary system ceases, reactor coolant pressure and core outlet temperature rise. The reactor coolant temperature increase combined with some voiding causes the pressurizer level to rise, and there is relief to the containment. Continued voiding in the primary system affects natural circulation flow, but core cooling is adequate to prevent melting until the core is uncovered (t_2) at about 9 hours. At this point, the pressurizer level has dropped because most of the primary system is voided. Within about 2 more hours (t_3) the core has melted and penetrated the reactor vessel, causing a containment pressure and temperature spike because of the rapid influx of steam and noncondensable gases from the melt. If containment survives that spike, the continued release of decay heat and the generation of combustible and non-combustible gas will continue to load the containment. Containment failure by overpressure in this sequence occurs about 19 hours into the accident (t_4).

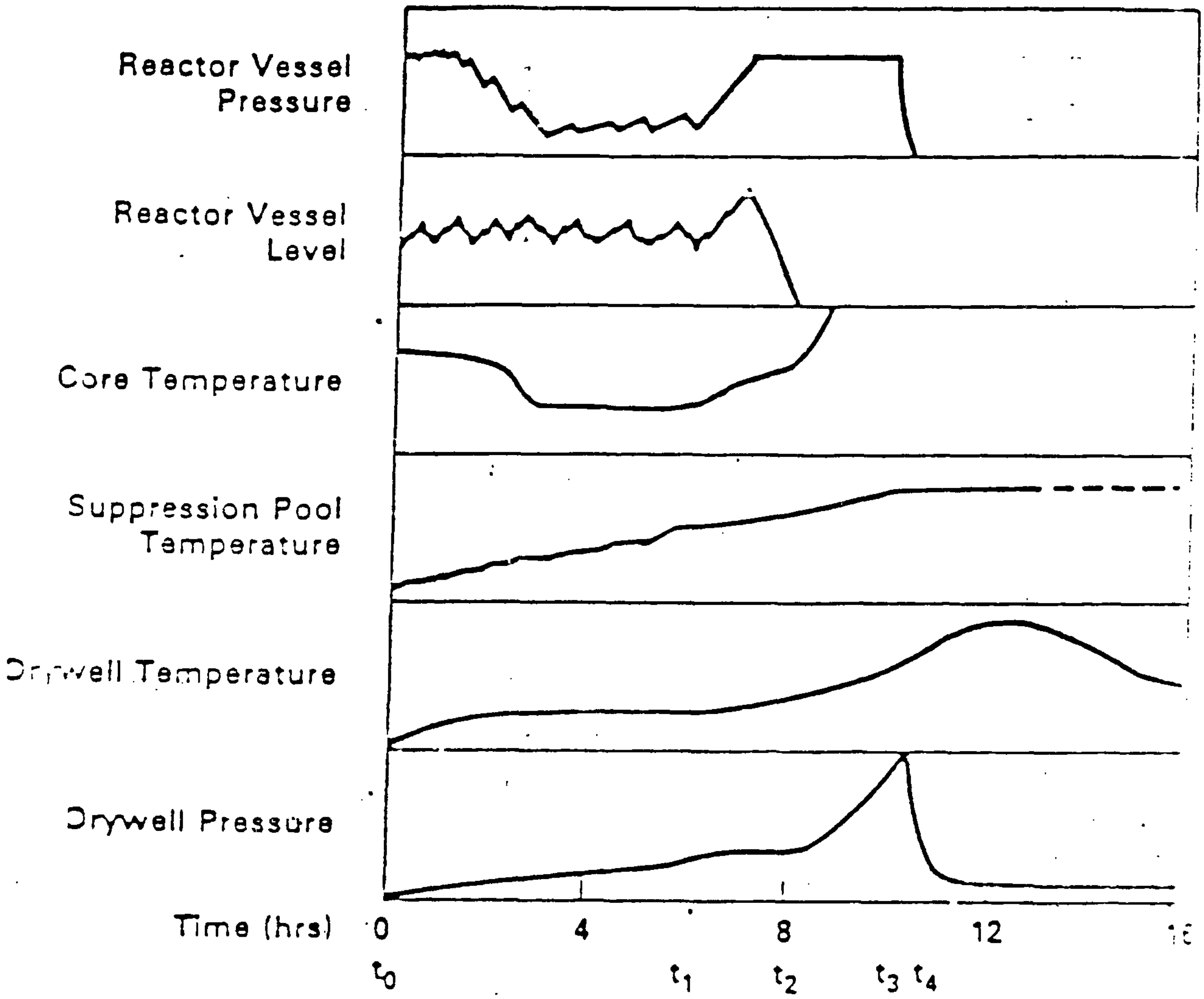
Figure 7.6 shows a BWR station blackout accident sequence progression. In this scenario for a BWR with Mark I containment, station blackout occurs at time zero (t_0). The reactor coolant system pressure and level are maintained within limits by RCIC and/or HPCI and relief valve actuations, which also transfers decay heat to the suppression pool. Both the suppression pool and drywell temperature begin to rise slowly; the latter is more affected by natural convection heat transport from the hot metal (vessel and piping) of the primary system. After 1 hour, when AC power restoration is not expected, the operator begins a controlled depressurization of the primary system to about 100 psi. This also causes a reduction in reactor coolant temperature from about 550°F to 350°F, which will reduce the heat load to the drywell as primary system metal components are also cooled. The suppression pool temperature increase is only slightly faster than it would have been without depressurization. Drywell pressure is also slowly increasing. At about 6 hours (t_1), DC power supplies are depleted, and HPCI and RCIC are no longer operable. Primary coolant heatup follows, with increases in pressure and level until the safety-relief valve set point is reached. Continued core heatup causes continued release of steam; this eventually depletes the primary coolant inventory to the point that the level falls and the core is uncovered, about 2 hours after loss of makeup (t_2). Core temperature then begins to rise rapidly, resulting in core melt and vessel penetration within another 2 or 3 hours (t_3). During the core melt phase, containment pressure and temperature rise considerably so that--nearly coincident with vessel penetration--containment failure occurs, either by loss of electrical penetration integrity (shown at t_4) or by containment overpressure shortly thereafter, around 11 hours into the accident.



Time	Sequence Event
t_0	Loss of all AC power
t_1	AFWS fails (or DC power depleted)
t_2	Core uncovery begins
t_3	Reactor vessel penetration
t_4	Containment failure

Figure 7.5 PWR station blackout accident sequence

RCIC/HPCI available, controlled depressurization



Time	Sequence Event
t_0	Loss of all AC power
t_1	DC power (batteries) depleted
t_2	Core uncovery begins
t_3	Reactor vessel penetration
t_4	Containment failure

Figure 7.6 BWR station blackout accident sequence

Estimates of the likelihood of these accident sequences were made to identify the potentially dominant contributors to the station blackout accident sequences (NUREG/CR-3226). Table 7.2 summarizes the results for the typical PWR and BWR. These results have been modified to account for better estimates of loss-of-offsite-power frequency and duration derived since NUREG/CR-3226 was completed (see Appendix A). In addition to identifying the dominant accident sequences and their likelihoods, the table also shows the major factors affecting the accident sequence frequency. For PWRs, an important contributor to the estimate of the likelihood core damage is the ability to restore AC power before the DC power needed to run the auxiliary feedwater system is lost or the condensate storage tank supplies are depleted. Another important contributor is the integrity of the reactor coolant system considering potential leaks from the reactor coolant pump seals following a station blackout. If reactor coolant pump seals leak and there is no way to supply makeup water to the reactor coolant system, the core will be uncovered. If reactor coolant pump seal leakage is large (more than 100 gpm per pump), the core could be uncovered within a few hours. Smaller leak rates (a few gpm per pump) are not a limiting factor. Adequate coolant inventory would be available to allow continued core cooling for a day or more without the need for makeup if other limitations (e.g., DC power) did not exist. The analyses performed for this program (NUREG/CR-3226) showed the reactor core was uncovered in approximately 8 hours, using the reactor coolant seal leakage information currently available (a leak rate of about 10 to 20 gpm per pump).

For BWRs with isolation condensers, a similar dominant failure mode exists. The failure of the DC power system is less important because the isolation condenser system operates passively once it is activated; little operator action is necessary thereafter. However, reactor coolant pump seal failure could cause depletion of reactor coolant inventory and, because the isolation condenser BWR typically does not have an AC-power-independent makeup system, the reactor core could be uncovered. This sequence was estimated to result in core damage in about 8 to 12 hours. BWRs with HPCI and RCIC are capable of coping with reactor coolant system leaks equivalent to that resulting from a stuck-open relief valve. However, they are subject to the effects of DC power depletion and other interactive failures associated with the lack of the ventilation system to maintain HPCI and RCIC room temperature, and suppression pool heatup phenomena that can result in a loss of core cooling in about 8 to 12 hours. For this type of plant, unattenuated suppression pool temperature increases during a station blackout transient can be a problem because of the potential for unstable condensation phenomena. These phenomena could cause containment structural failure, with the potential for subsequent loss of reactor coolant from the suppression pool resulting in loss of recirculation capability. However, recent test data provided by General Electric in support of the BWR Owners Group suggest there is no unstable condensation regime (General Electric Topical Report NEDO-30832). Perhaps more important is the effect that high suppression pool temperature would have on HPCI pumps during recirculation. These pumps are not usually qualified for operation with fluid temperatures in excess of 160°F. In addition, NPSH requirements may not be satisfied if suppression pool temperatures exceed 200°F.

Table 7.2 Summary of potentially dominant core damage accident sequences

Generic plant	Sequence	DHR system/component contributors	Time in which AC power must be recovered to avoid core damage, hr	Typical core damage frequency
PWR (all	TML_1B_1	Steam driven AFWS unavailable		
	TML_2B_2	DC power or condensate exhausted	4 to 16	1×10^{-5}
	TMQ_2B_2	Reactor coolant pump seal leak	4 to 16	1×10^{-5}
BWR w/isolation condenser	TMU_1B_1	Isolation condenser unavailable	1 to 2	2×10^{-6}
	TMQ_1B_1	Stuck-open relief valve	1 to 2	3×10^{-6}
	TMQ_2B_2	Reactor coolant pump seal leak	4 to 16	2×10^{-5}
BWR w/HPCI-RCIC	TMU_1B_1	HPCI/RCIC unavailable	1 to 2	2×10^{-6}
	TMU_2B_2	DC power or condensate exhausted, component operability limits exceeded (HPCI/RCIC)	4 to 16	2×10^{-5}
BWR w/HPCS-RCIC	TMU_1B_1	HPCS/RCIC unavailable	1 to 2	5×10^{-7}
	TMU_1B_1	HPCS unavailable, DC power or condensate exhausted, component operability limits exceeded (RCIC)		

For BWRs with HPCS, which has its own AC and DC power systems, both the effects of depletion of the DC supply and reactor coolant leakage are minimal contributors to sequence core melt probability. However, suppression pool temperature limitations may cause some equipment operability problems during longer duration station blackouts.

In all of the accident sequences evaluated for this program, the early failure of decay heat removal because of the initial unreliability of these systems was a relatively small, but not insignificant, contributor to core melt frequency. This is not surprising, because, since the accident of Three Mile Island Unit 2 (TMI-2), most nuclear power plants have been required to have at least one AC-power-independent decay heat removal train available. However, very little has been done at nuclear power plants to determine the capability and reliability of systems during a sustained loss of AC power. Thus, it is not inconsistent that most of the dominant failure modes that have been identified are associated with the inability to operate decay heat removal systems because of support system failures or capacity limits on support and auxiliary systems needed to maintain decay heat removal during station blackout.

With the consideration of containment failure, station blackout events can represent an important contributor to reactor risk. In general, active containment systems are unavailable during a station blackout event. These systems are usually required for pressure suppression through steam condensation to maintain the containment pressure below the appropriate limits and for the removal of radioactivity from the containment atmosphere following an accident. The time to containment failure after the onset of core damage and the containment failure mode is an important factor in determining fission product release and ultimately public risk.

Table 7.3 summarizes containment failure insights derived from the analyses performed for the severe accident research program at the NRC (NUREG-1150). It shows the different types of containment, the estimated time of containment failure following the onset of core damage, and the consequences of containment failure resulting from a station blackout accident. For the large, dry PWR containment, long-term failure (by overpressure or basemat meltthrough) or no failure is more likely than early failure. The potential for early failure is principally associated with uncertainties in the phenomena related to "direct containment heating," as discussed in draft NUREG-1150. Because of its smaller volume and pressure capacity, the PWR ice condenser containment is less capable in handling steam or hydrogen combustion loads during station blackout accidents. In NUREG/CR-3226, it was estimated that the containment would fail in about 1 or 2 hours for several possible reasons including hydrogen burn, steam pressure spike, or containment overpressure as a result of noncondensables and noncondensed steam. The recent analyses show a lesser likelihood of containment failure. Analyses performed as part of the Industry Degraded Core Rulemaking Program (IDCOR, 1984), show containment failure times of more than 1 day and significant reductions in perceived consequences.

The BWR Mark I and II containments offer some pressure suppression capability during a station blackout accident, but after a core melt, they may fail by one of several modes. Because of the small size of these containments, direct contact of molten core material with the containment wall has been identified as a potential failure mode. In addition, temperature-induced failure of penetrations or the steel containment structure has been identified as a potential threat. Absent effective containment venting strategies during station blackout,

Table 7.3 Containment performance and consequence results for station blackout accident sequences

Plant	Sequence	Containment performance: Failure mode	Probability: Mean	Probability: Range	Timing (hr)	Population dose: Mean	Population dose: Range
Surry	Station blackout w/seal LOCA (SNNN)	Early	0.3	0-1	3	1E+07	4E6-2E7
		Late	<0.01	--	--	--	--
		Basemat melt-through	0.3	0-0.4	>24	2E+04	*
		None	0.37	0.01-0.6	N/A	2E+04	*
	Station blackout no seal LOCA (TNNN)	Early	0.3	0-0.9	3	1E+07	4E6-2E7
		Late	<0.01	--	--	--	--
		Basemat melt-through	0.2	0-0.5	>24	2E+04	*
		None	0.4	0-0.9	N/A	2E+04	*
Zion	Station blackout w/seal LOCA (SE)	Early	0.3	*	2	3E+07	*
		Late	0.5	*	15	1E+07	*
		Basemat melt-through	0.16	*	>24	3E+04	*
		None	<0.01	--	N/A	--	--
	Station blackout no seal LOCA (TEC)	Early	0.2	*	3	3E+07	*
		Late	<0.01	--	--	--	--
		Basemat melt-through	<0.01	--	--	--	--
		None	0.7	*	N/A	3E+04	*

See footnotes at end of table.

Table 7.3 (Continued)

Plant	Sequence	Failure mode	Containment performance: Probability Mean	Range	Timing (hr)	Population dose Mean	Range
Sequoyah	Station blackout w/seal LOCA (S2NNNN)	Early	0.56	*	2	5E+06	*
		Late	0.4	*	**	2E+06	*
		None	0.03	*	N/A	1E+04	*
	Station blackout no seal LOCA (TNNNN)	Early	0.56	*	3	5E+06	*
		Late	0.4	*	**	2E+06	*
		None	0.01	*	N/A	1E+04	*
Peach Bottom	Station blackout --slow (6-hr battery depletion) (TB)	Early	0.6	0.01-0.8	12	2E+07	3E6-4E7
		Late	0.3	0.1-0.6	15	7E+06	2E6-1E7
		None	0.1	0.05-0.2	N/A	1E+04	*
	Station blackout --fast (TBU/TBUX)	Early	0.6	0.01-0.8	3	2E+07	3E6-4E7
		Late	0.3	0.1-0.6	6	7E+06	2E6-1E7
		None	0.1	0.05-0.2	N/A	1E+04	*
Grand Gulf	Station blackout --slow (6-hr battery depletion (TB)	Early	0.3	0.25-0.4	12	9E+05	1E5-8E6
		Late	0.6	0.5-0.7	**	6E+05	1E5-2E6
		None	0.1	0.05-0.15	N/A	3E+05	*
	Station blackout --fast (TBU/TBUX)	Early	0.3	0.25-0.4	3	7E+05	1E5-8E6
		Late	0.6	0.5-0.7	**	5E+05	1E5-2E6
		None	0.1	0.05-0.15	N/A	3E+05	*

*Not currently available from NUREG-1150 analyses.

**Dependent on timing of power restoration, spray operation, and hydrogen burning.

NOTE: N/A = not applicable.

overpressure of the containment has also been predicted within 5 to 15 hours. (IDCOR estimates a Mark I containment will fail in about 18 hours as a result of temperature loadings.) Because these containments are generally inerted, hydrogen burn is not considered a likely failure mode. For Mark III containments, which are low pressure, large volume containments, failure in about 20 hours has been estimated in NUREG-1150 analyses for late overpressure scenarios not involving hydrogen combustion. The IDCOR estimate is 47 hours for this type of containment failure.

One item of interest should be noted for both the ice condenser containment and the Mark III containment, where hydrogen ignitors must be installed to meet hydrogen rule requirements and the post-Construction Permit Manufacturing Licensee (CPML) rule. For these containments, there is the potential that an inactive ignitor could be turned on following the restoration of AC power at a time when the hydrogen concentration is essentially at an explosive level. This consieration has been accounted for in the probability and consequence estimates shown in Table 7.3. However, this potential problem can be addressed and somewhat suppressed through proper procedures and by instructing the operators on how to control the hydrogen burning with ignitor systems following the restoration of AC power.

Substantial uncertainties exist regarding containment performance during a core melt accident. Based on the best information available at this time, it can be seen that station blackout accidents can potentially result in substantial consequences. However, the reader is cautioned that there are some technical disagreements between NRC and IDCOR and that ongoing research could cause revision of these recent findings.

8 EVALUATION OF DOMINANT STATION BLACKOUT ACCIDENT CHARACTERISTICS

The important factors that affect the probability of station blackout accidents have been identified on the basis of the previous work presented on dominant station blackout accident sequences. The principal parts of the station blackout sequence include: the likelihood or frequency of loss of offsite power; the probability that the emergency or onsite AC power supplies will be unavailable; the capability and reliability of decay heat removal systems that must function during a loss of AC power; and the likelihood that a source of offsite power will be restored before the core is damaged as a result of the loss of core cooling and the failure of systems that cannot operate without AC power. Reactor type, by itself, has not been found to be a dominant factor in determining likelihood of core damage as a result of station blackout because the capabilities of auxiliary and support systems needed for decay heat removal during station blackout can vary considerably (and still meet current safety requirements). The important factors in determining the likelihood of core damage as a result of station blackout are reliability of the AC power system (offsite and onsite) and the performance of these auxiliary systems (DC power, compressed air), as well as such plant characteristics as pump seal design, natural circulation capability, and suppression pool temperature effects.

Because of these differences, core damage frequency estimates for station blackout accident sequences could vary considerably. Therefore, the NRC staff analyzed the sensitivity of core damage frequency estimates to design variations different from the reference plant analyses performed by Sandia National Laboratories (NUREG/CR-3226). The models used were based on insights obtained from previous studies; they are described in Appendix C. Station blackout sequences were divided into two groups. The first included sequences involving the failure of AC-independent decay heat removal and, for plants without AC-independent makeup, loss of reactor coolant integrity at the onset of or soon after a station blackout. For these early core cooling failure sequences, AC power must be restored in 1 or 2 hours to avoid core damage and ultimately core melt. The second group of sequences identified included failures during an extended station blackout of 4 to 8 hours or more. These failures include a smaller rate of reactor coolant loss, support system capacity limitations (e.g., batteries, makeup water inventory, compressed air), and other station blackout capability limitations in decay heat removal systems (e.g., natural circulation and suppression pool temperature limitations).

Several sensitivity analyses have been performed by NRC staff to evaluate variations in LWR plant designs for both decay heat removal capability and system reliability, including offsite power. Because the ability to cope with a station blackout may vary considerably, results are provided to show the effect of limitations in maintaining decay heat removal during station blackouts of 2 to 16 hours. First, Figure 8.1 shows the sensitivity to offsite power system design and location as represented by different offsite power groups (clusters). The importance of higher frequency and long-duration losses of offsite power can be seen. It is also worthwhile to note that the highly reliable (redundant) AC-independent decay heat removal systems provide added value when ability to cope for long durations exists and very low core melt frequencies are estimated.

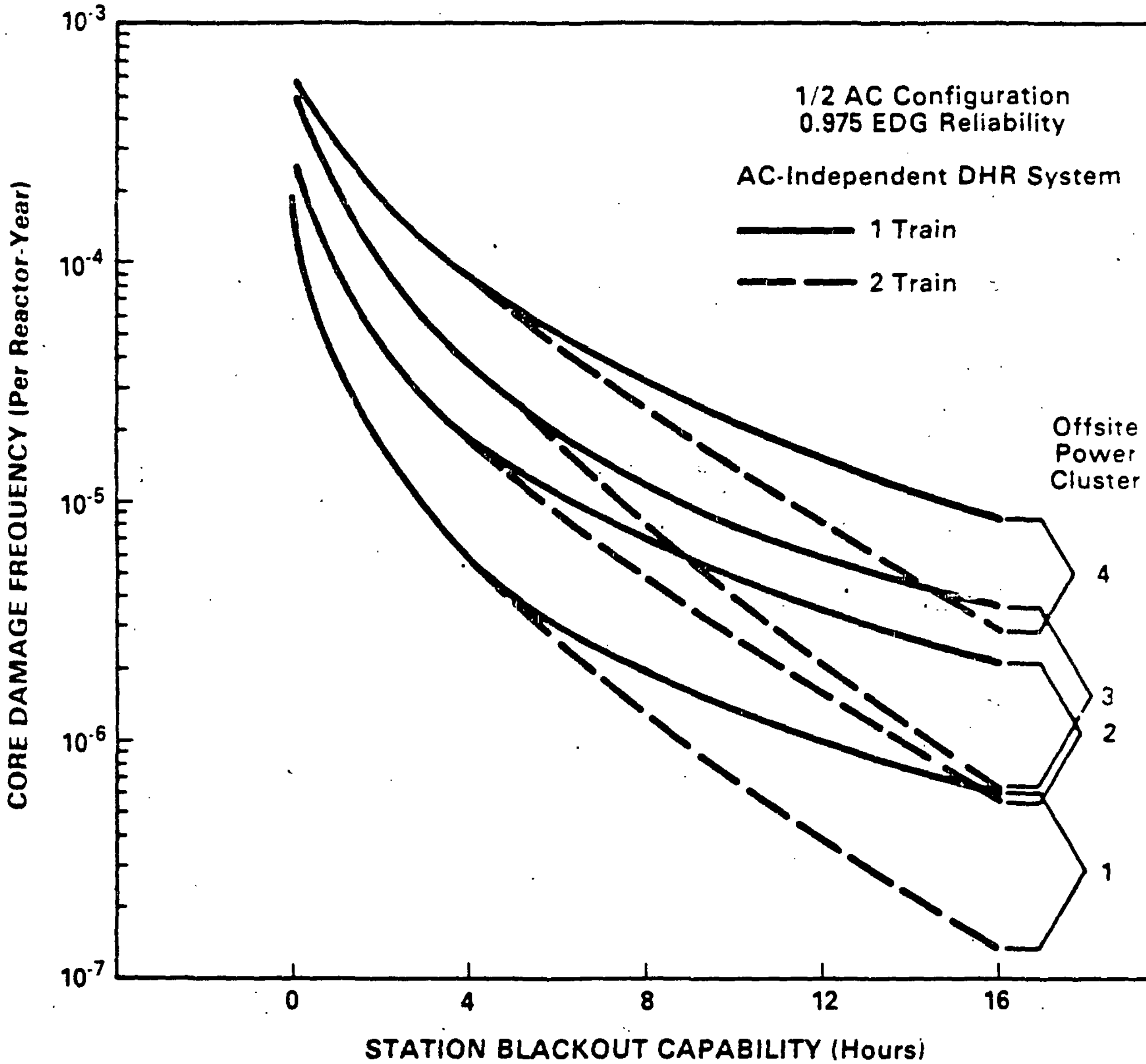


Figure 8.1 Sensitivity of estimated station blackout--core damage frequency to offsite power cluster, AC-independent decay heat removal reliability, and station blackout coping capability

Figure 8.2 shows the relationship between various emergency diesel generator reliability levels and estimated core damage frequency. A combination of reasonably good diesel generator reliability and the ability to cope with a station blackout lasting several hours results in estimated core damage frequencies on the order of 10^{-5} per year or less. The effect of a plant's emergency AC power configuration is shown in Figure 8.3. A substantial difference in core damage frequency may exist between plants with three emergency diesel generators, depending on the minimum number (1 or 2) needed to maintain core cooling and decay heat removal during a loss of offsite power. Again, frequencies drop rapidly as station blackout coping capabilities extend to cover longer AC power outages. Figure 8.4 shows the variations in emergency diesel generator failure rate from both independent and common causes. In this figure, common cause failures in support systems (e.g., service water, DC power) are estimated on the basis of the industry experience (see Appendix B). These results show that estimated core damage frequency can be kept low by maintaining highly reliably emergency AC power systems. Estimated core damage frequencies as low as 10^{-6} per year may be possible if the emergency AC power system is maintained in a high state of operational reliability and there is some capability of coping with an unlikely station blackout.

The results described above and additional sensitivity analyses can be used to assess the effectiveness of certain strategies in dealing with station blackout concerns. For instance, if PWR reactor coolant pump seals were known to fail early during station blackout and the reactor coolant system leakage were the factor limiting the ability to cope with station blackout, core damage could occur 1 or 2 hours after the loss of AC power, even if the AC-independent decay heat removal system (the AFWS) were operating properly. Table 8.1 has been developed from the sensitivity analyses to show the effect of providing a "fix" to maintain reactor coolant pump seal integrity to allow successful core cooling for station blackouts from 2 to 4 and 4 to 8 hours.

The results provided up to this time represent point estimates of probability or, more properly, frequency. NUREG/CR-3226 shows the effect of using log normal distributions to represent basic event probabilities on mean probability estimates, calculated medians, and uncertainty ranges. When that work was completed, the magnitude of the uncertainty in the loss of offsite power frequency and duration estimates was not known. Because the uncertainty bounds are now perceived to exceed those used in NUREG/CR-3226, the accident sequence uncertainty ranges derived using the most recent uncertainty estimates for loss of offsite power frequency may be larger than previously estimated. The loss of offsite power frequency and duration estimates are most uncertain for the very low frequency, long duration losses of offsite power. The uncertainty on the probability of accident sequences which result from the shorter duration losses of offsite power should not be significantly different from the previous estimates.

Some typical station blackout core damage probabilities and uncertainty ranges representing a 90% confidence interval have been provided in Figure 8.5 for reference. The sequence mean is typically 3 to 8 times larger than the point estimate and the upper and lower bounds are typically within a factor of 5 to 20 of the median estimate. The large difference in point estimate and mean can be attributed to the use of a log-normal distribution. When sequences are combined into a single core damage probability, the proportional distance between mean and point estimate tends to decrease somewhat.

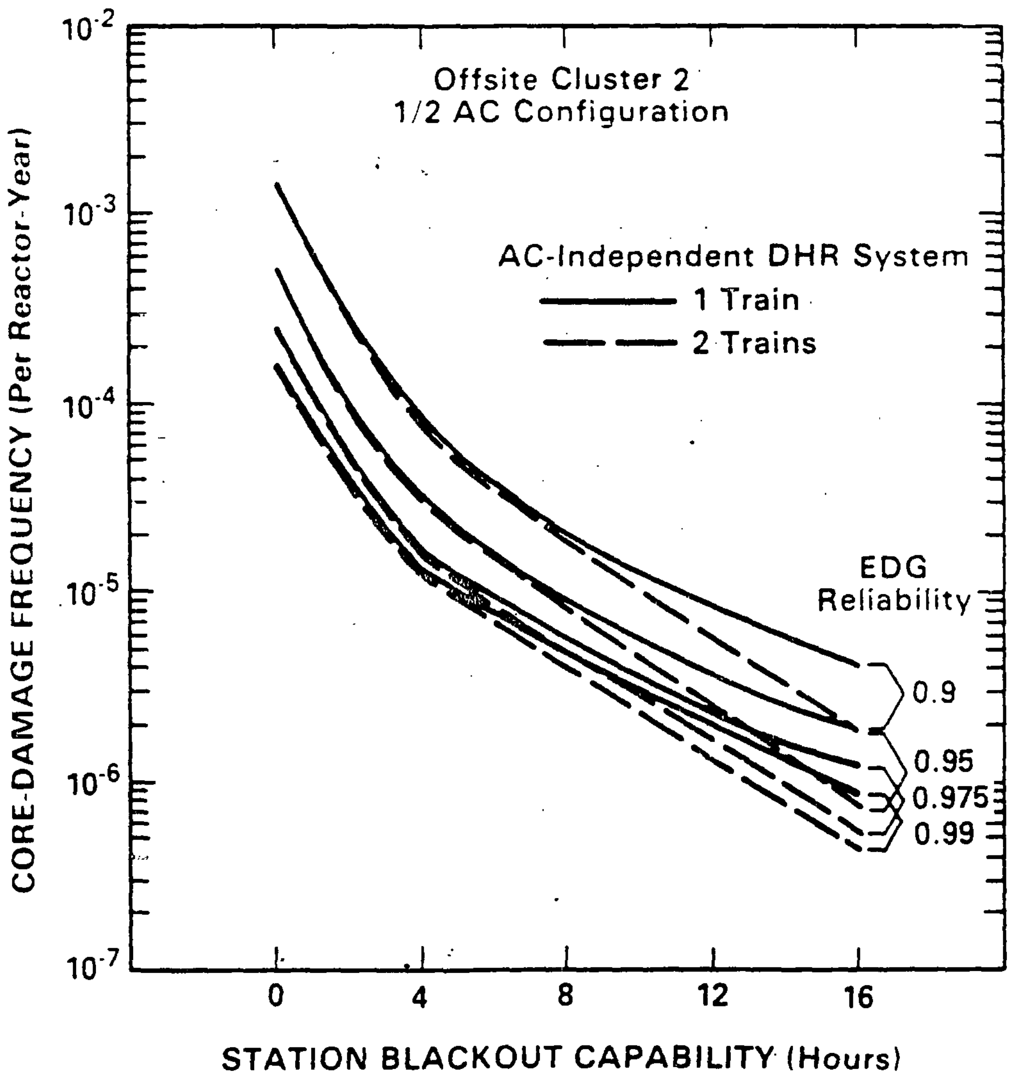


Figure 8.2 Sensitivity of estimated station blackout--core damage frequency to emergency diesel generator reliability, AC-independent decay heat removal reliability, and station blackout coping capability

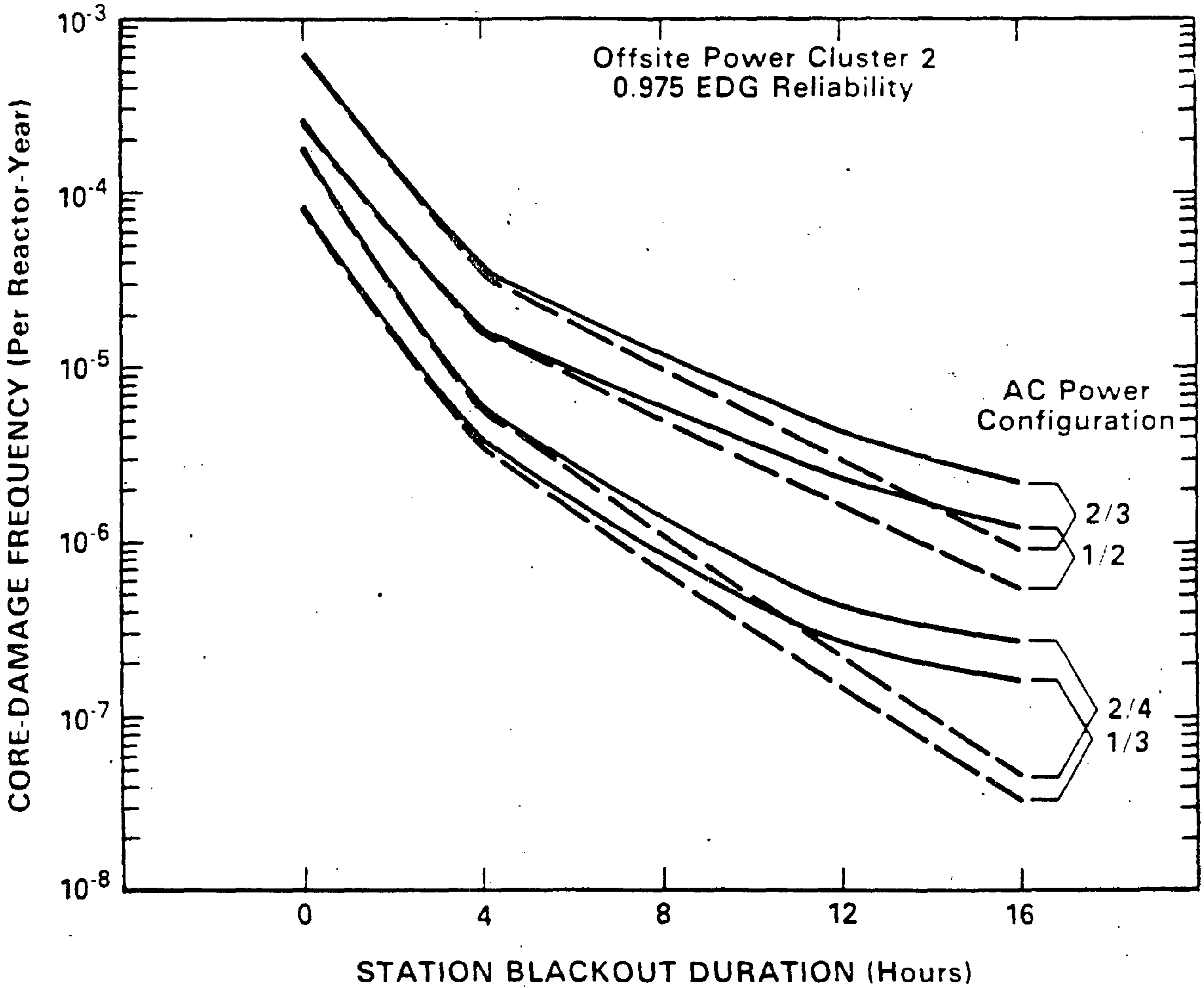


Figure 8.3 Sensitivity of estimated station blackout--core damage frequency to emergency AC power configurations, AC-independent decay heat removal reliability, and station blackout coping capability

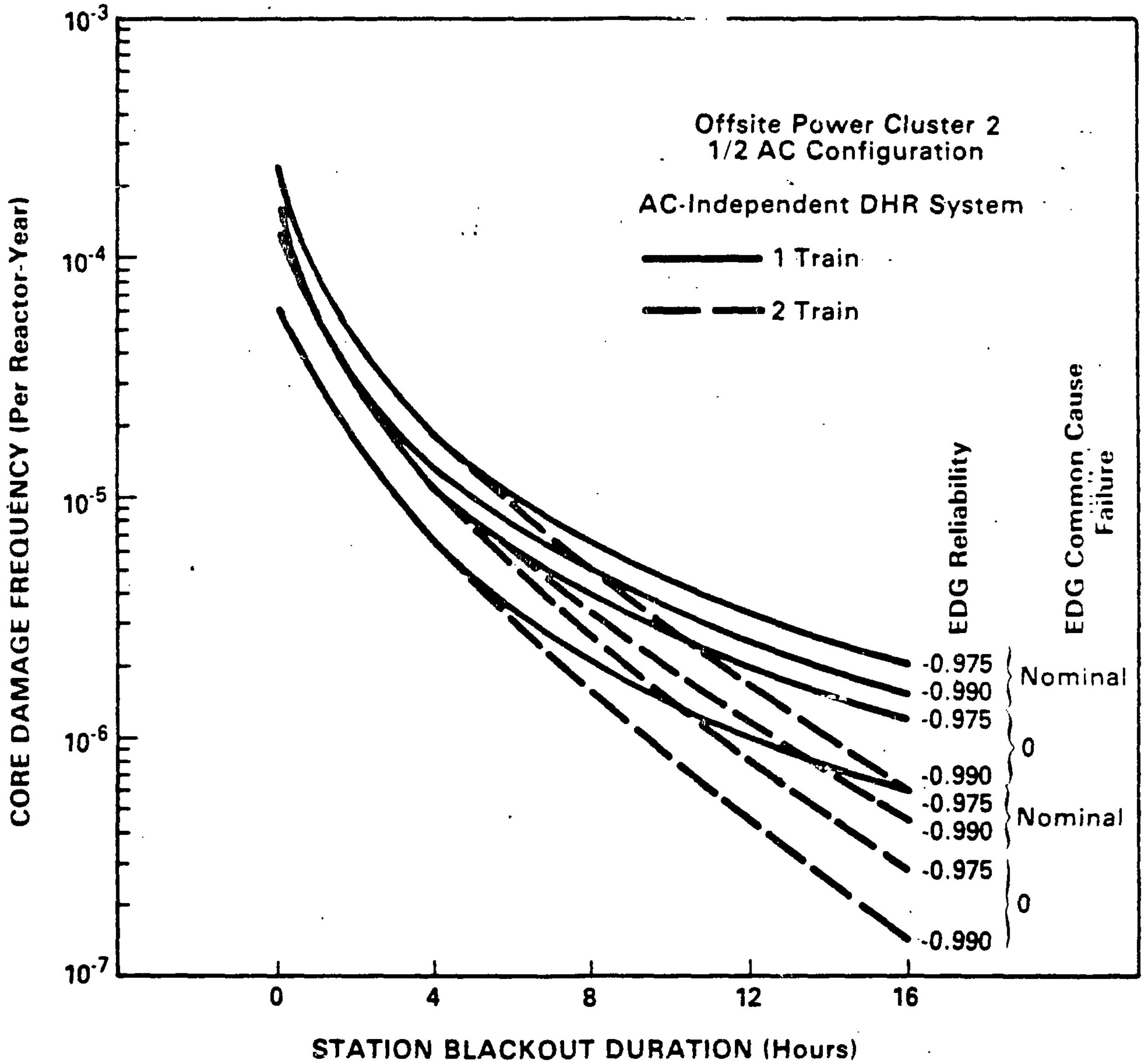


Figure 8.4 Sensitivity of estimated station blackout--core damage frequency to reducing the common cause failure susceptibility of emergency diesel generators, their reliability, and station blackout coping capability

Table 8.1 Sensitivity of estimated core damage frequency reduction for station blackout accidents with reactor coolant pump seal failure delay from 2 to 4 hours and 4 to 8 hours

Configuration and Cluster	Estimated core damage frequency (per reactor-year) EDG$\overline{R}$* = 0.025 2 to 4 hr	EDG$\overline{R}$* = 0.025 4 to 8 hr	EDG$\overline{R}$ = 0.05 2 to 4 hr	EDG$\overline{R}$ = 0.05 4 to 8 hr
1/2 configuration:				
1	6.8×10^{-6}	3.5×10^{-6}	1.2×10^{-5}	5.9×10^{-6}
2	2.1×10^{-5}	1.2×10^{-5}	4.0×10^{-5}	1.9×10^{-5}
3	4.7×10^{-5}	2.6×10^{-5}	8.8×10^{-5}	4.5×10^{-5}
4	8.1×10^{-5}	5.1×10^{-5}	1.2×10^{-4}	8.5×10^{-5}
1/3 configuration:				
1			2.4×10^{-6}	9.9×10^{-7}
2			7.7×10^{-6}	3.2×10^{-6}
3			1.8×10^{-5}	7.3×10^{-6}
4			2.7×10^{-5}	1.4×10^{-5}

*EDG$\overline{R}$ = emergency diesel generator unreliability (i.e., failure rate per demand)

The measure of risk associated with a station blackout accident can be obtained by multiplying the estimated core damage likelihood by the estimated dose that would result from containment failure during the accident. The recovery of AC power during the accident would provide the potential for terminating core damage before core melt and the potential for reducing fission product releases by delaying containment failure or by actuating containment sprays before containment failure.

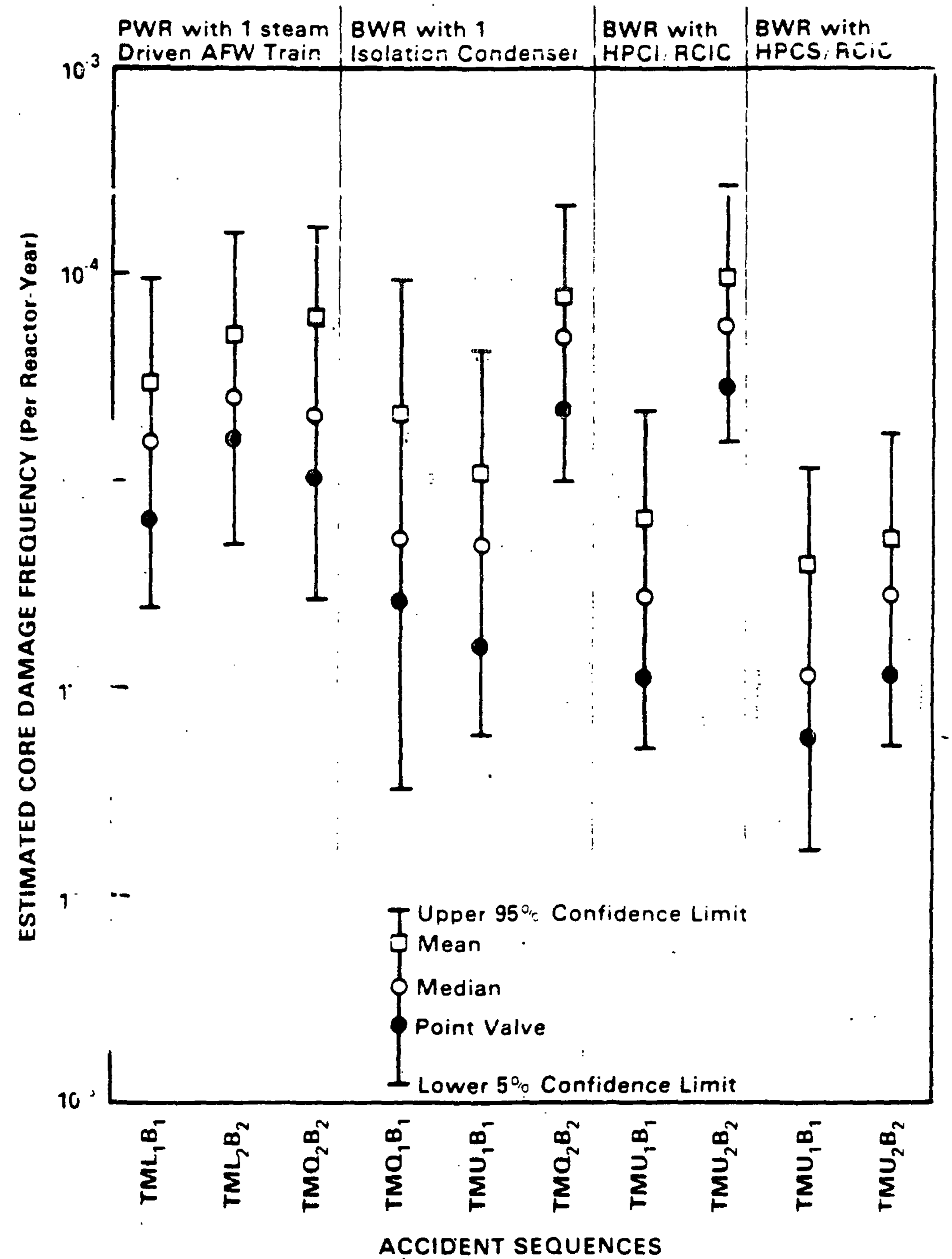


Figure 8.5 Estimated core damage frequency showing uncertainty range for four reference plants

Source: NUREG/CR-3226

9 RELATIONSHIP OF OTHER SAFETY ISSUES TO STATION BLACKOUT

The implications of station blackout on several other safety issues were reviewed for significance. These include: loss-of-coolant-accident initiators; anticipated transients without scram; external hazards, such as seismic events and severe weather; and internal hazards associated with fire or extreme environments, such as flooding or high steam temperature resulting from pipe breaks within the plant. In general, it was concluded that, if the likelihood of station blackout were independent of any of these other safety considerations, the potential risk of a station blackout concurrent with one of these other safety concerns is very small. However, if as a result of common cause failure or interactive failure, the initiation of an accident by one of those other mechanisms described causes a station blackout, then the safety implications of those safety issues on station blackout are fairly large. Each of these safety issues is discussed below.

9.1 Loss-of-Coolant Accidents

Loss-of-coolant accidents (LOCAs) induced by a station blackout transient have already been included in the accident sequence analyses described in Section 7; these will not be discussed further here. LOCAs concurrent with a loss of offsite power are usually included in the design basis of nuclear power plants in accordance with the general design criteria of Appendix A to 10 CFR 50. The likelihood of a LOCA followed by and concurrent with a station blackout has been considered and is discussed below.

Although no strong coupling could be found between the initiation of a LOCA and a subsequent failure of the offsite or onsite AC power system, one potential mechanism has been identified. If a LOCA were to occur at a nuclear power plant, the reactor would trip; subsequently the turbine generator would be tripped and a grid instability could follow, or the site could be isolated by switching activities in the switchyard to provide onsite safety-related or alternative sources of preferred power to the emergency power safety buses. Historical experience collected about loss-of-offsite-power events at nuclear power plants suggests that given a transient or an accident situation that would cause a trip of the turbine generator, the likelihood of a failure of the offsite power supply is on the order of 10^{-4} to 10^{-2}, depending on the strength of the grid and the offsite power design at the site.

Estimated LOCA frequencies range from 10^{-2} per reactor-year for small loss-of-coolant accidents down to less than 10^{-4} per reactor-year for large diameter pipe breaks. The frequency of small LOCAs is dominated by pump seal LOCAs on pressurized-water reactors and stuck open safety-relief valves on boiling-water reactors. These situations do not require rapid actuation of AC-powered emergency safety feature equipment and have been addressed previously. The most likely small LOCA that has not been incorporated in the station blackout accident analyses is a small pipe break (less than 2 inches in diameter) with a frequency of about 10^{-3} per reactor-year.

The low LOCA frequency combined with the likelihood of losing offsite power on turbine-generator trip results in an estimated frequency of occurrence ranging from 10^{-5} per reactor-year to 10^{-7} per reactor-year. When this frequency is combined with a conservative estimate of emergency AC power system unreliability of 10^{-2} per demand, it is easily shown that accident sequences of this type represent a small element of reactor risk (less than 10^{-7} per reactor-year). The variability of the frequency of station blackout caused by a LOCA could be as much as two orders of magnitude higher and still represent one of the smaller station blackout accident threats. Although, at this higher level, these accidents could represent a noticeable fraction of reactor risk. Large pipe break LOCAs with initiating frequencies on the order of 10^{-4} per reactor-year combined with the probability of subsequent failure of all AC power do not appear to represent an appreciable fraction of accident likelihood or public risk, at least in comparison to other station blackout sequences.

9.2 Anticipated Transients Without Scram

Another safety consideration that was investigated is anticipated transients without scram. In this case, the anticipated transient is a loss of offsite power. If the probability of a loss of offsite power is taken as the generic average, 0.1 per year, and the probability of reactor scram failure is taken as the historical average, about 10^{-4} per demand, then the probability of a loss of offsite power followed by a failure to scram is about 10^{-5}. This is a level of accident sequence likelihood that might be considered important. However, in order for a station blackout to occur, the onsite emergency AC power system also must fail. In the worst case, one might find an unreliability of the emergency AC power system of about 10^{-2} per demand. Thus, the frequency of an anticipated transient without scram involving loss of offsite power and a failure of the onsite emergency AC power system is on the order of 10^{-7} per reactor-year or less. Even if the level of uncertainty were an order of magnitude higher, this accident sequence would not be of concern in comparison to the dominant station blackout accident sequences that have been identified.

9.3 Extreme Internal Environment

A safety area in which there does appear to be a potential for station-blackout-type accident sequences being induced by other causes involves fire and other extreme environments internal to a nuclear power plant. The concern associated with internal environmental hazards is that their occurrence can represent a common cause accident initiator that also affects the ability to cope with the incident. Specifically of concern is the likelihood of a fire, flood, or other extreme environmental condition generated by internal events that would cause a loss of all AC power. In general, for this to occur, portions of AC power systems must be in a common location where these hazards are present, or protection barriers and AC power system design requirements must be insufficient to control the spread or failure resulting from these hazards. Therefore, the likelihood of internal hazards causing a station-blackout-type accident is heavily dependent on the plant's design and, in particular, on the location of equipment. If separation and internal environmental protection barriers are maintained, or adequate AC system design is provided, the likelihood of these internal environmental hazards causing a station-blackout-type accident would be very small, probably less than 10^{-6} per reactor-year. On the other hand, if commonality of location or a lack of protection exists at a plant, then the safety significance of these internal hazards would have to be evaluated for

plant damage susceptibility and likelihood of occurrence. The frequency of occurrence of these hazards can be as high as once per 100 to once per 1,000 reactor-years. Therefore, the vulnerability to station-blackout-type accidents resulting from these hazards can be of concern.

9.4 External Hazards

Another potentially significant safety consideration that could be related to station blackout involves external hazards to the plant, particularly those resulting from seismic- and weather-induced failures. To date, a seismically induced loss of offsite power has not been observed at a nuclear power plant. Failure of offsite power because of severe weather has been observed at nuclear power plants; in fact, severe weather was included as a major factor in determining the likely duration of an extended offsite power outage at nuclear power plants, as described in Section 3. The greatest potential for safety significance exists where there is a direct coupling or common cause failure associated between a transient-initiating external hazard causing loss of offsite power and the reliability of the onsite and offsite power systems. It can be expected that significant seismic and severe-weather events will cause a loss of the offsite power system. On the other hand, the plant, and in particular the emergency AC power system, is typically designed to withstand, or is protected from the effects of, these severe phenomena. Therefore, for severe external hazards that are within the design basis of the plant, the failure of the emergency AC power system can be considered as an independent failure event. For example, if the likelihood of a safe shutdown earthquake that could cause a loss of offsite power were approximately 10^{-3} per year or less, and one assumes that it would take approximately 8 to 24 hours to restore offsite power from such an incident, then a typical estimate of core damage or core melt frequency as a result of a safe shutdown earthquake and a station blackout would be about 10^{-6} per reactor-year or less. For severe weather, the likelihood of the weather-induced failure of the offsite power system could be as high as 10^{-2} per year, and the outage could be expected to be on the order of several hours. Again, if the severe-weather event is within the design basis of the plant, the likelihood of a weather-induced station blackout accident causing core damage or core melt would be on the order of 10^{-5} per reactor-year.

Table 9.1 provides a summary of the typical internal and external accident hazards of a nuclear power plant and identifies some potential points of failure that could result in a coupling between these accident initiators and a station blackout. If such interactions or points of commonality do not exist, then it is concluded that the contribution of these accident initiators to station blackout accident sequences results in core melt frequencies that are no larger, and probably much less, than those previously considered.

Table 9.1 Coupling between external (and internal) events and potential plant failures

Event	Potential plant "weakness"
Seismic	Switchyard, control, non-seismically designed equipment
Fire, flood	Areas with multiple divisions, inadequate protection barriers
Severe weather	Transmission lines and towers, switchyard, non-safety structures

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APPENDIX A

DEVELOPMENT OF LOSS-OF-OFFSITE-POWER FREQUENCY AND DURATION RELATIONSHIPS

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APPENDIX A

DEVELOPMENT OF LOSS-OF-OFFSITE-POWER FREQUENCY AND DURATION RELATIONSHIPS

INTRODUCTION

This appendix provides the details and results of analyses performed by NRC staff to develop the cause, frequency, and duration relationships for loss of offsite power at nuclear power plants. The purpose of this work was to develop generic loss of offsite power relationships that would allow differentiation of plant design, operational, and location factors that can significantly affect the expected frequency and duration of loss-of-offsite-power events. Within this study, the loss of offsite power has been defined as the interruption of the preferred power supply to the essential and nonessential switchgear buses necessitating or resulting in the use of emergency AC power supplies. A total loss of offsite power is said to have occurred when non-emergency AC power sources become unavailable requiring some diagnosis or special recovery actions, including correcting switching errors, fixing or bypassing faulted equipment, or otherwise making available an alternate standby source of non-emergency AC power.

Although total loss of offsite power is a relatively infrequent occurrence at nuclear power plants, it has happened a number of times, and a data base of information has been compiled (Wyckoff, 1986; NUREG/CR-3992). From these data and a review of relevant design and operational characteristics, the frequency and duration relationships for loss-of-offsite-power events at nuclear power plants have been developed. Historically, a loss of offsite power has occurred with a frequency of about once per 10 site-years. The typical duration of these events has been on the order of one-half hour. However, at some power plants the frequency of loss of offsite power has been substantially higher than the average, and in other instances the duration of offsite power outages has been much longer than the norm. In some cases, licensees have and are taking corrective action to limit the recurrence of these longer and more frequent losses of offsite power.

A summary of the data on the total loss-of-offsite-power events is provided in Table A.1. Because design characteristics, operational features, and the location of nuclear power plants within different grids and meteorological areas can have a significant effect on the likelihood and duration of loss-of-offsite-power events, it was necessary to analyze the nuclear industry experience in more detail. The data have been categorized into plant-centered events and area- or weather-related events. Plant-centered events are those in which the design and operational characteristics of the plant itself play a role in the likelihood or duration of the loss-of-offsite-power event. Area or weather effects include the reliability of the grid and external influences on the grid or at the site (such as severe weather) that have an effect on the likelihood and duration of the loss of offsite power. The data show that plant-centered events account for the majority of the loss-of-offsite-power events. Although the area-blackout- and weather-related events are less frequent, they typically

Table A.1 Summary of loss-of-offsite-power experience

Category	No. of events ($\geq$ ½ hr)	Frequency (yr^{-1}) ($\geq$ ½ hr)
Plant centered	46 (15)	0.087 (0.028)
Grid	12 (7)	0.018 (0.011)
Weather	6 (6)	0.009 (0.009)
Total	64 (28)	0.114 (0.048)

Note: The number of reactor-critical site-years through December 1985 is 527, and the number of site-years is 664.

account for the longer duration outages, with storms the major contributor to long outages. Because plant-centered events that occurred when reactors were shut down were screened from the event count, reactor-critical site-years were used to derive plant-centered event frequencies. Reactor-critical site-years are the number of years that reactors were at power conditions at the site.

Figure A.1 provides a plot of the frequency and duration of loss-of-offsite-power events resulting from plant-centered faults, grid blackout, and severe weather, based on past experience at nuclear plant sites. The curves were developed by fitting data to a two-parameter Weibull function of the following form:

$$\lambda_{LOP_i}(t) = \lambda_{LOP_i} e^{-(\alpha_i t^{\beta_i})}$$

where $\lambda_{LOP_i}(t)$ is the frequency of losses of offsite power of type "i," which are equal to or greater than duration "t." That is, the recovery time equals or exceeds "t" hours. The term λ_{LOP_i} is the frequency of occurrence of losses of offsite power of type "i," which have greater than zero duration. Parameters α_i and β_i are curve-shaping constants that vary according to the data being curve fitted.

