BROWNS FERRY NUCLEAR PLANT FIRE

HEARINGS

BEFORE THE

JOINT COMMITTEE ON ATOMIC ENERGY
CONGRESS OF THE UNITED STATES

NINETY-FOURTH CONGRESS

FIRST SESSION

SEPTEMBER 16, 1975

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BROWNS FERRY NUCLEAR PLANT FIRE

Part 1

TUESDAY, SEPTEMBER 16, 1975

CONGRESS OF THE UNITED STATES,
JOINT COMMITTEE ON ATOMIC ENERGY,
Washington, D.C.

The Joint Committee met at 10 a.m., pursuant to call, in room S-407, the Capitol, Senator Joseph M. Montoya, presiding.

Present: Senators Montoya, Baker, and Case; and Representatives Price, Young, Horton, and Anderson.

Also present: George F. Murphy, Jr., executive director; James B. Graham, assistant director; William J. Parler, committee counsel; Norman P. Klug and Stephen J. Lanes, technical consultants; and James K. Asselstine, assistant counsel.

OPENING REMARKS BY SENATOR MONTOYA

Senator MONTOYA. The Joint Committee will be in order.

The committee meets this morning to receive testimony on the circumstances surrounding the fire which occurred on March 22, 1975, at the TVA's Browns Ferry Nuclear Plant, located near Athens, Ala.

In the development of our Nation's civilian nuclear power program, the paramount consideration should and must be nuclear safety. I believe that we have had an excellent safety record, as witnessed by the fact that no one has been killed or injured in a nuclear accident at a commercial powerplant. However, the fire at Browns Ferry is a striking example of how things can and do go wrong. In this connection, we want to examine three basic issues:

First, the factors that caused the fire to get out of control and the response to it.

Second, the impact of the fire on the nuclear reactors at the site and how many of the defenses in depth were breached.

And, third, the relationship of this accident to our nationwide nuclear power program.

As I understand it, the fire began by the use of an almost primitive inspection technique of holding a candle close to a hole which contained flammable material. The initial attempt to beat out the flames was made by using a flashlight. When that failed, rags were used to attempt to stifle the flames. And when that effort was unsuccessful, one of the men attempted to actuate the carbon dioxide fire control system, only to find that the power had been cut off and a metal plate had been placed over the CO_2 controls so they could not be immediately activated. This became the genesis of the disabling of the multimillion-dollar plant.

I might say parenthetically that I wonder what odds Dr. Rasmussen might assign to such a chain of events occurring at one of the Nation's newest nuclear facilities.

On the broader issue of whether or not the fire could have resulted in the loss of effective control over the nuclear reactors, even more disturbing questions must be explored today. We would particularly like to find out which systems were available for cooling the reactors and which were not available at various times during the fire.

In this connection, there are indications that there may have been a period of several hours after the start of the fire when only one normally operating system capable of supplying cooling water was available to remove the heat that continued to be generated after the control rods were inserted to shut down the reactors.

The scope, magnitude, and implications of this fire are of considerable importance for the development of commercial nuclear power in the United States. The direct cost of the fire is estimated at about $10 million. The indirect cost of providing replacement power may run as high as $10 million per month.

The information presented to the committee raises serious questions concerning the quality of the regulatory review and inspection system. Prior to issuing a license to operate a nuclear power reactor, the NRC must find that "the facility authorized has been constructed and will operate in conformity with the application as amended in conformity with the provisions of this act and the rules of the Commission * * *"

Serious questions appear to be raised regarding compliance with the foregoing by the following:

a. The final design of the reactor which was approved apparently did not meet regulatory requirements.

b. The inspection procedures apparently did not reveal that certain aspects of the plant did not meet regulatory requirements.

c. Apparently the regulatory review process did not require effective emergency procedures either on the part of the applicant's on-station personnel or steps to insure that coordinated actions would be taken by State and local officials.

We have reports of the Nuclear Regulatory Commission,[1] the Tennessee Valley Authority,[2] and the Nuclear Energy Liability and Property Insurance Association (NELPIA),[3] which we will include in the record of today's proceedings, together with other related documents.

Our first witness today will be Chairman William Anders and his staff of the NRC, followed by TVA Chairman Wagner and his associates, and Aubrey V. Godwin, who will be representing the State of Alabama.

Before we proceed any further I would like to have incorporated in the record at this point an opening statement by Senator Baker who is attending another committee meeting and is unable to be here with us this morning at the opening of this session. The statement will be made a part of the record at this point.

[1] See Appendix 6.
[2] See Appendix 7.
[3] See Appendix 8.

[The prepared statement of Senator Baker follows:]

STATEMENT BY SENATOR BAKER

I want to thank the witnesses who are appearing before the committee today for their participation in the development of the record on a very important subject. I look forward particularly to hearing from representatives of the Tennessee Valley Authority which has long served the people of my State and region.

As a member of the Joint Committee and one of the congressional representatives of the area served by TVA, I personally visited and inspected the Browns Ferry site after the unfortunate fire. I am therefore looking forward to hearing and reviewing the testimony to learn how the details of the preliminary investigation compare with the impressions I may have formed on the basis of my visit to the site. I want to recognize and express my admiration for the devoted and heroic efforts of the TVA and other personnel who fought the fire under the most difficult circumstances and shut down the plants under extreme and extraordinary conditions and did so without any off-site damage to property or injury to people.

I am sure that there will be explanations in the hearing which would be viewed either as passing the buck for the cause of the fire and its regrettable economic consequences, or as Monday morning quarterbacking. But I would hope that the record which will be developed at these hearings will clearly show that the ultimate objective of them is to learn for the future by the experiences of the past. Certainly in an area such as nuclear reactor safety, it is important that all channels of commuication—with the Executive Branch, the nuclear industry, the Joint Committee, interested States and interested members of the public—be used so that nuclear power can continue to be developed safely and reliably.

Senator MONTOYA. Mr. Young, do you wish to make a statement?

Representative YOUNG. Mr. Chairman, you prefaced your opening statement by naturally saying what you understood the facts to be. I just want to say that you gave a very graphic description of what I understand the facts to be, too. I just wanted to ask, Chairman Anders, is that your understanding of those facts generally speaking?

Mr. ANDERS. The facts as portrayed by Senator Montoya generally reflect my understanding of the situation. Some of the conclusions the Senator expressed which were drawn from those facts might be discussed further. The purpose of our being here today is to bring out the facts that we have found to date and to suggest to you some of the conclusions that are emerging with regard to improving our own regulatory process that the Senator mentioned.

Representative YOUNG. I appreciate that and I understand that to be correct. I wanted to be sure we were getting off with a common understanding of the facts as they did occur, separate and apart from the conclusions that might be reached.

Thank you, Mr. Chairman.

Senator MONTOYA. You may proceed, Mr. Anders.

STATEMENT OF HON. WILLIAM A. ANDERS, CHAIRMAN, ACCOMPANIED BY DR. EDWARD A. MASON, COMMISSIONER; DR. DONALD F. KNUTH, DIRECTOR OF OFFICE OF INSPECTION AND ENFORCEMENT; BENARD C. RUSCHE, DIRECTOR, OFFICE OF NUCLEAR REACTOR REGULATION; AND DR. STEPHEN H. HANAUER, TECHNICAL ADVISOR TO THE EXECUTIVE DIRECTOR FOR OPERATIONS

Mr. ANDERS. Chairman Montoya and members of the committee, first I would like to introduce the other members of the Nuclear Regulatory Commission and the staff who are with me at the front table.

I have on my left Commissioner Mason, whom you have met before.

I have on my right Dr. Don Knuth, who is the Director of our Office of Inspection and Enforcement—that part of our organization conducting the investigation.

On his right is Mr. Benard Rusche, who is Director of the Office of Nuclear Reactor Regulation.

And on my far left is Dr. Steve Hanauer, who is Chairman of the Special Review Group which we will talk about here a little later.

Gentlemen, I plan to present a very brief overview of the fire that occurred at TVA Browns Ferry Units 1 and 2 located near Decatur, Ala. Members of the NRC staff will follow my remarks with additional details of the fire inspection and subsequent NRC actions. In general, we would plan, with your permission, to summarize the more voluminous and detailed remarks submitted for the record if you wish.

Senator MONTOYA. Your statement will be made part of the record if there is no objection and you may proceed to summarize your statement in any way that you wish.

Mr. ANDERS. Mr. Chairman, as for our more detailed and technical statements, they are quite voluminous. We would plan to summarize them if that is all right with you.

Senator MONTOYA. Without objection, you may do so.

Mr. ANDERS. In summary, the fire started about noon on March 22, 1975, by workers using a candle to test the effectiveness of an air seal around electrical cable penetrations between the cable-spreading room, which is located below the control room, and the Unit 1 secondary containment building. Air flow through the seal resulted in the fire spreading along the insulation on the cables into an adjoining equipment room of the Unit 1 secondary containment building. The fire burned in electrical cable trays for approximately 7 hours until it was extinguished.

Both operating reactors were manually shut down shortly after the fire started. We will have more detailed information on that from Dr. Knuth as we go along.

Damage to certain electrical power and control cables prevented the use of normal and some backup cooling systems, including ECCS components, which, of course, were not needed to perform their design mission. However, the reactor core was adequately cooled at all times, and thus, there was no damage to the nuclear fuel nor release of radioactivity as a result of the fire.

Damage was primarily to certain electrical cables, and was localized in an area roughly 40 by 20 feet in an equipment room within the Unit 1 secondary containment building. Outside this area, there was essentially no other damage, though some equipment required cleaning as a result of soot. There was no fire or soot damage within the primary containment of the reactor.

The Nuclear Regulatory Commission immediately initiated an extensive investigatory program, part of which has been completed, which we are here to report to you today.

Other elements are still being carried out and we will be talking about those.

I might add, Senator, that shortly after the fire I went down to the Browns Ferry site to get a firsthand view of the situation and also to insure that our investigation was proceeding adequately. I might also note that Senator Baker was there with me.

NRC INVESTIGATION ACTIVITIES

Now, our investigatory program consists of three major and essentially parallel efforts.

The first of these was an investigation to determine what happened at the Browns Ferry nuclear plant. This investigation was conducted by the Office of Inspection and Enforcement with the assistance of outside experts. It considered the events leading to the fire, the firefighting efforts, the sequence of operational events, and problems experienced with the reactors, the interaction between the two reactors, and the response of TVA, State, and local authorities.

This phase of the three-part effort has been completed and summarized in a report published July 25. This report has been made publicly available and widely distributed.

In addition, shortly after the fire, two bulletins[1] were issued to licensees of all operating plants to assure that they were aware of the Browns Ferry fire and that their attention was directed to areas of concern based on the initial evaluation of the fire. Dr. Knuth, Director of our Office of Inspection and Enforcement will present additional details of this investigation and related actions.

The second effort is being performed by the Office of Nuclear Reactor Regulation and has as its objective to assure that a safe plant configuration was attained and maintained subsequent to the fire.

Additionally, this effort is to assure safety during equipment removal and to approve design changes for restoration of these plants to operational status by TVA. The first part of this objective has been attained, and the last part is currently under way. Mr. Rusche, Director of our Office of Nuclear Reactor Regulation, will further describe the status of the licensing effort to restore these two reactors to an operating condition after you finish with Dr. Knuth.

The third effort concerns that of a special review group established to evaluate the results of the various investigations and other input in order to develop appropriate recommendations concerning improvements of a generic nature in NRC technical requirements, policies, and procedures.

I might add that in contrast to the fire investigation which was focused upon the Browns Ferry units in Alabama, this phase of the investigation is focused backward and introspectively on the Nuclear Regulatory Commission itself.

This effort is well under way and most of the technical evaluation of the causes of the fire and subsequent events have been completed. I expect their report documenting these recommendations to be published in early November. Dr. Hanauer, the chairman of the Special Review Group, will discuss the progress and direction of this effort in more detail as our last witness today.

As I noted, two of these major efforts, the licensing activity and the generic review are not yet complete and therefore the views at this time must be somewhat general and tentative in nature. But, we can state now that there was no nuclear fuel damage nor radiological consequences as a result of this fire and that the safety margins in-

[1] See Appendix 3 for inspection bulletins.

herent in the Browns Ferry reactors were sufficient to protect public health and safety—although they were not adequate to prevent localized damage within the plant and loss of generating capacity.

We believe that this unfortunate and serious occurrence has shown that the reliance on the defense-in-depth concept is sound for the protection of public health and safety. It is also our view that such a fire at another nuclear plant is unlikely, and even if one would occur, public health and safety is not likely to be affected.

Further, both the likelihood and consequences of any future fire should be reduced as a result of this fire investigation and related corrective actions. In this regard, our studies of this fire and its implications is expected to result in a determination that some additional plant measures and changed NRC practices are needed, as will be discussed in subsequent NRC testimony. Based upon the results of work to date, staff has concluded that there is no need for NRC to suspend or restrict operation at other nuclear powerplants.

Mr. Chairman, I would like to now introduce Dr. Knuth who will present further details concerning the investigation and actions by the Office of Inspection and Enforcement which he heads. As I said earlier, he will be followed by Mr. Rusche and Dr. Hanauer.

[The formal statement of Chairman Anders follows:]

STATEMENT OF WILLIAM A. ANDERS, CHAIRMAN, U.S. NUCLEAR REGULATORY COMMISSION

Mr. Chairman, and members of the Committee, let me first introduce the other Commissioners and the members of the staff who are with me.

I will present a very brief overview of the fire that occurred at TVA's Browns Ferry Units 1 and 2 located near Decatur, Alabama. Members of the NRC staff will follow my remarks with additional details of the fire inspection and subsequent NRC actions. In general, we would plan with your permission to summarize the more voluminous and detailed remarks submitted for the record.

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Both operating reactors were manually shutdown shortly after the fire started. Damage to certain electrical power and control cables prevented the use of normal and some backup cooling systems, including ECCS components, which, of course, were not needed to perform their design mission. However, the reactor core was adequately cooled at all times, and thus, there was no damage to the nuclear fuel nor release of radioactivity as a result of the fire. Damage was primarily to certain electrical cables, and was localized in an area roughly 40 ft. by 20 ft. in an equipment room within the Unit 1 secondary containment building. Outside this area, there was essentially no other damage, though some equipment required cleaning as a result of soot. There was no fire or soot damage within the primary containment of the reactor.

The NRC immediately initiated an extensive investigatory program, part of which has been completed. Other elements are still being carried out. The program consists of three major parallel efforts.

The first of these was an investigation to determine what happened at the Browns Ferry nuclear plant. This investigation was conducted by the Office of Inspection and Enforcement with the assistance of outside experts. It considered the events leading to the fire, the fire fighting efforts, the sequence of operational events and problems experienced with the reactors, the interaction between the two reactors, and the response of TVA, state and local authorities. This phase of the three-part effort has been completed and summarized in a report published July 25. This report has been made publicly available and widely distributed. In addition, shortly after the fire, two bulletins were issued to licensees of all operating plants to assure that they were aware of the Browns Ferry fire and that their attention was directed to areas of concern based on the initial evaluation of the fire. Dr. Knuth, Director of our Office of Inspection and Enforcement will present additional details of this investigation and related actions.

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Let me now introduce Dr. Knuth who will present further details concerning the investigation and actions by the Office of Inspection and Enforcement, which he heads. He will be followed by Mr. Rusche, Director of our Office of Nuclear Reactor Regulation, the group involved in assuring the safe design of the Browns Ferry reactors and approving the needed plant modifications, and then by Dr. Hanauer, Chairman of the Browns Ferry Special Review Group, who will discuss the generic implications.

Senator MONTOYA. Chairman Anders, may I ask you just three or four questions before we proceed with Dr. Knuth.

Now, are there any fundamental changes in design criteria or safety approaches called for as a result of this fire?

CHANGES EXPECTED

Mr. ANDERS. Mr. Chairman, as I suggested in my testimony and as will be discussed and summarized a little more graphically by the following speakers, we would expect some changes in design criteria as well as operational procedural changes to be identified for the Nuclear Regulatory Commission and our licensees.

When you use the term "fundamental," it seems to imply that we would expect to cause forced outages of the various plants that we license. We would not expect our required changes to be of that fundamental nature.

On the other hand, our analysis is not yet complete and we reserve the right to change our mind if more data emerge.

LENGTH OF INVESTIGATION

Senator MONTOYA. Now, the fire occurred, as I understand, in March of this year.

Mr. ANDERS. March 22.

Senator MONTOYA. And you state that the investigation is still going on. Now, why has it taken so long to conduct this investigation and why hasn't NRC concluded it and come up with some definite recommendation?

Mr. ANDERS. Without seeming to be argumentative, Mr. Chairman, I would say that this investigation is moving along very rapidly—some would suggest possibly too rapidly. This is a very complicated technology. These units are intricately designed. This fire, though very simple in its initiation, is very interwoven with the control and emergency and operational circuitry and electrical control of the reactor.

If one compares this investigation with others that I have been familiar with in the past—say in the aircraft investigation field—they take much longer.

We had a rather basic policy decision to make initially in the Commission as to how we would conduct the investigation. Should we—in the pursuit of the more traditional approach—say nothing until we had not only investigated the facts at the site of the accident; and then, drawn from these facts, to go through the considerably more difficult process of establishing to what the facts pertain—that is the kind of work Dr. Hanauer is doing—or should we make our findings available in a step-wise manner because of the interest in this event? Because of this accident's potential significance we felt we should break from the more normal and traditional approach and come out in this three-phase manner.

We believe that the investigation is moving along very efficiently and we would hope that the significant activity that Dr. Hanauer has under way would be out forthwith, but we will not rush it in the sense of moving too rapidly and drawing improper conclusions from a very complicated situation.

Senator MONTOYA. Well, do you have a time frame that you can tell us about within which you will complete the investigation?

Mr. ANDERS. As I suggested in my testimony, we would expect to have Dr. Hanauer's report to us in early to mid November. On the other hand, it is quite difficult to unravel the various complicated factors that were involved. As I think you will see as we get into the description of the event, it is quite complicated indeed.

Senator MONTOYA. Isn't it of the essence that you expedite this investigation?

Mr. ANDERS. Certainly.

Senator MONTOYA. And make certain recommendations. Because what affects Browns Ferry will affect other reactors in the country.

Mr. ANDERS. Yes, certainly. Of course, some recommendations are already apparent, Mr. Chairman, and we are expediting this aspect.

Senator MONTOYA. Are you passing on some of these recommendations to other people?

Mr. ANDERS. Yes. We are, of course, feeding all information which develops into our licensing group which Mr. Rusche heads. That information is feeding back into his analysis of the repair of Browns Ferry reactors and of course into other actions and licensing activities that they perform, as well as being used by Dr. Knuth's inspection and enforcement organization, which maintains cognizance over the operating reactors.

Senator MONTOYA. Now, which specific recommendations are you sending out to these other people, to these other reactor licensees throughout the system, throughout the United States?

Mr. ANDERS. There have been two bulletins released in addition to the report and we would expect more direction to be released as the generic process that Dr. Hanauer is conducting unfolds.

Senator MONTOYA. All right.

Would you submit a digest of those new instructions and recommendations for the record?

Mr. ANDERS. Yes, sir, Dr. Knuth, who will be testifying as soon as you like, has some of that information included in his prepared testimony which he has submitted for the record. If that is inadequate, we will certainly develop it further.

Senator MONTOYA. Also please include the very specific things that you recommend in addition to the digest.

Mr. ANDERS. Yes, sir.

[Information subsequently received follows:]

The information and recommendations provided to operating plants relative to the Browns Ferry fire are contained in the following Office of Inspection and Enforcement Bulletins which are enclosed for the record: [1]

IE Bulletin No. 75–04, Dated March 24, 1975.
IE Bulletin No. 75–04A, Dated April 3, 1975.

In addition, the Office of Nuclear Reactor Regulation has issued changes to the Browns Ferry Nuclear Plant Technical Specifications as a result of the fire-related safety analyses. These changes are detailed in Attachments 2 and 3 to Mr. Ben Rusche's prepared testimony.

NELPIA REPORT

Senator MONTOYA. Now, the Nuclear Energy Liability and Property Insurance Association conducted an internal investigation of the Browns Ferry fire, and their report [2] makes some strong recommendations for preventing and controlling the impact of future cable fires.

Would NRC comment as to what impact, if any, this report is having on its review of the situation?

Mr. ANDERS. Yes. Could I have Mr. Rusche answer that because he was more closely involved.

Senator MONTOYA. Sure.

Mr. RUSCHE. Thank you, Mr. Chairman.

We have had the opportunity to review the report to which you referred. I can't recall immediately, I believe there are some 35 to 40 recommendations made in the report. We certainly have studied those recommendations carefully and I believe as you hear my testimony and if you have the opportunity to review in detail the written testimony we prepared for the record, I think you will see that we have

[1] See Appendix 3, this volume.
[2] See Appendix 8 for NELPIA report.

concluded that some 20 or more of their recommendations clearly deserve adoption. On our own analysis we have included those kinds of recommendations in the studies and licensing effort we have made for Browns Ferry so we certainly have used the report. I might add that we have not had the report available for a long time.

Senator MONTOYA. Now, were you in this type of work checking these reactors before ERDA came into being and before the Nuclear Regulatory Commission came into being?

Mr. ANDERS. Mr. Chairman, Mr. Rusche was brought onto the Nuclear Regulatory Commission staff at the senior position he now holds after the reorganization. He comes from an extensive nuclear engineering management and environmental background. Dr. Knuth, whose organization actually conducted the reactor inspections, was with the Atomic Energy Commission prior to that time.

Senator MONTOYA. Now, the initial inspection of the TVA plant was made on July 17 to 18, 1967. This is from Dr. Knuth's testimony. As of March 20, 1975, before the fire, 78 AEC or NRC inspections had been performed at Unit 1, 73 at Unit 2 and 58 at Unit 3.

NELPIA CRITIQUE

Now, the reports of these inspections are publicly available and they document the inspection findings, items of noncompliance identified and the resolution of enforcement actions that were taken. I would like you to amplify upon these and also comment on your inspections in light of the report of the Nuclear Energy Liability and Property Insurance Association and more specifically their particular and well-noted critique which is as follows:

1. The cable spreading room (CSR) at Browns Ferry was considered to be poorly designed from an operational and fire protection standpoint.

2. The congestion in this CSR was inexcusable.

3. Such a massive array of cable trays in the absence of aisles avoids any realistic fire-fighting effort while subjecting firemen to possible injury or even loss of life.

4. Unless automatic fixed fire extinguishment is provided to protect against the hazards inherent in these rooms, a loss beyond imagination should be anticipated.

To overcome such a catastrophe a first-class, designed and approved, automatic fire protection system is needed.

Now then, there are some recommendations with respect to design which I will not go into at this point. In light of the many inspections that were conducted, in fact I have a number here, 209 inspections——

Representative YOUNG. Excuse me, Mr. Chairman. That was prior to the fire?

Senator MONTOYA. That is right, prior to the fire.

Now, why didn't these inspections detect these deficiencies and why was something not done by way of enforcement of any recommendation that you might have made in response to any findings pursuant to these inspections?

Dr. KNUTH. Mr. Chairman, Dr. Knuth from the Office of Inspection and Enforcement.

I would like to try to answer that question specifically. I do have, of course, the data and the background summarized in my testimony as to the part of the Office of Inspection and Enforcement. Our philosophy in the inspection arena is to examine the performance of licensees to insure that they are meeting regulatory requirements as imposed by the regulations and license conditions that appear in the technical specifications of the license.

Our philosophy is to take a sampling of selected areas for an inspection, and to do an in-depth inspection in those areas to find out if the licensee is adhering to the requirements.

Senator MONTOYA. Let me ask you this question at this point then. Do I understand from your testimony that you are saying that the inspection system is strictly designed to make sure that the people who are operating the reactor are complying with the regulations then in force which you have promulgated and which they are subject to?

Dr. KNUTH. Yes, sir.

Senator MONTOYA. Now, shouldn't your inspection be extended beyond this sphere to find out what defects exist that are not covered by the regulations?

Dr. KNUTH. Yes.

Senator MONTOYA. Did you find any defects during your inspections that might have called for stringent enforcement measures on your part?

Dr. KNUTH. Our inspection history of the Browns Ferry and other plants during this period of time show that we did inspect against the cable separation criteria. We selectively sampled certain areas of the cable routing to make sure that the circuits were separate and met the separation criteria, but in our sampling program that was in effect at the time of Browns Ferry we did not specifically look at fire protection equipment.

The fire protection inspections, of course, in commercial nuclear powerplants are done by NELPIA and insurance companies who do look at fire protection and they are looking at it mainly from the aspect of property damage and protecting property.

Senator MONTOYA. Are you going to continue to depend on their inspections or are you going to provide for in-house inspection by the NRC?

Dr. KNUTH. Let me just continue with my other thought. In the case of Browns Ferry, they are self-insured. In that the NELPIA fire underwriters did not inspect their plant, they are self-inspected as a Government agency.

To respond to your second area, in our inspection program which we now have before us, yes, we have included selective sampling of fire protection equipment as part of our routine program but again it is a selective sample. It is our philosophy again to make sure that licensees are doing their job in management—in exercising their management responsibility—so it is an in-depth inspection.

Mr. ANDERS. The short answer is yes, Mr. Chairman.

Representative HORTON. Mr. Chairman, on that subject would you yield?

Senator MONTOYA. Yes.

Representative HORTON. I just really have one question to ask on that subject with regard to the recommendations as to the cable fire.

Has there been a recommendation that we eliminate the inspection of these types of leaks by candles?

Mr. ANDERS. Yes.

Dr. KNUTH. Yes.

Representative HORTON. Could somebody spell that out for us briefly?

Dr. KNUTH. Yes. We issued a bulletin [1] essentially the first working day after the day of the fire, the 24th, which did indicate the cause of the fire was by a candle. In our bulletin we emphasized the control of fires not only from candles but from welding operations, cutting operations, wherever you might initiate a fire. They should be a part of the procedure for the close control—work permit control—of the use of open flames.

LEAKAGE INSPECTION TECHNIQUES

Representative HORTON. How do you inspect these types of leaks now? I would think with all of the technology we have, we ought to be able to devise something better than a lighted candle.

Dr. KNUTH. Yes. There are smoke generators. There are a lot of techniques available that do not use an open flame.

Representative HORTON. That has been recommended by NRC?

Dr. KNUTH. Yes, sir.

Representative HORTON. Thank you, Mr. Chairman.

Representative PRICE. But up to this time has the use of candles been a general practice?

Dr. KNUTH. In the utility industry in general the use of a candle for detecting air leaks into condensers was a fairly common practice. I believe there were a number of utilities that used this technique not only in checking the condensers for vacuum but checking leakage in other areas. I would not say it was common but it was not an uncommon practice.

Mr. ANDERS. Mr. Chairman, just to pin down the fundamental question that you asked about, that is whether we are evaluating our procedures, I don't believe I fully answered. Indeed, we are doing that. In addition, I want to make it clear that our present inspection philosophy, the one that has been used for years and has been developed by the inspection and enforcement staff, follows from the approval of the reactor design and of the utility's proposed quality assurance system by our licensing section. Once that approval is made and the plant is licensed for construction, the licensee's quality assurance system is set up and operating, our inspector's main task under the philosophy we operate is to inspect that quality assurance process.

Now, additionally, in their inspections of the plant which are focused on the utility's quality assurance management as opposed to quality assurance product, the inspectors do witness activities and look for unusual situations in the plant.

In the past we have relied on the independent fire insurance underwriters as third party inspectors to review the firefighting equipment and firefighting procedures. As Dr. Knuth has mentioned, the TVA

[1] See Appendix 3.

as a self-insurer did not have this outside activity. As a result of our review of this fire, we have expanded our present inspection activities to include firefighting procedures and equipment in all plants. We have the basic question of just how much we should do in this area vis-a-vis the third party fire insurance underwriters to implement our overall mission of protecting public health and safety.

Senator MONTOYA. Yes, but are you going to rely strictly on them when we have a national responsibility of your organization to undertake whatever inspection might be necessary even though it might be to inspect the inspection activities of private industry?

Mr. ANDERS. Well, certainly we would not do it strictly that way, Senator, and that is what I meant by saying that we have expanded our present activities to include a more thorough inclusion of the firefighting equipment and procedures. Just what the appropriate balance between the NRC activity and the very highly qualified activity of the fire underwriters where they actually do their inspection is something that we want to think about more and possibly talk to you more as we proceed to strike that balance.

Of course, the problem here is that those third-party investigators were not involved with TVA plants because TVA is self-insured.

DELAY IN USE OF WATER

Representative ANDERSON. Mr. Chairman, may I ask a question at this point?

I have read the charge that this fire could have been put out in 20 minutes, but that it burned for 6 hours because the people who were fighting the fire insisted on using, I think it was, carbon dioxide and chemicals and that had they used water as was recommended and to which the plant superintendent would not give his consent initially, that it would have been put out, as I say, very promptly.

Were there instructions in the event of fire that required this plant superintendent to take this apparently erroneous position that he did?

Mr. ANDERS. Mr. Anderson, those factors are among the information we planned to bring out in a little more structured way by the oral summary.

Representative ANDERSON. I appreciate that, but we seem to have wandered off in the area of asking all kinds of questions so I thought it would not be inappropriate in view of the fact that we are talking about this general subject of criteria for fighting fires.

Mr. ANDERS. Let me give you at least a partial answer to your question. Our investigators in their report, laid out the facts and time sequence of the fire damage, this report noted that water was used on the fire at approximately 6 or 7 hours after it started and it went out. One conclusion you could draw from this is that water should have been used on the fire earlier. The licensee's considerations of why water was not used or was used is best asked of them. It was, as I understand, an allowable firefighting agent in their procedures. As I understand it, their concern at the time was that possible additional shorting of electrical control cables might have occurred during the more critical early phase of the fire when the plant was being shut down and cooled down.

So whether it was right or wrong, may not be answered since we will never see the alternate course in history. Some people have one view, some have another. I would be reluctant to draw a conclusion based on the information I have seen so far.

Representative ANDERSON. But you say subsequent witnesses are going to address themselves.

Mr. ANDERS. Yes, and I would be——

Representative ANDERSON. It seems to me pretty basic that your licensees have some knowledge as to how they fight one of these fires.

Mr. ANDERS. Yes. Personally I don't think there was any unawareness that water is generally not used in fighting electrical fires. The question is what is more important at the time, putting out the fire quickly and possibly exposing your system to more shorting by the presence of water or getting the system shut down and letting the fire burn a little bit longer?

Representative ANDERSON. Thank you, Mr. Chairman.

Senator MONTOYA. Did you have any instructions with respect to the use of water and in what circumstances it could be used?

Dr. KNUTH. No, we do not have instructions in our own internal organization that requires licensees to use or not use water. It is a decision that the licensee makes.

Senator MONTOYA. Why don't you?

Mr. ANDERS. Mr. Chairman, there is an infinite combination of events associated with problems in firefighting and our policy has generally been to support the various rules of good practice established by such organizations as the Fire Underwriters, and the American Society of Electrical Engineers.

Senator MONTOYA. What is their position on that?

Mr. RUSCHE. Mr. Chairman, I doubt that the particular organizations to which you refer make a specific recommendation that is applicable to every case. We are looking into this, however, in the review of the restoration of the plant considering design features and I will discuss these in my testimony, but we are going to recommend and require that Browns Ferry have water available for fighting fires in this area as well as in the adjacent room in the reactor building.

Senator MONTOYA. Does the Insurance Underwriters Association that we have mentioned heretofore agree with that recommendation?

Mr. RUSCHE. I believe that is so.

FLAMMABILITY OF POLYURETHANE

Senator CASE. Mr. Chairman, since we are on this question I wish you would comment on why this material is used since it is highly flammable.

Mr. ANDERS. One of the things that Dr. Hanauer's group is reviewing is just what information is available regarding fire retarding and fire insulation material and to compare that against our specifications which may possibly be inadequate in that respect.

Senator MONTOYA. Well, following Senator Case, it would appear that previous lessons in fire prevention relative to the use of polyurethane have not been learned by your organization and this raises concern over whether other matters are improperly handled.

Now, specifically the AEC reported in a health and safety bulletin [1] in December 1963 that foamed polyurethane fires are difficult to control. Recommendations made then seem directly applicable to Browns Ferry but there is no evidence that they had ever been considered by NRC.

Now, the 1963 incident report pointed out the need for (a) a better appreciation of polyurethane fire risks by all personnel involved, (b) smoke ejecting equipment and additional portable breathing equipment, (c) prompt notification of the fire department in case of any small fire and (d) consideration of attaching polyurethane so that it can be removed more easily in case of fire.

Now, what comment do you have on that?

Mr. ANDERS. Well, the report recommendations in this area stand and I support them. I would point out that this material can perform a difficult mission in the sense of sealing intricate flow paths around control cables going into reactor containment and therefore has merit for use there. The question is, is such merit for use outweighed by the points that you have raised and as mentioned in that report?

TVA was required to test the particular material that had been proposed by the designers—that is foamed polyurethane.

They were also required to put a fireproof coating over this material. This specification did not approve, as I understand it, the use of sheet polyurethane or rags or whatever in stuffing the air flow paths in order to prevent the flow temporarily.