Analyses were also performed to determine the trends in the frequency of loss of offsite power. Figure A.2 shows a plot of the rolling average loss of offsite power for nuclear plants included in Table A.1 and Figure A.1. These results show that over a period of 20 years, from 1966 through 1985, the general

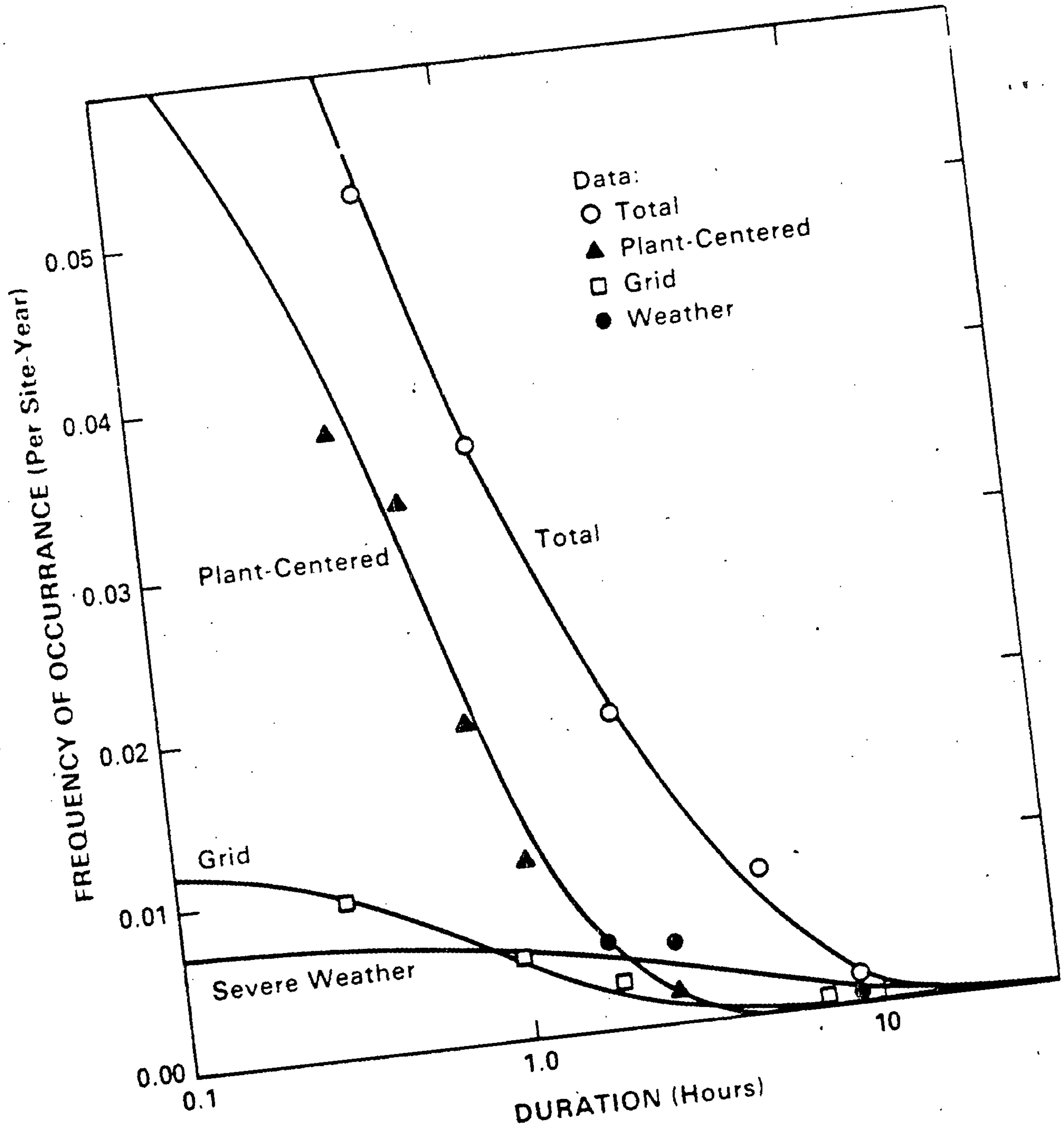


Figure A.1 Frequency of loss-of-offsite-power events exceeding specified durations

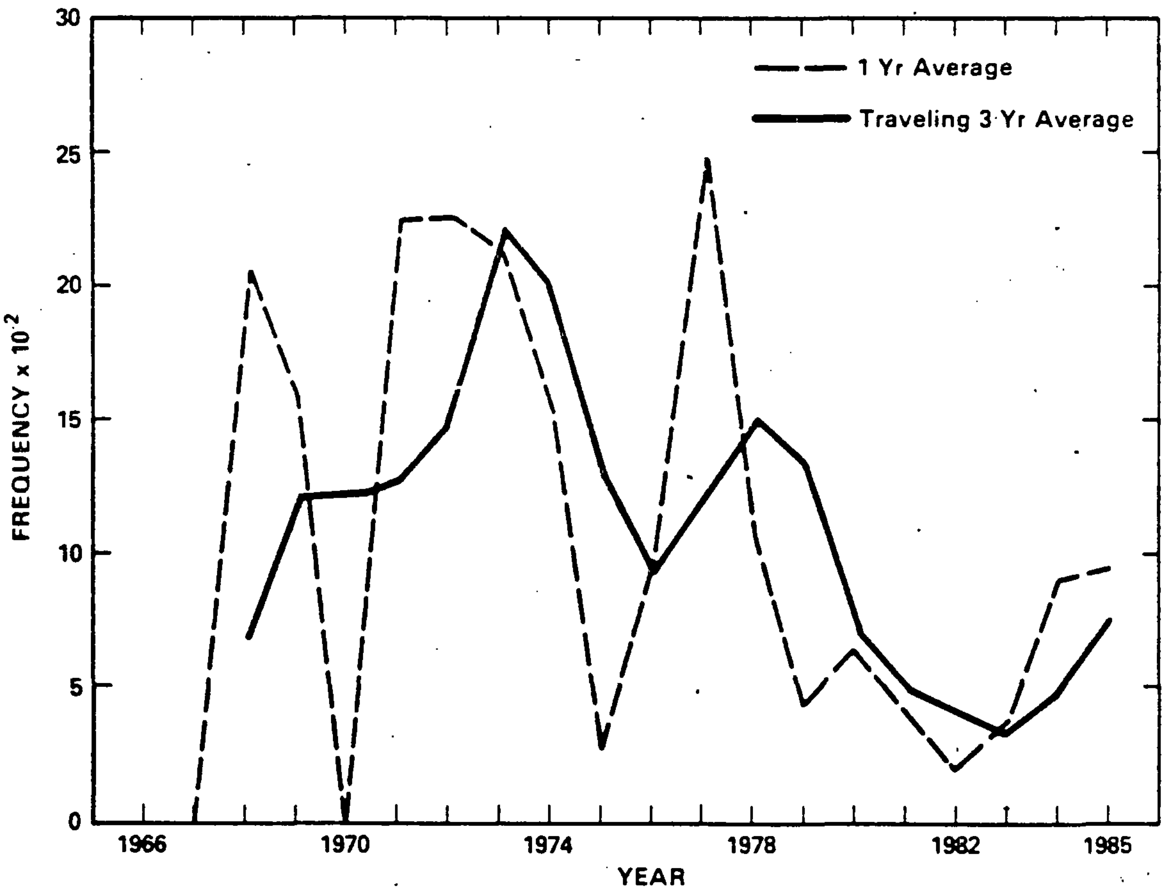


Figure A.2 Annual frequency of loss of offsite power

trend has been toward a reduction in loss-of-offsite-power frequency. However, that reduction in frequency has been modest. The results also show that fluctuations occur so that trends and averages indicated in any given interval of 2 or 3 years can be considerably different than the cumulative results. As of the end of 1985, the cumulative average frequency of loss of offsite power was about 0.1 while the trends from Figure A.2 indicate an industry-wide frequency variation ranging between 0.25 and 0.05 over the period.

LOSS OF OFFSITE POWER FROM PLANT-CENTERED CAUSES

Plant-centered failures typically involve hardware failures, design deficiencies, human errors (in maintenance and switching), localized weather-induced faults (lightning), or combinations of these failure types. Plant-centered failures can be recovered by switching or repairing faulted equipment at the site. An effort was made to screen out events that occurred when plants were shut down and offsite power configurations are not required to meet requirements for availability of immediate and delayed access circuits.

For the plant-centered losses, an attempt was made to determine any correlation between offsite power design characteristics and frequency and duration of losses of offsite power. Two offsite power design features were identified as potentially significant with regard to frequency and duration of loss of offsite power: (1) the independence of incoming offsite power sources and (2) the number of immediate and delayed access circuits and their transfer schemes to the Class 1E buses. Table A.2 defines the design differences associated with these features. The designs of offsite power sources were further subdivided into groups, and the number of shutdown sources were subdivided into different possible design combinations (NUREG/CR-3992).

The relationship between the listed design features and the frequency of loss of offsite power was analyzed using the Failure Rate Analysis Code (FRAC) (NUREG/CR-2434) to correlate loss-of-offsite-power frequency with various design features. These analyses showed no statistically significant correlations between frequency of plant-centered losses of offsite power and the design features analyzed.

An analysis was also performed to determine if any relationship exists between offsite power design characteristics and the duration of losses of offsite power. Analyses were performed using the generalized linear model (GLM) procedure of the Statistical Analysis System (SAS) (SAS Institute, 1979). The data for all of the different design factors were analyzed to check for any statistical interactions using analysis of variance. One data point--a 5.83-hour restoration time for an event at the Calvert Cliffs plant on April 13, 1978--was found to cause a strong interaction. Without that event, there was no significant interaction. The Calvert Cliffs event involved a latent design flaw that has since been corrected; it is not expected to typify future occurrences with regard to design feature, type of failure, or duration. With the data "corrected," the independence of offsite power sources was found to be an important determinant of the restoration time associated with plant-centered losses of offsite power. The number and type of transfer schemes were found to be less significant. It was concluded that various combinations of these design features could be used to define a set of design characteristics with different recovery times for plant-centered losses of offsite power. On the basis of this analysis and a

Table A.2 Definitions of offsite power system design factors

Major design factor	Design features
A. Independence of offsite power sources to the nuclear plant	1. All offsite power sources are connected to the plant through one switchyard.
	2. All offsite power sources are connected to the plant through two or more switchyards, and the switchyards are electrically connected.
	3. All offsite power sources are connected to the plant through two or more switchyards or separate incoming transmission lines, but at least one of the AC sources is electrically independent of the others.
B. Automatic and manual transfer schemes for the Class 1E buses when the normal source of AC power fails and when the backup sources of offsite power fail	1. If the normal source of AC power fails, there are no automatic transfers and there is one or more manual transfers to preferred or alternate offsite power sources.
	2. If the normal source of AC power fails, there is one automatic transfer but no manual transfers to preferred or alternate offsite power sources.
	a. All of the Class 1E buses in a unit are connected to the same preferred power source after the automatic transfer of power sources.
	b. The Class 1E buses in a unit are connected to separate offsite power sources after the automatic transfer of power sources.
	3. After loss of the normal AC power source, there is one automatic transfer. If this source fails, there may be one or more manual transfers of power sources to preferred or alternate offsite power sources.
	a. All of the Class 1E buses in a unit are connected to one preferred power source after the first automatic transfer.
	b. The Class 1E buses in a unit are connected to separate offsite power sources after the first automatic transfer.

Table A.2 (continued)

Major design factor	Design features
	4. If the normal source of AC power fails, there is an automatic transfer to a preferred source of power. If this preferred source of power fails, there is an automatic transfer to another source of offsite power. a. All of the Class 1E buses in a unit are connected to the same preferred power source after the first automatic transfer. b. The Class 1E buses in a unit are connected to separate offsite power sources after the first automatic transfer of power sources.

review of the design features, the staff concluded (1) that plants with switchyard designs that are normally operated as an interconnected system could be separated, as a group, from those with designs offering electrical independence, and (2) that sites with two or more alternate offsite power circuits (immediate or delayed access) in addition to the normally energized power circuit to the Class 1E buses (offsite or unit generator source) could be grouped. Table A.3 shows design combinations obtained with the mean-time-to-repair (MTTR) values for each group.

Other groupings can be derived that have at least some statistical significance and are physically valid. However, data limitations and small differences in MTTR that occur for more detailed breakdowns suggest that the design groups obtained represent a reasonable and valid compromise between completely generic and more design-specific breakdowns.

Table A.3 Mean time to restore offsite power

Group designation	Design factor*	Mean time to restore
I1	A1, A2, or A3 and B4	0.20
I2	A1 or A2 and B2b or B3	0.39
I3	A1 or A2 and B1 or B2a	0.78

*See Table A.2 for design features.

A plant-centered loss-of-offsite-power-frequency-vs.-duration curve was developed for each of the three design groups by fitting the corresponding data to a two-parameter Weibull distribution. A list of the data used for each curve fit is given in Table A.4. The actual curves generated by this analysis are in Figure A.3. The curves show the probability and frequency of events that exceed a specified duration. Figure A.4 shows the 90% confidence limits for two of the correlations (I1 and I3) derived using the extreme value theory.

GRID-RELATED LOSS OF OFFSITE POWER

Grid reliability has traditionally been the most prominent factor associated with a loss of offsite power at nuclear power plants. Yet, the historical data show that losses of offsite power as a result of grid-related problems account for no more than 19% of all losses of offsite power. Attempts to find characteristics to classify site, design, and location features that affect the expected frequency of grid loss have not been successful. An investigation into the various utility transmission and distribution system reliability characteristics was beyond the scope of this study. Such a study is likely to involve an extensive state-of-the-art analysis of grid stability, the results of which would be of questionable validity considering limitations on current methodology. In its place a more pragmatic and experience-based approach to estimating nuclear plant site susceptibility to grid loss was taken. Both frequency of grid loss and time to restore power were considered.

It was recognized that the Florida Power and Light (FPL) grid has represented the upper end of utility grid failure frequency during the past 10 to 15 years, although some recent improvements seem to have been effective. Very few other nuclear plant sites have experienced even one or two loss-of-offsite-power events as a result of grid blackout. The great majority of nuclear power plants have not experienced grid failure. A systemic weakness identified after a grid failure is usually corrected as soon as possible. Thus, it is usually a new and previously unidentified systemic weakness that results in future failures. Therefore, in the absence of known and uncorrected systemic weaknesses, the occasional, non-recurring type of grid failure may not be a good indicator of future trends within a utility system. With this in mind, the FPL experience was separated from the balance of the U.S. nuclear utility experience to estimate grid-failure frequency. Because a set of design or location factors could not be identified that could effectively differentiate the expected reliability of the various utility grids, grid reliability was categorized by failure frequency ranges characteristic of past experience. The FPL experience suggests an upper end to the grid-failure frequency of once per 2 to 5 site-years, although there have been recent improvements. In a few utility systems, the occasional grid failures have occurred at a frequency of about once per 10 to once per 20 site-years. The national average is about once per 100 site-years, excluding FPL experience. Table A.5 lists grid-related losses of offsite power and site-specific frequencies calculated from the data. Two grid undervoltage events are discussed in a footnote to the table. Although these events were not counted as grid failures, offsite power sources were momentarily unavailable during these events.

Two factors that have been identified as significant in determining the duration of grid-related losses of offsite power at nuclear power plant sites are: (1) the availability of adequate restoration procedures and (2) the availability of "black start" power sources that are able to supply power to a nuclear

Table A.4 Data used for plant-centered loss-of-offsite-power-duration curve fits*

Group**	Plant	Date	Duration (hr)
I1	Davis-Besse	11/29/77	0.002
	Nine Mile Point	11/17/73	0.003
	Oconee	01/04/74	0.013
	Haddam Neck	07/19/72	0.017
	Millstone	07/21/76	0.080
	Haddam Neck	07/15/69	0.150
	Haddam Neck	08/01/84	0.167
	Susquehanna	07/26/84	0.183
	Monticello	04/27/81	0.250
	Haddam Neck	06/26/76	0.270
	Haddam Neck	01/19/74	0.330
	Davis-Besse	10/15/79	0.430
	Haddam Neck	04/27/68	0.480
	Indian Point 2,3	06/03/80	0.500***
I2	Oyster Creek	09/08/73	0.003
	Point Beach	04/27/74	0.020
	Brunswick	03/26/75	0.070
	Dresden	08/16/85	0.083
	Point Beach	02/05/71	0.130
	Turkey Point	02/12/84	0.250
	Turkey Point	02/16/84	0.250
	Beaver Valley	07/28/78	0.280
	McGuire	08/21/84	0.334
	Ginna	03/04/71	0.500
	Ginna	10/21/73	0.670
	Prairie Island	07/15/80	1.030
	Arkansas Nuclear One	09/16/78	1.480
I3	San Onofre	11/22/80	0.004
	Fort Calhoun	08/22/77	0.015
	San Onofre	11/21/85	0.067
	Palo Verde	10/07/85	0.200
	Palo Verde	10/03/85	0.400
	Palisades	09/24/77	0.500
	Quad Cities	06/22/82	0.570
	Farley	09/16/77	0.900
	Fort Calhoun	02/21/76	0.900
	Palisades	09/02/71	0.930
	Quad Cities	11/06/77	1.150
	Indian Point	06/03/80	1.750***
	Farley	10/08/83	2.750

(See next page for footnotes)

Table A.4 - Footnotes

*Not included in the duration analysis were the Palisades events of 11/25/77 and 12/11/77 (recurring failures), the Calvert Cliffs event of 04/13/78 (outlier), the Big Rock Point event of 11/25/72 (insufficient plant design information), and the Crystal River event of 06/16/81, the Vermont Yankee event of 12/17/72, and the Turkey Point event of 04/04/79 (incomplete reporting of duration).

**Group designations are explained in Table A.3.

***The Indian Point event of 06/30/80 lasting 1.75 hours, included in Group I3, is also included as a 0.50-hour event in Group I1 on the basis that had the available gas turbine been employed, offsite power would most likely have been recovered in approximately 30 minutes.

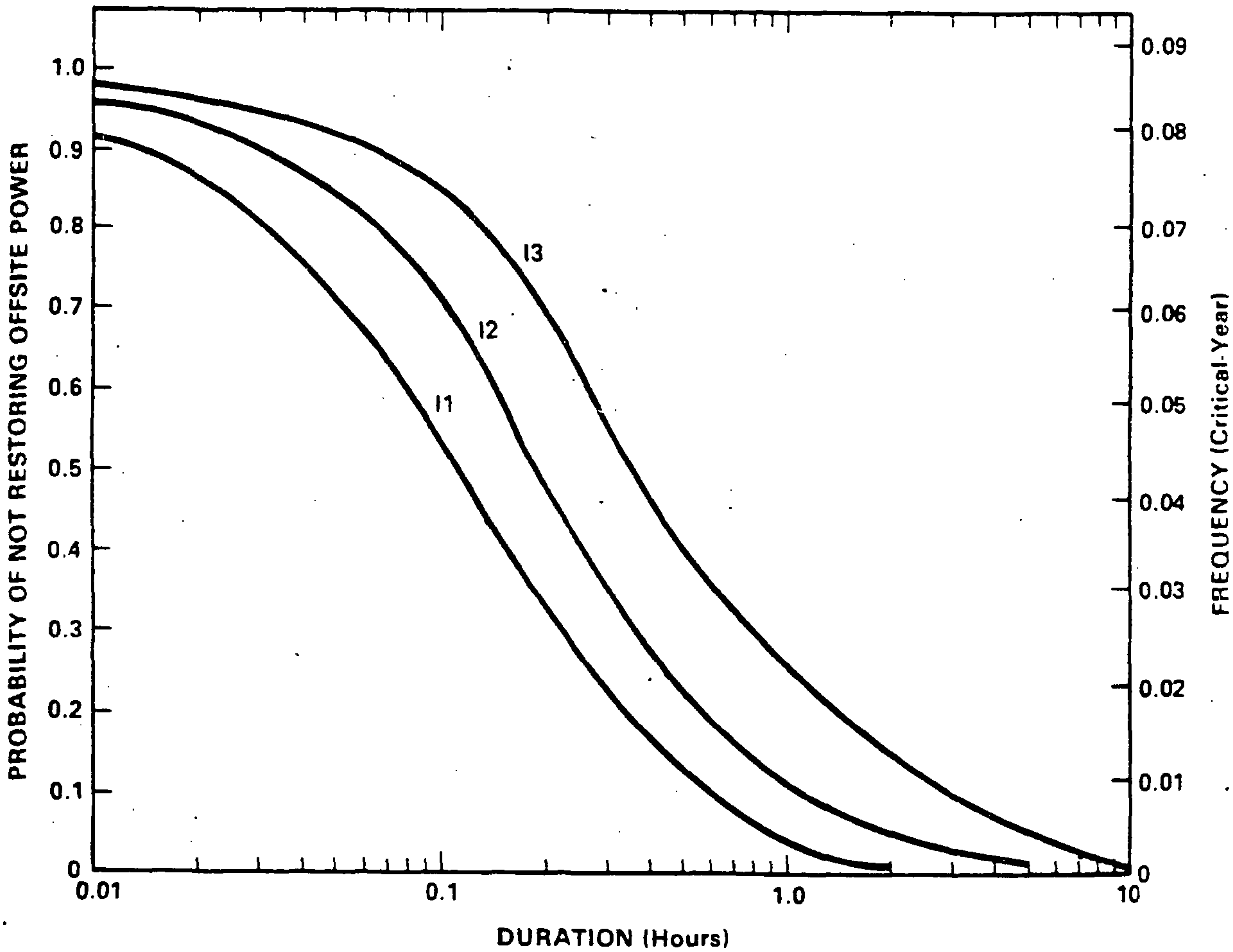


Figure A.3 Estimated frequency of occurrence of plant-centered losses of offsite power exceeding specified durations (for offsite power groups as shown in Table A.3)

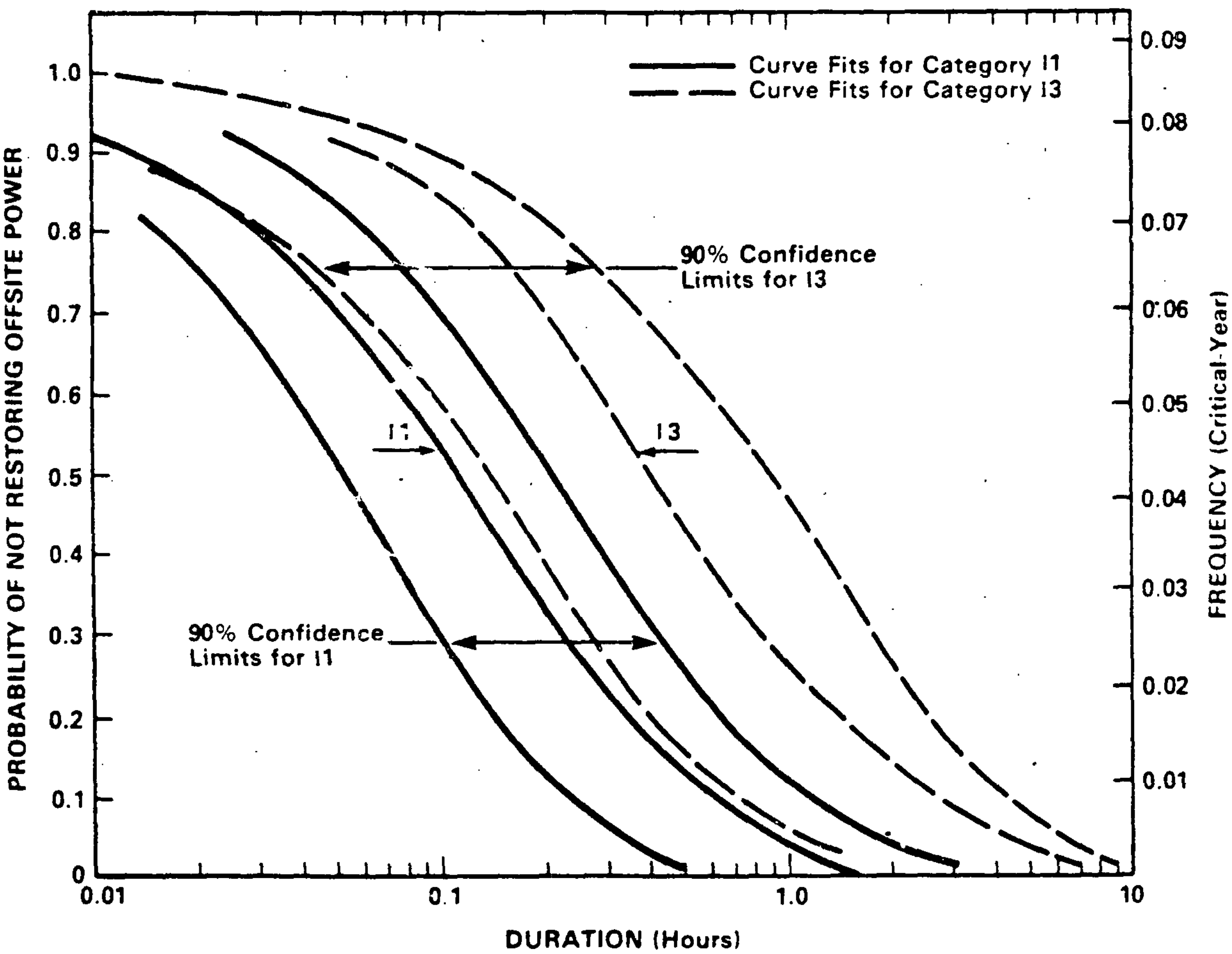


Figure A.4 90% confidence limits for two categories of plant-centered losses of offsite power

Table A.5 Grid-related loss-of-offsite-power frequency versus duration, through December 1983

Site	Date of occurrence	Duration (hours)	Site frequency (per year)
Turkey Point:	06/28/74	0.180	0.444 (6 events in 13.5 site-years)
	04/04/73	0.250	
	04/03/73	0.300	
	04/25/74	0.330	
	05/16/77*	1.030	
	05/16/77*	2.000	
	05/17/85	2.083	
Indian Point:	07/20/72	0.920	0.126 (3 events in 23.8 site-years)
	07/13/77	6.470	
	11/09/65	**	
St. Lucie:	05/14/78	0.130	0.20 (2 events in 9.8 site-years)
	05/16/77*	0.330	
	05/16/77*	1.500	
Yankee Rowe:	11/09/65	0.550*	0.039 (1 event in 25.5 site-years)
60 sites:	none***		(no events in 0.3 to 26.3 site-years)
Total for 64 sites			0.018 (12 events in 664.4 site-years)
Total excluding FPL			0.006 (4 events in 664.4 site-years)

*The Turkey Point and St. Lucie events of 05/16/77 were counted as one event for each plant for frequency calculations.

**Actual duration not reported.

***The undervoltage event at Millstone on 07/21/76 was treated as a plant-centered design problem; the undervoltage event at Quad Cities on 02/13/78 was treated as a degradation with a usable offsite power source available throughout the incident.

power plant in isolation of a grid disturbance. Both of these factors can contribute to a significant reduction in the expected duration of grid-related losses of offsite power, as reported in the Indian Point Safety Study (Power Authority of the State of New York, 1982). In 1981 the NRC sent Generic Letter 81-04 to all nuclear power plant licensees requesting them to develop and implement procedures to enhance restoration of offsite power. Responses to that generic letter have indicated that power could be preferentially restored to many nuclear power plant sites within 1 or 2 hours, even if the grid remained in a blackout condition.

The time to restore offsite power following a grid failure can be estimated by past experience. However, if an appropriate set of procedures is provided and power sources are available and capable of supplying power during grid blackout, a more prompt recovery may be possible. Human reliability and the availability of alternate power sources may limit the recovery potential to as low as 60% recovery in about an hour. If multiple reliable sources of power that can be isolated from a blacked-out grid are available, the potential may be as high as 95% recovery in less than one-half hour. For this study, an offsite-power-restoration likelihood of 80% within one-half hour of a grid failure was assumed for the analysis of plant sites with enhanced recovery capabilities (e.g., procedures and at least one power source available for prompt recovery). The recovery probabilities for grid-related losses of offsite power were developed by fitting past operating data to a two-parameter Weibull distribution. The data used in the curve fit are provided in Table A.5. Figure A.5 provides a curve showing the probability of not restoring offsite power versus the duration of losses of offsite power as a result of grid blackouts. It also shows the potential for improvement with enhanced recovery capability over past operating experience.

The correlations for grid reliability and offsite power restoration were developed by combining the occurrence frequencies representative of operating experience and the calculated recovery probabilities. Table A.6 provides the grid failure frequency and duration groups obtained. Figure A.6 shows the discrete loss-of-offsite-power frequency and duration curves corresponding to the groups identified in Table A.6.

LOSS OF OFFSITE POWER AS A RESULT OF SEVERE WEATHER

Severe weather conditions, such as local or area-wide storms, have caused losses of offsite power at nuclear power plants. Weather-related causes of offsite power failure have been divided into two groups

(1) those for which the weather caused the event but did not affect the time to restore power

(2) those for which the weather initiated the event and created conditions so that power was not or could not have been restored for a long time

Group (1) includes lightning and most other weather events that do not cause severe or extensive physical damage at or near the site. They can cause a loss of offsite power, but their severity does not contribute in any significant way to long-duration losses of offsite power. These types of weather-related offsite power outages are usually considered in the plant-centered or, possibly, the grid category. Group (2) includes losses of offsite power that result from

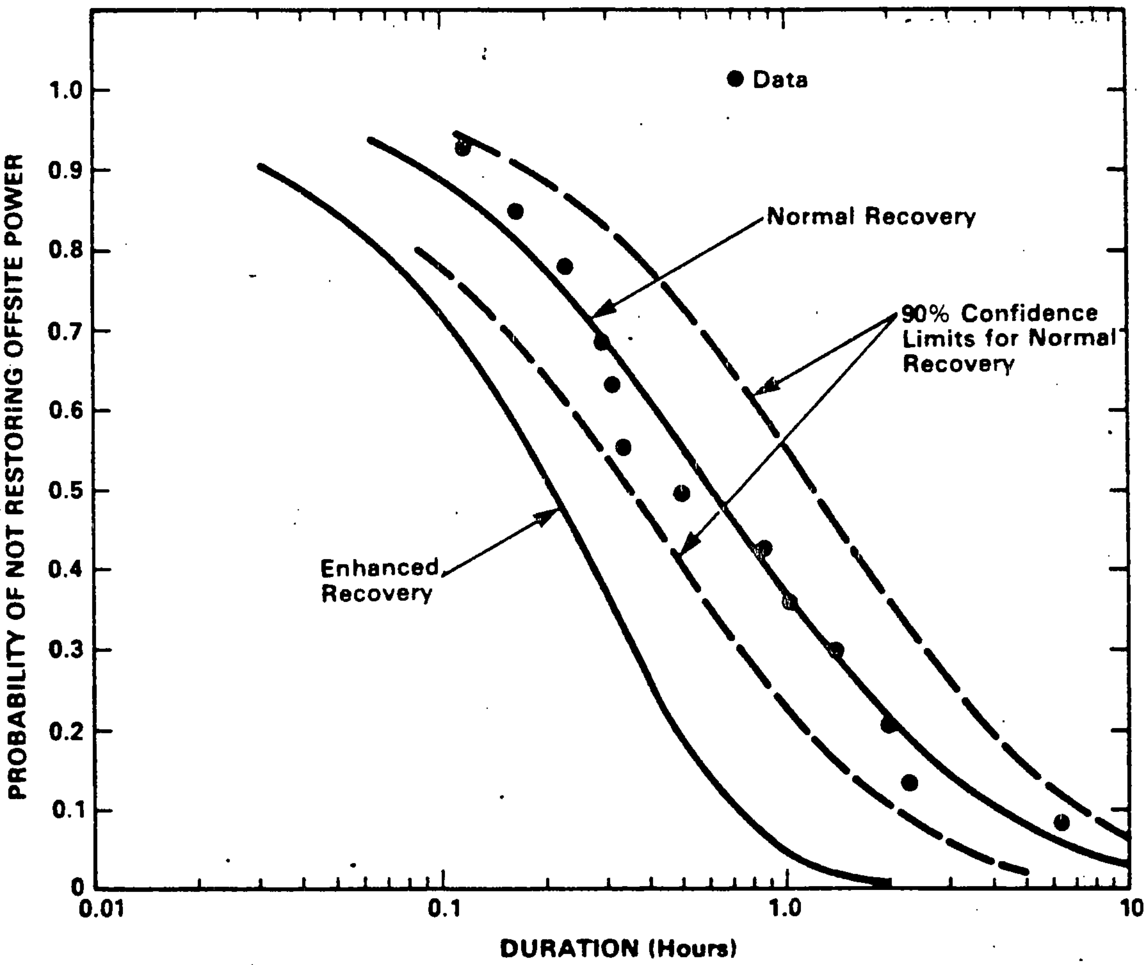


Figure A.5 Restoration probability for grid-related losses of offsite power

Table A.6 Grid reliability/recovery

Group	Grid loss frequency, reliability recovery	
Grid reliability group (G):	Frequency of grid loss:	
G1	Less than 1 per 60 site-years (0.01/site-year)	
G2	≥ 1 per 60 site-years and ≤ 1 per 20 site-years (0.03/site-year)	
G3	> 1 per 20 site-years and ≥ 1 per 6 site-years (0.1/site-year)	
G4	Greater than or equal to 1 per 6 site-years (0.3/site-year)	
Recovery from grid blackout group (R):	Recovery capability:	
R1	Plant has capability and procedures to recover offsite (nonemergency) AC power to the site within 1/2 hour following a grid blackout.	
R2	All other plants not in R1.	
Grid reliability/recovery group (GR):	Grid reliability group (G):	Recovery from grid blackout group (R):
GR1	G1	R1
GR2	G2	R1
GR3	G3	R1
GR4	G4	R1
GR5	G1	R2
GR6	G2	R2
GR7	G3	R2

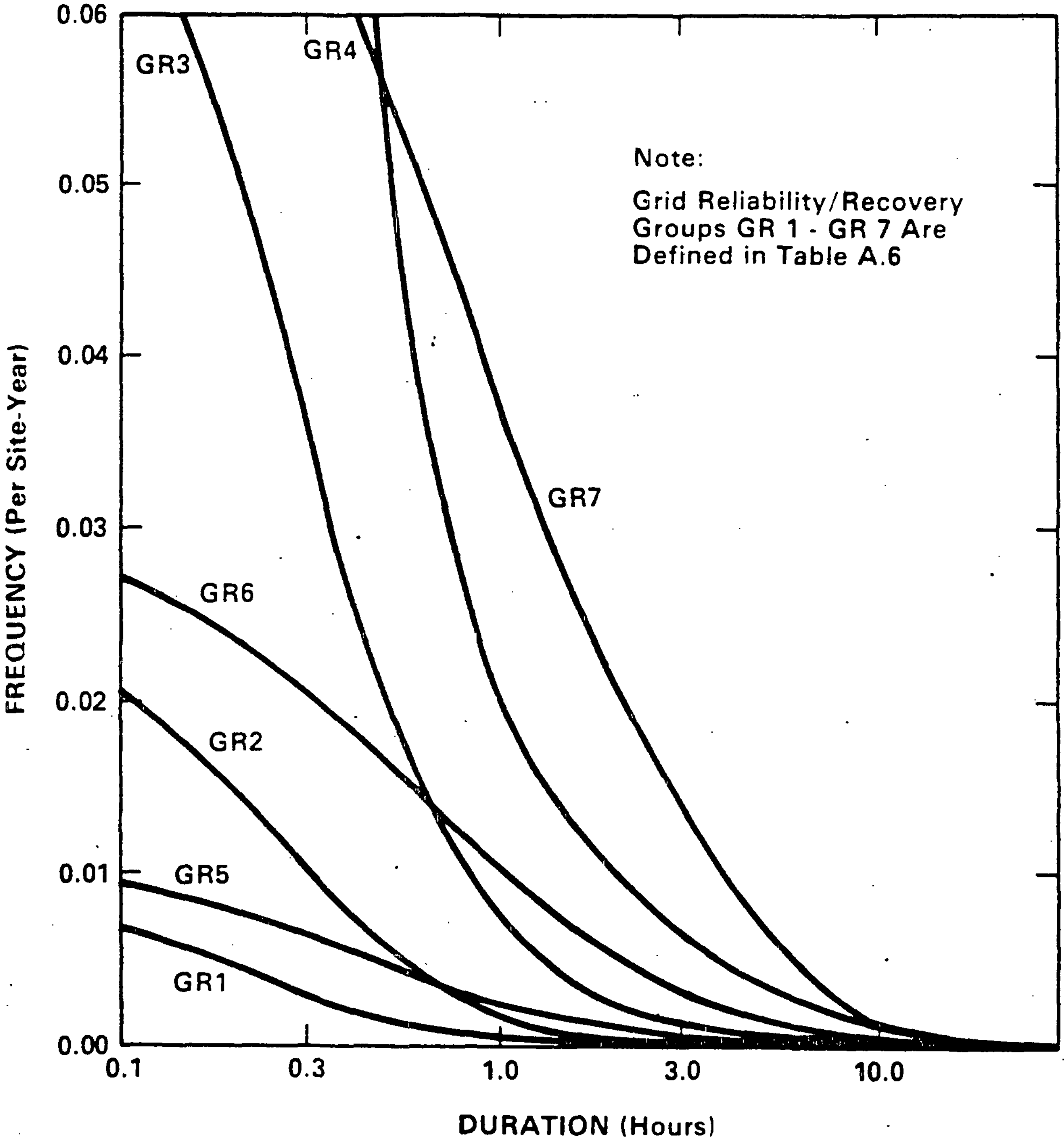


Figure A.6 Estimated frequency of occurrence of grid-related losses of offsite power exceeding specified durations

major storms, hurricanes, high winds, accumulations of snow and ice, and tornadoes. The expected frequency of loss of offsite power of this group is relatively small; on the other hand, for this group the likelihood of restoring offsite power in a short time is also relatively small.

To estimate the likelihood and duration of loss of offsite power as a result of severe weather, it is necessary to (1) identify the set of weather hazards to be considered, (2) determine the likelihood of failure for a given hazard intensity, and (3) determine the repair or restoration time for the various failure modes associated with severe weather-related power losses. Although utilities and regional power pools normally keep extensive data on transmission line, terminal, and customer outages from all causes, including weather, little information has been obtainable that can be used to derive the likelihood of loss of all offsite power at nuclear plants or for similarly designed incoming transmission lines and switchyards at non-nuclear plants. In light of this limitation, the objective of this study was to derive some general frequency and duration characteristics that could be applied to the design and location of nuclear power plant offsite power systems generically or on a case-by-case basis, considering specific susceptibility to the various weather hazards.

The approach taken was to develop a range of loss-of-offsite-power frequency and duration relationships based on weather hazard rate and past operating experience. First, data for all loss-of-offsite-power events involving both partial or total failures were reviewed. Weather-related total loss-of-offsite-power events and significant partial loss-of-offsite-power events, such as those causing the complete loss of power to or from a switchyard, were included. These data are provided in Table A.7. Here again, as with grid reliability experience, this data base is too small to be used to derive plant location and design-dependent conclusions regarding the expected frequency of loss of offsite power as a result of severe weather.

Normally, regression analyses would be used to correlate failure rate, design factors, and weather hazards. However, the losses of offsite power are so rare that the available data are too limited to take such an approach. The method used to correlate loss-of-offsite-power frequency to weather hazards is based on the assumption that the frequency of loss of offsite power as a result of severe-weather events is proportional to the weather hazard rates at a site. The weather hazard rate is a measure of the frequency of conditions that have the potential to cause loss of offsite power. The following weather hazard rate indicators were selected:

- snow/ice: inches of snowfall per year
- tornado: frequency of tornadoes per year
- hurricane and wind: frequency of storms per year with wind speeds of tropical storm strength or greater

These factors are called indicators because no mechanistic cause and effect analysis has been performed to associate their occurrence with a loss of offsite power. Rather, it has been observed that losses of offsite power have occurred when these types of weather conditions were present. For instance, winter and spring snowstorms, which can be measured according to inches of snowfall, also bring conditions involving ice accumulations on lines and terminals. Windy conditions may also accompany these storms. Thus, a hazard indicator of inches

Table A.7 Severe-weather-induced losses of offsite power used in the analysis

Type loss/site	Date	Duration (hours)	Weather type
Total Losses of Offsite Power:			
Fort St. Vrain	05/17/83	1.75	Snow/Ice
Pilgrim	05/10/77	2.67	Snow/Ice
Dresden	11/12/65	4.00	Tornado
Millstone	08/10/76	5.00	Salt Spray
Millstone	09/27/85	5.50	Salt Spray
Pilgrim	02/06/78	8.90	Snow/Ice
Major Partial Losses of Offsite Power:			
Browns Ferry	03/01/80		Snow/Ice
D. C. Cook	02/04/78		Snow/Ice
Pilgrim	10/12/82		Salt Spray
San Onofre	02/24/69		High Wind
Brunswick	09/13/84		Hurricane/Wind
Arkansas Nuclear One	02/22/75		Tornado
Arkansas Nuclear One	04/07/80		Tornado
Browns Ferry	04/03/74		Tornado

of snowfall is merely a factor used to correlate loss-of-offsite-power occurrences with locations most susceptible to winter and spring storms involving snow and ice accumulations and associated windy conditions.

A similar situation exists with regard to tornado hazards. The expected frequency of tornadoes in the vicinity of the plant was used as a factor to correlate actual losses of offsite power resulting from tornado strikes.

Hurricane and high wind conditions can cause losses of offsite power by blowing debris, falling trees, and other possible modes of falling lines and shorting terminals. Storms are classified as hurricanes when wind speeds sustain 75 mph. The frequency of this wind speed was used as a correlation point to determine the variability of hurricanes and high wind hazards at various locations (sites).

A special subgroup was identified for hurricane and wind losses at plants adjacent to the seacoast or large bodies of salt water. This subgroup was formed in response to experience at the Millstone and Pilgrim sites where high winds associated with storms and hurricanes caused salt buildup on switchyard insulators, which then resulted in arcing and faulting of the switchyard.

By dividing the number of losses of offsite power that have occurred by the cumulative historical weather hazards for each weather type at nuclear power plant sites, an offsite power failure proportionality factor for each weather type was derived. This process can be represented as follows:

$$P_i = \frac{N_i}{\Sigma H_{ji}}$$

where

P_i = the proportionality factor for weather type "i"

N_i = the observed number of offsite power losses as a result of weather type "i"

H_{ji} = the cumulative weather hazard factor for weather type "i" at site "j"

$$H_{ji} = h_{ji} \, \Delta t_j$$

where

h_{ji} = the weather hazard rate for type "i" weather at site "j"

Δt_j = the cumulative site-years since commercial operation began at site "j"

The expectation frequency of loss of offsite power can then be computed by

$$S_{ji} = P_i \, h_{ji}$$

where S_{ji} is the estimated frequency of loss of offsite power at site "j" for weather type "i", and P_i and h_{ji} are defined as before.

Weather-induced failure proportionality factors were derived using the data from Table A.7 and cumulative weather hazards data for U.S. nuclear power plant sites through 1985. The weather hazard factors for each site were derived from National Weather Service data where available (Batts et al., 1980; National Oceanic and Atmospheric Administration, 1980; Neumann et al., 1985; Shaefer et al., 1985; Simiu et al., 1979) and from site-specific probability calculations performed by the National Severe Storms Forecast Center. The proportionality factors from hurricane/high wind and tornadoes were derived for several subgroups to account for plant design or location features which may result in variations in the probability of offsite power losses resulting from these weather conditions.

As discussed previously, hurricane and high wind conditions which can induce salt spray to unprotected switchyard components near bodies of salt water were separated from other potential causes of hurricane/high wind induced losses of offsite power (e.g., falling trees and blowing debris). Since no total losses of offsite power were reported for the latter type of hurricane/high wind conditions, the median value of the chi-square for zero failures and two degrees of freedom was used as a bound.

A tornado hazards loss-of-offsite-power proportionality factor was derived for plants with single or closely spaced rights-of-way emanating from the plant and for plants with multiple, divergent rights-of-way. The data in Table A.7 involve losses of lines on single rights-of-way or multiple line losses on some but not all rights-of-way. Therefore, these data were used to derive the proportionality factor for sights with single or closely grouped rights-of-way. Since no occurrences of tornadoes causing total loss of offsite power at sites with multiple, divergent rights-of-way have been reported, the median value of the zero failure chi-square statistic was used to approximate this proportionality factor.

On the basis of the analyses described above, the following weather-induced failure proportionality factors were derived:

$P_{S/I}$ = 1.3×10^{-4} inches of snowfall

$P_{H/W}$ = 1.2×10^{-2} for windspeeds $\geq$ 75 mph

P_{SS} = 0.783 for windspeeds $\geq$ 50 mph

P_{T1} = 72.3 for single rights-of-way or equivalent

P_{T2} = 12.5 for multiple divergent rights-of-way

where subscripts S/I = snow/ice, H/W = hurricane/high wind, SS = salt spray, and T1 and T2 refer to tornadoes.

Normally this type of correlation would be supported by a statistical validity test. As pointed out previously, because there have only been a few weather-related losses of offsite power at nuclear plants, the statistical validity could not be ascertained. However, as a test of the reasonableness of this formulation, a plot of cumulative weather hazard factor for each site (H_i) versus total cumulative weather hazard factor tabulated for all applicable nuclear plant sites (ΣH_i) was made, and the severe weather-related operating experience for both total and major partial loss-of-offsite-power events was identified. A comparison was also made of the number of sites falling within subdivisions of the range of cumulative weather hazard factors. This information is provided in Figure A.7, where the number of losses of offsite power followed by a "T" represent total losses of offsite power and those followed by a "P" represent major partial losses of offsite power. Because frequency of loss of offsite power as a result of weather has been assumed to be proportional to the magnitude of weather hazards, the occurrence of weather-related losses of offsite power should favor the sites with the highest cumulative weather hazard. In general it does.

The events identified in Table A.7 are typified by durations of several hours. The failures are somewhat localized, able to be isolated, or repairable with modest effort. Design factors such as transmission line right-of-way separation, structural strength of transmission and switchyard components, insulation from effects of adverse environments, and operational factors related to repair capability or use of alternate, available power sources will impact the likelihood and duration of loss-of-offsite-power events of this type. Events of this type will be referred to as severe-weather events throughout this appendix.

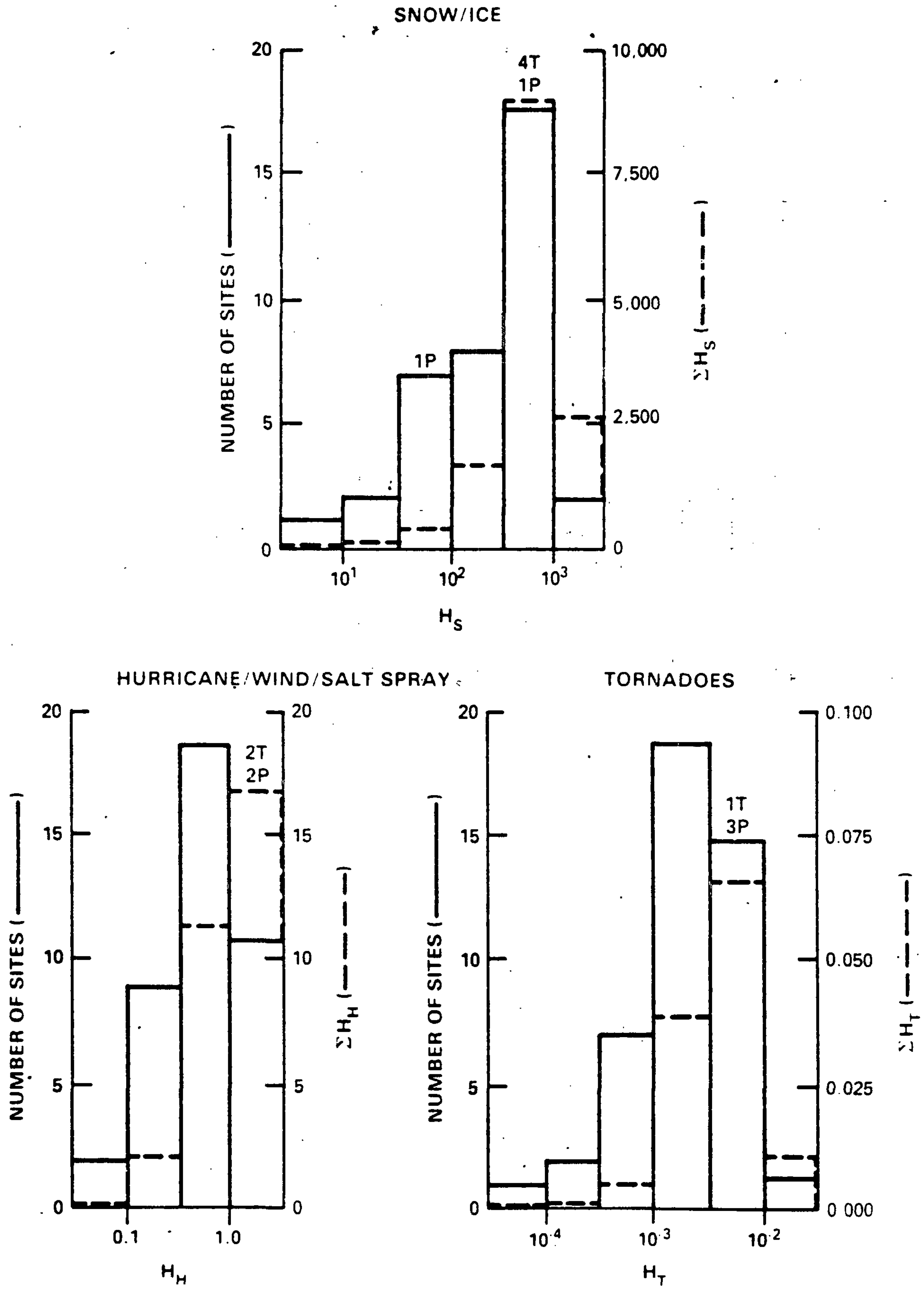


Figure A.7 Weather hazard expectation histograms

None of the events identified in Table A.7 involved tornado or hurricane/high wind conditions that severely damaged structural elements of all transmission and/or switchyard components of sources of offsite power to the plant. Although such an occurrence is rarely expected, many hours or days could be required to repair and restore offsite power.

The frequency of these more extreme weather-related power losses can be estimated by determining the frequency of weather conditions that are severe enough to damage all offsite power sources. The same design factors noted above for the more repairable loss-of-offsite-power events will determine the susceptibility, and thus frequency, or hazard rate, of weather conditions that could result in area-wide transmission and/or switchyard failures. Based on the National Electric Safety Code, power plant transmission systems should be designed for wind speeds on the order of 125 mph. High wind speeds could cause extensive power transmission losses, although this will vary, depending on the specific design. Another potential hazard, tornado(es), must strike all rights-of-way or switchyards with sufficient intensity to damage the minimum number of components required to supply offsite power in order to cause a long-duration loss of offsite power. The probability of equipment failure given the occurrence of these extreme weather conditions is assumed to be unity, or nearly so; thus the likelihood of loss of offsite power can be approximated by the frequency of occurrence of the extreme weather condition. The frequencies of the extreme hurricane (known as great hurricanes) and high winds are available from National Weather Service data.

To estimate the frequency of single or multiple tornado strikes damaging all transmission lines or switchyard components requires modeling of the offsite power transmission line geometry (Anders et al., 1984; Teles et al., 1980) and using site/area data for tornado frequency, intensity, and direction. This type of mechanistic, probabilistic analysis was not performed as part of this work. A simpler approach was used. The frequency of tornadoes of intensity F2 or greater (> 113 mph wind speeds) striking at any point within the site was obtained. Since this frequency for tornado strikes can be considered to occur any where at the site, it has been used as the frequency of tornado strikes at the switchyard or transformers. This represents the frequency of losses of offsite power as a result of tornado strikes that require significant repair effort and time. Since tornado strikes crossing all rights-of-way are not included in this simplified approach, the frequencies estimated will underpredict the actual frequency of long-repair-time losses of offsite power as a result of tornado strikes. However, the repairable losses of offsite power resulting from tornado strikes have been included in the overall model previously discussed, using the hazard and proportionality factor approach. And the median repair time of about 4 hours should adequately account for repairable tornado-associated losses in light of the overall uncertainty of the simplified modeling and analyses used.

Events of the types discussed in the preceding two paragraphs are referred to as extreme weather events throughout this appendix. Although the frequency of these extremely severe-weather events could be as high as 0.01 per site-year, it will more typically be less than 0.001 per site-year.

The time necessary to restore a source of offsite power for weather-related failures will depend on the severity of damage caused by the event. Major

structural damage can typically require 8 to 24 hours or longer for repair. Data obtained from the Mid-America Interpool Network (MAIN) and the Mid-Continental Area Power Pool (MAPP) (MAIN, 1983; MAPP, 1983) indicate that it takes on the order of 8 to 12 hours to restore transmission or terminal point outages that resulted from severe weather. For this study, nuclear power plant outage time data for losses of offsite power that resulted from severe weather were used to estimate restoration likelihood for the less-than-catastrophically-damaging weather events. Data for total loss-of-offsite-power events were fitted to a two-parameter Weibull distribution and used to generate the restoration likelihood curve shown in Figure A.8. Also shown in Figure A.8 is an example of an "enhanced" recovery curve that can be used to differentiate plants with practicable power restoration procedures for these weather types. The applicability of enhanced recovery shown depends on the capability and procedures to restore power within about 2 hours for a given weather hazard.

An estimate of the total severe-weather-related frequency of loss of offsite power was derived by summing the values for each weather hazard type at all nuclear plant sites. Plant-specific design or procedural details can affect the estimated frequency of weather-related losses of offsite power. Therefore, an attempt was made to derive the range of possibilities rather than to provide site-specific estimates. It should be noted, however, that, because of a lack of data, not all weather hazards could be accounted for at every site. Moreover, some weather data extrapolations were necessary when data from weather stations near a site were not available. The frequency range derived was large, and determining where a particular site/design combination would fall in that range requires evaluation of the site-specific details identified previously. For the purpose of this work, the range was subdivided into groups with approximately a factor of 3 difference in median frequency. The subranges so derived are provided in Table A.8. This partitioning allowed generic evaluation of the effects of severe weather hazard on loss-of-offsite-power frequency while at the same time providing perspective on the potential for plant-specific differences. Figure A.9 shows the severe weather frequency and duration combinations corresponding to the groups defined in Table A.8.

For losses of offsite power caused by extremely severe weather--such as great hurricanes, very high winds (greater than 125 mph), and major damage from tornado strikes to a switchyard--restoration of offsite power was not assumed to occur before 24 hours after the start of the outage. The frequency breakdowns, derived in a manner similar to that for severe weather, are provided in Table A.9. Again it must be noted that a site-specific assessment of the susceptibility to these weather hazards must be performed to determine the site-specific expectation frequency.

GENERIC LOSS-OF-OFFSITE-POWER CORRELATIONS

Combinations of design, grid, and weather factors derived in the previous sections provide a wide spectrum of possibilities for loss-of-offsite-power frequency and duration. Each of these factors was subdivided to account for known or hypothetical but reasonable differences in frequency and duration; typically, a factor of 2 to 5 difference was maintained for these subdivisions. The intent was to develop a discrete set of frequency and duration groups that could account for actual and potential differences in both design and location (grid and weather) for the spectrum of nuclear power plant sites. The frequency of losses

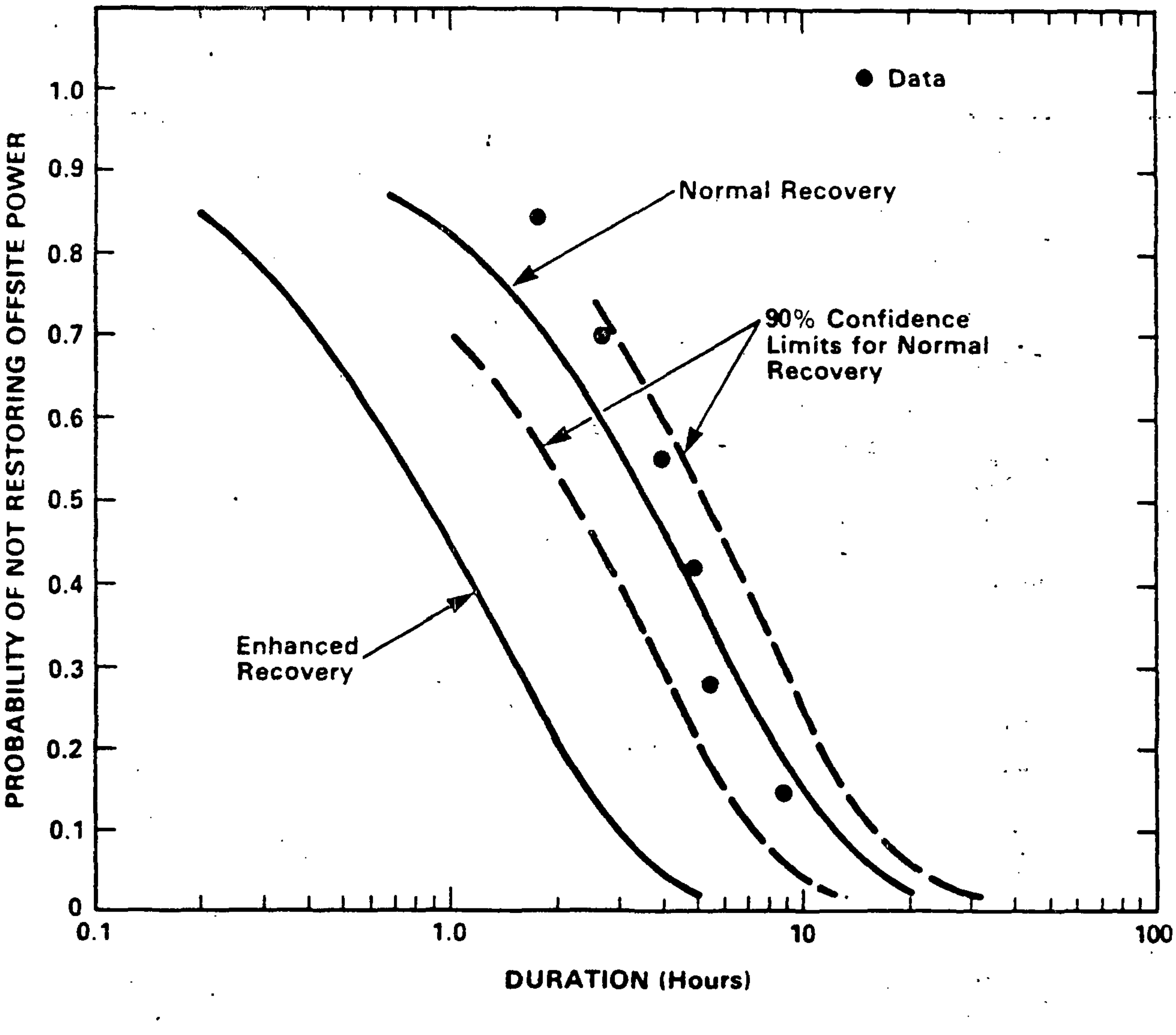


Figure A.8 Restoration probability for severe-weather-induced losses of offsite power

Table A.8 Severe-weather-induced loss-of-offsite-power frequency/recovery

Group	Duration combination	
Frequency of severe-weather-induced loss of offsite power group(s):	Frequency:	
S1	Less than 1 per 333 site-years (0.002)	
S2	1/333 to 1/100 site-years (0.005)	
S3	1/100 to 1/33 site-years (0.02)	
S4	1/33 to 1/10 site-years (0.05)	
S5	1/10 to 1/3 site-years (0.2)	
Recovery from severe-weather-induced loss-of-offsite-power group (R):	Recovery capability:	
R1	Plant has capability and procedures to recover offsite (nonemergency) AC power to the site within 2 hours following a severe-weather-induced loss of offsite power.	
R2	All other plants not in R1.	
Severe-weather-induced loss-of-offsite-power frequency/recovery group (SR):	Frequency group (S):	Recovery group (R):
SR1	S1	R1
SR2	S2	R1
SR3	S3	R1
SR4	S4	R1
SR5	S5	R1
SR6	S1	R2
SR7	S2	R2
SR8	S3	R2
SR9	S4	R2
SR10	S5	R2

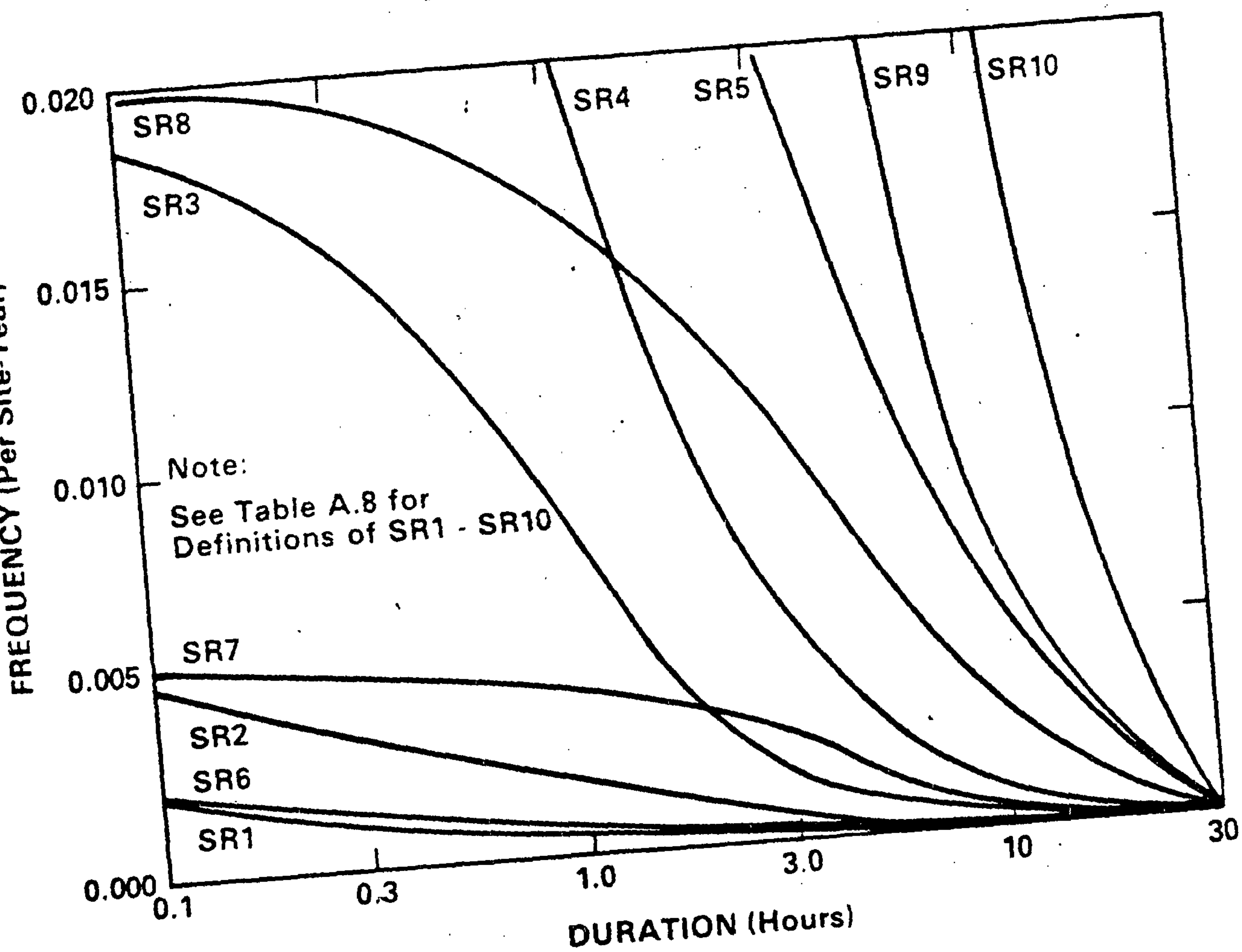


Figure A.9 Estimated frequency of occurrence of severe-storm-induced losses of offsite power exceeding specified durations

Table A.9 Extremely severe-weather-induced loss-of-offsite-power frequency

Extremely severe-weather-induced loss-of-offsite-power frequency group (SS)	Frequency
SS1	Less than 1 per 3333 site-years (0.0002/site-year)
SS2	$\geq$ 1 per 3333 site-years and $<$ 1 per 1200 site-years (0.0005/site-year)
SS3	$\geq$ 1 per 1000 site-years and $<$ 1 per 333 site-years (0.002/site-year)
SS4	$\geq$ 1 per 333 site-years and $<$ 1 per 100 site-years (0.005/site-year)
SS5	Greater than or equal to 1 per 100 site-years (0.02/site-year)

of offsite power lasting duration "t" or longer can be estimated by an appropriate combination of the correlations that were developed in this appendix and can be represented by the following equation:

$$\lambda_{LOP}(t) = I_i(t) + GR_j(t) + SR_k(t) + SS_1$$

where

$I_i(t)$ = the plant-centered loss-of-offsite-power frequency correlation, defined in Table A.3 and Figure A.3

$GR_j(t)$ = the grid-related loss-of-offsite-power frequency correlation defined in Table A.6 and Figure A.6

$SR_k(t)$ = the severe-weather-related loss-of-offsite-power frequency correlation defined in Table A.8 and Figure A.9

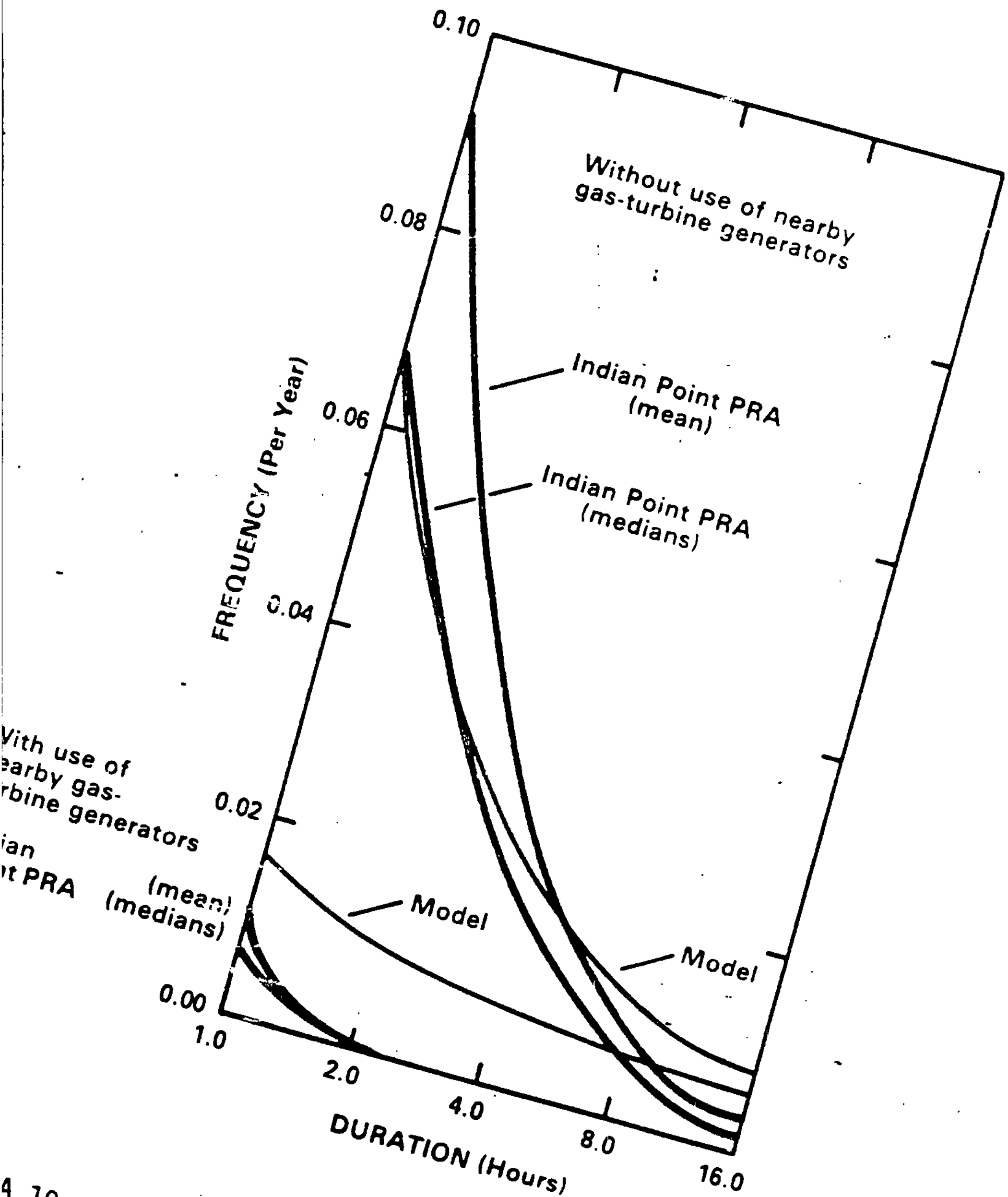
SS_1 = the extremely severe-weather-related loss-of-offsite-power frequency defined in Table A.9

The identification of the I_i factor is the most straightforward because it is based on configuration. As a first cut, the appropriate GR_j factor can be identified by dividing nuclear sites in the United States into two categories:

(1) FPL sites, approximated by GR3, GR4, or GR7, and (2) all other sites representing average frequency expection of grid failure, approximated by GR1 or GR4. The SR_k and SS_1 factors are not so easily identified because both design specifics and hazard rate must be determined. It is possible, however, to bracket these factors with a range that can be used to judge importance of station blackout considerations using hazard rates and proportionality factors for severe weather and using the upper range of the estimated failure rate for extreme weather hazards.

A test of the loss-of-offsite-power correlations that were developed was made by comparison with plant-specific results from published probabilistic risk assessments (PRAs). Figures A.10 through A.14 provide these comparisons. The degree of conformity between the results from the published PRAs and results based on the models developed in this appendix varies. Reasonable agreement was achieved for Indian Point (with credit for nearby gas turbine generators), Shoreham, and Limerick. The difference between the Indian Point PRA with credit for nearby gas turbine generators and this model is primarily due to the reliability associated with those power sources. In the Indian Point PRA, the combined reliability of the two gas turbine generators was on the order of 99%. In the model developed for this study, a fixed value for alternate offsite power sources of 80% was used. With regard to the Millstone PRA, the differences are primarily due to the use of data from other sites that do not appear to have the susceptibility to salt spray that the Millstone site has. In the model developed in this study the operating experience at sites other than Millstone, and to some extent Pilgrim, was not considered to be relevant and thus the two long losses of offsite power at the Millstone site contribute significantly to the estimated occurrence frequency of long-duration outages. The differences with the Zion PRA results could stem from one of several possibilities: design and procedural factors are more reliable than assumed in the comparison; the Zion PRA results are optimistic; or the models and correlations derived for generic analyses have limitations when applied to some plant-specific cases. Because of these considerations, a generic analysis must be used with caution in plant-specific applications. However, the generic models can usually provide good "ball park" results for generic applications and perspectives. Clearly the more details available and included in the models regarding design, procedures, alternate power sources, and protection provided from severe weather conditions, the more likely that the generic results will closely equate to plant-specific results.

The development of a more limited number of generic loss-of-offsite-power frequency and duration relationships that could be used for regulatory analysis involved the clustering of the site/design factors to determine if combinations of these factors could be grouped into a more limited, but still representative, set. A set of five cluster groups was derived from the set of site/design possibilities using the Fastclus procedure of the SAS package (SAS Institute, 1979). To limit the number of cluster groups, the clustering had to be based on loss-of-offsite-power durations of 2 to 8 hours. Figure A.15 provides a plot of the cluster groups derived from this analysis, and Table A.10 identifies combinations of each of the four factors (GR, I, SR, and SS) included in the nine cluster groups. For example, a plant with GR1, I1, SR1, and SS2 would be in cluster group 1. Grid reliability groups were limited to GR1, GR3, GR5, and GR7 to generate the clusters. Table A.11 provides a tabulation of cluster mean, median, and range values.



4.10 Estimated frequency of losses of offsite power exceeding specified durations for Indian Point

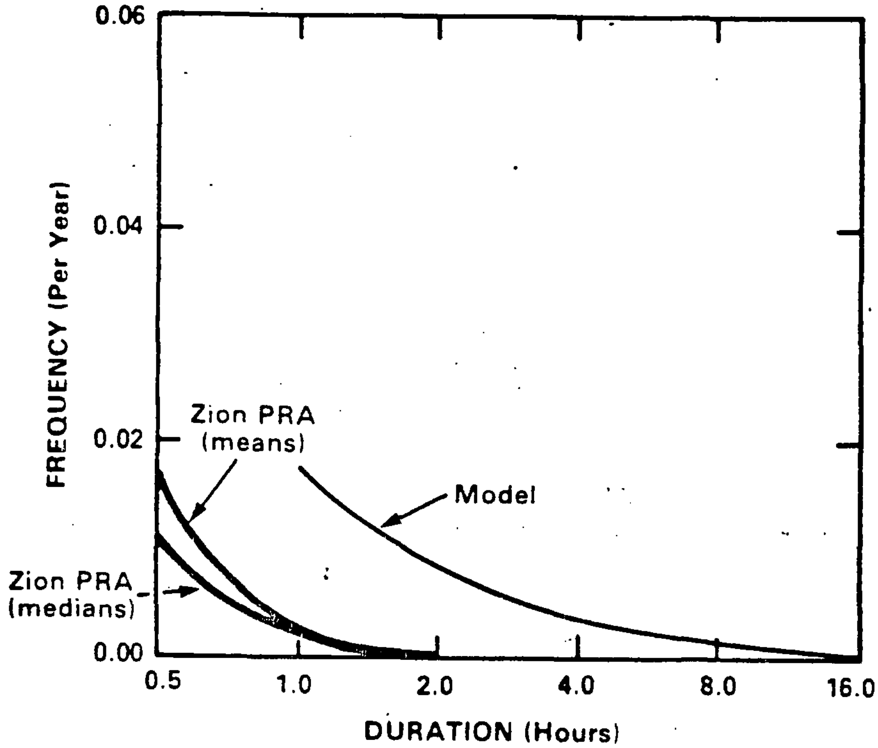


Figure A.11 Estimated frequency of losses of offsite power exceeding specified durations for Zion

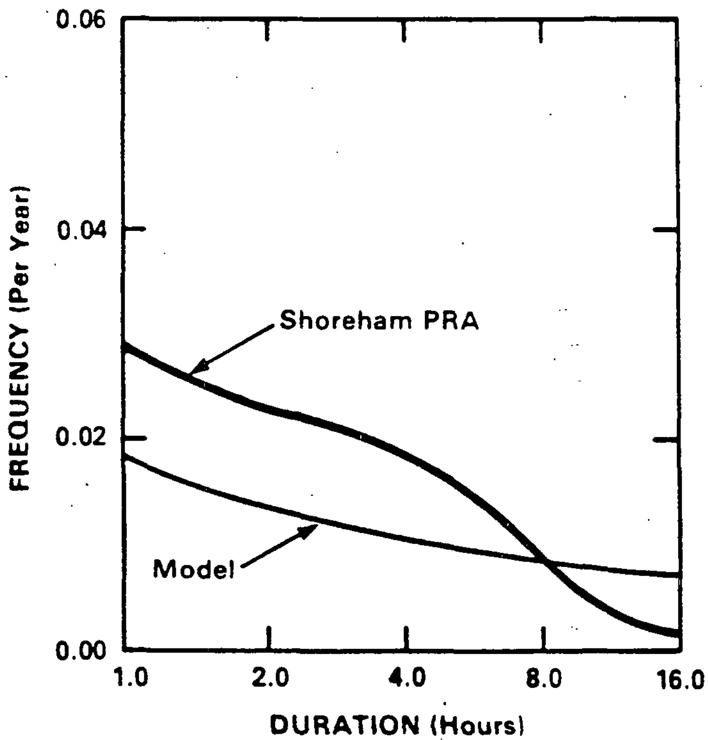


Figure A.12 Estimated frequency of losses of offsite power exceeding specified durations for Shoreham

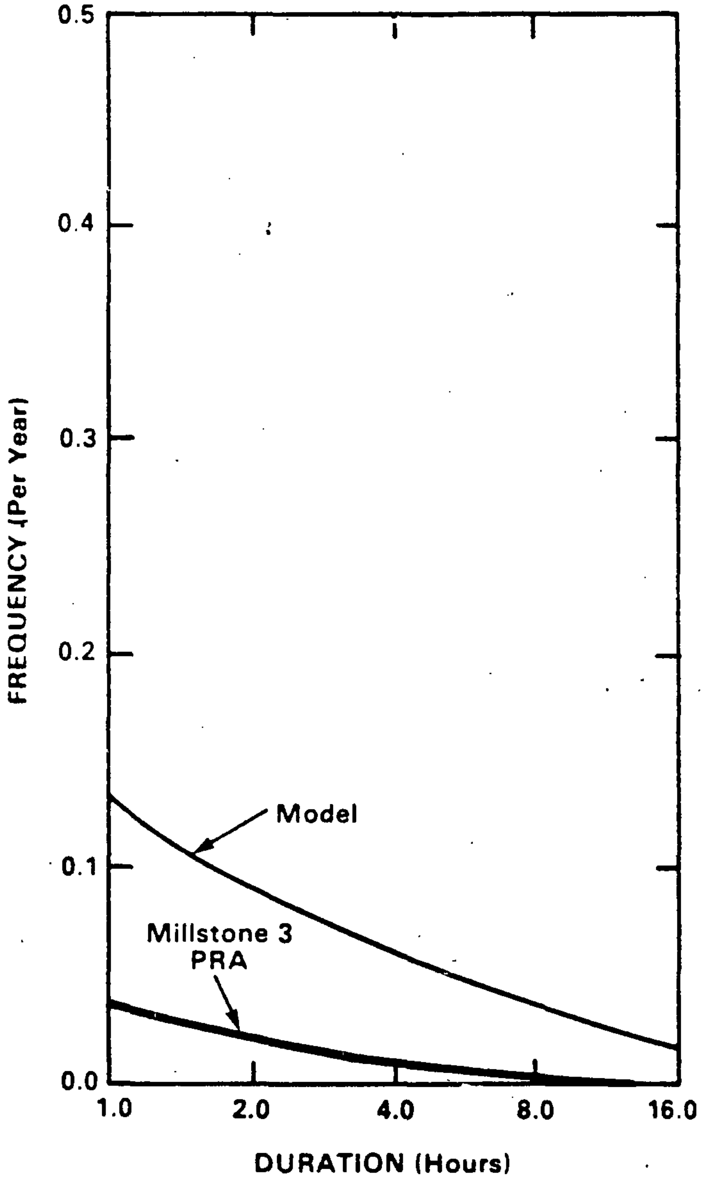


Figure A.13 Estimated frequency of losses of offsite power exceeding specified durations for Millstone 3

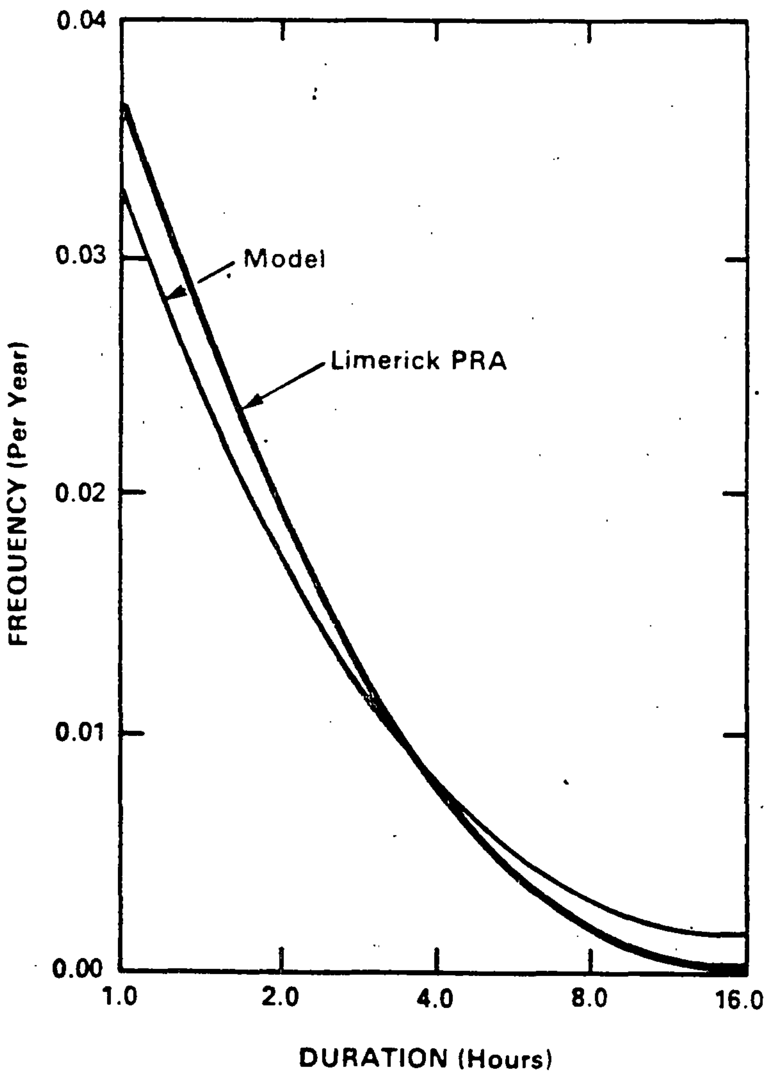


Figure A.14 Estimated frequency of losses of offsite power exceeding specified durations for Limerick

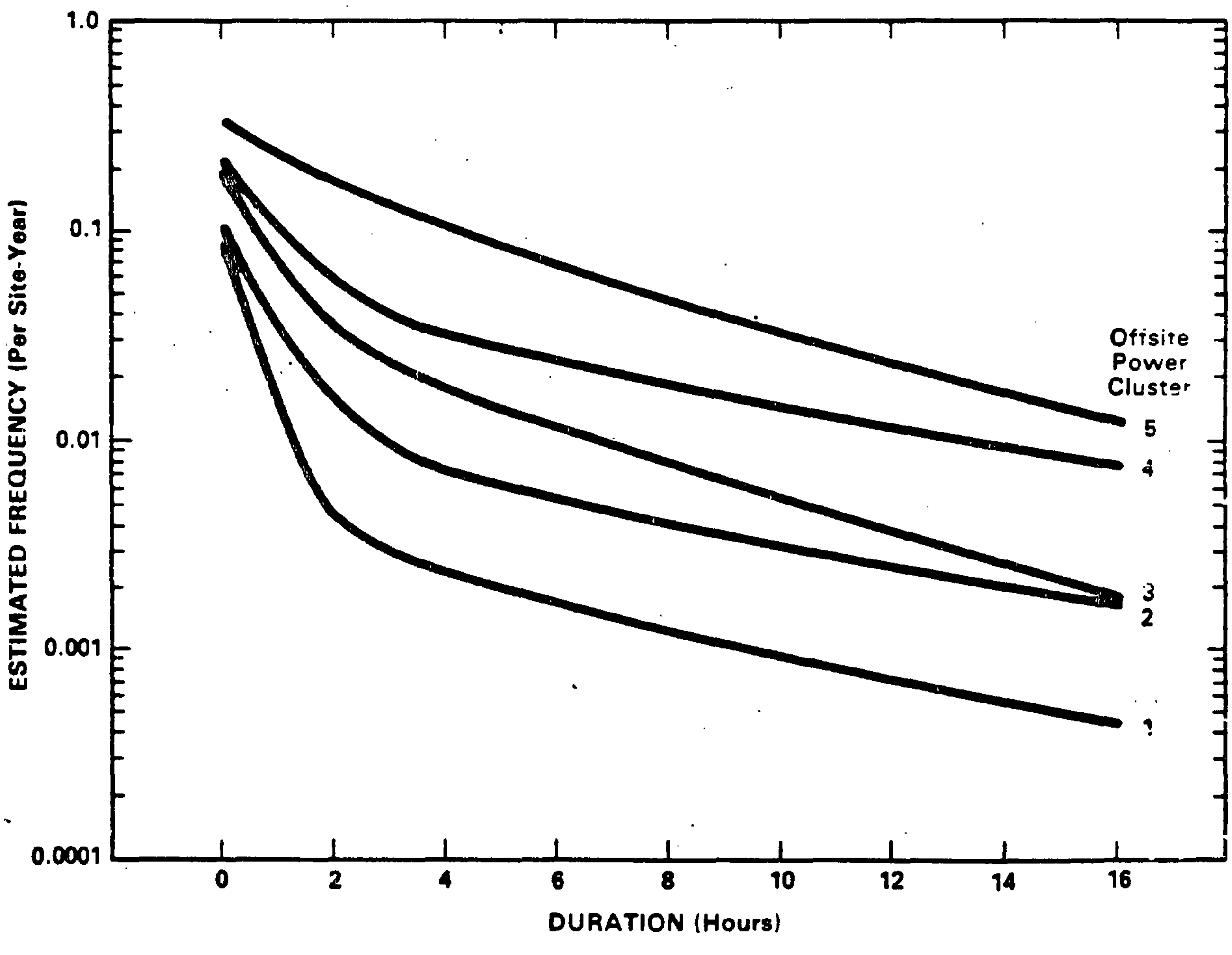


Figure A.15 Estimated frequency of occurrence of losses of offsite power exceeding specified durations for nine offsite power clusters

Table A.10 Identification of grid (GR), offsite power system design (I), severe weather (SR), and extremely severe weather (SS) factors included in five cluster groups

Cluster group	I	GR	SR	SS
1	1,2	1,3,5	1,2,6,7	1,2
	1,2	1,3,5	1,6	3
	1,2	1,3,5	3	1,2
2	1,2	1,3,5	8	1,2,3
	1,2	1,3,5	4	1-4
	1,2	1,3,5	2,3,7	3,4
	1,2	1,3,5	1,6	4
	3	1,3,5	1,2,6,7	1-4
	3	1,3,5	3,8	1,2
	3	1,3,5	3	3,4
	3	1,3,5	4	1-4
3	Same as cluster 2 and 1	7	Same as cluster 2 and 1	Same as cluster 2 and 1
4	1,2,3	1,3,5,7	1-9	5
	1,2,3	1,3,5,7	5,9	1-4
	1,2	1,3,5,7	8	4
	3	1,3,5,7	8	3,4
5	1,2,3	1,3,5,7	10	1-5

Table A.11 Loss-of-offsite-power frequency distribution per cluster group

Cluster group/value:	Duration (hrs) 0	2	4	8	16
Cluster 1:					
Upper Bound	0.1895	0.0102	0.0050	0.0031	0.0022
Mean	0.1157	0.0057	0.0027	0.0014	0.0007
Median	0.0845	0.0052	0.0025	0.0012	0.0005
Lower Bound	0.0812	0.0013	0.0005	0.0003	0.0002
Cluster 2:					
Upper Bound	0.2240	0.0271	0.0142	0.0077	0.0058
Mean	0.1297	0.0144	0.0075	0.0044	0.0027
Median	0.1040	0.0141	0.0070	0.0040	0.0022
Lower Bound	0.0812	0.0037	0.0026	0.0007	0.0002
Cluster 3:					
Upper Bound	0.2277	0.0447	0.0232	0.0104	0.0060
Mean	0.1892	0.0307	0.0159	0.0063	0.0024
Median	0.1798	0.0303	0.0153	0.0057	0.0017
Lower Bound	0.1749	0.0218	0.0113	0.0037	0.0006
Cluster 4:					
Upper Bound	0.3927	0.0909	0.0563	0.0340	0.0230
Mean	0.2113	0.0447	0.0273	0.0175	0.0126
Median	0.1978	0.0043	0.0253	0.0186	0.0080
Lower Bound	0.1010	0.0191	0.0140	0.0065	0.0023
Cluster 5:					
Upper Bound	0.3927	0.1838	0.1242	0.0647	0.0287
Mean	0.3306	0.1504	0.1006	0.0477	0.0140
Median	0.3343	0.1466	0.0970	0.0449	0.0123
Lower Bound	0.2792	0.1354	0.0909	0.0412	0.0086

Because design, grid, and weather all play a role in the frequency and duration relationship for each cluster, it is difficult to generalize about the dominant factors affecting loss of offsite power. It is possible to say that the higher frequency at longer duration groups (clusters) are most heavily influenced by weather hazard susceptibility. The highest frequency and duration correlation developed in this study (cluster 5) is driven by the high occurrence frequency (location) and susceptibility (design) to salt spray at coastal sites.

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APPENDIX B

EMERGENCY AC POWER RELIABILITY AND STATION.BLACKOUT FREQUENCY: MODELING AND ANALYSIS RESULTS

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APPENDIX B

EMERGENCY AC POWER RELIABILITY AND STATION BLACKOUT FREQUENCY: MODELING AND ANALYSIS RESULTS

This appendix provides the details and results of emergency AC power system reliability analyses and station blackout frequency/duration estimates. The models and analysis results were developed to confirm and extend the findings of a previous study (NUREG/CR-2989) and to be used in regulatory analyses. Modeling has been done at a generic level, but it could be made plant-specific by adjusting failure rate parameters to reflect site location, system design, and operational factors. The term generic, as used here, is meant to imply that the insights derived are generally applicable to a large number of plants. Modeling and component failure rate variations are used to account for plant differences in design and operational features that are most important to system reliability. Sensitivity analyses were used to explore the effect of design and operational differences on system reliability for a realistic spectrum of differences.

ELEMENTS OF EMERGENCY AC POWER RELIABILITY MODEL

The diesel generators--including all the subsystems and the auxiliary systems required to start, load, and run the diesels--are the components that have the highest impact on system reliability. Specifically the following have been identified as the largest contributors to AC power system availability:

(1) diesel generator configuration
(2) reliability of each diesel generator
(3) vulnerability to common cause failure
(4) support/auxiliary system dependence

In general, the details of the emergency AC power distribution system design from the Class 1E engineered safety feature buses to the safety system components using emergency AC power have not been found to be important contributors to system unreliability. With this in mind, emergency diesel generators (EDGs), DC power supplies, and service water cooling systems were the principal system elements included in the emergency AC power reliability models. A relatively high level (super component) modeling approach was used that could account for major differences in equipment configuration and support system dependencies while using support system reliability estimates developed in other studies.

Three generic emergency AC power system designs were selected as roughly representing the spectrum of operating nuclear plant systems. These systems are described by the number of diesel generators in the system and the number required to maintain core cooling during a loss of offsite power. These generic systems have been designated 2/3, 1/2, 2/4, and 1/3, indicating the number of diesel generators required per number available. Some other configurations do exist, but, emergency AC power system reliability is generally encompassed and well characterized by the three systems modeled, especially if the variability of

failure rates of the major components and auxiliary systems is accounted for. Configurations with a higher degree of redundancy and/or diversity are the exception, not the rule, in current U.S. designs. The simplified reliability logic models for the generic configurations were developed from fault trees and insights on what factors are important contributors to AC system reliability. The simplified logic models are provided below:

$$
\begin{aligned}
R_{EAC1/2} &= 1 - P_{EAC2/2} \\
&= 1 - [(P_{EDG})^2 + P_{CCF2/2}] \\
R_{EAC1/3} &= 1 - P_{EAC1/3} \\
&= 1 - [(P_{EDG})^3 + 3P_{EDG}\, P_{CCF2/3} + P_{CCF3/3}] \\
R_{EAC2/3} &= 1 - P_{EAC2/3} \\
&\cong 1 - [3(P_{EDG})^2 + 3P_{CCF2/3} + P_{CCF3/3}] \\
R_{EAC2/4} &= 1 - P_{EAC3/4} \\
&\cong 1 - [4(P_{EDG})^2 + 12P_{EDG}\, P_{CCF2/4} + 6(P_{CCF2/4})^2 + 4P_{CCF3/4} + P_{CCF4/4}]
\end{aligned}
$$

Where $R_{EACi/j}$ is the AC power reliability of an "i" out of "j" diesel generator system, and $P_{EACi/j}$ is the probability that "i" out of "j" diesels will fail or be unavailable when required, P_{EDG} is the probability that a single diesel generator will fail or be unavailable when required, and $P_{CCFi/j}$ is the probability that "i" out of "j" diesel generators will fail and be unavailable as a result of common causes when required.

A more complete logic model can be developed using Markov modeling techniques (Husseing, 1982) when failure and repair rates are exponentially distributed in time. However, the simplifications inherent to the models used are in keeping with the approach of accounting for dominant factors affecting system reliability.

Both random independent component failures and common cause or dependent failures are included in the model. Failure mode considerations included hardware faults and human errors for start and run failures, component repair, and component out-of-service time for maintenance. The least detailed level of modeling was at the support systems, which vary considerably in design. These systems have been modeled in detail in several probabilistic risk assessments (PRAs). The reliabilities of the support systems were treated as a super component or undeveloped event in the logic models with a failure rate indicative of results from other studies (NUREG/CR-3226).

Failure to run was treated as a constant failure rate process, and emergency diesel generator repair was treated as a constant repair rate process. With

these approximations, the probability that a diesel generator will be unavailable for τ_{SB} hours during a loss of offsite power lasting τ_{LOP} is given by

$$P_{EDG} = P_{FTS}\, e^{-\tau_{SB}/\tau_R} + \int_0^{\tau_{LOP}-\tau_{SB}} \lambda_{FTR}\, e^{-\lambda_{FTR} t}\, e^{-\tau_{SB}/\tau_R}\, dt$$

where τ_R is the mean repair time and λ_{FTR} is the failure-to-run rate. The failure-to-start probability, P_{FTS}, includes the standby demand failure likelihood of the emergency diesel generator to start and load, plus the unavailability because of scheduled and unscheduled maintenance, and the probability that auxiliary systems will fail or be unavailable (out of service) at the time of the demand. Although the second term of the equation can be integrated easily, the integral is maintained for applications relating to estimating station blackout frequency and duration to follow.

The probability of failure to start, load, and run for a time, τ_{SB}, because of common cause failures is developed similarly to that for independent failures. It is given by:

$$P_{EDGCCF} = P_{CCF}\, e^{-\tau_{SB}/\tau_{CCFR}} + \int_0^{\tau_{LOP}-\tau_{SB}} \lambda_{CCFTR}\, e^{-\lambda_{CCFTR} t}\, e^{-\tau_{SB}/\tau_{CCFR}}\, dt$$

Here, P_{CCFTS} represents the common cause failure-to-start probability, λ_{CCFTR} represents the common cause failure-to-run rate, and τ_{CCFR} is the associated repair time constant.

For simplicity, the repair rate for auxiliary systems that are required for successful diesel operation has been assumed to be approximately equal to that of the emergency diesel generator. Double component out-of-service conditions limited by technical specification were eliminated from the final expression through inspection. However, the possibility of such outages occurring as a result of human errors or simultaneous failures was treated as a common cause unavailability contributor.

Recall that the unreliability of a two diesel generator system was given by

$$P_{EAC2/2} = (P_{EDG})^2 + P_{CFF2/2}$$

where

$$(P_{EDG})^2 = F_1 + F_2 + F_3$$

and where

$$F_1 = (P_{FTS})^2 e^{-2\tau_{SB}/\tau_R}$$

$$F_2 = 2P_{FTS} e^{-\tau_{SB}/\tau_R} \int_0^{\tau_{LOP}-\tau_{SB}} \lambda_{FTR} e^{-\lambda_{FTR}t} e^{-(t+\tau_{SB})/\tau_R} dt$$

$$F_3 = 2e^{-\tau_{SB}/\tau_R} \int_0^{\tau_{LOP}-\tau_{SB}} \int_{t_1}^{\tau_{LOP}-\tau_{SB}} (\lambda_{FTR})^2 \cdot e^{-\lambda_{FTR}t_2} e^{-(t_2+\tau_{SB}-t_1)/\tau_R} dt_2 \, e^{-\lambda_{FTR}t_1} dt_1$$

with

$$P_{CCF2/2} = P_{CCFTS2/2} e^{-\tau_{SB}/\tau_{CCFR}}$$

$$+ \int_0^{\tau_{LOP}-\tau_{SB}} \lambda_{CCFTR2/2} e^{-\lambda_{CCFTR2/2}t} e^{-\tau_{SB}/\tau_R} dt$$

and

$$P_{FTS} = Q_{EDG1} + U_{EDG1} + P_{DC1} + P_{SW1}$$

$$P_{CCFTS} = Q_{CCF2/2} + U_{CCF2/2} + P_{DCCCF} + P_{SWCCF}$$

where Q_{EDG1} is the probability of a diesel generator failing on demand, U_{EDG1} is the maintenance unavailability of the diesel generator, P_{DC1} is the probability of DC power supply failure causing a diesel to fail on demand, and P_{SW1} is the probability of a service water system failure causing a diesel generator failure on demand. Terms with subscript CCF represent common cause failure contributions.

The term $(U_{EDG1})^2$ is not allowed. It is accounted for in the term $U_{CCF2/2}$. In a similar manner, the correlations for three or four diesel generator systems requiring one or two diesels for success can be derived.

COMMON CAUSE FAILURE OF THE EMERGENCY AC POWER SYSTEM

There has been a concern for years that the reliability of redundant systems may be limited by single point and common causes of failure resulting in simultaneous unavailability of two or more trains. Several techniques for modeling

and quantifying the major contributors and their likelihood have been, and continue to be, developed. Some of these techniques are aimed at a qualitative evaluation of common cause failure potential (Rasmuson, 1982), while others are primarily used to estimate common cause failure likelihood (Fleming and Raabe, 1978). Existing techniques have been used in this study to model and quantify common cause failures on a generic level, with sensitivity analyses used to evaluate realistic variations in common cause failure likelihood and the effect on emergency AC power reliability.

Emergency diesel generator operating experience for the years 1976 through 1980 was reviewed and documented in NUREG/CR-2989. Other reviews [Electric Power Research Institute (EPRI), 1982, and Steverson and Atwood, 1981] also show relevant operating experience and analysis of common cause failures of emergency diesel generators. Based on information from these sources and limited review through 1985 of licensee event reports (LERs) dealing with common cause failures, an updated list and classification of multiple emergency diesel generator failures and outages has been prepared. When enough information exists, the common cause failures can usually be identified as falling into one of four groups: (1) design/hardware, (2) operations/maintenance, (3) support systems/dependence, and (4) external environment. A further breakdown of this classification scheme is provided in Table B.1. The list of common cause failures taken from LERs is in Table B.2. In NUREG/CR-2989 these were classified somewhat more generally in two broad categories of hardware and human-error-related failures. These two categories were then classified more specifically into generic and plant-specific design groups and into generic human error or plant-procedure-specific human error.

Table B.1 Areas of potential common cause failure

Common cause failure group	Types of potential failures
Design/hardware	Mechanical/structural design inadequacy Subsystems (fuel, cooling, start, actuation) Environment (normal)
Operations/maintenance	Inadequate procedures Errors of omission/commission Wrong procedure
Support/dependence systems	DC control power Service water cooling EDG room heating, ventilation, and air conditioning Electrical interface
External	Fire Flood Severe weather Seismic Other internal environmental extremes

Table B.2 Emergency diesel generator (EDG) common cause failures

Plant	Date of event	LER number	Description of event
ANO	08/27/79 09/11/79	79-016 79-017	Water in lube oil caused failure of two EDGs 2 weeks apart.
Arnold	05/10/77 05/12/77	77-037 77-043	Maintenance caused control system failures on both EDGs within 2 days.
Browns Ferry 1, 2	05/06/81 05/06/81	81-019 81-020	Left bank air start motors failed to start three EDGs.
Browns Ferry 3	01/03/84	84-001	Clam shell movement on overchlorination failed emergency service water (ESW) coolers and three of four EDGs.
Brunswick 1, 2	01/04/77	77-001	Low lube oil pressure tripped two of four EDGs after starting.
Crystal River 3	01/04/79	---	Low ambient room temperature (28°F) failed both EDGs.
Dresden 3	10/23/81	81-033	ESW check valve failures caused two of the three EDGs to trip on high temperature.
Farley 1	09/13/77 09/16/77	77-026 77-027	Dirty air start circuit failed two EDGs within 3 days.
Farley 1, 2	09/18/81 09/27/81	81-043 81-067	Scored cylinder linings failed two EDGs 9 days apart.
FitzPatrick	02/07/85	85-003	ESW pump trip failed two EDGs.
Millstone 2	05/15/77	77-020	Both EDG fuel supply valves found closed.
North Anna 2	02/18/81	81-020	Batteries failed surveillance test, caused both EDGs to be inoperable.
North Anna 2	12/09/84	84-013	Damaged cylinders and high crankcase pressure failed both EDGs, caused unit shutdown.
Peach Bottom	06/13/77	77-026	Air-start compressor trip caused two EDGs to fail while another was unavailable.
Quad Cities	05/01/77	*	Improper ESW valve lineup degraded three EDGs.

Table B.2 (Continued)

Plant	Date of event	LER number	Description of event
Salem 1	07/30/77	77-059	Fuel rack lubrication leak and subsequent linkage binding caused failure of two EDGs.
Salem 1	10/08/80	80-060	All three EDGs failed to start because of a misaligned service water valve. Operator disabled service water from train 2 while train 1 was down for maintenance.
Sequoyah 1, 2	08/09/80	80-140	Operator error caused relay coils to fail on all EDGs.
Susquehanna	01/21/85	85-002	Low ambient room temperature failed two EDGs.
Vermont Yankee	10/22/84	84-022	Failed Zener diodes caused all EDGs to lock out.
WNP-2	07/09/84	84-008	Slip ring and bearing design weakness caused failure of two EDGs.
Yankee Rowe	08/02/77	77-042	Sludge-plugged cooling water radiator tubes caused failure of two EDGs

*Reported in PLG-400, Pickard, Lowe and Garrick Inc.

Common cause failure rates were estimated in NUREG/CR-2989 using the binomial failure rate (BFR) computer code (Atwood and Smith, 1982). The estimated common cause failure rates varied by about an order of magnitude depending on plant design and procedural dependencies. If individual emergency diesel generator reliability is maintained at or above industry average levels, common cause failure contributed on the order of one-half the system unavailability for the less redundant configurations and most of the unavailability for the more redundant designs, especially when demand failure rates are low (<0.03). At lower reliability levels, independent diesel generator failures are the major contributor to the unavailability of the onsite AC power system.

A technique that has been used to estimate the likelihood of emergency diesel generator common cause failure is the beta factor method (Fleming, 1975) and its extension known as the multiple Greek letter (MGL) method (Fleming and Kalinowski, 1983). This method was used to estimate common cause failure rates from the updated LER review. Table B.3 provides the MGL parameter estimates and common cause failure rate estimates that were derived by the MGL method. It also compares these estimates with "generic" rates derived in NUREG/CR-2989 using the BFR method. Differences result more from data classification than from analytical method.

Table B.3 Common cause failure rate parameter estimates

Results of MGL method*	BFR method
2 EDG configuration: $\beta = 0.035$	
P_{CCFTS} (2/2) = 5.7×10^{-4}	7.1×10^{-4}
P_{CCFTR} (2/2) = 1.0×10^{-4}/hr	
3 EDG configuration: $\beta = 0.087$, $\gamma = 0.351$	
P_{CCFTS} (2/3) = 4.62×10^{-4}	5.6×10^{-4}
P_{CCFTS} (3/3) = 5.00×10^{-4}	1.8×10^{-4}
P_{CCFTR} (2/3) = 8.19×10^{-5}/hr	
P_{CCFTR} (3/3) = 8.85×10^{-5}/hr	
4 EDG configuration: $\beta = 0.147$, $\gamma = 0.528$, $\delta = 0.505$	
P_{CCFTS} (2/4) = 3.79×10^{-4}	
P_{CCFTS} (3/4) = 2.10×10^{-4}	
P_{CCFTS} (4/4) = 6.43×10^{-4}	
P_{CCFTR} (2/4) = 6.71×10^{-5}	
P_{CCFTR} (3/4) = 3.71×10^{-5}	
P_{CCFTR} (4/4) = 1.14×10^{-4}	

*The following equations were used to perform the above calculations:

$$P_{CCF}\ (2/2) = \beta Q$$

$$P_{CCF}\ (2/3) = \frac{(1-\gamma)\ \beta Q}{2}$$

$$P_{CCF}\ (3/3) = \gamma\beta Q$$

$$P_{CCF}\ (2/4) = \frac{(1-\gamma)\ \beta Q}{3}$$

$$P_{CCF}\ (3/4) = \frac{(1-\delta)\gamma\ \beta Q}{3}$$

$$P_{CCF}\ (4/4) = \delta\gamma\beta Q$$

EMERGENCY AC POWER RELIABILITY EVALUATION

The reliability estimates for the generic emergency AC power systems were derived for instantaneous availability on demand and mission reliability. (The latter is the likelihood that emergency AC power will be available for a specified mission length, such as the duration of a loss-of-offsite-power event or for the duration of a test.) System reliability analysis parameters were selected to represent the average of the operating reactor population as well as the variations within that population. The population average and ranges for the system reliability analysis parameters are described below.

(1) Emergency Diesel Generator Failure To Start

Based on data reported in NUREG/CR-2989 and NSAC/108 (Wyckoff, 1986), the failure rate can vary considerably from plant to plant. The following probability of failure/demand rates have been identified:

Average	0.02
High	0.08
Low	0.005

(2) Emergency Diesel Generator Failure To Run

A constant failure rate of 0.0024 per hour was estimated in NUREG/CR-2989, while more recent data obtained from NSAC/108 and a review of LERs from 1983 through 1985 resulted in a revised estimate of 0.0032 per hour. For the period 1976 through 1985 the average was 0.0028 per hour. A range of 0.001 to 0.01 is reasonably representative of other published estimates (EPRI, 1982).

(3) Emergency Diesel Generator Repair Time

Approximately 50% of all diesel generator failures reported in NUREG/CR-2989 were repaired within 8 hours. If two diesel generators failed as a result of independent causes and operators could diagnose the problems to select the quickest possible repair, in 50% of these cases, one of two diesel generators would be repaired in approximately 4 hours. These two cases have been used as representative of the repair rate.

(4) Common Cause Failure

Common cause failure rates were obtained from NUREG/CR-2989 for diesel generator hardware and human-error-related causes; however, only failure-to-start estimates were made in that study. Subsequently, the MGL method has been used to estimate generic common cause failure rates for both failure to start and failure to run. Human errors causing a simultaneous out-of-service state for two or more diesel generators were included in estimates of failure to start. The MGL estimates are consistent with the generic estimates made in NUREG/CR-2989.

The common cause failure rates, for support systems--such as DC power, service water, and component cooling water--were obtained from NUREG/CR-3226.

(5) <u>Common Cause Failure Repair Rates for Components and Subsystems</u>

When the inadvertent removal from service of more than two diesel generators is excluded, the failure mode and repair rates appear similar to those for independent failure causes. In this case, however, the same repair time could be expected for both units. For inadvertent removal from service, repair (or restoration) can be accomplished usually in less than 1 hour and many times even more promptly (within minutes). Repair rates for hardware failure and maintenance outages have been based on median repair times of 2 to 8 hours.

The effect of system reliability parameter variations covering the realistic range was analyzed to determine the sensitivity within the generic models and the variability that is possible in plant-specific cases.

The first sensitivity analysis shown in Figure B.1 includes the effect of a mission time of 8 hours for various emergency diesel generator starting reliability values and for variations in common cause failure rates by a factor of 3. These results show that starting reliability of individual emergency diesel generators is most important when lower-than-average diesel generator performance exists or when system configurations represent nominal redundancy (e.g., 2/3 and 1/2). Common cause failures dominate system failure probability when individual diesel generator reliability levels are above average or when a higher level of redundancy (2/4 and 1/3) is introduced.

Figure B.2 shows the sensitivity of emergency AC power system unavailability as a function of individual diesel unavailability. This unavailability is due to out-of-service time for normal maintenance and for repairs necessary to fix incipient, degraded, or catastrophic failures of diesel generators which are detected by surveillance or other activities during normal plant operations. Only when the diesel generator out-of-service unavailability approaches or exceeds the starting failure rate does a significant effect on system unavailability become apparent.

Figure B.3 shows the AC power system unavailability variation as a function of diesel generator repair time for a mission time of 8 hours. This repair time represents the time it would take to repair 50% of all diesel generator failures during an actual demand situation assuming an exponential rate of repair. Also it has been assumed that sufficient resources and expertise are available to ensure selection of the diesel generator which can be repaired most quickly. The most significant affect on system unavailability is due to variations in common cause failure repair times especially where common cause failures are the dominant contributor to system unavailability (e.g., 1/3 system configuration).

The last sensitivity analysis performed is shown in Figure B.4. In this case the potential range of unavailability for emergency AC power systems was estimated by using combinations of above and below average reliability performance parameters discussed previously in this appendix. Not surprisingly, the range is large, especially for the more redundant configurations.