In retrospect, there was at least one aspect of this test which was not adequate and that was that it was not conducted under the similar air flow conditions. So at least there was an attempt to ascertain whether the points made in that report were completely applicable to this job of insulating or packing the passageway for the cables.

This whole subject, of course, is under review by Dr. Hanauer's group.

Senator MONTOYA. There have been previous incidents where fires occurred with respect to cables. We had in Peach Bottom Unit No. 1 in February 1, 1965, a cable fire that was started as a result of sparks from a welding operation. In San Onofre Unit No. 1 there were two fires in 1968 and both involved cable trays. In the Peach Bottom fire, polyurethane was involved.

Now, in light of those occurrences, why wasn't something designed or promulgated by way of regulation to establish a better design for the containment of these cables?

Mr. ANDERS. Mr. Rusche can comment in detail on the design of the containment or cables and plans to in his summary.

I might say, though, that the requirements of the NRC criteria which were received from the AEC, does take into account fire considerations and of course certain firefighting systems.

I might add, though, before I pass the question to Mr. Rusche to answer in detail, that the record of fires in the nuclear industry has been quite good. This is an extremely sensitive area of the facility with massive amounts of wiring. This industry has been operating successfully for years and Browns Ferry represents the most serious fire by a good measure that has occurred to date. I think it would be unreal-

[1] See Appendix 12.

istic to expect that you would not have fires in electrical cabling whether it would be in a nuclear or nonnuclear plant.

Mr. Rusche, do you have anything to add?

Senator MONTOYA. I want to add also the cable fire in the Salem Unit No. 1 nuclear plant. That was in April 1974.

Now, in light of all these, Dr. Knuth, just tell us why there were really no precautionary measures taken to warn people operating reactors on how to handle these fires.

Let's also take into consideration that you are now talking to this subject after 209 inspections—after 209 inspections—and after these three and perhaps more fires.

Dr. KNUTH. Yes. Well, I would answer by saying that the fact that the fires occurred, the fact that certain materials were involved in these fires were widely publicized at the time of the fires. They were sent by way of what were called reactor operating experience documents back in those days to make known that the fires had occurred.

Of course, the cable separation criteria were established to separate the redundant systems into separate systems, partly as a consequence of these fires.

Now, in hindsight you can say another one occurred but those are the facts.

Senator MONTOYA. Well, please provide for the record at this point just exactly what you have done by way of increasing your precautionary measures and by way of advising the industry as to what to do under these circumstances and more especially what to do when you are faced with cable using a polyurethane inner sheath.

Now, would you supply for the record just exactly what you did during these years when these fires were occurring and what regulations you promulgated; what changes of design you recommended, what advice you gave to the industry; and what enforcement procedures you launched to see to it that your recommendations were carried through.

[Information later supplied follows:]

The results of previous cable tray fires, starting with the Peach Bottom 1 fire and the San Onofre fire in 1965 and 1968 respectively, were reviewed and considered by the NRC staff in the formulation of staff positions on separation and isolation criteria which were then applied to ongoing and subsequent reviews. Review efforts were also directed to assuring that existing industry standards were being applied, e.g., that cable ampacity and derating were in accordance with IPCEA (Insulated Power Cable Engineers Association) standards and that overload and short circuit protection were in accordance with the National Electric Code. Additional information requirements addressing cable installations were developed and issued in the form of Information Guide 2 (1971) and the Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants (October 1972). Concurrently, NRC staff efforts were directed towards further development and refinement of cable separation and isolation criteria. These efforts culminated in the issue of Regulatory Guides 1.75 (February 1974) and 1.81 (June 1974) and IEEE Standards 383–1974 and 384–1974 which are presently being applied. The Browns Ferry fire has shown that the criteria and standards developed as a result of previous fire experiences were not adequate. Efforts are underway to upgrade these criteria and standards.

Dr. MASON. If I may comment.

The practice has been to make flammability tests on single wires. Even before the Browns Ferry fire we had concerns as to whether the configuration of a number of wires and cables to provide the same assurance against ignition and then later flammability and we have tests under way now to do this.

So, in addition to the directives that have been given to licensees for some time, we could have been concerned about this but our insulations are variously classified under various categories of flammability rating. By hindsight greater assurance against fires of this type, especially when put into these packed configurations and trays, is needed, and we are looking at that in addition to the formal orders we are giving.

Senator MONTOYA. Congressman Price.

Representative PRICE. Most industrial organizations have continuing programs on fire prevention, and on how to handle a fire when you do have one. Is there any such program in any of the nuclear complexes?

Dr. KNUTH. Yes. In fact, the TVA did have such a program in that their plant protection force was charged with fighting fires and they conducted periodic drills in fire fighting. So, to answer your question, yes, the nuclear powerplants do have training sessions and they do have sessions on various classes of fires and what to use to extinguish them.

Senator MONTOYA. I would like to ask Dr. Knuth this question.

Did your inspectors ever inspect this room where the fire broke out before the fire took place?

Dr. KNUTH. Yes. Our inspectors inspected the room and as part of the inspection procedures they looked for cable separation. As a result of previous fires, critera setting minimum separation were established for redundant cables supplying safety equipment, and part of our inspection program was to inspect to make sure that the separation criteria that were required of the licenses were being met.

Senator MONTOYA. How many times did your inspectors enter into this room?

Dr. KNUTH. I will have to supply the answer for the record. I don't know how many times they entered the room.[1]

Senator MONTOYA. Will you supply a copy of the inspection report pursuant to their entering the room.

Dr. KNUTH. Yes, sir.[2]

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[1] NRC has advised the Committee that records are not maintained of visits to specific locations within a plant, but NRC estimates there were a minimum of about 10 visits "to these areas." They were unable to provide the specific number of times that NRC inspectors visited the room prior to the fire.

[2] Based on information provided by NRC, there are no specific inspection reports available on the cable spreading room where the fire began. However, NRC has provided 37 reports of inspections conducted over the period of September 1969 through January 1975 that include information related to cables. Due to their voluminous nature, these reports have not been included in this print, but are being retained for public inspection in the Joint Committee's files.

Senator MONTOYA. And also any recommendations that you might have pursuant to such inspection.

Dr. KNUTH. Yes, sir.[3] [4]

Senator CASE. Mr. Chairman, on that question it is my understanding that there is not any redundancy in this matter of cable separating rooms. Is this not a matter which ought to be very carefully reconsidered as to desirability, as to cost and as to feasibility? We are supposed to be protected by redundancy in these plants.

Mr. ANDERS. Yes, Senator. There are criteria for cable routing related to the question of redundancy. Of course, if you have a combined control room for two units, then you have to have, by definition, a combined cable spreading room to spread the cables out into not only one unit but also to separate the two units as the cables flow out.

Senator CASE. Is that done?

Mr. ANDERS. That is the situation at Browns Ferry. There is a combined control room and underneath that room the cables go out to the various units. Now, there are criteria to separate the control circuits of each individual unit, the redundant control circuits of Unit 1 and the redundant control circuits of Unit 2. If these circuits come together it causes not only interlocking of the two units but may also involve the interlocking of the redundant systems of either one of them.

Senator CASE. So that is redundancy.

Mr. ANDERS. Yes, though it is reduced somewhat in that regard.

As part of the investigation that Dr. Hanauer's group is conducting, our criteria will be reviewed to see if they are adequate to provide the kind of safety required where redundancy is reduced to some degree.

Senator CASE. I would ask, Mr. Chairman, that when the report is made to us finally, it go into this very specifically.

Mr. ANDERS. It will.

Senator CASE. Including what it would cost to retrofit new plants in the country with separate cable spreading rooms for each reactor unit.

Mr. ANDERS. We will include our estimate of what the cost is. I would expect it to be a massive cost.

Senator CASE. This would be a massive disaster, too, if you don't have true redundancy right through it.

Mr. ANDERS. Yes, sir. The principle of redundancy is one I feel reasonably familiar with and it is certainly something that one should work toward. However, I cannot think of any operational system that can be completely redundant through all the systems.

Our job will be, of course, to try to reduce the exposure to single point failure in nuclear powerplants as we do in other high technology systems.

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[3] NRC did not list any specific recommendations made pursuant to inspection of the cable spreading room. They advised the Committee that "Specific recommendations to, or requirements on, licensees resulting from these inspections are discussed in the inspection reports and associated transmittal correspondence with TVA."

[4] The reader is referred to page 180 for the specific information provided by NRC in response to Senator Montoya's questions on this matter.

Senator CASE. I think everybody thinks that even if you are not redundant once or twice or three times that you still have defenses in depth.

Mr. ANDERS. Certainly in some cases you are. Where you have to reduce the concept of redundancy, you have to provide other safety systems.

Senator CASE. I do think this is a very important factor and I am sure that we will get a full report on it, including the whole question not only of cost but the feasibility of other methods that are used if redundancy doesn't exist.

Mr. ANDERS. Yes, sir, I am certainly in sympathy with that thought.

RASMUSSEN REPORT

Senator CASE. Then another point. Again, I know that we are jumping in on you ahead of time but most of us are pretty darn busy and we want to have the questions answered for the record.

The Rasmussen report on reactor safety assumes that this is going to play a large role in safety. This plant came pretty close to melt down but the local civil defense director didn't learn about it until 2 days later. The local sheriff said he only learned about it 2 days after it was out. I think this is not good to say the very least and I would be glad to know what your plans are to avoid them in the future.

Mr. ANDERS. Senator, I don't want to sound argumentative but I would disagree with your statement that it came pretty close to melt down. Regardless of that——

Senator CASE. Quite regardless of that, and you can call it a trivial thing but in any event the same thing would have been true if it had been a much more major thing.

Mr. ANDERS. Our system, in my mind, should provide for adequate alerting of local authorities and State authorities for an event or an accident of this kind. We are prepared to discuss it more fully with you. There are members of both groups here, not only TVA but the local authorities.

Dr. Knuth, maybe you would like to add a couple of points to Senator Case's question.

Dr. KNUTH. Yes. The Alabama authorities will be here later this afternoon. The notifications that did occur are documented in our investigation and in our written testimony stating who was notified and the time scale. We did, as you indicate, find that there were, in our view, some lapses and it could have proceeded better, but on the other hand, the people who made the decision to make the call didn't at the time they made the call believe that there was going to be a release of radioactivity. It was a precautionary call and indeed hindsight shows there was not a release of radioactivity. However, they were calling people and we do see the need for improvement.

Senator CASE. Thank you, Mr. Chairman.

Senator MONTOYA. Dr. Knuth, you may proceed with your statement. If it is all right, we will insert your statement in the record. You may proceed to summarize your statement and then we will ask you some questions unless the members here have other questions at this point.

Representative ANDERSON. I will defer.

Representative YOUNG. Mr. Chairman, you covered in your previous questions the major points I was concerned with. I was glad to see the chairman correct the record with regard to this major aspect of the melt down because I don't think the record at this point bears out the hypothesis that a melt down nearly occurred.

Senator MONTOYA. Dr. Knuth, you may proceed.

STATEMENT OF DR. DONALD F. KNUTH, DIRECTOR OF THE OFFICE OF INSPECTION AND ENFORCEMENT, NRC

Dr. KNUTH. Mr. Chairman and members of the committee, during the next few minutes, I will briefly summarize those aspects of the Browns Ferry fire which are under the scope of responsibility of the Office of Inspection and Enforcement. It should be noted that because of its very nature, our investigation concentrated mainly on the negative aspects of the fire and so, therefore, will this testimony.

The historical background of the Browns Ferry Nuclear Plant; the chronology of events on the day of the fire; and the actions of the Office of Inspection and Enforcement are set forth in greater detail in my written statement and the investigation report,[1] both of which have been previously provided to the committee.

The Browns Ferry Nuclear Plant, located on the north shore of Wheeler Lake, which is part of the Tennessee River, near Decatur, Ala., is owned and operated by the Tennessee Valley Authority (TVA). The plant comprises three boiling water reactors designed by the General Electric Co. Each unit is rated at 3,293 megawatts, thermal, and can produce approximately 1,100 megawatts of electric power at design capacity.

In March of this year, an extensive program was underway for resealing cable penetrations between the cable spreading room, located beneath the control room for Units 1 and 2 and the secondary containment building. The resealing operation involved stuffing pieces of plastic polyurethane foam into each affected cable penetration, followed by installation of either liquid or spray-type polyurethane foam. A final step involved application of a fire-resistant material.

The method used by TVA to check the effectivness of the sealing operation was to hold a lighted candle near the penetration opening to see if air flow from the cable spreading room to the secondary containment building existed. If the opening was not fully sealed, the lower pressure in the secondary containment building would cause a draft through the opening and thus cause the candle flame to be deflected toward the opening.

CHRONOLOGY OF THE FIRE

On March 22, while both Units 1 and 2 were operating at 100 percent of rated power, this leak-checking technique was in use in connection

[1] See Appendix 6.

with the resealing operation. The flame of the candle used was pulled strongly into the opening and ignited a fire within the penetration at approximately 12:15 p.m. Prompt attempts to extinguish the fire locally were unsuccessful. The fire was detected by plant personnel in the secondary containment building only a few minutes after it was ignited. However, fire-fighting efforts exerted in the secondary containment building were frustrated by the inaccessibility of the fire, the dense smoke and the limited availability of breathing apparatus for use by personnel fighting the fire. TVA has stated that water was not used on the fire during the initial fire-fighting efforts because of concern about personnel injury from electrical shock and possible adverse impact on operating systems.

A decision was made about 45 minutes after the fire started to initiate operation of the carbon dioxide fire extinguishing system installed in the cable spreading room. Some temporary difficulty was encountered in initiation of that system, but this difficulty did not have significant impact on the ultimate outcome of fire fighting activities. The installed carbon dioxide system was not effective in extinguishing the fire and by this time the coverings on cables in the penetration had been ignited and the fire had burned through the penetration into the secondary containment building.

The fire in the cable spreading room was extinguished about 4:30 p.m. but for a period of about 3½ hours there was little or no coordination of effort to fight the fire in the secondary containment building. Shortly after 1 o'clock, assistance was requested of the local fire department from Athens, Ala.; they arrived at the plant about 1:30. The Athens Fire Chief recommended that water be used to fight the fire about half an hour after arriving at the scene. Permission to use water to fight the fire was given by TVA management between 5:30 and 6 p.m. About 7 o'clock water was directed at the fire. After a period of about 15 minutes, there were no further observations of burning. The fire was declared to be extinguished at 7:45 p.m. In our opinion, the delay in the decision to use water in fighting the fire contributed significantly to the extent of the damage incurred.

The fire damage extended from the point of ignition in the cable penetration in one direction a few feet into the cable spreading room and in the opposite direction along the cables about 40 feet into the secondary containment building. All 10 cable trays in the penetration and over 1,600 cables were affected, including cables in 26 trays in the secondary containment building.

EFFECTS OF THE FIRE

Let us now rezero the clock and consider the highlights of the effects of the fire on operation of Unit 1.

In the common control room for Units 1 and 2, the control room operators were notified of the fire about 20 minutes after the fire started. About 5 minutes later, the Unit 1 operator noted anomolous behavior of controls and instrumentation for systems designed to provide emergency cooling of the reactor core. About 10 minutes later, the main recirculating pumps on Unit 1 stopped operating and the operator immediately shut down the Unit 1 reactor by activating the control rod drive system, and he verified that all control rods had been fully inserted.

Shortly after 1 o'clock, the normal methods of providing high pressure cooling water to the core became unavailable. TVA used one of two electrically driven pumps readily available in the control rod drive system to provide somewhat over 100 gallons of water per minute to the vessel. This was insufficient to maintain the required water level in the reactor vessel.

As the pressure in the Unit 1 reactor coolant system increased as a result of steam produced by radioactive decay heat in the core, the relief valves opened automatically to release steam, thus preventing overpressurization of the reactor coolant system.

When system pressure was reduced sufficiently, low pressure pumps were operated to maintain water level in the reactor vessel. This mode of core cooling was successful until about 6 p.m., when control was lost for the four operating relief valves. This caused these relief valves to close, resulting in repressurization of the reactor coolant system. Because of this repressurization, the low pressure pumps could not continue to supply water, and the operator chose to rely on one control rod drive system pump as the source of coolant injection for a period of about 4 hours until temporary modifications re-established the capability to open the four affected relief valves and reduce system pressure.

About 13 hours after the fire started, temporary repairs were made which permitted cooling of the pressure suppression pool, and normal means of cooling the Unit 1 reactor core were restored by about 4:10 a.m. on March 23.

The effect of the fire on Unit 2 operation was less pronounced. The Unit 2 control operator shut down the reactor manually at approximately 1 o'clock when numerous alarms occurred, most of which were associated with loss of d.c. power, and verified that all control rods has been inserted fully. Since there were no major problems encountered in cooling the Unit 2 reactor, I will not repeat the sequence of events of Unit 2 set forth in my written statement.

The Browns Ferry Emergency Plan was activated at about 1 o'clock when it became clear that operation of the Units 1 and 2 reactors had been affected by the fire. At that time, it was not known whether or not there would be offsite consequences as a result of the fire. Actually, there were none, as we later determined. Although we later identified some deficiencies in the execution of the plan, they did not contribute to the effects of the fire.

The Atlanta Regional Office of the NRC's Office of Inspection and Enforcement was notified of the fire shortly after 4 p.m. on March 22. Inspectors from the Regional Office were dispatched promptly to the Browns Ferry site. The Regional Office immediately notified NRC Headquarters, and key members of the NRC's Offices of Inspection and Enforcement, Nuclear Reactor Regulation and Public Affairs were assembled. Other cognizant Federal agencies were alerted and contact was maintained with the Regional Office and NRC inspectors at the Browns Ferry site. An investigation was initiated immediately by the Atlanta Office. The field investigation required 280 man-days of effort. It was completed and a detailed report thereon, dated July 25, 1975, was prepared and released publicly on July 30.

Shortly after the Browns Ferry fire, the Office of Inspection and Enforcement prepared and distributed to all power reactor licensees a bulletin (75–04) and supplement (75–04A) [1] requesting generalized information on policies and procedures relating to work control practices at operating plants, particularly as they relate to fire prevention.

Senator MONTOYA. Mr. Murphy has a question at this point.

Mr. MURPHY. Dr. Knuth, was there any onsite radiation above normal level?

RADIATION LEVELS

Dr. KNUTH. There was no radiation that was above limits. The highest radiation that I am aware of that occurred was airborne activity which reached about 35 percent of the allowable maximum permissible concentration inside the reactor building but the direct radiation levels were not above normal, no, sir.

Mr. MURPHY. The reason why I asked the question was in your report it says that at 6:35 p.m. health physicists reported to the CECC that the radiation levels in Unit 1 control room were increasing and that some of the individuals in the room did not have respirators. You went on to say the director, CECC was recontacted at 6:50 p.m. and advised that the turbine building activity levels were increasing.

Subsequently there is a statement on the following page that the CECC related that radiation levels had dropped below mask requirements.

Dr. KNUTH. I believe the radiation levels increased in the control room but they were at a low level. They did not exceed the maximum concentrations allowable in any of the spaces inside the reactor building or the control room.

Mr. MURPHY. Are you going to describe the various fans and offsite radiation detection devices that were not operable or samples that were not obtained?

Dr. KNUTH. I had not planned to go into that detail. It is a part of our written investigation report and I could refer you to the page of the report for these details. It is in section II of our detailed report and it is in about 10 pages of discussion.

Mr. MURPHY. Thank you, Mr. Chairman.

[The information later supplied follows:]

During the course of firefighting activities, radiation monitoring of personnel involved was provided using portable, hand-held beta-gamma instruments to survey the work areas, and normal film badging and dosimeters. In addition, post-event whole-body counting of potentially affected personnel was conducted. This particular aspect of the event was handled in a straightforward application of routine radiation control procedures in effect at the Browns Ferry plant. There were no problems associated with direct radiation affecting the workers.

The Unit 1 reactor building (secondary containment) ventilation system was inoperable as a result of the fire from about 12:45 p.m. until about 4:00 p.m. The Unit 2 system was inoperable from about 2:00 p.m. until about 9:00 p.m., when it was restored to operation. During the period when the reactor building ventilation systems were out of service, "grab" samples of air in the working area were taken approximately each hour. Analyses of these samples indicated the presence of Rubidium-88 (a daughter product of Krypton-88 with a 17-minute half-life) in concentrations of about 35% of levels permitted by NRC regulations. No other radioisotopes were detected. It should be noted that the requirement for use of self-contained breathing apparatus was dictated by the presence of

[1] See Appendix 3 for inspection bulletins.

noxious and toxic combustion products—not by any need for protection from airborne radioactivity.

Smear samples taken in the work area showed no surface radioactive contamination as a result of the fire.

Samples taken hourly from the reactor zone stack effluent revealed a maximum release rate of slightly over 9,000 microcuries per second (compared to a Technical Specification limit of 130,000 microcuries per second). Particulate and charcoal filter samples were also well below limits set forth in the Technical Specifications. Air particulate samples were taken from the plant environs; they revealed nothing in excess of results of routine environmental monitoring.

Further details are available in the NRC Investigation Report (Section J, pages I–13 through I–18).

Senator MONTOYA. If the members have any questions as we go along, I would allow such questions.

Representative ANDERSON. At this time?

Senator MONTOYA. At any stage of his presentation or the summary.

Representative ANDERSON. I have a couple of questions and I think they are fairly brief, Mr. Chairman.

Senator MONTOYA. All right.

TIMING OF REACTOR SHUTDOWN

Representative ANDERSON. In examining the chronology of this fire, the thing that is amazing to me, if that is the proper word to use, is why these reactors continued apparently to operate at full power for a full 45 minutes after the fire was discovered. Have new operating procedures since been put into effect that would call for the immediate shutdown in the event of the discovery of a fire in the control room?

Dr. KNUTH. The procedures that are in effect do require that the operator immediately shut down a reactor when he has a safety concern for the continued operation of the plant. In this particular case, the fire started in a room underneath the control room and the operators actually were not aware of the location of the fire or the specifics of it until some 20 minutes after the fire was initiated.

Any situation where something anomolous is occurring and the lights are appearing or coming on indicating something is going wrong, requires evaluation. In this particular case, the operator did not elect to shut down the reactor; it was a decision he made.

Again, in hindsight you may question that but they are instructed that if there is a concern for safety the plant is to be shut down and put in a safe mode.

Representative ANDERSON. It seems to me that is an awfully loose criterion. If the operator has a concern for safety, he is instructed to shut down the plant. Obviously nobody can tell, I suppose, the moment a small fire starts whether it is going to just smolder and remain a small blaze or whether it is going to become a larger fire. I would think, and realizing somewhat as I do the consequences of shutting down a reactor, that it is expensive and loss of power can have certain consequences to be sure. But given the subject matter that we are dealing with here I just find it incredible that an operator would be under that very loose, broad general guideline.

Well, if you feel there is a safety problem, shut down the reactor. Or when you have a fire in a strategic place—you say right under the control room—it seems to me that that ought to give any reasonable person cause for concern and yet for 45 minutes things continued to pump away.

Mr. ANDERS. Mr. Anderson, let me make a general comment, if I may. The guidance boils down to the general comment that Dr. Knuth made. The general guide that we have to all operators and licensees of plants is that their foremost responsibility is to protect the public health and safety and they should act conservatively. Part of the problem, as we look back on this event, is that it is not just a question of the decisionmaking process at the operator level but also the information that he has presented to him and indeed the information——

Representative ANDERSON. He didn't get any information for what—20 minutes, 25 minutes?

Mr. ANDERS. That is a separable part of the problem, and we should focus on the operator's actions after that. This now comes down to some 20 to 25 minutes.

You must keep in mind that reactors are designed to be interlocked with an amazing blanket of automatic shutdown systems. When these systems sense that there is a potential, even a very minor potential, for some more consequential event, they will shut the reactor down automatically. So it does not necessarily follow that for any fire any place it would be advisable to shut down when the first piece of information was presented—though certainly in retrospect it does from this one.

We are, of course, going to look into our criteria with respect to that point.

Representative ANDERSON. Is somebody going to describe to us then what the triggers are in that control room for shutting down the control room in the event of a fire? Is that control room considered a critical area so that if you have a fire anywhere in the vicinity this activates certain triggers? I would like a better understanding, I think, of that.

Dr. KNUTH. Yes, there are general requirements that we are enumerating and that have been enumerated here. There are detailed procedures that a plant operator has to operate the machine within. These are written by the company and do indicate what the expected response of an operator should be for various types of emergencies, not only for fires. I don't want to minimize the effect but I think one of the criticisms that we had in our investigation was that the operator could have shut the plant down more promptly, but again that is 20-20 hindsight.

FIREFIGHTING EFFORTS

Representative ANDERSON. The other question I have deals with a statement that you made, Dr. Kruth, in your testimony that there was little or no coordination of effort to fight the fire. As I have briefly reviewed the chronology it does seem to me that that is almost an understatement on your part.

More importantly at this stage, are there now mandatory requirements by the Commission that in a plant of this kind we have what back in the days when I was going to grammar school we called fire drills that you put people through, procedures periodically, so that if something happens they don't take this disjointed, haphazard series of responses that seem to have occurred here at the Browns Ferry plant?

Dr. KNUTH. Yes, I will respond to that, Mr. Anderson. In the second bulletin—75-04A—one of the items discussed was the ability and the response of firefighting members when operations were being conducted which could be considered—I will not say hazardous but I will say could result in a fire—and it did require the operators of plants to develop adequate procedures to adequately monitor and provide adequate fire protection when they were engaged in activities of this nature.

REQUIREMENTS FOR EMERGENCY DRILLS

Representative ANDERSON. But I am still not quite sure that you answered my question. Is there a requirement that periodically they actually go through the motions of doing these things? Now, obviously you can't anticipate every type of emergency maybe but I would think there would be some value in mandating that people involved in the operation of one of these plants periodically actually go through the motions of carrying out their assigned function of what they are supposed to do in reporting a fire or in acting in the event of an emergency.

Dr. KNUTH. Yes. I can answer that this way. There are requirements that the licensee exercise his emergency procedures and there are requirements that he do it on a specific basis. One of the procedures is for fires, but I don't believe it is down to the specifics that he must simulate a fire and exercise that on a given time frame. There are general requirements that list emergencies to be simulated and we inspect to find out if he is doing that.

Representative ANDERSON. Will you give some thought to the suggestion that I think maybe having these people actually conduct drills from time to time would be a very useful device and impressing upon them the seriousness of this whole matter and also preparing them mentally and otherwise for the job of actually executing their assigned role should such an emergency develop?

Mr. ANDERS. Mr. Anderson, let me try to put it in perspective, the main question is are those items in our specific regulations?

Representative ANDERSON. That is what I am trying to get to, yes.

Mr. ANDERS. My understanding is they are not. My understanding further is that that very point is being reviewed in some depth at the moment and that additional measures have been taken to shore that up. The fact of the matter is, though, that all nuclear and nonnuclear utilities and TVA in this case have had an extensive fire prevention program including drills.

As to the adequacy of it and as to the aspect or the charge or comment here, if you will, about the coordination of the fire, I understand that TVA has a different view of that and I think you should hear from them on that point.

Representative ANDERSON. I have one other little question to follow on Senator Case's questions on this separation of cables.

Now, I have been informed that it is impossible perhaps to totally separate all of these cables and still operate from a single control room, but in that event what thought, if any, Dr. Knuth, has been given to the possibility of inerting the atmosphere, using inert gases in that atmosphere where these cables necessarily must be brought together which would render it impossible to have a conflagration of this kind?

Dr. KNUTH. I would like to have either Mr. Rusche or Dr. Hanauer comment on that since that is part of their ongoing studies in this regard.

Mr. RUSCHE. Mr. Anderson, I would like to comment that we have in the review of the efforts to restore the plant given serious consideration to methods that will improve the separation, but I agree with you that we know of no practical method for absolute separation in a common control room. The objective is to provide separation that is adequate to allow time for detection systems and corrective systems to function.

If you recall, in the testimony that you have heard from Chairman Anders and from Dr. Knuth there was and is a CO_2 system which has as its function that very purpose, to inert the atmosphere in the room. We are, as you will hear me testify later on, going to require the TVA to make that system automatically a functioning system. It is now a remote functioning system. That, coupled with other detection and correction systems, we believe will achieve the objective I have and that you are seeking.

Representative ANDERSON. One other question. This whole concept that you have of defense in depth is based very heavily, if I understand it correctly, on the use of computers. What happens, though, if you get a fire in the computer? How do you put that out? How do you control that?

Mr. ANDERS. I think that is a misconception, Mr. Anderson. The defense in depth goes to all aspects in design of the plant.

Representative ANDERSON. I will rephrase my question. But certainly that is an important tool in connection with the operation of the reactor, the computer.

Mr. ANDERS. Well, of course, as we move into the more sequenced movements involving different systems of the reactor, whether a computer is used or not, then you come into exposures to possible failure of interconnections like we saw here and then you must take an additional means such as firefighting procedures, to give you the kind of defense in depth you need from this kind of a system.

Representative ANDERSON. But in this whole study that you say is ongoing is any consideration being given to that problem of the possible fire in a computer and how you would control it?

Mr. RUSCHE. May I comment.

I would like to comment first that in TVA and in general the computer functions as an operational tool and is not a primary safety device. It can initiate annunciators and provide information to the operators for the purpose of evaluating system conditions. It is not the primary sensor for producing corrective action and that is clearly the case in TVA.

Now, in this case, it is the concept of defense in depth, not only redundancy as we have heard discussed by Senator Montoya and others within a particular system but it is the array of electrical, mechanical, and hydraulic devices which also contribute to the defense in depth. The systems that are required to initiate safety corrective actions are independent of the computer and it is this combination of both redundancy and multiplicity of types of systems that really comprise the defense in depth concept.

Representative ANDERSON. Thank you, Mr. Chairman.

POTENTIAL FOR CORE DAMAGE

Senator MONTOYA. Mr. Murphy, do you have a question?

Mr. MURPHY. Thank you, Mr. Chairman.

Dr. Knuth, in following up on what Congressman Anderson just mentioned, the inoperable equipment as I understand it was very briefly, the high pressure coolant injection system, reactor core isolation cooling, reactor feedwater pumps, control rod drive pumps and under the low pressure systems, the residual heat removal, core spray, condensate and condensate booster pumps, condensate pumps, standby coolant supplies, residual heat removal unit crosstie and several other components. There were also two pieces of equipment that were available on a nonstandard system operation and they were the reactor core isolation cooling and the standby liquid control.

Now, in light of this and in light of some taped recordings that we have here that the TVA people provided in an appendix where one of the individuals says, "We have some things out of extreme, we are trying to reestablish control," and at another place "This situation in Unit 1 right now is still not good, the only way we are putting water in is control rod drive."

In light of these things could you tell the committee how close you were to damaging the fuel or having a significant problem?

Mr. ANDERS. Mr. Murphy, I think that would be a more appropriate question for Mr. Rusche if that would be all right with you.

Dr. Knuth could review the facts.

Mr. RUSCHE. Mr. Murphy, Mr. Chairman, I would like to comment first. When one attempts to assess the closeness to core damage you should keep in mind the very important factor that the fission process was shut off by the insertion of the rods and the shutdown of the reactor. I think there is a tendency when one attempts to evaluate this closeness of approach to damage to put it in a short timeframe which is the timeframe in which most reactor safety studies are done.

I would like to clearly indicate that in the first place when you evaluate this we are in a different time context, not minutes but hours, and in fact tens of hours in some cases.

I believe that in addition to the one control rod drive pump that you mentioned was available, there were in addition two others that were available and could have, by relatively straightforward manual operation, been made available to provide water for the core.

Mr. MURPHY. I think I mentioned that some equipment was available using nonstandard systems operations.

Mr. RUSCHE. I would like to say nonstandard does not imply heroic, it is not something that is done momentarily or every day. Clearly, on a time sequence the water could have been added to the core well before any occasion for real need occurred. I think the assesment that we have made is that there was maintained a supply of water and there was maintained a removal of steam and/or water throughout the course of the event and that the core did not approach damage.

Mr. MURPHY. Specifically, how many systems were you down to, just to clear the record?

Mr. RUSCHE. I believe at one point, if you will accept the systems that you described, there were three control rod drive pumps. These, I guess, comprise about five water injection systems at high pressure.

Now, if the system were reduced to low pressure, there were additional systems that could have been brought to bear.

There were two elements in providing coolant to the core, of course. One is to get water into the core, the other is to remove steam or water from the core to discharge the heat. There was available through the relief valves and through the safety valves—operating at different times in sequence, as well as an additional steam line drain valve which was brought in to service early in the second sequence of pressurization. I plan to discuss that in some detail in my testimony.

Mr. MURPHY. The basis for my question is figure 4, Browns Ferry Nuclear Plant Unit No. 1 in the statement by Mr. Gilleland, Assistant Manager of Power, TVA.

It clearly shows on the chart that the systems under low pressure—as far as I can determine all of them were "inoperable due to high reactor vessel pressure." And it shows on the high pressure systems the high pressure coolant injection out, the reactor feed pumps out and the reactor core isolation cooling and standby liquid control available using nonstandard system operations.

As you pointed out, nobody suggested "heroic" and that is the reason for my question.

In other words, taking away all the additional words that you use, if you were to take the worst case, how many systems—one, three? What is it?

Mr. RUSCHE. There were three control rod drive pumps and the additional coolant system and the standby liquid control system that were available to inject water.