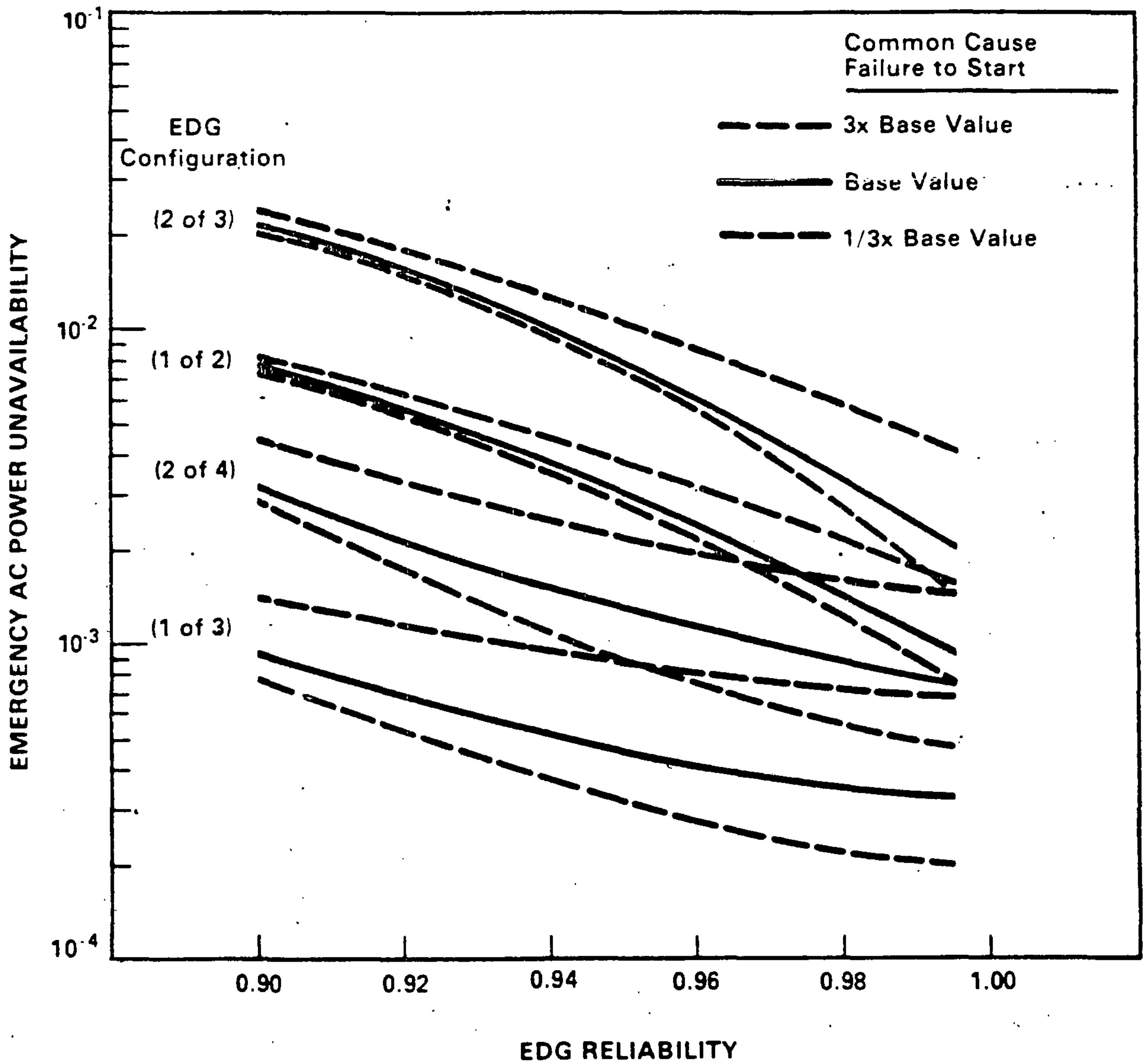

Figure B.1 Emergency AC power unavailability as a function of individual EDG reliability and common cause failure to start for three emergency AC configurations

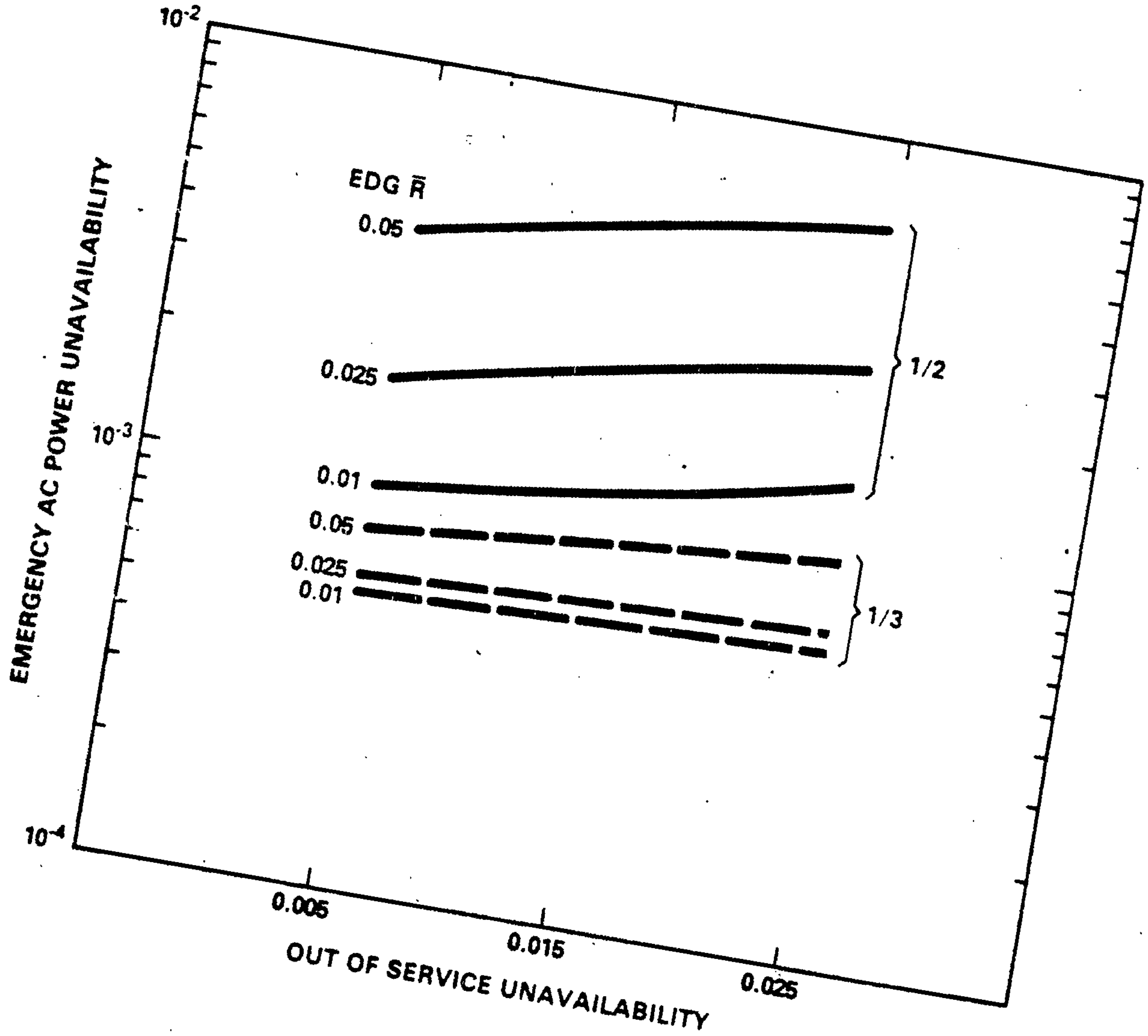

Figure B.2 Emergency AC power unavailability as a function of out-of-service unavailability for three EDG unreliabilities

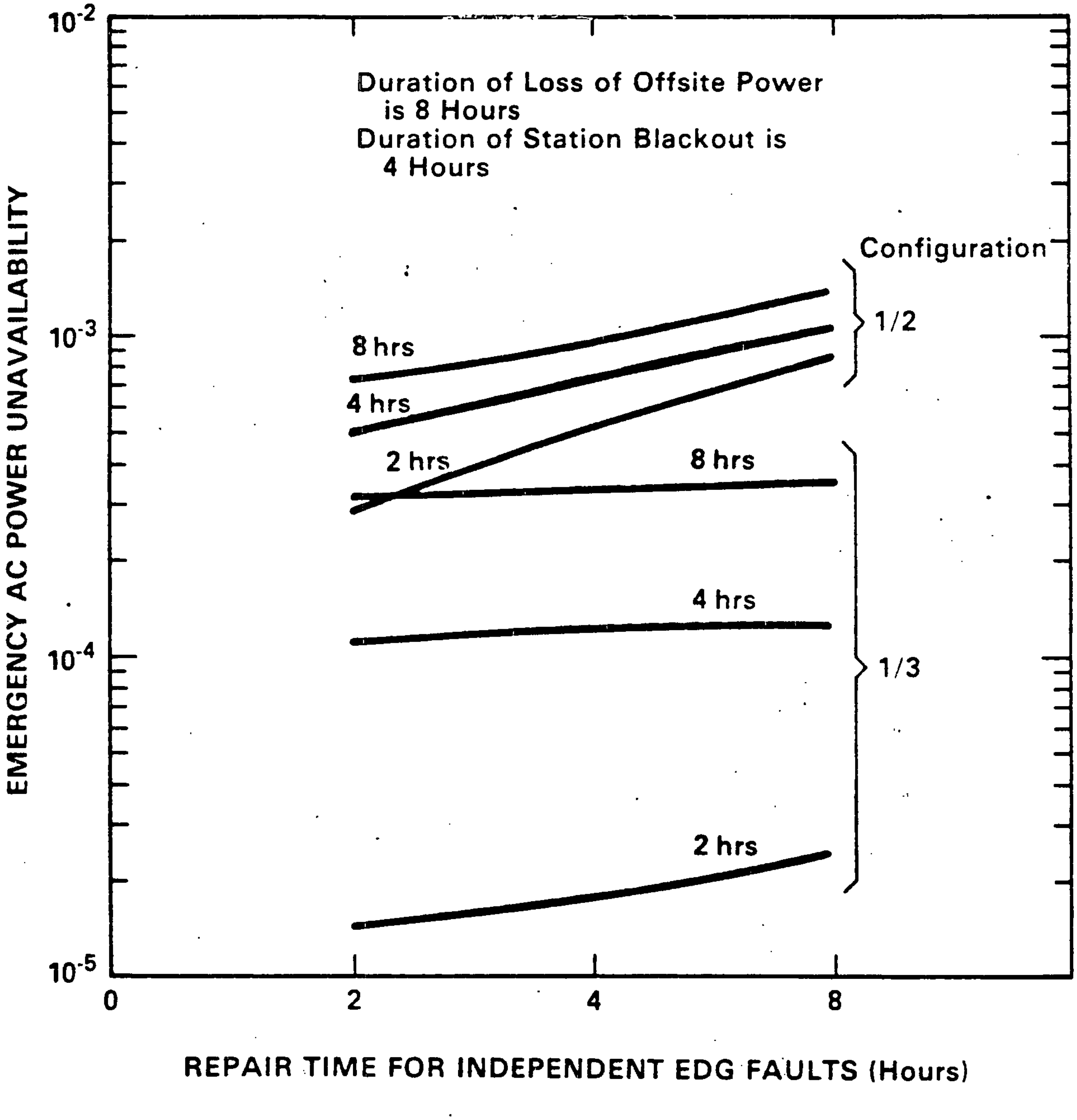

Figure B.3 Emergency AC power unavailability as a function of repair time for both independent EDG faults and common cause failure to start

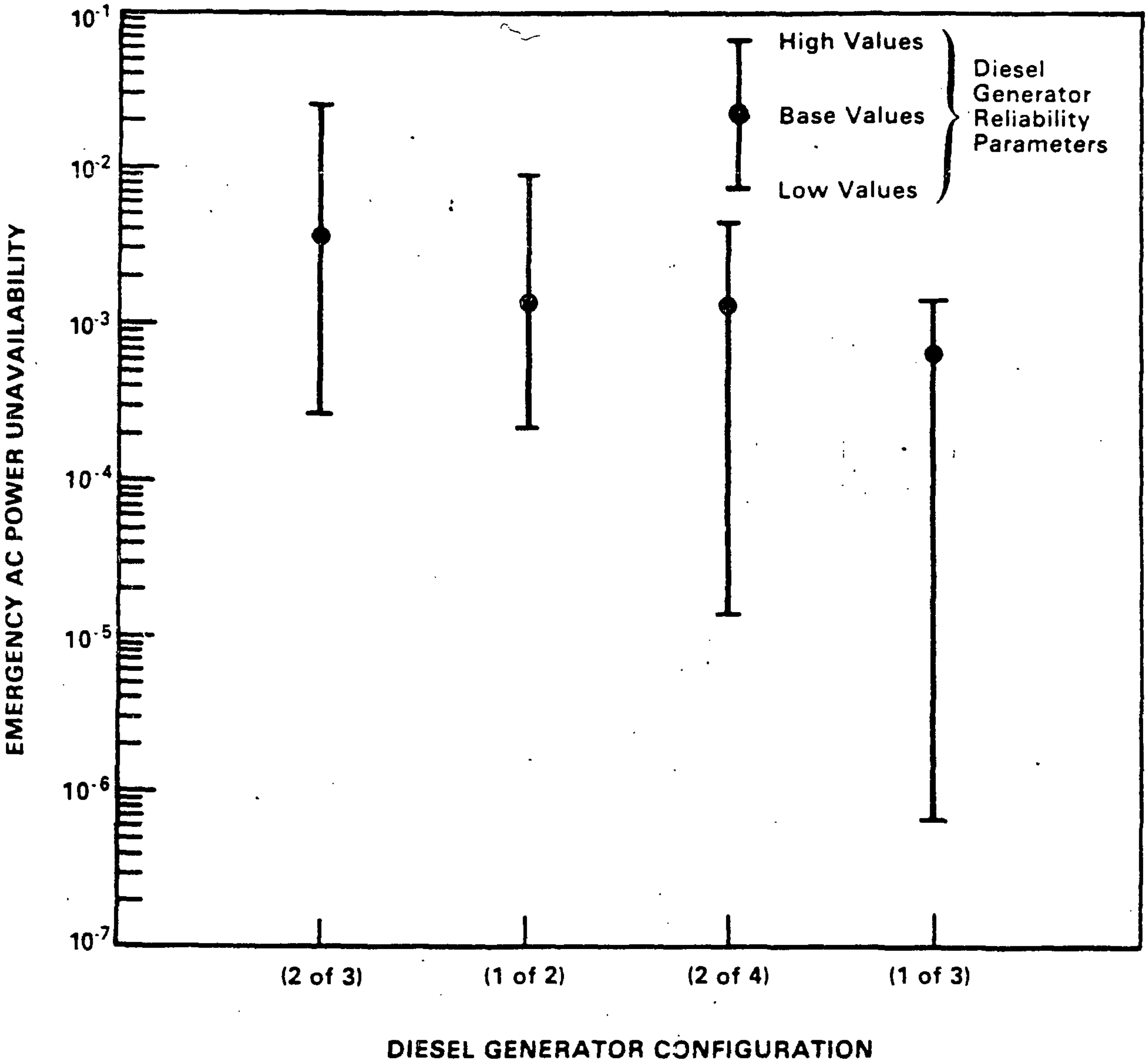

Figure B.4 Estimated range of emergency AC power system reliability for different diesel generator configurations

STATION BLACKOUT FREQUENCY

Station blackout has been defined as the loss of all AC power supplies from both offsite and safety-related sources. Also, a station blackout must exist for sufficient time to incur core damage and result in containment failure if the sequence is to be of risk significance. Therefore, station blackout models incorporate duration as a parameter in frequency estimates. Although in some instances it is possible to have a station blackout initiated by failure of, or operational efforts associated with, DC control power, this type of event is more rare than the station blackout sequence beginning with loss of offsite power and followed by failure of the safety-related AC power supplies. DC power reliability is the subject of another generic safety issue, designated A-30, "Adequacy of Safety-Related DC Power Supplies."

Station blackout frequency estimates can be made by combining the loss-of-offsite-power models developed in Appendix A with the emergency AC power reliability models of this appendix.

The loss-of-offsite-power frequency and duration correlations were derived in Appendix A. In the derivations that follow, let $\lambda_{LOP}(t)$ represent a loss-of-offsite-power frequency correlation. The frequency of a station blackout is derived by combining the loss-of-offsite-power duration (repair frequency with the rate of emergency AC power system failures of duration τ_{SB} over the time period of interest for which a loss of offsite and emergency AC power can occur. This is the same general approach that has been taken in other studies [Evans and Parry, 1983; Power Authority of the State of New York (PASNY), 1982] to estimate the frequency of total losses of offsite and emergency AC power for risk analysis. For the 1/2 emergency diesel generator configuration, the equation for the frequency of a station blackout lasting τ_{SB} or longer can be written as

$$\Lambda_{SB1/2}(\tau_{SB}) = \lambda_{LOP}(\tau_{SB})(P_{FTS})^2\, e^{-2\tau_{SB}/\tau_R}$$

$$+\ \lambda_{LOP}(\tau_{SB})\, P_{CCFTS2/2}\, e^{-\tau_{SB}/\tau_{CCFR}}$$

$$+\ 2P_{FTS}\, e^{-\tau_{SB}/\tau_R} \int_0^{\tau_{LOP}-\tau_{SB}} \lambda_{LOP}(t+\tau_{SB})\, \lambda_{FTR}\, e^{-\lambda_{FTR}t}\, e^{-(t+\tau_{SB})/\tau_R}\, dt$$

$$+\ 2e^{-\tau_{SB}/\tau_R} \int_0^{\tau_{LOP}-\tau_{SB}} \int_{t_1}^{\tau_{LOP}-\tau_{SB}} \lambda_{FTR}\, e^{-\lambda_{FTR}t_2}\, e^{-(t_2+\tau_{SB}-t_1)/\tau_R}\, dt_2$$

$$\cdot\ \lambda_{LOP}(t_1)\, \lambda_{FTR}\, e^{-\lambda_{FTR}t_1}\, dt_1$$

$$+ \int_0^{\tau_{LOP}-\tau_{SB}} \lambda_{LOP}(t+\tau_{SB})\, \lambda_{CCFTR2/2}\, e^{-\lambda_{CCFTR2/2}t}\, e^{-\tau_{SB}/\tau_{CCFR}}\, dt$$

In a similar manner, the station blackout frequency equations for three and four diesel generator systems requiring one or two diesels for success can be derived.

Analyses have been performed to estimate the sensitivity of station blackout frequencies and durations to various site characteristics. The loss-of-offsite-power cluster correlations developed in Appendix A were combined with the emergency AC power system reliability models using nominal values for emergency diesel generator failure to start and run, repair, and common cause failure rates. Results are in the main report in Figures 5.1, 5.2, and 5.3.

Additional analyses were performed to determine the sensitivity of station blackout results to potential variations in plant-centered loss-of-offsite-power frequency. Cluster correlations 2 and 4 (see Appendix A of this report) were selected. The plant-centered loss-of-offsite-power frequency was varied from a high value of 0.15 to a low value of 0.04. This represents a reasonable variation in the plant-centered frequency based on actual operating experience. Figure B.5 provides the results of these analyses. This figure shows that modest variations (factor of 2) in the plant-centered loss-of-offsite-power frequency will have essentially no noticeable effect on results at sites dominated by weather-induced losses of offsite power (cluster 4). Only a small effect would be noticeable at sites which have a more typical blend of failure causes (cluster 2), and that effect is only noticeable for short duration blackouts. Thus potential variations in plant-centered loss-of-offsite-power frequency will generally result in small changes in station blackout results when typical or more substantial contributions from grid and particularly weather exist.

Another sensitivity analysis was performed to estimate the impact of variations in grid reliability and restoration capability. For cluster 4, grid loss frequencies of 0.01 and 0.1 per year were analyzed with enhanced recovery (see Appendix A). For cluster 2, the same frequencies were analyzed but this time with normal recovery. The results are shown in Figure B.6. Potential variations in grid-related loss-of-offsite-power frequency have a small effect on the station blackout frequency and duration in most cases where typical or more substantial contributions from plant-centered and particularly weather exist.

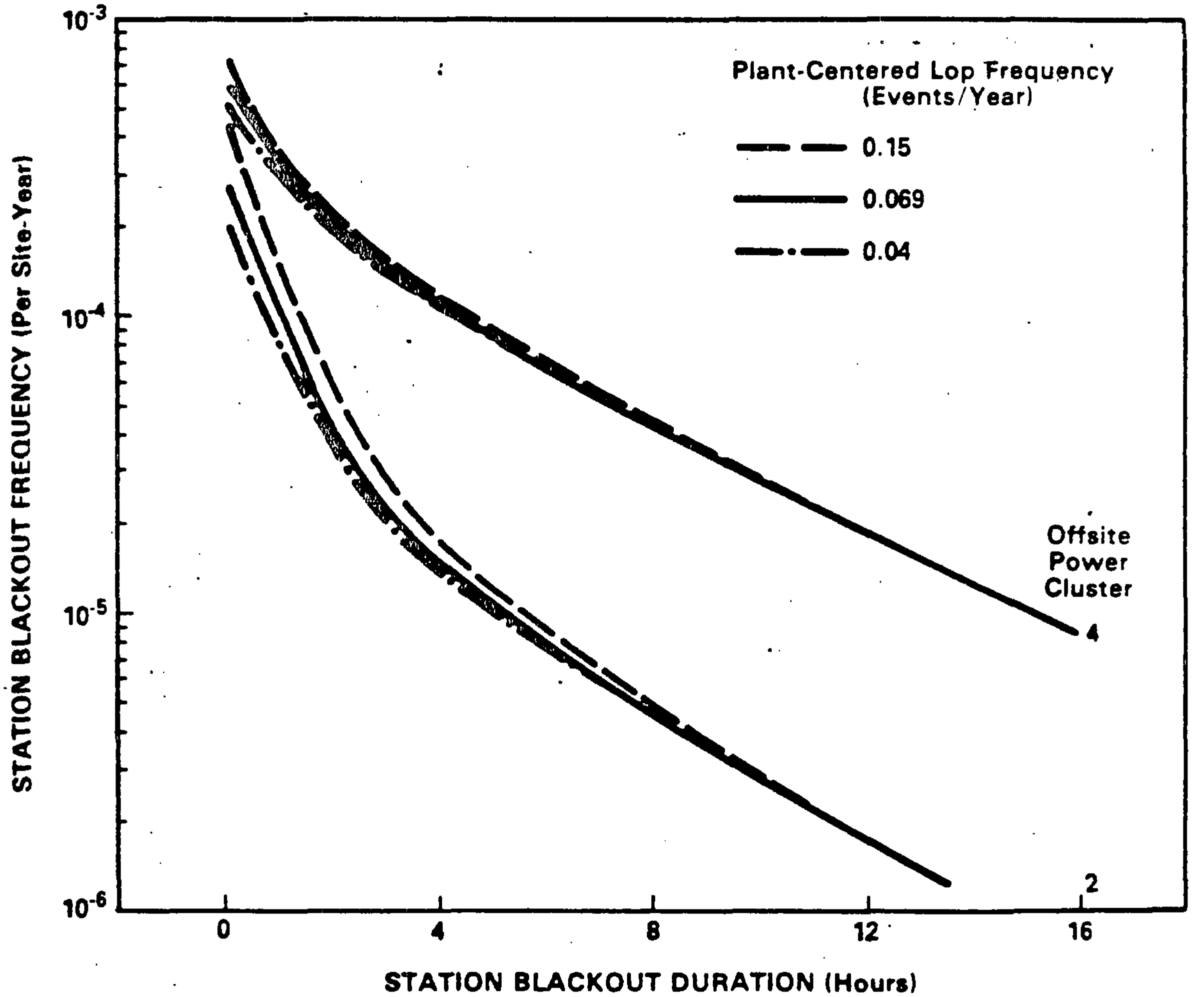

Figure B.5 Sensitivity of station blackout results to potential variation in plant-centered loss-of-offsite-power frequency

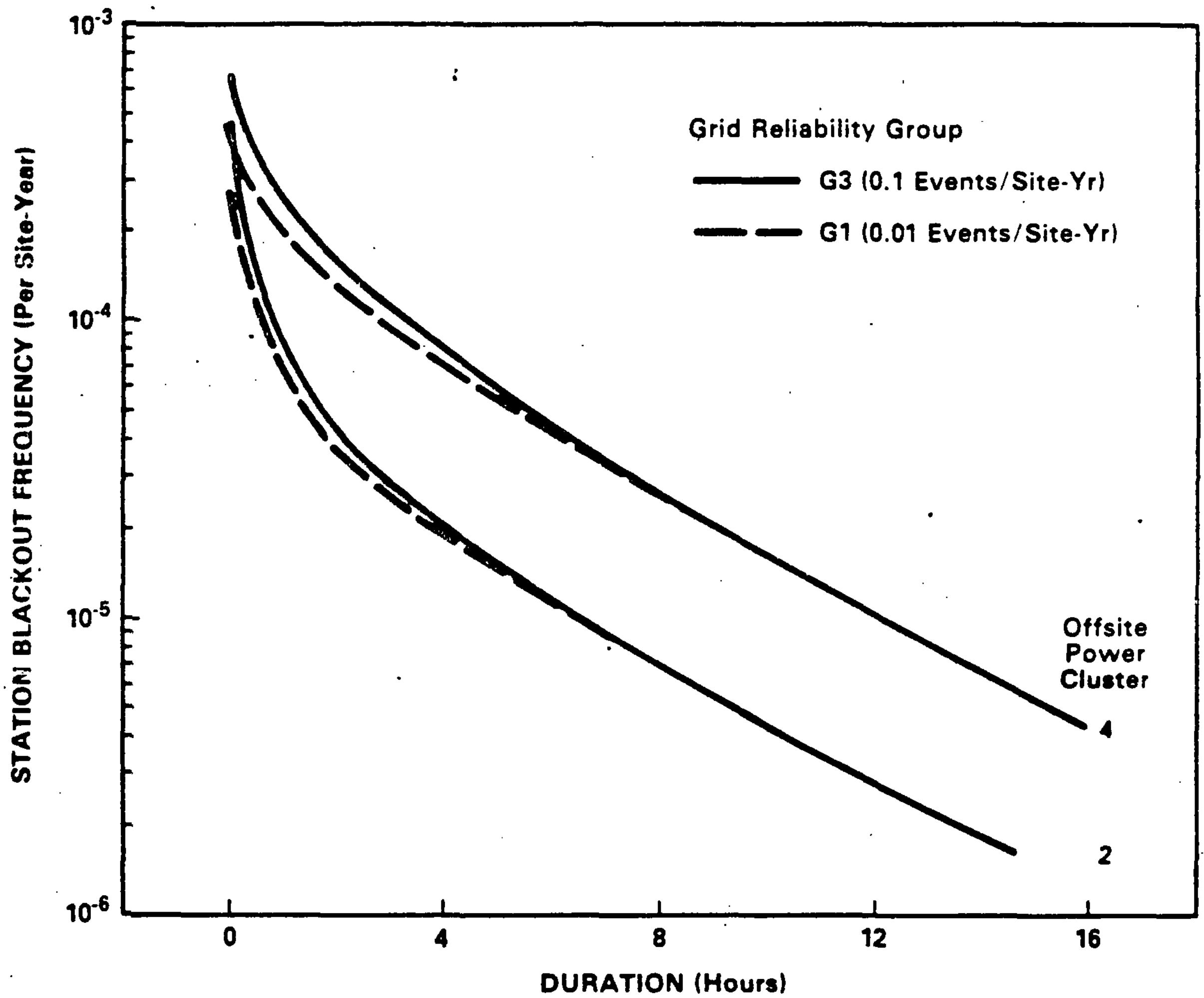

Figure B.6 Sensitivity of station blackout results to potential variation in grid-related loss-of-offsite-power frequency

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APPENDIX C

STATION BLACKOUT CORE DAMAGE LIKELIHOOD AND RISK

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APPENDIX C

STATION BLACKOUT CORE DAMAGE LIKELIHOOD AND RISK

This appendix provides a description of the simplified method used to estimate station blackout core damage likelihood, and risks from station blackout transients. The models and results are generic in nature and intended for use in regulatory analyses. The station blackout frequency estimation models described in Appendix B of this report were integrated into sequences involving failure of decay heat removal systems with AC power unavailable, thus allowing the estimation of the frequency of core damage as a result of station blackout events. When core damage proceeds to core melt and containment failure, fission products may be released to the environs, causing risk to public health and safety.

The likelihood of station blackout transients involving core damage and the dominant accident sequences have been identified by Kolaczkowski and Payne in NUREG/CR-3226, using event tree and fault tree analyses of several typical plant designs. However, the variability of station blackout frequency and duration was not evaluated systematically as part of that work. In this appendix, the station blackout models have been combined with the decay heat removal and core cooling failure sequences to obtain a more complete evaluation of the sensitivity of station blackout core damage likelihood and risk estimates to variations in plant design.

STATION BLACKOUT CORE DAMAGE LIKELIHOOD

The dominant station blackout sequences are provided in Table C.1. Both pressurized water reactors (PWRs) and boiling water reactors (BWRs) have sequences that involve early core cooling failure (essentially on demand) and time-dependent failures related to capacity, capability, and transient phenomenological conditions associated with a loss of all AC power. For the dominant accident sequences, the core damage times have been characterized as falling into two groups: (1) a core damage time of 1 to 2 hours for the early core cooling failure types of sequences or (2) core damage in the 2-to-16-hour range for the sequences involving capability and capacity limitations causing loss of core cooling during extended blackouts. Sequences involving longer duration blackouts than these have not been found to be nearly as important.

Thermal hydraulic analyses have been performed to determine event timing for both types of sequences (Fletcher, 1981; Schultz and Wagoner, 1982). In general, it has been estimated that it will take between 1 and 2 hours to uncover the reactor core following a station blackout and loss of all core cooling, and perhaps another 1 to 2 hours for the reactor core to melt and penetrate the reactor vessel after the core is uncovered. If decay heat removal is initially successful during station blackout and then is lost several hours into the transient because of design limitations, the time to core uncovery and melt will be somewhat extended as a result of lower primary coolant temperatures and reduced decay heat levels.

Table C.1 Summary of potentially dominant core damage accident sequences

Generic plant type	Sequence	DHR system/component contributors	AC recovery time to avoid core damage (hr)
PWR (all)	TML_1B_1	Steam-driven AFWS unavailable	1 to 2
	TML_2B_2	DC power or condensate exhausted	4 to 16
	TMQ_2B_2	RCS pump seal leak	4 to 16
BWR w/isolation condenser	TMU_1B_1	Isolation condenser unavailable	1 to 2
	TMQ_1B_1	Stuck open relief valve	1 to 2
	TMQ_2B_2	RCS pump seal leak	4 to 16
BWR w/HPCI-RCIC	TMU_1B_1	HPCI/RCIC unavailable	1 to 2
	TMU_2B_2	DC power or condensate exhausted, component operability limits exceeded (HPCI/RCIC)	4 to 16
BWR w/HPCS-RCIC	TMU_1B_1	HPCS/RCIC unavailable	1 to 2
	TMU_2B_2	HPCS unavailable, DC power or condensate exhausted, component operability limits exceeded (RCIC)	4 to 16

Notes:
DHR = decay heat removal
AFWS = auxiliary feedwater system
RCS = reactor coolant system
HPCS = high pressure core spray
RCIC = reactor core isolation cooling
HPCI = high pressure coolant inspection

The dominant accident sequences were modeled as either an early core cooling failure or as a subsequent loss of core cooling. In the former case, the likelihood of the accident sequence is given by the probability of a station blackout combined with the probability of failure to maintain adequate core cooling or decay heat removal by AC-independent means long enough to cause core damage. For PWRs and most BWR-2 and -3 plants that do not have a makeup capability independent of AC power, there are two paths to inadequate core cooling early during station blackout. The first involves failure of the turbine-driven train of the auxiliary feedwater system (AFWS) in PWRs or failure of the isolation condenser in the BWR-2 and -3 plants. Because neither of these reactor types has a makeup capability independent of AC power, the core will be uncovered early by a major loss of reactor coolant system (RCS) integrity such as a stuck-open

relief valve or gross failure of reactor coolant pump seals, either of which could result in leak rates upwards of several hundred gpm. BWRs with reactor core isolation cooling (RCIC) systems, steam turbine-driven high pressure coolant injection (HPCI) systems, or high pressure core spray (HPCS) systems with a dedicated diesel generator can cool the reactor core and have the potential to make up losses of coolant equal to or greater than those identified above. The latter type of sequence was modeled as the likelihood of a station blackout of a duration sufficient to exceed core cooling systems capabilities and allow core damage to occur. If decay heat removal is initially successful, if reactor coolant leakage rates do not exceed makeup capability, and if primary coolant inventory requirements are met, operators should be able to establish a relatively stable decay heat removal mode. However, decay heat removal capability during longer blackouts may be limited by the capacity of support systems such as DC power or compressed air, by reactor coolant leakage when makeup is unavailable or insufficient, or by thermal limitations on component operability as a result of the loss of heating, ventilation, and air conditioning systems.

In light of the above discussion, the general form of the core damage accident likelihood equation considering both early phase and longer term decay heat removal failure is as follows:

$$P_{SBCD} = P_{SB}(t_1)\,(P_{DHR/SB} + P_{LOCA/SB}) + P_{SB}(t_2) \qquad (1)$$

where P_{SBCD} is the probability of core damage due to station blackout, $P_{SB}(t_1)$ is the probability of a station blackout of duration t_1 and t_1 is a time sufficient for core damage to occur if all decay heat removal capability is lost at the onset of a station blackout. $P_{DHR/SB}$ is the probability of decay heat removal failure on demand given station blackout. $P_{LOCA/SB}$ is the probability of a station-blackout-induced loss of reactor coolant integrity that would cause an early core cooling loss. $P_{SB}(t_2)$ is the probability of a station blackout of duration t_2, where t_2 is a time sufficient for core damage to occur because decay heat removal capability limits are exceeded during an extended duration station blackout.

In terms of the notation used to describe the dominant accident sequences for the various types of light water reactors (LWRs) identified in Table C.1, the equation can be written as follows:

for PWRs: $$P_{SBCD} = TMB_1(L_1 + Q_1) + TMB_2 \qquad (2)$$

for BWR 2/3s: $$P_{SBCD} = TMB_1(U_1 + Q_1) + TMB_2 \qquad (3)$$

for BWR 4/5/6s: $$P_{SBCD} = TMB_1U_1 + TMB_2 \qquad (4)$$

The probabilities for $(L_2 + Q_2)$, $(U_2 + Q_2)$, and U_2 have been set equal to 1.0, because the time of B_2 was selected to represent loss of decay heat removal capability as a result of design limitations. The probability contribution to Q_1 from reactor coolant pump seals degradation during station blackout is not well known. Based on material reviewed in NUREG/CR-3226, the impact of reactor

coolant pump seal leakage was assumed to represent a potential limit on the TMB_2 type of sequences.

The TMB_1 portion of equations 2, 3 and 4 above can be estimated from the first term failure-to-start portion of the station blackout equations in Appendix B of this report. The TMB_2 term of these equations can be estimated from the complete station blackout equations in Appendix B. Probability estimates for L_1, U_1 and Q_1 were derived from NUREG/CR-3226 and are summarized in Table C.2.

Table C.2 Decay heat removal failure probability for loss of core cooling early during station blackout

System/train/component	Probability of failure
Auxiliary feedwater systems	
1 steam turbine-driven train	0.04
2 steam turbine-driven trains	0.002
Isolation condenser	0.01
Stuck-open safety relief valve (BWR)	0.025
HPCI/RCIC	0.005
HPCS/RCIC	0.001

Estimated values of the early loss of core cooling term of equations 2, 3, and 4 are provided in Table C.3. This table shows the sensitivity of the estimated frequency of early core cooling failure during station blackout on loss-of-off-site power characteristics (clusters 1 through 5), emergency AC power unreliability (EDGR (i.e., failures per demand) and decay heat removal unreliability (DHR). The second term estimates of equations 2, 3, and 4 are the same as the station blackout frequency and duration assessments provided previously, given that t_2 is defined. Because the capability limitations vary from plant to plant, so will t_2. Some example estimates for the total core damage frequencies given capacity limitations which equate to station blackout durations of 2, 4, 8, and 16 hours are provided in Table C.4. These estimates include the early core cooling failure frequencies from Table C.3.

The results in Tables C.3 and C.4 show that the frequency and duration probabilities of offsite power failures, emergency AC power configuration, and reliability of the diesels are the most important factors in limiting the likelihood of core damage. These results also show that the likelihood of significant core damage may exist at some plants if the capability to cope with station blackout of modest durations (2 to 8 hours) does not exist. Moreover, the results show that the demand reliability of AC-independent decay heat removal systems is important, but it is not the most dominant factor in limiting the likelihood of core damage for station blackout.

Table C.3 Estimated frequency of early core cooling failure during station blackout, per reactor-year

DHR	EDGR	Offsite power cluster 1	2	3	4	5
		1/2 EDG configuration				
0.05	0.1	2.5E-6	8.0E-6	1.9E-5	3.0E-5	9.0E-5
	0.1	2.5E-6	8.0E-6	1.9E-5	3.0E-5	9.0E-5
	0.05	1.0E-6	3.4E-6	7.5E-6	1.3E-5	3.8E-5
	0.025	5.5E-7	1.9E-6	4.1E-6	7.5E-6	2.2E-5
	0.01	4.0E-7	1.4E-6	2.8E-6	5.5E-6	1.5E-5
0.01	0.1	5.0E-7	1.6E-6	3.7E-6	5.9E-6	1.8E-5
	0.05	2.0E-7	6.7E-7	1.5E-6	2.5E-6	7.5E-6
	0.025	1.1E-7	3.8E-7	8.2E-7	1.5E-6	4.3E-6
	0.01	8.0E-8	2.7E-7	5.6E-7	1.1E-6	3.0E-6
0.005	0.1	2.5E-7	8.0E-7	1.9E-6	3.0E-6	9.0E-6
	0.05	1.0E-7	3.4E-7	7.5E-7	1.3E-6	3.8E-6
	0.025	5.5E-8	1.9E-7	4.1E-7	7.5E-7	2.2E-6
	0.01	4.0E-8	1.4E-7	2.8E-7	5.5E-7	1.5E-6
		2/3 EDG configuration				
0.05	0.1	7.0E-6	2.3E-5	5.0E-5	8.0E-5	2.5E-4
	0.05	2.6E-6	8.5E-6	1.9E-5	3.1E-5	9.5E-5
	0.025	1.3E-6	4.3E-6	9.5E-6	1.7E-5	4.9E-5
	0.01	8.5E-7	2.8E-6	6.0E-6	1.1E-5	3.2E-5
0.01	0.1	1.4E-6	4.5E-6	1.0E-5	1.6E-5	4.9E-5
	0.05	5.2E-7	1.7E-6	3.8E-6	6.2E-6	1.9E-5
	0.025	2.6E-7	8.6E-7	1.9E-6	3.3E-6	9.7E-6
	0.01	1.7E-7	5.5E-7	1.2E-6	2.1E-6	6.3E-6
0.005	0.1	7.0E-7	2.3E-6	5.0E-6	8.0E-6	2.5E-5
	0.05	2.6E-7	8.5E-7	1.9E-6	3.1E-6	9.5E-6
	0.025	1.3E-7	4.3E-7	9.5E-7	1.7E-6	4.9E-6
	0.01	8.5E-8	2.8E-7	6.0E-7	1.1E-6	3.2E-6
		1/3 EDG configuration				
0.05	0.1	3.6E-7	1.2E-6	2.6E-6	9.3E-6	1.3E-5
	0.05	1.8E-7	6.0E-7	1.3E-6	2.3E-6	6.5E-6
	0.025	1.5E-7	4.9E-7	1.1E-6	1.9E-6	5.5E-6
	0.01	1.4E-7	4.6E-7	1.0E-6	1.8E-6	5.0E-6
0.01	0.1	7.1E-8	2.3E-7	5.2E-7	8.6E-7	2.6E-6
	0.05	3.6E-8	1.2E-7	2.6E-7	4.5E-7	1.3E-6
	0.025	2.9E-8	9.7E-8	2.1E-7	3.7E-7	1.1E-6
	0.01	2.7E-8	9.1E-8	2.0E-7	3.5E-7	1.0E-6

Table C.3 (continued)

DHR	EDG$\overline{R}$	Offsite power cluster 1	2	3	4	5
		<u>2/4 EDG configuration</u>				
0.01	0.1	2.3E-7	7.5E-7	1.7E-6	2.7E-6	8.3E-6
	0.05	8.6E-8	2.8E-7	6.2E-7	1.1E-6	3.2E-6
	0.025	5.7E-8	1.9E-7	4.1E-7	7.2E-7	1.8E-6
	0.01	4.8E-8	1.6E-7	3.4E-7	6.1E-7	1.1E-6
0.005	0.1	1.2E-7	3.8E-7	8.5E-7	1.4E-6	4.2E-6
	0.05	4.3E-8	1.4E-7	3.1E-7	5.5E-7	1.6E-6
	0.025	2.9E-8	9.5E-7	2.1E-7	3.6E-7	9.0E-7
	0.01	2.4E-8	8.0E-7	1.7E-7	3.1E-7	5.5E-7

Table C.4 Tabulated estimated values of total core damage frequency for station blackout accidents as a function of emergency diesel generator configuration, EDG unreliability, offsite power cluster, and ability to cope with station blackout

EDG$\overline{R}$ and t(hr)	Offsite power cluster 1	2	3	4	5
	<u>1/2 AC configuration</u>				
<u>EDG$\overline{R}$ = 0.1</u>					
2	5.1E-5	1.7E-4	3.8E-4	6.1E-4	1.9E-3
4	2.0E-5	6.8E-5	1.5E-4	2.9E-4	9.0E-4
8	6.3E-6	2.2E-5	4.0E-5	1.0E-4	2.5E-4
16	5.0E-7 to 2.4E-6	2.0E-6 to 8.2E-6	2.4E-6 to 1.6E-5	9.6E-6 to 3.2E-5	1.2E-5 to 8.4E-5
<u>EDG$\overline{R}$ = 0.05</u>					
2	2.1E-5	6.9E-5	1.5E-4	2.5E-4	7.7E-4
4	8.7E-6	2.9E-5	6.2E-5	1.3E-4	3.8E-4
8	2.8E-6	1.0E-5	1.7E-5	4.5E-5	1.1E-4
16	2.2E-7 to 1.0E-6	9.1E-7 to 3.5E-6	1.1E-6 to 6.7E-6	4.4E-6 to 1.4E-5	6.8E-6 to 3.5E-5
<u>EDG$\overline{R}$ = 0.025</u>					
2	1.2E-5	3.9E-5	8.3E-5	1.6E-4	4.4E-4
4	5.2E-6	1.8E-5	3.6E-5	7.9E-5	2.2E-4
8	1.7E-6	6.1E-6	1.0E-5	2.8E-5	6.2E-5
16	1.4E-7 to 5.8E-7	5.8E-7 to 2.0E-6	6.3E-7 to 3.7E-6	2.8E-6 to 8.6E-6	4.2E-6 to 2.0E-5

Table C.4 (continued)

EDG$\overline{R}$ and t(hr)	Offsite power cluster 1	2	3	4	5
	1/2 AC configuration				
EDG$\overline{R}$ = 0.01					
2	8.3E-6	2.8E-5	5.7E-5	1.1E-4	3.1E-4
4	3.8E-6	1.3E-5	2.5E-5	5.9E-5	1.6E-4
8	1.3E-6	4.5E-6	7.1E-6	2.1E-5	4.6E-5
16	1.1E-7 to 4.1E-7	4.5E-7 to 1.5E-6	4.7E-7 to 2.6E-6	2.2E-6 to 6.4E-6	3.2E-6 to 1.5E-5
	1/3 AC configuration				
EDG$\overline{R}$ = 0.1					
2	7.3E-6	2.4E-5	5.3E-5	8.8E-5	2.7E-4
4	2.5E-6	8.1E-6	1.8E-5	3.5E-5	1.1E-4
8	5.9E-7	2.1E-6	3.8E-6	9.2E-6	2.3E-5
16	3.0E-8 to 8.0E-7	1.1E-7 to 9.9E-7	1.7E-7 to 2.2E-6	5.0E-7 to 3.8E-6	9.8E-7 to 1.1E-5
EDG$\overline{R}$ = 0.05					
2	3.7E-6	1.2E-5	2.7E-5	4.6E-5	1.4E-4
4	1.3E-6	4.2E-6	9.2E-6	1.9E-5	5.6E-5
8	3.1E-7	1.1E-6	1.9E-6	4.8E-6	1.2E-5
16	1.5E-8 to 1.5E-7	5.7E-8 to 5.1E-7	8.6E-8 to 1.1E-6	2.6E-7 to 2.0E-6	5.0E-7 to 5.6E-6
EDG$\overline{R}$ = 0.025					
2	3.0E-6	9.9E-6	2.2E-5	3.8E-5	1.1E-4
4	1.1E-6	3.6E-6	7.5E-6	1.6E-5	4.6E-5
8	2.6E-7	9.0E-7	1.5E-6	4.0E-6	9.7E-6
16	1.2E-8 to 1.2E-7	4.8E-8 to 4.2E-7	6.8E-8 to 8.7E-7	2.1E-7 to 1.6E-6	4.1E-7 to 4.5E-6
EDG$\overline{R}$ = 0.01					
2	2.8E-6	9.3E-6	2.0E-5	3.6E-5	1.1E-4
4	9.7E-7	3.3E-6	6.9E-6	1.5E-5	4.3E-5
8	2.4E-7	8.3E-7	1.5E-6	3.7E-6	8.9E-6
16	1.1E-8 to 1.3E-7	4.3E-8 to 3.9E-7	6.4E-8 to 8.1E-7	2.0E-7 to 1.5E-6	3.8E-7 to 4.2E-6

Table C.4 (continued)

EDG$\bar{R}$ and t(hr)	Offsite power cluster				
	1	2	3	4	5
			2/3 AC configuration		
EDG$\bar{R}$ = 0.1					
2	1.4E-4	4.6E-3	1.1E-3	1.7E-3	5.0E-3
4	5.4E-5	1.8E-4	4.1E-4	7.6E-4	2.4E-3
8	1.7E-5	5.8E-5	1.1E-4	2.6E-4	6.6E-4
16	1.3E-6 to 6.5E-6	5.1E-6 to 2.2E-5	6.4E-6 to 4.5E-5	2.4E-5 to 8.5E-5	4.0E-5 to 2.3E-4
EDG$\bar{R}$ = 0.05					
2	5.3E-5	1.8E-4	3.9E-4	6.4E-4	2.0E-3
4	2.1E-5	6.9E-5	1.6E-4	3.0E-4	9.4E-4
8	6.5E-6	2.3E-5	4.1E-5	1.0E-4	2.6E-4
16	4.9E-7 to 2.5E-6	2.0E-6 to 8.4E-6	2.4E-6 to 1.7E-5	9.4E-6 to 3.3E-5	1.5E-5 to 8.7E-5
EDG$\bar{R}$ = 0.025					
2	2.7E-5	8.9E-5	2.0E-4	3.4E-4	1.0E-3
4	1.2E-5	3.7E-5	8.0E-5	2.7E-4	4.9E-4
8	3.4E-6	1.2E-5	2.1E-5	5.5E-5	1.3E-4
16	2.5E-7 to 1.3E-6	1.0E-6 to 4.3E-6	1.2E-6 to 8.5E-6	4.9E-6 to 1.7E-5	7.8E-6 to 4.5E-5
EDG$\bar{R}$ = 0.01					
2	1.7E-5	5.1E-5	1.3E-4	2.2E-4	6.4E-4
4	7.3E-6	2.4E-5	5.1E-5	1.1E-4	3.1E-4
8	2.2E-6	7.7E-6	1.3E-5	3.6E-5	8.4E-5
16	1.6E-7 to 8.0E-7	6.5E-7 to 2.8E-6	7.6E-7 to 5.3E-6	3.1E-6 to 1.1E-5	4.9E-6 to 2.9E-5
			2/4 AC configuration		
EDG$\bar{R}$ = 0.1					
2	2.4E-5	7.7E-5	3.5E-5	2.8E-4	8.5E-4
4	7.2E-6	2.5E-5	1.1E-5	1.1E-4	3.5E-4
8	1.8E-6	6.2E-6	2.1E-6	2.7E-5	7.0E-5
16	9.6E-7 to 9.8E-8	3.4E-7 to 3.2E-6	9.3E-8 to 1.4E-6	1.5E-6 to 1.2E-5	3.1E-6 to 3.5E-5

Table C.4 (continued)

EDG$\bar{R}$ and t(hr)	Offsite power cluster 1	2	3	4	5
	2/4 AC configuration				
EDG$\bar{R}$ = 0.05					
2	8.8E-6	2.9E-5	6.3E-5	1.1E-4	3.3E-4
4	2.9E-6	1.0E-6	2.1E-5	4.2E-5	1.3E-4
8	6.5E-7	2.3E-6	4.1E-6	1.0E-5	2.5E-5
16	3.2E-8 to 3.6E-7	1.2E-7 to 1.2E-6	1.9E-7 to 2.6E-6	5.2E-7 to 4.6E-6	1.1E-6 to 1.3E-5
EDG$\bar{R}$ = 0.025					
2	5.8E-6	2.0E-5	4.2E-5	7.3E-5	2.2E-4
4	1.9E-6	6.4E-6	1.4E-5	2.9E-5	8.2E-5
8	4.2E-7	1.5E-6	2.6E-6	6.5E-6	1.6E-5
16	2.0E-8 to 2.4E-7	7.3E-8 to 7.9E-7	1.2E-7 to 1.7E-6	3.2E-7 to 3.1E-6	6.6E-7 to 8.7E-6
EDG$\bar{R}$ = 0.01					
2	4.8E-6	1.6E-5	3.6E-5	6.2E-5	1.8E-4
4	1.5E-6	5.3E-6	1.1E-5	2.4E-5	6.8E-5
8	2.5E-7	1.2E-6	2.1E-6	5.4E-6	1.3E-5
16	6.1E-9 1.6E-8	5.8E-8 6.6E-7	9.3E-8 1.4E-6	2.5E-7 2.6E-6	5.3E-7 7.2E-6

The point estimates obtained from NUREG/CR-3226 and a comparable plant design analyzed in this study are shown in Table C.5. The differences in results primarily result from lower loss-of offsite-power frequencies supported by most recent evaluations of the data (see Appendix A).

The results provided up to this time represent point estimates of probability per year or, more properly, frequency. The effect on the mean probability estimates of using log-normal distributions to represent basic event probabilities, calculated medians, and uncertainty ranges was shown in NUREG/CR-3226. The sequence mean estimates derived in that document were typically 3 to 8 times larger than the point estimates, and the upper and lower bounds were typically within a factor of 5 to 20 of the median estimates. The large difference between point estimates and means can be attributed to the use of a log-normal distribution.

The potential effect of operator error causing loss of decay heat removal has not been found to be a large contributor, if adequate training and procedures exist. Another consideration that has not been found to be a significant factor is the difference in time to core uncovery for the various LWR designs on loss of all decay heat removal.

Table C.5 Comparison of results with NUREG/CR-3226

Plant type and sequence	Core damage frequency (per reactor-year)	
	NUREG/CR-3226	NUREG-1032
PWR with one steam-driven AFW train		
TML_1B_1	5×10^{-6}	1.5×10^{-6}
$TMB_2(L_2 + Q_2)$	2×10^{-5}	9.2×10^{-6}
BWR with isolation cooling		
$TM(U_1 + Q_1)B_1$	5×10^{-6}	1.3×10^{-6}
TMQ_2B_2	2×10^{-5}	9.2×10^{-6}
BWR with HPCI/RCIC		
TMU_1B_1	2×10^{-6}	1.9×10^{-7}
TMU_2B_2	2×10^{-5}	9.2×10^{-6}
BWR with HPCS/RCIC		
TMU_1B_1	5×10^{-7}	3.8×10^{-7}
TMU_2B_2	1×10^{-6}	5.2×10^{-6}

Note: All B_2 sequences except the BWR with HPCS/RCIC are assumed to result in loss of core cooling and decay heat removal in 6 hours from the start of station blackout for the NUREG-1032 results. Core damage frequencies in this table (NUREG-1032 column) are based on offsite power cluster 2, 1/2 diesel generator configuration and 0.975 diesel generator reliability.

STATION BLACKOUT RISK

The potential risk associated with station blackout accidents can be estimated by extending the core damage probabilistic results through to accident consequence estimates. The potential for terminating core damage before core melt and coping with core melt before containment failure is currently a matter of extensive research and evaluation. In most probabilistic risk assessments (PRAs), the probability of core damage has been equated with core melt. Acknowledging that this is a possible conservative assumption, to estimate risk in these PRAs, containment failure modes and probabilities are applied as if the core has melted.

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ENCLOSURE F

Beckjord for Appropriate Action

Cys: Stello
Taylor
Rehm
Murley
Thompson
Jordan
Murray
Matt Taylor
CFiles

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

June 9, 1987

The Honorable Lando W. Zech, Jr.
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Chairman Zech:

SUBJECT: ACRS COMMENTS ON THE NRC STAFF PROPOSAL FOR THE RESOLUTION OF USI A-44, "STATION BLACKOUT"

During the 326th meeting of the ACRS, June 4-6, 1987, and in our 325th meeting on May 7-9, 1987, we discussed the resolution of USI A-44, "Station Blackout," that is being proposed by the NRC Staff. We also discussed the Nuclear Utility Management and Resources Committee (NUMARC) initiatives directed at reducing the risk from "Station Blackout." A Subcommittee meeting was also held to discuss this issue with the NRC Staff on May 6, 1987. During these meetings, we had the benefit of presentations by representatives of the NRC Staff and NUMARC. We also had the benefit of the documents referenced.

Since March 30, 1982, members of the ACRS have considered and discussed this issue at nine meetings, and offered comments to the Executive Director for Operations in letters dated July 13, 1983 and March 12, 1985. The ACRS has been generally receptive to and supportive of the Staff's efforts in seeking resolution of the issue.

We consider the proposed resolution of USI A-44, "Station Blackout," to be workable, and we commend the Staff for its efforts. However, we do not recommend issuance of the final rule at this time.

We believe that the NUMARC initiatives may be a viable alternative for dealing with this issue on an expeditious schedule and may require the least expenditure of resources on the part of the industry. We believe that the electric utility industry has a strong incentive to deal with "Station Blackout."

One shortcoming of the proposed NUMARC initiatives is the absence of a requirement for any assessment of a plant's ability to cope with station blackout for a specified length of time. A letter from NUMARC has advised us that they are developing a methodology to do this, but that industry-wide agreement will have to be obtained. They expect that the development of their initiatives will be substantially completed by September of this year.

We recommend that the Staff continue to work with NUMARC on the technical aspects of the NUMARC efforts. If by September of this year it is determined by the Staff that the NUMARC initiatives will not be effective or timely in reducing the risk from "Station Blackout" to acceptable levels, or that the NUMARC initiatives will be unduly difficult to evaluate on a plant-to-plant basis, we then recommend issuance of the final rule.

Additional remarks by ACRS Members Glenn A. Reed and Charles J. Wylie are presented below.

Sincerely,

WKerr

William Kerr
Chairman

Additional Remarks by ACRS Members Glenn A. Reed and Charles J. Wylie

We believe the NRC Staff has done a commendable job in bringing A-44 to resolution. However, we continue to support two previous ACRS letters (July 13, 1983 and March 12, 1985) recommending in part that A-44 implementation should be integrated with A-45, "Shutdown Decay Heat Removal Requirements." Unfortunately A-45 has not arrived at the same status, and the NRC Staff wishes to proceed now with a rule and guide on station blackout which deal with A-44 only. But, the root issue is not station blackout but rather decay heat removal to limit core melt risk to an appropriate level.

We do not consider it in the best interest of nuclear safety to proceed now with an NRC rule and guide on station blackout, which could compromise future desirable and more effective action for decay heat removal. Since it appears that NUMARC-Nuclear Utilities Group on Station Blackout (NUGSBO) has also been moving forward with an industry effort, and since the electric utilities should have premiere capabilities to upgrade vulnerabilities to station electrical blackout, we recommend NUMARC-NUGSBO carry the ball, with NRC Staff interfacing and monitoring -- but without an NRC rule. This arrangement would leave the NRC uncompromised to act appropriately on A-45 when its resolution is completed. In our opinion there may be some outlier units for which it is more preferable to focus and expend funds on the root issue of decay heat removal without diverting effort to station blackout; and such focusing may be more harmonious with the backfit rule.

References:
1. U.S. Nuclear Regulatory Commission, Federal Register Notice (51 FR 9829) for the proposed Station Blackout Rule (10 CFR 50.63), published on March 21, 1986.
2. U.S. Nuclear Regulatory Guide on "Station Blackout," dated March 30, 1987.
3. U.S. Nuclear Regulatory Commission, NUREG-1109, "Regulatory/Backfit Analysis for the Resolution of Unresolved Safety Issue A-44," submitted March 30, 1987.
4. U.S. Nuclear Regulatory Commission, NUREG-1032, "Evaluation of Station Blackout Accidents at Nuclear Power Plants," draft, submitted April 16, 1987.
5. U.S. Nuclear Regulatory Commission, NUREG/CR-3226, "Station Blackout Accident Analyses," dated May 1983.
6. U.S. Nuclear Regulatory Commission, NUREG/CR-2989, "Reliability of Emergency AC Power Systems at Nuclear Power Plants, dated July 1983.

ENCLOSURE G

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

June 23, 1987

MEMORANDUM FOR: Victor Stello, Jr.
Executive Director for Operations

FROM: Edward L. Jordan, Chairman
Committee to Review Generic Requirements

SUBJECT: MINUTES OF CRGR MEETING NUMBER 115

The Committee to Review Generic Requirements (CRGR) met on Wednesday, May 27, 1987, from 1-5 p.m. A list of attendees for this meeting is enclosed (Enclosure 1). The following items were addressed at the meeting:

1. W. Minners (RES), P. Baranowsky (NRR), and A. Rubin (RES) presented for CRGR review the proposed final rule and regulatory guide resolving USI A-44, "Station Blackout." The CRGR recommended EDO approval for transmittal to the Commission, subject to some changes in the documents that would be coordinated with the CRGR staff prior to transmittal to the EDO. The CRGR also recommended that the rule not be implemented until certain additional implementation guidance for the staff is prepared and reviewed by the CRGR. Further, the CRGR recommended that the staff's proposed implementation schedule should be shortened. This matter is discussed in Enclosure 2.

2. The proposed final rule amendments to 10 CFR Parts 30, 40, 50, 51, 70, and 72 concerning general requirements for decommissioning nuclear facilities, which were scheduled for review, were not reviewed at this meeting. The review was postponed until the next scheduled CRGR meeting.