Mr. ANDERS. But in regard to the question of whether it was a near approach to danger, we must emphasize that the time factors in this kind of a situation are very long and the opportunity for actions such as to manually depressurize or crossvalve other pumps certainly was there.

Mr. MURPHY. Well, not to prolong this, but I believe it would appear that on a time sequence between 12 and 1 o'clock, which was the time of greatest pressure, at 1300 precisely it looks to me by reading the chart that you are down to three.

Mr. ANDERS. Yes.

Mr. MURPHY. Is your answer yes?

Mr. ANDERS. Certainly.

Mr. MURPHY. Thank you.

Mr. ANDERS. And the opportunity to depressurize and get into the other part of the chart.

Mr. MURPHY. I just didn't want any obfuscation.

Mr. RUSCHE. Mr. Chairman, to avoid any obfuscation may I point out three is not the correct answer. There are three control rod drive pumps which were each capable in addition to the other two systems.

Mr. MURPHY. So the answer is five?

Mr. RUSCHE. Five.

Mr. MURPHY. All right.

Senator MONTOYA. You may proceed with your statement, Doctor.

Dr. KNUTH. Yes. Returning now to my oral summary, as I indicated, the two bulletins were sent to the licensees requesting generalized information on policies and procedures relating to work control practices at operating plants, particularly as they relate to fire prevention.

The objective of course was to ensure that all operating plants were aware of the Browns Ferry event and that their attention was directed to areas of concern based on our initial evaluation of the fire.

Senator MONTOYA. Would you submit the exact communication you sent out for the record? [1]

Dr. KNUTH. Yes, sir, we will.

Most responses to the bulletins have described the overall procedures and review system for handling work which could affect safety. All responses from licensees are publicly available in the NRC's public document room.

Special inspections have been made by NRC inspectors at all of the other operating plants to determine what requirements exist for compartment boundary fire barriers and seals at electrical cable penetrations and the extent to which the facilities conform to these requirements. The results of these special inspections are available to the public in the NRC's public document room.

Based on the information received in response to the bulletins and on the nature and required rectification of the deficiencies discovered during the special inspections conducted as a result of the Browns Ferry fire, we concluded that there was sufficient assurance of continuing protection of public health and safety that no immediate NRC action to suspend or restrict operation at other nuclear power plants was warranted.

Mr. Chairman, this does complete my oral summary. We do have a number of slides, about 30, which show what happened on the day of the fire, and if the committee would desire we could go through them. They are all a matter of record as exhibits to my written testimony.

[The formal statement of Dr. Donald F. Knuth follows:]

STATEMENT OF DR. DONALD F. KNUTH, DIRECTOR, OFFICE OF INSPECTION AND ENFORCEMENT, U.S. NUCLEAR REGULATORY COMMISSION

This statement presents a brief historical background concerning the Browns Ferry Nuclear Plant; describes the chronology of events on the day of the fire; and summarizes the actions taken by the NRC's Office of Inspection and Enforcement as a result of the fire. The findings and conclusions of the Inspection and Enforcement staff involved in our investigation are set forth in the Summary of the publicly available Investigation Report. It should be borne in mind that our investigation highlights negative aspects of this event, based on the perspective of hindsight. The positive aspects of the licensee's reaction are not included in our report. Copies of our Investigation Report have been previously supplied to the Committee. This statement does not repeat our findings and conclusions.

The Browns Ferry Nuclear Plant, located on the north shore of Wheeler Lake, which is part of the Tennessee River, near Decatur, Alabama (Exhibit 1) is owned and operated by the Tennessee Valley Authority (TVA). The plant comprises three boiling water reactors designed by the General Electric Company. Each unit is rated 3293 megawatts, thermal, and can produce approximately 1100 megawatts of electrical power at design capacity.

Applications for construction permits for Browns Ferry Nuclear Plant (BFNP), Units 1 and 2, were filed with the AEC on July 1, 1966. Construction permits were issued by the AEC for these units on May 10, 1967, and construction was started on September 9, 1967. The application for the Unit 3 construction permit was filed on July 13, 1967, and the permit was issued on July 31, 1968.

Preoperational testing of Unit 1 began in early 1971. Operating License No. DPR-33 was issued by the AEC on June 26, 1973, and initial criticality, that is, the first self-sustaining chain reaction, was achieved on August 17, 1973. Low

[1] See appendix 3.

power and physics tests were completed and the unit attained commercial operation status on August 1, 1974. Approximately 6,900,000 megawatt hours of electric power had been generated by Unit 1 at the time of the fire on March 22, 1975.

Preoperational testing of Unit 2 began in early 1973. Operating License No. DPR-52 was issued on June 29, 1974, and initial criticality was achieved on July 20, 1974. Following low power and physics testing, the unit was placed in commercial operation on March 1, 1975. Approximately 2,500,000 megawatt hours of electric power had been generated by Unit 2 at the time of the fire.

The initial inspection of the BFNP by the AEC's Division of Compliance was made on July 17–18, 1967. As of March 20, 1975, 78 AEC or NRC inspections had been performed at Unit 1; 73 at Unit 2; and 58 at Unit 3. The reports of these inspections are publicly available; they document the inspection findings, items of noncompliance identified and the resolution of enforcement actions that were taken.

In March of this year, preparations were underway for removal of a temporary wall that had been erected to isolate construction associated with Unit 3 from operational activities involving Units 1 and 2. The licensee anticipated that possible increases in the leak rate from the secondary containment building as a result of removal of this temporary wall, combined with unrelated leaks through cable penetrations installed in the containment walls for Units 1 and 2, might result in total leakage exceeding acceptable limits. The possibility of such leaks through cable penetrations was associated with post-construction modifications made on Units 1 and 2 that involved changing cable runs or installing new cables, and an extensive program was underway for resealing penetrations through the wall between the cable spreading room, located beneath the control room for Units 1 and 2 and the secondary containment building for Units 1 and 2. Exhibit 2 shows the general location of the cable spreading room in relation to other portions of the facility.

The sealing operation involved stuffing pieces of plastic (polyurethane) foam and, in some cases, other materials, into each affected cable penetration, followed by installation of either liquid or spray-type polyurethane foam against the previously installed foam sheet dam. The liquid or spray-type foam expands to fill any remaining openings against air flow through the penetration. A final step involved application of a fire resistant material over the expanded foam.

The method used by TVA to check the effectiveness of the sealing operation was to hold a lighted candle near the penetration opening to see if air flow from the cable spreading room to the secondary containment building persisted. If the opening was not fully sealed, the lower pressure in the secondary containment building would cause a draft through the opening and thus cause the candle flame to be deflected toward the opening. The use of an open flame to test for air leakage in checking condenser vacuum has been a relatively common practice in the utility industry.

On March 22, while both Units 1 and 2 were operating at 100% of rated power, this leak checking technique was in use in connection with the sealing operation, and led to ignition of either the expanded foam or the polyurethane foam sheet installed in the first step of the operation. The flame of the candle used was pulled strongly into the opening and ignited a fire within the penetration (Exhibit 3: Fire and Fire Fighting[1]) at approximately 12:15 p.m. Prompt attempts to extinguish the fire locally were hampered by the difficult access to the initial flame location and the chimney effect of the air flow through the penetration. These attempts were unsuccessful. The fire was detected by plant personnel in the secondary containment building (Exhibit 4: Fire and Fire Fighting) only a few minutes after it was ignited. However, fire fighting efforts exerted in the secondary containment building (Exhibit 5: Fire and Fire Fighting) were frustrated by the inaccessibility of the fire—20 to 30 feet above the floor and accessible only by ladder—the dense smoke, and the limited availability (Exhibit 6: Fire and Fire Fighting) of breathing apparatus for use by personnel fighting the fire. Water was not used on the fire during these initial fire fighting efforts (Exhibit 7: Fire and Fire Fighting) for fear of personnel injury from electrical shock and to avoid possible adverse impact on operating systems.

A decision was made about 45 minutes after the fire started (Exhibit 8: Fire and Fire Fighting) to initiate operation of the fire extinguishing system installed

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[1] In subsequent references to exhibits, the reader will be directed to additional details relating to these aspects: Fire and Fire Fighting, Unit 1 Operations and Unit 2 Operations which occurred concurrently during the time period discussed.

in the cable spreading room. Some temporary difficulty was encountered in initiation of the installed carbon dioxide system (Exhibit 9: Fire and Fire Fighting) because metal plates that had been installed behind the breakout glass at the manual initiation stations to prevent inadvertent actuation during construction activities had not been removed, and because power to the electrical initiation stations had been disconnected as a personnel safety measure while men were working in the cable spreading room. Power was restored to the electrical system and the carbon dioxide system was placed in operation within about ten minutes, but it was not effective in extinguishing the fire which by this time had ignited the covering of cables in the penetration and had burned through the penetration into the secondary containment building. The temporary difficulty in initiating operation of the carbon dioxide system does not appear to have had significant impact on the outcome of fire fighting activities.

For a period of about 3½ hours, from shortly after 1:00 o'clock (Exhibit 10: Fire and Fire Fighting) until about 4:30, there was little or no directed, organized effort to fight the fire in the secondary containment building although some sporadic individual efforts may have been made, and the fire in the cable spreading room was fought separately. Shortly after 1:00 o'clock (Exhibit 11: Fire and Fire Fighting) the local fire department from Athens, Alabama (ten miles from the site) was called for assistance. The Athens fire trucks arrived at the plant about 1:30. The Athens Fire Chief recommended that water be used to fight the fire about half an hour after he arrived at the scene (Exhibit 12: Fire and Fire Fighting).

Fire fighting continued in the cable spreading room throughout the afternoon and by 4:30 the fire there had been put out (Exhibit 13: Fire and Fire Fighting). Permission to use water to fight the fire in the secondary containment building was given (Exhibit 14: Fire and Fire Fighting) between 5:30 and 6:00 p.m. About 7:00 o'clock water was used on the fire by two men who wedged the nozzle of the fire hose in a position to direct water on the fire after which they left the scene. Fifteen minutes later, two men returned to the scene and found no further evidence of burning. The fire was declared to be extinguished at 7:45 p.m. The delay in the decision to use water in fighting the fire contributed significantly to the extent of the damage incurred.

The fire damage extended from the point of ignition in the cable penetration in one direction a few feet into the cable spreading room and in the opposite direction along the cables about 40 feet into the secondary containment building. All ten cable trays in the penetration and over 1600 cables were affected, including cables in 26 trays in the secondary containment building.

Exhibit 15 depicts, in simplified form, some important BWR systems. Throughout this and other statements reference will be made to ECCS, the emergency core cooling system. This frequently used term involves a number of independent systems: the Low Pressure Coolant Injection (LPCI) system (LPCI is actually one mode of operation of the Residual Heat Removal (RHR) system); the Core Spray system; the High Pressure Coolant Injection (HPCI) system; and the Automatic Depressurization system, which operates to open the relief valves automatically under certain conditions and releases steam to the pressure suppression pool.

Other systems, not normally considered as ECCS are also available to provide sources of cooling water to the reactor core. Later in this statement, reference will be made to the control rod drive system pumps, which provide high pressure water to operate the hydraulic control rod drives during normal operation. In the process of doing so, they also pump water into the reactor vessel.

Let us now "rezero" the clock and consider the effects of the fire on operation of Unit 1.

In the common control room for Units 1 and 2, the control room operators were notified of the fire about twenty minutes after the fire started. About five minutes later (Exhibits 6 and 7: Unit 1 Operations), the Unit 1 operator noted anomolous behavior of controls and instrumentation for systems designed to provide emergency cooling of the reactor core. About ten minutes later (Exhibit 8: Unit 1 Operations), the main recirculating pumps on Unit 1 shut down and the operator manually shut down the Unit 1 reactor and verified that all control rods had been fully inserted. This stopped the chain reaction and eliminated nuclear fission as a direct source of heat although heat generation in the core continued as a result of the radioactive decay of fission products in the reactor fuel. The principal concern for the safety of the plant following shutdown of the reactor was to provide continuous cooling of the fuel to remove this decay heat.

As the water level in the reactor vessel dropped as a result of the reduced boiling in the core, the feedwater pumps, designed to provide high pressure water to the reactor vessel, increased their flow rate, as designed, and raised the water level to the point that the operator elected to shut down two of them as well as the HPIC and RCIC steam-driven pumps, which had started on the low water level signal, to prevent overfilling the vessel.

Shortly after 1:00 o'clock (Exhibit 9: Unit 1 Operations), the main steam isolation valves closed. This action shut off the steam to the main condenser thus eliminating this means of rejecting decay heat. The closing of the main steam isolation valves also caused the loss of the remaining steam-driven feedwater pump providing high-pressure feedwater to the core. The normal methods of providing high pressure cooling water to the core were unavailable due to loss of electrical equipment and instrumentation caused by the fire. TVA used one of two electrically-driven pumps readily available in the control rod drive system (Exhibit 10: Unit 1 Operations) to provide somewhat over 100 gallons of water per minute to the vessel.

As the pressure in the Unit 1 reactor coolant system increased as a result of steam produced by the decay heat in the core, the relief valves opened automatically to release steam to the pressure suppression pool thus preventing overpressurization of the reactor coolant system.

When it became apparent that the Unit 1 reactor coolant level could not be maintained at high pressure using only the CRD pump (Exhibit 11: Unit 1 Operations), the operator elected to reduce the pressure in the reactor coolant system by opening the relief valves to release additional steam from the reactor vessel to the pressure suppression pool. When system pressure was reduced sufficiently, low pressure pumps were operated to maintain water level in the reactor vessel (Exhibit 12: Unit 1 Operations). This mode of core cooling was successful until about 6:00 p.m., when control was lost for the four relief valves that had been operable up to that point (Exhibit 14: Unit 1 Operations). This caused these relief valves to close, permitting repressurization of the reactor coolant system. Because of this repressurization, the low pressure pumps could not continue to supply water and the operator chose to rely on one control rod drive system pump as the source of coolant injection for a period of about four hours (Exhibit 16: Unit 1 Operations) until temporary modifications reestablished the capability to open the four affected relief valves and reduce system pressure. An additional spare control rod drive system pump was also available for use. Subsequent analysis has shown that other methods were also available.

Attempts were continued to establish more conventional methods of providing cooling of the reactor core and the water in the pressure suppression pool. About 13 hours after the fire started, temporary repairs were made which permitted cooling of the pressure suppression pool and normal means of cooling the Unit 1 reactor core were restored by about 4:10 a.m. on March 23.

Let us again turn back the clock to consider Unit 2.

The effect of the fire on Unit 2 operation was less pronounced. The Unit 2 control operator shut down the reactor manually forty-five minutes after the fire started, when numerous alarms occurred, most of which were associated with loss of DC power (Exhibit 9: Unit 2 Operations). He verified that all control rods had been inserted fully and shut down the feedwater pumps when the water level in the reactor increased to the designated level for such action. About three minutes later, the Unit 2 main steam isolation valves closed automatically. The Unit 2 control operator successfully initiated operation of systems designed to supply high pressure water to the core (Exhibit 10: Unit 2 Operations). For a period of about 45 minutes, the loss of some electrical supply panels caused the loss of manual control of the Unit 2 relief valves as well as the loss of certain other motor-operated valves (Exhibit 11: Unit 2 Operations). The relief valves continued to operate to prevent overpressurizing the Unit 2 reactor coolant system during this period. About 2:10 the reactor coolant system pressure began to drop, probably as a result of a relief valve failing to close after opening automatically (Exhibit 12: Unit 2 Operations). Approximately five minutes later, control of the relief valves was restored and controlled depressurization of the reactor coolant system was continued. After depressurization, low pressure pumps were used to provide cooling water to the Unit 2 reactor core (Exhibit 13: Unit 2 Operations). By about 6:30, conditions on Unit 2 were stabilized (Exhibit 14: Unit 2 Operations). Normal means for cooling the core were reestablished about 10:45 p.m.

The Browns Ferry Emergency Plan was activated at about 1:00 o'clock when it became clear that operation of the Units 1 and 2 reactors had been affected by the fire. At that time, it was not apparent that no offsite consequences were to result from the fire, as we now know was the case. Communication among the various groups and agencies involved in coping with this event could have been better than it was. Some deficiencies in performance of emergency plan functions were identified in our investigation, but inasmuch as no offsite release occurred, these did not contribute directly to the effects of this event.

The Atlanta Regional Office of the NRC's Office of Inspection and Enforcement was notified of the fire shortly after 4:00 p.m. on March 22. Inspectors from the Regional Office were dispatched promptly to the Browns Ferrey site. The Regional Office immediately notified NRC Headquarters, and key members of the NRC's Offices of Inspection and Enforcement, Nuclear Reactor Regulation and Public Affairs were assembled. Other cognizant federal agencies were alerted and contact was maintained with the Regional Office and NRC inspectors at the Browns Ferry site. An investigation was initiated immediately by the Atlanta Office. The field investigation required 280 man-days of effort. It was completed and a detailed report thereon, dated July 25, 1975, was prepared and released publicly on July 30. Copies of the Investigation Report have been previously furnished to the Committee.

As a result of our investigation, we took enforcement action in the form of a Notice of Violation to the licensee, TVA, on July 29, 1975. This Notice of Violation is publicly available. It identifies those items we believe to require corrective action. In accordance with our standard practice, TVA was given twenty days to respond to the Notice of Violation. TVA requested, and was granted, an extension of that period to September 2. We have received TVA's response and we are evaluating it. TVA takes exception to some of our findings and points out some parts of our report which they believe contain factual errors. If our evaluation confirms that there are errors in our report, the record will be corrected.

It is perhaps appropriate to note parenthetically that the purpose of our enforcement action known as Notices of Violation, such as the one issued to TVA, is not punitive but rather corrective. Indeed, our entire enforcement program has as its aim the correction of deficiencies rather than punishment of offenders, although punitive sanctions are available as a means of reinforcing our concerns, when such action is necessary. We do not believe that the use of the more severe sanctions is warranted in this case.

Shortly after the Browns Ferry fire, the Office of Inspection and Enforcement prepared and distributed to all licensees a Bulletin (75-04) and supplement (75-04A) requesting generalized information on policies and procedures relating to work control practices at operating plants, particularly as they relate to fire prevention. Specific information on certain safety questions raised by the Browns Ferry experience was also requested. The objective here was to ensure that all operating plants were aware of the Browns Ferry event and that their attention was directed to areas of concern based on our initial evaluation of the fire.

Most of the responses to the bulletins that have been received have described the overall procedure and review system for handling work which could affect safety. All responses from licensees are publicly available in the NRC's Public Document Room. Typically, such work requires written procedures spelling out hazards, precautions or prohibitions, and requiring various levels of review and approval.

Construction and modification work in operating plants is generally handled quite formally, requiring engineering and safety committee review and approval and Plant Superintendent approval. Higher management approval may or may not be required. Maintenance activities are generally handled somewhat less formally, using job orders or trouble tickets. Usually, general maintenance procedures which have previously been reviewed and approved at the various levels are followed. Such activities also require approval of the responsible foreman, the shift operating supervisor, and sometimes the department supervisor, Chief Engineer or Plant Manager.

Special inspections have been made by NRC inspectors at all of the other 51 operating plants to determine what requirements exist for compartment boundary fire barriers and seals at electrical cable penetrations and the extent to which the facilities conform to these requirements. The results of each of the special inspections are available to the public in the NRC's Public Document Room.

Forty-three of the plants had construction specifications establishing detailed requirements for fire barriers, including thirty-nine plants which are committed by their Safety Analysis Reports to install fire barriers. These requirements varied in complexity from multilayer barriers with non-flammable insulation, sheathing and fire retardant to simple fiberglass packing around the cables. For the remaining eight plants no documented fire barrier requirements were available. Three of these were found to be well protected by fire barriers.

Deviations from fire barrier requirements were found in varying degrees at thirty-eight of the plants. These included missing barriers, improper construction, improper materials, and barriers that had been opened for one reason or another or had deteriorated to some degree, but which had not been repaired. Most of the deviations were from construction specifications, although twelve of the plants also had deviations from commitments made in their Safety Analysis Reports. At twenty-three of the plants the deviations were few in number (10 or less out of up to 2000 locations) or of a minor nature. The remaining sixteen had numerous deviations with some plants having 20% or more of the required barriers deficient in some manner. Where appropriate, action to enforce requirements has been taken based on findings of the special inspections and completion of corrective action is being verified by our inspectors.

Based on the information received in response to the Bulletins and on the nature of the deficiencies discovered during the special inspections conducted as a result of the Browns Ferry fire, we concluded that with the additional procedural controls implemented by those licensees at whose facilities deficiencies were identified, there was sufficient assurance of continuing protection of public health and safety that no immediate NRC action to suspend or restrict operation at other nuclear power plants was warranted. The extent to which any new or changed NRC criteria or practices that may arise from this event will be applied to presently licensed plants will be determined when such criteria are developed.

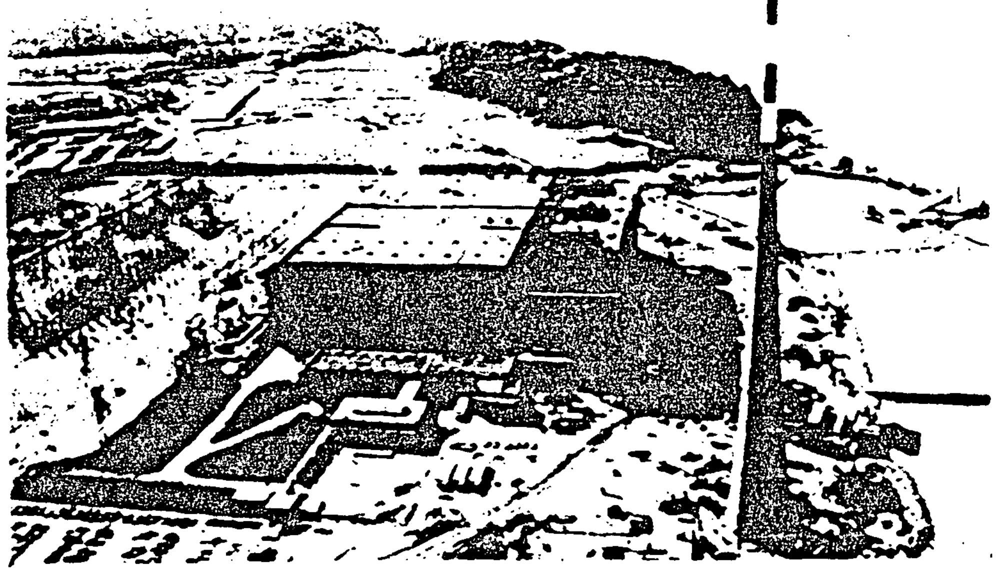


EXHIBIT 1.—GENERAL VIEW LOOKING EAST

BROWNS FERRY NUCLEAR GENERATING PLANT

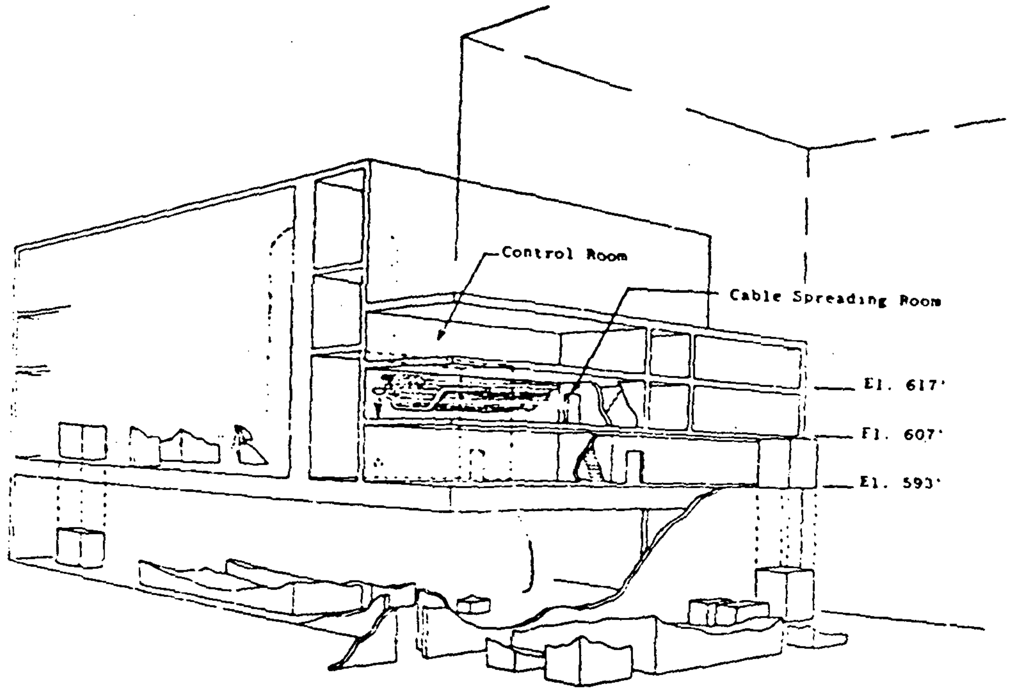


EXHIBIT 2

CHRONOLOGY – BROWNS FERRY FIRE, MARCH 22, 1975

Activity	12 15 p.m.
Fire and Fire Fighting	Polyurethane sealant in cable spreading room penetration was ignited by candle flame at approximately 12:15. Fire was initially fought by construction workers using portable CO_2 and dry chemical extinguishers.
Unit 1 Operations	Unit operation normal at 1098 MWe
Unit 2 Operations	Unit operation normal at 1098 MWe

EXHIBIT 3

CHRONOLOGY — BROWNS FERRY FIRE, MARCH 22, 1975

Activity	12:30 p.m.
Fire and Fire Fighting	Public Safety Officer reported the fire to an Assistant Shift Engineer (ASE) at 12:30-12:35. ASE initiated fire alarm and advised Shift Engineer at 12:34-12:35. Construction workers continued to fight fire which had spread to reactor building.
Unit 1 Operations	Unit operation continued normal.
Unit 2 Operations	Unit operation continued normal.

EXHIBIT 4

CHRONOLOGY — BROWNS FERRY FIRE, MARCH 22, 1975

Activity	12:35 p.m.
Fire and Fire Fighting	Unit operator began announcing fire over PA system at about 12:35. The ASE arrived in the reactor building and began fighting fire at about 12:40.
Unit 1 Operations	Unit operation continued normal.
Unit 2 Operations	Unit operation continued normal.

EXHIBIT 5

CHRONOLOGY — BROWNS FERRY FIRE, MARCH 22, 1975

Activity	12:40 p.m.
Fire and Fire Fighting	Between 12 40 and 12 43 firefighters called control room for breathing apparatus. Unit 1 operator requested assistance. Shift Engineer and an Assistant Shift Engineer left reactor building fire area to go to control room.
Unit 1 Operations	At about 12 40 the Unit 1 operator observed alarms associated with the Emergency Core Cooling Systems that were contrary to the status of the systems. By about 12 42 to 12 43 Residual Heat Removal (RHR), Core Spray (CS), High Pressure Coolant Injection (HPCI), Low Pressure Coolant Injection (LPCI), and Reactor Core Isolation Cooling (RCIC) pumps were all running. The unit operator stopped the pumps. The alarms, however, would not reset.
Unit 2 Operations	Unit operation continued normal.

EXHIBIT 6

CHRONOLOGY — BROWNS FERRY FIRE, MARCH 22, 1975

Activity	12:45 p.m.
Fire and Fire Fighting	Fire fighting efforts continued in spreading room and reactor building.
Unit 1 Operations	Between about 12 45 12 48 the RHR, CS, and HPCI initiated again. Random lights on the ECCS panel began glowing abnormally bright and dim. At about 12 48 the recirculating pump flow began decreasing causing a reactor power level decrease. The operator attempted to stop the RHR and CS pumps but they could not be stopped from the control bench board. Indicated water level in reactor was normal.
Unit 2 Operations	Unit operation continued normal.

EXHIBIT 7

CHRONOLOGY — BROWNS FERRY FIRE, MARCH 22, 1975

Activity	12:50 p.m.
Fire and Fire Fighting	The first steps of the Browns Ferry Emergency Plan were implemented at 12:56 when the plant superintendent was called. At about 1:00 the ASE initiated the cable spreading room evacuation alarm from the control room. He then proceeded to the cable spreading room to actuate the CO_2 system.
Unit 1 Operations	Unit operator began zeroing the recirculating pump controllers in preparation for tripping the reactor. The recirculating pumps tripped and the Unit operator manually tripped the reactor at 12:51 from a power level of 704 MWe. The ASE tripped the turbine generator at about 12:53. The unit operator tripped two of three feedwater pumps. He also verified that all control rods had inserted. Subsequent to the reactor trip, the indicated water level decreased to plus 5 inches and then increased to plus 47 inches and the unit operator tripped the RCIC and HPCI to control the water level. Diesel generators "C" and "D" had started and were feeding the shutdown boards. Diesel generators "A" and "B" had started but were not connected to the shutdown boards.
Unit 2 Operations	Unit operation continued normal. Two spurious alarms were received on the Turbine Control Panel at 12:50.

EXHIBIT 8

CHRONOLOGY — BROWNS FERRY FIRE, MARCH 22, 1975

Activity	1:00 p.m.
Fire and Fire Fighting	The ASE attempted to actuate the CO_2 system from the station at the Unit 1 entrance to the cable spreading room. The power had been shut off at the Disable Switch at the Unit 2 entrance. He went to that station, restored power, and when automatic initiation did not appear successful he attempted to manually actuate the system. He found that a metal plate had been installed under the break out glass and the system could not be actuated. The automatic initiation did function and after about 3 minutes CO_2 began discharging.
Unit 1 Operations	By about 1:00 power had been lost to the three 480V motor operated valve (MOV) boards, two of three 250V MOV boards, the two 480V Shutdown Boards, Preferred Power Board, and Shutdown Bus No. 1. At about 1:03 the main steam isolation valves (MSIV) closed isolating the reactor from the condenser heat sink and cutting off the steam supply to the feedwater pump turbines. Possible cause of valve closure was loss of power to solenoids or because of low reactor water level. With the closure of the MSIV's reactor pressure rose rapidly to 1100 psig and the relief valves began operating and maintained pressure between 1080 psig and 1100 psig.
Unit 2 Operations	At about 1:00 operator observed numerous alarms associated with DC power failure and reactor protective system. He pushed the control button to trip the reactor but it is not certain whether his action or automatic action resulted in the trip. He confirmed that all rods had fully inserted. The turbine was tripped at about 1:01. Reactor water level first dropped to about 0 inches indicated, then rose to about 40 inches at about 1:03. These fluctuations caused HPCI and RCIC to start and stop. When the water level reached 40 inches the operator tripped all three feedwater pumps. At 1:03 the MSIV's closed. This caused the reactor coolant system pressure to increase.

EXHIBIT 9

CHRONOLOGY — BROWNS FERRY FIRE, MARCH 22, 1975

Activity	1:00 p.m.
Fire and Fire Fighting	Between 1:00 and 1:10 the ASE that had been directing the firefighting activities in the reactor building ordered the fire fighters out because of the lack of breathing apparatus, dense smoke and the failure of the lighting system. Although there may have been sporadic individual efforts, there was no centrally organized firefighting in the reactor building until about 4:30. Firefighting efforts did continue in the cable spreading room during this period.
Unit 1 Operations	Prior to 1:10 the reactor coolant system pressure relief valves began opening and closing automatically, maintaining system pressure between 1080 psig and 1100 psig. The reactor coolant system was being supplied water by the control rod drive (CRD) water pump. At this time the standby liquid control system (SLCS), the nuclear instrumentation and about one half the CRD position indicators were inoperable. The CRD pump capacity of about 100 gpm was insufficient to maintain reactor water level. Level began decreasing.
Unit 2 Operations	At 1:10 the RCIC system was started to supply water to the reactor. HPCI was started to relieve steam from the reactor. The control rod drive pump was also supplying water to the reactor. At this time the relief valves were opening automatically to maintain reactor pressure at or below 1100 psig.

EXHIBIT 10

CHRONOLOGY — BROWNS FERRY FIRE, MARCH 22, 1975

Activity	1:10 p.m.
Fire and Fire Fighting	The Athens, Alabama, Fire Department was called at 1:09. The firemen arrived at the site at about 1:30. Fire fighting efforts, using portable extinguishers, continued in the cable spreading room.
Unit 1 Operations	At about 1:30 the Unit 1 operator decided that the reactor water level could not be maintained with the CRD pump and that reactor pressure must be reduced to 350 psig, by manually controlling the operation of the relief valves in order to use the condensate booster pump.
Unit 2 Operations	At about 1:20 manual actuation capability of all relief valves was lost, but automatic operation at 1179 psig continued. Diesel generator D tripped. At this time power was lost to all 480V Shutdown and MOV Boards for 45 minutes.