3. G. Arlotto (RES) and G. Millman (RES) presented for CRGR review the proposed rule amendments to 10 CFR 50.55a, "Codes and Standards." The CRGR recommended that the EDO approve issuance of the proposed amendments for public comment. This matter is discussed in Enclosure 3.

In accordance with the EDO's July 18, 1983 directive concerning "Feedback and Closure on CRGR Reviews," a written response is required from the cognizant office to report agreement or disagreement with CRGR recommendations in these minutes. The response, which is required within five working days after receipt of these meeting minutes, is to be forwarded to the CRGR Chairman and if there is disagreement with the CRGR recommendations, to the EDO for decisionmaking.

Questions concerning these meeting minutes should be referred to Tom Cox (492-4148).

Edward L. Jordan, Chairman
Committee to Review Generic
Requirements

Enclosures:
As stated

cc: Commission (5)
SECY
Office Directors
Regional Administrators
CRGR Members
W. Parler
B. Sheron
G. Arlotto

Enclosure 1
LIST OF ATTENDEES
CRGR MEETING NO. 115

May 27, 1987

CRGR MEMBERS

E. Jordan
R. Bernero
T. Martin
J. Scinto
T. Speis
J. Sniezek

OTHERS

J. Zerbe
T. Cox
J. Conran
A. Rubin
P. Baranowsky
K. Kniel
B. Colmar
D. Jones
A. Serkiz
M. El-Zeftawy
J. Flack
J. Larkins
M. Federline
J. Austin
M. Taylor
J. Clifford
R. Fonner
P. Norian
M. Malsch
W. Minners
R. Bosnak
S. Treby
G. Millman
G. Arlotto
T. Dorian

Enclosure 2 to the Minutes of CRGR Meeting No. 115
Proposed Final Resolution for USI A-44, "Station Blackout"
May 27, 1987

TOPIC

W. Minners (RES), P. Baranowsky (RES), and A. Rubin (RES) presented for CRGR review the proposed final resolution for Unresolved Safety Issue (USI) A-44, "Station Blackout." The proposed resolution calls for amendments to 10CFR50 (including an addition to GDC-17) and issuance of an associated Reg. Guide which would require that licensees and applicants: (a) assess the vulnerability of their plants to blackout (i.e. simultaneous loss of offsite AC power and unavailability of onsite AC power), and (b) demonstrate the capability of their plants to cope with the effects of blackouts of specified duration if they occur. Copies of briefing slides used by the staff to guide their presentations and discussions with the Committee at this meeting are attached to this Enclosure.

BACKGROUND

The proposed resolution for USI A-44 was reviewed by CRGR earlier, at the draft stage, in Meeting Nos. 59, 61, and 60; the complete account of that earlier review and the resulting recommendations of the Committee are documented in the minutes of Meeting No. 60 dated May 8, 1984. The current package submitted for review by CRGR at this final stage was transmitted by memorandum dated April 6, 1987, H. R. Denton to J. E. Zerbe; the current package included the following review documents:

1. Enclosure 1 — Proposed Federal Register Notice, dated March 30, 1987, (Response to public comments, proposed rule amendments, backfit analysis)

2. Enclosure 2 — Regulatory Guide, dated March 30, 1987, "Station Blackout"

3. Enclosure 3 — NUREG-1109, undated, "Regulatory/Backfit Analysis for the Resolution of Unresolved Safety Issue USI A-44, Station Blackout"

4. Enclosure 4 — NUREG-1032, undated, "Evaluation of Station Blackout Accidents at Nuclear Power Plants"

5. Enclosure 5 — "Significant Changes to the USI A-44 Package," undated (circa March/April 1987)

The Commission received 53 letters commenting on the draft resolution package reviewed earlier by CRGR; those comments are summarized and responded to by the staff in the proposed Federal Register Notice (FRN) included in the current review package (Background Item 1 above). One of the public comment letters, from the Nuclear Utility Management and Resources Committee (NUMARC), put forward as a possible alternative to the staff's proposed resolution for USI A-44 four initiatives designed to address the more important contributors

to blackout. Those initiatives were endorsed broadly by the utilities currently operating nuclear power plants, and they have been pursued actively by the utilities in parallel with the staff's work on this issue over the last couple of years. The NUMARC initiatives are included as one of the five possible alternative courses of action examined by the staff in regulatory and backfit analyses done in connection with USI A-44; those analyses are summarized and documented in NUREG-1109 (Background Item 3 above). A method acceptable to the NRC staff for licensees to determine plant-specific blackout coping durations for their facilities is provided in the proposed Reg. Guide (Background Item 2 above) included in the current review package. Technical studies funded by NRC to examine the most important parameters of the station blackout issue (i.e., frequency of loss of offsite power; probability that onsite AC will fail to provide power for core cooling; capability and reliability of shutdown cooling systems during prolonged blackout; ability of containment to withstand pressure/temperature buildup during prolonged blackout; and probability of occurrence of blackout accident sequences) are summarized in NUREG-1032 (Background Item 4 above). Significant changes made to the USI A-44 package since the Committee's earlier review at the draft stage are summarized in Background Item 5.

Briefly, the staff's proposed resolution for USI A-44 includes: (a) amendments to 10CFR50 and Appendix A to Part 50 (General Design Criteria), which would require that all nuclear power plants be able to cope with a station blackout for a specified duration and have procedures and training for such an event, and (b) issuance of a supporting Regulatory Guide which would provide an acceptable method for determining the station blackout duration for each plant. The staff's analyses indicate that an improvement of about 2.6E-5 in the frequency/reactor year of core damage events will be realized by implementing the proposed A-44 fix (decreasing from about 4.2E-5 before fix to 1.6E-5 after fix). This corresponds to a best-estimate total averted risk benefit of about 145K person-rem for a population of 100 reactors over their remaining lifetime (about 30 years average). The cost of implementing the proposed fix is estimated by the staff to be about $600K per plant, for a total cost to the industry of about $60M (i.e., 60 million dollars). The benefit/cost ratio for the proposed action, then, is 2400 person-rem/$M (or about $420/person-rem, compared with $1000/person-rem commonly used as the breakeven guideline).

DISCUSSION

Major points of discussion at this meeting regarding the proposed resolution for USI A-44 were as follows:

1. A major change in the A-44 package from the draft stage to the current version is that licensees would no longer be required to determine the maximum duration that their plants could cope with blackout conditions. Instead, licensees would determine, by applying the screening/categorization guidance in the proposed Station Blackout Reg. Guide, a specified coping duration (i.e., 2 hours, 4 hours, 8 hours or 16 hours) appropriate for each facility. They would then be required to demonstrate by further analyses that each plant could actually cope with blackout conditions for that specified duration. The staff did not, however, include in the revised package the proposed acceptance criteria/guidance (e.g. SRPs) that would be used by staff reviewers for judging the acceptability of the

coping analyses to be submitted by the licensees under the proposed resolution, or the inspection guidance (e.g., TIs) that would be applied by NRC inspectors in determining ultimately licensees' conformance to the proposed rule. In the absence of such criteria, the Committee felt that the proposed coping demonstration requirement is much too open-ended, and that the staff's estimate of cost to implement the proposed fix (i.e., about $250K/plant) is not verifiable. In this context, it was noted that the actual cost of analyses done by the St. Lucie 2 applicant to demonstrate a 4-hour blackout coping capability (after Station Blackout was made a design basis event by the Licensing Board for that facility) was said by NUGSBO to be in the neighborhood of about $2 million.

The staff stated that they had no detailed knowledge of actual costs incurred in that instance. They were, however, generally aware of the circumstances involved; and they said that the extensive analyses done by the licensee in that case (which included even some thermodynamic analyses) were done in the process of demonstrating the 4-hour coping duration specified by the Board. Extensive analyses were apparently done by the licensee in a futile attempt to avoid having to make significant physical modifications to the plant. The staff felt strongly that those circumstances were not representative for the remaining plants that would be affected by this proposed blackout rule; and they stated categorically that the coping analyses envisioned in connection with the currently proposed USI A-44 resolution were not intended to be as extensive as those done by the St. Lucie applicant.

After much discussion, the Committee concluded that the A-44 package did not have to be held up from going forward because of their concerns on this point; but they recommended that both the acceptance criteria and guidance to be used by staff reviewers (i.e., SRPs), and the guidance to be used by NRC inspectors (i.e., TIs) in implementing the new rule, should be reviewed by the CRGR prior to implementation of any station blackout rule that is finally approved. The staff agreed with this recommendation; RES and NRR will cooperate in this effort, and will undertake immediately the expeditious development of the acceptance criteria and review/inspection guidance needed by staff reviewers and inspectors. The Commission currently is not scheduled to consider this matter until late summer or early fall of this year; it was felt, therefore, that there is a good chance that the recommended criteria/guidance could be available at the time the Commission gives final consideration to the proposed resolution package.

2. The credit to be given to "alternate" AC sources for coping with blackout was another major area of discussion at this meeting. This had been a major point of comment by NUMARC in putting forward their initiatives as an alternative to the staff's approach for resolving A-44. The staff is willing to credit such AC sources for coping purposes; but they have had some difficulty in specifying detailed acceptance criteria, i.e., what degree of independence, diversity, reliability, etc. is required for such AC sources to qualify as adequate for coping. Although they have not fully developed their thinking in this regard, the staff indicated in these discussions, that a "swing" diesel generator configuration would not qualify as an alternate AC source adequate for coping with blackout,

because of its lack of independence from the normal safety-related AC systems and its common-mode failure potential. Generally, however, if an alternate source can be shown to be reliable, independent, and available within 10 minutes or so of the onset of blackout, the staff will credit it fully. The staff stated that such a plant would, in effect, fall into the so-called "zero hour" category that was the subject of much discussion with CRGR at the draft review stage. That is to say, a fully credited alternate AC capability will be considered to be adequate demonstration of the required blackout coping capability for a plant, without the need for any additional coping analyses.

3. In discussing the utilities' reluctance to accept the proposed coping analysis requirement (as indicated above in Discussion Item 1), the staff offered the view that a major concern underlying the utilities' opposition on this point was that the staff might raise such extensive equipment operability questions in the review of such analyses as to effectively reopen the EQ issue. With respect to such concerns, the staff stated that, if the blackout coping capability claimed by a licensee relies on use of equipment that previously did not require any sort of environmental qualification, then its availability and operability in the blackout context would have to be reasonably established/demonstrated. This could be done by providing an evaluation of equipment characteristics and specifications vs. anticipated environmental conditions and equipment loads that could result from blackout, and by providing for administrative controls to reasonably assure availability of equipment designated to be relied on in coping with blackout. The staff felt that the licensees' concerns regarding the likelihood of extensive environmental qualification effort being required in connection with blackout were misplaced.

The CRGR staff noted that, in recent discussions with NUGSBO representatives on this point, the concern expressed was that the extensive coping analyses required under the staff's proposal would not actually fix any equipment or procedural deficiency that contributed to station blackout potential. The emphasis in the NUMARC initiatives was on physical equipment and procedural improvements that would directly address blackout concerns; they felt that this represented the most effective and cost beneficial expenditure of resources in addressing the blackout issue. With respect to demonstration of blackout coping capability, the utilities felt that staff's criteria for determining what was adequate in that regard (i.e., detailed analyses) were too stringent, and more like what is normally required to demonstrate adequacy of equipment/procedures judged necessary to assure public health and safety. The staff has not claimed that any of the proposed USI A-44 fixes are necessary to assure safety; they have only said that such fixes provide significant improvement to safety in a cost beneficial manner. In view of the relative importance (thus indicated) of A-44 fixes in general, the utilities believe that adequate demonstration of 2-hour, 4-hour, etc. coping capability should be provided by simply meeting the criteria specified by the staff in the proposed Station Blackout Reg. Guide for determining that a plant is a 2-hour plant, or a 4-hour plant, etc. The Committee agreed generally with the view that analyses in themselves do not fix equipment or procedural deficiencies that result in vulnerability to blackout. The staff

reiterated strongly, however, that the analyses specified are required to provide a high level of assurance that a given plant has the capability to cope with blackout conditions for the duration specified for that facility. The Committee asked whether the staff had considered allowing utilities to take credit for demonstrating operability of such equipment by actual testing under simulated blackout conditions during a plant outage. The staff responded that they would consider licensee proposals to demonstrate operability in that manner, and would revise the wording of the FRN to indicate this; but they would not require (or even necessarily encourage) that approach.

As a final point in regard to equipment operability and reliability, and their relationship to the overall demonstration of required coping capability, the Committee noted that, although the staff had stated explicitly at least once in the A-44 package that it was not intended that any equipment relied on for blackout coping must be safety grade, or that full scope Appendix B QA requirements must be applied to such equipment (e.g., see proposed FRN, at p. 12, last paragraph), the wording of the package in other places was such as to suggest that comparably detailed and stringent QA requirements might apply to such equipment. For example, the proposed FRN (at p. 12, second to last paragraph) states that:

> "...the equipment must meet certain quality assurance criteria to ensure a high level of reliability and operability during station blackout events."

The Committee commented that, because the objective of proposed rule is to further reduce risk in a cost beneficial manner (not to achieve an acceptable level of risk where reasonable assurance of that is currently lacking), they agreed with the staff approach of not requiring equipment relied on for coping with blackout to be safety grade. In this context, the Committee recommended that the staff clarify their intent with regard to intended safety classification of, and QA requirements for, such equipment by revising the wording of the package in the specific paragraphs noted, to identify explicitly and unambiguously the "certain QA criteria" that the staff is referring to there. The Committee also suggested that the remainder of the package be carefully reviewed for similar potentially confusing wording, and be revised as necessary to avoid possible confusion on this important point. The staff agreed that the wording cited was potentially confusing and should be changed (e.g., by deleting the word "certain" in the specific passage cited, and by indicating more clearly that the level of treatment given to quality in the Station Blackout Reg. Guide and in draft ANS Standard 58.12 is representative of what is expected by the staff for equipment relied on for coping with blackout). The staff will also review other pertinent sections of the A-44 package for other such inconsistencies or ambiguities.

4. With regard to the proposed requirement specifying that the Director, NRR, must make a determination of the specified station blackout duration for each affected operating plant (see proposed FRN, p. 45, third paragraph), the Committee asked what standard is to be used by the Director in making that determination. The staff responded that the operable standards are set forth in the proposed Station Blackout Reg. Guide in the guidance for categorizing existing plants (i.e., as 2-hour, 4-hour, 8-hour or 16-hour

plants). Beyond those explicit deterministic standards, there is also the probabilistic core damage frequency objective of about 10E-5 which underlies the staff's overall approach to resolution of this issue, as reflected in NUREG-1032 and NUREG-1109. The Committee agreed that either or both of these would be acceptable for use as the standard to be used for making the required determination; but they also stated that the intended standard(s) must be incorporated explicitly into the rule that requires such a determination to be made. The staff agreed to modify the wording of the package to reflect this recommendation.

5. The Committee questioned the staff regarding the proposed implementation schedule (see proposed FRN, p. 45, last two paragraphs). They noted that a stated objective of the NUMARC initiatives is to upgrade all 8-hour plants into the 4-hour category within one year; while under the staff's proposed resolution the upgrading of affected plants could drag on into the 1990's. The Committee recommended that NRR work out a more expeditious implementation schedule for their proposed resolution, consistent with the general objective of reducing schedules for high priority generic actions.

6. The Committee asked the staff to reaffirm that the wording in the next-to-last paragraph of the transmittal letter for this review package, and the wording in the proposed FRN (at p. 50, first paragraph), was intended to satisfy the requirement in 10CFR50.109 for an explicit finding by the Director, NRR that a proposed backfit will provide a substantial increase in the overall protection of the public health and safety, and is justified in view of the direct and indirect costs of implementing the backfit. It was noted that the referenced wording actually states that the results of the staff's analyses in NUREG-1032 and NUREG-1109 <u>support</u> such a determination, but do not state explicitly that that <u>is</u> the Director's determination. The staff agreed to verify that it was the (former) Director's intent to make the required finding when the review package was transmitted to CRGR, and to revise the wording of the package to more clearly reflect that intent.

7. The Committee asked if the staff had concluded <u>on the basis of actual analyses</u> that it is safer to shutdown a plant in the event of imminent severe weather (e.g., hurricane, tornado, blizzard, etc.) and rely on diesel generators and/or offsite power sources for power to maintain safe shutdown conditions, as strongly suggested by the staff's treatment of severe weather procedures in the A-44 context. It was suggested in this context, in view of the problems that have been experienced with DG reliability and the expectation that offsite power is likely to fail in the event of very severe weather, that it might be prudent to keep reactors critical but operating at reduced power levels in such circumstances. The staff responded that they had not done detailed comparative risk analyses on this question. Where such shutdown procedures exist, or in specific instances where shutdown actions have been taken in imminent severe weather situations (e.g., during a hurricane last year), those procedures/actions have been the result of licensees' evaluations of specific situations and alternatives, with the staff only deferring to or not objecting specifically to the licensees' judgment. It was agreed that it should not

be assumed automatically that shutdown of the reactor and reliance on DG power is the preferred course of action in all severe weather situations, but that this question should be examined on a case basis; and this approach will be taken in implementing any approved resolution for USI A-44.

8. The Committee inquired where in the rule is the concept of "alternate AC" sources addressed. Specifically, where is the basis for the staff accepting a qualified alternate AC source as demonstration of adequate blackout capability in lieu of the requirement for analyses to demonstrate the ability of a plant to cope for the specified duration (as indicated in Discussion Item 2 above). The staff responded that an interpretation of factors (1) and (2) of the four factors mentioned in the new proposed section (e) of GDC-17 (see proposed FRN, at p. 47) might be considered the basis for doing so. The Committee recommended that both the concept of "alternate AC," and the provision for waiving the requirement for analyses to demonstrate coping capability for the specified duration if a fully qualified alternate AC source is provided, should be included explicitly in the proposed rule. (For alternate AC sources, as well as other blackout coping equipment, "fully qualified" does not mean that the equipment must meet safety-grade, Seismic I, Appendix B QA requirements, etc.) The staff agreed to work out with OGC the appropriate wording changes needed to incorporate this recommendation by the Committee.

9. After much discussion of the proposed A-44 resolution with the staff, the Committee agreed that it appeared that, to the extent that the staff's cost estimates are accurate and can be actually realized in practice, the costs involved in implementing the proposed resolution are justified. A major qualitative factor in the Committee's conclusion in this regard was the observation that the AC electrical systems affect so pervasively plant operations. There is a high likelihood, therefore, that significant improvement in AC systems reliability and independence will provide a significant improvement in safety. Assuming the proposed resolution for A-44 is approved and implemented as expected, the Committee inquired about the prospect of any further cost beneficial fixes being imposed under USI A-45. The staff responded that, in view of the small residual risk levels remaining after implementation of the proposed resolution for A-44 and other generic fixes imposed over the last few years, and in further view of the projected cost of the type of fixes that have been most seriously considered in working that issue, it is not clear at this time that the staff can propose further cost beneficial fixes under USI A-45.

10. The Committee felt that the wording of the last paragraph on p. 36-37 of the proposed FRN is too negative with regard to the potential role and contribution of probabilistic analyses in better understanding the blackout issue, and with regard to the use of reliability goals in resolving important generic problems such as USI A-44. They recommended that this wording be modified to reflect a more balanced agency view in this regard prior to issuance of the proposed resolution package. The staff agreed to work with OGC in doing so.

11. The Committee noted that the proposed Station Blackout Reg. Guide provides for 2-hour, 4-hour, 8-hour and 16-hour categories of operating plants; but the FRN refers only to 4-hour and 8-hour plants. They recommended that the wording of the proposed FRN (e.g., at pp. 23, 26 and 28) be revised so that the wording of the proposed resolution package is consistent throughout. The staff agreed to do so.

12. With reference to the wording of the last sentence of the second paragraph on p. 14 of the proposed FRN, the Committee asked if the staff were sure that no exemptions to GDC-17 had been approved by the Commission for any operating plant, as suggested by the wording of that sentence. The staff agreed to check this point specifically, and will adjust the wording of the sentence in question, if necessary, to reflect GDC-17 exemption status accurately.

13. The Committee noted that OGC has not formally concurred in the A-44 package at this point; in accordance with the CRGR charter, this must be done prior to final approval and issuance of the proposed package.

14. In addition to the recommendations made in the preceding regarding a number of general kinds of changes that should be made to the proposed resolution package in various topical areas, the Committee recommended also the following revisions to the wording of specific sections of the documents provided to CRGR for review:

    a. Federal Register Notice, at p. 12, last sentence. Change to read as follows:

    > "However, the equipment must meet the quality assurance criteria needed to establish an appropriate level of reliability and operability during station blackout events."

    (See also Discussion Item 3 for indication of additional nonspecific wording changes needed to this section of the FRN.)

    b. Federal Register Notice, at p. 36, next-to-last and last paragraphs. Change to read as follows:

    > "...However, the Commission recognizes that there may be potential drawbacks from relying on this approach on an industry-wide basis for the reasons given below.
    >
    > One detrimental aspect...is that it might lead to overemphasis on efforts to assess the adequacy of the analysis, rather than concentrating on adequacy of the design. There can be too strong an emphasis on fine tuning the model...to achieve results directed mainly toward meeting a numerical criterion....On balance, in implementing...goals...."

    (See Discussion Item 11 above, for context of this recommendation.)

c. Federal Register Notice, at p. 44, under "(b) Limitation of Scope," Change the ending of the sentence to read as follows:

> ".... if the capability to withstand blackout was specifically addressed in the operating license proceeding and was explicitly approved by the NRC."

(Alternatively, consider simply stating that these new requirements do not apply to the St. Lucie plant, if that is what is really intended by the wording of this section.)

d. Federal Register Notice, at p. 45, last paragraph. Change to read as follows:

> "A final schedule for implementing modifications...shall be developed by the NRC staff in consultation and coordination with the licensee."

(See Discussion Item 5 above for context of this recommendation.)

e. Proposed Station Blackout Reg. Guide, at p. 6 and p. 11 (Footnote).

Revise the wording referring to "plant specific technical guidelines" to clarify that this actually means Emergency Operating Procedures for the plants.

f. Proposed Station Blackout Reg. Guide, at p. 8.

Clarify to indicate that references to "full power" mean that the reactor has operated long enough at 100% power to reach equilibrium Xenon conditions.

RECOMMENDATIONS TO THE EDO

On the basis of their overall review of this matter, including the presentations and discussions at this meeting, the CRGR recommended to the EDO that the proposed resolution for USI A-44 be approved for implementation, subject to a number of revisions recommended by the Committee, in particular the following:

1. The staff should resolve wording issues in several topical areas, and make specific revisions in several sections of the documents in the package, as indicated in the preceding Discussion Items.

2. The staff should develop, and submit for review by CRGR, the acceptance criteria and review/inspection guidance documents (i.e., SRPs and TIs) to be used by NRC reviewers and inspectors prior to implementing any action finally approved for resolving USI A-44, as discussed in the preceding Discussions Items.

3. The staff should address explicitly in the body of the proposed rule the "alternate AC source" concept, and also make clear in the rule itself that a fully qualified alternate AC source will be accepted as demonstration of adequate blackout coping capability in lieu of extensive analyses otherwise required by the proposed rule.

4. The staff should develop in coordination with affected licensees more expeditious schedules for overall implementation of the proposed resolution than are indicated in the current package.

ENCLOSURE H

SEE ATTACHED LIST

Dear Mr. Chairman:

Enclosed for the information of the Subcommittee on ________________________ ________________ are copies of: (1) the Federal Register Notice for the final rule on Station Blackout, (2) Regulatory Guide 1.155 entitled "Station Blackout," (3) NUREG-1032 entitled "Evaluation of Station Blackout Accidents at Nuclear Power Plants, Technical Findings Related to Unresolved Safety Issue A-44," and (4) a supporting Regulatory/Backfit Analysis (NUREC-1109).

These documents serve as the staff's resolution of Task A-44, which has been identified as an "Unresolved Safety Issue" in the 1978 (1980 for A-45 through A-48) Annual Report pursuant to Section 210 of the Energy Reorganization Act of 1974. The final rule incorporates consideration of public comments received in response to the Federal Register Notice of Proposed Rulemaking dated March 21, 1986 (51FR9829).

The proposed rule requires all licensees and applicants:

1) to assess the capability of their plants to cope with a station blackout (that is, determine that the plant could maintain core cooling and containment integrity with AC power unavailable for a minimum specified time period);

2) to have procedures and training to cope with such an event; and

3) to make modifications, if necessary, to cope with an acceptable minimum duration station blackout.

The implementation schedule stated in 10CFR50.63 will become effective immediately upon promulgation of the rule.

Sincerely,

Eric S. Beckjord, Director
Office of Nuclear Regulatory Research

Enclosure I - Public Notice

NRC ADOPTS FINAL RULE TO REQUIRE NUCLEAR POWER PLANTS TO BE CAPABLE OF WITHSTANDING "STATION BLACKOUT"

The Nuclear Regulatory Commission has amended its regulations to require that commercial nuclear power plants be capable of withstanding a total loss of alternating current (AC) electric power--called "station blackout"--for a specified time and to maintain reactor cooling during that period.

Previous regulations established requirements for the design and testing of onsite and offsite power systems intended to minimize the probability of losing all AC power. However, the previous regulations did not require explicitly that nuclear power plants be designed to assure that the reactor core can be cooled for any specified period of loss of all AC power.

Station blackout has been studied as an unresolved safety issue since 1980 and results show that loss of offsite and onsite AC power systems can be an important contributor to the overall plant risk. These systems provide power for various safety systems, including reactor decay heat removal and containment heat removal. If a station blackout persists for a sufficient time so that the capability of the AC-independent systems (for example, batteries and steam-driven auxiliary feedwater systems) to remove heat decay is exceeded, core melt and containment failure could result.

The final rule requires all licensees and applicants:

1) to assess the capability of their plants to cope with a station blackout (that is, determine that the plant could maintain core cooling and containment integrity with AC power unavailable for a minimum specified time period);

2) to have procedures and training to cope with such an event; and

3) to make modifications, if necessary, to cope with an acceptable minimum duration station blackout.

Written comment on the proposed rule were received in response to a Notice of Proposed Rulemaking published in the Federal Register on March 21, 1986 (51FR9829). The comments have been taken into account in development of the final rule as described in the Federal Register Notice of Final Rulemaking, , 1988.

NORTHEAST UTILITIES

NU

THE CONNECTICUT LIGHT AND POWER COMPANY
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November 23, 1987

Mr. Themis P. Speis
Deputy Director for Generic and
Regulatory Issues
Office of Nuclear Regulatory Research
U.S. NUCLEAR REGULATORY COMMISSION
Washington, D.C. 20555

Re: NUMARC-8700, "Guidelines and Technical Bases for NUMARC Initiatives Addressing Station Blackout at Light Water Reactors."

Dear Mr. Speis:

Enclosed with this letter are 10 copies of NUMARC-8700, "Guidelines and Technical Bases for NUMARC Initiatives Addressing Station Blackout at Light Water Reactors," dated November 20, 1987. This document replaces the October 19, 1987 version previously provided to you. NUMARC-8700 provides guidance and methodologies that utilities will use to implement the Nuclear Management and Resources Council (NUMARC) station blackout initiatives. Discussions with the NRC Staff indicate that this document essentially addresses the NRC's concerns with a station blackout event and provides a reduction in risk at least comparable to that associated with the proposed rule. It has been further indicated that this document will be viewed by the Staff as providing an acceptable means for meeting the requirements of the proposed rule.

Consistent with past practice, we expect this transmittal will be placed in the public document room.

Very truly yours,

John Opeka

John F. Opeka, Chairman
NUMARC SBO Working Group

cc: Aleck Serkiz, Senior Task Manager
Reactor and Plant Safety Issues Branch

J. F. Colvin, Executive Vice President
NUMARC

ENCLOSURE J
USI A-44 EDO PKG
11-20-87

NUMARC-8700

GUIDELINES AND TECHNICAL BASES

FOR NUMARC INITIATIVES ADDRESSING

STATION BLACKOUT AT LIGHT WATER REACTORS

NOVEMBER 20, 1987

NUCLEAR MANAGEMENT
AND RESOURCES COUNCIL

NUMARC-8700

GUIDELINES AND TECHNICAL BASES

FOR NUMARC INITIATIVES ADDRESSING

STATION BLACKOUT AT LIGHT WATER REACTORS

NOVEMBER 20, 1987

NUCLEAR MANAGEMENT
AND RESOURCES COUNCIL

CONTENTS

1. INTRODUCTION

1.1 GUIDANCE AND DOCUMENT STRUCTURE

The objective of this document is to provide guidance and methodologies for implementing the Nuclear Management and Resources Council (NUMARC) station blackout initiatives. Section 1 provides an introduction and discussion of the initiatives.

Section 2 provides a set of baseline assumptions concerning the course and nature of a station blackout. Each assumption is accompanied by a basis discussion. These assumptions define the major topics concerning station blackout which the initiatives are intended to address.

Section 3 provides guidance for determining the required coping duration category consistent with the NRC Staff's draft Regulatory Guide 1.155.

Section 4 provides guidelines for assuring plant specific procedures adequately address station blackout response.

Section 5 describes industry's attention to reduce cold starts of diesel generators during testing of emergency standby diesel generators.

Section 6 describes industry's EDG unavailability monitoring program.

Section 7 provides a simplified methodology for reviewing basic plant coping features.

The appendices provide additional information concerning various topics:

Appendix A provides definitions.
Appendix B provides Alternate AC power criteria.
Appendix C provides sample AAC configurations.
Appendix D discusses an EDG performance program.
Appendix E analyzes the effects of loss of ventilation.
Appendix F describes methods for assuring equipment operability under station blackout conditions.
Appendix G provides references.

1.2 NUMARC INITIATIVES

Late in 1985, NUMARC established a working group on station blackout to address USI A-44. The Nuclear Utility Group on Station Blackout (NUGSBO) has provided the major portion of the technical support for the NUMARC station blackout working group. NUMARC determined that many of the concerns related to station blackout could be alleviated through industry initiatives to reduce overall station blackout risk.

In light of these considerations, on June 10, 1986, the NUMARC executives endorsed four industry initiatives to address the more important contributors to station blackout risk. These initiatives were described to the Commission by letter dated June 23, 1986 which also forwarded comments concerning the proposed station blackout rule. On October 22, 1987, NUMARC approved one additional initiative and a modification to one of the original initiatives. The initiatives are:

(1) Initiative 1A — RISK REDUCTION

Each utility will review their site(s) against the criteria specified in NRC's revised draft Station Blackout Regulatory Guide, and if the site(s) fall into the category of an eight-hour or sixteen-hour site after utilizing all power sources available, the utility will take actions to reduce the site(s) contribution to the overall risk of station blackout. Non-hardware changes will be made within one year. Hardware changes will be made within a reasonable time thereafter.

This initiative was changed by the October 22, 1987 NUMARC vote to reflect changes in NRC's criteria from those in NUREG-1109 which were incorporated in the original Initiative 1.

(2) Initiative 2 — PROCEDURES

Each utility will implement procedures at each of its site(s) for:

(a) coping with a station blackout;

(b) restoration of AC power following a station blackout event; and,

(c) preparing the plant for severe weather conditions (e.g., hurricanes) to reduce the likelihood and consequences of a loss of off-site power and to reduce the overall risk of a station blackout event.

(3) Initiative 3 --- COLD STARTS

Each utility will, if applicable, reduce or eliminate cold fast-starts of emergency diesel generators through changes to technical specifications or other appropriate means.

(4) Initiative 4 --- AC POWER AVAILABILITY

Each utility will monitor emergency AC power unavailability, utilizing data provided to INPO on a regular basis.

(5) Initiative 5 --- COPING ASSESSMENT

Each utility will assess the ability of its plant(s) to cope with a station blackout. Plants utilizing alternate AC power for station blackout response which can be shown by test to be available to power the shutdown busses within 10 minutes of the onset of station blackout do not need to perform any coping assessment. Remaining alternate AC plants will assess their ability to cope for one-hour. Plants not utilizing an alternate AC source will assess their ability to cope for four-hours. Factors identified which prevent demonstrating the capability to cope for the appropriate duration will be addressed through hardware and/or procedural changes so that successful demonstration is possible.

1.3 SUPPORTING INFORMATION

Utilities are expected to ensure that the baseline assumptions are applicable to their plants. Further, utilities are expected to ensure that analyses and related information are available for review.

2. GENERAL CRITERIA AND BASELINE ASSUMPTIONS

This section contains general criteria and a listing of the base line assumptions, a brief description of their bases, and appropriate references to source material. The topics in this section are:

Section 2.1 —	general criteria
Section 2.2 —	initial plant conditions
Section 2.3 —	the initiating event
Section 2.4 —	station blackout transient
Section 2.5 —	reactor coolant pump inventory loss
Section 2.6 —	operator action
Section 2.7 —	effects of the loss of ventilation
Section 2.8 —	system cross-tie capability
Section 2.9 —	instrumentation and controls
Section 2.10 —	containment isolation valves
Section 2.11 —	hurricane preparations.

2.1 GENERAL CRITERIA

Procedures and equipment in light water reactors relied upon in a station blackout should ensure that satisfactory performance of necessary decay heat removal systems is maintained for the required station blackout coping duration. For a PWR, an additional requirement is to keep the core covered. For a BWR, no more than a momentary core uncovery is allowed. For both BWRs and PWRs, appropriate containment integrity should also be provided in a station blackout to the extent that isolation valves perform their intended function without AC power.

2.2 INITIAL PLANT CONDITIONS

2.2.1 Assumptions

(1) The station blackout event occurs while the reactor is operating at 100% rated thermal power and has been at this power level for at least 100 days.

(2) Immediately prior to the postulated station blackout event, the reactor and supporting systems are within normal operating ranges for pressure, temperature, and water level. All plant equipment is either normally operating or available from the standby state.

2.2.2 Basis

(1) The potential for core damage from a station blackout is bounded by events initiated from 100% power due to the presence of substantial decay heat.

(2) Transients initiated from normal operating conditions are considered most probable.

2.3 INITIATING EVENT

2.3.1 Assumptions

(1) The initiating event is assumed to be a loss of off-site power (LOOP) at a plant site resulting from a switchyard-related event due to random faults, or an external event, such as a grid disturbance, or a weather event that affects the off-site power system either throughout the grid or at the plant.

LOOPs caused by fire, flood, or seismic activity are not expected to occur with sufficient frequency to require explicit criteria and are not considered.

(2) The LOOP is assumed to affect all units at a plant site. At a multi-unit site with normally dedicated emergency AC power sources, station blackout is assumed to occur at only one unit. At multi-unit sites with normally shared emergency AC power sources, where the combination of AC sources exceeds the minimum redudancy requirements for normal safe shutdown (non-DBA) of all units, the remaining emergency AC power sources may be used as alternative AC power sources provided they meet the alternate AC power criteria in Appendix B. If there are no remaining emergency AC power sources in excess of the minimum redundancy requirements, station blackout must be assumed to occur at all the units.

(3) Emergency AC (EAC) power sources are assumed to be available as Alternate AC power sources to cope with the station blackout under the following conditions:

(a) For the blacked-out unit, any emergency AC power source(s) in excess of the number necessary to meet minimum redundancy requirements (i.e. single failure) for safe

shutdown is assumed to be available and may be designated as an Alternate AC (AAC) power source(s) provided it meets the AAC criteria provided in Appendix B.

(b) For multi-unit sites, EAC sources available from a non-blacked-out unit, after assuming a single failure at the non-blacked-out unit, may be designated as Alternate AC, if they meet the AAC criteria provided in Appendix B and are capable of meeting the necessary shutdown loads of both units.

(4) No design basis accidents or other events are assumed to occur immediately prior to or during the station blackout.

2.3.2 Basis

(1) NRC analysis separates LOOP events into three categories: plant-centered, grid disturbance, and severe weather. Plant-centered events involve hardware failures, design deficiencies, human errors in maintenance and switching, and localized weather-induced faults, such as due to lightning, salt spray, and ice. These plant-centered events reportedly occur at a frequency of 0.056 events per site-year, with a median duration of 0.3 hour. Grid disturbance events have been shown to be of much lesser concern for most plants. Events in this category reportedly have a frequency of 0.020 events per site-year, with a median duration of 0.7 hour. Severe weather events have a lesser experience with 0.011 events and a median duration of 2.6 hours. (Section 3, including Table 3.1, NUREG-1032)

Seismic, fire, and flooding events include accident scenarios for which current licensing requirements specify protective measures. For example, the potential for a fire-induced station blackout is extremely remote due to the effectiveness of current fire protection programs and 10 CFR 50 Appendix R separation requirements imposed on shutdown systems. In fact, some plants installed an alternate or dedicated shutdown capability in response to Appendix R which may also be used to respond to a station blackout event. NRC analysis concludes that fire-induced station blackout is not a generic concern, citing a station blackout frequency of less than $1x10^{-6}$ per reactor-year for most plants. Consequently, station blackout events that may occur at a particular site involving fire initiators are not likely to occur, and are not addressed in this document.

The seismic and flooding issues are similar to the fire risk concern regarding the potential for causing station blackout. The Class 1E power system is currently designed to withstand seismic events. Similarly, flooding protection is addressed in the plant's licensing basis. As a result, the potential for seismically-induced or flooding-induced station blackout is on the same order as fire-induced events, and are not addressed in this document.

For these reasons, seismic, flooding, and fire-induced station blackout events are not addressed in these guidelines.

(Appendix J, NUREG/CR-3226)

(2)-(3) *The major contributor to overall station blackout risk is the likelihood of losing off-site power and the duration of power unavailability. A LOOP may occur as a result of a switchyard problem either affecting a single unit, or possibly multiple units at a site. Alternatively, the cause of the LOOP may be a grid or area-wide disturbance associated with severe weather conditions. Although these events are a much smaller fraction of the total number of events (in fact, weather-related events represent on the order of 10% of all LOOPs experienced to date), they can be significant because of the longer time to restore off-site power following such events. To be conservative, the LOOP is assumed to affect all units at a site.*

The next most important contributor to station blackout risk for a given plant is low EDG availability. EDG availability varies among operating sites, based on the number of EDGs on-site, the reliability to start from a standby state, the overall availability of the machine, and the potential for dependent failures. Industry EDG reliability to start from a standby state is typically in the range of 0.98-0.99. It is very unlikely to have average EDG reliability for all machines at a site below 0.95 over a sustained period. Consequently, the contribution of EDG reliability to station blackout risk is well below that of LOOP for most plants.

EDG failures may also occur due to dependent causes (i.e., common cause events). These failures may result from design or operating deficiencies that manifest themselves in a concurrent failure. The potential for these deficiencies affecting all EDGs for multiple unit sites is considered remote since most reactors have staggered operating cycles. Staggered operating cycles also make it less likely that major maintenance activities are scheduled at the same time. Similarly, redundant units are often designed and constructed on independent schedules, with initial commercial operation dates separated by up to several years in time.

Generally high EDG reliability and low dependent failure rates provide a basis for screening EDG configurations. In support of this perspective, NUGSBO analyzed the likelihood of failure on demand for standby systems, such as for typical emergency AC power systems. The potential for simultaneously failing two identical EDGs with each machine at industry average reliability (i.e., approximately 2% average failure rate on demand for each machine) and nominal susceptibility to dependent failure (i.e., 2%) is approximately 7.8×10^{-4}*. The likelihood of three identical EDGs simultaneously failing is even lower, at about* 4.1×10^{-4} *for machines with 0.98 reliability.*

These results suggest that the potential for more than two EDGs failing at a unit is very low. Consequently, assuming failure of EDGs in excess of those required for minimum redundancy is not necessary to assure that the risk of a station blackout is sufficiently low. For multi-unit sites (assuming an EDG single failure at the non-

blacked-out unit), the marginal probability of an additional EDG failure at the non-blacked-out unit is so low that the remaining EDGs are assumed available if they meet the applicable AAC criteria. One-out-of-two shared (1/2S) and two-out-of-three shared (2/3S) configurations do not meet Alternate AC power criteria. At single unit sites with EDGs in excess of the number necessary to meet the minimum redundancy requirements (such as units with 3 or more diesels), these additional EDGs are candidates for Alternate AC. At multi-unit sites, where the combination of emergency AC sources exceeds the minimum redundancy requirements for normal safe shutdown (non-DBA) for all units, the remaining emergency AC power sources may be used as alternate AC power sources provided they meet the AAC power criteria of Appendix B.

The availability of EDGs as an Alternate AC source may be assumed if the machine satisfies the Alternate AC power source criteria provided in Appendix B. This includes criteria designed (1) to minimize the potential for dependent failure events adversely affecting the Alternate AC power source in station blackout scenarios, and (2) to provide requirements for power source availability.

The Staff's stated objective of the proposed station blackout rule is to reduce the core damage frequency due to station blackout to approximately 10^{-5} per year for the average site. As provided in the proposed rule, this objective could be obtained by extending the current nominal two-hour coping capability to four hours. Comparable safety benefits may exist from the utilization of an AAC power source. To investigate these benefits, NUGSBO extended the emergency AC power system model to include the contribution of off-site power system failure frequency and power restoration. A composite LOOP duration distribution was constructed based on the LOOP events reported in NUREG-1032. Assuming a LOOP frequency of 0.1 per year, industry average power restoration distributions, a 1/3 EDG configuration, and failure likelihoods of 2% for each machine and 2% dependent failure, a two-hour coping capability yields a station blackout core damage frequency of well below the 10^{-5} per year. This frequency is below the threshold sought by the Staff in the station blackout rulemaking. (Section 4, NUREG-1032; see also NUREG-1109, page 9 wherein the Staff assumes "... that all plants, as currently designed, can cope with a station blackout for 2 hours, and, with proper procedures and training, plants could cope with a 4-hour station blackout without having to make major modifications.")

(4) *The likelihood of a design basis accident or other event coincident with a station blackout is considered remote and is not addressed in this document.*

2.4 STATION BLACKOUT TRANSIENT

2.4.1 Assumptions

(1) Following the loss of all off-site power, the reactor automatically trips with sufficient shutdown margin to maintain subcriticality at safe shutdown (i.e. hot standby or hot shutdown as appropriate). The event ends when AC power is restored to shutdown busses from any source, including Alternate AC.

(2) The main steam system valves (such as main steam isolation valves, turbine stops, atmospheric dumps, etc.) necessary to maintain decay heat removal functions operate properly.

(3) Safety/Relief Valves (S/RVs) or Power Operated Relief Valves (PORVs) operate properly. Normal valve reseating is also assumed.

(4) No independent failures, other than those causing the station blackout event, are assumed to occur in the course of the transient. The potential for mechanistic failures resulting from the loss of HVAC in a station blackout event is addressed in Section 7 of this document.

(5) AC power is assumed available to necessary shutdown equipment within four hours from either the off-site or blacked-out unit's Class 1E sources or is available within one hour from an Alternate AC source.

2.4.2 Basis

(1)-(3) These assumptions outline some of the more important features of the station blackout transient. The basic considerations are a normal LOOP transient, proper unit trip with full reactivity insertion, and MSIV closure as appropriate for the design of the plant. In addition, the likelihood of PORV or S/RV malfunction in a station blackout is on the order of 1-2% (See Section 2, NUREG/CR-1988; Section 2 and 6, NUREG/CR-2182; and NUREG-1032)

(4) Imposing additional independent failures on the station blackout response capability has diminishing safety significance for most power plants. This is because the dominant accident contributors to a station blackout event generally involve off-site power system reliability, the reliability and level of redundancy of the emergency AC power system, and the station blackout coping capability, in that order. Since a number of failures must occur to result in a station blackout event, additional independent failures are of secondary importance. The station blackout response capability also depends on systems that are highly reliable due to the design and maintenance standards used. Consequently, the potential for random failure in these systems is low. Finally, the safety effects of response

capability loss are of most significance only if they are experienced early in the station blackout transient (i.e., primarily in the first 30 minutes). This potential has been addressed in NRC Staff analysis which estimates the probability of decay heat removal system failure early in a station blackout event to range from 0.001 for High Pressure Core Spray (HPCS)/RCIC combinations to 0.04 for a single steam turbine-driven train auxiliary feedwater system (AFW). These results underscore the lower significance of additional non-mechanistic failures in the station blackout scenario. (Appendix C, particularly Table C.2, NUREG-1032)

(5) *Historically, the vast majority of LOOP events are of short duration. NRC Staff analysis reports the median AC power restoration time for all LOOP events to be about 1/2 hour, with off-site power restored in approximately 3 hours for 90% of all events. Consequently, assuming a four hour restoration time addresses the bulk of postulated station blackout events. For AAC systems, one hour is considered an acceptable period of time to lineup the AAC power source and restore power to a shutdown bus. (Off-site power restoration times are taken from Supplementary Information, Proposed Station Blackout Rule, 51 FR 55, at 9830)*

2.5 REACTOR COOLANT INVENTORY LOSS

2.5.1 Assumptions

Sources of expected PWR and BWR reactor coolant inventory loss include (1) normal system leakage, (2) losses from letdown, and (3) losses due to reactor coolant pump seal leakage. Expected rates of reactor coolant inventory loss under station blackout conditions do not result in core uncovering for a PWR in the four hour time period. Therefore, makeup systems in addition to those currently available under blackout conditions are not required. There exists sufficient head to maintain core cooling under natural circulation.

2.5.2 Basis

Normal system leakage is limited by technical specifications to a low rate. These rates are not assumed to increase under station blackout conditions. Emergency operating procedures developed in accordance with NSSS vendor Emergency Procedure Guidelines or individual plant analysis should be used to direct operators to take appropriate action. RCP seal leakage is assumed not to exceed 25 gpm per pump for the duration of the station blackout event. However, this assumption is currently the subject of a resolution program (NRC Generic Issue 23).

If the final resolution of Generic Issue 23 results in higher RCP leakage rates, then the coping duration analysis will need to be reevaluated.

Generic NSSS vendor analyses and studies listed below show that for the assumed leakage rates core uncovery does not occur in the four hour time period. These studies also show that sufficient head exists to maintain core cooling under natural circulation for a PWR, and that decay heat removal capability is maintained for a BWR.

(1) *Analyses submitted in response to the TMI accident and emergency procedure guidelines, including IEB 79-05, NUREG-0578, NUREG-0660, and NUREG-0737;*

(2) *Analyses submitted in response to NRC Generic Letter 81-04 concerning station blackout response procedures;*

(3) *C. D. Fletcher, "A Revised Summary of PWR Loss of Offsite Power Calculations", EGG-CAAD-5553, EG&G Idaho, September 1981;*

(4) *D. H. Cook, et. al., "Station Blackout at Browns Ferry Unit One - Accident Sequence Analysis", NUREG/CR-2182, Oak Ridge National Laboratory, November 1981; and*

(5) *A. M. Kolaczkowski and A. C. Payne, Jr., "Station Blackout Accident Analyses", NUREG/CR-3226, Sandia National Laboratories, May 1983.*

2.6 OPERATOR ACTION

2.6.1 Assumptions

Operator action is assumed to follow the Plant Operating Procedures for the underlying symptoms or identified event scenario associated with a station blackout.

2.6.2 Basis

NRC analyses supporting the proposed station blackout rulemaking assume that a reasonable set of operator actions will occur. The governing document for defining operator actions is the plant's procedures. (Appendix H, NUREG/CR-3226)

2.7 EFFECTS OF LOSS OF VENTILATION

2.7.1 Assumptions

(1) Equipment Operability Inside Containment

Temperatures resulting from the loss of ventilation are enveloped by the loss of coolant accident (LOCA) and high energy line break environmental profiles.

(2) Equipment Operability Outside Containment

(a) Areas containing equipment required to cope with a station blackout need only be evaluated if (a) the area is a dominant area of concern, and (b) the dominant area of concern has not been previously evaluated as a harsh environment due to a high or moderate energy line break. The dominant areas of concern are:

(i)	HPCI/HPCS and RCIC rooms (BWR only)	—	decay heat removal equipment
(ii)	Steam driven AFW pump room (PWR only)	—	decay heat removal equipment
(iii)	Main steam tunnel (BWR only)	—	high temperature cutout for decay heat removal equipment.

Assumptions concerning the potential for thermal-induced equipment failure in a station blackout for the dominant areas of concern are separated into three distinct conditions based on bulk air temperatures:

Condition 1

Equipment located in Condition 1 rooms are considered to be of low concern with respect to elevated temperature effects and will likely require no special actions to assure operability for a 4-hour station blackout. This condition is defined by a steady state temperature of 120° F.

Condition 2

Equipment located in Condition 2 rooms generally require no forced cooling in order to assure operability for a 4-hour station blackout. If additional cooling is needed, such actions as opening doors may be sufficient to support equipment operation to mitigate a station blackout event. This condition is defined by a steady-state temperature of 150° F.

Condition 3

Equipment located in Condition 3 rooms require plant-specific treatment of the potential for thermal-induced failure. Such treatment may include (1) further plant-specific analysis, (2) providing forced cooling, and (3) replacement by equipment designed or qualified for the environment.

NOTE: Plant procedures need to reflect the operator actions necessary to enhance cooling for rooms in above conditions.

The control room complex (i.e., area(s) containing instrument indications and associated logic cabinets which the control room operator relies upon to cope with a station blackout) is considered to be in Condition 1. By opening cabinet doors, adequate air mixing is achieved to maintain internal cabinet temperatures in equilibrium with the

control room temperature. Therefore, cabinets containing instrumentation and controls required for achieving and maintaining safe shutdown in a station blackout are considered to be in Condition 1. As such, additional cooling may be provided in a station blackout by opening cabinet doors within 30 minutes of the event's onset.

For multi-unit control room complexes (i.e., area(s) containing instrument indications and associated logic cabinets which the control room operator relies upon to cope with a station blackout) where a portion of the HVAC is powered from the non-blacked-out unit, no significant temperature rise above normal operating conditions is expected. For this situation, the effects of loss of ventilation need not be considered further.

(b) Loss of heating in the battery room does not result in a decrease in battery electrolyte temperature sufficient to warrant battery capacity concern for a four-hour period.

(3) <u>Control Room Habitability</u>
Loss of cooling in the control room for a four hour period does not prevent the operators from performing necessary actions.

2.7.2 Basis

(1) <u>Equipment Operability Inside Containment</u>
No design basis accidents (DBAs) (i.e., LOCAs or steam line breaks) or beyond DBAs (i.e., resulting in core damage) are assumed coincident with a station blackout. Therefore, environmental concerns inside containment are limited to (1) loss of cooling water, and (2) loss of ventilation systems. In both cases, no sudden onset of extreme temperature conditions or humidity is expected. Station blackout results in a slow heatup of containment due to loss of ventilation. Absent DBA conditions, temperatures in a four-hour station blackout are expected to be bounded by thermal profiles considered for the high energy line break events.

The response of a large, dry containment to a station blackout was previously analyzed in the course of preparing Emergency Procedure Guidelines (see Westinghouse ECA-0.0). For two, three, and four loop plants, assuming 50 gpm per pump RCP seal leakage, containment temperature rises less than 15° F from the initial temperature.

Other PWR containments can be expected to perform within an acceptable thermal range, based on the relative volume of other containments to the large dry containment. For example, ice condensers offer a somewhat smaller amount of free volume, combined with several million pounds-mass of ice. Even ignoring the cooling capacity of the ice baskets, containment heating is not expected to result in excess temperatures substantially greater than 50-60° F above normal operating conditions. These temperature increases are well below the thermal profiles

established for ice condenser containments.

For BWRs, analyses indicate that conditions inside containment under station blackout conditions will be within typical thermal limits established for equipment qualification for pressure suppression containments (e.g., see letter from Mr. N. W. Curtis (Pennsylvania Power and Light Company) to Mr. A. Schwencer (NRC), dated June 15, 1982).

(2) Equipment Operability Outside Containment

(a) *As with inside containment, the temperature rise in a station blackout outside containment over a four-hour period is not expected to exceed conditions associated with a high or moderate energy line break. With reactor shutdown and station blackout initiation, a significant amount of equipment is de-energized with a resultant reduction in heat load. Process piping and other high temperature surfaces do not efficiently transfer heat to air, particularly when forced ventilation is not present. Consequently, the potential for significant heatup is negligible in a four-hour period.*

Under station blackout conditions, the effects of the loss of ventilation are less severe due to the associated loss of lighting and AC powered equipment heat loads. The potential for mechanistic failures of systems and components due to loss of ventilation is dependent on the time required for temperatures to rise in closed rooms and cabinets. Temperature buildup in a compartment is a slow process due to the normally large thermal lag associated with natural convection and the loss of AC-supplied heat sources. This large thermal lag allows sufficient time for operator actions to supplement cooling in order to limit the thermal buildup. NUGSBO has analyzed the potential for temperature buildup in closed rooms over a four hour period. The results show that opening doors early in a station blackout (i.e., within approximately 30 minutes) significantly limits any temperature rise due to loss of forced ventilation.

Occasionally, supplemental cooling measures (such as opening doors to increase natural circulation and ventilation) may conflict with other safety or administrative considerations. For example, procedural requirements may exist for keeping fire or flooding doors closed. Despite these procedural considerations, opening doors would be acceptable during a station blackout to increase natural circulation for necessary shutdown instrumentation. Other techniques, such as using permanently mounted small battery-operated fans inside cabinets, could also be considered (Section 5 and Appendix I, NUREG/CR-3226).

Condition 1

Condition 1 rooms are assumed to have a relatively small potential for thermally-induced failure

during a four hour station blackout. This assumption is based on operating experiences and studies concerning the operability of various classes of equipment exposed to elevated temperatures.

In a station blackout, forced cooling will be lost to most plant areas and the potential exists in Condition 1 areas for bulk air temperatures to rise up to 120° F. For most mechanical and electrical equipment and instrumentation found in Condition 1 dominant areas, temperature rises up to 120° F would likely not adversely affect operability.

<u>Condition 2</u>

Condition 2 rooms are likely to include a relatively substantial heat generation source and a small room geometry. These conditions are more typical of rooms containing steam-driven makeup pumps, such as RCICS and AFWS which are generally qualified or designed to operate in elevated temperatures.

The NRC has considered equipment operability during station blackout conditions (see Jacobus [1987]). One of the conclusions of this review is that certain classes of components (e.g., relays and switches) will likely remain operable in thermal environments of 150° F to 300° F for up to eight hours. While the Jacobus study was not extensive, the general assumption of equipment operability for Condition 2 thermal environments is considered valid because (1) only a four hour station blackout event is considered, and (2) in practice, less than the full four hours would be involved since there would be a period of thermal buildup during the front-end of the station blackout transient.

<u>Condition 3</u>

Condition 3 rooms represent classes of thermal environments where plant-specific consideration may be appropriate.

Appendix F provides a method for assessing the operability of equipment exposed to Condition 1, 2, and 3 environments.

The operability of a representative set of control room complex (i.e., area(s) containing instrument indications and associated logic cabinets which the control room operator relies upon to cope with a station blackout) cabinet equipment was established with actual experience involving loss of control room ventilation for several hours (see Chiramal [1986]). During this extended loss of ventilation event at McGuire, there was negligible operability effects

on equipment or instrumentation.

(b) *Battery capacity is reduced if the electrolyte temperature drops significantly below design temperatures. Class 1E batteries are housed in seismic Category 1 structures, and are not typically subjected to the direct effects of the external environment. Therefore, the temperature decrease in the battery room is not significant over a four-hour period. Also, the mass of battery electrolyte is sufficient to resist significant temperature drops over a four-hour period due to lower battery room temperatures since battery cell materials are not efficient thermal conductors. Therefore, a decrease in battery capacity due to temperature decreases in electrolyte under station blackout conditions does not warrant further consideration.*

(3) Control Room Habitability

Control room habitability is not an important contributor to station blackout risk, particularly for events of 4-hour durations. NUREG-1032 points out that the dominant accident sequences involve either an early core cooling failure or a subsequent loss of core cooling (see Appendix C, NUREG-1032 for a more complete discussion of station blackout accident sequences). Both sequences are dominated by the failure of automatic equipment to properly function on demand. Even these events have failure probabilities of less than 1% per event, reflecting the exceptionally high reliability of these systems and components. With respect to human error, such as due to habitability concerns, NUREG-1032 states: "The potential effect of operator error causing loss of decay heat removal has not been found to be a large contributor to core damage frequency, if adequate training and procedures exist." (draft NUREG-1032, page C-15). Since NUMARC Initiative 2, as provided in Section 4 guidelines, assures adequate training and procedures will exist, the concern regarding operators' ability to perform cognitive tasks is insignificant.