EXHIBIT 11

CHRONOLOGY — BROWNS FERRY FIRE, MARCH 22, 1975

Activity	1:30 p.m.
Fire and Fire Fighting	The Athens firemen were [illegible] at [illegible] and by 1:45 were [illegible] to assist in fire fighting. The Athens Fire Chief examined the fire area and at about 2:00 initially recommended the use of water. Fire fighting efforts in the spreading room continued.
Unit 1 Operations	Depressurization of reactor initiated at about 1:40. Depressurization to 350 psig took approximately 20 minutes. The indicated minimum water level [illegible] was about 120 inches, about 48 inches above the active fuel. After initial fluctuations, water level was controlled at an indicated +33 inches using the condensate booster pumps. Relief valves were kept open to maintain pressure below 350 psig. Feedwater valve breakers were opened to preclude inadvertent closure of valves.
Unit 2 Operations	One relief valve stuck open at about 2:10, and reactor began depressurizing. Manual control of valves was restored at 2:15. Decision was made to continue to depressurize in the event that the Unit 2 systems degraded further. At about 2:30 torus cooling was initiated using RHR pump D.

EXHIBIT 12

CHRONOLOGY — BROWNS FERRY FIRE, MARCH 22, 1975

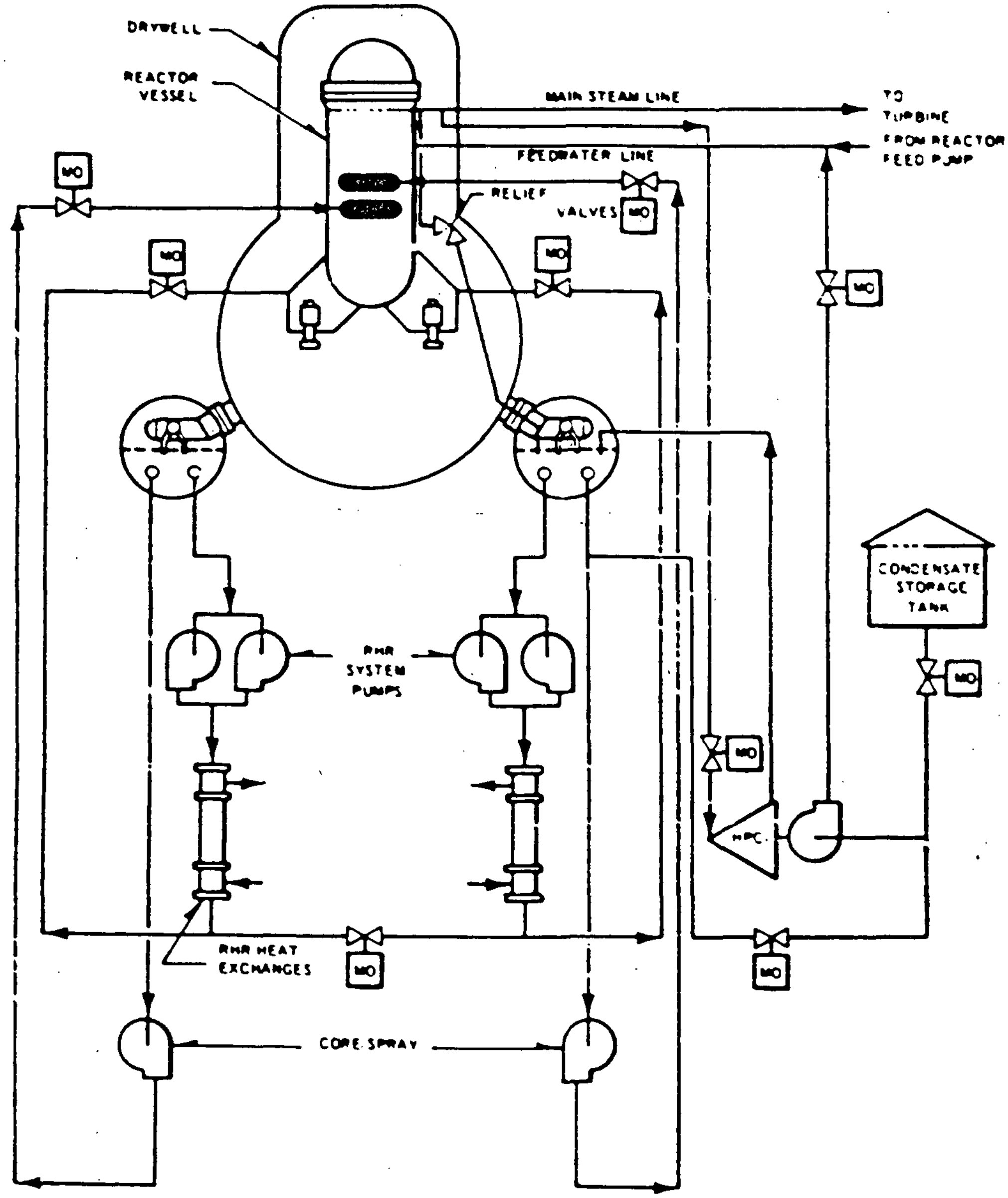
Activity	2:30 p.m.
Fire and Fire Fighting	Fire fighting efforts in the spreading room were being assisted by the Athens Fire Department. The CO_2 system had been initiated on three separate occasions. By 4:00 the fire in the spreading room had been contained and by 4:30 had been extinguished. Organized fire fighting efforts in the reactor building were reinitiated by about 4:30 with the stringing of temporary lights and hand lines.
Unit 1 Operations	By 3:15 all torus level and temperature indication was inoperative. Inoperable equipment at this time included all ECCS, the main steam isolation valves, manual control of seven relief valves, standby gas treatment system train B, diesel generators C and D, and [illegible] blowers. Equipment that had been restored to service by 4:30 or shortly there after included three RHR valves, MOV Board 1A, and Reactor Protection System MG Set 1A.
Unit 2 Operations	At about 3:00 torus level control was established using the RHR drain pump. At this time reactor pressure was 200 psig. Prior to 4:10 use of the condensate booster pump to maintain reactor water level was initiated.

EXHIBIT 13

CHRONOLOGY — BROWNS FERRY FIRE, MARCH 22, 1975

Activity	4:30 p.m.
Fire and Fire Fighting	When fire fighting efforts were reinitiated at about [illegible] dry chemicals, the fire was going strong in two places and smoldering in several others. Public Safety Officers [illegible] Athens firemen assisted in the efforts but did not use the extinguishers. The use of breathing apparatus was [illegible]ssary. The plant superintendent requested and received permission from his supervisor to use water on the fire at about 5:30. He [illegible] discussed the use of water with the Athens Fire Chief at about 6:00, and the Fire Chief again recommended using water. The plant superintendent gave permission for the use of water at about 6:40. Some time was required for preparation and water was used beginning between 7:00 and 7:20. The fire was declared extinguished at 7:45.
Unit 1 Operations	At 6:30 neither torus nor normal reactor shutdown cooling had been established. At about 6:00 manual control of the four relief valves had been lost, and there was not at that time [illegible] method for controlling pressure below 350 psig. Pressure increased as follows: 6:40 - pressure - 300 psig; [illegible]:30 - pressure - 460 psig 6:55 - pressure - 400 psig; [illegible]:20 - pressure - 540 psig 7:00 - pressure - 420 psig; [illegible] - pressure - 600 psig
Unit 2 Operations	At about 6:30 the reactor conditions were stabilized and no further problems were experienced with cool down.

EXHIBIT 14



Emergency Core Cooling System

Exhibit 15

CHRONOLOGY — BROWNS FERRY FIRE, MARCH 22, 1975

Activity	4:30 p.m.
Fire and Fire Fighting	
Unit 1 Operations	After pressure increased above 350 psig the condensate booster pumps no longer had the capability to supply water to the reactor, and the control rod drive pump was used to maintain reactor water level. The plant operators determined that a solenoid valve had failed closed due to the fire, and the closure of this valve had cut off the control air supply to the valve supplying air to the relief valves causing closure of this valve. Control air was restored to the supply valve, and by 9:50 control of the relief valves was restored. The reactor was depressurized, and no further problems were experienced in the cooldown of the reactor.
Unit 2 Operations	

EXHIBIT 16

Senator MONTOYA. How long will this take?

Mr. ANDERS. Senator, if I may, we also have Mr. Rusche and Dr. Hanauer who have summaries of their testimony and it depends on your time.

Senator MONTOYA. Well, we are going to go on this afternoon, but I would like to ask Dr. Knuth a few questions.

Mr. ANDERS. There are about 20 minutes of slides.

Senator MONTOYA. I would like to ask you a few questions on your summary and your statement, Dr. Knuth.

Dr. KNUTH. Yes, sir.

Senator MONTOYA. Now, it is very apparent that there was no definite instruction as to what to do in this kind of an emergency either from the underwriters or from NRC or this would not have occurred or at least the extent of the damage would not have taken place.

Now, there were apparently disagreements on how to proceed, whether or not to use the water. There were further complications by virtue of poor training and equipment, very bad response, and by the difficulty in gaining access to the affected area.

However, there was no nuclear component to the accident and fire-fighting was conducted without the presence of a radiation environment.

Now, can you speculate how accident recovery activities might have been affected if a nuclear event had taken place?

Now, I will ask you another question in this same context.

Are utilities prepared to take necessary countermeasures when operating personnel themselves might be subject to radiation exposures?

Dr. KNUTH. It is a very difficult question to answer. I will have to start off by putting it in the context that the emergency response of a utility in response to any emergency, or accident situation is part of a procedure which is maintained by the utility. The utility is required as part of its license to develop an emergency plan in conformance with Appendix E of part 50 of our regulations.

USE OF WATER IN FIREFIGHTING

Senator MONTOYA. Was there an emergency plan in this particular case and did it call for the use of water?

Dr. KNUTH. There was an emergency plan and I don't believe the manual for Browns Ferry they were operating under required the use of water but I do not——

Senator MONTOYA. Did it prohibit it?

Dr. KNUTH. No, it did not prohibit the use of water.

Mr. ANDERS. It did allow it—it did not prohibit it, Senator, even with the fact that the NRC is reviewing the fire prevention area in more and more depth. I doubt that we will find it advisable to get down to the specificity of requiring or not requiring water for this kind of particular event. We should allow some element of judgment by people on the scene.

Senator MONTOYA. Mr. Anders, what I am trying to develop is was there any specific authorization or instruction for the use of water in this type of a situation?

Mr. ANDERS. It is my understanding that the TVA procedure——

Senator MONTOYA. In the manual or instructions.

Dr. KNUTH. Yes, sir. It was authorized, it was not precluded.

Dr. MASON. I think one point that should be brought out is in some firefighting installations there are fixed nozzles. In the absence of fixed nozzles in TVA's plant they were reluctant to instruct that they hand spray the fire, because the operators could be injured.

Now, that is a judgment call and it was approved or permitted by the manual. The plant superintendent and I think TVA, as the Chairman has already suggested here, you should ask them about this. In his judgment he did not wish to take what he considered to be these risks. Later on after the opportunity had been provided to verify that the cables had been de-electrified, water, using hand-held nozzles, was permitted.

Now, our judgment is that perhaps it could have been used sooner but it was not that there was a lack of interest, let's say, in using water; there was a considered judgment on the part of TVA. I think they could better provide their reasons for delaying water.

Senator MONTOYA. Will you be a little more specific and state for the record under what circumstances would the instructions from TVA be triggered to use water in that situation? Is it the presence of a nozzle close by or what?

Dr. MASON. No. Some of the factors are, as I understand them, the evaluation of the cables—are they apt to be flooded or is the water apt to drain off. Are the cables electrified or not. What is the class of fire.

It was deemed that the combustion of this insulation would permit the use of water. There are other classes of fire in which the regulations, as I understand them, would not have permitted water.

Senator MONTOYA. Why was there hesitancy?

Mr. ANDERS. There are two reasons that TVA has brought forth. One, the danger to personnel fighting the fire and two, the concern that the use of water at the early stage of the event might compound the problem of controlling the reactor if water might short out or increase the interconnection of cables.

Senator MONTOYA. Well, in other words, were they right in waiting?

Mr. ANDERS. They were right in the sense that there was no radioactive release and there was no core damage. Whether the fire could have been put out sooner with the same results in a shorter period of time can never be fully determined but our investigators expressed such a view in their report.

Senator MONTOYA. Which is what, that the fire could have been put out earlier?

Mr. ANDERS. The report suggests that the fire could have been put out earlier without adverse consequences but personally I am not sure that we have looked into all the facets involved to be able to make that statement as conclusively as it was made. That aspect is also part of the ongoing review.

Senator MONTOYA. Hasn't the statement been made by some in your agency and by others that the fire could have been put out earlier and it would have caused less damage than actually occurred?

Mr. ANDERS. That position tends to be reflected in the report, Mr. Chairman. I personally would not make the statement quite as specific as that sounds—but that's a matter of professional judgment.

Senator MONTOYA. What statement would you make then?

Mr. ANDERS. I would say the procedures taken by TVA, which includes areas where we have recommended improvements, in fact did not result in a core damage or radiation leaks. I believe that sooner action and more coordinated firefighting procedures and some increased availability of equipment probably could have reduced the time the fire burned.

Senator MONTOYA. Now, in what depth does NRC review and approve a reactor licensee's own onsite emergency procedures?

Mr. RUSCHE. Mr. Chairman, in the course of our review of applicants we determined that onsite and offsite emergency procedures are developed according to a published manual that NRC has provided. Our requirements are that their procedures comply with the contents of this manual, and we had done that for TVA and that continues to be the case.

Senator MONTOYA. But you didn't have anything for this situation in your manual, did you?

Mr. RUSCHE. The manual that I am referring to?

Senator MONTOYA. Requiring the use of water.

Mr. RUSCHE. That is correct.

Senator MONTOYA. You didn't?

Mr. RUSCHE. We didn't.

Senator MONTOYA. Do you have it now?

Mr. RUSCHE. The manual has to do with the concept and scope of the plan, not the details of the procedure.

Senator MONTOYA. Would you give us the content of that for the record.

Mr. RUSCHE. We will do that.

[The following material was later submitted:]

NRC requirements are set forth in the Code of Federal Regulations, Chapter 1, Title 10, Part 50, Appendix E. This Appendix establishes the information regarding plans for coping with emergencies to be contained in the Preliminary Safety

Analysis Report and the Final Safety Analysis Report in license applications. It also establishes the minimum requirements for emergency plans.

Appendix E was published in 1970. The staff evaluated the Browns Ferry emergency planning program against these requirements and found it acceptable. This finding was presented in the Safety Evaluation Report of June 26, 1972.

More detailed guidance to applicants is provided in the NRC Standard Review Plan, Section 13.3, "Emergency Planning". A copy of this document is attached.

[Attachment follows:]

U.S. ATOMIC ENERGY COMMISSION,
REGULATORY STANDARD REVIEW PLAN,
DIRECTORATE OF LICENSING.

SECTION 13.3—EMERGENCY PLANNING

REVIEW RESPONSIBILITIES

Primary—Industrial Security and Emergency Planning Branch (ISEPB)
Secondary—None

I. Areas of review

The applicant's emergency planning, as described in his safety analysis report (SAR), is reviewed by ISEPB. This review of this section of the SAR involves evaluation of evidence of preliminary planning (in the preliminary safety analysis report, PSAR) or substantive evidence of planning (in the final safety analysis report, FSAR) for emergency preparedness directed primarily at situations involving real or potential radiological hazards.

At the PSAR stage the review covers each of the seven sub-parts A–G of 10 CFR Part 50, Appendix E, Part II. Particular attention is given to the following areas, applicable to the sub-parts indicated.

With respect to sub-part B, the designation by the Governor of the state in which the facility is to be located of an agency that has the primary responsibility for planning for radiological emergency response in the (public) environs of the plant is verified and evidence of the arrangements that have been made by the applicant with this agency for the preparation of coordinated emergency response plans in the environs of the facility is reviewed.

With respect to sub-part C, one of the protective measures considered is the evacuation of persons from the exclusion area and from potentially affected sectors of the environs. An analysis of the implications for evacuation of the most severe design basis accident postulated is reviewed to assure that it includes explicit findings or information necessary for emergency planning.

With respect to sub-part E, the review includes a determination that at least two off-site hospital facilities are identified, with evidence that preliminary contacts have established agreements and potential capability to receive and treat individuals affected by radiological emergencies.

At the FSAR stage, a comprehensive emergency plan document is reviewed. The emergency plan should demonstrate implementation of the objectives and requirements of 10 CFR Part 50, Appendix E, Parts I, III, and IV.

II. Acceptance criteria

At the PSAR stage, this section is considered acceptable (1) if it conforms to the requirements of 10 CFR Part 50, Appendix E, Part II, (2) if the emergency planning information, submitted in accordance with section 13.3 of Revision 2 of the "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," is consistent with facility design features, analyses of postulated accidents, and characteristics of the proposed site location, and (3) if it provides reasonable assurance that appropriate protective measures can be taken in the event of a serious accident within and beyond the site boundary.

ISEPB considers that the last of the above is satisfied if preliminary planning and analysis shows that there is reason to expect that the emergency plans for the facility can be designed to meet, at minimum, the following objectives, based upon calculated radiological dose consequences of an airborne release following the most serious design basis accident:

1. Completion of evacuation of persons within the exclusion area within two hours from the onset of release. In this connection, ISEPB considers that the required assurance cannot be given if non-plant related activities, e.g. recreational

activities, are permitted anywhere within the exclusion area where siting dose guidelines of 10 CFR Part 100 might be reached in less than two hours, as shown by calculation.

2. Completion of evacuation of persons within 45 sectors of the environs beyond the exclusion radius boundary within two hours from the onset of release, or within the times calculated as a function of distance for a potential dose to reach the upper limit of the range of protective action guide levels to be adopted as warranting evacuation as protective measure for the general public, whichever is larger at each distance considered. ISEPB considers that the minimum range of acceptable distances within which this determination is to be made is the distance at which the referenced protective action guide level is reached in 8 hours from the onset of release.

3. Completion of initial accident assessment measures, including dose projection, and notification to offsite authorities within fifteen minutes or within the calculated time at which the dose at the exclusion radius would reach the lower limit of the range of protective action guide levels to be adopted (for evacuation), whichever is larger.

At the FSAR stage, the organization and content of a generally acceptable emergency plan for a nuclear power plant to implement the requirements of 10 CFR Part 50, Appendix E, Parts III and IV, is given in Appendix A to this standard review plan.

III. Review procedures

At the PSAR stage, the review consists of an evaluation of the information submitted by comparison of this information with the foregoing Acceptance Criteria. The reviewer should determine that all of these criteria are satisfied, exercising his judgment as to the reasonableness and adequacy of the qualitative factors involved, in the light of emergency planning objectives.

The reviewer should gain familiarity with the proposed site, including the exclusion area, low population zone, demography, and land use factors, with the proposed plant design and layout, and with the calculated dose consequences of design basis accidents postulated by the applicant. To this end the reviewer should examine relevant sections of the PSAR, particularly Chapters 1.0, 2.0 and 15.0. This information may be supplemented by the use of United States Geological Survey grid maps, road maps, and a personal visit to the site of the reviewer.

With respect to the applicant's analysis and findings relative to emergency planning for evacuation, the reviewer should assess the credibility and adequacy of time factors presented by the applicant in the light of emergency operations experience and should analyze them to determine that the time estimates or allocations for sequential actions are consistent with the objectives and criteria set forth in II above. In addition he should assure that calculational methods and assumptions used by the applicant for dose projections are generally consistent with those found acceptable to the staff for purposes of demonstrating conformance with 10 CFR Part 100 siting criteria. Consultation with other members of the staff may be necessary to gain this assurance.

For cases in which the reviewer determines that there are site-related population, road network, or land use factors, or unique accident considerations which present potential problems for emergency planning, he may develop and recommend independent techniques to determine certain acceptable emergency plan design objectives for that site.

At the FSAR stage, the review consists of a careful examination of the applicant's emergency plan. The requirements of 10 CFR Part 50, Appendix E, Parts III and IV, and the elements of emergency planning set forth in Appendix A to this standard review plan should be used as checklists for detailed comparisons with the applicant's plan.

IV. Evaluation findings

At the conclusion of the PSAR stage review, a finding of acceptability of the applicant's defined low population zone with respect to the definition in 10 CFR § 100.3(b), should be transmitted to the Accident Analysis Branch.

The evaluation finding for this section at the PSAR stage should be substantially equivalent to the following statement:

"The applicant has described his preliminary plans for coping with emergencies. An onsite Emergency Coordinator will direct the implementation of the Emergency Plan in accordance with detailed written emergency procedures. Initial contacts and arrangements have been made with the following agencies: (listing by name). The (identity of state agency) has been identified as having primary responsibility for radiological emergency planning in the environs of the proposed facility.

"In-plant monitors will provide the first indication of a radiological emergency. Provisions will be made for surveys by portable meters and air sampling devices on a timely basis. The plant control room has been designed for continuous occupancy and will be the principal emergency control center. One alternate center will be designated. Emergency kits will be stored at the primary assembly area. Decontamination facilities and a first aid room will be provided. Arrangements have been initiated with area hospitals to treat contaminated injury cases. All plant personnel will receive training in emergency procedures and periodic drills will be conducted.

"Analyses have been performed to confirm the practicability of taking protective measures, including evacuation, within and beyond the site boundary during the expected lifetime of the plant, and appropriate criteria have been identified for the design of an acceptable emergency plan.

"We have reviewed the applicant's preliminary plans for coping with emergencies and consider that they meet the requirements of 10 CFR Part 50, Appendix E, and are acceptable."

The evaluation finding for this section at the FSAR stage should be substantially equivalent to the following:

"The applicant has formulated and submitted an Emergency Plan which describes the program for coping with emergencies within and beyond the site boundary. The plan includes a description of the organizational control extending from the on-site emergency organization to off-site agencies, specific emergency measures to be taken as indicated by defined accident assessment techniques, including protective measures, for persons subject to potentially excessive radiological exposures, and facilities and supplies needed for coping with emergencies, including redundant communications equipment. The plan also describes arrangements made for providing necessary medical attention for persons with contaminated injuries, and provisions for maintaining an adequate emergency preparedness posture throughout the expected lifetime of the plant through training, exercises, and drills.

"The plan has been determined to be acceptably coordinated with the radiological response planning of the (state name and agency identification).

"We have reviewed the applicant's Emergency Plan and consider that it meets the requirements of 10 CFR Part 50, Appendix E, is responsive to the specific requirements of the staff, and provides an adequate basis for an acceptable state of emergency preparedness. Details and procedures to implement the Emergency Plan require inspection and evaluation by the Directorate of Regulatory Operations prior to the issuance of an Operating License."

Modifications or additions to this statement may be necessary to highlight features of the review of emergency planning which are unique to the plant or site in question.

V. *References*

1. Appendix A, "Emergency Plans for Nuclear Power Plants", attached hereto.
2. 10 CFR Part 50, Appendix E, "Emergency Plans for Production and Utilization Facilities".
3. Regulatory Guide 1.70, "Standard Format and content of Safety Analysis Reports for Nuclear Power Plants," Rev. 2.

APPENDIX A—STANDARD REVIEW PLAN 13.3

EMERGENCY PLANS FOR NUCLEAR POWER PLANTS

Discussion

Regulatory concern for emergency planning is directed primarily at situations involving real or potential radiological hazards. Such hazards may place the health and safety of one or more persons in jeopardy. Emergency planning should aim to diminish the degree of jeopardy by preparing for timely action on the part of individuals who constitute a coordinated emergency organization. Although it is not practicable to develop a completely detailed response procedure for every conceivable type of emergency situation, advance planning can create a high order of preparedness and assure an orderly and timely decision-making process at times of stress as well as the availability of equipment, supplies and essential services.

An important element of emergency planning for nuclear power plants is the recognition of a need to cope with a very broad spectrum of potential consequencies. Federal, state, and local agencies as well as the applicant-licensee have responsible roles to play in both the planning and the implementation of emergency preparedness procedures. Federal interagency responsibilities for nuclear incident planning have been set forth in a Federal Register notice of January 24, 1973, by the former Office of Emergency Preparedness (now the Federal Disaster Assistance Administration). To a large extent, these responsibilities are directed toward a coordination of effort to provide assistance to state and local governments in their planning. This policy is based upon the recognition that state and local governments have the necessary authority to implement emergency measures in their jurisdictions. Although federal agencies can and will respond to emergencies arising from nuclear power plant activities if necessary, such response should be regarded primarily as backup and not a substitute for responsible action by licensees and state and local governments.

In the preparation of an emergency plan for a specific nuclear power plant, the applicant should be guided by the following criteria to clarify the scope, content, and purpose of the document which describes the plan. The emergency plan should incorporate sufficient detail so that other participating organizations and agencies with related plans may review it and determine that they are coordinated effectively with one another. Detail which can reasonably be expected to change from time to time, e.g., names and telephone numbers, equipment and supplies inventory lists, or step-by-step procedures or check lists which may be altered as a result of experience or test exercises, should not be incorporated in the plan. The document itself should not be considered as a primary working document to be used during an emergency. Implementing procedures documents, keyed to the plan, should be available for this purpose. The latter documents should not be necessary for licensing review. However, they should be available for inspection by the Commission's Directorate of Regulatory Operations, and should be transmitted, if applicable, to appropriate state or local agencies.

The plan document should also clarify its scope relative to interfacing plans and procedures within the operating organization, e.g., emergency and off-normal operating procedures within the plant, radiation protection program and procedures, and security plans.

Although a part of the final safety analysis report, it is recommended that the plan be prepared as a separate document.

Branch recommendations

A. Each applicant's emergency plan should include provisions for handling emergencies both within the site of his plant and in the environs of the site. Responsibility for planning and implementing all emergency measures for persons within the site boundaries rests with the licensee. Planning and implementation of emergency measures in the environs of the site arising from onsite activities should be coordinated with local, county, state, and federal agencies having emergency responsibilities and should be described in the applicant's emergency plan. Such planning should generally be increasingly definitive in its provisions for emergency measures as the regions of consideration get closer to the site and the plant itself.

B. The scope and content of a nuclear power plant emergency plan should be substantially equivalent to that outlined in the following section, entitled "Organization and Content of a Nuclear Power Plant Emergency Plan".

ORGANIZATION AND CONTENT OF A NUCLEAR POWER PLANT EMERGENCY PLAN

CONTENTS

ORGANIZATION AND CONTENT OF A NUCLEAR POWER PLANT EMERGENCY PLAN

(In the following, the decimal notation identifies recommended major subject headings for the organization of an emergency plan document. The text, including portions identified by alphabetic notation, gives specific guidance or recommendations as to the content of the section or sub-section.)

For clarity, certain terms are employed with specific definitions as follows:

Definitions

Assessment actions—means all of those actions taken after an accident has occurred which are collectively necessary to make decisions to implement specific emergency measures.

Corrective actions—means emergency measures taken to ameliorate or terminate an emergency situation at or near the source of the problem.

Protective actions—means those emergency measures taken after an accident or an uncontrolled release of radioactive materials has occurred, for the purpose of preventing or minimizing radiological exposures to persons which would be likely to occur if the actions were not taken.

Population-at-risk—means those persons for whom protective actions are or would be taken.

Affected persons—means persons who have been radiologically exposed or physically injured as a result of an accident, to a degree requiring special attention as individuals, e.g., decontamination, first aid, or medical services.

Recovery actions—means those actions taken post-emergency to restore property to its pre-emergency condition as nearly as possible.

Protective action guides—means projected radiological dose, or dose commitment, values to individuals in the general population which warrant protective action following a contaminating event.

Emergency action levels—means radiological dose rates, specific contamination levels of airborne, waterborne, or surface-deposited concentrations of radioactivity, or specific instrument readings, which may be used to prescribe specific emergency measures.

1.0 SCOPE AND APPLICABILITY

This section of the plan should define the unit, plant, station, or area to which the plan is applicable, and a summary of its inter-relationships with (1) its implementing procedures, (2) plant operating, radiological control, and industrial security procedures, (3) other emergency plans of the company, e.g., an overall corporate plan, and (4) emergency plans of other participating agencies, particalarly the responsible state agency.

2.0 SUMMARY OF EMERGENCY PLAN

This should describe the key elements of overall emergency planning logic incorporating graded emergency classifications of increasing severity and their relationship to the participating status of onsite and offsite personnel and agencies.

3.0 EMERGENCY CONDITIONS

3.1 Classification System

An emergency plan should characterize several classes of emergency situations. The system of classification employed should consist of mutually exclusive groupings (to avoid ambiguity) but should cover the entire spectrum of possible situations. Each class should incorporate (1) a specific emergency organization alerting and mobilization procedure, and (2) a set of predefined preliminary actions to be taken by designated emergency organization personnel. Succinct descriptive rather than numerical or alphabetical classification designations are recommended to give better immediate clues to personnel as to the scope and character of the situation.

An acceptable classification scheme is described below in qualitative terms. This part of the emergency plan should describe the criteria for recognizing and declaring each class, including specific emergency action levels for the last three classes.

(a) *Personnel Emergency.*—Accidents or occurrences onsite may require emergency treatment of individuals. This classification applies to situations which have no potential for escalation to more severe emergency conditions. There may be no effect on the plant, nor does it necessarily involve immediate operator action to alter plant status. A personnel emergency does not activate the entire plant emergency organization but may activate teams such as first aid. It may also require special local services such as ambulance and medical.

Implementing procedures for the handling of this class of emergency may also be incorporated in the plant's radiation protection procedures and general industrial safety procedures.

Included in this class are injuries which may be complicated by contamination problems or excessive radiation exposures to onsite personnel.

The recognition of this class of emergency is primarily a judgment matter for plant staff supervisory or management personnel. Its importance as part of the classification scheme rests to some extent on its "negative" information content, viz. that the incident giving rise to an emergency is restricted in its scope of involvement. This section of the plan should designate the classification criteria, and enumerate discrete accident situations which would give rise to the use of this class.

(b) Emergency Alert.—Specific situations may arise that can be recognized as creating a hazards potential that was previously non-existent or latent. In and of itself the situation has not yet caused damage to the plant nor harm to personnel and does not necessarily require an immediate change in plant operating status. Inherently, then, this is a situation in which time is available to take precautionary and constructive steps to prevent the realization of an accident and to mitigate the consequences should it occur. An emergency alert situation may be brought on by either man-made or natural phenomena.

Emergency alert conditions imply a rapid transition to a state of readiness by the plant personnel, the possible cessation of certain routine functions or activities within the plant which are not immediately essential, and possible precautionary actions which the specific situation may require. Examples of situations which might be placed in this class are: threats to or breaches of plant security measures such as bomb threats or civil disturbance; severe natural phenomena in the plant environment such as floods, earthquakes, tsunamis, hurricanes, or tornadoes; emergency situations such as fires at adjacent facilities; release of a toxic or noxious gas in or near the plant; or flooding offsite caused by malfunction or failure in some part of the plant cooling water system. This section of the emergency plan should identify specific candidate situations for emergency alerts and the quantitative criteria that would guide the descision to implement each. Qualitative criteria should be added for other candidate situations to guide the decision of on-site supervisory personnel.

(c) Plant (Unit) Emergency.—This class incorporates physical occurrences within the plant requiring full plant staff emergency organization response. The initial information and assessment indicates that it is very unlikely that an offsite hazard will be created. However, substantial modification of plant operating status is a highly probable corrective action if this has not already taken place by the actions of automatic protective systems. Although it is judged that the emergency situation can be corrected and controlled by the plant staff, notification of corporate headquarters staff to put them on an alert status is prudent. In turn, notification of appropriate offsite agencies as to the nature and extent of the incident is advisable. Evacuation of the plant is not anticipated in this class although protective evacuations or isolations of certain plant areas may be necessary.

Examples of situations which might fall into this class are those accidents which have been analyzed in the FSAR as events which are predicted to have no radiological consequences offsite. Fires, explosions or explosive gas releases, or in-plant flooding conditions, may also fall into this class.

Activation levels for declaring plant emergencies should be based upon the recognition of an immediate need to implement in-plant emergency measures to protect or provide aid to affected persons in the plant and to mitigate the consequenses of damage to plant equipment, coupled with a positive observation that (a) effluent and other radiological monitors do not indicate the possibility of a site emergency, and (b) there is no apparent breach of any fuel cladding, primary system boundary, or containment. This section should describe the alarm conditions or combinations of alarm conditions and the emergency action levels for initiating a plant emergency and their bases.

(d) Site (Station) Emergency.—This class involves an uncontrolled release of radioactive materials into the air, water, or ground to an extent that initial information and assessment indicates that protective actions offsite may be desirable. Mobilization and readiness of offsite emergency organizations is prudent. Protective actions are likely to include evacuation of plant areas other than control rooms and emergency stations, and should include provisions for evacuation of construction personnel during those periods when additional units are under construction on the same site. Assessment actions will include monitoring of the environment.

Situations which are likely to fall into this class include those accidents analyzed in the FSAR which are predicted to have small to moderate releases at the exclusion radius. It should be anticipated that site emergencies would not normally be preceded by a plant emergency although this evolution should not be excluded.

Emergency action levels declaring a site emergency should be defined in terms of instrument readings or alarms in the control room. To avoid false alarms or to minimize their frequency of occurrence, the levels may be defined so as to require corroborating evidence from two independent sources having input to the control room. Indications from effluent monitors should be included. Site emergencies should also be declared on the basis of evidence of apparent breaches in fuel cladding, primary system boundaries, or containment when otherwise a plant emergency would be declared. The bases and criteria used to define the instrument alarm levels should be described. Suitable criteria would be protective action guide values at a security fence, or exclusion area or site boundary and the bases would show how the effluent monitor readings relate to such values. Protective action guides selected for this purpose should be below the siting guideline values of 10 CFR Part 100 and should have the concurrence of state authorities. Federal agency guidance is available to assist in the selection of acceptable protective action guides.