As to the expected environment within the control room, it has been shown that temperatures are not likely to exceed 110° F should a station blackout event actually occur (Chiramal [1986]). In the McGuire event discussed by Chiramal, habitability was never an issue. Studies suggest that long term occupancy in higher temperature environments does not prevent performance of tasks of various difficulties (see Eichna [1945] and Humphries and Imalis [1946] for military applications). Such studies have been the basis of guidance in the heating and ventilating industry handbooks (e.g., ASHVE [1950] and ASHRAE [1985]). ASHRAE, in particular, correlates temperature, humidity, and pressure and concludes that light work at 110° F and relative humidities up to 50% would not be intolerable.

Before the station blackout event, it is assumed that the control room is at 78° F and about 35% relative humidity. Although temperature increases may be expected due to loss of HVAC, the relative humidity actually decreases to approximately 30%. Guidance provided for military applications may establish a technical basis for defining

habitability standards for power plants in a station blackout. The operative standard, MIL-STD-1472C, concludes that a dry bulb temperature of 110°F is tolerable for light work for a four hour period while dressed in conventional clothing assuming the relative humidity is approximately 30%. Loss of HVAC would impose a slow heatup on the control room. It is expected that steady-state control room air temperatures will be well below 110° F for most plants under loss of HVAC conditions. For the conservative case when it is assumed that a control room is initially at 78° F and experiences an exponential temperature rise to a steady-state 110° F should HVAC be lost in a station blackout, the bulk air temperature at the end of the first hour would be approximately 97° F. At the end of the second hour, the air temperature would be approximately 104° F. At the end of the third hour, it would be approximately 108° F. Since it would take some time for a control room to heatup once HVAC is lost, the operator is not exposed to the thermal limit for the duration of the event. Therefore, it is not expected that operator actions would be impacted significantly by projected temperature and humidity conditions and, further, that a dry bulb temperature of 110° F appears to be a conservative limit for control room habitability.

2.8 SYSTEM CROSS-TIE CAPABILITY

2.8.1 Assumptions

Under station blackout conditions it is assumed that multiunit sites with fluid or DC electrical system cross-tie capability will be able to achieve and maintain safe shutdown in the affected unit by procedurally utilizing the unaffected unit's cross-tied systems. Systems of the unaffected unit must be electrically independent of the blacked-out unit as appropriate in order to credit their availability to bring the affected unit to safe shutdown.

2.8.2 Basis

NRC analyses supporting other rulemakings (i.e., 10 CFR 50 Appendix R) permit multiunit sites to rely on cross-tie capability of fluid systems to bring the affected unit to safe shutdown conditions.

2.9 INSTRUMENTATION AND CONTROLS

2.9.1 Assumptions

Actions specified in Emergency Procedure Guidelines for station blackout are predicated on use of instrumentation and controls powered by vital buses supplied by station batteries. Appropriate actions will be taken by operations personnel to assess plant status in the event of erratic performance or failure of shutdown instrumentation.

2.9.2 Basis

NSSS emergency procedure guidelines identify instrumentation and controls requirements to achieve and maintain safe shutdown. Operator training includes the use of backup instrumentation and methods for identifying erratic performance.

2.10 CONTAINMENT ISOLATION VALVES

2.10.1 Assumptions

Containment isolation valves either fail in the safe condition in accordance with the design bases of the plant or can be manually closed.

2.10.2 Bases

10 CFR 50 General Design Criteria (GDC) 55 through 57 specify requirements for isolating piping systems penetrating containment, including reactor coolant pressure boundaries. These requirements call for combinations of redundant locked closed and automatic isolation valves for reactor coolant pressure boundaries and any containment penetration line directly connected to the containment atmosphere. In cases where automatic isolation valves are used, the GDC specifies that the valves fail upon loss of power in a position which provides greater safety. All other containment penetration valves must meet the requirements of the GDC by being automatic, or locked closed, or capable of remote manual operation.

Most containment isolation valves are in the normally closed or failed closed position during power operation. These valves can also be closed manually. Loss of AC does not affect the design bases for these valves. Some valves, such as MSIVs, charging and letdown lines, and reactor water cleanup lines, are normally open. Typically, these valves are air-operated, failed closed valves and do not need AC power to close. A few DC operated containment isolation valves exist, such as valves in the shutdown cooling or residual heat removal systems. These DC operated valves are normally closed during power operations, and generally are locked or have DC breaker control power removed by racking out the circuit breaker for the valve operator. The position of these DC operated valves is not affected by the station blackout.

2.11 HURRICANE PREPARATIONS

2.11.1 Assumptions

Procedural actions taken in anticipation of the effects of a hurricane provide significant safety benefits and reduce the risk of a station blackout. Plants which are impacted in their "extremely severe weather" grouping primarily due to the

effects of a hurricane have a basis for classifying their "off-site power design characteristic group" (P2*, or P3*) in a lower group.

2.11.2 Bases

NUMARC Guidelines in Section 4.2.3 specify actions to be taken to prepare a plant to cope with a station blackout due to an anticipated hurricane-induced LOOP. These actions can be separated into two groups: (1) actions taken in the 24-hour period prior to anticipated hurricane arrival, and (2) a commitment to be in safe shutdown two hours before the anticipated hurricane arrival at the site. These actions result in a coping categorization consistent with Section 3.2.1, Part 1E(B) and Section 3.2.5 of these guidelines.

The following actions are important for achieving an enhanced coping capability under hurricane conditions:

(1) Plant in safe shutdown at least two hours before the anticipated hurricane arrival at the site (i.e., sustained winds in excess of 73 m.p.h.) so that major decay heat loads can be dissipated using non-emergency plant equipment prior to the occurrence of a LOOP;

(2) Enhancement and verification of EDG reliability by prewarming, prelubricating, starting and load-testing (see, Section 4.2.3);

(3) Topping off condensate storage tank inventory and placing battery systems on charge; and,

(4) Expediting the restoration of important plant systems and components needed to cope with a hurricane-induced LOOP;

and other actions as detailed in Sections 4.2.3 and 4.3.3. Such actions have the capability to enhance the coping capacity for the reasons discussed below.

The timing of anticipatory actions is tied to hurricane tracking performed by both utilities and the National Weather Service. Hurricane tracking normally begins when tropical depressions are first detected far out in the Atlantic Ocean. Forward motion does not normally accelerate until the hurricane approaches the Eastern seaboard or Gulf coast. Even at landfall, hurricane forward speeds are generally below 35 knots-speeds that permit adequate tracking and warning.

Continuous position information for hurricanes is provided to the National Hurricane Center by reconnaissance aircraft and geostationary satellites, and are updated at six hour intervals. This tracking permits National Weather Service analysts to project the time and location of landfalls and to issue hurricane watches and warnings for affected areas (see NWS [1987]). Hurricane watches are issued for an area 36 hours prior to the expectation of hurricane conditions. Hurricane warnings are issued for an area 24 hours prior to the expectation of wind speeds in excess of 73 mph. With the institution of a hurricane warning, plant operators will have sufficient time to take action prior to hurricane arrival.

During the 24-hour period prior to hurricane arrival, NUMARC station blackout initiatives direct plant operators to take actions to enhance the normal EDG reliability and coping capability. These actions include reviewing procedures, restoring systems and components to service, warming, lubricating, starting and load testing EDGs, increasing CST levels, and charging batteries. The safety benefits offered by these actions result from increased EAC availability, above-normal available coping resources (and extended coping times), and lower potential for operator error for hurricane events.

With EDG testing in advance of hurricane arrival, the average EDG will realize a reduction in EDG failures up to 50% depending on mode of failure (i.e., stress versus demand), based on industry EDG failure data reported in NUREG-1032. This data suggests that approximately 50% of failures can be repaired within four hours in non-emergency situations with normal staffing. With 24-hours available, Figure 4.6 of NUREG-1032 indicates up to 75% of EDG failures may be repairable under normal conditions. Enhancement and verification of high EDG reliabilities is one of two major improvements that can reduce the risk of a station blackout. The other is plant safe shutdown in sufficient advance of an anticipated hurricane-induced LOOP to dump a significant portion of the decay heat load.

The relative amount of decay heat removed in a two hour period by the main condenser is approximately 60% of the energy generated in the first four hours following shutdown. By removing this energy through the main condenser, the station blackout coping resources normally reserved for processing this decay heat would be preserved, permitting longer coping times for a four hour water supply.

During the hurricane warning period, "topping-off" water supplies can also extend a normal four hour water supply by several hours. For example, increasing the condensate available for coping above a technical specification level of 65% to 100% available capacity can add several hours of coping time to a rated four hour capability. Topping off the condensate storage tank and placing the plant in a safe shutdown several hours before the hurricane induced LOOP reduces the likelihood of core damage from a subsequent station blackout event.

Actions are also available to extend the time to battery depletion in order to support enhanced coping. Analysis demonstrates that for a typical plant, pre-hurricane actions can effectively support enhanced coping capability for hurricane events. With the plant in an early shutdown, certain loads would not be needed should a station blackout subsequently occur as a result of a hurricane-induced LOOP. Further, other loads could reasonably be stripped after initiation of a LOOP in order to extend the effectiveness of the available charge.

During early shutdown, many air-operated valve operations necessary for decay heat removal following shutdown would also be accomplished while air compressors are available. These operations would result in fewer air-operated valve

operations in a station blackout and longer coping capability involving this resource. In any event, air-operated valves necessary for shutdown can be manually operated or are equipped with backup means for ensuring proper positioning in a station blackout.

The combined effects of these actions (i.e., implementation of plant specific pre-hurricane shutdown requirements and procedures) provide an eight-hour enhanced coping capability under anticipated hurricane conditions.

3. REQUIRED COPING DURATION CATEGORY

3.1 PROCEDURE OVERVIEW

This section provides a methodology for determining the required station blackout coping duration.

3.2 PROCEDURE

Five steps are provided for determining the required coping duration category:

Step 1 Determine the Off-site AC Power Design Characteristic Group
Plant weather, grid, and switchyard features are grouped into three categories of susceptibility to losing off-site power labeled P1, P2, and P3.

Step 2 Classify the EAC Power Supply System Configuration
The redundancy of the emergency AC power system is evaluated and classified among four available groups labeled A, B, C, and D.

Step 3 Determine the Calculated EDG Reliability
The current EDG reliability is determined consistent with NSAC-108 criteria.

Step 4 Determine the Allowed EDG Target Reliability
Based on current EAC reliability, a method is provided for determining an acceptable EAC target reliability.

Step 5 Determine Coping Duration Requirement
Based on the allowed EDG target reliability determined in Step 4, a coping duration category is calculated.

3.2.1 Step One: Determine The Off-site AC Power Design Characteristic Group

The objective of this first step is to distinguish between sites having particular susceptibilities to losing off-site power due to plant-centered, grid-related, and weather-related events. Three off-site power design groups are provided:

P1 - *Sites characterized by redundant and independent power sources that are considered less susceptible to loss as a result of plant-centered and weather-initiated events;*

P2 - *Sites whose off-site power sources are less redundant or independent, or that are more susceptible to extended off-site power losses due to weather-initiated events or more frequent losses due to plant-centered events; and,*

P3 - *Sites whose off-site power sources are (1) least redundant or independent combined with moderate severe weather potential, (2) most susceptible to extended off-site power losses due to weather-initiated or grid-related events, or (3) susceptible to grid-related events.*

These categories are provided by the Staff in the draft station blackout regulatory guide and are designed to be mutually exclusive. Further discussion concerning independence of offsite sources is provided in Section 3.3.4.

THERE ARE FIVE PARTS IN STEP ONE TO DETERMINING THE OFF-SITE POWER DESIGN CHARACTERISTIC GROUP:

PART 1.A *DETERMINE THE SITE SUSCEPTIBILITY TO GRID-RELATED LOSS OF OFFSITE-POWER EVENTS;*

PART 1.B *ESTIMATED FREQUENCY OF LOSS OF OFF-SITE POWER DUE TO EXTREMELY SEVERE WEATHER (ESW GROUP);*

PART 1.C *DETERMINE THE ESTIMATED FREQUENCY OF LOSS OF OFF-SITE POWER DUE TO SEVERE WEATHER (SW GROUP);*

PART 1.D *EVALUATE INDEPENDENCE OF OFF-SITE POWER SYSTEM (I GROUP); AND,*

PART 1.E *DETERMINE OFF-SITE AC POWER DESIGN CHARACTERISTIC GROUP (P GROUP).*

Part 1.A: Determine Site Susceptibility to Grid-Related Loss of Off-site Power Events

Grid-related loss of off-site power events are defined as LOOPs that are strictly associated with the loss of the transmission and distribution system due to insufficient generating capacity, excessive loads, or dynamic instability. Although grid failure may also be caused by other factors, such as severe weather conditions or brush fires, these events are not considered grid-related since they were caused by external events.

The industry average frequency of grid-related events is approximately 0.020 per site-year, with most events isolated to a few systems. According to NUREG-1032, the average occurrence for the majority of systems is about once per 100 site-years. NUREG-1032 notes sites having a frequency of grid-related events at the once per 20 site-year frequency are limited to St. Lucie, Turkey Point, and Indian Point. Accordingly, no other sites are expected to exceed the Once per 20 site-year frequency of grid-related loss of off-site power events.

PLANTS SHOULD BE CLASSIFIED AS P3 SITES IF THE EXPECTED FREQUENCY BASED ON PRIOR EXPERIENCE OF GRID-RELATED EVENTS EXCEEDS ONCE PER 20 YEARS. THIS DOES NOT INCLUDE EVENTS OF LESS THAN 5 MINUTES DURATION. EVENTS OF LONGER DURATION MAY BE EXCLUDED IF THE RESULTS OF ANALYSIS CONCLUDES THE EVENT IS NOT SYMPTOMATIC OF UNDERLYING OR GROWING GRID INSTABILITY.

PLANTS CLASSIFIED AS P3 SITES ON THE BASIS OF GRID EXPERIENCE NEED NOT COMPLETE THE REMAINING PARTS OF THIS STEP IN ORDER TO DETERMINE COPING DURATION REQUIREMENTS.

Part 1.B: Estimated Frequency of Loss of Off-site Power Due to Extremely Severe Weather (ESW Group)

The estimated frequency of loss of off-site power due to extremely severe weather is determined by the annual expectation of storms at the site with wind velocities greater than or equal to 125 mph. These events are normally associated with the occurrence of great hurricanes where high windspeeds may cause widespread transmission system unavailability for extended periods. Since electrical distribution systems are not designed for these conditions, it is assumed that the occurrence of such windspeeds will directly result in the loss of off-site power.

USE METHOD "A" OR "B" BELOW TO DETERMINE THE ESTIMATED FREQUENCY OF LOSS OF OFF-SITE POWER DUE TO EXTREMELY SEVERE WEATHER AT THE SITE AND SELECT AN ESW GROUP:

A. **Site-specific data** provides the most accurate source for calculating the annual frequency of storms with wind velocities greater than or equal to 125 mph, and can be **used in calculating the estimated frequency of loss of off-site power due to extremely severe weather.**

Once the frequency (e) is calculated, use Table 3-1 to assign the site to an ESW Group.

Table 3-1

EXTREMELY SEVERE WEATHER GROUPS (ESW)

ESW GROUP	ANNUAL WINDSPEED EXPECTATION ≥ 125 MPH
1	$e < 3.3 \times 10^{-4}$
2	$3.3 \times 10^{-4} \leq e < 1 \times 10^{-3}$
3	$1 \times 10^{-3} \leq e < 3.3 \times 10^{-3}$
4	$3.3 \times 10^{-3} \leq e < 1 \times 10^{-2}$
5	$1 \times 10^{-2} \leq e$

B. If site data is not readily available to perform this calculation, the annual estimated frequency of loss of off-site power due to extremely severe weather may be derived from data recorded at local weather stations. Alternatively, a loss of off-site power frequency estimate for extremely severe weather may be based on data obtained from the National Oceanic and Atmospheric Administration (NOAA). Site-specific NOAA data is summarized in Table 3-2 along with the appropriate ESW Group.

Table 3-2

EXTREMELY SEVERE WEATHER DATA[a]

SITE	STORMS 125 MPH+	ESW GROUP	SITE	STORMS 125 MPH+	ESW GROUP
ARKANSAS NUCLEAR ONE	0.0002	1	MONTICELLO	0.0003	1
ARNOLD	0.0008	2	NINE MILE POINT	0.0001	1
BEAVER VALLEY	0.0001	1	NORTH ANNA	0.0034	4
BELLEFONTE	0.0001	1	OCONEE	0.0011	3
BIG ROCK POINT	0.0001	1	OYSTER CREEK	0.005	4
BRAIDWOOD	0.001	3	PALISADES	0.0006	2
BROWNS FERRY	0.0001	1	PALO VERDE	0.0004	2
BRUNSWICK	0.013	5	PEACH BOTTOM	0.0026	3
BYRON	0.0002	1	PERRY	0.0001	1
CALLAWAY	0.0001	1	PILGRIM	0.0068	4
CALVERT CLIFFS	0.0038	4	POINT BEACH	0.0036	4
CATAWBA	0.0011	3	PRAIRIE ISLAND	0.002	3
CLINTON	0.0002	1	QUAD CITIES	0.0002	1
COMANCHE PEAK	0.0001	1	RANCHO SECO	0.0005	2
COOK	0.0006	2	RIVER BEND	0.0068	4
COOPER	0.0014	3	ROBINSON	0.0036	4
CRYSTAL RIVER	0.006	4	SALEM	0.0038	4
DAVIS-BESSE	0.0004	2	SAN ONOFRE	0.0001	1
DIABLO CANYON	0.0001	1	SEABROOK	0.0038	4
DRESDEN	0.0001	1	SEQUOYAH	0.0007	2
FARLEY	0.002	3	SHOREHAM	0.01	5
FERMI	0.0001	1	SOUTH TEXAS	0.012	5
FITZPATRICK	0.0001	1	ST LUCIE	0.017	5
FORT CALHOUN	0.0014	3	SUMMER	0.0011	3
FORT ST. VRAIN	0.0001	1	SURRY	0.006	4
GINNA	0.0001	1	SUSQUEHANNA	0.0018	3
GRAND GULF	0.004	4	THREE MILE ISLAND	0.002	3
HADDAM NECK	0.01	5	TROJAN	0.0011	3
HARRIS	0.01	5	TURKEY POINT	0.023	5
HATCH	0.0009	2	VERMONT YANKEE	0.0034	4
HOPE CREEK	0.0038	4	VOGTLE	0.0006	2
INDIAN POINT	0.0079	4	WATERFORD	0.0068	4
KEWAUNEE	0.0036	4	WATTS BAR	0.0001	1
LASALLE	0.0002	1	WNP-2	0.0001	1
LIMERICK	0.002	3	WOLF CREEK	0.0003	1
MAINE YANKEE	0.0028	3	YANKEE ROWE	0.0056	4
MCGUIRE	0.0001	1	ZION	0.0001	1
MILLSTONE	0.012	5			

Note (a): NRC STAFF PROVIDED THE DATA IN TABLE 3-2 USING CLIMATOLOGICAL SOURCES CITED IN THE REFERENCES TO THIS PROCEDURE. NUMARC HAS NOT VERIFIED THE ACCURACY OF THIS DATA.

Part 1C: Determine the Estimated Frequency of Loss of Off-site Power Due to Severe Weather (SW Group)

Four factors are used to calculate the estimated frequency of loss of off-site power due to severe weather:

(1) *Annual expectation of snowfall for the site, in inches [h_1];*

(2) *Annual expectation of tornadoes of severity f2 or greater at the site (i.e., windspeeds greater than or equal to 113 miles per hour), in events per square mile [h_2];*

(3) *Annual expectation of storms for the site with wind velocities between 75 and 124 mph [h_3]; and,*

(4) *Annual expectation of storms with significant salt spray for the site [h_4].*

These factors are combined in the following relationship to yield the estimated frequency of loss of off-site power due to severe weather:

$$f = (1.3 \times 10^{-4}) * h_1 + b * h_2 + (1.2 \times 10^{-2})\ h_3 + c * h_4$$

where:

b	=	*12.5 for sites with multiple rights of way*
b	=	*72.3 for sites with a single right of way*
c	=	*0.78 if site is vulnerable to effects of salt spray*
c	=	*0 for other sites*

Sites which are determined to be susceptible to the effects of salt spray may remedy this situation through design or procedures to minimize the loss of off-site power.

DETERMINE THE ESTIMATED FREQUENCY OF LOSS OF OFF-SITE POWER DUE TO SEVERE WEATHER AS FOLLOWS:

A. Determine the total amount of snowfall in inches which falls on the site in any year. NOAA data for snowfall are provided in Table 3-3. Label the data used as h_1.

B. Determine the expected frequency of "f2+" tornadoes per square mile for the site using plant-specific data. NSSFC data are also provided in Table 3-3. Label the data used as h_2.

C. Determine the expected frequency of storms with winds between 75 and 124 mph at the site. NOAA data are also provided in Table 3-3. Label the data used as h_3.

D. Determine the expected frequency of hurricanes and tropical storms with significant salt spray for the site. NOAA data for sites vulnerable to the effects of salt spray are also provided in Table 3-3. Label the data used as h_4.

E. Calculate the estimated frequency of loss of off-site due to severe weather, f, in events per year.

F. Use Table 3-4 to determine the Severe Weather Group (SW Group).

Table 3-3

SEVERE WEATHER DATA[b]

SITE	SNOWFALL	TORNADO	STORMS	SALT SPRAY	SITE	SNOWFALL	TORNADO	STORMS	SALT SPRAY
	(h1)	(h2)	(h3)	(h4)		(h1)	(h2)	(h3)	(h4)
ARKANSAS NUCLEAR ONE	6	0.00045	0.067	0	MONTICELLO	46	0.0001218	0.08	0
ARNOLD	33	0.000257	0.25	0	NINE MILE POINT	89	0.0000058	0.06	0
BEAVER VALLEY	45	0.0000692	0.03	0	NORTH ANNA	15	0.0000367	0.08	0
BELLEFONTE	4	0.000253	0.029	0	OCONEE	6	0.000038	0.12	0
BIG ROCK POINT	97	0.0000183	0.006	0	OYSTER CREEK	17	0.0000038	0.063	0
BRAIDWOOD	40	0.000205	0.08	0	PALISADES	48	0.0001845	0.1	0
BROWNS FERRY	4	0.000415	0.029	0	PALO VERDE	0	0.0000018	0.125	0
BRUNSWICK	2	0.000087	0.12	0	PEACH BOTTOM	22	0.0000601	0.026	0
BYRON	35	0.000118	0.01	0	PERRY	38	0.000066	0.08	0
CALLAWAY	24	0.000106	0.05	0	PILGRIM	42	0.000005	0	0.08
CALVERT CLIFFS	9	0.0000077	0.062	0	POINT BEACH	42	0.000035	0.1	0
CATAWBA	6	0.000104	0.12	0	PRAIRIE ISLAND	46	0.0001713	0.08	0
CLINTON	24	0.000305	0.1	0	QUAD CITIES	40	0.000089	0.15	0
COMANCHE PEAK	4	0.000109	0.05	0	RANCHO SECO	0	0.0000006	0.1	0
COOK	48	0.000145	0.1	0	RIVER BEND	0	0.000154	0.09	0
COOPER	30	0.000248	0.5	0	ROBINSON	1	0.000197	0.09	0
CRYSTAL RIVER	0	0.000013	0.1	0	SALEM	22	0.0000275	0.045	0
DAVIS-BESSE	38	0.000083	0.11	0	SAN ONOFRE	0	0.0000153	0.001	0
DIABLO CANYONO		0.0000001	0.07	0	SEABROOK	65	0.0000291	0.045	0
DRESDEN	40	0.000181	0.08	0	SEQUOYAH	4	0.0001499	0.1	0
PARLEY	0	0.000081	0.05	0	SHOREHAM	26	0.0000251	0.08	0
FERMI	32	0.0000939	0.05	0	SOUTH TEXAS	0	0.000031	0.12	0
FITZPATRICK	89	0.0000057	0.06	0	ST LUCIE	0	0.000013	0.15	0
FORT CALHOUN	29	0.000141	0.5	0	SUMMER	2	0.000106	0.12	0
FORT ST. VRAIN	59	0.000013	0.02	0	SURRY	8	0.000044	0.1	0
GINNA	89	0.0000054	0.06	0	SUSQUEHANNA	44	0.0000292	0.028	0
GRAND GULF	1	0.000392	0.03	0	THREE MILE ISLAND	35	0.0000392	0.027	0
HADDAM NECK	27	0.000089	0.08	0	TROIAN	7	0.0000004	0.14	0
HARRIS	8	0.000222	0.13	0	TURKEY POINT	0	0.000012	0.18	0
HATCH	0	0.000029	0.022	0	VERMONT YANKEE	79	0.0000871	0.04	0
HOPE CREEK	22	0.0000275	0.045	0	VOGTLE	2	0.000036	0.022	0
INDIAN POINT	29	0.0000141	0.08	0	WATERFORD	0	0.000032	0.09	0
KEWAUNEE	42	0.000006	0.1	0	WATTS BAR	10	0.0001422	0.1	0
LASALLE	40	0.000221	0.08	0	WNP-2	53	0.0000002	0.03	0
LIMERICK	22	0.0000285	0.027	0	WOLF CREEK	20	0.0003815	0.23	0
MAINE YANKEE	74	0.000001	0.034	0	YANKEE ROWE	79	0.000068	0.063	0
MCGUIRE	6	0.0000102	0.03	0	ZION	40	0.00005	0.01	0
MILLSTONE	27	0.000046	0	0.18					

NOTE (b): NRC STAFF PROVIDED THE DATA IN TABLE 3-3 USING CLIMATOLOGICAL SOURCES CITED IN THE REFERENCES TO THIS PROCEDURE. NUMARC HAS NOT VERIFIED THE ACCURACY OF THIS DATA.

Table 3-4

SEVERE WEATHER GROUPS (SW)

SW GROUP	ESTIMATED FREQUENCY OF LOSS OF OFFSITE POWER
1	$f < 0.0033$
2	$0.0033 \leq f < 0.0100$
3	$0.0100 \leq f < 0.0330$
4	$0.0330 \leq f < 0.100$
5	$0.10 \leq f$

Part 1D: Evaluate Independence of Off-site Power System (I Group)

The potential for long duration loss of off-site power events can have a significant impact on station blackout risk and required coping durations. Long duration LOOP events are associated with grid failures due to severe weather conditions or unique transmission system features. Shorter duration LOOP events tend to be associated with specific switchyard features. Two features, in particular, are of special importance: (1) the independence of the off-site power sources constituting the preferred power supply to the shutdown buses on-site, and (2) the power transfer schemes when the normal source of AC power is lost.

Two plant groupings are specified in this part for classifying the interface of the preferred power supply to the safe shutdown bus: I1/2 and I3. The I1/2 group is characterized by features associated with greater independence and redundancy of sources, and a more desirable transfer scheme. I3 sites have simpler, less desirable off-site power systems and switchyard capabilities. The importance of the site groupings becomes evident when combined with the potential for losing off-site power due to severe and extremely severe weather.

THE OFF-SITE POWER SYSTEM IS IN THE I3 GROUP IF:

(1) A "YES" ANSWER CAN BE ASSIGNED TO CONDITION "A" BELOW,

AND

(2) A "YES" CAN BE ASSIGNED TO EITHER CONDITIONS "B(1)" OR "B(2)", BELOW.

A. All off-site power sources are connected to the unit's safe shutdown buses through (1) one switchyard, or (2) two or more electrically connected switchyards.

B(1) The normal source of AC power is from the unit main generator and there are no automatic transfers and one or more manual transfers of all safe shutdown buses to preferred or alternate off-site sources.

B(2) The normal source of AC power is from the unit main generator and there is one automatic transfer and no manual transfers of all safe shutdown buses to one preferred or one alternate off-site power source.

OTHERWISE THE SITE IS ASSIGNED TO THE I1/2 GROUP.

Part 1E: Determine Off-site AC Power Design Characteristic Group

Site susceptibility to loss of off-site power is separated into three basic groups, based on combinations of features. The determining features are: (1) independence of off-site power, (2) severe weather potential, measured either by experience or recurrence intervals, and (3) extremely severe weather potential. The following tables establish the off-site power design characteristic group.

A. REVIEW THE INDEPENDENCE OF OFF-SITE POWER GROUP, SW GROUP AND ESW GROUP, AND

USE THE FOLLOWING TABLES TO DETERMINE THE OFF-SITE AC POWER DESIGN CHARACTERISTIC GROUP.

OFF-SITE AC POWER DESIGN CHARACTERISTIC GROUP MATRIX

I1/2 SITES

SW GROUP \ ESW GROUP	1	2	3	4	5
1	P1	P1	P1	P2	P3
2	P1	P2	P2	P2	P3
3	P2	P2	P2	P3	P3
4	P3	P3	P3	P3	P3
5	P3	P3	P3	P3	P3

Table 3-5a

I3 SITES

SW GROUP \ ESW GROUP	1	2	3	4	5
1	P2	P2	P2	P2	P3
2	P2	P2	P2	P2	P3
3	P2	P2	P3	P3	P3
4	P3	P3	P3	P3	P3
5	P3	P3	P3	P3	P3

Table 3-6a

NOTE: Coastal plants are susceptible to long duration LOOPS as a result of extremely severe weather associated with hurricanes. As a result, plants with otherwise sufficient EDG reliability and configuration and lower susceptibility to severe weather events may be in a higher coping duration category solely due to the probability of a hurricane induced LOOP.

B. IF A PLANT IS SUSCEPTIBLE TO A HURRICANE INDUCED LOOP AND HAS HURRICANE RESPONSE PROCEDURES WHICH MEET THE GUIDELINES OF SECTION 4.2.3 OF THIS DOCUMENT, USE THE FOLLOWING TABLES TO DETERMINE THE OFF-SITE POWER DESIGN CHARACTERISTIC GROUP.

OFF-SITE AC POWER DESIGN CHARACTERISTIC GROUP MATRIX

For Hurricane Exposed Plants

I1/2 SITES

SW GROUP \ ESW GROUP	1	2	3	4	5
1	P1	P1	P1	P2*	P3*
2	P1	P2*	P2	P2	P3*
3	P2	P2	P2	P3*	P3
4	P3	P3	P3	P3	P3
5	P3	P3	P3	P3	P3

Table 3-5b

I3 SITES

SW GROUP \ ESW GROUP	1	2	3	4	5
1	P2	P2	P2	P2	P3*
2	P2	P2	P2	P2	P3*
3	P2	P2	P3	P3	P3
4	P3	P3	P3	P3	P3
5	P3	P3	P3	P3	P3

Table 3-6b

*DENOTES SITE UPGRADE ATTRIBUTED TO IMPLEMENTATION OF PLANT SPECIFIC PRE-HURRICANE SHUTDOWN REQUIREMENTS AND PROCEDURES WHICH PROVIDE AN ENHANCED 8-HOUR COPING CAPABILITY UNDER ANTICIPATED HURRICANE CONDITIONS.

3.2.2 Step Two: Classify The Emergency AC Power Supply System Configuration

After the likelihood of losing off-site power, the redundancy of the emergency AC power system is the next most important contributor to station blackout risk. With greater EAC system redundancy, the potential for station blackout diminishes, as does the likelihood of core damage. The importance of EAC redundancy is reflected in this procedure through the use of four distinct EAC configuration groups:

A - Characterized by highly redundant and independent EAC sources to safe shutdown equipment;

B - Having better than typical redundant and independent EAC sources to safe shutdown equipment;

C - Having typical redundant and independent EAC sources to safe shutdown equipment; and,

D - Having the lowest level of independency and redundancy in EAC sources powering safe shutdown equipment.

Placement in one of the groups listed depends on the number of EAC standby power supplies available and the number required to operate AC-powered decay heat removal equipment necessary to achieve and maintain safe shutdown in a station blackout. Overall, the greater the level of EAC redundancy, the less restrictive are the station blackout coping durations and maximum EDG failure rates before longer coping durations are required, or corrective actions become necessary.

The potential for excess EAC power sources to be used as Alternate AC is directly related to the existing level of EAC redundancy. Since EAC redundancy is an important parameter for determining station blackout coping duration categories, EAC power sources relied upon as Alternate AC power sources must not also be considered when assessing the required coping duration.

Accordingly, the following process precludes the use of an EAC power source as both an input to determine the EAC group and an Alternate AC source. This process eliminates the potential for "double counting" the value of an individual EAC power source, both as preventing the station blackout, and in responding to its occurrence.

To illustrate this point, consider a single unit site that has three EAC power sources, and needs only one for safe shutdown. This site can be classified as either a one-out-of-three site (EAC Group A); or a one-out-of-two site (EAC Group C) with the third EAC power source available as a potential Alternate AC power source, if it meets the criteria for Alternate AC specified in Appendix B.

THIS STEP CONSISTS OF THREE PARTS:

PART 2.A	DETERMINE THE NUMBER OF EAC POWER SUPPLIES NOT CREDITED AS ALTERNATE AC POWER SOURCES;
PART 2.B	IDENTIFY THE SMALLEST NUMBER OF EAC POWER SOURCES NECESSARY FOR SAFE SHUTDOWN; AND,
PART 2.C	SELECT THE EAC POWER CONFIGURATION GROUP.

Part 2.A Determine the Number of EAC Power Supplies Normally Available

A. SINGLE UNIT OR MULTI-UNIT SITES WITH NORMALLY DEDICATED POWER SUPPLIES

Count the total number of standby power supplies (see Appendix A) normally available to the blacked-out unit's safe shutdown equipment that are not being used as an Alternate AC power source.

B. MULTI-UNIT SITES WITH NORMALLY SHARED POWER SUPPLIES

Count the total number of dedicated and shared standby power supplies normally available to safe shutdown equipment at each site that are not being used as an Alternate AC power source.

Part 2.B Determine the Number of Necessary EAC Standby Power Supplies

The number of EAC standby power supplies required for station blackout is based on the AC loads needed at each unit to remove decay heat (including the heat generated by AC-powered decay heat removal systems) in order to achieve and maintain safe shutdown with off-site power unavailable.

The number of EAC standby power sources necessary to operate safe shutdown equipment may be less than that required for LOCA loads.

The number of necessary EAC standby power sources should be determined by accounting for the individual safe shutdown loads, or inferred from the site's design basis for operating Class 1E AC equipment without off-site AC power.

A. SINGLE UNIT OR MULTI-UNIT SITES WITH NORMALLY DEDICATED POWER SUPPLIES

Count the total number of EAC standby power supplies necessary to operate safe shutdown equipment during a station blackout on a per unit basis.

B. MULTI-UNIT SITES WITH SHARED NORMALLY POWER SUPPLIES

Count the total number of EAC standby power supplies necessary to operate safe shutdown equipment during a station blackout for all units at the site.

Part 2.C Select the EAC Power Configuration Group

USE THE TABLE PROVIDED BELOW TO SELECT THE EAC GROUP:

Table 3-7

EAC GROUP	SHARED AND DEDICATED SUPPLIES NECESSARY FOR SAFE SHUTDOWN	SUPPLIES AVAILABLE
A	1	3 DEDICATED
A	1	4
B	2	5
B	2	4
C	1	2 DEDICATED
C	1	3 SHARED
D	3	4
D	3	5
D	2	3
D	1	2 SHARED

Dedicated -- for EAC standby power supplies not normally shared with other units at a site

Shared -- for EAC standby power supplies in which some number are normally capable of providing AC power to safe shutdown equipment at more than one unit at a site concurrently.

*** If any of the EAC power sources are normally shared among units at a multi-unit site, this is the total number of shared and dedicated sources for those units at the site.**

3.2.3 Step Three: Determine The Calculated EDG Reliability

The unit EDG reliability is used in conjunction with the site's off-site power design characteristics (i.e., P1, P2, or P3), and the EAC configuration (A, B, C, or D) to determine the unit's required station blackout coping duration. The unit EDG reliability is calculated by averaging the individual EDG reliability for the last 20, 50, and 100 demands for each machine. However, if the total number of valid demands is less than 100 (e.g., newly licensed plants, EDGs which have undergone intensive maintenance or a reliability requalification program), the EDG reliability over the last 20, and the last 50 if available, can be averaged and compared to the evaluation criteria in Section 3.2.4. If the unit's EDG reliability over the last 20 demands is > 0.90, or > 0.94 over the last 50 demands, then the unit may select an EDG target reliability of either 0.95 or 0.975 as detailed in Section 3.2.4.

The objective of the three-tier approach to reliability measurement is to provide greater depth of understanding regarding reliability trends. The 20-demand sample set is the most volatile, and offers a very sensitive indication of EDG performance. Since this indicator moves with each incremental failure or success, it is not considered a reliable measure of long-term performance. Similarly, the 100-demand sample set offers a long-term trend indication, while providing limited insight to recent trends due to data smoothing effects. The 50-demand sample set bridges the two indicators while also providing an intermediate level. Taken together, the set of indicators provides a fairly complete picture of EDG reliability.

DETERMINE THE CURRENT UNIT EDG RELIABILITY:

(1) CALCULATE THE MOST RECENT EDG RELIABILITY FOR EACH EDG BASED ON THE LAST 20, 50, AND 100 DEMANDS (USING NSAC-108 DEFINITIONS AND METHODOLOGY CONTAINED IN SECTION 2 OF THAT DOCUMENT OR EQUIVALENT).

(2) CALCULATE THE NUCLEAR UNIT AVERAGE EDG RELIABILITY FOR THE LAST 20 DEMANDS BY AVERAGING THE RESULTS FROM (1), ABOVE.

CALCULATE THE NUCLEAR UNIT AVERAGE EDG RELIABILITY FOR THE LAST 50 DEMANDS BY AVERAGING THE RESULTS FROM (1), ABOVE.

CALCULATE THE NUCLEAR UNIT AVERAGE EDG RELIABILITY FOR THE LAST 100 DEMANDS BY AVERAGING THE RESULTS FROM (1), ABOVE.

3.2.4 Step Four: Determine Allowed EDG Target Reliability -

The minimum EDG reliability should be targeted at 0.95 per demand per EDG for plants in EAC Groups A, B, C, and 0.975 per demand per EDG for plants in EAC Group D. These reliability levels should be considered minimum target reliabilities. Each plant should establish an EDG Reliability Program as outlined in Appendix D to this document. Plants which select a target EDG reliability of 0.975 should utilize this target level in their reliability program. If the diesel generator performance falls below the target reliability level specified, action should be taken through an EDG reliability program such as set forth in Appendix D to restore the target reliability level.

The unit EDG reliability for the last 20, 50, and 100 demands calculated in the previous step provides the allowed target reliability used in determining minimum required station blackout coping durations in the next step.

ALLOWED TARGET RELIABILITIES ARE DETERMINED AS FOLLOWS:

(1) COMPARE THE CALCULATED AVERAGE NUCLEAR UNIT EDG RELIABILITY DETERMINED IN SECTION 3.2.3 TO THE CRITERIA BELOW:

<u>Evaluation Criteria</u>

LAST 20 DEMANDS > 0.90 RELIABILITY
LAST 50 DEMANDS > 0.94 RELIABILITY
LAST 100 DEMANDS > 0.95 RELIABILITY

(2) IF THE EAC GROUP IS A, B, OR C, <u>AND</u> ANY OF THE THREE EVALUATION CRITERIA IN SECTION 3.2.4, STEP FOUR, PART (1) ARE MET, THEN THE NUCLEAR UNIT MAY SELECT AN EDG RELIABILITY TARGET OF EITHER 0.95 OR 0.975 FOR DETERMINING THE REQUIRED STATION BLACKOUT COPING DURATION. IF THE EAC GROUP IS D, AND ANY OF THE THREE EVALUATION CRITERIA IN SECTION 3.2.4, STEP FOUR, PART (1) ARE MET, THEN THE ALLOWED EDG RELIABILITY TARGET IS 0.975.

(3) IF THE EAC GROUP IS A, B, OR C, AND NONE OF THE THREE EVALUATION CRITERIA IN SECTION 3.2.4, STEP FOUR, PART (1) ARE MET, THEN 0.95 SHOULD BE USED AS THE RELIABILITY TARGET FOR DETERMINING THE REQUIRED STATION BLACKOUT COPING DURATION.

ADDITIONALLY, IF THE RELIABILITY IS LESS THAN 0.90 BASED ON THE LAST 20 DEMANDS, THEN ACCEPTABILITY OF THE COPING DURATION RESULTING FROM USING 0.95 MAY REQUIRE FURTHER JUSTIFICATION.

IF THE EAC GROUP IS D AND NONE OF THE THREE EVALUATION CRITERIA IN PART (1) ARE MET, THE REQUIRED COPING DURATION CATEGORY CALCULATED IN STEP FIVE, SECTION 3.2.5 SHOULD BE INCREASED TO THE NEXT HIGHEST LEVEL (I.E., FOUR HOURS BECOMES EIGHT HOURS; EIGHT HOURS BECOMES 16 HOURS).

3.2.5 Step Five: Determine Coping Duration Category

USE THE TABLE PROVIDED BELOW TO DETERMINE THE COPING DURATION REQUIREMENT IN HOURS:

Table 3-8

OFFSITE POWER GROUP (From Section 3.2.1)	EAC GROUP (From Section 3.2.2)	ALLOWED EDG TARGET RELIABILITY (Per Demand) (From Section 3.2.4)	REQUIRED COPING DURATION CATEGORY
P1	A	0.950	2
P1	B	0.950	4
P1	C	0.950	4
P1	D	0.975	4
P2	A	0.950	4
P2	B	0.950	4
P2	C	0.975	4
P2	C	0.950	8
P2*	C	0.950	4
P2	D	0.975	8
P2*	D	0.975	4
P3	A	0.975	4
P3	A	0.950	8
P3*	A	0.950	4
P3	B	0.975	4
P3	B	0.950	8
P3*	B	0.950	4
P3	C	0.975	8
P3*	C	0.975	4
P3	C	0.950	16
P3*	C	0.950	8
P3	D	0.975	8
P3*	D	0.975	4

*** Denotes site upgrade attributable to implementation of plant specific pre-hurricane shutdown requirements and procedures which provide an enhanced coping capability under anticipated hurricane conditions.**

3.2.6 Required Action

Step Five (Section 3.2.5) yields one of the four coping duration categories discussed in the NRC Station Blackout Regulatory Guide 1.155: two hours, four hours, eight hours, or 16-hours. Plants in the eight and 16-hour categories should undertake actions to reduce risk consistent with NUMARC Station Blackout Initiative 1.

THE FOLLOWING COURSES OF ACTION ARE AVAILABLE TO REDUCE THE ASSESSED RISK OF STATION BLACKOUT:

(1) IMPLEMENT ACTION TO REDUCE THE REQUIRED COPING DURATION TO AT LEAST THE FOUR HOUR CATEGORY BY:

(a) REVIEWING PLANT-SPECIFIC WEATHER DATA;

(b) MODIFYING THE SWITCHYARD TO CHANGE THE I-GROUP; AND/OR,

(c) MODIFYING THE PLANT TO CHANGE THE EDG CONFIGURATION; AND/OR,

(d) IMPROVING EDG RELIABILITY.

(2) INSTALL OR UTILIZE AN EXISTING ALTERNATE AC POWER SOURCE THAT MEETS THE CRITERIA PROVIDED IN APPENDIX B.

4. STATION BLACKOUT RESPONSE PROCEDURES

4.1 OVERVIEW

Most existing plant procedures are based on procedure guidelines generated by NSSS vendors or plant-specific analysis, and provide the operator with substantial direction for responding to a station blackout event. Plant procedures may also address power restoration and severe weather concerns. Actions that may not be addressed in existing procedures, but are important considerations during a station blackout, are addressed below. Utilities should review their plant procedures to assure these considerations are addressed.

As provided by NUMARC Station Blackout Initiative 2, plant staffs should review, and revise as appropriate, their operating procedures using the technical bases and associated guidelines provided in this document. Appropriate plant personnel should be trained on any new or revised procedures resulting from this initiative.

4.2 OPERATING PROCEDURES GUIDELINES

4.2.1 Station Blackout Response Guidelines (NUMARC Station Blackout Initiative 2.a)

This section provides guidance for operator actions to be taken in a station blackout event. Section 4.3.1 contains additional information and bases for the guidelines provided in this section.

These guidelines assume a single path to achieve and maintain safe shutdown conditions in a station blackout. In addition to repeated attempts at restoring AC power to a shutdown bus, the path consists of performing operations designed to stabilize the plant using available equipment. Guideline (1) reflects attempts at AC power restoration which may be made from either the preferred or a standby (Class 1E) power source. If an AAC power source is available, it may also be used to restore power. Guidelines (2) through (13) address items to be considered in stabilizing the plant until AC power is restored.

(1) Plant procedures should identify site-specific actions necessary to restore off-site or standby (Class 1E) AC power sources. If an AAC power source is available, it should be started as soon as possible. Plants relying on AAC power sources should start the AAC power source and commence loading shutdown equipment within the first

hour of a station blackout.

(2) Plant procedures should specify actions necessary to assure that shutdown equipment (including support systems) necessary in a station blackout can operate without AC power.

(3) Plant procedures should recognize the importance of AFWS/HPCIS/HPCS/RCICS during the early stages of the event, and direct the operators to invest appropriate attention to assuring their continued, reliable operation throughout the transient since this ensures decay heat removal.

(4) Plant procedures should identify the sources of potential reactor inventory loss and specify actions to prevent or limit significant loss.

(5) Plant procedures should ensure that a flowpath is promptly established for makeup flow from the CST to the steam generator/nuclear boiler and identify backup water sources to the CST in order of intended use. Additionally, plant procedures should specify clear criteria for transferring to the next preferred source of water.

(6) Plant procedures should identify individual loads that need to be stripped from the plant DC buses (both Class 1E and non-Class 1E) for the purpose of conserving DC power.

(7) Plant procedures should specify actions to permit appropriate containment isolation and safe shutdown valve operations while AC power is unavailable. These actions may include:

(a) providing additional bottled air or nitrogen at the valves;

(b) specifying manual valve operation to maintain shutdown (e.g., manual valve seating to reduce system losses)

(c). ensuring appropriate containment integrity.

(8) Plant procedures should identify the portable lighting necessary for ingress and egress to plant areas containing shutdown or AAC equipment requiring manual operation.

(9) Plant procedures should consider the effects of AC power loss on area access, as well as the need to gain entry to other locked areas where remote equipment operation is necessary.

(10) Plant procedures should consider loss of ventilation effects on specific energized equipment necessary for shutdown (e.g., those containing internal electrical power supplies or other local heat sources that may be

energized or present in a station blackout). These procedures should address:

(a) specific room or cabinet temperatures or symptoms (e.g., alarms or indication of loss of cooling) readily identifiable by the operator, and the response thereto;

(b) methods for providing necessary ventilation and/or supplemental cooling within 30 minutes;

(c) the potential need for operator action to override HPCIS/RCICS steam line isolation on high temperature;

(d) opening cabinet doors containing instrumentation in control rooms necessary for safe shutdown in a station blackout within 30 minutes, as required; and,

(e) effects of actuation of fire protection features due to elevated temperature.

(11) Plant procedures should consider habitability requirements at locations where operators will be required to perform manual operations.

(12) Non-Class 1E equipment relied upon to cope for the required station blackout coping duration should be addressed in a maintenance program.

(13) Plant procedures should consider loss of heat tracing effects for equipment necessary to cope with a station blackout. Alternate steps, if needed, should be identified to supplement planned action.

4.2.2 AC Power Restoration (NUMARC Station Blackout Initiative 2.b)

This section provides guidance for operations and load dispatcher personnel concerning the proper course of action for restoring AC power in a station blackout. Section 4.3.2 contains additional information and bases for the guidelines provided in this section.

(1) Load dispatchers should give the highest possible priority to restoring power to nuclear units. Procedures and training should consider several potential methods of transmitting power from blackstart capable units to the nuclear plant.

(2) Should incoming transmission lines to a nuclear power plant be damaged, high priority should be assigned to repair and restoration activities to at least one line capable of feeding shutdown equipment.

(3) Repair crews engaging in power restoration activities for nuclear units should be given high priority for manpower, equipment, and materials.

(4) Portable AC generators should be designated as backup sources, if available, and directed to nuclear power plant sites. Procedures should address pre-planned actions and identify required equipment.

(5) Once preferred and/or standby (Class 1E) AC power becomes available, station procedures should specify the sequence of circuit breaker operations required to restore AC power to shutdown equipment. Any additional actions such as pulling or replacing fuses should also be identified.

4.2.3 Severe Weather Guidelines (NUMARC Station Blackout Initiative 2.c)

This section provides guidance for operators to determine the proper course of action due to the onset of severe weather, particularly hurricanes. Section 4.3.3 contains additional information and bases for the guidelines provided in this section.

The characteristics of hurricanes which allow them to be tracked provides advance warning and the opportunity for actions to put the plant into a shutdown condition. These actions can greatly reduce the consequences of a hurricane-induced LOOP with a subsequent station blackout. With sufficient warning, actions may also be taken to enhance the reliability of AC power sources.

Actions for Hurricane

(1) The plant procedures should identify site-specific actions necessary to prepare for the onset of a hurricane. These actions should be initiated when a hurricane warning is issued for the plant site area and should include:

- (a) inspecting the site for potential missiles and reducing this potential;
- (b) reviewing the adequacy of site staff to support operations and repair;
- (c) expediting the restoration of important plant systems and components to service;
- (d) warming and lubricating standby (Class 1E) AC power sources;
- (e) determining the status of Alternate AC sources (if available) and taking necessary actions to ensure their availability;
- (f) increasing CST inventory;
- (g) placing battery chargers in service (if applicable); and,
- (h) start and load test EDGs.

(2) Utility procedures should identify additional plant support staff and the method of contacting them once a hurricane notice has been issued by the National Weather Service.

(3) Plant procedures should specify actions necessary to ensure equipment required for station blackout response is available.

(4) Plant procedures should address the following items prior to a hurricane arrival at a site:

(a) the site-specific indicator should ensure that the plant would be in safe shutdown two hours before the anticipated hurricane arrival at the site (i.e., sustained windspeeds in excess of 73 mph);

(b) operator review of station blackout procedures; and,

(c) operator review of procedures to line up and operate the switchyard spraydown system (if installed).

The actions identified in Items 1-4 above result in a coping categorization consistent with Section 3.2.1, Part 1E(B) and Section 3.2.5 of these guidelines.

Actions for Tornado

Plant procedures should identify site-specific actions necessary to prepare for the onset of a tornado. These actions should include:

(a) inspecting the site for potential missiles and reducing this potential, and

(b) expediting the restoration of important plant systems and components to service.

4.3 SUPPORTING INFORMATION

4.3.1 Station Blackout Response Guidelines

This section provides the bases and related supplemental information for the operating procedure guidelines of Section 4.2.1.

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(1) Plant procedures should identify site-specific actions necessary to restore offsite or standby (Class 1E) AC power sources. If an AAC power source is available it should be started as soon as possible. Plants relying on AAC power sources should start the AAC power source and commence loading shutdown equipment within the first hour of a station blackout.

These actions include:

(a) Early commitment of available staff to restore AC power

This should occur within the first few minutes of a station blackout

(b) Isolating the shutdown bus to be loaded onto the AAC system from the preferred power supply and blacked out unit's Class 1E power sources

This can be achieved by circuit breaker operation, and pulling fuses at the switchgear disabling circuit breaker control power or by manual interlocks.

(c) Starting and/or preparing the AAC source for loading.

(d) Transferring the designated shutdown bus to the AAC system.

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(2) *Plant procedures should specify actions necessary to assure that shutdown equipment (including support systems) necessary in a station blackout can operate without AC power.*

Cooling functions provided by such systems as auxiliary building cooling water, service water, or component cooling water may be required in order for shutdown systems to perform their safety function. For example, after TMI it was recognized that a steam driven auxiliary feedwater pump might have relied on component cooling or service water to cool the bearing lubricating oil rather than relying on sump cooling provided by pump discharge. Systems potentially supplemented in this manner may include component/auxiliary cooling water, service water, and auxiliary/reactor building cooling water systems.

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(3) *Plant procedures should recognize the importance of AFWS/HPCIS/HPCS/RCICS during the early stages of the event and direct the operators to invest appropriate attention to assuring its continued, reliable operation throughout the transient since this ensures decay heat removal.*

The risk of core damage due to station blackout can be significantly reduced by assuring the availability of AFWS/HPCIS/HPCS/RCICS, particularly in the first 30 minutes to one hour of the event. A substantial portion of the decay and sensible reactor heat can be removed during this period. AFWS/HPCIS/HPCS/RCICS availability can be assured by providing a reliable supply of condensate, monitoring turbine conditions (particularly lubricating oil flow and temperature), and maintaining nuclear boiler/steam generator water levels. This step helps to ensure that the core remains adequately covered and cooled during a station blackout event.

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(4) *Plant procedures should identify the sources of potential reactor inventory loss, and specify actions to prevent or*

limit significant loss.

Actions should be linked to clear symptoms of inventory loss (e.g., specific temperature readings provided by sensors in relief valve tail pipes), associated manual or DC motor driven isolation valves, and their location. Procedures should establish the priority for manual valve isolation based on estimated inventory loss rates early in the event. If manual valves are used for leak isolation, they should be accessible, sufficiently lighted for access and use, and equipped with a handwheel, chain, or reachrod. If valves are locked in position, keys or cutters should be available in the control room. Procedures should identify the location of valves, keys, and cutters.

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(5) *Plant procedures should ensure that a flowpath is promptly established for makeup flow from the CST to the steam generator/nuclear boiler and identify backup water sources to the CST in order of intended use. Additionally, plant procedures should specify clear criteria for transferring to the next preferred source of water.*

All stored water sources may be assumed to be available in a station blackout at their nominal capacities, including water stored in non-safety tanks. Alternate water delivery systems can be considered available on a case by case basis. In general, all condensate storage tanks should be used first. The main condenser may be assumed to be available if a pump can be operated and is capable of making up (1) to the AFW/HPCI/HPCS/RCIC pump suction with sufficient head and flow, or (2) directly to a CST (safety or non-safety). After the CSTs are exhausted, demineralized or borated water tanks may be used as appropriate. Heated torus water should be used only if sufficient NPSH can be established. Finally, when all other preferred water sources have been depleted, lower water quality sources may be pumped as makeup flow using available equipment (e.g. a diesel driven fire pump). Procedures should clearly specify the conditions when the operator is expected to resort to increasingly impure water sources.

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(6) *Plant procedures should identify individual loads that need to be stripped from the plant DC buses (both Class 1E and non-Class 1E) for the purpose of conserving DC power.*

DC power is needed in a station blackout for such loads as shutdown system instrumentation, EDG field flashing, circuit breaker operations, and motor-driven valve operators. Emergency lighting may also be powered by safety-related batteries. However, for many plants, this lighting may have been supplemented by Appendix R and security lights, thereby allowing the emergency lighting load to be eliminated. Station blackout procedures should direct operators to conserve DC power during the event by stripping nonessential loads as soon as practical. Early load stripping can significantly extend the availability of the blacked-out unit's Class 1E

batteries. For plants with turning gear loaded on the batteries, stripping this load early in the transient can also significantly extend battery availability. In certain circumstances, AFW/HPCI/HPCS/RCIC operation may be extended by throttling flow to a constant rate, rather than by stroking valves in open-shut cycles.

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(7) *Plant procedures should specify actions to permit appropriate containment isolation and safe shutdown valve operations while AC power is unavailable.*

Compressed air is used to operate (cycle) some valves used for decay heat removal and in reactor auxiliary systems (e.g. identifying letdown valves or reactor water cleanup system valves that need to be closed). Valves requiring manual valve operations are identified in Section 7.2.3. Most containment isolation valves are in the normally closed or failed closed position during power operation. Many other classes of containment isolation valves are not of concern during a station blackout. Section 7.2.5 provides guidance on determining valves of concern which need to be capable of being closed.

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(8) *Plant procedures should identify the portable lighting necessary for ingress and egress to plant areas containing shutdown or AAC equipment requiring manual operation.*

Areas requiring continuous occupancy for instrumentation monitoring or equipment operation may require portable lighting as necessary to perform essential functions. Lighting provided to meet the requirements of Section III.J, 10 CFR 50 Appendix R, for achieving safe shutdown is generally adequate if it is independent of the preferred and emergency AC power system.

---

(9) *Plant procedures should consider the effects of AC power loss on area access, as well as the need to gain entry to other locked areas where remote equipment operation is necessary.*

At some plants, the security system may be adversely affected by the loss of the preferred or Class 1E power supplies in a station blackout. In such cases, manual actions specified in station blackout response procedures may require additional actions to obtain access.

---

(10) *Plant procedures should consider loss of ventilation effects on specific energized equipment necessary for shutdown (e.g., those containing internal electrical power supplies or other local heat sources that may be energized or present in a station blackout).*

Station blackout procedures should identify specific actions to be taken to ensure that equipment failure does not

occur as a result of a loss of forced ventilation. Actions should be tied to either the actual loss of AC power or upon reaching certain temperatures in the plant. Plant areas requiring additional cooling are likely to be locations containing shutdown instrumentation and power supplies, turbine-driven decay heat removal equipment, and in the vicinity of the inverters. These areas include: steam driven AFW pump room, HPCIS/HPCS and RCICS pump rooms, the control room, and logic cabinets. Cooling may be accomplished by opening doors to rooms and electronic and relay cabinets, and/or providing supplemental cooling.

Air temperatures may be monitored during a station blackout event through the use of locally mounted thermometers inside cabinets and in plant areas where cooling may be needed. Alternatively, procedures may direct the operator to take action to provide for alternate cooling in the event normal cooling is lost. Upon loss of these systems, or indication of temperatures outside the maximum normal range of values, the procedures should direct supplemental cooling be provided to the affected cabinet or area, and/or designate alternate means for monitoring system functions.

For the limited cooling requirements of a cabinet containing power supplies for instrumentation, simply opening the back doors is effective. For larger cooling loads, such as HPCIS/HPCS, RCICS, and AFWS pump rooms, portable engine-driven blowers may be considered during the transient to augment the natural circulation provided by opening doors. The necessary rate of air supply to these rooms may be estimated on the basis of rapidly turning over the room's air volume.

Temperatures in the HPCI pump room and/or steam tunnel for a BWR may reach levels which isolate HPCIS or RCICS steam lines to protect against a steam line break. Supplemental cooling or the capability to override the isolation feature may be necessary at some plants. The procedures should identify the corrective action required, if necessary.

Actuation setpoints for fire protection systems are typically at 165 - 180 °F. It is expected that temperature rises due to loss of ventilation during a station blackout will not be sufficiently high to initiate actuation of fire protection systems. If lower fire protection system setpoints are used or temperatures are expected to exceed these temperatures during a station blackout, procedures should identify actions to avoid such inadvertent actuations.

---

(11) Plant procedures should consider habitability requirements at locations where operators will be required to perform manual operations.

Due to elevated temperatures in some locations where manual valve actions are required, procedures should identify the protective clothing or other equipment or actions necessary to protect the operator from high temperatures on valve handwheels or other control equipment as appropriate. Control room habitability is discussed in Section 2.7.2.

---

(12) *Non-Class 1E equipment relied upon to cope for the required station blackout duration should be addressed in a maintenance program.*

Typical maintenance programs for non-Class 1E equipment consider vendor recommendations or other industry programs for maintenance and surveillance activities as well as procurement for spare parts. Such programs provide assurance of the application of appropriate quality standards providing an acceptable confidence in the availability of equipment.

---

(13) *Plant procedures should consider loss of heat tracing effects for equipment required to cope with a station blackout. Alternate steps, if needed, should be identified to supplement planned action.*

Heat tracing is used at some plants to ensure cold weather conditions do not result in freezing important piping and instrumentation systems with small diameter piping. Procedures should be reviewed to identify if any heat traced systems are relied upon to cope with a station blackout. For example, additional condensate makeup may be supplied from a system exposed to cold weather where heat tracing is needed to ensure control systems are available. If any such systems are identified, additional backup sources of water not dependent on heat tracing should be identified. Control room habitability is discussed in Section 2.7.

4.3.2 AC Power Restoration Guidelines

This section provides the bases and related supplemental information for the AC-power restoration procedure guidelines of Section 4.2.2.

---

(1) *Load dispatchers should give the highest possible priority to restoring power to nuclear units. Procedures and training should consider several potential methods of transmitting power from blackstart capable units to the nuclear plant.*

During a complete loss of AC power, other power stations may be affected by the initiating event. Grid load dispatchers should give high priority to locating alternate transmission sources in order to restore power to the affected nuclear unit.

---

(2) Should incoming transmission lines to a nuclear power plant be damaged, high priority should be assigned to repair and restoration activities to at least one line capable of feeding shutdown equipment.

Multiple incoming transmission lines to a plant switchyard exist at most nuclear utilities. However, it is not necessary to restore all lines in order to feed the necessary shutdown equipment. Transmission line repair should be prioritized in such a way as to ensure that the most efficient manner of AC power restoration is achieved.

---

(3) Repair crews engaging in power restoration activities for nuclear units should be given high priority for manpower, equipment, and materials.

During severe weather conditions, repair activities will be competing for repair resources and manpower. Procedures should be implemented to ensure that repair crews are assigned on a priority basis to tasks related to power restoration to nuclear units. Manpower, equipment, and materials should also be allocated to these crews on a priority basis.

---

(4) Portable AC generators should be designated as backup sources if available and directed to nuclear power plant sites. Procedures should address pre-planned actions and identify required equipment.

The use of portable generators as backup sources of AC power, whether located on-site or locally contracted, should be considered whenever possible. Procedures should be in place to instruct plant operations personnel concerning:

(a) backup generator location and contact personnel;

(b) means of transporting portable generators from outside the plant (e.g., tractor trailer); and,

(c) location of equipment necessary to connect the backup generator to the plant's electrical system.

---

(5) Once preferred and/or standby (Class 1E) AC power becomes available, station procedures should specify the sequence of circuit breaker operations required to restore AC power to shutdown equipment. Any additional actions such as pulling or replacing fuses should also be identified.

Numerous circuit breaker trips will likely occur in the event of a loss of AC power. Plant procedures should address breaker operation sequencing to facilitate AC power restoration as well as identify any additional operator

actions such as pulling or installing fuses.

4.3.3 Severe Weather Guidelines

This section provides the bases and related supplemental information for the operating procedure guidelines of Section 4.2.3.

Actions For Hurricane

---

(1) *The plant procedures should identify site-specific actions necessary to prepare for the onset of a hurricane. These actions should be initiated when a hurricane warning is issued for the plant site area.*

The likelihood of core damage due to station blackout can be significantly reduced by taking actions in anticipation of a hurricane. The National Weather Service issues hurricane warnings approximately 24 hours prior to the expected onset of hurricane conditions. This provides ample time for verifying the availability of the Class 1E and Alternate AC power sources and can reduce the potential for a station blackout should a hurricane cause the loss of off-site power. If diesels are load tested, procedures should specify that they should be load-tested sufficiently prior to the anticipated hurricane arrival at the site to preclude the possibility of common cause failure (resulting from potential storm effects) involving the diesel generator and the preferred power supply. Similarly, enhancing the ability to respond to a station blackout event can further reduce the likelihood of core damage. Note that if the EDGs are load-tested within a few hours of the expected hurricane arrival at the site, they should not be run in parallel with offsite power to preclude the potential for EDG loss upon the LOOP.

---

(2) *Utility procedures should identify additional plant staff to be recalled in order to support the present staff and the means to contact them once a hurricane notice has been issued by the National Weather Service.*

The normal plant operations staff may not be adequate to deal with the added activities necessary to mitigate the effects of a hurricane. Utility procedures should be responsive to the need to recall additional personnel.

---

(3) *Plant procedures should specify actions necessary to ensure equipment required for a possible station blackout is available.*

With the onset of a severe weather conditions, the potential for a LOOP increases. It is, therefore, necessary to verify the availability and operability of equipment necessary for responding to a station blackout. Any equipment

testing in progress should be completed as soon as practical and no unnecessary testing (i.e., testing not associated with surveillance requirements) started until the severe weather warning has been lifted.

Equipment important to station blackout response should include but not be limited to:

(a) *Emergency diesel generators* - EDGs should be kept in a warm standby condition with circulating water and lubricating oil, if possible. Pre-lubricating should also be accomplished if such means are provided.

(b) *Station batteries* - Station batteries should be checked to verify they are charged.

(c) *Decay Heat Removal Systems* - The status of systems supplied from DC or emergency AC power should be determined and appropriate actions should be taken to ensure the availability of such systems.

---

(4) Plant procedures should address the following items prior to a hurricane arrival at a site:

(a) the site-specific indicator should ensure that the plant would be in safe shutdown two hours before the anticipated hurricane arrival at the site (i.e., sustained winspeeds in excess of 73 mph);

(b) operator review of station blackout procedures; and,

(c) operator review of procedures to line up and operate the switchyard spraydown system (if installed).

The possibility of sustaining core damage from a station blackout can be greatly reduced if the plant has been placed in safe shutdown before the anticipated hurricane arrival at the site. Prior to the hurricane-induced LOOP, decay heat is removed by means of the feedwater pump supplying water to either the steam generator (PWR) or directly to the reactor (BWR) and then condensing the steam that has been subsequently generated through the main condenser.

Section 7 of these guidelines provides a methodology for determining the resources needed to cope with a 4-hour station blackout assuming the blackout occurred with the plant operating at 100% power. Removing decay heat prior to the anticipated hurricane-induced LOOP can significantly extend these resources beyond four hours. A plant that is in a coping duration category in excess of four hours because of extremely severe weather associated with a hurricane can provide risk reduction equivalent to the enhanced coping duration periods by undertaking the actions noted in Section 4.2.3 to achieve an enhanced coping capability.

The time to the anticipated hurricane arrival at the site can be estimated to allow sufficient time to extend the nominal 4-hour coping capability to the enhanced eight hour duration. For example, if an off-site power system's transmission towers are designed for hurricane force winds, the LOOP would not be expected prior to exceeding hurricane conditions. The expected arrival of hurricane conditions at a site can be estimated knowing the hurricane's location, the radius of hurricane force winds about the center, the forward speed of the storm, and its likely path. This knowledge can be used to develop a site-specific indicator.

During a station blackout it would be necessary to commence plant shutdown independent of AC power. All personnel involved in the operation of the plant should review the appropriate procedures dealing with an AC independent shutdown. Specific duties, such as manual valve and breaker operations, should be assigned to eliminate confusion or duplication of tasks.

Some utilities have installed spraydown systems to reduce salt spray accumulation on switchyard equipment during severe weather. The alignment and operation of these systems should be reviewed by the appropriate plant personnel.

Actions For Tornado

Plant procedures should identify site-specific actions necessary to prepare for the onset of a tornado. These actions should include:

(a) inspecting the site for potential missiles and reducing this potential, and

(b) expediting the restoration of important plant systems and components to service.

The warning associated with impending tornadoes may not be of sufficient duration to allow extensive actions. However, the above mentioned activities should be undertaken as a minimum as well as any additional actions that may be deemed prudent by plant personnel.

5. COLD STARTS

5.1 DISCUSSION

NUMARC Station Blackout Initiative 3 was structured to provide utility attention toward reducing, as much as possible, cold starting of emergency diesel generators during test conditions. This initiative was prompted by the NRC Staff attention to this issue in NRC Generic Letter 84-15.

For this review, a cold start is considered to be an attempt to start an emergency diesel generator from ambient conditions without the presence of pre-warmed circulating water or pre-lubrication. A continuously pre-warmed and pre-lubed machine would not be considered to have cold starts.

5.2 ACTION

Each plant should ensure that emergency diesel generator tests are performed in a pre-warmed and pre-lubed condition except during an actual demand test required approximately once each scheduled refueling outage unless the emergency diesel generator is normally pre-warmed and pre-lubricated. Plants with EDG cold starts more frequently than once each scheduled refueling outage should either reduce the cold fast start frequency or provide justification for necessary cold starts. Manufacturer recommendations, operational requirements, or regulatory requirements are examples of acceptable justifications. If more frequent testing is currently required by technical specifications, consideration should be given to applying for technical specification relief.

6. EMERGENCY AC POWER AVAILABILITY

6.1 DISCUSSION

NUMARC Station Blackout Initiative 4 calls for monitoring of plant emergency generator unavailability. Further attention to a more comprehensive diesel generator reliability program is addressed in Appendix D of this document.

6.2 ACTION

Each plant, through participation in the industry-wide Plant Performance Indicator program that is managed by INPO, provides regular reports of diesel generator unavailability data. This "other indicator" in the Plant Performance Indicator Program is trended and provided to each plant semiannually. Through this Program each plant monitors plant specific diesel generator unavailability and can compare its performance to an Industry average.

7. COPING WITH A STATION BLACKOUT EVENT

7.1 OVERVIEW

This section provides an overview of a simplified assessment procedure for coping with a station blackout. There are five steps to the procedure, addressing the following topics:

(1) Condensate inventory for decay heat removal;
(2) Assessing the Class 1E battery capacity;
(3) Compressed air;
(4) Effects of loss of ventilation; and,
(5) Containment isolation.

The procedure is structured to utilize information readily available from licensing documents (e.g., FSAR, licensing submittals), existing calculations, purchase specifications, and drawings. For most units, no additional computation or analysis is anticipated. Plant specific analysis may be relied upon as supplemental to or in lieu of the coping assessment in Section 7.2 for the topics listed above.

7.1.1 Coping Methods

For purposes of this assessment, coping methods are separated into two different approaches. The first is referred to as the "AC-Independent" approach. In this approach, plants rely on available process steam, DC power, and compressed air to operate equipment necessary to achieve safe shutdown conditions (i.e., Hot Standby or Hot Shutdown, as appropriate) until off-site or emergency AC power is restored. A second approach is called the "Alternate AC" approach. This method is named for its use of equipment that is capable of being electrically isolated from the preferred off-site and emergency on-site AC power sources. Station blackout coping using the Alternate AC power approach would entail a short period of time in an AC-Independent state (up to one hour) while the operators initiate power from the backup source. Once power is available, the plant would transition to the Alternate AC state and provide decay heat removal until off-site or emergency AC-power becomes available. The AC power sources used in the Alternate AC power approach would be subject to the Appendix B criteria including electrical isolation requirements in order to assure their availability in the event of a station blackout.

Appendix A provides a definition of Alternate AC power sources. Appendix B provides detailed acceptance criteria for an

Alternate AC power source.

7.1.2 Coping Duration

AC-Independent plants must meet the requirements of this methodology for at least four hours (or at least two hours for plants in both emergency AC group A and off-site power group P1). Plants using an Alternate AC power source must assess their ability to cope for one hour. However, if an Alternate AC power source can be shown by test to be available within 10 minutes of the onset of station blackout, then no coping assessment is required. Available within 10 minutes means that circuit breakers necessary to bring power to safe shutdown buses are capable of being actuated in the control room within that period.

7.2 COPING ASSESSMENT

7.2.1 Condensate Inventory for Decay Heat Removal

Discussion

The purpose of this procedure is to ensure that each plant has adequate condensate inventory for decay heat removal during a station blackout for the required duration.

The necessary condensate inventory is assessed by a bounding analysis. If this quantity is less than the Technical Specification minimum requirement for the condensate storage tank (CST), then the plant's current condensate inventory is adequate. If not, other sources of water that can be aligned and transferred under station blackout conditions are identified and considered.

Procedure

Step 1: Plant Rating

Record in A the unit's licensed reactor output in megawatts thermal (Mwt) from the unit's operating license. A = ________

Step 2: Required Condensate

Determine the number of gallons of water required for decay heat removal as follows: B = ________

$$B = A^*(22.12\ GAL/MWt) + C$$

If emergency operating procedures do not require a primary system cooldown to minimize reactor coolant pump leakage or to maintain decay heat removal capability, then C = zero.

If emergency operating procedures require a primary system cooldown to minimize reactor coolant pump seal leakage or to maintain decay heat removal capability, then C is the amount of water required to support the cooldown.

Record the result as B.

Step 3: Technical Specification for CST Volume D = ________

Obtain the minimum permissible usable gallons of water in the CST as found in the unit's Technical Specifications. Record this value as D.

Step 4: Review for Adequacy — CST Quantities Alone

Compare the value of B with the value of D.

(a) If B is less than D, adequate condensate is available

(b) Otherwise, continue to Step 5.

Step 5: Additional Water Sources

In this step, additional water sources are identified as backup condensate makeup sources for decay heat removal. The following are examples of sources of water which may serve in this role:

Hotwell
Adjacent unit water sources
Fire water tanks
Cooling water pond

River or lake water

NOTE: **Plant procedures need to reflect actions and water sources relied upon in responding to a station blackout event.**

The following criteria must be met prior to assuming the availability of any backup water sources:

(a) A physical connection and transfer capability is provided independent of the preferred power supply and blacked-out unit's Class 1E AC power sources and capable of providing a source of water to the CST or the makeup pump suction.

(b) Plant procedures must exist to accomplish this makeup to the CST.

(c) The source must be be able to be connected before the CST is emptied.

(d) After one hour, an AAC source may be used to provide power to pumps and valves if the equipment is powered from the AAC source.

NOTE: **Water relied on from adjacent units at a multiunit site must be capable of being transferred to the blacked-out unit without adversely affecting adequate decay heat removal activities at the non-blacked-out unit.**

Record below the usable volume of each additional source of water satisfying Criteria (a)-(d), above.

Source 1:________________ Amt. Water (gal.)__________

Source 2:________________ Amt. Water (gal.)__________

Source 3:________________ Amt. Water (gal.)__________

Source 4:________________ Amt. Water (gal.)__________

Total the amount of water from Sources 1 to 4 and record this amount after E. E = ________

Step 6: **Condensate Available**

Sum the values of D and E. Record the result as F. F = ________

(i.e. F=D+E)

Step 7: **Test for Adequacy — With Backup Sources**

Compare the value of F to the value of B

(a) **If B is less than F, adequate condensate is available.**

(b) **If B is greater than F, return to Step 5 and identify additional water sources.**

Supporting Information

This section provides the analytical basis for the thermal normalized condensate requirement presented in the condensate inventory procedure. The analysis determines the amount of water necessary to remove decay heat for a given duration. The amount of water is then normalized with respect to the thermal rating of the the reactor to obtain the thermal normalized condensate requirement.

Analysis

Figure 7-1 represents a PWR steam generator or BWR reactor vessel with decay heat (Q'), an inlet mass flow rate (m_i') from the condensate storage tank (CST), and exit mass flow rate (m_e').

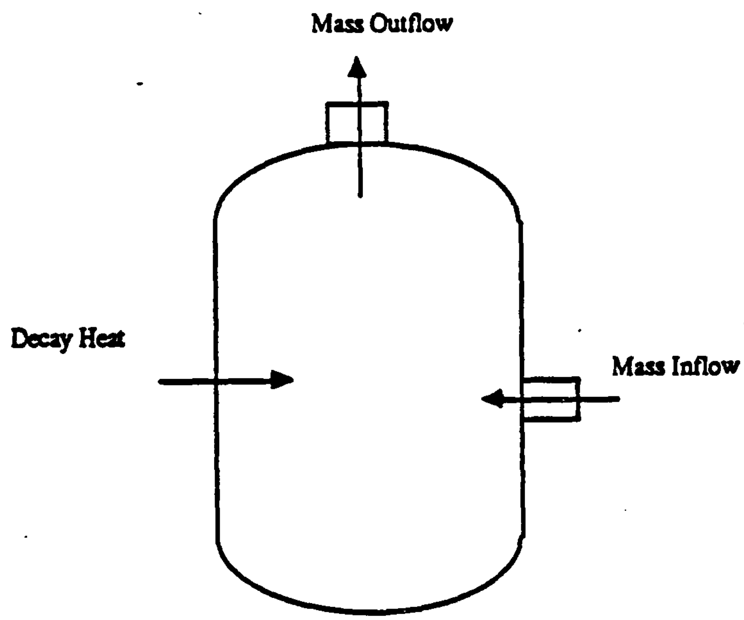


Figure 7-1

Assuming steady state and adiabatic conditions, the equations of mass and energy conservation which describe this system are as follows:

$$m_i' = m_e' = m'$$

$$Q' + m_i'h_i - m_e'h_e = 0$$

These equations can be used to derive an expression for the rate of change of mass in the CST (m_{CST}):

$$Q' = m'(h_e - h_i) = -dm_{CST}/dt\,(h_e - h_i)$$

The amount of condensate necessary to remove decay for a given duration (T) is then given by:

$$m_{CST}(0) = \int_0^T Q'/(h_e - h_i)\,dt$$

Results

The right hand side of this equation is evaluated using the following assumptions:

- the reactor has been operating at full power for 100 days
- decay heat is calculated using ANS standard ANS-5.1/N18.6
- there are no stuck open PORVs
- the inlet enthalpy corresponds to the enthalpy of saturated liquid at atmospheric pressure
- the exit enthalpy corresponds to the enthalpy of saturated vapor -- approximately 1200 Btu/lbm for pressures between 100 and 1200 psia

A fourth order Runge-Kutta scheme was used to integrate this equation.

7.2.2 Assessing the Class 1E Battery Capacity

Discussion

The purpose of this section is to ensure that each plant has adequate battery capacity to support decay heat removal during a station blackout for the required coping duration.

This procedure offers two analytical methods that can be used to ensure sufficient capacity exits at each unit. IEEE-STD-485, or other design basis battery analysis updated as necessary to reflect current loads, should be used. The two alternatives are outlined below.

(a) Use an existing battery capacity calculation or perform one that verifies sufficient coping capacity under station blackout conditions.

(b) Use an existing battery capacity calculation or perform one that verifies sufficient coping capacity by stripping loads in order to extend the battery life in a station blackout.

NOTE: All calculations should use the lowest electrolyte temperature anticipated under normal operating conditions.

Procedure

Battery Capacity Calculation --- No Load Stripping

Step 1: **Review for Battery Adequacy**

Review an existing battery capacity calculation for a station blackout event or perform one for either (a) or (b) below:

(a) AC-Independent: four hours.

(b) Alternate AC: one hour.

NOTE: If an existing battery exceeds the above capacity, the rated capacity should not be reduced solely on the basis of the above station blackout criteria.

Step 2: **Review for Adequacy - Without Load Stripping**

Compare results of Step 1 to the battery manufacturers capacity curves. If sufficient capacity exists, no further action is required. Otherwise, go to the Load Stripping Case.

Battery Capacity Calculation --- With Load Stripping

Step 1: **List DC Loads to Be Stripped**

List loads on the Class 1E batteries that are not required to cope with a station blackout and can be stripped commencing 30 minutes after the initiation of the station blackout event:

____

____

____

NOTE: Loads listed above to be stripped must be based on actions reflected in plant procedures and which can be accomplished under station blackout conditions.

Step 2: **Adjust Duty Cycle Curves**

Delete the loads listed in Step 1 from the unstripped load duty cycle curves in the battery capacity calculation. Recalculate the maximum section size and follow the steps used in the calculation to assess Class 1E battery capacity.

Step 3: Review for Adequacy --- With Load Stripping

Compare results of Step 2 to the battery manufacturer's capacity curves. If sufficient battery capacity exists with load shedding, no further action is required.

Otherwise, battery capacity (IEEE-STD-485, or equivalent) must be extended further to meet the required station blackout coping duration. Acceptable means for extending battery capacity include the addition of batteries or the addition of a battery charging system for the existing batteries provided the source of power for the charging system is independent of both the preferred power source and the blacked out unit's Class 1E power system. Assure that the required additional capacity is achieved.

Supporting Information

The total DC power requirements for a four hour station blackout (one hour for AAC plants) depend on the required loads, their duration of operation, and the capacity of the batteries to hold a charge. The batteries' capacity varies with the rate of discharge, which also varies with the loads. Consequently, the amount of energy recoverable from the batteries depends to a large measure on the rate of discharge associated with the station blackout response loads and initial electrolyte temperature. The battery's ability to discharge stored energy is defined in a series of battery curves provided by the manufacturer. Capacity curves are generally provided for discharge periods ranging from five minutes to upwards of 16 hours.

The capacity of storage batteries varies with electrolyte temperature. This temperature depends on room temperature which may vary in certain circumstances with the season of the year. Calculations should be performed assuming the lowest temperature normally expected for the battery.

The station blackout loads can be estimated from design basis accident loads since they are generally a subset of these loads. They may be classified as being one of three categories: (1) continuous, (2) discontinuous, and (3) momentary loads. Continuous loads are required for the duration of the station blackout event. Inverters and annunciators required for instrumentation and control are common examples of continuous loads. Discontinuous loads are required for short durations throughout the event. Examples of these loads include motor-operated valves and loads necessary to support circuit breaker operations. Momentary loads are of a temporary nature and are required only once or for a limited number of cycles. EDG field flashing is an example of a momentary load that should be considered when determining battery loads.

Knowing the magnitude and timing of loads, it is possible to use the battery capacity curves provided for each plant to determine whether sufficient capacity is available for a four hour station blackout (one hour for AAC plants).

The DC power requirements for a required station blackout may be estimated using the same methodology for which the plant is licensed. The generally accepted methodology is *IEEE Recommended Practice for Sizing Large Lead Storage Batteries for Generating Stations and Substations* (IEEE-STD-485). IEEE-STD-485 incorporates design margins for aging and temperature correction that are addressed in various other industry standards such as *IEEE Recommended Practice for Maintenance, Testing, and Replacement of Large Stationary Type Power Plant and Substation Lead Storage Batteries* (IEEE-STD-450). This methodology calculates battery load requirements for various sections of time. The magnitude of DC loads for each such section of time is referred to as the section size. Various section sizes are calculated in order to construct a battery duty cycle. The battery is then sized to address the maximum section calculated for the entire duty cycle.

7.2.3 Compressed Air

Discussion

The purpose of this section is to ensure that air operated valves required for decay heat removal have sufficient reserve air or can be manually operated under station blackout conditions for the specified duration.

The loss of instrument air in a station blackout can be minimized through a strategy for operator actions. This procedure provides the requisite information for developing that strategy.

Procedure

Step 1: Identify Air-Operated Valves Necessary for Decay Heat Removal

List below all air-operated valves that are required to be cycled during a station blackout.

____

____

____

____

____

____

____

____

____

____

Step 2: **Backup Air Supplies**

Review each valve listed and identify the valves that are not supplied with air within design pressure and moisture limits from at least one of the following sources for at least four hours (one hour for AAC plants if the compressor is on the AAC supply, otherwise four hours).

(a) Backup air systems (compressors with associated valves, instruments etc.) that are:

- supplied from the adjacent unit (if it is independent of the preferred and blacked out unit's Class 1E power supply), or
- powered from Alternate AC power sources, or
- powered by DC power.

(b) Backup local sources of compressed air or nitrogen located at the valves.

Step 3: **Criteria for Manual Operation**

For all valves not supplied by backup sources, determine whether they meet all of the following criteria for manual operation:

(a) Procedures specify manual operation for valves in a station blackout;
(b) Accessible in a station blackout;
(c) Identifiable in a station blackout;
(d) Necessary tools, reachrods, or chains are normally present;
(e) Appropriate indication and means for communication are provided; and,
(f) Sufficient manpower is available onshift to accomplish specified tasks

Step 4: Review for Adequacy --- Manual Valve Operation

If all air-operated valves required for decay heat removal have backup sources or may be manually operated in a station blackout, no further action is required.

Otherwise, return to Step 2.

NOTE: Plant procedures need to reflect the manual actions and backup air supplies assumed to be relied upon in responding to a station blackout event.

Supporting Information

With the initiation of a station blackout, instrument air systems lose their air compressors and begin to depressurize. As the air headers bleed down, operability of the air-operated valves also degrade ultimately resulting in their unavailability. With prolonged loss of instrument air systems, it is possible that decay heat removal and reactor coolant inventory may be adversely affected.

The amount of air needed for decay heat removal depends on the expected number of valve cycles, the failure mode for air operated valves on the reactor coolant system boundary, and the ability to manually cycle or close air operated valves. Atmospheric dump valves on PWRs generally require air for prolonged operation. In contrast, most other valves, such as feedwater regulator valves, generally fail in the "as-is" position. Similarly, reactor coolant pressure boundary valves generally fail as-is, or closed, in order to limit reactor coolant inventory loss. Valves failing in such a manner do not normally require repositioning in a station blackout.

7.2.4 EFFECTS OF LOSS OF VENTILATION

Discussion

The purpose of this section is to determine the average steady state temperature in dominant areas containing equipment necessary to achieve and maintain safe shutdown during a station blackout. Appendix E provides the basis for the procedure contained in this section. This temperature provides a reference point for reasonably assuring the operability of equipment needed to cope with a station blackout using the methodologies outlined in Appendix F.

Plants utilizing an Alternate AC capability need not complete this review if the Alternate AC source is used to power ESF ventilation systems and is available within 10 minutes (see Section 7.1.2).

Procedure

Step 1: **Dominant Area Geometry**

Record in A(1) , A(2), A(3), and A(4), as appropriate, the estimated total room surface area, excluding floors but including ceilings and walls, ***measured in square meters***, for the following rooms/quadrants (as applicable):

(1) Steam Driven AFW Pump Room (PWRs only)
A(1) = ________

(2) HPCI/HPCS Room (BWRs only)
A(2)= ________

(3) RCIC Room (BWRs only)
A(3)= ________

(4) Main steam tunnel (BWRs only)
A(4) = ________

Step 2: **Dominant Area Heat Generation Rates**

Record in Q(1), Q(2), Q(3), and Q(4), as appropriate, the heat generation rates, ***measured in Watts***, for the following rooms/quadrants (as applicable):

(1) Steam Driven AFW Pump Room (PWRs only)
Estimate the heat generation rate for this room/quadrant and enter in Q(1)
Q(1)= ____________

(2) HPCI/HPCS Room (BWRs only)
Estimate the heat generation rate for this room/quadrant and enter in Q(2)
Q(2)= ____________

(3) RCIC Room (BWRs only)

Estimate the heat generation rate for this room/quadrant and enter in Q(3)

Q(3)=__________

(4) Main steam tunnel (BWRs only)

Estimate the heat generation rate for the tunnel and enter in Q(4). The heat transfer correlation presented in Appendix E is adequate to estimate the heat transfer from hot steam pipes to the surrounding air. Thermal radiation heat transfer may be neglected.

Q(4) = __________

NOTE: **See supporting information for the methodology used to determine the generation rates for various equipment configurations.**

Step 3: **Determine the Wall Temperatures:**

Determine the upper bound for wall temperature, *in* °C, prior to loss of ventilation. A temperature of 40° C (104° F) may be reasonable for nearly all rooms/quadrants. This temperature is later used as the initial air temperature for the room/quadrant in Steps 4 and 5. A different temperature may be used if it can be justified based on actual measurement. It is assumed that the wall temperature does not change appreciably throughout the transient as shown in Appendix E (2.5° C or 4.5° F).

(1) Steam Driven AFW Pump Room (PWRs only)

T(1) = __________

(2) HPCI/HPCS Room (BWRs only)

T(2)=__________

(3) RCIC Room (BWRs only)

T(3)=__________

(4) Main steam tunnel (BWRs only)

$T(4) =$ ________

Step 4: Calculate the Steady State Room Temperature Following Loss of Ventilation

Calculate the steady state ambient air temperature, *in* °C, using Equation (E-18) for the following rooms/quadrants, assuming no additional cooling or natural circulation to the outside environment:

(1) Steam Driven AFW Pump Room (PWRs only)

$T_f(1) = [Q(1)/A(1)]^{(3/4)} + T(1) =$ ________

(2) HPCI/HPCS Room (BWRs only)

$T_f(2) = [Q(2)/A(2)]^{(3/4)} + T(2) =$ ________

(3) RCIC Room (BWRs only)

$T_f(3) = [Q(3)/A(3)]^{(3/4)} + T(3) =$ ________

(4) Main steam tunnel (BWRs only)

$T_f(4) = [Q(4)/A(4)]^{(3/4)} + T(4) =$ ________

Note that this equation is a simplified form of the complete steady state solution. Heat transfer coefficients and thermal properties have been evaluated in MKS units. Therefore, this dimensionally inconsistent equation is valid only with the units specified in Steps 1, 2, and 3.

Step 5: Calculate the Effect of Opening Area Doors

If it is feasible to open a door during the event to allow removal of heat through natural circulation, perform the following steps to determine the effect of opening the door.

5.1 Record in H(1), H(2), H(3), and H(4), as appropriate, the height of the door, *measured in meters.*

(1) Steam Driven AFW Pump Room (PWRs only)
H(1) = ________

(2) HPCI/HPCS Room (BWRs only)
H(2)= ________

(3) RCIC Room (BWRs only)
H(3)= ________

(4) Main steam tunnel (BWRs only)
H(4) = ________

5.2 Record in W(1), W(2), W(3), and W(4), as appropriate , the width of the door *measured in meters.*

(1) Steam Driven AFW Pump Room (PWRs only)
W(1) = ________

(2) HPCI/HPCS Room (BWRs only)
W(2)= ________

(3) RCIC Room (BWRs only)
W(3)= ________

(4) Main steam tunnel (BWRs only)
W(4) = ________

5.3 Calculate the door factor F for the following rooms/ quadrants:

(1) Steam Driven AFW Pump Room (PWRs only)
$F(1) = H(1)^{3/2}W(1)$

(2) HPCI/HPCS Room (BWRs only)

$F(2) = H(2)^{3/2}W(2)$

(3) RCIC Room (BWRs only)

$F(3) = H(3)^{3/2}W(3)$

(4) Main steam tunnel (BWRs only)

$F(4) = H(4)^{3/2}W(4)$

5.4 Calculate the ***steady-state ambient air temperature, in °C,*** using Equation (E-27) for the following rooms/quadrants:

(1) Steam Driven AFW Pump Room (PWRs only)

$$T_f(1) = 4 + T(1) + \left[Q(1)^{3/4}/[A(1)^{3/4} + 16.18F(1)^{0.8653}]\right]$$

(2) HPCI/HPCS Room (BWRs only)

$$T_f(2) = 4 + T(2) + \left[Q(2)^{3/4}/[A(2)^{3/4} + 16.18F(2)^{0.8653}]\right]$$

(3) RCIC Room (BWRs only)

$$T_f(3) = 4 + T(3) + \left[Q(3)^{3/4}/[A(3)^{3/4} + 16.18F(3)^{0.8653}]\right]$$

(4) Main steam tunnel (BWRs only)

$$T_f(4) = 4 + T(4) + \left[Q(4)^{3/4}/[A(4)^{3/4} + 16.18F(4)^{0.8653}]\right]$$

Note that this equation is a simplified form of the complete steady state solution. Heat transfer coefficients and thermal properties have been evaluated in MKS units. Therefore, this dimensionally inconsistent equation is valid only with the units specified in Steps 1, 2, and 3.

Step 6: Reasonable Assurance of Equipment Operability

Use the methodologies in Appendix F to provide a basis that the equipment relied upon to cope with a station blackout will operate at the steady-state temperatures determined in Steps 4 or 5 for the required duration.

Supporting Information

General Discussion

Since station blackout is not considered to be a design basis event, reasonable assurance of equipment operability need not be provided to the same level of precision and detail required by 10 CFR §50.49 for safety related equipment located in harsh environments. For this reason, a representative analysis approach is provided, with attention concentrated on the few situations where equipment operability is especially important to core cooling in a station blackout.

This procedure provides the results of representative analyses for such areas to be reviewed against. Plants that do not conform with the acceptance envelopes for thermal conditions may either perform plant specific analysis, provide additional assurance that equipment survivability can be assured, or provide alternative means of cooling in a station blackout.

The representative analysis provided in this section addresses a limited set of plant areas deemed to be potentially susceptible to heatup upon loss of ventilation, such as would occur in a station blackout. These areas are defined by three factors: (1) their containing equipment normally required to function early in a station blackout to remove decay heat, (2) the presence of significant heat generation terms (after AC power is lost) relative to their free volume (i.e., process steam or DC electrical power supplies in small rooms or enclosures), and (3) the absence of heat removal capability in a station blackout without operator action. These areas and their respective equipment consist of:

(1)	HPCI/HPCS and RCIC rooms (BWR only)	-	decay heat removal equipment
(2)	Steam Driven AFW pump rooms (PWR only)	-	decay heat removal equipment
(3)	Main steam tunnel (BWR only)	-	high temperature cutout for decay heat removal equipment

Areas not addressed in this list are viewed as posing a significantly reduced concern for a variety of reasons. Safe shutdown equipment in many plant areas is already qualified to operate in a harsh environment. The containment is one such harsh environment area. The station blackout event is expected to be bounded by analyses previously performed for these areas. Other plant areas will not be exposed to significant heat generation terms since: (1) a station blackout

results in the elimination of process steam from most plant areas, or (2) these areas do not contain equipment required for decay heat removal. In addition, the loss of AC power will eliminate AC motors, switchgear, and lighting from the list of heat generation sources. Finally, the equipment needed to function in a station blackout is limited to a turbine driven feedwater makeup system, atmospheric dump or steam relief valves, batteries, and a small set of instrumentation cabinets.

The loss of ventilation concern is significantly reduced for plants using Alternate AC sources, provided these plants also ensure that sufficient forced ventilation is available to safe shutdown equipment. For these plants, the period of ventilation loss would be limited to the time necessary to restore AC power from an Alternate AC source. This period is no greater than 1 hour, and during this time loss of ventilation is not anticipated to cause equipment problems.

Methodology for Determining Heat Generation Rates

To determine heat generation rates, it is necessary to evaluate electrical and steam equipment.

(1) Electrical Equipment - identify the nameplate rating of the equipment; convert this rating to Watts, for example:

(Nameplate in Horsepower) X (745.7) = watts

(2) Steam-driven Equipment - use standard formula to determine heat generation rates for the applicable configuration, for example:

Pipes

$$Q=\{0.1[0.4 + 15.7(T_s-T_{air})^{1/6} D^{1/2} + 170.3(T_s-T_{air})^{1/3}D](T_s-T_{air}) + 1.4E\text{-}7D(T^4_s - T^4_w)\}L$$

where

Q = the heat generation rate of the pipe in watts

D = the diameter of the pipe in meters

T_s = the surface temperature of the pipe in °K

T_{air} = the air temperature of the room at station blackout onset in°K

L = the length of the pipe in meters

Pumps

$$Q = 0.1(2 + 37.0(T_s - T_{air})^{1/4} D^{3/4}) D(T_s - T_{air}) + 1.4E\text{-}7\, D^2 (T_s^4 - T_{air}^4)$$

where

Q = the heat generation rate of the pump in watts

D = the equivalent diameter of the pump in meters

T_s = the surface temperature of the pump in °K

T_{air} = the air temperature of the room at station blackout onset in °K

Note that this equation models a pumps as a sphere. The equivalent diameter of the pump is determined by using the volume of space that is occupied by the pump to calculate the equivalent diameter of a sphere. This is accomplished by the following relationship:

$$D = (6 * V/\pi)^{1/3}$$

where

V = volume occupied by the pump in meters cubed

π = 3.1415927

7.2.5 Containment Isolation

Discussion

The purpose of this procedure is to ensure that appropriate containment integrity can be provided during a station blackout event for the required duration.

Appropriate containment integrity is defined such that the capability for valve position indication and closure of certain containment isolation valves is provided independent of the preferred or Class 1E power supplies. The containment isolation valves requiring this capability are valves that may be in the open position at the onset of a station blackout. Acceptable means of position indication includes local mechanical indication, DC-powered indication (including AC-powered indicators powered through inverters), and Alternate AC-powered indication. Acceptable means of closure include manual operation, air-operation (including air-operated valves that are mechanically closed on loss of air), DC-powered operation, and Alternate AC-powered operation.

Procedure

Step 1: Valve Identification

Review the list of containment isolation valves and exclude the following valves from consideration:

(1) valves normally locked closed during operation;
(2) valves that fail closed on loss of AC power or air;
(3) check valves;
(4) valves in non-radioactive closed-loop systems not expected to be breached in a station blackout (with the exception of lines that communicate directly with the containment atmosphere); and,
(5) all valves less than 3-inch nominal diameter.

The remaining valves are the containment isolation valves of concern.

Step 2: Containment Isolation Valves Requiring Manual Operation

List valves from Step 1 that are of concern and which need to be operated to cope with a station blackout event for the required duration (i.e., 2 or 4-hours).

Ensure that these valves can be operated independent of the preferred and Class 1E power supplies and have valve position indication (e.g., local mechanical, DC powered, or Alternate AC powered) that is independent of the preferred and blacked-out unit's Class 1E power supplies.

Step 3: Containment Isolation Valves Requiring Closure Capability

List valves from Step 1 not identified in Step 2. Ensure that these valves can be closed independent of the preferred and Class 1E power supplies and have valve position indication (e.g., local mechanical, DC powered, or Alternate AC powered) that is independent of the preferred and blacked-out unit's Class 1E power supplies.

APPENDIX A. DEFINITIONS

Terms defined below were specifically developed for these guidelines and are of special importance to its use.

ALTERNATE AC POWER SOURCE - An alternating current (AC) power source that is available to and located at or nearby a nuclear power plant and meets the following requirements:

- (i) is connectable to but not normally connected to the preferred or on-site emergency AC power systems;
- (ii) has minimal potential for common cause failure with off-site power or the on-site AC power sources;
- (iii) is available in a timely manner after the onset of station blackout;
- (iv) has sufficient capacity and reliability for operation of all systems necessary for coping with a station blackout and for the time required to bring and maintain the plant in safe shutdown (Hot Shutdown or Hot Standby, as appropriate); and,
- (v) is inspected, maintained, and tested periodically to demonstrate operability and reliability as set forth in Appendix B.

PREFERRED POWER SUPPLY - that power supply from the transmission system to the Class 1E distribution system which is preferred to furnish electric energy under accident or post-accident conditions. *IEEE-STD-765; IEEE-STD-308; and NUREG/CR-3992, page 2.*

REQUIRED COPING DURATION - the time between the onset of station blackout and the restoration of off-site AC power to safe shutdown buses.

SAFE SHUTDOWN - For the purpose of this procedure safe shutdown is the plant conditions defined in plant technical specifications as Hot Standby or Hot Shutdown, as appropriate (plants have the option of maintaining the RCS at normal operating temperatures or at reduced temperatures).

SEVERE WEATHER - the occurrence of annual average snowfall, tornado of F2 severity or greater, hurricane with salt spray potential, and wind speeds in excess of 75 mph. *NUREG-1032.*

STANDBY POWER SUPPLY - the Class 1E power supply that is selected to furnish electric energy to shutdown equipment when the preferred power supply is not available. *Based on IEEE-STD-308.*

STATION BLACKOUT - means the complete loss of alternating current (AC) electric power to the essential and nonessential switchgear buses in a nuclear power plant (i.e., loss of off-site electric power system concurrent with turbine trip and unavailability of on-site emergency AC power system). Station Blackout does not include the loss of available AC power to buses fed by station batteries through inverters or by Alternate AC power sources as defined in this appendix, nor does it assume a concurrent single failure or a design basis accident. At a multi-unit site with normally dedicated emergency AC power sources, station blackout is assumed to occur in only one unit.

At single unit sites, any emergency AC power source(s) in excess of the number required to meet the minimum redundancy requirements (i.e. single failure) for safe shutdown is assumed to be available and may be designated as an Alternate AC Power Source(s) provided it meets the Alternate AC power criteria in Appendix B.

At multi-unit sites with normally shared emergency AC power sources, where the combination of emergency AC sources exceeds the minimum redundancy requirements for normal safe shutdown (non-DBA) of all units, the remaining emergency AC power sources may be used as alternative AC power sources provided they meet the alternate AC power criteria in Appendix B. If there are no remaining emergency AC power sources in excess of the minimum redundancy requirements, station blackout must be assumed to occur at all the units.

APPENDIX B. ALTERNATE AC POWER CRITERIA

This appendix describes the criteria that must be met by a power supply in order to be classified as an Alternate AC power source. The criteria focus on ensuring that station blackout equipment is not unduly susceptible to dependent failure by establishing independence of the AAC system from the emergency and non-Class 1E AC power systems.

AAC Power Source Criteria

B.1 The AAC system and its components need not be designed to meet Class 1E or safety system requirements. If a Class 1E EDG is used as an Alternate AC power source, this existing Class 1E EDG must continue to meet all applicable safety-related criteria.

B.2 Unless otherwise provided in this criteria, the AAC system need not be protected against the effects of:

(1) failure or misoperation of mechanical equipment, including (i) fire, (ii) pipe whip, (iii) jet impingement, (iv) water spray, (v) flooding from a pipe break, (vi) radiation, pressurization, elevated temperature or humidity caused by high or medium energy pipe break, and (vii) missiles resulting from the failure of rotating equipment or high energy systems; or

(2) seismic events.

B.3 Components and subsystems shall be protected against the effects of likely weather-related events that may initiate the loss of off-site power event. Protection may be provided by enclosing AAC components within structures that conform with the Uniform Building Code, and burying exposed electrical cable run between buildings (i.e., connections between the AAC power source and the shutdown busses).

B.4 Physical separation of AAC components from safety related components or equipment shall conform with the separation criteria applicable for the unit's licensing basis.

Connectability to AC Power Systems

B.5 Failure of AAC components shall not adversely affect Class 1E AC power systems.

B.6 Electrical isolation of AAC power shall be provided through an appropriate isolation device. If the AAC source is connected to Class 1E buses, isolation shall be provided by two circuit breakers in series (one Class 1E breaker at

the Class 1E bus and one non-Class 1E breaker to protect the source).

B.7 The AAC power source shall not normally be directly connected to the preferred or on-site emergency AC power system for the unit affected by the blackout. In addition, the AAC system shall not be capable of automatic loading of shutdown equipment from the blacked-out unit unless licensed with such capability.

Minimal Potential for Common Cause Failure

B.8 There shall be minimal potential for common cause failure of the AAC power source(s). The following system features provide assurance that the minimal potential for common cause failure has been adequately addressed.

(a) The AAC power system shall be equipped with a DC power source that is electrically independent from the blacked-out unit's preferred and Class 1E power system.

(b) The AAC power system shall be equipped with an air start system, as applicable, that is independent of the preferred and the blacked-out unit's preferred and Class 1E power supply.

(c) The AAC power system shall be provided with a fuel oil supply, as applicable, that is separate from the fuel oil supply for the onsite emergency AC power system. A separate day tank supplied from a common storage tank is acceptable provided the fuel oil is sampled and analyzed consistent with applicable standards prior to transfer to the day tank.

(d) If the AAC power source is an identical machine to the emergency onsite AC power source, active failures of the emergency AC power source shall be evaluated for applicability and corrective action taken to reduce subsequent failures.

(e) No single point vulnerability shall exist whereby a likely weather-related event or single active failure could disable any portion of the onsite emergency AC power sources or the preferred power sources, and simultaneously fail the AAC power source(s).

(f) The AAC power system shall be capable of operating during and after a station blackout without any support systems powered from the preferred power supply, or the blacked-out unit's Class 1E power sources affected by the event.

(g) The portions of the AAC power system subjected to maintenance activities shall be tested prior to returning the AAC power system to service.

Availability After Onset of Station Blackout

B.9 The AAC power system shall be sized to carry the required shutdown loads for the required coping duration determined in Section 3.2.5, and be capable of maintaining voltage and frequency within limits consistent with established industry standards that will not degrade the performance of any shutdown system or component. At a multi-unit site, except for 1/2 Shared or 2/3 emergency AC power configurations, an adjacent unit's Class 1E power source may be used as an AAC power source for the blacked-out unit if it is capable of powering the required loads at both units.

Capacity and Reliability

B.10 Unless otherwise governed by technical specifications, the AAC power source shall be started and brought to operating conditions that are consistent with its function as an AAC source at intervals not longer than three months, following manufacturer's recommendations or in accordance with plant-developed procedures. Once every refueling outage, a timed start (within the time period specified under blackout conditions) and rated load capacity test shall be performed.

B.11 Unless otherwise governed by technical specifications, surveillance and maintenance procedures for the AAC system shall be implemented considering manufacturer's recommendations or in accordance with plant-developed procedures.

B.12 Unless otherwise governed by technical specifications, the AAC system shall be demonstrated by initial test to be capable of powering required shutdown equipment within one hour of a station blackout event.

B.13 The Non-Class 1E AAC system should attempt to meet the target reliability and availability goals specified below, depending on normal system state. In this context, reliability and availability goals apply to the overall AAC system rather than individual machines, where a system may comprise more than one AAC power source.

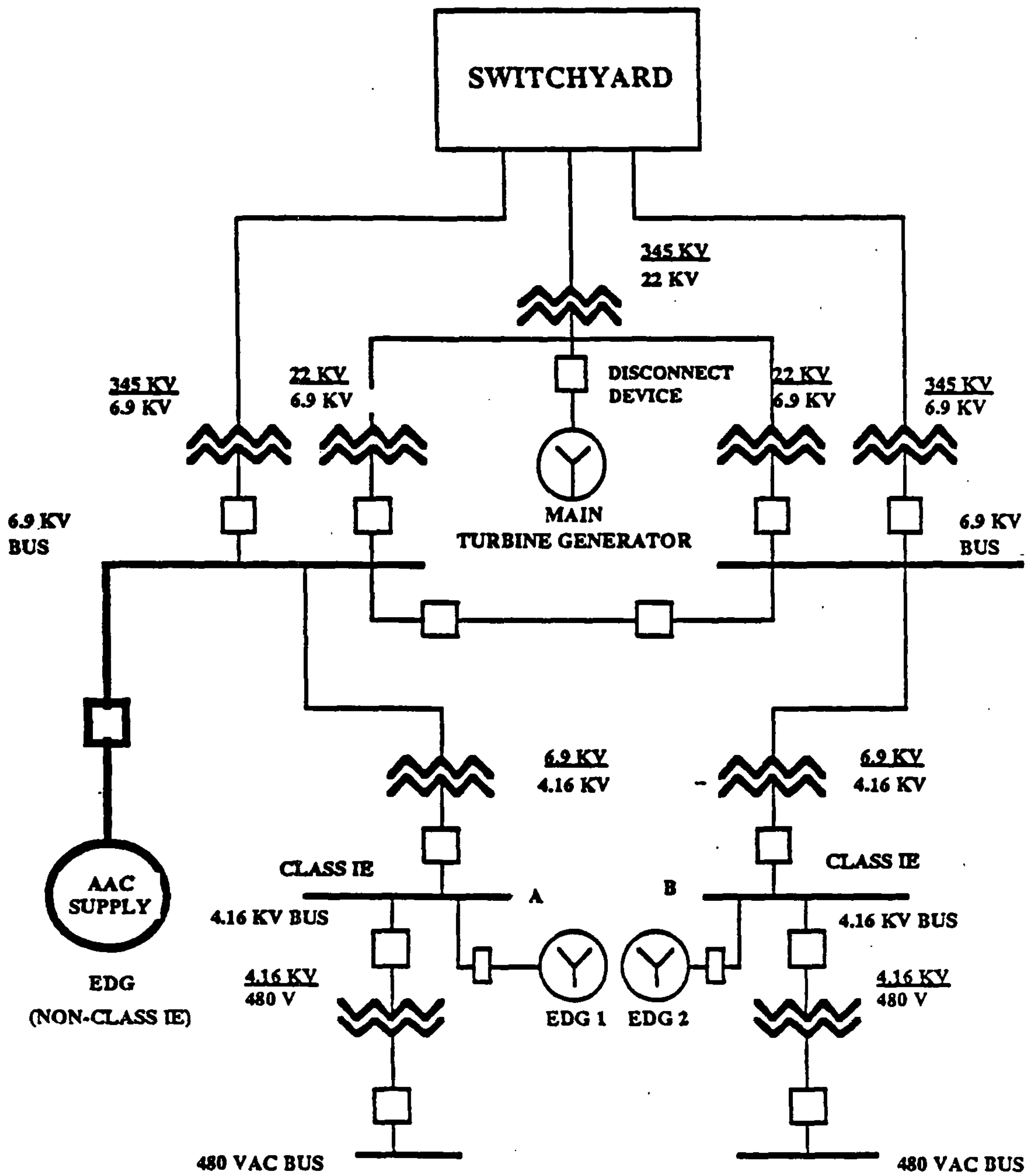
(a) Systems Not Normally Operated (Standby Systems)
System reliability should be maintained at or above 0.95 per demand, as determined in accordance with NSAC-108 methodology (or equivalent).

(b) Systems Normally Operated (Online Systems)

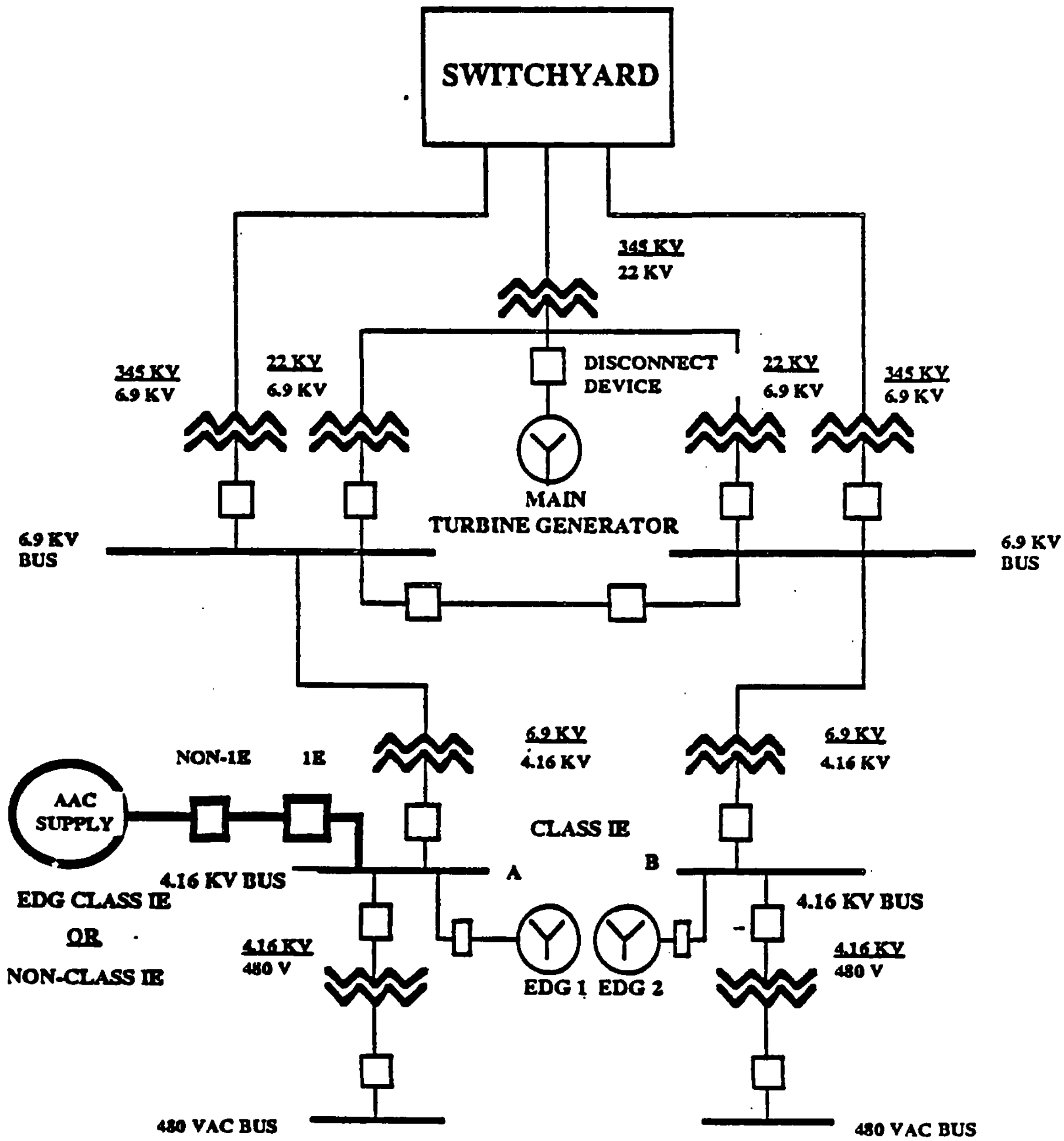
Availability	AAC systems normally online should attempt to be available to its associated unit at least 95% of the time the reactor is operating.
Reliability	No reliability targets or standards are established for online systems.