(c) *General Emergency.*—This is an occurrence characterized by offsite consequences requiring protective action measures as a matter of prudence or necessity. Evacuation of the site may also be necessary under extreme circumstances. Emergency action levels for declaring a general emergency should be defined.

Two categories, short term and long term, should be recognized. The former is guided by direct radiation or inhalation hazards, while the latter is guided primarily by contamination hazards. General emergency action levels may be based upon confirmatory measurements taken in the field to the extent that it can be shown that they can be taken and evaluated rapidly enough to permit adequate time for the protective actions to be accomplished. The levels for severe short term situations require definition in terms of effluent and other onsite monitor indications. As in the previous case, the bases and criteria used to define the relevant instrument levels should be described.

3.2 Spectrum of Postulated Accidents

Accident analysis sections of safety analysis reports are primarily concerned with the design responses of a plant to postulated malfunctions or equipment failure and include estimates of the radiological consequences of discrete accidents. By contrast, emergency planning is concerned with individual and organizational responses to the continuum of potential accident situations which must include those discrete accidents which have been hypothesized. This section of the emergency plan should show that each is encompassed within the emergency characterization classes and provide a summary analysis of their implications for emergency planning. Implications to be considered include:

(a) Instrumentation capability for prompt detection and continued assessment, including functional applicability, range, response time, locations of sensing and readout elements (including alarms), and backup or redundant capability.

(b) Manpower requirements for assessment, including record keeping; for corrective actions; for protective actions including communications requirements; and for aid to affected persons.

(c) The timing of and the time required for the implementation of each emergency measure which may be brought into play.

4.0 ORGANIZATIONAL CONTROL OF EMERGENCIES

Starting with the normal operating organization as a base, this section of the plan should describe the emergency organization that would be activated on the site and its augmentation and extension offsite. Authorities and responsibilities of key individuals and groups should be delineated. The communication links established for notifying, alerting, and mobilizing emergency personnel should be identified.

4.1 NORMAL OPERATING ORGANIZATION

Both day and night shift operating staffs (crews) should be described, indicating clearly who is in the immediate onsite position of responsibility for the plant and station (normally a shift supervisor) and his authority and responsibility for declaring an emergency.

4.2 ONSITE EMERGENCY ORGANIZATION

This section should describe the mobilization billets of plant staff personnel for controlling each class of emergency for both day and night shift situations.

4.2.1 *Direction/Coordination*

The position title of that person who is designated to take charge of emergency control measures onsite should be clearly identified. A specific line of success for this function should also be given. A policy statement describing the scope of authority and responsibility vested in that role by the company (applicant) should be included. Functional responsibilities assigned to this individual should be described, and should include a summary of those preliminary assessment procedures that would be followed to prescribe or guide his decision to classify and declare an emergency.

4.2.2 *Plant Staff Emergency Assignments*

The plan should specify the functional areas of emergency activity to which members of the plant staff are assigned, including an indication of how the assignments are made for both day and night shifts, and for plant staff members both onsite and away from the site. Functional areas should include:

1. Plant systems operations.
2. Radiological survey and monitoring.
3. Fire fighting.
4. Rescue operations.
5. First aid.
6. Decontamination.
7. Security of plant and access control.
8. Repair and damage control.
9. Personnel accountability.
10. Record keeping.
11. Communications.

4.3 AUGMENTATION OF ONSITE EMERGENCY ORGANIZATION

This section should describe two categories of offsite supporting assistance to the plant staff emergency organization. These can be either directed, authorized or requested by the company management to perform special emergency assistance functions.

4.3.1 *Headquarters support*

Headquarters management, administrative, and technical personnel should be prepared to augment the plant staff, both in emergency planning and in the performance of certain functions required to cope with an emergency. The following special functions are considered appropriate for headquarters support and should be incorporated in the overall plan, although company policy and organizational features may dictate variations in modes of assigning responsibilities for these functions among headquarters personnel, plant staff personnel, and outside support organizations.

1. Environs monitoring.
2. Logistics support for emergency personnel, e.g., transportation, temporary quarters, food and water, sanitary facilities in the field, and special equipment and supplies procurement.
3. Technical support for planning reentry/recovery operations.
4. Notification of governmental authorities.
5. Public relations and information release, coordinated with governmental authorities, including steps taken to inform visitors to the plant or information center, and to occupants in the environs of the site, of how the emergency plans provide for notification to them and how they can expect to be advised as to what to do.

The emergency organizattion status of supporting headquarters personnel should be specified, relative particularly to the person directing the plant emergency organization.

In some instances, companies may provide for certain emergency supporting services to their plants by contract with private organizations. Where this is the case, the nature and scope of the support services should be characterized here. (The Commission may find it necessary to request evidence of the qualifications of such contractors.) Specific services by the contractors should be identified as such at the appropriate places in the emergency plan document.

4.3.2 *Local services support*

This section should identify the extension of the organizational capability for handling emergencies to be provided by ambulance, medical, hospital, fire, and police organization. Evidence of the arrangements and agreements reached with such organizations should be included in an appendix and referenced here, along with references to the parts of the plan in which their functions are primarily described.

4.4 COORDINATION WITH PARTICIPATING AGENCIES

This section should identify the principal state agency (designated state authority) and other governmental (local, county, state, and federal) agencies having planning and action responsibilities for emergencies, particularly for radiological emergencies, in the area in which the plant is located. If the boundary line between two political entities, e.g., counties or states, passes within the low population zone or approximately four miles of the site, agencies from both entities should be included. Subsections for each such agency should describe the following:

(*a*) Identity of agency.

(*b*) Summary of written agreement with agency which clearly defines the authority and responsibility of the agency for emergency preparedness planning, and for emergency response in the public domain, particularly relative to those of the licensee and to those of other agencies. (Copies of such agreements should be included in an appendix, along with a copy or summary of relevant parts of that agency's emergency plan.)

(*c*) Activation of agency function, including titles and alternates of both ends of the communications links, and primary and alternate means of communication.

(*d*) The designation and location of the emergency operations center of each agency.

(*e*) Support of the agency that may be provided by the company emergency organization, which may include (1) information on plant status, monitoring results, dose predictions, (2) recommendations or requests for specific actions, and (3) logistics support.

Typical agencies to be included here are: law enforcement agencies (not included above, e.g., state police/highway patrol), departments of health and environmental protection, civil defense and emergency/disaster control agencies, AEC regional operations offices, and the AEC regional office of Regulatory Operations.

5.0 EMERGENCY MEASURES

Specific emergency measures should be identified in this section and related to action levels of criteria that specify when the measures are to be implemented. They should be organized with respect to each emergency classification. Preplanned action levels and criteria should be designed to assist and guide, or in some cases specifiy, the decision-making functions.

The planning represented by this section should lead to more detailed emergency procedures and assignments for executing tasks by appropriate members of the total emergency organization. Emergency measures begin with the activation of an emergency class and its associated emergency organization. The additional measures may be organized into assessment actions, corrective actions, protective actions, and aid to affected persons.

5.1 Activation of Emergency Organization

The emergency conditions classified in Section 3.1 involve the alerting or activation of progressively larger segments of the total emergency organization. This section should describe how the necessary communications steps are taken to alert or activate emergency personnel under each class, including, in particular, action levels for notification of offsite agencies.

5.2 Assessment Actions

Effective coordination and direction of all elements of the emergency organization require continuing assessment throughout the duration of an emergency situation. Assessment functions should be incorporated in explicit procedures for each emergency classification They should be identified in this section and may include the following:

(*a*) Surveillance of control room instruments and emergency control center monitors, radiological and meteorological, installed, pursuant to General Design Criteria 13 and 64 of 10 CFR Part 50, Appendix A.

(*b*) Surveillance of containment integrity.
(*c*) In-plant radiological surveys.
(*d*) Site and site boundary surveys.
(*e*) Environs surveys and monitoring.

1. Plume and other effluent surveillance for short term assessment. Planning should consider type of data sought; instrument and equipment requirements; monitoring team transportation facilities, e.g., aircraft, boats, vehicles; methods and accuracy of plume location; and potential use of fixed off-site monitoring facilities.

2. Contamination surveillance. Planning should consider the timing, frequency, and types of samples to be collected, such as soil, vegetation, food, milk and water supplies, and potential locations for reconcentration, e.g., in air intake filters.

(*f*) Data reporting, reduction and analysis.
(*g*) Interviewing evacuees or other witnesses of the accident.
(*h*) Notification of assessment results for modification of emergency measures in progress, if necessary.

5.3 Corrective Actions

Many emergency situations involve actions which can be taken to correct or mitigate the situation at our near the source of the problem. This section should identify those actions, such as fire control, and repair and damage control, which would be implemented when necessary. Emergency exposure criteria for personnel undertaking corrective actions should be included.

5.4. Protective Actions

This section should describe the nature of protective actions which the plan contemplates, the protective action levels, the area involved, and the means of notification to the population-at-risk. Protective actions to be taken offsite by other agencies should be described.

5.4.1 Protective Cover, Evacuation, Personnel Accountability.—The emergency plan should provide for timely relocation of persons to prevent or minimize exposure to direct radiation or airborne hazards. The following items should be included:

1. Plant Site:

(a) Action criteria.
(b) The means and the time required to notify persons involved. These should include:
(1) Employees not having emergency assignments.
(2) Working and non-working visitors.
(3) Contracter and construction personnel.
(c) Control of public access areas on or passing through site or within exclusion area.
(d) Evacuation routes, transportation of personnel, and reassembly areas, including inclement weather and high traffic density alternatives.
(e) Missing persons check.
(f) Radiological monitoring of evacuees.

2. Off-Site Areas:

(a) Action criteria including inclement weather alternatives.
(b) Company emergency organization responsibilities.
(c) Agency responsibilities.
(d) The means and the time required to notify and the expected response of persons involved. These should include:
(1) Adjacent businesses, property owners, and tenants.
(2) Nearby schools or recreational facilities.
(3) General public, in the environs.

5.4.2 Use of Protective Equipment and Supplies.—Additional protective actions which should be considered in emergency planning include measures for minimizing the effects of radiological exposures or contamination problems through the distribution of special equipment or supplies. Measures to be considered include:

1. Individual respiratory protection.
2. Use of protective clothing.
3. Individual thyroid protection.

For each measure which might be used, a description should be given of:

1. Criteria for issuance.

2. Location(s) of items.
3. Means of distribution to onsite and offsite persons.

5.4.3. Contamination Control Measures.—Provisions should be made for preventing or minimizing ingestion of or exposure to contaminated areas or materials. (Control of in-plant contamination should be described in the facility radiological protection procedures and need not be repeated here.) Measures for the protection of onsite persons outside of fenced security areas and offsite persons should include:

1. Isolation or quarantine and area access control.
2. Control of the distribution of affected commercial agricultural products.
3. Control of public water supplies.
4. Means for providing advisory information regarding the use of potentially affected home food and water supplies.
5. Criteria for permitting return to normal use.

Action levels and responsibility for execution of each measure contemplated should be described.

5.5 Aid to Affected Personnel

This section of the emergency plan should describe measures which will be used to provide necessary assistance to persons injured or radiologically exposed. The following matters should include:

5.5.1 Emergency Personnel Exposure Criteria.—Exposure limits should be specified for voluntary entry or reentry of areas to remove injured persons and limits for emergency personnel who may provide first aid, decontamination, ambulance, or medical treatment services to injured persons.

5.5.2 Decontamination and First Aid.—Capabilites for decontaminating personnel for their own protection and to prevent or minimize further spread of contamination should be included, along with a brief description of first aid capabilities of appropriate members of the emergency organization.

5.5.3 Medical Transportation.—Arrangements for transporting injured personnel, who may also be radiologically contaminated, to medical treatment facilities should be specified.

5.5.4 Medical Treatment.—Arrangements made for local and back-up hospital and medical services, and the capability for radiation exposure and uptake evaluations should be described.

For both hospital and medical services, the plan should incorporate assurance that the required services are not only available, but also that persons providing them are prepared and qualified to handle radiological emergencies. Written agreements with respect to arrangements made by the applicant, which should be included in the appendix, would facilitate this determination.

6.0 EMERGENCY FACILITIES

This section of the emergency plan should identify, describe briefly, and give the locations of the following categories of items.

6.1 Emergency Control Centers

This should include the principal and, if provided for, alternate onsite location from which effective emergency control direction is given. One alternate offsite location under the jurisdiction of the applicant should also be described. Their descriptions should also specify prevailing wind direction and evacuation routes.

6.2 Communications Systems

Brief descriptions should be given of both internal and external communications systems that would perform vital functions in transmitting and receiving information throughout the course of an emergency.

6.3 Assessment Facilities

Many of the emergency measures described in Section 5.0 will depend upon the availability of monitoring instruments and laboratory facilities. This section should list monitoring systems that are to be used to initiate emergency measures as well as those used for continuing assessment. Organization of the listing should be as follows.

6.3.1 Onsite Systems and Equipment.—

1. Natural phenomena monitors, e.g., meteorological, hydrologic, seismic.
2. Radiological monitors, e.g., process, area, emergency, effluent, portable monitors and sampling equipment.

3. Non-radiological monitors, e.g., reactor coolant system pressure, temperatures, containment pressure, temperature, liquid levels, flow rates, status or lineup of equipment components.

4. Fire detection devices.

6.3.2 Environs Monitoring Facilities and Equipment—

1. Natural phenomena monitors.
2. Radiological monitors.
3. Laboratory facilities, fixed and mobile.

Reference may be made to the applicable part of the safety analysis report for more detailed descriptions, if applicable.

6.4 Protective Facilities

Specific facilities mentioned in Section 5.4.1 which are intended to serve a protective function should be described, emphasizing those features of the facility which assure its adequacy with respect to capacity for accommodating the number of persons expected, and with respect to shielding, ventilation, and inventory of supplies. Such facilities might include fallout shelters or similar areas, and reassembly points. If design details have been provided elsewhere in the safety analysis report, a brief summary only need be given here, along with a reference to the detail.

6.5 First Aid and Medical Facilities

A summary description of onsite facilities should be provided. Offsite medical facilities should be described in the appendix, along with the agreements providing for their use.

7.0 MAINTAINING EMERGENCY PREPAREDNESS

This section of the plan should describe the means to be employed to assure that the plan continues to be effective throughout the lifetime of the nuclear facility.

7.1 Organizational Preparedness

7.1.1 Training.—This section should include a description of periodic training programs to be given to all categories of emergency personnel. Specialized training for the following categories should be included:

1. Directors or coordinators of the plant emergency organization.
2. Personnel responsible for accident assessment, including control room shift personnel.
3. Radiological monitoring teams.
4. Fire, and repair and damage control teams.
5. First aid and rescue team members.
6. Local services personnel.
7. Medical support personnel.

7.1.2 Drills.—Periodic (at least annual) announced drills should be incorporated in the emergency plan. These should be pre-planned simulations of accidents to test the adequacy of timing and content of specific implementing procedures and to test emergency equipment. Arrangements should be made for critiques of the drills. Coordinating drills should be made with participating agencies at least annually, testing at a minimum the communications links. An initial coordinated drill with participating agencies should be planned and carried out prior to fuel loading of the first unit at any site.

7.2 Review and Updating of the Plan and Procedures

Provision should be made for an annual review of the emergency plan and for updating and improving procedures based upon training, drills, and changes onsite or in the environs. Means for maintaining all coordinate elements of the total emergency organization informed of revisions to the plan or relevant procedures should be described.

7.3 Emergency Equipment and Supplies

The operational readiness of all items of emergency equipment and supplies should be assured. The plans and schedules for performing maintenance, surveillance testing, and inventory of emergency equipment and supplies should be described.

8.0 RECOVERY

This section should describe general plans, including applicable criteria, for restoring property as nearly as may be possible to its pre-emergency status.

9.0 APPENDIX

The appendix should include the following items:

1. Copies of agency agreement letters and copies or summaries of interfacing emergency plans.

2. Plots of calculated time-distance-dose for the most serious design basis accident as required by Revision 2 of the "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants".

3. Listings by title of written procedures which implement the plan.

4. Listings by category of protective equipment and supplies.

The written procedures themselves and detailed cataloguing of protective equipment and supplies should be available at the plant site for inspection at any time by a representative of the Commission's Directorate of Regulatory Operations.

Senator MONTOYA. I will ask you this question. Do you provide in your manual authorization for the use of water in this kind of a situation for the future?

Mr. RUSCHE. Mr. Chairman, we neither authorize nor preclude it. Now, in the case of particular situations I am going to discuss in a few moments, what we propose for Browns Ferry in the particular circumstances that exist there and I will tell you in a few moments that we are going to require that water be available under the right circumstances for initiation in the cable spreading room for the return of Browns Ferry to operation.

Mr. ANDERS. And of course Dr. Hanauer's group is looking not only at Browns Ferry but more importantly at the implications of Browns Ferry situation to all nuclear power plants and our own procedures.

Senator MONTOYA. What kind of measures have you provided for in the event that there is radiation and personnel exposure?

Mr. ANDERS. You mean our personnel treatment and evacuation requirements?

Senator MONTOYA. Yes. If there is radiation in the environment or an escape of radiation, what countermeasures have you provided for?

Dr. KNUTH. Two aspects. The question is if there was radiation associated in the area where work had to be done, what would occur. As part of the operating license there is a requirement that the licensee have health physics coverage, they have fixed monitors, they have airborne samplers and this type of thing to determine the radiation levels in a particular area where people have to go to do whatever they have to do. In this particular case some of the monitors failed as a result of the fire. However, the personnel at the site also had hand-held equipment, they also took grab samples in particular areas; to keep a surveillance of what the radioactivity levels were in buildings where personnel were working. If a person has to go into a radiation environment the procedures do account for establishing what the environment is he has to work under, and what length of time he is allowed to be in that area from a personnel protection standpoint.

So the procedures do allow work to be conducted, such as putting out a fire or realigning valves or what-have-you, in a radiation environment.

Mr. ANDERS. They are pretty specific requirements, Senator.

Senator MONTOYA. Are utilities prepared to take the necessary countermeasures when operating personnel themselves might be subject to radiation exposure?

Dr. KNUTH. Yes. As part of their emergency procedures if a person does receive exposure they are required to have working relationships with local hospitals and so forth to treat such individuals, yes.

EFFICIENCY OF OPERATORS' ACTIONS

Senator MONTOYA. Did the plant operators generally do the right things during the fire to insure the health and safety of the public? What is your opinion?

Mr. RUSCHE. Mr. Chairman, our evaluation of the safety during the incident and following the incident confirms that the operators did in the main take corrective actions that were effective.

Senator MONTOYA. What do you mean in the main? What about the other, off the main? [Laughter.]

Mr. RUSCHE. Let me answer your question bluntly. Yes.

Senator MONTOYA. Yes, what?

Mr. RUSCHE. They took corrective actions that protected the plant very effectively.

Senator MONTOYA. Then "in the main" is out of context here.

Mr. RUSCHE. Yes.

Mr. ANDERS. Well, Senator, I would add that in a complicated event like this, there are probably several possible variations in the course of action, at least minor variations in dealing with such a problem. The analysis that Dr. Hanauer is doing may indicate that some other route may have been more efficient or effective. But I would say that, in the main, the actions they took and the sequence taken, even though—as our report indicates—there might have been some nonproductive activity, such as efforts to manually position valves, in retrospect, the net result of their activity was the maintenance of a safe plant.

Senator MONTOYA. What about that, Dr. Hanauer? What have you to say about this?

Mr. ANDERS. Dr. Hanauer.

Dr. HANAUER. My answer is the same. Overall the actions were, I would say, highly successful. You can go back and look at the details of what each person did at each moment and find a few actions that in hindsight would have been better off done some other way, but overall they did very well.

Senator MONTOYA. Now, what does your hindsight tell you that should have been done?

Dr. HANAUER. One of the principal areas where hindsight is being exercised is the question of whether they should have put water on the fire earlier.

Senator MONTOYA. What does your hindsight recommend?

Dr. HANAUER. The review group hindsight recommends that water should have been put on earlier and in this respect we disagree with the judgment of the operators.

Now, I would like to say that we are sitting in a nice quiet room here. We are not in a nuclear powerplant with a fire. We would not want to imply either that we might have done better exposed to this same stress or that they were being silly or anything like that. We just now think, knowing what we do, that water would profitably have been applied earlier and that perhaps the outage of equipment could have been minimized by so doing.

Senator MONTOYA. Give me some other instances of where hindsight would have played a better role.

Dr. HANAUER. Well, another example is the arrangements that were available to provide for air-breathing apparatus in the very bad atmosphere where these people had to work. It turns out that some air canisters were mistakingly gotten from the shop where they had been put for repair. Now that is the sort of thing that in hindsight—no, you should not get the ones that are broken in the shop; that is counterproductive. But it really, in the long run, didn't hurt anything and there were some that were available for use and were used and they put the fire out.

Senator MONTOYA. What else?

Dr. HANAUER. Well, it is very obvious you should not use candles to detect leaks around flammable materials.

Senator CASE. When was the vacuum shut off, if it was shut off, that drew the candle flame into the chamber?

OPERATION OF THE VENTILATION SYSTEM

Dr. KNUTH. I believe we have to check that and give an answer for the record. Right now my best recollection is that the ventilation system was on and off at various times during the period in question. It was certainly running at about 4:30 in attempts to remove smoke so people could get into the area again, but between the period of 1:00 and 4:00 my recollection is that it was on and off at various times, but it may have been off the entire time.

Senator CASE. The regular ventilating system that operates in a negative way?

Dr. KNUTH. Yes.

Mr. ANDERS. It is designed to insure that air goes through filters.

Senator CASE. I understand but the effect was to draw the candle flame into the chamber and perhaps fan the fire.

Mr. ANDERS. It did indeed.

Dr. MASON. Prevent the out-leakage of air.

Senator CASE. How long did it run after the fire started?

Mr. ANDERS. I believe we will have to get that for the record. I don't believe the flow was stopped immediately.

Dr. KNUTH. We are going to have to get that for the record. It was on from about 4:30 on because that was when they resumed fire fighting activities. Between the period of 1:00 and 4:30—we will have to check the record.

Senator CASE. Between the time the fire started?

Dr. KNUTH. It didn't shut off until after about 1 o'clock.

[Material later received follows:]

The Unit 1 reactor building ventilation system was inoperable from approximately 12:45 p.m. until 4 p.m.; the Unit 2 system was out of service from about 2 p.m. until about 9 p.m., when it was returned to service.

For a short time after 4:40 p.m., the Unit 1 reactor building ventilation system was placed in operation in an attempt to clear smoke from the fire area. This was stopped after about 20 minutes when it appeared that it was fanning the fire. Routine operation of the ventilation system was initiated at about 7 p.m.

It should be noted that during the time these ventilation systems were out of service, natural draft continued to provide a small inflow of air into the reactor building.

The reactor building ventilation system maintains a negative air pressure in the secondary containment building with respect to the surrounding areas in order to insure that leakage is into the building. A limit on the amount of inleakage is specified in the license Technical Specifications. It is appropriate to note that the fanning effect of the draft due to the pressure differential was most pronounced during the first few minutes of the fire.

Senator CASE. Would it not have been a good idea to shut it off and shut that draft?

Mr. ANDERS. It was in that very first few minutes, I would wager, where the presence of the ventilation system causing the draft played its major negative role. That system, of course, is itself part of the safety system. Had it not been for that draft, I would bet that the maintenance men could have snuffed the fire out very easily. But, the fire was drawn out of the worker's reach and into the next building. Once that happened, the significance of the ventilation system was reduced as far as exacerbating the fire situation.

Senator CASE. I am just wondering whether this indicates there ought to be some kind of automatic shift.

Dr. MASON. That system, sir, is installed to insure that in the event of a release of radioactivity which did not occur in this case, the leakage would be through the filters which are installed for this purpose so I think we would have to look very carefully before shutting that off to be sure that the public would not be exposed to radiation. Indeed the fire was prolonged but the first concern was the protection of public health and safety, so we have to go through that very carefully.

Senator CASE. This might not happen again if you don't use open flames.

Mr. ANDERS. One of our major actions is to reduce the use of open flames in nuclear facilities.

Senator CASE. Mr. Chairman, I have to go. I will just leave one question to be answered for the record if I may and that relates to the report made by the insurance industry. This was completed some time before the regulatory agency made its report and I would like to know why that happened and also whether these gentlemen here have any substantial disagreement with the recommendations of the insurance report.

Mr. ANDERS. Let me note first that there is little comparison between the nature of these two reports. Ours is a very detailed investigation; it goes not only into the rather apparent fire causing or fire exacerbating situations at that particular plant, which I would expect could be written down rather promptly but this investigation is one phase of a very complicated three-phase review. Just unraveling the facts on which you must base future licensing actions is not an easy task.

A lot was happening. Some of the data recording computers dropped off the line—these were computers that would record time sequences. Frankly, I believe that the output of our inspection people designated to do our report did it in a very rapid manner and, indeed, possibly exposed themselves or the agency to some inaccurate conclusions which we may revise once we have had time to review the response from the utility itself.

Senator CASE. What about the question of your agreement or disagreement with the recommendations in the insurance report?

Mr. RUSCHE. Senator Case, earlier I commented, I believe, that report contains some 35 or 40 recommendations. In the review that we have made so far of the TVA proposal for returning the equipment to operation, we are in agreement with about 20 to 25 of those recommendations and the others are under review. We have reached that determination not because of the report, but having the report available was valuable to us.

Senator CASE. So your final report will deal with these specific recommendations and each one of them?

Mr. RUSCHE. Well, the requirements that we are imposing upon TVA for returning the plant to operation will include the consideration of those recommendations, and so far those recommendations include concurrence with about 20 to 25 of them.

Senator CASE. Will there be any documents or anything that will let us know with what recommendations you agree and with what recommendations you don't agree, not the TVA or anybody else?

Mr. RUSCHE. To the best of my knowledge there is no such document in existence but we would be glad to give you a point by point comparison.

Senator CASE. I would like to have that, Mr. Chairman.

Senator MONTOYA. Would you supply it for the record?

Mr. RUSCHE. Yes.

Senator MONTOYA. Would you supply the 25 on which you have agreed?

Mr. RUSCHE. Yes. My assessment will tend to show both what we have done and whether or not we agree with them. Obviously the ones we have agreed with we have adopted.

Senator CASE. Thank you, Mr. Chairman.

[Material subsequently supplied follows:]

This comparison has been prepared on the basis of the Nuclear Energy Liability and Property Insurance Association recommendations contained in their report on the fire that occurred at the Browns Ferry Nuclear Plant. As such, it must be recognized that the NRC comments are specific to the Browns Ferry Nuclear Plant and do not necessarily apply to other operating plants or plants under construction. It should be noted, however, that the NRC staff will consider the NELPIA recommendations during its review of these other plants as well as in establishing criteria and standards for the review of future plants.

A. *Cable Spreading Room* (CSR) (Cable Spreading Room and similar rooms)

NELPIA recommendations

1. A CSR for each unit should be provided. Each spreading room should be cut off and arranged totally independent of other CSR's by a fire barrier wall of 3-hours' fire resistance rating.

2. At least one 3-foot wide, 8-foot high aisle should be provided the length and width of the CSR to insure fire fighting access. Class A fire doors should be installed at each entrance (at least two) into the room.

NRC comments

The staff has reviewed the above recommendations and does not consider them necessary for Browns Ferry. TVA has committed to (a) making the CO_2 fire extinguishment system in the CSR automatic; (b) liberal use of flamastic on all cable trays in the CSR, and (c) investigating possible installation of fixed manual water spray or other alternative means of augmenting the fire extinguishment capabilities. The staff feels that these commitments, in addition to those administrative changes proposed to reduce the probability of a fire, provide sufficient protection in this area. Moreover, the arrangement of the Browns Ferry CSR is such that the cost and downtime to accomplish such modifications would not be justified in view of the limited additional benefit that would result.

NELPIA recommendation

3. A standard installation of open-head, water spray sprinklers controlled by an automatic deluge valve and products-of-combustion actuated detectors should be provided in each CSR. The deluge valve should be located outside the room and connected to the station's annunciator system.

NRC comments

The staff is requiring that the existing manual CO_2 system be made automatic and that a fire retardent material such as flamastic be utilized liberally in this area. In addition, TVA has committed to investigate the possibility of installing manual fixed nozzle water sprays or other alternative means for fire extinguishment in the CSR. The staff will evaluate this study and determine the need for additional fire extinguishment systems prior to plant startup.

NELPIA recommendation

4. One-inch hose connections, equipped with 75 ft of 1½'' woven jacket lined fire hose and adjustable spray nozzle should be provided in the CSR and located at approximately 100 ft intervals.

NRC comments

NRC agrees with the intent of this recommendation and will require manual fire hose availability. However, the specific details of such an installation must be carefully reviewed in conjunction with personnel and critical equipment safety.

NELPIA recommendation

5. The concrete floor should be pitched to drain the sprinkler and hose steam discharge to a suitable drainage facility.

NRC comments

NRC agrees. Suitable drainage capability will be provided as required based on installation of water systems.

NELPIA recommendation

6. Approved smoke and heat venting of the CSR utilizing a powered mechanical exhaust system actuated by products-of-combustion detectors should be provided.

NRC comments

The existing design is based on utilization of a CO_2 system as the primary means for fire extinguishment in the CSR. Accordingly, it is necessary to provide for isolation of the CSR from ventilation system in the event a fire is detected. TVA has committed to ensuring such isolation and NRC believe that such a system is acceptable. Modifications of this approach will be considered, based on resolution of Item 3 above.

NOTE.—These recommendations are also pertinent to cable penetrations rooms and other areas where concentrations of cable exist.

NRC positions regarding pertinent areas outside the CSR are covered in detail under Items C, D, and E which follow.

B. Cable Construction

NELPIA recommendations

1. A re-evaluation of current cable testing requirements should be made to establish the pass-fail criteria for flame propagation of "real-life" cable tray systems.

2. As a minimum requirement today only those cable constructions that will pass the current IEEE 383 flame test should be used.

NOTE.—This does not infer that cables passing this test do not require fire protection nor that certification of cables passing more realistic tests are not essential for tomorrow's cables.

3. Whenever practical, cables that do not liberate copious quantities of corrosive gases should be used particularly in strategic relatively inaccessible and highly susceptible areas.

NRC comments

The NRC staff is accepting replacement cabling purchased to the standards existing at the time of construction. A change in cable materials is not considered to be necessary, by NRC, in view of measurer being taken to reduce the probability of fire through administrative changes and to provide fire detection and extinguishment systems in areas of cabling where a fire in one division could affect redundant safety equipment.

C. Cable Tray Protection

NELPIA recommendations

1. Cable tray systems should be protected by automatic, zoned, open-head, water spray sprinkler systems arranged to discharge directly onto the cables in the trays.
2. An approved fast-acting products-of-combustion type detection system should be provided to actuate the deluge system having sectional control.
3. Adequate floor drains and curbs should be provided to safely remove discharged sprinkler water. Drainage water should be monitored for radioactive materials before being released to the environment. Curbs should be provided around all floor penetrations.
4. Approved noncombustible fire stop constructions should be located in each cable tray and spaced at maximum intervals of 10 ft. horizontally and 10 ft. vertically. NOTE: Cable derating should be given consideration when installing fire stops.
5. Wherever practical, isolate, shield, relocate water damageable equipment.

NRC comments

NRC agrees in principle. Outside of the CSR, NRC has required TVA to meet the above requirements where redundant engineered safety features would be affected by direct or exposure fires. Final details have not yet been resolved and minor deviations could exist. For example, although fire stops will be installed, the spacing may not necessarily agree with the NELPIA recommendations.

D. Indoor Hose Connections

NELPIA recommendations

1. Fire protection equipment including hose, nozzles, standpipe valves, and hydrants should have compatible threads with existing eqeuipment and the local fire department.
2. Combination spray-straight steam nozzles should be provided on each hose connection to effectively combat Class A fires normally inaccessible, e.g., cable tray fires.
3. Standpipe risers should be sealed on each floor to prevent smoke and corrosive gases from penetrating into areas normally unexposed to the effects of fire.

NRC comments

NRC agrees. The above recommendations are to be implemented. In providing Item D2 above, due consideration will be given to personnel and critical equipment safety.

E. Smoke and Heat Removal

NELPIA recommendations

1. Approved smoke and heat venting facilities independent of the station's normal ventilation system should be provided throughout areas having a combustible occupancy. Each system should be actuated by products-of-combustion detectors and arranged to contain the release of radioactive materials.
2. The mechanical exhaust system should be powered from electrical feeders run *outside* the fire area. If inside wiring is necessary, mineral-insulated metal sheathed cable should be used.
3. Additional preventive measures outlined in the "International Guidelines for the Fire Protection of Nuclear Power Plants" should be implemented. Specifically Sections 6.1 (Extraction of Smoke and Heat) and 6.2 (Preventing Corrosion) are applicable.

NCR comments

NRC is still considering the above recommendations. The staff has specifically asked TVA to study means of implementing Items 1 and 2. There is potential here for conflicting requirements between those ventilation features necessary to assure secondary containment integrity and those features which would assist in fighting fires. The staff is considering implementing those measures outlined in "International Guidelines for the Fire Protection of Nuclear Power Plants."