APPENDIX C. SAMPLE AAC CONFIGURATIONS

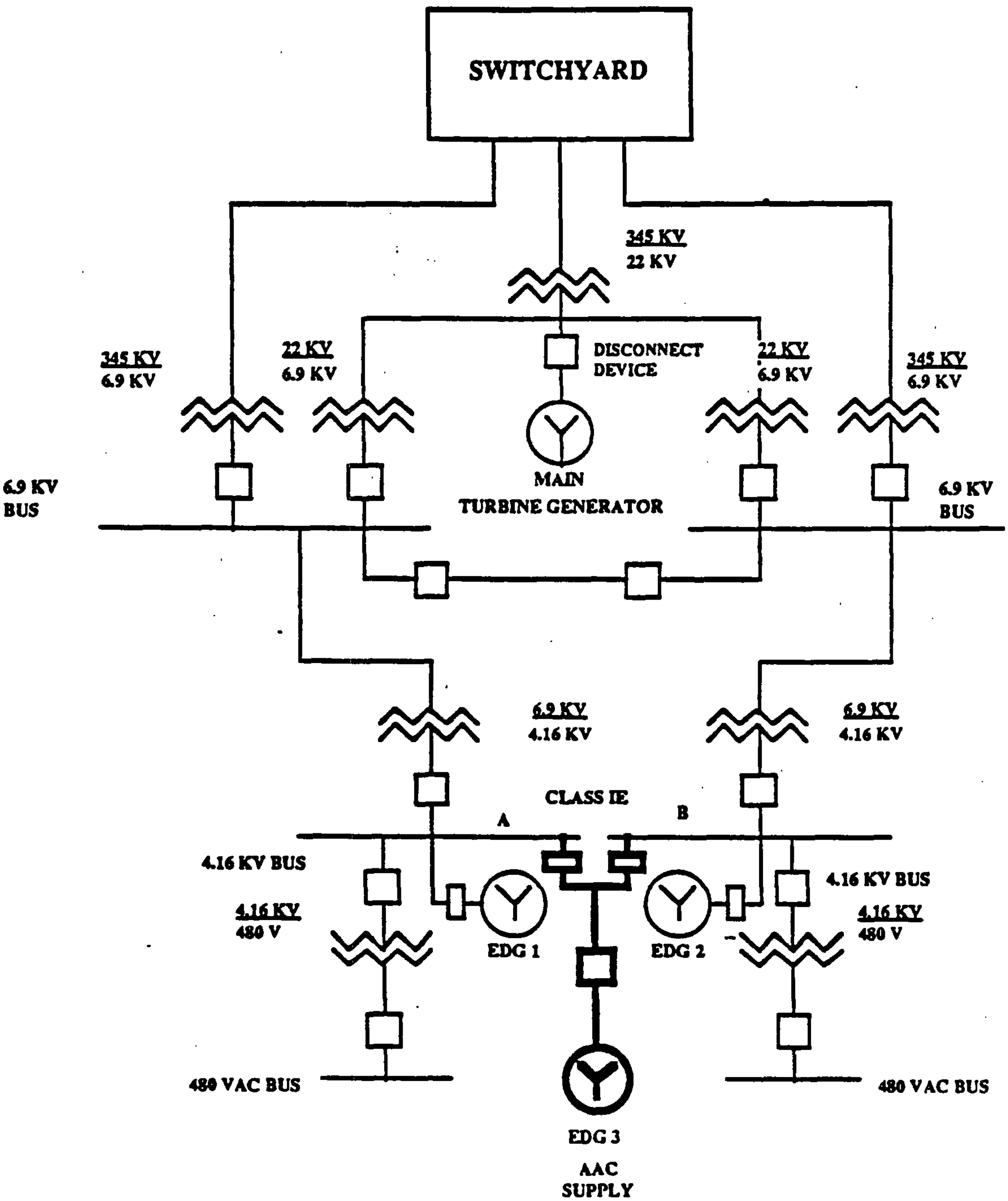
AAC Configuration 1A: Non-Class 1E Power Source



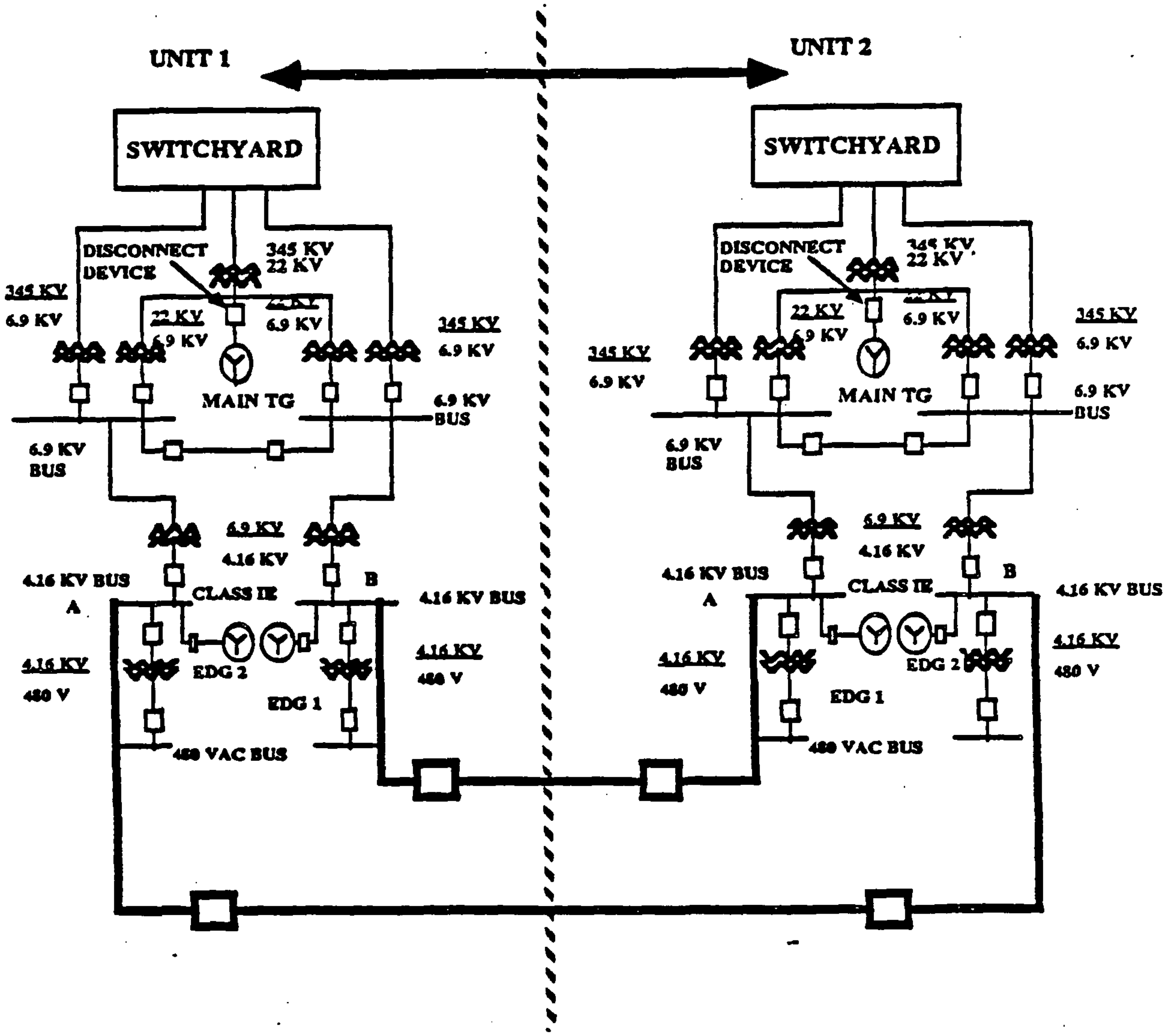
AAC Configuration 1B: DG Class 1E or Non-Class 1E



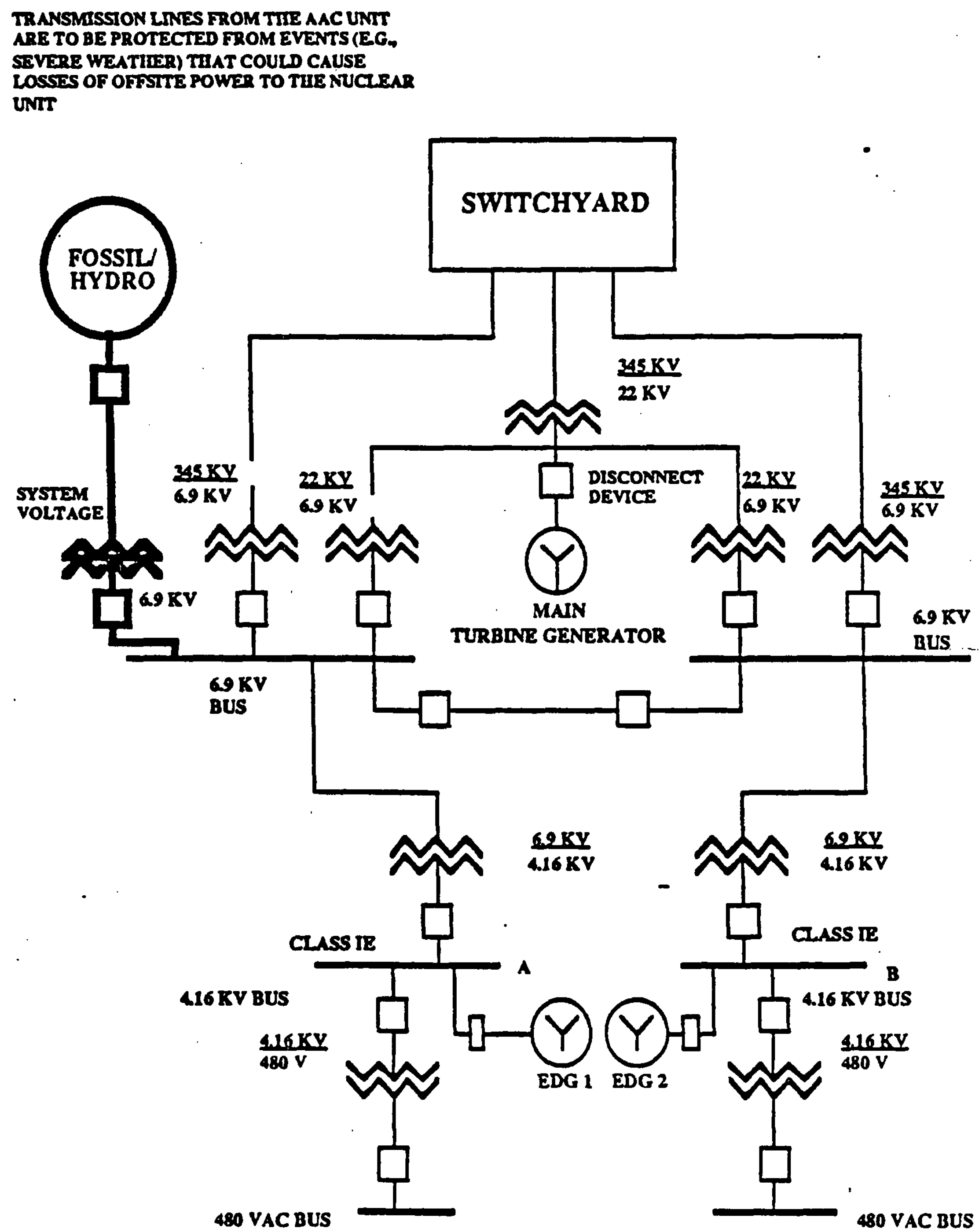
AAC Configuration 2A: Swing Diesel



AAC Configuration 2B: Dedicated Diesels with Cross-tie at Multiunit Site

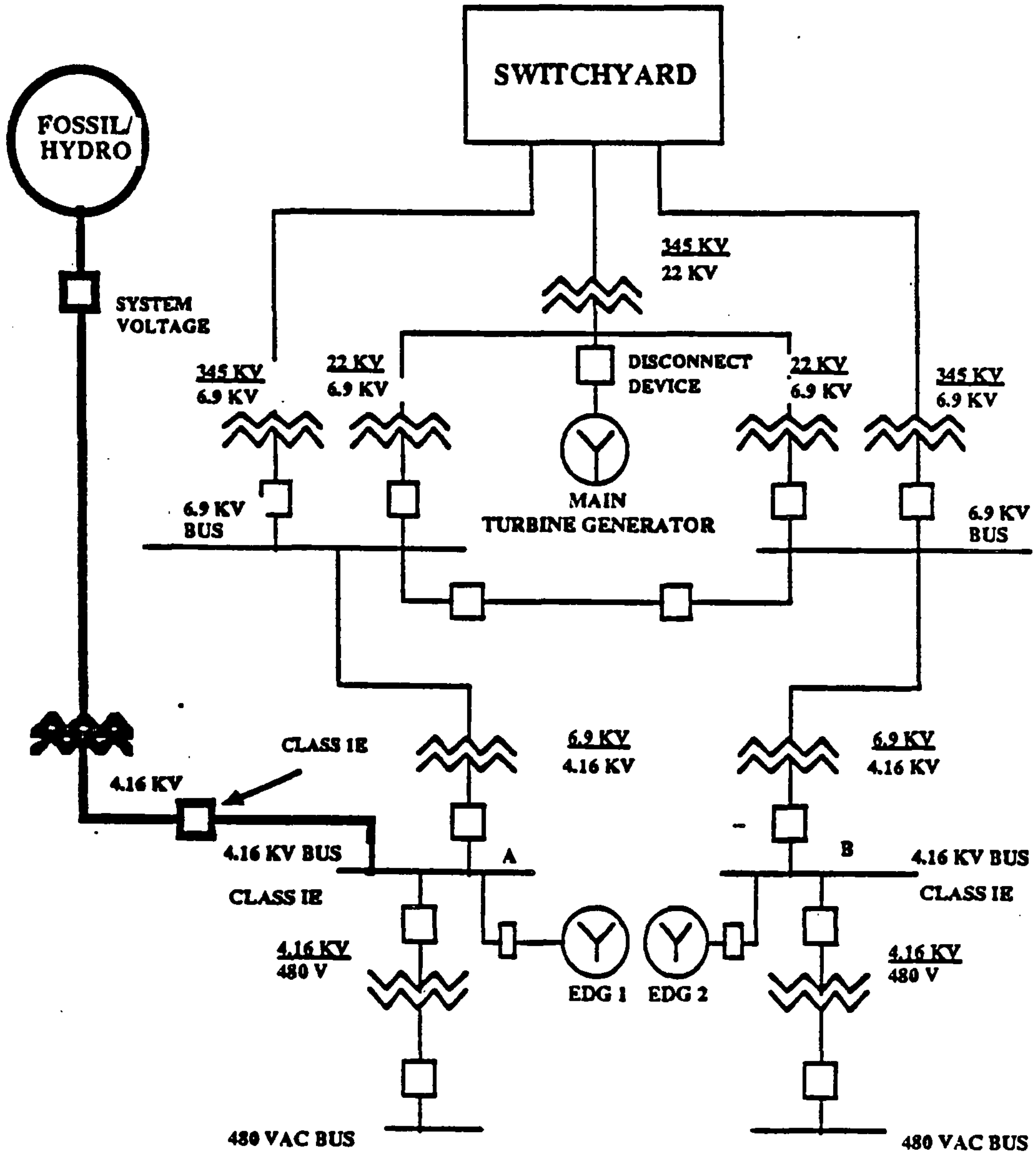


AAC Configuration 3A: Nearby Power Source Connected to Non-Class 1E Bus



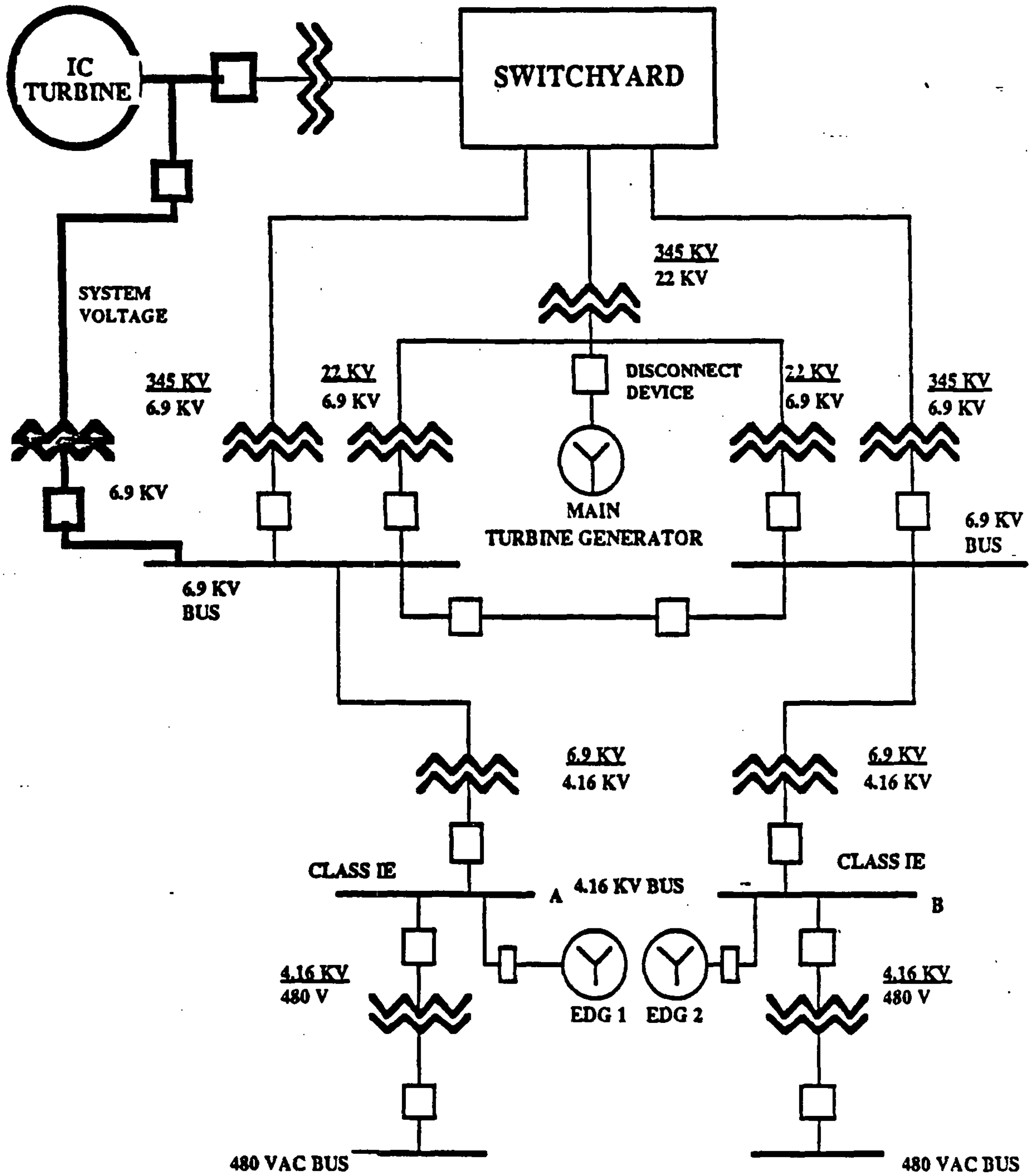
AAC Configuration 3B: Nearby Power Source Connected to a Class 1E Bus

TRANSMISSION LINES FROM THE AAC UNIT ARE TO BE PROTECTED FROM EVENTS (E.G., SEVERE WEATHER) THAT COULD CAUSE LOSSES OF OFFSITE POWER TO THE NUCLEAR UNIT



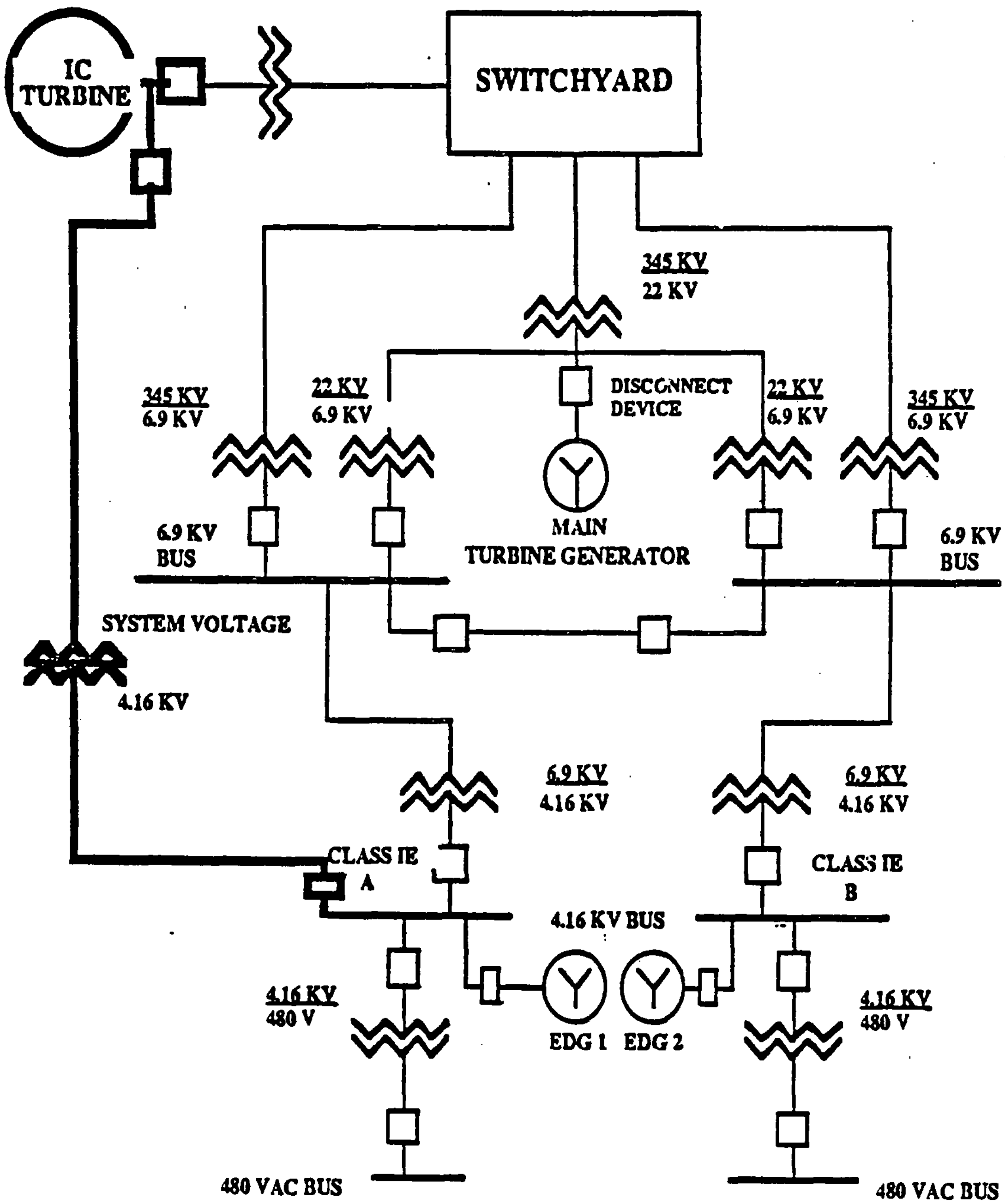
AAC Configuration 4A: Onsite IC Turbine Connected to a Non-Class 1E Bus

THE IC TURBINE AND CABLES FROM THE IC TURBINE TO THE 6.9 KV BUS ARE TO BE PROTECTED FROM EVENTS (E.G., SEVERE WEATHER) THAT COULD CAUSE LOSSES OF OFFSITE POWER TO THE NUCLEAR UNIT.



AAC Configuration 4B: Onsite IC Turbine Connected to a Class 1E Bus

THE IC TURBINE AND CABLES FROM THE IC TURBINE TO THE 4.16 KV BUS ARE TO BE PROTECTED FROM EVENTS (E.G., SEVERE WEATHER) THAT COULD CAUSE LOSSES OF OFFSITE POWER TO THE NUCLEAR UNIT



APPENDIX D. EDG RELIABILITY PROGRAM

D.1 BACKGROUND

The NRC proposed resolution to station blackout is based on a risk analysis presented in NUREG-1032. An important input parameter in the risk analysis is emergency diesel generator (EDG) reliability. While the NRC recognizes that the industry average EDG reliability is acceptably high, they are concerned that some plants have marginal machines and that current high reliability at some plants may degrade in the future. In order to ensure that EDG performance is maintained at a high level and improved for those machines that are currently marginal, the NRC is pursuing the resolution of Generic Issue B-56 *Emergency Diesel Generator Reliability.* The NRC Staff maintains that the resolution of USI A-44 *Station Blackout* should include (1) the identification of target EDG reliabilities and (2) the commitment to implement an EDG reliability program. An outline of a possible EDG reliability program to be developed under Generic Issue B-56 is described below.

D.2 EDG RELIABILITY PROGRAM

The reliable operation of on-site emergency AC power sources should be ensured by a reliability program. For emergency diesel generators, such a program might be comprised of the following elements (or equivalent):

(1) Establishment of individual EDG target reliability levels consistent with the plant category and coping duration determined in Section 3.2.5.

(2) Surveillance testing and reliability monitoring programs designed to track EDG performance and also support maintenance activities.

(3) A maintenance program which ensures that the target EDG reliability is being achieved and which also provides a capability for failure analysis and root cause investigations.

(4) An information and data collection system capability which services the elements of the reliability program, and which monitors achieved EDG reliability levels against target values.

(5) Identified responsibilities for the major program elements and a management oversight program for reviewing reliability levels being achieved and assuring that the program is functioning properly.

APPENDIX E: ANALYSIS OF THE EFFECTS OF LOSS OF VENTILATION UNDER STATION BLACKOUT CONDITIONS

E.1 Introduction

This appendix provides the technical basis for the methodology used in Section 7.2.4 to calculate 4-hour steady state temperatures for the dominant areas of concern.

E.2 Dominant Areas of Concern

Since normal ventilation is unavailable during a station blackout, equipment needed to achieve and maintain safe shutdown in a blackout may be subjected to elevated temperatures. Only a limited set of equipment, however, is needed to provide core cooling and decay heat removal during a station blackout. Loss of ventilation concerns are thus, limited to rooms and cabinets housing this equipment.

This approach focuses on rooms and plant areas labeled Dominant Areas of Concern. These rooms are limited to areas that will have significant heat load in a station blackout, and also contain safe shutdown equipment. AC-driven equipment will not be operable in a station blackout and process steam will not be in the plant. Similarly, only a few plant areas will contain safe shutdown equipment.

For PWRs, the pump room for the steam driven auxiliary feedwater system is the Dominant Area of Concern. The Dominant Areas of Concern in BWRs are the HPCI/HPCS and RCIC pump rooms and the main steam tunnel. In general, the size of RCIC turbines relative to AFW turbines, and relative room geometries make the RCIC results bounding.

E.3 Analysis of Compartment Heatup

Analytical models have been developed to estimate the temperature rise in compartments during a station blackout. In this analysis, a lumped parameter model is used to calculate the average air temperature as a function of time after loss of ventilation. The effect of mitigating actions, such as opening doors to promote air circulation, are also considered.

E.3.1 Model Description

A simple lumped parameter model of compartment heatup can be used to estimate the bulk air temperature as a function of time after loss of ventilation. The rate of change of the air temperature can be calculated by an energy balance if the appropriate heat sources and sinks can be described. An equation for the rate of change of the air temperature is given by:

$$\rho c_p V dT_{air}/dt = Q_{sources} - Q_{sinks} \qquad \text{(E-1)}$$

where:

ρ is the air density;

c_p is the constant pressure specific heat of air;

V is the volume of the compartment.

The sources of heat considered are hot steam pipes, and to a lesser extent in a station blackout, DC switchgear and equipment. The heat from either bare or insulated steam pipes is dissipated to the air and walls by natural convection and thermal radiation. Since the absorptivity of air is very small, the heat dissipated by thermal radiation will be absorbed primarily by the concrete walls.

There are two heat sinks available to remove heat from compartment air: concrete walls act as a large heat sink; and, compartment doors can be opened to remove heat by convection. Heat transfer to walls can be estimated by heat transfer coefficient correlations for natural convection along a vertical plate. Heat transfer to the walls via thermal radiation from steam pipes can be estimated if the temperature of the steam pipes and wall are known.

Equation (E-1) can then be expanded to:

$$\rho c_p V dT_{air}/dt = Q_{elec} + Q_{pipe} + Q_{pump} - Q_{wall} - Q_{door} \qquad \text{(E-2)}$$

where:

Q_{elec} is the heat load from major DC electrical equipment;

Q_{pipe} is the heat dissipated to the air from steam pipes by natural convection;

Q_{pump} is the heat dissipated to the air from steam driven pumps by natural convection;

Q_{wall} is the heat transferred to walls from the air by natural convection; and,

Q_{door} is the heat convected out of compartment openings.

Each term on the right hand side is discussed below.

The heat dissipated to the air from electrical equipment for a particular compartment can be found by adding up the power dissipated by major DC loads.

The heat transferred to the air from the steam pipes by natural convection can be estimated from the correlation for a long horizontal cylinder in a quiescent fluid. The heat transferred from steam driven pumps can be estimated from the correlation for a sphere in a quiescent fluid. The correlations for convective heat transfer rates of Churchill and Chu are used (Incropera [1981]):

For a cylinder:

$$Nu_D = (h_p D)/k = CRa^n{}_d \qquad \text{(E-3)}$$

where:

Nu_D is the Nusselt number

h_p is the heat transfer coefficient for free convection from a pipe;

D is the diameter of the pipe;

k is the thermal conductivity of air;

Ra_d is the Rayleigh number based on the pipe diameter; and,

C and n are empirical constants.

The Rayleigh number is defined as:

$$Ra_d = g\beta(T_{pipe} - T_{air})D^3/\alpha\nu \qquad \text{(E-4)}$$

where:

g is gravitational acceleration;

β is the volumetric thermal expansion coefficient;

T_{air} is the bulk temperature of the air;

T_{pipe} is the temperature of the pipe;

α is the thermal diffusivity of air;

v is the kinematic viscosity of air.

For an ideal gas, $\beta = 1/T_{air}$ (based on absolute temperatures).

Churchill and Chu have recommended a single correlation for a wide Rayleigh number range ($10^{-5} < Ra_d < 10^{12}$):

$$Nu_D = \left\{ 0.60 + \frac{0.387\,(Ra_d)^{1/6}}{[1 + (0.559/Pr)^{9/16}]^{8/27}} \right\}^2 \qquad \text{(E-5)}$$

where Pr is the Prandlt number.

The convective heat transfer coefficient for a cylinder can then be expressed as:

$$h_p = (k/D)\,[0.60 + 0.321(Ra_d)^{1/6}]^2 \qquad \text{(E-6)}$$

Similarly, for a sphere, the Nusselt number can be represented by:

$$Nu_D = 2 + \frac{0.589(Ra_d)^{1/4}}{[1 + (0.469/Pr)^{9/16}]^{4/9}} \qquad \text{(E-7)}$$

For $Pr \geq 0.7$ and $Ra_d \leq 10^{11}$

The convective heat transfer coefficient for a sphere can then be expressed as:

$$h_s = (k/D)[2 + 0.454(Ra_d)^{1/4}] \tag{E-8}$$

Natural convection to the walls is described by the Churchill and Chu correlation for free convection for a vertical plate (Incropera [1981]). This correlation is given by:

$$h_w = (k/L)\,[.825 + .324(R_{al})^{1/6}]^2 \tag{E-9}$$

where l is the height of the wall. The Rayleigh number is based on L and the difference in temperature between the air and wall surface:

$$R_{al} = g\beta(T_{air} - T_{wall})L^3/\alpha\nu \tag{E-10}$$

The temperature of the inside wall surface also varies as a function of time. An explicit finite difference model of one–dimensional transient heat conduction in a plane wall can be used to describe the wall temperature. This model considers a time dependent heat flux to the wall as a boundary condition to be satisfied at each time step. A fine mesh is used near the wall surface to accurately predict the temperature gradients resulting from the heat flux to the wall. Deep into the wall, where temperature gradients are relatively small, a coarse mesh is employed.

Application of this model to a typical room containing a heat flux of 80 KW, a total surface area of 514 m^2, and an 8 inch thick concrete wall resulted in a change in wall temperature of approximately 2.5 °C (4.5 °F) over a period of 4 hours.

The heat flux specifying the boundary condition for the wall conduction model is the sum of the convective flux, found by use of (E-9), and the radiative flux from hot steam pipes, which is found using the Stefan-Boltzmann law.

Correlations have been developed to estimate heat flow through openings by convection. These correlations are applicable to the compartment heatup scenario, where heat is convected from a room with a higher temperature to a room with a lower temperature through an opening. The following correlation is suitable for the dimensionless parameters associated with compartment heatup scenarios:

$$h_d = 2\,(k/H)\,Gr^{1/2}\,Pr \tag{E-11}$$

where the Grashof number Gr is based on the door height H in the following manner:

$$Gr = gH^3\,(T_{air} - T_{\infty})/v^2 T_{ave} \tag{E-12}$$

The temperature of the outside air is denoted by T_{∞}, and T_{ave} is the average of T_{air} and T_{∞}.

Equation (E-2) can now be expanded to:

$$\rho c_p V dT_{air}/dt = Q_{elec} + h_p(T_{air})A_p(T_{pipe} - T_{air}) + hs(T_{air})As(T_{pump} - T_{air}) - h_w(T_{air}, T_{wall})A_w\,(T_{air} - T_{wall}) - h_d\,(T_{air})\,A_d(T_{air} - T_{\infty}) \tag{E-13}$$

This equation can be solved numerically for a particular geometry if the initial conditions, namely the initial temperatures of the room air, walls, and outside air, are specified.

This steady state (i.e. $dT_{air}/dt = 0$) solution to this equation assuming no open doors can be derived by setting $Q_{door} = 0$. Equation (E-13) then becomes:

$$Q_{total} = h_w(T_{air}, T_{wall})\,A_w\,(T_{air} - T_{wall}) \tag{E-14}$$

where, Q_{total}, represents the total amount of heat deposited in the building.

The natural convection coefficient can be calculated with the correlation shown in equation (E-9). For the range of air temperatures under consideration, (i.e., 22-50^{o} C) the $.825$ term is much smaller than $.324R_{al}^{(1/6)}$, and therefore can be neglected. Equation (E-9) becomes:

$$h_w = (k/L)\,[.324(R_{al})^{1/6}]^2 \tag{E-15}$$

Substituting in the Rayleigh number as defined in equation (E-10):

$$h_w = .1k[g\beta(T_{air} - T_{wall})/\alpha v]^{1/3} \qquad \text{(E-16)}$$

Substituting this result into equation (E- 14) yields:

$$Q_{total} = .1k[g\beta/\alpha v]^{1/3} A_w(T_{air} - T_{wall})^{4/3} \qquad \text{(E-17)}$$

Finally, for the temperature ranges being discussed:

$$.1k[g\beta/\alpha v]^{1/3} \sim 1$$

Hence, rearranging equation (E-17):

$$T_{air} = (Q_{total}/A_w)^{3/4} + T_{wall} \qquad \text{(E-18)}$$

This is the result used in Step 4 of Section 7.2.4.

Scaled experiments performed by Brown [1962] suggest the following correlation for predicting heat transfer coefficients through vertical openings:

$$Q_{door} = V' \rho_{ave} C_p(T_{air} - T_{\infty}) \qquad \text{(E-19)}$$

where

$$V' = .2\, W\, (g\, \Delta\rho/\rho_{ave})^{1/2}\, H^{3/2} \qquad \text{(E-20)}$$

ρ_{ave} = the average of the densities of the air inside and outside of the room

$\Delta\rho$ = the change in density between the air inside and outside of the room

C_p = the specific heat of the room air at constant pressure

T_{∞} = the temperature of the air outside of the room

T_{air} = the temperature of the air inside of the room

W = the width of the door

H = the height of the door

g = the acceleration due to gravity

It should be recognized that the Brown experiments are small-scale and should be used with an understanding of their underlying basis.

Since air can be modeled as a perfect gas:

$$\rho_{ave} = P/2R(1/T_\infty + 1/T_{air})$$

$$\Delta\rho = P/R(1/T_\infty - 1/T_{air})$$

where

P = atmospheric pressure

R = the universal gas constant

Equation (E-19) can then be written as:

$$Q_{door} = .2\, W H^{3/2} g^{1/2} [2(T_{air}-T_\infty)/(T_\infty + T_{air})]^{1/2} \rho_{ave} C_p (T_{air}-T_\infty) \qquad \text{(E-21)}$$

For a typical RCIC room numerical analysis predicts a rapid temperature buildup within the first half hour. The temperature will increase by only a few degrees for the next three and one-half hours where it then approaches steady state. This rapid thermal buildup appears to characterize room geometries for dominant areas of concern.

Once a door is opened, analysis indicates the temperature of the room will decrease rapidly and approach steady state conditions in approximately 20 minutes for similar room geometries. From this analysis, it is apparent that the time at which a door is opened past thirty minutes will have little effect on either the peak temperature or the final steady state temperature that is achieved. For this reason the steady state solution of equation (E-2) will apply when calculating the final temperature for a four hour event.

The steady state solution to equation (E-2) is:

$$0 = Q_{elec} + Q_{pipe} + Q_{pump} - Q_{wall} - Q_{door}$$

$$Q_{elec} + Q_{pipe} + Q_{pump} = Q_{wall} + Q_{door}$$

$$Q_{total} = Q_{wall} + Q_{door}$$

From Equation (E-17)

$$Q_{wall} = A_w(T_{air} - T_{wall})^{4/3}$$

Therefore

$$Q_{total} = A_w(T_{air} - T_{wall})^{4/3} + .2\, W\, H^{3/2}\, g^{1/2}\, [2(T_{air}-T_\infty)/(T_\infty + T_{air})]^{1/2}\, \rho_{ave}\, C_p\, (T_{air}-T_\infty) \qquad \text{(E-22)}$$

By substituting

$$\Delta T = T_{air} - T_\infty$$

and

$$T_{wall} = T_\infty$$

$$Q_{total} = A_w\, \Delta T^{4/3} + .2\, W\, H^{3/2}\, g^{1/2}\, \rho_{ave}\, Cp\, (2)^{1/2}\, [(\Delta T)^{3/2}/(\Delta T + 2\, T_{wall})^{1/2}] \qquad \text{(E-23)}$$

After performing a Binomial expansion on the term $(\Delta T + 2\, T_{wall})^{1/2}$, Equation (E-23) becomes:

$$0 = -4\, T_{wall}\, Q_{total} - Q_{total}\, \Delta T + 4T_{wall}\, A_w\, \Delta T^{4/3} + A_w\, \Delta T^{7/3} + 4\, (.2WH^{3/2}g^{1/2})C_p\, \rho_{ave}\, T^{1/2}{}_{wall}\, \Delta T^{3/2} \qquad \text{(E-24)}$$

This transcendental equation cannot be solved explicitly for ΔT, although a numerical solution is possible. By using the results of this numerical solution for a wide range of input parameters, a correlation can be developed that approximates the actual solution as follows:

$$0 = -A - B\Delta T + C\Delta T^{4/3} + D\Delta T^{7/3} + E\Delta T^{3/2} \tag{E-25}$$

By solving this equation for $WH^{3/2}$, we find that:

$$WH^{3/2} = [\,(A + B\Delta T - C\Delta T^{4/3} - D\Delta T^{7/3})\,/\,(4\rho_{ave}C_pT^{1/2}{}_{wall})(.2g^{1/2})\Delta T^{3/2}\,] \tag{E-26}$$

Where for the temperature ranges considered:

$\rho_{ave} \sim 1$

$C_p \sim 1$

Therefore equation (E- 26) becomes:

$$WH^{3/2} = [\,(A + B\Delta T - C\Delta T^{4/3} - D\Delta T^{7/3})\,/\,(4T^{1/2}{}_{wall})(.2g^{1/2})\Delta T^{3/2}\,]$$

This equation can be solved for various values of Q_{total}, A_w, and ΔT. Several graphs of the results were plotted in order to obtain a factor for ΔT based on $WH^{3/2}$ using a power series curve fit of the data. The data used to develop these graphs is presented in Table E-1.

Table E-1

Data Used for Power Series Curve Fit

Qtotal (KW)	Aw (sq.m)	DELTA T (°K)
65	400	10
65	400	20
65	400	30
65	400	40
65	500	10
65	500	20
65	500	30
65	500	40
65	750	10
65	750	20
65	750	30
65	750	40
65	1000	10
65	1000	20
65	1000	30
65	1000	40
80	400	10
80	400	20
80	400	30
80	400	40
80	500	10
80	500	20
80	500	30
80	500	40
80	750	10
80	750	20
80	750	30
80	750	40
80	1000	10
80	1000	20
80	1000	30
80	1000	40

The factor $(16.18(F_{door})^{0.8653})$ was incorporated into Equation (E-18) to obtain:

$$T_{air} = (Q_{total})^{3/4} / [(A_w)^{3/4} + 16.18(F_{door})^{0.8653}] + T_{wall} + 4 \qquad \text{(E-27)}$$

where

$$F_{door} = H^{3/2} W$$

This relationship has been shown to have a correlation coefficient (r^2) equal to 0.99. Since most heat transfer correlations contain r^2 factors between 0.9 and 1.0, this correlation is well within acceptable limits. To account for any uncertainties within this correlation a plot of temperatures obtained using equation (E-27) was compared to a plot of equation (E-23) with a specified temperature difference. It was found that, in general, equation (E-27) predicts temperatures from 3 -3.5 °C lower than predicted from this plot. To account for this uncertainty a correction factor of 4°C was added. This is the result used in Step 5 of Section 7.2.4. Note that this equation is a simplified form of the complete steady state solution. Heat transfer coefficients and thermal properties have been evaluated in MKS units. Therefore, this dimensionally inconsistent equation is valid only with the equation parameters in the following units:

Q_{total} = Watts

T_{wall} = °K or °C (units of T_{air} will result accordingly)

A_w = square meters

H = meters

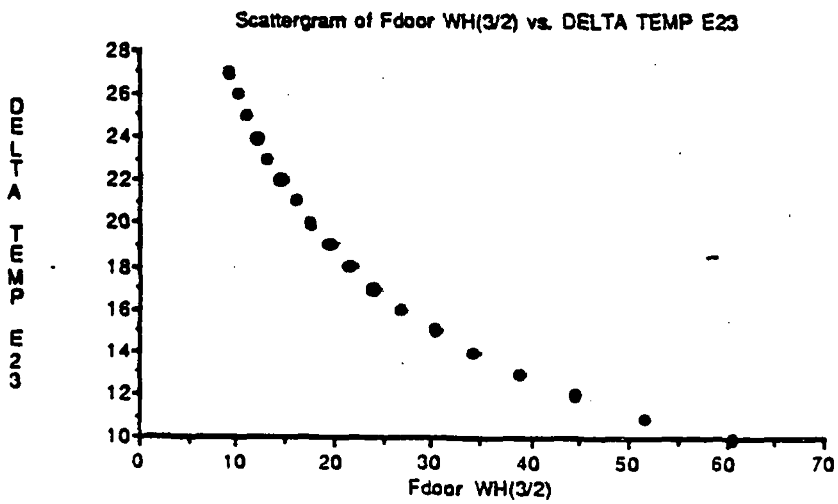
W = meters

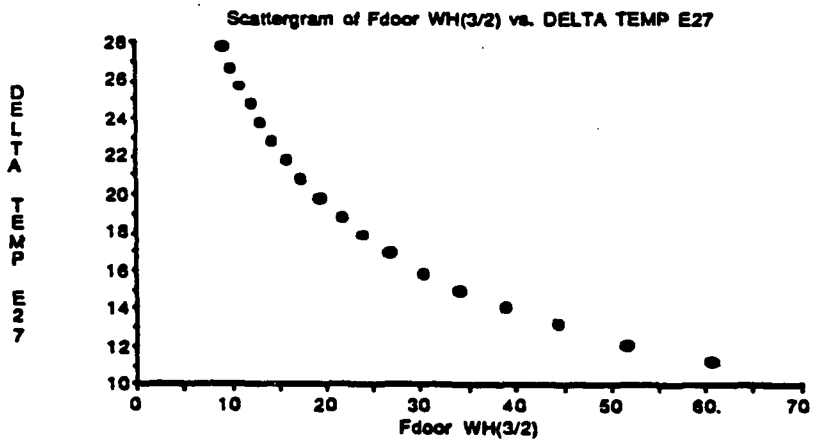
By testing the sensitivity of equation (E-26) it was found that this correlation is valid for the following parameter ranges:

$$24000W < Q < 100000W$$

$$0\ °C < \Delta T < 50\ °C$$

Plots for a sample case that illustrates the results of Equations (E-23) and the corrected form of Equation (E-27) are shown below:





E.3.2 RCIC Pump Room with No Openings

To provide an upper bound for the potential temperature rise which may be expected during a station blackout event, Equation (E-13) is solved with the assumption that A_d is zero -- effectively representing a closed compartment. The RCIC room chosen for analysis by Jacobus et. al. [1986] has been reevaluated for comparison. The geometry and initial conditions are specified in Table E-2, below.

Table E-2

RCIC Room Geometry and Initial Conditions

Wall Height	7.5 m
Surface Area	514 m^2
Volume	892 m^3
Steam Pipe dia.	.2 m
Steam Pipe length	15.4 m
Initial Air Temp.	40° C (104° F)
Initial Wall Temp.	40° C (104° F)
Pipe Temp. (uninsulated)	288° C (550° F)
Pipe Temp. (insulated)	93° C (200° F)
Pipe Emissivity	0.8
Electric energy dissipated	63 KW

If the steam pipes are insulated the temperature rise will be smaller, since less heat will be convected to the air and a relatively smaller heat flux will strike the wall surface. This case is considered more representative than the case with uninsulated steam pipes. Assuming insulated steam pipes, the calculated temperature rise is about 37° C (67° F) after 4 hours, and about 38° C (72° F) after 8 hours. This result compares to the 44° C temperature rise obtained after 8 hours by Jacobus.

E.3.3 RCIC Pump Room with a Compartment Opening

NUMARC has determined that the effect of opening doors to promote natural circulation is determined to prevent compartments from reaching excessive temperatures during station blackout conditions. A heat transfer term accounting for convection of heat to outside air has been included into the energy balance.

Calculations have been made assuming that a 2m by 1m door is opened one-half hour into the station blackout event, and that heat transfer is described by (E-11). It is also assumed that the flow of exiting hot air is matched by the flow of incoming cool air from outside. Outside air is assumed to be constant at 40° C (104° F). For the case with insulated pipes, a temperature rise of 28° C (50° F), (using equation (E-27) 31.5 °C or 56.8 °F), was obtained after 4 hours, peak temperature of 165° F is predicted by simulation at the time the door is opened. If two 2m by 1m doors are opened one-half hour into the event, a temperature rise of 23°C (41°F), (using equation (E-27) 25.7°C or 46.3 °F), will be obtained after four hours. In comparison, a temperature rise of 37° C (67° F) was obtained for the same case with the door closed.

Sensitivity analyses were performed to measure the effect of changes in the heat transfer correlation used to describe convection through open doors. The heat transfer coefficient was reduced by 25% to account for the normal range of uncertainty associated with empirical heat transfer correlations. Little effect on temperature rises after four hours was observed -- reducing the heat transfer coefficient by 25% resulted in an increase in temperature rise of about one degree Centigrade for the two cases just discussed.

In conclusion, opening doors will reduce the temperature rise during station blackout to a level where equipment operability can be assured to a sufficiently high level of confidence. For the example case, the estimated temperature after four hours into a station blackout can be reduced from 171° F to 155° F if a door is opened one-half hour into the event. If two doors are opened, the estimated temperature can be reduced to 146° F. Furthermore, calculations reveal that rooms will cool rather quickly after doors are opened, and peak temperatures will exist for only a few minutes.

E.4 Nomenclature

ρ = density (Kg/m^3)

C_p = constant pressure specific heat (KJ/Kg °K)

V = volume (m^3)

T_{air} = temperature of the air inside of the room (°K)

T_∞ = temperature of the air outside of the room (°K)

T_{wall} = temperature of the room walls (°K)

g = acceleration due to gravity (m/s^2)

β = expansion coefficient (1/°K)

α = diffusivity (m^2/s)

ν= kinematic viscosity (m^2/s)

A_w = area of the wall (m^2)

L = length (m)

l = height of the wall (m)

W = width (m)

H = height of the opening (m)

D = diameter (m)

F_{door} = door factor ($m^{5/2}$)

Q = heat transfer rate (W)

k = thermal conductivity (W/m°K)

h_p, h_d, h_w = convective heat transfer coefficients (W/m^2°K)

R_{ad}, R_{al} = Rayleigh number (dimensionless)

P = pressure (bar)

R = universal gas constant (0.08314 (m^3 bar/Kmol °K))

r^2 = correlation coefficient

APPENDIX F: ASSESSMENTS OF EQUIPMENT OPERABILITY IN DOMINANT AREAS UNDER STATION BLACKOUT CONDITIONS

F.1 Introduction

This appendix outlines a methodology for providing reasonable assurance of the operability of equipment used to cope with a station blackout in the dominant areas of concern. The approaches identified in this appendix are discussed conceptually with additional details being developed as indicated below.

Station blackout is not a design basis accident, and, therefore, is not subject to the requirements of 10 CFR §50.49 and the rigorous certification process for equipment operability. However, since station blackout coping equipment needs to operate in order to achieve safe shutdown, reasonable assurance should be provided that no thermal-induced failures will result due to loss of forced ventilation. Station blackout environments in the dominant areas of concern outside containment are expected to experience increases in air temperature. The resulting temperatures are expected to range from slight to moderate increases in temperature, in most cases not exceeding 150° F.

Most equipment is expected to operate in these station blackout environments with no loss of function for the short duration expected, (i.e., 4-hours). The basis for this general conclusion can be traced to previous studies and analyses performed, as well as plant operating experience. The approaches discussed in this appendix provide acceptable bases for reaching this conclusion on a plant-specific basis. In particular, the approaches justify removing classes of equipment (e.g., relays, switches) from further consideration and focusing attention on those components of concern. The approaches may be used individually, or in combination in reaching a conclusion that an acceptable basis exists for equipment operability in a station blackout environment.

Six approaches may be used to establish equipment operability in a station blackout:

(1) equipment previously evaluated;
(2) equipment design capability;
(3) materials;
(4) equipment inside instrumentation and control cabinets;
(5) generic studies and experience; or,
(6) plant-specific experience and tests.

F.2 Equipment Previously Evaluated

Equipment that is similar to equipment already qualified under 10 CFR §50.49 need not be further evaluated if the station blackout environments do not exceed the qualification temperatures.

F.3 Equipment Design Capability

Equipment vendors generally provide a design temperature associated with the continuous operation of their equipment. A margin may exist above design temperature which varies according to equipment class (e.g., smaller margins for electronic equipment relative to electromechanical devices) and the expected operating conditions (e.g., temperature levels, time at these elevated temperatures, duty cycle, etc.). For the station blackout coping duration, equipment design and operational requirements should be reviewed in terms of the Conditions 1, 2, and 3 defined in Section 2.7 of this report. Reasonable assurance for equipment operability may be determined if the design temperature plus the margin for the equipment or component class has been evaluated against the temperatures and considerations in Conditions 1, 2, and 3. Identification of expected margins for various equipment or component classes is under development.

Reasonable assurance for equipment operability is provided if it is shown that the design temperature plus the expected margin for the equipment or component class does not exceed the bulk air temperature expected in a 4-hour station blackout.

F.4 Materials

The primary consideration for equipment operability in a station blackout is the potential for thermal-induced failure. Most materials used in plant equipment and components are not subject to physical or chemical changes in the range of temperatures expected to result in a station blackout. Materials or combinations of materials that are susceptible to significant changes in these ranges will be identified and used to screen components that are potentially sensitive to station blackout conditions.

Reasonable assurance for equipment operability is provided if it is shown that the station blackout coping equipment does not contain materials that are susceptible to significant physical or chemical changes in a station blackout

environment.

F.5 Equipment Inside Instrumentation and Control Cabinets

Components located inside instrumentation and control cabinets are normally exposed to the heat generated by electrical power supplies. Most cabinets are not equipped with forced ventilation, relying, instead, on natural convection through louvers in the cabinet. Guidelines direct operators to open doors for cabinets containing energized equipment relied upon to cope with a station blackout within 30 minutes in order to provide more extensive mixing with the general area. This action is expected to reduce the potential for building up higher air temperatures in the immediate vicinity of electrical and electronic equipment and components.

Reasonable assurance for equipment operability is provided if it is shown that the station blackout coping equipment and components inside instrumentation and control cabinets with open doors will not be exposed to a thermal environment that exceeds normal operating conditions with the doors closed.

F.6 Generic Studies and Experience

The current state of knowledge concerning equipment operability in elevated thermal environments provides a substantial basis for concluding that plant equipment can properly function in thermal environments above design conditions. Three recent studies support the conclusion that plant equipment can operate under loss of forced ventilation for periods longer than a four hour station blackout:

(1) *Letter Report on Equipment Operability During Station Blackout Events*, M. J., Jacobus, V. F. Nicolette , and A. C. Payne, Sandia National Laboratories, (1986); and

(2) *Effects of Ambient Temperature on Electronic Components in Safety-Related Instrumentation and Control Systems*, M. Chiramal, AEOD/C604, United States Nuclear Regulatory Commission, (1986).

The Jacobus report also found that certain classes of equipment will not be affected when exposed to temperatures above 150° F for eight hours or longer. The McGuire report concluded that an actual loss of ventilation event of a duration longer than four hours did not adversely affect the performance of equipment needed for safe shutdown.

These studies can be used to support conclusions of equipment operability under elevated temperature conditions

estimated for the station blackout coping duration.

If the site-specific application is not covered by the above data base or other generic studies, then other sources of data such as Licensee Event Reports (LERs), or the Nuclear Plant Reliability Data System (NPRDS) can be used to support conclusions of equipment operability under elevated temperature conditions estimated for the station blackout coping duration.

F.7 Plant-Specific Experience and Tests

Some plants have actually experienced the effects of loss of ventilation or have studied the issue for specific applications. For such cases, reasonable assurance for equipment operability is provided if no failures of equipment needed to cope with a station blackout resulted from exposing the equipment to temperatures expected from a four hour station blackout during tests or operational events.

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