F. Cable Penetration

NELPIA recommendation

1. All wall and floor openings through which electrical cables or conduits penetrate should be protected against the passage of flame and smoke by devices and constructions approved or listed by recognized testing laboratories.

NRC comments

NRC agrees with the above protection requirements. TVA is currently conducting in-house test programs to develop such a seal design and will submit a final report for staff review and approval.

NELPIA recommendations

2. Temporary wall and floor openings should be sufficiently sealed with a noncombustible material at the end of each workday to insure the fire integrity of the wall or floor.

3. Open flames should never be used to check the installation, gas tightness and integrity of penetration seals. Whenever protected openings are examined, fire extinguishers should be immediateley available to those checking for openings on both sides of the wall.

NRC comments

NRC agrees. These items have been corrected by revised plant procedures.

G. *Self-Contained Breathing Apparatus*

NELPIA recommendation

1. Self-contained breathing apparatus approved by the United States Bureau of Mines and described in NEPA No. 19B should be provided for all fire fighting and control room personnel. Preferably, their service or operating life should be one hour.

NRC comments

NRC agrees; however, the service or operating life of one hour is being reviewed.

NELPIA recommendation

2. On-site reserve air supply should be available and arranged to expediently replenish the supply of air in each unit so that the designed service life is available.

NRC comment

NRC agrees. TVA has proposed change in the systems utilized for recharging self-contained breathing units.

H. *Physical Independence of Redundant Circuits*

NELPIA recommendations

1. All redundant Class IE circuits and the equipment served by these circuits should be separated from the primary Class IE circuits by a minimum three-hour fire wall. This will require that a redundant cable spreading room be constructed.

2. Mineral-insulated metal sheathed cable or equivalent fire resistant cables should be used in one of the two Class IE electrical circuits.

NRC comments

NRC does not believe these actions are necessary in light of other actions taken. (Please see responses given for Items A.1, A.2, and B.1, B.2 and B.3.)

I. *Cardox Total Flooding System*

NELPIA recommendations

1. The ventilation system in the CRS should be arranged to shut down whenever the Cardox system is discharged.

2. The Cardox system should be rearranged to operate automatically upon actuation of the ionization detection system. NOTE: A one-minute delay should be incorporated into the system to allow workers ample time to leave.

3. A written procedure and permit system should be adopted that would require employees to obtain written permission to impair fire protection equipment.

4. An acceptance test of the fire protected system, including a complete discharge, should be conducted and witnessed by the installer and authority having jurisdiction.

5. An investigation into the compatability of the ionization detectors with the products-of-combustion generated by the burning cable should be made to insure that the detectors will, in fact, operate during the incipient stages of the fire.

NRC comments

NRC agrees. Item 1 through 3 were proposed by TVA. The staff has required items 4 and 5.

J. Control Room (CR)

NELPIA recommendation

1. All floor openings between the CSR and CR should be sealed airtight with a material that will insure the fire resistance integrity of the floor. Only penetration seals listed by Underwriter's Laboratories or approved by the Factory Mutuals should be considered. Cellular concrete, and inorganic assemblies as described in the "International Guidelines" may also be considered.

NRC comments

NRC agrees with intent to seal floor openings between CSR and CR and will require seals as defined above or equivalent.

NELPIA recommendation

2. Self-contained breathing apparatus approved by the United States Bureau of Mines should be located in the CR to insure an orderly station shutdown and to minimize breathing hazards to personnel. The supply should be sufficient for the number of operators and the time it takes to effect a safe shutdown.

NRC comment

NRC agrees.

K. *Stairwells, Vertical Opening, Mechanical Penetrations*

NELPIA recommendation

1. All stairwells, elevators, chutes and other vertical openings should be enclosed in approved masonry towers with airtight, automatic closing Class B fire doors at each opening into the buiding.

NRC comment

NRC is evaluating per our response given above to items in Category E. This recommendation will be implemented to the extent practical.

NELPIA recommendation

2. All unprotected vertical openings between floors (hoistways, steam pipes, etc.) should be sealed airtight.

NRC comment

NRC agrees with intent. All openings in fire barriers will be sealed airtight.

Senator MONTOYA. Now, in your opinion, were there any actions taken by plant personnel that in retrospect may be classified as having significantly added to or alleviated the potential danger of the situation?

Mr. RUSCHE. Mr. Chairman, I think we have discussed many of these as Dr. Hanauer mentioned a moment ago. There certainly are some individual actions considered in isolation which might have been more optimum. On the other hand, the total effect of the operators was quite effective as I tried to say a moment ago. I think the operators under the circumstances did an excellent job with the information they had at hand. They protected the public health and safety which is their first concern. I think it was an execllent combination.

PUNITIVE SANCTIONS

Senator MONTOYA. Now in your testimony you indicate that the use of punitive sanctions against TVA is not warranted. Without meaning to question your judgment, could you describe the basis on which you reached this conclusion, considering the many deficiencies noted in your investigation report and the serious situation that resulted?

Dr. KNUTH. Yes, sir. That refers to the notice of violation which went to TVA with the release of our investigation report, wherein we did enumerate the items [1] of noncompliance. Based upon the response [2] that we have gotten from TVA—our initial review of it and the corrective action they are taking—as indicated in my testimony at this time, we do not believe the more severe sanctions, such as a civil penalty are warranted in this case. We do not believe that that step is necessary. We do believe that they have taken and are taking corrective action and we do not plan the more severe enforcement actions. We have already taken the first enforcement action.

NRC INSPECTIONS AND APPROVALS

Senator MONTOYA. Now apparently the installation of the cable tray penetration sealant and the fire retardant coating had not been completed prior to the operations of Units 1 and 2. That is reflected in your inspection report.

Now how did this manage to slip past detection by both TVA and the NRC?

Dr. KNUTH. Yes. Of course, I cannot speak to why TVA missed it but as I indicated earlier, during the period of the construction of Browns Ferry Units 1 and 2, we did in our inspections a sampling of areas, where we checked to find whether or not the licensee was meeting his responsibility. In this particular respect, the fire sealing of the penetration was not included in our inspection sample program.

Senator MONTOYA. Why wasn't it?

Dr. KNUTH. In general the penetrations were sealed but there were some exceptions.

Mr. ANDERS. And in this particular case, it involved a rerunning of wire long after the plant had begun operation—thus it was more like a repair.

Senator MONTOYA. Well, when they deviate from the original specifications don't they have to come to you for authorization?

Dr. KNUTH. No, sir.

Senator MONTOYA. For approval?

Dr. KNUTH. No, sir. The regulations—which I believe are 50.59—do, when changes are made in the facility configurations, allow the utility or company to make a safety evaluation of the changes that they are making and it does not require prior approval from the NRC unless it involves a change either in the technical specifications or involves a "significant safety hazard."

Senator MONTOYA. But do they notify you that they are doing it even though it is not required that you give them authorization or approval?

[1] See Appendix 6.
[2] See Appendix 14.

NRC PERMISSION FOR SPECIFICATIONS CHANGES

Mr. ANDERS. Senator, we have not communicated a full answer to the first question. Let me try it.

You asked whether a utility must notify us or get our permission for a change of specifications. In general, for a level of specification which directly relates to public health and safety, the answer is yes. But in this case we are not talking about such a change——

Senator MONTOYA. I was talking about—and in fact I Ispecified and he so stated—under what circumstances there was no authorization. Now you say that they do have to come in for approval.

Mr. ANDERS. I have not finished. We had approved the use of polyurethane foam with the fire retardant material over it for sealing these reactor penetrations. The repair or upgrading of this material, if it is done according to the original specifications—which in hindsight themselves may have been inadequate—would not normally require a utility to come and ask us to approve that particular maintenance action. For some modifications such as a rerouting of wires, it is well understood they are required to do a safety analysis though on their own to insure that at least in their own mind, that this is not a significant——

Senator MONTOYA. Did they do that?

Mr. ANDERS. In my understanding they did not make such an analysis in this case since they did not, at the time, feel it was safety related. We question as to whether that was a good judgment.

Senator MONTOYA. Well, what do you mean you questioned it?

Mr. ANDERS. I understand that they assumed this was not a safety——

Senator MONTOYA. Did you conclude it was bad judgment?

Mr. ANDERS. It was assumed by TVA that this was not a significant safety question; therefore, they did not do the safety analysis in detail.

Senator MONTOYA. Apparently it was.

Mr. ANDERS. In retrospect it certainly looks like it was. Possibly they could have asked us to agree with that conclusion in advance but of course we can't turn back the clock.

Senator MONTOYA. What are you going to do about similar situations in the future? Are you going to leave it to the option of the managing authority?

Mr. RUSCHE. Mr. Chairman, I think the only practical way to operate such a facility is that the managing authority does have to have the responsibility for making those judgments.

Senator MONTOYA. But it has to have monitoring from NRC, does it not?

Mr. RUSCHE. Yes, sir.

Senator MONTOYA. What kind of monitoring will you provide?

Mr. RUSCHE. It is our intent to provide the kind of monitoring that both recognizes the potential for such events and analyzes them.

Mr. ANDERS. It is done on a statistical basis. We do not oversee everything that the utility does.

OPERATION WITHOUT PENETRATION SEALANT

Senator MONTOYA. Was TVA allowed to operate without a penetration sealant?

Dr. KNUTH. Yes, they were allowed to operate. The technical requirement was that the leakage be maintained at a specified value, and there is leakage allowed from one compartment of the building to another as long as they meet the requirement. I believe it was on the order of 7,000 cubic feet per minute or a quarter of an inch of water differential pressure. So, yes, they are allowed to operate with a controlled leakage.

Senator MONTOYA. Does this lend itself to an escape of radiation?

Dr. KNUTH. It is a design consideration that the fans are to be able to maintain a negative pressure so that the leakage would be into this area, where the ventilation system holds the negative pressure, and then up the stack. So the leakage is controlled to a certain level with the ability to hold this negative pressure.

So the answer to the question is yes, they can have leakage up to a specified amount.[1] It is allowed in their license.

Senator MONTOYA. Who is going to determine what the amount is? Do you have monitoring devices?

Dr. KNUTH. Yes. That is in the technical specifications for the plants; yes.

Senator MONTOYA. What is the degree of tolerance?

Dr. KNUTH. It is a limit. Maintain a given pressure. I believe it is a quarter inch of water. They have to maintain at least a quarter inch of water and that is the limit.

Senator MONTOYA. Now in light of the experience, what are you doing now? Are you still allowing operation without the penetration sealant?

Mr. RUSCHE. Mr. Chairman, the specifications that existed before the fire are still in existence—and I will speak to some modifications—would not allow operation without a proper sealant. I think we have recognized already that this connection, which was the subject of the fire, was in the process of being reconstructed and did not have the sealant on it. The answer would be that there should be no operation unless the specifications that are in the license conditions are met.

Mr. ANDERS. But the key factor, Senator, is what should not be allowed. The licensee is not allowed to operate the reactor without the quarter inch of water pressure differential in the secondary containment. Now how you get that condition is left to their judgment. One way is to have openings sealed. Now certainly you are going to have to use sealing material around penetrations. It does not have to be perfect because we do allow some tolerances. But, when the sealant is used it should (1) be adequate and this polyurethane may not be and (2) if polyurethane is used it should be covered with the fireproofing material.

In this particular case where this one cable went through the one hole in a larger sealed area, obviously the polyurethane was not sufficiently covered at that point with the fire retardant material. It was my understanding that the workers' intention was that once they had checked the sealing with the candle, which certainly we all agree is not the way to do it, they would then have covered the polyurethane with the fire retardant material. We then would be up to the design or specifications that TVA proposed and that we had accepted in our licensing review.

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[1] NRC has subsequently advised in response to this question that means are provided to prevent the release of radiation, although there can be inward leakage of air up to a specified amount.

Senator MONTOYA. Was it not apparent that the monitoring capability was inoperable? That is reflected in your Inspection Report on pages I-13 through I-18.

Dr. KNUTH. Are you speaking about the radiological exposures?

Senator MONTOYA. Yes.

Dr. KNUTH. Yes; certain of the monitoring equipment did become inoperable, particularly the stack monitors, during the period of the fire. It was restored that evening. When the equipment did become inoperable, TVA took grab samples of the radioactive material present in the room or going up the stack and took them back to the laboratory and did an analysis so they knew what it was. Yes; there was certain radiation monitoring equipment that did become inoperable because of the fire.

Mr. ANDERS. But there were actions taken by TVA to offset that loss.

Senator MONTOYA. Now what assurances can you give us that there was no radiation exposure in light of the delayed countermeasures in monitoring or detection?

Dr. KNUTH. There are really two aspects of this. There were samples taken of the environment which were analyzed and we know what the radioactive materials were. It was established that the concentrations were less than the maximum permissible concentrations allowed in the technical specifications. After the incident was concluded, the plant health physics staff did determine which of the individuals were fighting the fire, and they actually checked these individuals to find out had they taken up any radioactivity. There was none.

They also had film badges which they monitored. The construction personnel and so forth are required to wear film badges and they were sent off for processing and none showed exposures out of the ordinary.

Senator MONTOYA. When was this equipment rendered inoperable?

Dr. KNUTH. The radiation monitors?

Senator MONTOYA. Yes.

Dr. KNUTH. I believe the stack monitor was out on the order of 9 hours. I believe it was restored to service at about 9 o'clock. The chronology does appear in our testimony.

Senator MONTOYA. It was during the course of the fire? That is what I am trying to establish. It was during the course of the fire?

Dr. KNUTH. Yes.

Senator MONTOYA. All right. Now you mentioned that there were quite a few inspections before this date. Why wasn't the lack of sealant detected through one of those inspections?

Dr. KNUTH. Well, I believe I indicated earlier that at the time of the previous inspections it was not part of our inspection procedures to look for the fire stops in the penetrations.

Senator MONTOYA. Is it now?

Dr. KNUTH. Yes, it is, on a selected basis. Again we do not check 100 percent of them but we do sample.

Senator MONTOYA. We will stand in recess until 2 o'clock this afternoon.

[Whereupon, at 12:10 p.m., the Joint Committee recessed, to reconvene at 2 p.m.]

AFTER RECESS

[The Joint Committee reconvened at 2 p.m., Representative John Young presiding.]

Committee members present: Representatives Young, Price, Horton, Anderson; and Senator Baker.

OPENING REMARKS OF REPRESENTATIVE YOUNG

Representative Young [presiding]. The committee will come to order.

This morning we heard from the chairman, Mr. Anders, and from Dr. Donald Knuth, both of the NRC.

This afternoon we are running a little late. We have first Mr. Benard Rusche and Dr. Stephen Hanauer. In deference to the time and the obvious problem that we have, I am going to suggest that those two witnesses might summarize their statements for us and then without objection we will enter their written statements in the record.

With that then, would you proceed, Mr. Rusche.

STATEMENT OF BENARD C. RUSCHE, DIRECTOR, OFFICE OF NUCLEAR REACTOR REGULATION, NRC

Mr. Rusche. Thank you, Mr. Chairman.

I would like to outline the events both during and immediately following the event and then I will discuss briefly the actions that NRC has taken and plans to take in connection with TVA's recovery and restoration activities and then close with a brief outline of our current thinking with respect to the fire itself.

First let me discuss plant safety during the fire. You recall this morning that we had a considerable amount of discussion in response to questions of this sort so I will move fairly rapidly with your permission.

A detailed discussion of the means by which the TVA operators achieved and maintained a safe shutdown condition of the plants and our analysis of the availability of alternate means that existed for achieving the same goal are described in Attachment I to my written statement. I would like to summarize these results very briefly and note particularly that they apply to Unit 1 and note that in all cases the situation in Unit 2 was more favorable.

About 15 minutes after the fire started, the fission process was stopped by the operator's action to rapidly insert all control rods. In our jargon, such a rod insertion is called a manual scram. The effects of the fire damage on protection circuitry would have subsequently caused an automatic scram had the operator not elected to take action at the time that he did. That is, the rod system was fail-safe. I would like to emphasize that this is a key and important action in converting the time response required from the operators from that of a few minutes to several hours and we had some discussion on that point this morning.

Following shutdown of the fission process, a reactor must continue to be cooled to remove heat produced by radiation fission product decay. Heat removal is required for an extended period of time after plant shutdown in order to prevent fuel damage. Immediately following the shutdown of Unit 1, the reactor coolant system was being maintained at a pressure near that of normal operation—that is, about 1,100 psig—by operation of relief valves. Because of fire damage at this time, there was no automatically available high pressure source of water that was of sufficient capacity to maintain the core covered with water.

In light of this situation, the Browns Ferry operators decided to depressurize the primary coolant system by opening relief valves. As noted in the detailed staff analysis in Attachment I, alternate methods for depressurizing the primary system were available and there was also available manual methods for supplying cooling water at high pressure.

The operators depressurized the reactor by discharging steam from the reactor to the suppression pool through the four relief valves that could be operated from the control room. The discharged steam heated up the water in the suppression pool as intended. Because of the large volume of water in the pool, about 12 hours of such discharge could be accommodated before the pool water would boil, even without the cooling system for the suppression pool which was not available at the time because of fire damage to its control system. Once the reactor pressure was reduced, low pressure pumps were used to provide an adequate source of water for cooling the fuel. Many low pressure pumps capable of supplying more than enough water to keep the core covered were available to the operators at low reactor pressure.

Some hours later, as was brought out in the questioning this morning, the supply of compressed air that actuates the relief valves was lost owing to continuing fire damage experienced in Unit 1 and the valves closed. Since the decay heat in the fuel continued to boil the water in the reactor vessel and the relief valves were closed, the reactor pressure increased again to the point that additional water could not be injected by the low pressure pumps.

At this point two courses of action were pursued. The first involved restoring the air supply to the relief valves so that a low reactor pressure could be reestablished. The second course of action pursued was to establish a depressurization path through the main steam line drain with water injection from control rod drive pump (Unit 1) and from control rod drive pump (Unit 2).

As described in Attachment I, the depressurization path through the drain line was established in about 1 hour. Analyses show that a satisfactory cooling condition was being achieved at that time. In about 3½ hours the operators restored the air supply to the relief valves and again reduced reactor pressure sufficiently to permit operation of the low pressure pumps again. From this point on the pressure was maintained low enough for the low pressure pumps to be effective in injecting water.

I would like to confirm Chairman Anders' former statement that the system was quite effective in protecting public health and safety. The core cooling was maintained with multiple systems available. I might, if you would permit me to interject, say in response to Senator Montoya's question this morning we do have an early estimate of the likelihood of such an event as estimated by our Rasmussen Study

Group and I would like to have that provided now or at some later time [1] if you prefer, sir.

Representative YOUNG. We will just have it later. Go ahead.

Mr. RUSCHE. Thank you, sir.

NRC ACTIVITIES FOLLOWING THE FIRE

Let me now describe our activities following the fire. The actions of the NRC during and immediately following the fire were directed toward determining the exact status of Units 1 and 2 and verifying that both units were in a safe and stable configuration.

Subsequently, the objectives of the Office of Nuclear Reactor Regulation have been (1) to assure that a safe plant configuration was maintained; (2) to assure safety during removal of fuel from Units 1 and 2; (3) to assure plant safety during removal and restoration of fire damaged cables and equipment; and (4) to determine that the restoration and associated design changes proposed by TVA are acceptable.

Thus far three licensing actions have been taken with respect to these objectives. The first established the conditions for maintaining the reactor in a safe condition following the fire. The second permitted removal of the fuel from both reactors and removal of damaged equipment. The third permitted certain restoration activities such as installing new cables and cable trays. Copies of these safety evaluations are provided for the record as Attachments II, III, and IV to my written testimony. These documents have been made available to the public already.

Let me now turn to summarize the future actions that we see before us as we continue to progress with Browns Ferry. The three major objectives of the TVA restoration program described in the NRC Safety Evaluation issued on September 2, 1975—Attachment IV—are:

1. To improve administrative actions that can prevent a fire from occurring;
2. To use separation of electrical cables, with physical barriers, as a mechanism by which to prevent a fire from damaging redundant safety equipment; and
3. To incorporate means to detect and extinguish a fire quickly.

The activities that remain to be accomplished by NRC to meet these objectives before the units can be returned to an operational status are, and I summarize again very briefly, sir:

1. Complete the review and approval of the total plant fire detection and protection system design changes—and we discussed some of those this morning.
2. Review and approve changes to procedures and administrative controls relating to operation, construction and repair, communications, and emergency planning—again a subject of extensive discussion this morning.
3. Review and evaluate plant ventilation systems as they relate to isolation and smoke control.
4. Review and approve new fire resistant materials to be used in fire stops and seals.

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[1] See page 94.

5. Review and evaluate the surveillance program for the long-range monitoring of the effects of chloride contamination of equipment and materials. I am sure you will recognize that the chloride contamination arose from the combustion products of the cable covers and insulation.

6. Review and evaluate the preoperational testing program for those systems and components modified or replaced as a result of the fire.

7. Prepare and issue a final Safety Evaluation Report and technical specifications upon completion of our review and evaluation of the six items described above.

Our objective is to complete these necessary NRC review and approval tasks on a schedule that is consistent with TVA plans to resume operation of Units 1 and 2, about January 1976.

Now, sir, if I may turn to the implication of the fire on other plants. The fire at the Browns Ferry Nuclear Plant raises the question as to what additional actions are warranted at other nuclear power stations to avoid or withstand the effects of a fire. Dr. Knuth noted in his remarks that after the fire, NRC issued bulletins to each of the other operating nuclear powerplants notifying them of the specific circumstances associated with the fire and requesting each of the licensees to consider policies and procedures related to various construction activities, control of flammable materials, and emergency actions that might be required following a fire and to examine equipment provided to cope with the effects of a fire.

In addition to eliminating deficiencies discovered as a result of these reviews, other actions taken by licensees include the acquisition of additional firefighting equipment; evaluations of the feasibility of installing fixed spray systems and modifications of existing administrative procedures.

The Office of Nuclear Reactor Regulation has initiated an evaluation of the implications on other plants of the Browns Ferry experience as well as fires previously reported at other nuclear facilities. Attachment V of my written testimony describes all the data that we have, in summary form, of fires that have occurred at other nuclear facilities. Dr. Hanauer will describe the activities of the NRC Special Review Group which will also be making recommendations in this regard.

Based on what we have learned to date, we expect that some improvements in operating plants will be needed but that there is no need to suspend and restrict their operation immediately. For these plants, as well as for plants in an advanced state of construction, we will consider measures that can be taken to improve the existing designs with respect to fire prevention, separation of redundant equipment, and fire fighting.

Powerplant designs now in the preliminary stages and designs submitted in the future may also be subjected to new requirements. For these new designs our emphasis will be to achieve as much separation and isolation of redundant safety electrical cabling as is practicable.

In summary I wish to emphasize that the public health and safety was not affected as a result of the Browns Ferry fire, although the fire damage to the facility was more severe than we would have expected. No radioactive release above normal operating levels was experienced. In spite of the fire damage, there was considerable remaining equipment available to keep the reactors in a cooled and safe configuration.

Mr. Chairman, this concludes my very brief summary and with your permission I would like to introduce for the record the detailed material as you had previously granted. I would also like, in conclusion, to compliment the TVA staff for their diligence and cooperation in the intensive effort that we have had underway as they have attempted to make plans and preparations to return the plant to service in an acceptably safe condition.

Thank you, Mr. Chairman.

Representative YOUNG. Thank you, Mr. Rusche. We appreciate your statement, and without objection the material will be made a part of the record at this point.

[The prepared statement follows:]

STATEMENT OF BENARD C. RUSCHE, DIRECTOR, OFFICE OF NUCLEAR REACTOR REGULATION, U.S. NUCLEAR REGULATORY COMMISSION

INTRODUCTION

Mr. Chairman and members of the Committee. The first part of my presentation will be concerned with the effects of the Browns Ferry fire on the safety of the facility during and immediately following the event. Then I will outline the actions NRC has taken and plans to take in connection with TVA's recovery and restoration activities. I will close with a brief outline of our current thinking with respect to the implications of this incident on other plants.

PLANT SAFETY DURING THE FIRE

Both Units 1 and 2 of the Browns Ferry Station were operating near full power at the time of the fire. Although the fire was localized it affected the operability of some major equipment at each unit. Unit 2 was shut down and was capable of being maintained in a safe shutdown condition using normal cooling systems. In the case of Unit 1, because of the significant losses of control of important equipment, the operating staff made use of backup equipment to maintain adequate cooling following the fire.

When a nuclear power plant is faced with a situation such as the Browns Ferry fire, there are two major concerns. One is to shut off the power production from the fission process and the other is to remove the nuclear heat to assure that the reactor fuel remains cooled at all times. A detailed discussion of the means by which the TVA operators achieved and maintained a safe shutdown condition and our analysis of the availability of alternate means that existed for achieving the same goal, which is a measure of the margin of safety that existed, are presented in Attachment I to my statement. I will only summarize the results at this time. These results apply to Unit 1. In all cases the situation in Unit 2 was more favorable.

About fifteen minutes after the fire started, the fission process was stopped by the operator's action to rapidly insert all control rods. In our jargon, such a rod insertion is called a "manual scram". After detailed examination of the reactor protection system design, we have confirmed that had the rods not been inserted manually, the effects of the fire damage on protection circuitry would have caused an automatic scram. That is, the rod system was fail safe.

Following shutdown of the fission process, a reactor must continue to be cooled to remove heat produced by radiation fission product decay. During the first few hours after shutdown, the decay heat level can be in the range of 2% to 3% of the heat output at full power, decreasing to about 1% after one day and declining very slowly after that. Heat removal is required for an extended period of time after plant shutdown in order to prevent fuel damage. Knowing this, and faced with the loss of much equipment, the Browns Ferry plant operators did a commendable job of providing adequate cooling water so that the reactor core remained covered with water at all times.

Immediately following the shutdown of Unit 1, the reactor coolant system was being maintained at a pressure near that of normal operation (i.e., about 1100 psig) by operation of relief valves. Because of fire damage at this time, there was no automatically available high pressure source of water that was of sufficient capacity to maintain the core covered with water.

In light of this situation, the operators decided to depressurize the primary coolant system by opening relief valves. The system normally used for shutdown cooling are shown on Figure 1. As noted in the detailed staff analysis in Attachment I, alternate methods for depressurizing the primary system were available and there also were available manual methods for supplying cooling water at high pressure.

Following their selected course, the operators depressurized the reactor by discharging steam from the reactor to the suppression pool through the four relief valves that could be operated from the control room (Figure 2). This discharge heated up the water in the suppression pool as intended. Because of the large volume of water in the pool, about 12 hours of such discharge could be accommodated before the pool water would boil, even though the cooling system for the suppression pool was not available at the time. Even if the pool water had boiled, the resultant steam could have been vented to the atmosphere. Once the reactor pressure was reduced, a low pressure condensate pump was used in conjunction with a condensate booster pump to provide an adequate source of water for cooling the fuel. A total of three of these condensate booster pumps and three condensate pumps were available to the operator at this time. Any of the condensate pumps would have been capable of supplying more than enough water to keep the core covered (Figure 3). Also, the Unit 1 control rod drive pump supplied some water to the reactor throughout reactor cooldown.

Some hours later, the supply of compressed air that actuates the relief valves was lost owing to continuing fire damage experienced in Unit 1 and the valves closed. Since the decay heat in the fuel continued to boil the water and since the relief valves were closed, the reactor pressure increased again to the point (about 600 psig) that additional water could not be injected by the low pressure pumps. In about 3½ hours the operators restored the air supply to the relief valves and reduced reactor pressure sufficiently to permit operation of the low pressure pumps again. From this point on, a stable condition was maintained. As noted in the staff analysis in Attachment 1, even if it had not been possible to restore the availability of the relief valves, operation of the control rod drive pump in conjunction with pressure reduction effected by opening of the main steam line drain was providing sufficient water to keep the core covered throughout the remainder of the incident.[1]

The sequence of events I have just described (Figure 4) resulted from choices the operators made during the incident. In looking back on the event, it is now clear that a number of other choices could have been made with equal success. I can confidently state, based on our study of the incident, that not only did the operators pursue a logical and effective course of action under difficult circumstances, but that alternate methods were always available for cooling the core, although some were not available at the touch of a switch. Some of these methods would have required adaptation of equipment for these functions by the operators. These methods are described in Attachment I to my statement.

A nuclear plant of this type has a significant amount of redundancy and flexibility in various modes of operation, and even in the event of a fire that caused loss of capability for automatic actuation of the emergency cooling equipment, there remained many alternate means for maintaining a safe shutdown condition. At no time during the event was there either immediate danger of damage to the fuel, or danger to the public health and safety. Nevertheless, the Browns Ferry fire was a serious event. It is appropriate that we take positive steps to prevent the occurrence of another fire of similar magnitude in a nuclear power plant.

NRR ACTIVITIES FOLLOWING THE BROWNS FERRY FIRE

The actions of the NRC during and immediately following the fire were directed toward determining the exact status of Units 1 and 2, and verifying that both units were in a safe and stable configuration.

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[1] Attachments included with Mr. Rusche's statement appear in this volume as appendix 15.

Subsequently, the objectives of the Office of Nuclear Reactor Regulation have been (1) to assure that a safe plant configuration was maintained; (2) to assure safety during removal of fuel from Units 1 and 2; (3) to assure plant safety during removal and restoration of fire damaged cables and equipment; and (4), to determine that the restoration and associated design changes proposed by TVA are acceptable. Thus far, three licensing actions have been taken with respect to these objectives. I will briefly describe each of these actions. As each action was approved, the staff's safety evaluation has been made available to the public. Copies of these safety evaluations are provided for the record as Attachments II, III, and IV.

Plant safety after the fire

Following the fire, Units 1 and 2 were maintained in a safe and stable condition by: (1) reconnecting power and control to some systems by routing cables outside of the fire damaged zone; (2) verifying that the original cables for systems being used were outside of the fire zone and therefore were unaffected by the fire; (3) converting some systems and components to manual control, and; (4) placing some unneeded systems and components in a desired safe configuration and then disconnecting power leads to prevent the possibility of spurious operation.

Our actions were directed toward verifying the availability and reliability of all vital systems, including the backup systems for providing required safety margins. Our technical specialists visited the site for visual observation of operating systems, audits of engineering work and testing performed, review of procedures developed for operation of systems in their post-fire configurations, and the observation of a test to demonstrate the availability of onsite standby power. During the visits, we ascertained that both cores were subcritical and that there were no equipment malfunctions or single operator errors that could cause them to be made critical.

Adequate core cooling was provided by the operation of one residual heat removal (RHR) system. Additional systems utilizing the suppression pool as an intermediate heat sink were also available for core cooling. Analyses were performed to demonstrate that in excess of 15 hours would be available to restore core cooling if for some unforeseen reason all cooling was interrupted. Backup cooling systems could be placed in operation manually in less than one hour. It was also established that the electrical energy to operate the systems required to maintain adequate cooling of the core could be provided by redundant offsite and onsite power supplies.

Finally, the Technical Specification requirements in the license were amended to reflect the changes that were necessary to account for the post-fire condition of the plant. Certain additional controls and equipment requirements not included in the pre-fire Technical Specifications were added to provide additional assurance that the plant would be maintained in a safe and stable shutdown condition during the activities associated with preparation for the restoration program.

Plant safety during fuel storage and damage removal

After discussions with the staff, TVA proposed to remove the fuel from the reactor vessels in Units 1 and 2 and place it in the respective fuel storage pools prior to start of damage removal and restoration operations. This had the effect of (1) virtually eliminating any potential for inadvertent criticality; (2) providing a substantially greater coolant inventory than in the vessels; and, (3) reducing the number of systems required to maintain the fuel in a safe condition.

We reviewed the proposed plan for transferring and storing the fuel and, as summarized in Attachment III, concluded that these actions were acceptable.

We also reviewed the criteria and procedures proposed by TVA governing cable cutting and damage removal operations. These included criteria for identification of vital systems, identification of fire damaged cable, and measures to be taken to preclude spurious operation of critical components. We concluded that the measures and procedures governing cable cutting and fire damage removal operations provided adequate assurance that these operations could be performed without undue risk to the health and safety of the public.

On June 13, 1975, we issued additional changes to the Technical Specifications and a supporting Safety Evaluation concerning fuel removal and storage and removal of damaged electrical equipment. These documents are provided for the record as Attachment III to my statement.

Plant design changes

The licensing actions I have just described limited the restoration work carried out at the Browns Ferry facility to various cleanup operations and removal of damaged electrical wiring and cable trays. The next step involved our review and approval of some proposed design features for the restoration of structures and equipment damaged by the fire, and the preparation of the facility for returning to power.

The three major objectives of the TVA restoration program for the Browns Ferry Units described in the NRC Safety Evaluation issued on September 2, 1975 (Attachment IV) are:

1. To improve administrative actions that can prevent a fire from occurring;
2. To use separation of electrical cables, with physical barriers, as a mechanism to prevent a fire from damaging redundant safety equipment; and,
3. To incorporate means to detect and extinguish a fire quickly.

Our approval authorized design changes and modifications in two major categories:

1. Limited approval of proposed electrical design changes, including cable rerouting and further separation of divisional cables by fire barriers; and,
2. Limited approval of the proposed additional fire detection and fire extinguishing systems.

The approval of restoration of structures and equipment damaged by the fire includes:

1. Approval to proceed with structural work including restoration of cable tray supports, pipe supports, process piping, and supports of affected mechanical equipment;
2. Approval to replace damaged cabling; and,
3. Approval of the program for cleaning and testing components affected by combustion products from the fire.

Before discussing the design changes to the Browns Ferry facility, a brief understanding of the basic design concept would be helpful. The electrical design is based on a two-division concept. The fundamental design objective of the safety systems is to insure that no single credible failure or event can result in the loss of the safety function. To accomplish this, designers have provided redundant sets of equipment. These redundant sets of equipment are called "divisions." The purpose of such a concept is to assure that all safety functions would be available even with the loss of one of the two electrical divisions. To achieve this, fire induced failures must be limited to one of the electrical divisions of the safety-related equipment needed for reactor shutdown. To assure that a fire would not cause a loss of both electrical divisions, some changes in the design and equipment layout at the Browns Ferry units have been determined to be necessary.

As a practical alternative to complete physical independence of the two electrical divisions, TVA has proposed an overall program embracing several elements that achieve the same objective. The essential changes involved (1) either increasing physical separation or installing fire barriers between divisions to insure that a fire in one division would not damage the second division prior to extinguishment of the fire, and (2) providing means to promptly detect and extinguish the fire.

The design changes and improvements in circuit separation and resulting improved fire protection, along with design improvements in fire detection systems and fire extinguishing systems proposed by TVA, will substantially enhance the capability of the facility to withstand fires. Our review of the safety of the restored facility is still in progress and, prior to return to operation, additional changes may be required as a result of our continuing review, or as a result of the additional studies that we have indicated are needed and which TVA plans to conduct.

Our safety evaluation pertaining to these actions is provided for the record as Attachment IV.

FUTURE ACTIONS FOR BROWNS FERRY

The efforts remaining to be accomplished to restore Browns Ferry Units 1 and 2 to operational status are as follows:

1. Complete the review and approval of the total plant fire detection and protection system design changes and establish schedules for their installation. The major design changes relate to: (*a*) The installation of an automatically-

actuated fixed spray system which was initially proposed by TVA to be manually actuated. Areas to be protected include selected cable tray runs, and certain cable spreading room penetrations; (*b*) The development of designs, before startup, of additional systems required to extend the fixed water spray in the facility; (*c*) The installation of automatic capability to the already installed manually actuated CO_2 system in the cable spreading room; (*d*) The investigation, before startup, of the possibility for the use of additional fire barriers to the extent reasonable and practicable.

2. Review and approve changes to procedures and administrative controls relating to operation, construction and repair, communications and emergency planning. The events that took place during and subsequent to the fire identified weaknesses in these procedures and controls. The NRC Inspection and Enforcement report discusses these weaknesses in detail. The objective here is to assure that the new and revised procedures and controls adequately correct these weaknesses.

3. Review and evaluate plant ventilation systems as they relate to isolation and smoke control. The ventilation system in the secondary containment building is designed to isolate the building upon an accident signal to limit any radioactivity releases. Experience gained from the fire demonstrated a need to remove the dense smoke that was present. This places conflicting requirements on the ventilation systems since for a release of radioactivity the system should fail closed and for a fire without such a release the system should fail open. TVA will investigate the possibility of modifying the ventilation system to accommodate these diverse functions.

4. Review and approve new fire resistant materials to be used in fire stops and seals. TVA is now conducting and will soon complete a testing program for selecting a new sealant material for the penetrations. The staff has required that a more suitable sealant be found and used in all pentrations that were damaged by the fire, that will be breached due to the restoration effort, and those yet to be made in Unit 3.

5. Review and evaluate the surveillance program for the long range monitoring of the effects of chloride contamination of equipment and materials. All of the equipment and components exposed to chloride contamination resulting from the soot and smoke from the burning cable have been cleaned. Surveys have been performed to verify that acceptably low levels have been achieved. Nonetheless, it is necessary to monitor these items for prompt detection of any corrosion damage.

6. Review and evaluate the preoperational testing program for those systems and components modified or replaced as a result of the fire. It is necessary to assure that those systems and components that were modified or replaced as a result of the fire are capable of functioning as designed. Also, the operability of new systems, such as the water spray system and fire detection system, must be demonstrated. TVA has submitted a general description of the procedures which will be used for preoperational testing of these systems and components. The staff has requested and received additional information on these tests and is in the process of reviewing and evaluating these test procedures.

7. Prepare and issue a final Safety Evaluation Report and Technical Specifications, upon completion of our review and evaluation of the six items described above. The report will: (*a*) Summarize the events which led up to and occurred after the fire; (*b*) Describe and evaluate the plant modifications which have been made; (*c*) Describe and evaluate the changes in procedures and administrative controls which have been made; and, (*d*) Prescribe operating limitations and requirements on the operation of Browns Ferry in the form of new Technical Specifications.

Our objective is to complete these necessary NRC review and approval tasks on a schedule that is consistent with TVA plans to resume operation of Units 1 and 2 about January 1976.

THE IMPLICATION OF THE BROWNS FERRY FIRE ON OTHER PLANTS

The fire at the Browns Ferry Nuclear Plant raises the question as to what additional actions are appropriate regarding the capability to withstand the effects of a fire at other nuclear power stations. Dr. Knuth noted in his remarks that after the fire, NRC issued bulletins to each of the 51 other operating nuclear powerplants notifying them of the specific circumstances associated with the fire, and requesting each of the licensees to consider policies and procedures related to various construction activities, control of flammable materials, and emergency actions that might be required following a fire, and to examine equipment provided to cope with the effects of a fire.

In addition to eliminating deficiencies discovered as a result of these reviews, licensees are taking a number of other actions that will reduce both the potential for, and consequences of, a fire. The actions include instances in which additional firefighting equipment is being ordered, studies are being performed to evaluate the feasibility of installing fixed spray systems, and modifications are being made to existing procedures to reduce the potential for a fire and provide more effective procedures for coping with the effects of a fire.

The Office of Nuclear Reactor Regulation has initiated an evaluation of the implications on other plants of the Browns Ferry experience, as well as fires reported at other nuclear facilities (Attachment V describes other significant fires). Dr. Hanauer will describe the activities of the NRC Special Review Group which will also be making recommendations in this regard. Although the NRR evaluation is not complete and the study group's recommendations are not yet completed, based on what we have learned to date, we expect that some improvements in operating plants will be needed. For these operating plants, as well as for plants in an advanced state of construction, we will consider measures that can be taken to enhance the existing designs with respect to (1) fire prevention; (2) separation of redundant equipment; and (3) firefighting. For example, the installation of additional barriers to prevent the spread of a fire from one portion of the electrical system to another will be considered, and improvements in the design of penetrations and fire stops will also be evaluated. Improvements in spray systems, hose connections, and portable fire extinguishing equipment also will be considered. The need for installing additional fire detection equipment will also be evaluated. Improvements in administrative controls and revisions to procedures are expected to be needed in some of these plants. As noted previously, many of these actions are already being taken by licensees as the result of our bulletins.

Powerplant designs now in the preliminary stages and designs submitted in the future, may also be subjected to new requirements. It is premature, at this stage of our studies, to attempt to specify in detail anticipated additional requirements. For these new designs our emphasis will be to achieve as much separation and isolation of redundant safety electric cabling as is practicable. More stringent standards for material specifications and qualifications are expected to be developed. The emphasis on firefighting will probably result in improved fire control systems within the new plants.

In summary, I wish to emphasize that the public health and safety was not affected as a result of the Browns Ferry fire, although the fire damage to the facility was more severe than we would have expected.

No radioactivity above normal operating levels was experienced. In spite of the fire damage, there was considerable equipment available to keep the reactors in a cooled and safe configuration.

Thank you Mr. Chairman. This concludes my prepared presentation but I would like to introduce for the record the detailed safety analyses and evaluations, prepared by the staff, which have formed the basis for my summary remarks. I would also, in conclusion, like to compliment the TVA staff for their diligence and cooperation in the intensive effort I have described to return the plants to service in an acceptably safe condition.

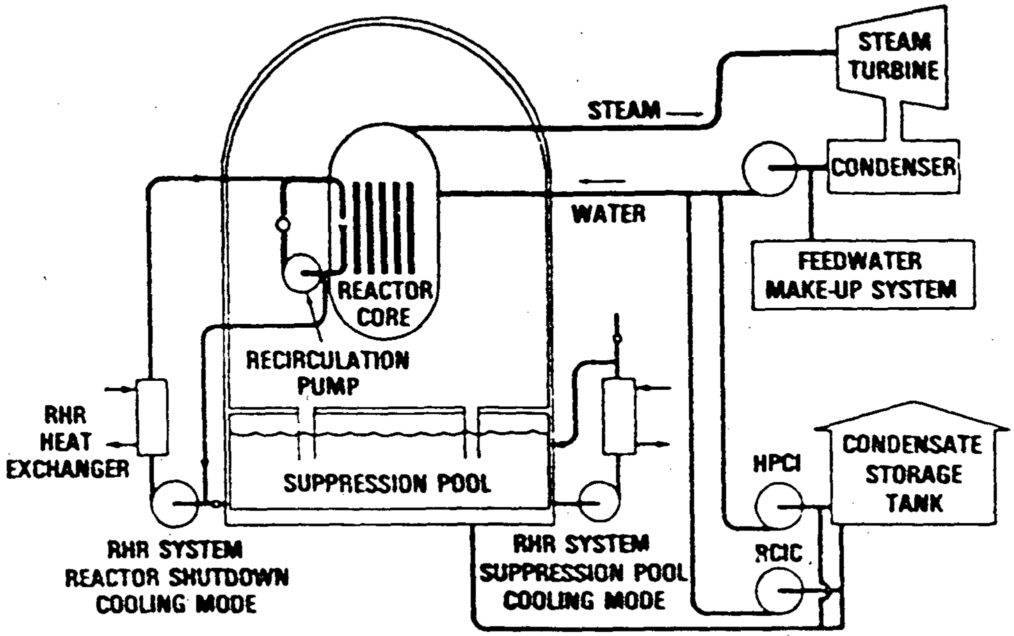

FIGURE 1.—Systems available for normal cooldown.

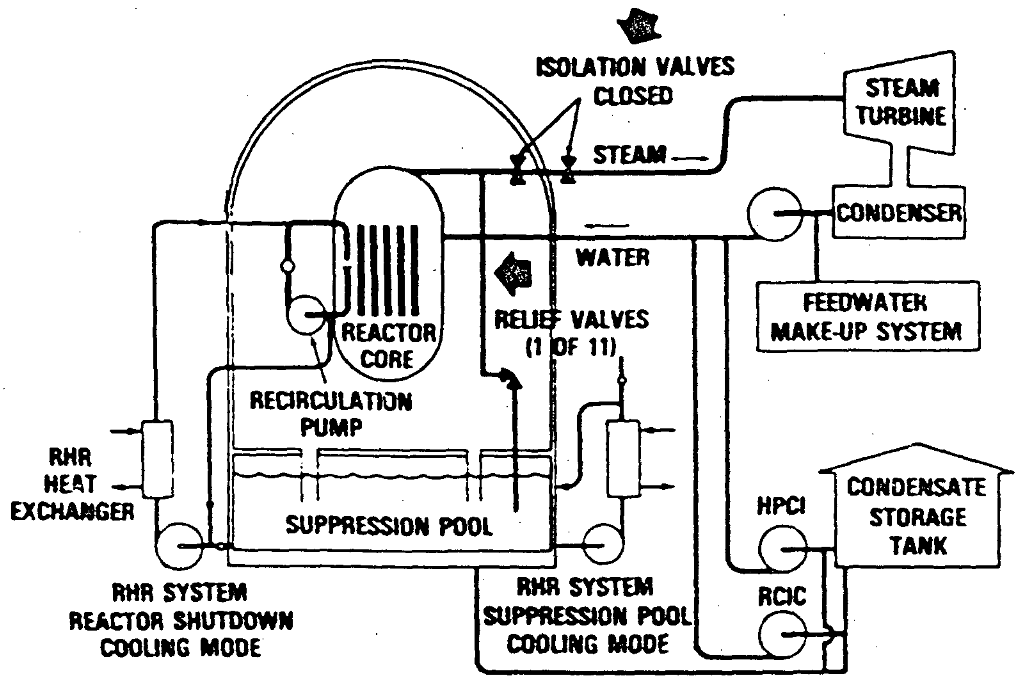

FIGURE 2.—Steam path to suppression pool via relief valves.

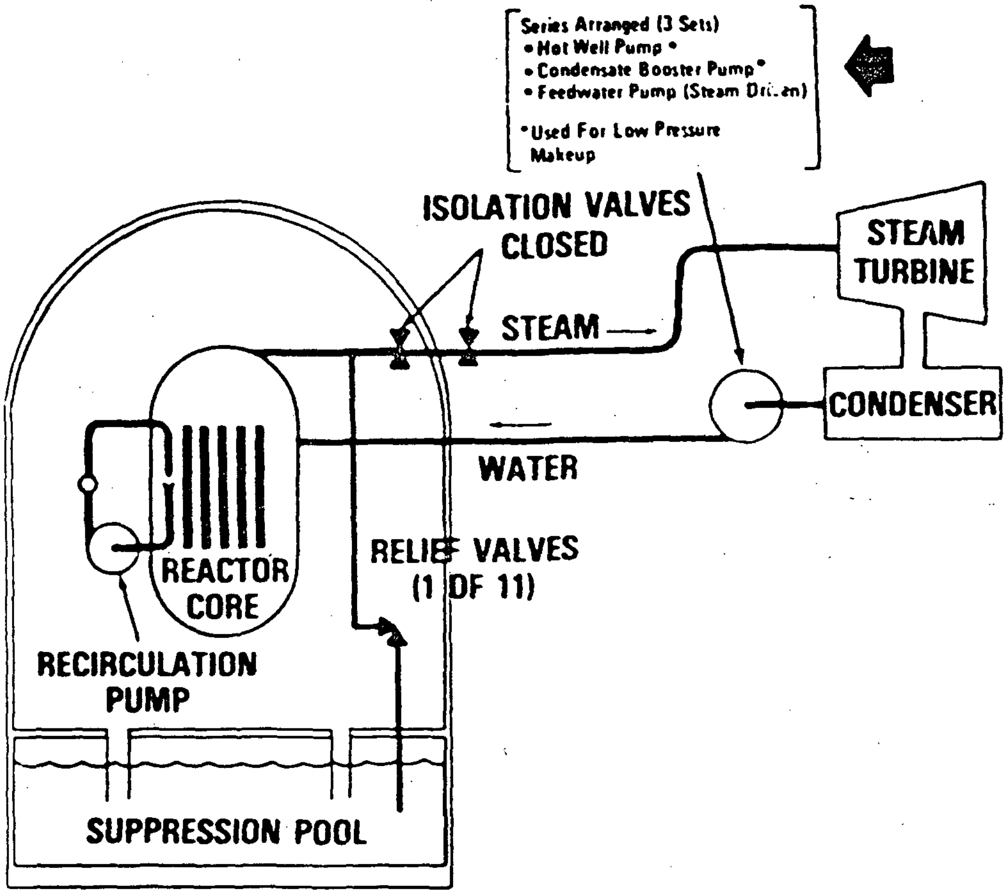

FIGURE 3.—Pumps used for low pressure water makeup.

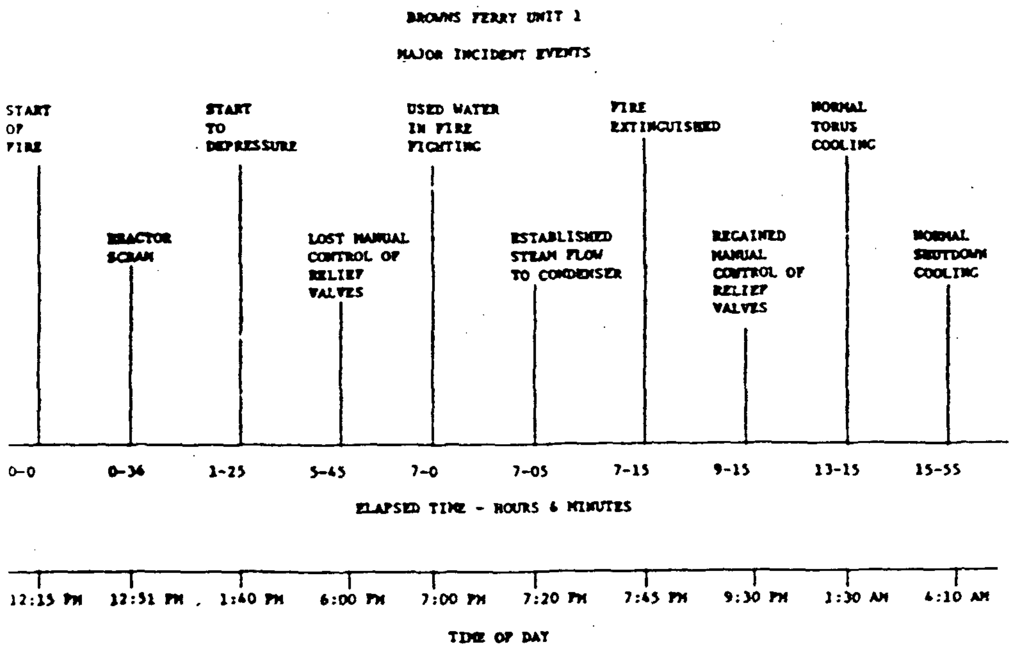

FIGURE 4

[Attachments included with Mr. Rusche's statement appear as appendix 15.]

Representative YOUNG. I have a whole series of questions that have been prepared here. However, I am only going to touch on two of them, Mr. Rusche.

One, you have made an earlier statement that the scram, had it not been effected manually, would have been caused by the failure of the circuitry automatically.

Mr. RUSCHE. Yes, sir.

Representative YOUNG. Mr. Rusche, is there any situation where that automatic scram would be negated?

Mr. RUSCHE. In our analysis, sir, we have found no circumstances in which that effect would not have occurred as a result of fire damage to the cables which occurred later on in the sequence. We have been unable to identify such circumstances, and therefore the basis for my statement that the system functions and appears to have been designed correctly fail-safe.

CABLE TRAYS INVOLVED IN PREVIOUS FIRES

Representative YOUNG. Mr. Rusche, further it appears that there have been quite a few previous fires involving nuclear powerplants, and nine of them involved cable trays. I would think that this would point up what we have discussed in some detail this morning with regard to—I think Mr. Anderson was bearing down pretty heavily on the need for routine practice sessions in this connection, and I am going to concur with him completely, too, sort of a Navy fashion fire drill, that type of thing.

Mr. RUSCHE. Yes, sir.

Representative YOUNG. I take it that you will be planning such procedures for these plants?

Mr. RUSCHE. That is correct. In our review of the administrative procedures that TVA has proposed and will be continuing to develop, this will be a key factor in our evaluation, and I can assure you it will receive our first attention.

Representative YOUNG. We went on at length this morning, and I don't mean to cover old ground or old trails, but all this business about when you can put water on an electric fire, it seems to me that it basically is a question of when can you turn off the electricity.

There are problems, too, with regard to these safety features.

Mr. RUSCHE. Yes.

Representative YOUNG. I have no further questions.

Does anybody else have any questions?

Representative PRICE. Only one question.

Was it unusual for the delay in notifying NRC that there was a fire? Was that anything unusual, or is there some regulation that these notifications are supposed to be immediate?

Mr. RUSCHE. Mr. Price, if you permit me, sir, I will ask Dr. Knuth of our staff to answer that question.

Dr. KNUTH. The time for notification is set forth to be immediately, but in no case longer than 24 hours after an event. In this case, of course, they notified us about 4 hours after the initiation of the fire, so, perhaps again in hindsight, we would like to have known sooner, but on the other hand, they were busy alerting other members and taking the emergency action, and so we do not have any particular quarrel with the timeliness of alerting the NRC office.

Representative YOUNG. Mr. Anderson.

Representative ANDERSON. I get the impression overall from what I have heard that carbon dioxide extinguishers were available, and there was other equipment available for fighting fire in the spreader room, but that when the fire then spread some 40 feet into the reactor room, that there was less equipment available for that purpose.

Are steps being taken to remedy that, if in fact what I have described was a correct description of the situation?

Mr. RUSCHE. Yes, sir, your description is correct to the best of my knowledge. There is firefighting equipment there. In our evaluation we have reviewed the TVA proposal to install extensive coverage for water-sprinkler systems initially to be installed manually with design work presently underway to see if this equipment might even be made automatic.

In any area where there is a question about the redundancy and the effectiveness of separation measures that might be taken, we will insist that there be readily available installed firefighting equipment such as water.

Representative ANDERSON. Thank you, Mr. Young.

Mr. RUSCHE. I might add that there was a water hose there for manual fighting of the fire.

Representative ANDERSON. But not any automatic system?

Mr. RUSCHE. No; no distribution system. That will be a feature of the redesign.

Representative YOUNG. Thank you, Mr. Rusche.

We will now hear from Dr. Stephen H. Hanauer, Technical Advisor to Executive Director for Operations, NRC.

Dr. Hanauer.

STATEMENT OF DR. STEPHEN H. HANAUER, TECHNICAL ADVISOR TO THE EXECUTIVE DIRECTOR FOR OPERATIONS, AND CHAIRMAN, SPECIAL REVIEW GROUP, U.S. NUCLEAR REGULATORY COMMISSION

Dr. HANAUER. Mr. Chairman, I have a summary of my written testimony.

Representative YOUNG. Thank you, Dr. Hanauer. Please proceed.

Dr. HANAUER. The Special Review Group was established by the Executive Director for Operations soon after the fire to identify the lessons learned from this event and to make recommendations for the future in the light of these lessons. The members of the review group are NRC employees with expertise in the various technologies involved. Group members and fire experts from NASA and insurance underwriters have visited the plant and the licensee's engineering offices to obtain information for the group's evaluation.

I might interpolate that we have had two meetings with the fire underwriters whose report was referenced in this morning's discussion.

Our review is still underway. At the present time, we have completed most of the technical evaluation of the causes of the failures experienced. We are reviewing the response of the licensee's organization and the outside organizations from which TVA needed help, and also how outside organizations involved in the Browns Ferry emergency planning responded to this incident. In addition, we are reviewing the adequacy of NRC criteria and procedures and the performance of our agency in the review and inspection of nuclear powerplants. The response of the NRC during and after the fire is also being examined. We are in the process of consolidating the lessons learned and formulating our recommendations. We expect to complete our work within the next month or two.

The Browns Ferry fire and its aftermath have shown up serious inadequacies. In addition to the direct fire damage, there were several kinds of failures. Some equipment did not function correctly, and, in hindsight, some people's actions were incorrect or at least not as productive as they should have been—another subject we discussed this morning.

There is another way of looking at the lessons of the Browns Ferry fire. The outcome with regard to the protection of public health and safety was successful.

The question naturally arises: How can a serious fire that involved inoperability of so many important systems result in no adverse effect on the public health and safety? The answer is to be found in the defense-in-depth approach used to provide safety in our nuclear powerplants today. It provides for achieving the required high degree of safety assurance by echelons of safety systems. The defense-in-depth afforded in this way does not depend on the achievement of perfection in any single system or component, but the overall safety is hig...

Let us now apply this perspective to fires in nuclear powerplants. With respect to fire the defense-in-depth principle is aimed at achieving an adequate balance in:

1. Preventing fires wherever you can.
2. Fighting those fires that occur, putting them out quickly and limiting their damage.
3. Designing the system so that a fire that gets started in spite of the fire prevention program and burns for a considerable time in spite of firefighting will not prevent vital functions from being performed.

No one of these echelons can be perfect or complete. Strengthening any one can compensate in some measure for weaknesses, known or unknown, in the others.

The lessons of Browns Ferry show that all three lines of defense had gaps, and yet the outcome of the Browns Ferry fire shows that the overall defense-in-depth was adequate to protect the public health and safety.

We turn now to the discussion of the gaps in the lines of defense—the failures that occurred at Browns Ferry. They included weaknesses in equipment design, operating procedures and quality control. The NRC must take its share of the responsibility for these failures. Browns Ferry was licensed after review of the proposed design and operation. So the lessons are for NRC as well as the industry.

Although our review is not complete, we believe that both fire prevention and firefighting should be improved to decrease the probability of fire getting started or, having gotten started, burning for a long time. We believe that simple and effective measures are available to effect this improvement.

With regard to the third element of defense-in-depth—providing safety functions in spite of postulated fires—our review is not as far along as it is in other areas.

We believe that the fire at Browns Ferry has revealed that some improvements in existing plants are prudent. We expect our recommendations to call for a reevaluation of each line of defense against fire in each existing nuclear plant. This reevaluation has already begun. Immediately after the Browns Ferry fire, the NRC issued a bulletin to all licensees to review their maintenance and construction procedures, their fire stop designs and their fire protection equipment and procedures. Licensees should now be more alert to the possibilities of fire and better prepared to fight fires. Due to this greater awareness of the possible causes and consequences of fires, prevention should thus be improved.

As interim measures, emergency shutdown procedures, firefighting procedures, and control of combustible material and ignition sources are being improved. For the longer term, this should be supplemented by those improvements in each line of defense which are revealed by a reevaluation of each plant and determined to be appropriate.

We have concluded that the Browns Ferry fire has not shown that present plants are unsafe; that is, while fires can cause severe damage to plant equipment, the health and safety of the public are adequately protected. However, subject to our further study and final recommendations, the review group believes that improvements may be required in fire prevention in firefighting, and in limiting the consequences of fires in nuclear powerplants.

Representative YOUNG. Dr. Hanauer, thank you very much for your statement. We will include it in the record at this point.

Dr. HANAUER. Thank you.

[Prepared statement follows:]

STATEMENT OF DR. S. H. HANAUER, TECHNICAL ADVISOR TO THE EXECUTIVE DIRECTOR FOR OPERATIONS AND CHAIRMAN, SPECIAL REVIEW GROUP, U.S. NUCLEAR REGULATORY COMMISSION

INTRODUCTION

The Special Review Group was established by the Executive Director for Operations soon after the fire to identify the lessons learned from this event and to make recommendations for the future in the light of these lessons. The members of the review group are NRC employees with expertise in the various technologies involved. Group members and fire experts from NASA and insurance underwriters have visited the plant and the licensee's engineering offices to obtain information for the group's evaluation. The names of the members of the review group are given in Attachment 1 to this Testimony.

We have tried not to duplicate the work of the investigation into the incident conducted by the Office of Inspection and Enforcement or the licensing activities of the Office of Nuclear Reactor Regulation described in other NRC testimony here today. Rather, we have used the technical information developed by these other NRC organizations. We have also made extensive use of the large amount of technical information developed and made available by Tennessee Valley Authority, the licensee, and information from other sources.

Our review is still underway. At the present time, we have completed most of the technical evaluation of the causes of the failures experienced. We are reviewing the response of the licensee's organization and the outside organizations from which TVA needed help, and also how outside organizations involved in the Browns Ferry emergency planning responded to this incident. In addition, we are reviewing the adequacy of NRC criteria and procedures and the performance of our agency in the review and inspection of nuclear power plants. The response of the NRC during and after the fire is also being examined. We are in the process of consolidating the lessons learned and formulating our recommendations. We expect to complete our work within the next month or two.

PERSPECTIVE

The Browns Ferry fire and its aftermath have shown up serious inadequacies. In addition to the direct fire damage, there were several kinds of failures. Some equipment did not function correctly, and, in hindsight, some people's actions were incorrect or at least not as productive as they should have been. The fire, although limited to one room in the plant, caused extensive damage to electric power and control systems, impeded the functioning of normal and standby cooling systems, degraded the capability to monitor the status of the plant, and caused both units to be out of service for many months. The history of previous small fires, the apparent ease with which the fire started and cable insulation burned, and the many hours that the fire burned—all direct our attention to fire prevention and fire fighting. The inoperability of redundant equipment for core and plant cool-down suggests that attention may also be needed with regard to adequacy of present separation and isolation requirements. Lapses in operating quality assurance programs also contributed to the event.

There is another way of looking at the lessons of the Browns Ferry fire. The outcome with regard to the protection of public health and safety was successful. In spite of the damage to the plant as a result of the fire, and the inoperable safety equipment, the reactors were shut down and cooled down successfully. Nobody on site was seriously injured. No member of the public was affected. No radioactivity above normal operating amounts was released. The nuclear fuel was not affected by the fire. The damage to the plant is being repaired. Based on our evaluation of the incident, we believe that even if a fire such as the one at Browns Ferry occurred in another existing plant, the most probable outcome would still be with no adverse effects on the public health and safety.

The question naturally arises: How can a serious fire that involved inoperability of so many important systems result in no adverse effect on the public health and safety? The answer is to be found in the defense-in-depth used to provide safety in our nuclear power plants today. It provides for achieving the required high degree of safety assurance by echelons of safety systems. The defense-in-depth afforded in this way does not depend on the achievement of perfection in any single system or component, but the overall safety is high.

Let us now apply this perspective to fires in nuclear power plants. With respect to fire the defense-in-depth principle is aimed at achieving an adequate balance in:

1. Preventing fires wherever you can.
2. Fighting those fires that occur, putting them out quickly and limiting their damage.
3. Designing the system so that a fire that gets started in spite of the fire prevention program and burns for a considerable time in spite of fire fighting will not prevent vital functions from being performed.

None of these echelons can be perfect or complete. Strengthening any one can compensate in some measure for weaknesses, known or unknown, in the others.

The lessons of Browns Ferry show that all three lines of defense had gaps, and yet the outcome of the Browns Ferry fire shows that the overall defense-in-depth was adequate to protect the public health and safety.

GENERAL RECOMMENDATIONS

We turn now to the discussion of the gaps in the lines of defense—the failures that occurred at Browns Ferry. They included weaknesses in equipment design, operating procedures, and quality control. The NRC must take its share of the responsibility for these failures. Browns Ferry was licensed after review of the proposed design and operation. So the lessons are for NRC as well as the industry.

Although our review is not complete, we believe that both fire prevention and fire fighting should be improved to decrease the probability of fire getting started, or having gotten started, burning for a long time. We believe that simple and effective measures are available to effect this improvement.

With regard to the third element of defense-in-depth—providing safety functions in spite of postulated fires—our review is not as far along as it is in other areas.

We believe that the fire at Browns Ferry has revealed that some improvements in existing plants are prudent. We expect our recommendations to call for a reevaluation of each line of defense against fire in each existing nuclear plant. This reevaluation has already begun. Immediately after the Browns Ferry fire, the NRC issued a bulletin to all licensees to review their maintenance and construction procedures, their fire stop designs and their fire protection equipment and procedures. Licensees should now be more alert to the possibilities of fire and better prepared to fight fires. Due to this greater awareness of the possible causes and consequences of fires, prevention should be improved.

As interim measures, emergency shutdown procedures, fire fighting procedures, and control of combustible material and ignition sources are being improved. Licensees are, where needed, upgrading firestops, improving fire-fighting training, tightening procedures involving ignition sources and combustible materials, and developing alternative cooldown methods. For the longer term, this should be supplemented by those improvements in each line of defense which are revealed by a reevaluation of each plant and determined to be appropriate.

BROWNS FERRY LESSONS—FIRE PREVENTION

We believe that the fire prevention lesson to be learned from Browns Ferry is that large bundles of electrical cables are more flammable than most people believed prior to the fire. The use of open flames to detect leaks, the frequency of occurrence of small fires as a part of the leak detection process, the ease with which the cable insulation was set afire, and the spread of flames all constituted a significant fire hazard. Our combustibility tests since the fire confirm that most cable insulation can burn and that even cables that pass various "flame retardancy" and "non-firepropagating" tests can ignite and propagate when grouped together in cable trays. Part of the reevaluation programs should be the recognition that groups of cables are combustible, and steps should be taken where needed to control this potential hazard. Cable insulation less flammable than that used at Browns Ferry is available, as are fire resistant materials that can be used to cover existing cables. Thus, improvements, where needed, are practicable in existing plants as well as future plants.

Improvements in procedures including special work permits and precautions for welding in such areas are being considered. Special control for construction activities involving these areas may need to be improved. In critical areas, additional or improved fire stops may be prudent in such places as openings in walls, floors, elevator shafts. Fire seals and all other fire prevention equipment may require improved inspection and better inventories of combustible materials in potentially hazardous areas may be needed.

The goal of these recommendations will be to reduce significantly the incidence of fires in nuclear power plants.

BROWNS FERRY LESSONS—FIREFIGHTING

While the fire in the cable spreading room was controlled by the installed carbon dioxide system, the fire in the secondary containment building was fought unsuccessfully with portable extinguishers using dry chemicals and carbon dioxide, and burned for several hours. It was then extinguished in a few minutes with water. Fire fighting plans may have to be revised in the light of this and such other fire-fighting experience. The present reluctance of some power plant personnel to fight fires with water must be considered. Additional and improved training may be required and provision of safety equipment to protect firemen using hoses on electrified wires may be needed. In vulnerable areas, such as cable spreading rooms and other cable concentration areas, fixed water deluge systems may be needed. Fire detection systems specific to the cable insulating material actually installed are being investigated. Improved periodic testing of fire fighting equipment should be considered.

The goal of these recommendations will be to improve significantly the capability to suppress and extinguish quickly those fires that may occur.

BROWNS FERRY LESSONS—LIMITING OF CONSEQUENCES

The Browns Ferry plant is typical in many respects of plants of its time. It was intended to have two separated and isolated divisions of cooldown equipment. Its provisions for local control of essential equipment go beyond those of some other plants in that this plant was designed to accommodate safely fire damage to the cable spreading room. Yet many items of redundant equipment became inoperable and the local control feature was not, in general, successful. The design of the plant, partially because of the duration of the fire, did not provide the expected degree of protection.

Separation criteria for more recent plants are superior to those used in the design of Browns Ferry, but further improvements may be prudent in the light of the Browns Ferry lessons. The reexamination of each existing plant must include a reevaluation of divisional separation of cables in the light of the Browns Ferry experience. We now know, for example, that cables in metal conduits can be damaged if the conduits are in the fire zone. Previous concepts of separation without barriers, or with thin metal barriers, must be reexamined. Some rerouting of cables may be necessary in some existing plants.

The design changes that may be needed to improve the capability to limit the consequences of fires depends on achieving a proper balance among the three elements of defense-in-depth—fire prevention, fire fighting, and limiting consequences of fires. To the extent that improvements in fire prevention and fire fighting make the likelihood of extensive fires very low, and to the extent that redundant systems are already provided, less emphasis may be needed on improvements in limiting the consequences of fires.

CONCLUSIONS

We have concluded that the Browns Ferry fire has not shown that present plants are unsafe; that is, while fires can cause severe damage to plant equipment, the health and safety of the public are adequately protected. However, subject to our further study and final recommendations, the review group believes that improvements may be required in fire prevention, in fire fighting, and in limiting the consequences of fires in nuclear power plants.

ATTACHMENT 1

BROWNS FERRY SPECIAL REVIEW GROUP

Chairman: S. H. Hanauer, Office of the Executive Director for Operations, Nuclear Regulatory Commission.

Members:

H. E. Collins, Office of International and State Programs Nuclear Regulatory Commission.

S. Levine, Office of Nuclear Regulatory Research Nuclear Regulatory Commission.

W. Minners, Division of Technical Review, Nuclear Regulatory Commission.

V. A. Moore, Division of Reactor Licensing, Nuclear Regulatory Commission.

V. W. Panciera, Office of Standards Development, Nuclear Regulatory Commission.

K. V. Seyfrit, Office of Inspection and Enforcement, Nuclear Regulatory Commission.

Representative YOUNG. Dr. Hanauer, I have several questions here, but I am really going to address myself to one. I mentioned about the Navy a minute ago and firefighting—that is, about all the fire drills and so forth in the Navy. It has been my experience that the ship was never more vulnerable to fire damage than when it was in the navy yard with all sorts of torches burning and everything in a general state of upheaval.

Dr. HANAUER. I believe industrial experience is the same.

Representative YOUNG. I was going to draw the parallel, Doctor. Surely you are mindful that where there is construction going on in these operating plants that some special attention must be given to fire prevention under those circumstances?

Dr. HANAUER. Yes, sir.

Representative YOUNG. And you are doing that?

Dr. HANAUER. We are doing that, and some of our detailed recommendations which are still being drafted treat this question.

Mr. ANDERS. As a matter of fact, Mr. Rusche, who has the current responsibility of licensing plants and the repair of the TVA plant, has this aspect under particular review.

Mr. RUSCHE. That is certainly correct, Mr. Chairman. If you recall, one of the areas I mentioned we were giving specific attention to was the development of administrative procedures; that is, a key area that needs attention and control.

I think we have a good example here in which the procedures could have been improved, and could have been improved already at TVA.

Representative YOUNG. I appreciate that, and I also appreciate the fact that one of your biggest problems, I suppose, is that nobody ever expects an accident to happen to them, and so you are going to have to ride herd on all these plants pretty constantly, or we are going to have continual recurrence of these things.

Are there any questions?

QUALITY CONTROL EFFORTS

Representative PRICE. Mr. Chairman, I just would like to say for years the Joint Committee emphasized quality control. I don't know how many hearings we have had on that. Most of the equipment in this particular plant should have benefited over the years. So there is a real tight quality control program; yet we find here that one of the failures that occurred at Browns Ferry was in quality control.

Would someone address themselves to that and tell us what type of equipment fell into that category?

Dr. KNUTH. Yes. This is, of course, one of the issues that we did highlight in our investigation report and our enforcement correspondence with TVA. In the quality assurance system, adequate procedures for controlling work in vital areas such as the cable spreading room were not adequately evaluated; and in our view, procedures were not detailed enough to exert quality control. There also was not sufficient independence of the quality control people observing and witnessing what the people were doing in the work area.

There are some of the issues that we did identify in our inspection report and enforcement correspondence with TVA.

Representative PRICE. It would seem over the years, considering the amount of pressure that was put into this particular area of quality control, that you would not be finding this situation today.

Mr. ANDERS. Mr. Price, I would say that I certainly agree with your point about the emphasis on quality control in the nuclear industry—which stands out in this regard among other industrial activities. The record of performance in the quality control area far exceeds that of many activities at this same stage of development, so your message has not been lost on the industry on the regulator. Quite obviously, though, there are still improvements that can be made, and you can be sure that we are taking efforts to make such improvements on a continuing basis.

Representative PRICE. That is all I have.

Representative YOUNG. Mr. Anderson.

COMMON MODE FAILURES

Representative ANDERSON. Has your special review group in connection with its analysis of this fire learned anything about common mode failures or where the failure of one system triggers the failure of another system, anything that has not already been analyzed in that regard in some detail by the Rasmussen study.

Dr. HANAUER. Yes and no. Yes in the sense that we have studied the common mode failures which certainly did occur during the Browns Ferry fire because of redundant elements which were independent and were all failed by the same thing; namely, the fire, in some instances.

No in the sense that we didn't find anything radically new.

Representative ANDERSON. You found out that the common mode failures are possible, in other words, but not necessarily why.

Dr. HANAUER. Of course, we already knew that, but we proved it all over again.

Mr. ANDERS. But that is the defense-in-depth concept.

We have with us today Mr. Saul Levine, who has been a primary participant in the study, which has looked at statistical failure probabilities. I understand that his group has reviewed this particular incident, and you may wish to hear his views related to common mode failures and problems of accidents with regard to fires.

Representative ANDERSON. That is up to the chairman, of course, but it seems to me that it is a question that may well be raised somewhere along the line as to whether or not this report on which we depend very heavily—of course, whether it really took into account the kind of failures that occurred in the Browns Ferry fire.

Mr. Chairman, would you mind if Mr. Levine were to very briefly address himself to that?

Representative YOUNG. Not at all.

Representative ANDERSON. I know it is a long hearing.

Representative YOUNG. Let Mr. Levine be recognized for that purpose.

Mr. LEVINE. Thank you, Mr. Chairman.

I have to start by talking about what we did in the draft version of our study, and then I will bring you up to date with what we have done since the fire has occurred.

In the draft portion of our study we were concerned about the potential effects of fire in the cable spreading area or in other places where there are large numbers of cables. We found no body of statistical data readily available with which to quantify the likelihood of such fires, so we were unable to make a coherent analysis of it.

However, we did know the relative frequency of fires in plants was quite low, that there had been not a large history of large fires, or in other words, there had been a small history of large fires that said that the probability of such a fire that would cause extensive damage was quite low. Further, there are, in fact, fire-fighting systems, and there are, in fact, design features to cope with the consequences of such a fire by limiting the amount of damage.

We therefore concluded that, in our opinion the likelihood of a fire such as this would not contribute significantly to the overall risk of a nuclear powerplant accident.

We have another study underway now where we are going to collect data on large fires from many sources and try to create a statistical model.

In lieu of having that available, however, we have analyzed the specific situation that occurred at the Browns Ferry plant, and we are able to say, in principle, that our judgment was confirmed by the fire. We have estimated that the likelihood of occurrence of such a fire is about five chances in a thousand per year, and that the likelihood that it would cause a core to melt would be about five chances in a million per year. This is about 10 percent of the probability that we have predicted for the occurrence of core melt accidents independent of consideration of the contribution of a fire, so that the fire risk would not be a significant contributor to the overall risk.

Now, that conclusion, as I just stated, is specifically applicable to the Browns Ferry plant and therefore the analysis does not apply to all reactors.

One would have to examine the adequacy of fire fighting capacity and fire prevention features at other plants to make a broader statement; this kind of a study is going forward now.

It is my view that because of certain improvements in separation criteria in newer plants and because of the inspection of fire prevention and fire-fighting capacity by an independent inspection group, namely, the NELPIA group, or other similar groups, that the likelihood of such fires in other plants is probably less than at Browns Ferry and that our statistical study which will be performed in a year or so will give us a better understanding of this contribution more explicitly.

Representative YOUNG. The question was suggested, when is the Rasmussen report due out?

Mr. LEVINE. We hope to have it out by the end of October of this year.

Representative YOUNG. Any further questions?

PAPERWORK PROBLEMS

Representative HORTON. I am the temporary Chairman of the Commission to study the possibility of eliminating paperwork or cutting back on paperwork in the Federal Government. I was appalled at the amount of paperwork that is here in front of us, and I can imagine that there is quite a bit more paperwork that has been generated as a result of this fire.

Do you have any idea as to how much has been generated?

Mr. ANDERS. Well, we certainly support what sounds like the thrust of your remarks, Mr. Horton. Any guidance you can give us on how to reduce it would be helpful. I would imagine it goes into the tons. We have not done a particular study on this, and as I say, this is a very complicated subject. Our licensing procedures and licensing amendments procedures are heavily involved in filling out the printed page.

Representative HORTON. I am going to ask you here publicly, and from your suggestion and your willingness to cooperate. I have already asked, we just have a temporary staff and we are just getting ready to tool up. The Commission is going to meet the 3d of October, and then the Commission will be set up. I am going to ask you, if you will assign at least some of your personnel, one or two people maybe, to analyze this situation and give us a report on the paperwork commission so we can get some idea.

Representative YOUNG. On paper.

Representative HORTON. So we can get some idea as to what is involved in such an investigation as this with regard to filling out forms. I notice that there is a tremendous number of statements, and I know from my background as a lawyer that you do have to get a lot of statements, but I wonder if maybe there was not a little overkill here.

Mr. ANDERS. We never know. There might well have been but we also want to observe that we are responsive to the Joint Committee and to our public responsibilities.

Representative HORTON. You do have a lot of statements from a lot of people here. I am not critical of what you have done. I am just making the point that there is a lot of paperwork generated as a result of something like this, and we have to be very mindful of that, I think, when we are dealing in these Federal agencies.

If you are willing to cooperate—and I am going to ask the staff to be in touch with your staff—and help us with regard to an analysis of what has been generated as a result of this.

Mr. ANDERS. We would be most pleased to. As a matter of fact, if you would like to extend an invitation to review not just this investigation, which is a little bit atypical of our operations, but also the more general activities we have in our overall licensing effects, where the paperwork for any one plant reaches the size of truckloads.

[Information subsequently received follows:]

Analysis of the Paperwork Required as a Result of Browns Ferry Fire

NRC studies relating to the fire at the Browns Ferry station consisted of three separate but interrelated investigations. First, the Office of Inspection and Enforcement investigated the facts leading up to the fire, and actions taken at the plant during and immediately following the fire. The Office of Nuclear Reactor Regulation is reviewing the safety of the units during repair and restoration of the two Browns Ferry units to operation. A Special Review Group, led by the Technical Advisor to the Executive Director for Operations, is studying the longer term implications for the purpose of recommending improvements in NRC policies, procedures, and technical requirements.

With respect to much of the paperwork needed for the three investigations, the same documents were used by all participants wherever possible. Reports received from the licensee and reports prepared by NRC were evaluated by the separate groups for their own purposes, which accounts for the somewhat large number of copies prepared. Each document is shown only once and is attributed to the study first requiring the material.

	Pages	Copies	Total Pages
Inspection & Enforcement Investigation			
IE Bulletins to licensees	8	1,000	8,000
Licensee responses to Bulletins	8(av.)	1,000	8,000
IE reports of 51 plants inspected	10(av.)	3,200	32,000
Notice of violation to TVA	8		
IE report of investigation	444	400	177,600
Knuth's testimony for JCAE hearing	44	140	6,200
Nuclear Reactor Regulation Investigation			
8 NRR letters to TVA enclosing license amendments, revised technical specifications, safety evaluations, etc.	267	64	17,100
Written summaries of meetings with TVA	118	86	10,100
Rusche's testimony for JCAE hearing	288	140	40,300
Final NRR Safety evaluation report (target - Dec. 1975 - estimated)	75	65	4,800
Special Review Group Investigation			
SRG final report (not yet drafted - estimated)	200	400	80,000
Hanauer's testimony for JCAE hearing	9	140	1,300
Other			
4 public announcements	15	5,000	75,000
Chairman's testimony	6	140	840
TOTALS	1,500	11,775	461,240

The above count of major documents produced does not include drafts, internal NRC notes and memoranda, or the frequent correspondence between NRC and TVA. As an indication of paperwork relating to this investigation, attached is a listing of 447 documents totaling approximately 3000 pages concerning the Browns Ferry fire and its implications for other plants, or earlier documents on fires and electric systems at nuclear power plants which were used in connection with the current investigations.

APPENDIX A

DOCUMENTS RELATING TO BROWN'S FERRY FIRE AND ITS IMPLICATIONS FOR OTHER PLANTS *

1. October 23, 1967, ROE 67-11
2. February 5, 1969, ROE 69-4.
3. March 23, 1975, Mailgram from TVA to NRC:II, Notification of fire.
4. March 24, 1975, IE Bulletin 75-04.
5. March 24, 1975, Memo Davis to Regional Directors, IE Bulletin No. 75-04.
6. March 24, 1975, Memorandum Chairman Anders to Gossick, Commendation of Staff.
7. March 24, 1975, Transcript of news conference held at TVA.
8. March 24, 1975, Memorandum from S. Maple to C. Sullivan, Actions by State of Alabama
9. March 24, 1975, Plan for Continued Coverage of Browns Ferry Fire.
10. March 26, 1975, Memorandum from J. Hufham to Files, Actions by State of Alabama.
11. Undated, Statement of Responsibilities.
12. March 26, 1975, Notification of an Incident or Occurrence No. 145.
13. March 26, 1975, Memorandum Davis to Case, Browns Ferry Investigation.
14. March 27, 1975, Memorandum Gossick to Davis, Commendation of Staff.
15. March 27, 1975, Press Release No. 75-69.
16. March 28, 1975, Memorandum Davis to Grier, Briefing of ACRS.
17. March 31, 1975, Letter E. Garrett to NRC:II, Request for information.
18. April 2, 1975, Memorandum Davis to Moseley, Response of State/Local Governments.
19. April 3, 1975, Memorandum Thornburg to Regional Directors, Summary of Information.

*List does not include documents such as the staff safety evaluation which are already in the Browns Ferry docket files (50-259, 50-260) in the NRC Public Document Room.

20. April 3, 1975, Memorandum Davis to Regional Directors, IE Bulletin No. 75-04A.

21. April 3, 1975, IE Bulletin 75-04A.

22. April 7, 1975, Memorandum Davis to Thornburg and Gover, Commendation of Staff.

23. April 9, 1975, Letter Moseley to Garrett, Response to inquiry.

24. Undated, Memorandum Davis to Hanauer, Interim Plans for Browns Ferry Investigation.

25. April 16, 1975, Memorandum Clark to Murphy, Public Information Response.

26. April 18, 1975, Memorandum Davis to Regional Directors, Request for Special Inspections.

27. April 18, 1975, Memorandum J. Ward to Murphy, Browns Ferry Investigation.

28. April 22, 1975, Memorandum Long to Sinkule et al., Special Fire Inspections.

29. April 22, 1975, Memorandum Long to Gover, Schedule of Special Inspections.

30. April 23, 1975, Note O'Reilly to Moseley, Inquiry by Philadelphia Bulletin.

31. April 23, 1975, Letter H. Green to Murphy, Load Centers.

32. April 23, 1975, Memorandum Long to Gover, Schedule of fire inspections.

33. April 24, 1975, Memorandum Gage to Dreher, Schedule of special inspections.

34. April 25, 1975, Memorandum Davis to Hanauer, et al., Initial draft information.

35. April 25, 1975, Memorandum Davis to Knuth, Events during Browns Ferry Fire.

36. April 29, 1975, Memorandum Davis to Hanauer, et al., Draft copy of sequence of events.

37. April 29, 1975, Memorandum Knuth to Cossick, et al., Sequence of events during fire.

38. Undated, Memorandum Long to Thornburg, Status of Systems as of April 30, 1975.

39. May 2, 1975, Memorandum Long to Sullivan et al., Browns Ferry Operations Inspections.

40. May 2, 1975, Letter Moseley to TVA.

41. May 2, 1975, Memorandum Davis to Moseley, Inspection History.

42. May 2, 1975, Memorandum Davis to Grier, Inspection Program Conducted at Browns Ferry.

43. May 5, 1975, Transfer of Lead Responsibility, Cable Fire.

44. May 6, 1975, Memorandum Davis to Hanauer, et al., Status of Systems as of April 30, 1975.

45. May 8, 1975, Memorandum Wilson to Rusche and Hanauer, Browns Ferry Incident.

46. May 8, 1975, Memorandum C. Weaver and R. Condra to Files, Actions by State of Tennessee.

47. May 12, 1975, Memorandum Davis to Grier, Generic Concern on Utilities' Electrical Fire Fighting Capabilities.

48. May 15, 1975, Letter R. Wolle to Hufham, Actions by State of Tennessee.

49. Undated, Memorandum Cantrell to Moseley, Recommendations re Browns Ferry.

50. May 15, 1975, Memorandum Wilson to Cossick, Plans and Schedules

51. May 19, 1975, Memorandum Davis to Hanauer, Protection of fire evidence.

52. May 19, 1975, Letter Knuth to TVA.

53. May 20, 1975, Memorandum Davis to Regional Directors, Initial Draft copies of reports.

54. May 21, 1975, Memorandum Wilson to Thornburg, Comments on preliminary draft report.

55. May 21, 1975, Memorandum Wilson to Cossick, Agenda for meeting on May 23, 1975.

56. May 23, 1975, Memorandum Davis to Rusche, et al., Plans and schedules.

57. May 23, 1975, Memorandum Davis to Grier, JCAE Hearing.

58. May 27, 1975, Memorandum Wilson to Gossick, Schedule.

59. May 28, 1975, Memorandum Davis to Hanauer, et al., Initial draft copy of information.

60. May 28, 1975, Memorandum Davis to Hanauer, et al., Draft of TVA and government agencies response.

61. May 28, 1975, Memorandum Thornburg to Fiorelli, Enforcement actions.

62. May 30, 1975, Letter Parks to Moseley, Transmittal of report.

63. May 30, 1975, Memorandum Moseley to Davis, Inspection History at Browns Ferry.

64. June 3, 1975, Memorandum Knuth to Biles, Activation of EACT.

65. June 5, 1975, Memorandum Davis to Hanauer, et al., Request for Comments.

66. June 6, 1975, Note Wilson to Knuth, et al., PERT Network.

67. June 9, 1975, Memorandum Wilson to Gossick, Browns Ferry Fire.

68. June 10, 1975, Memorandum Wilson to Davis, Comments on Draft Report.

69. June 10, 1975, Note Wilson to Knuth, IE briefing of EDO.

70. June 11, 1975, Note Wilson to Gossick, Commission briefing.

71. June 13, 1975, Note Knuth to Gossick, Briefing.

72. June 13, 1975, Memorandum Grier to Coller, Responses to Bulletins and Inspection Reports.

73. June 13, 1975, Memorandum Wilson to Ippolito, Comments on Draft Document.

74. June 13, 1975, Memorandum T. Young to Murphy, Recommendation re drills.

75. June 16, 1975, Memorandum Grier to Davis, Inspection History at Browns Ferry.

76. June 16, 1975, Memorandum Thornburg to Hanauer, et al., Draft Report.

77. June 17, 1975, Memorandum Seyfrit to Thornburg, Comments on draft report.

78. June 19, 1975, Memorandum Hanauer to Davis, Comments on report.

79. June 25, 1975, Memorandum Moseley to Davis, Inspection History at Browns Ferry.

80. June 30, 1975, Memorandum Ward to Murphy, Recommendation re Investigations.

81. July 1, 1975, Memorandum Davis to Crier, et al., Draft Report.

82. July 1, 1975, Memorandum Davis to Hanauer, et al., Draft Report.

83. May 13, 1975, Memorandum Davis to Moseley, Inspection History at Browns Ferry.

84. July 1, 1975, Memorandum Seyfrit to Crier, Special Inspections.

85. July 1, 1975, Memorandum Crier to Davis, Special Inspections.

86. July 1, 1975, Memorandum Davis to Hanauer, Comments on Report.

87. July 2, 1975, Memorandum Knuth to Cossick, Investigation Report.

88. July 2, 1975, Memorandum Davis to Regional Directors, Special Inspections.

89. July 3, 1975, Memorandum Grier to Arlotto, Development of Nuclear Fire Code.

90. July 3, 1975, Memorandum Tripp to Reinmuth, Fire Protection Inspection Programs.

91. July 3, 1975, Memorandum Grier to Davis, Electric Fire Fighting Capabilities.

92. July 7, 1975, Memorandum Higginbotham to Kuhlman, Comments on Report.

93. July 7, 1975, Memorandum Kuhlman to Davis, Comments on Report.

94. July 8, 1975, Memorandum Moseley to Hanauer, Browns Ferry Fire Information Request.

95. July 8, 1975, Memorandum Moseley to Hanauer, Browns Ferry Fire Information Request.

96. July 9, 1975, Memorandum W. Swan to Files, TVA Fire Test

97. July 10, 1975, Memorandum Murphy to Gover, Comments on Proposed Notice of Violation.

98. July 11, 1975, Memorandum Moseley to Thornburg, Proposed Notice of Violation.

99. July 15, 1975, Memorandum Schwartz to Gossick, Contact with the State of Alabama.

100. July 30, 1975, Press Release 75-50 (Region II).

101. August 1, 1975, Memorandum Davis to Knuth, Management of Browns Ferry Report.

102. August 4, 1975, Letter Moseley to TVA, 50-259/75-9, 50-260/75-9.

103. August 5, 1975, Letter TVA to Moseley.

104. August 8, 1975, Memorandum Thornburg to Long, Comments from C. C. Long.

105. August 13, 1975, Letter TVA to Moseley.

106. August 15, 1975, Memorandum F. Long to Thornburg, Comments from C. C. Long.

107. August 15, 1975, Memorandum Davis to Grier, Congressional Testimony.

108. August 19, 1975, Memorandum Fiorelli to Seyfrit, Requirements for Penetration Seals, etc.

109. August 20, 1975, Memorandum Moseley to Davis, Fire Tests Being Conducted by TVA.

110. August 20, 1975, Letter Moseley to TVA.

APPENDIX B

DOCUMENTS RELATING TO BROWN'S FERRY FIRE AND
ITS IMPLICATIONS FOR OTHER PLANTS

1. March 1975 memorandum from J. Davis to Directors of Region Offices IE Bulletin No. 75-04 "Cable Fire at Browns Ferry Nuclear Power Station."

2. March 26, 1975 memorandum from L. V. Gossick to all NRC Employees "Appointment of Special Review Group."

3. March 30, 1975 article from the Tennessee (Nashville) entitled "You Can Blow Out Candle, but N-Power?" by Dolph Honicker.

4. April 3, 1975 memorandum from J. Davis to Directors of Region Offices IE Bulletin No. 75-04A "Cable Fire at Browns Ferry Nuclear Plant."

5. April 4, 1975 chart entitled "Standby Auxiliary Power System."

6. April 9, 1975 memorandum from A. Giambusso to E. Case entitled, "Browns Ferry Return to Operation."

7. April 10, 1975 memorandum from S. Hanauer to Review Group Members entitled "Independence of Review."

8. April 10, 1975 report "Interim Report Materials Flammability Testing For Nuclear Regulatory Commission."

9. April 17, 1975 letter from B. Rusche to TVA re: NRC's views concerning the review and approval requirements for the restoration of fire-affected features at Browns Ferry 1 & 2."

10. April 17, 1975 report from H. Russel TVA - Report on "Physical Damage to Electrical Cables and Raceways Involved in the Browns Ferry Nuclear Plant Fire on March 22, 1975."

11. April 22, 1975 letter from J. Gilleland of TVA to B. Rusche entitled, "Material to be provided Formally for Nuclear Regulatory Commission Review of the March 22, 1975 Browns Ferry Nuclear Plant Units 1 & 2 Fire."

12. April 22, 1975 letter from E. Thomas of TVA to listed distribution entitled, "Plan for Evaluation, Repair and Return to Service of Browns Ferry Nuclear Plant Units 1 & 2 (March 22, 1975, Fire) - Revisions of April 20, 1975."

13. April 22, 1975 report from D. Patterson of TVA entitled, "Safety Analysis of the Browns Ferry Nuclear Plant Units 1 & 2 During Operations Related to Removal of Damaged Cabling, Cable Trays, and Conduits."

14. April 22, 1975 memorandum from K. Murphy to R. Houston entitled, "Inflight Radiation Monitoring Systems."

15. April 22, 1975 tests from Okonite Co. entitled, "Tray Cable Ampacity Tests."

16. April 23, 1975 letter from J. Gilleland of TVA to B. Rusche entitled, "Revision 1 to Plan for Evaluation, Repair, and Return to Service of Browns Ferry Units 1 & 2 (March 22, 1975-Fire).

17. April 25, 1975 Browns Ferry Fire Inspection and Enforcement Investigation.

18. April 29, 1975 memorandum from D. Knuth to Directors entitled, "Tennessee Valley Authority-Sequence of Events during the Browns Ferry Fire."

19. April 30, 1975 memorandum from T. Wambach to TVA entitled, "Summary of Meeting Held on April 24, 1975 at BFNP to Discuss the Safety Analysis for Removal of Fire-Damaged Components."

20. April 30, 1975 memorandum from J. Knight to T. Ippolito entitled, "Summary of April 16, 1975 Meeting on Browns Ferry Cable Fire of March 22, 1975."

21. April 30, 1975 Instruction letter from J. Studdard entitled, "TVA Browns Ferry Nuclear Plant Operations Instruction Letter No. 33."

22. Report from TVA entitled, "Plan for Evaluation, Repair, and Return to * Service of Browns Ferry Units 1 & 2 (March 22, 1975)."

23. Report from TVA entitled, "Division of Engineering Design Report No. * BT-DED (BHP-1) (Browns Ferry March 22, 1975-Fire)."

24. Appendix A to Facility Operating Licenses DPR-33 and 52 Technical * Specification and Bases for Browns Ferry Nuclear Plants Units 1 & 2 Limestone County, Alabama.

25. May 1, 1975 Recovery Plan - Browns Ferry Nuclear Plant Restoration * Schedule.

26. May 2, 1975 paper from P. J. Long entitled, "Browns Ferry Units 1 & 2 Status of Systems as of 4/30/75."

*Indicates Document is in Docket File in NRC Public Document Room.

27. May 2, 1975 memorandum from K. Goller to ADs entitled, "Browns Ferry."

28. May 3, 1975 memorandum from V. Stello to Files entitled, "Browns Ferry."

29. May 5, 1975 letter to K. Goller entitled, "Cable Fire."
*

30. May 5, 1975 Draft Facility Operating Licenses DPR-33 and 52 Temporary Technical Specification and Bases for B. :r Ferry Plant Units 1 & 2.

31. May 7, 1975 memorandum from J. Pittman ' /A entitled, "Control Rod Position Verification."

32. May 8, 1975 Technical Specification Appli. bility (from trip).

33. May 8, 1975 Facility Operating Licenses DPR-33 and 52 Temporary
* Technical Specifications and Bases for Browns Ferry 1 & 2. (issued to NRC staff on site).

34. May 8, 1975 memorandum from T. Ippolito to Distribution entitled, "Browns Ferry Task Group Assignments."

35. May 8, 1975 memorandum from T. R. Wilson to B. Rusche and S. Hanauer, entitled "Request for Information on Fire at Browns Ferry in Order to Carry out Recent Assignment."

36. May 9, 1975 letter from R. Purple to TVA issuing Amendment No. 9 &
* 6 to Browns Ferry Technical Specificaztions.

37. May 9, 1975 memorandum from B. Grimes to V. Stello entitled, "Browns Ferry Fire."

38. May 12, 1975 memorandum from S. Hanauer to Review Group Members "Notes from Review Group Meeting, May 9, 1975."

38a. May 12, 1975 memorandum from N. Dube to V. Stello concerning RMS codes
* for Browns Ferry.

39. May 13, 1975 Browns Ferry Nuclear Plant Status of Refueling Equipment Preliminary Report."

40. Working Papers developed at Browns Ferry Trip (5/2-5/75).

41. May 15, 1975 memorandum from T. R. Wilson to L. V. Gossick entitled, "Browns Ferry Fire, Plans and Schedules."

42. May 15, 1975 Revision 4 to report entitled, "Plan for Evaluation, Repair,
* and Return to Service of Browns Ferry, 1 & 2."

43. May 16, 1975 memorandum from J. Knight to T. Ippolito entitled, "Summary of May 15, 1975 Meeting on Defueling Program Plan for Browns Ferry."

44. May 19, 1975 meeting notice with TVA.

45. May 19, 1975 letter from D. Knuth to TVA asking for clarification, * information and schedule for submitting material to NRC on fire.

46. May 20, 1975 TVA Safety Analysis of Browns Ferry Nuclear Plant Units 1 & 2 During Operations Related to Fuel Removal, Fuel Storage, and Plant Restoration.

47. May 20, 1975 memorandum from F. Rosa to T. Ippolito entitled, "Summary of Site Visit Made on May 5-9, 1975 to Browns Ferry Nuclear Plant (in regard to Cable Fire Incident of March 22, 1975).

48. May 21, 1975 memorandum from V. Stello to B. Rusche thru R. Heineman entitled "Summary of SAR Audit of Fire Stop Requirements."

49. May 22, 1975 Recommendations Related to the Browns Ferry Fire of March 22, 1975 (Outline by Special Study Group).

50. May 23, 1975 memorandum from V. Stello to T. Ippolito entitled, "Browns Ferry."

51. May 23, 1975 letter from J. E. Gilleland to B. Rusche entitled, * "Startup Report - Browns Ferry Nuclear Plant Unit 2 - Docket No. 50-260 Operating License DPR-52."

52. May 24, 1975 Final Draft TVA Safety Analysis of Browns Ferry Plant Units 1 & 2 During Operations Related to Fuel Removal, Fuel Storage, and Plant Restoration.

53. May 27, 1975 memorandum from T. R. Wilson to L. V. Gossick entitled, "Schedule NRC Actions Regarding Browns Ferry Station."

54. May 27, 1975 Revision 5 to report entitled, "Plan for Evaluation, Repair, * and Return to Service of Browns Ferry."

55. May 27, 1975 Revision 6 to report entitled, "Plan for Evaluation, Repair, * and Return to Service of Browns Ferry."

56. May 29, 1975 memorandum from R. Boyd to B. Rusche regarding TVA press release.

57. May 30, 1975 memorandum with enclosures from T. Ippolito to V. Stello entitled, "Summary of Field Trip Made on 5///2-5/75 to Browns Ferry Nuclear Plant."

58. May 30, 1975 Revision 7 to report entitled, "Plan for Evaluation, Repair, and Return to Service of Browns Ferry."

59. May 30, 1975 letter from E. F. Thomas to B. Rusche entitled, "Tennessee Valley Authority - Browns Ferry Nuclear Plant Unit 1 - Docket No. 50-259 - Facility Operating License DPR-33 - Abnormal Occurrence Report BFAO-50-259/758W."

60. May 30, 1975 letter from E. F. Thomas to B. Rusche entitled, "Tennessee Valley Authority - Browns Ferry Nuclear Plant Unit 2 - Docket No. 50-260 - Facility Operating License DPR-52 - Abnormal Occurrence Report BFAO-50-260/757W."

61. June 2, 1975 memorandum from E. Case to S. Hanauer entitled, "Protection of Fire Evidence in Restoration of Browns Ferry Station."

62. June 2, 1975 amendment to Facility Operating Licenses DPR-33 and 52 Interim Technical Specifications and Bases for Browns Ferry Nuclear Plant Units 1 & 2.

63. June 3, 1975 letter from J. Gilleland to B. Rusche requesting a delay in submission of ECCS information.

64. June 4, 1975 press release from TVA.

65. June 5, 1975 letter from J. Gilleland to B. Rusche requesting an extension on completion of construction date due to fire at Units 1 & 2.

66. June 5, 1975 package from R. Scholl to T. Ippolito regarding Best Estimate of Sequence of Events.

67. June 6, 1975 Revision 9 to report entitled, "Plan for Evaluation, Repair, and Return to Service of Browns Ferry."

68. June 6, 1975 TVA Technical Specifications input from C. Long and L. Riani.

69. June 6, 1975 memorandum from T. R. Wilson to D. Knuth, B. Rusche, and S. Hanauer entitled, "Pert Network for Activities Related to Browns Ferry."

70. June 6, 1975 letter from E. F. Thomas to B. Rusche entitled, "TVA-
* BFNP Unit 1 - Docket No. 50-259 - Facility Operating License DPR-33 - Abnormal Occurrence Report BFAO-50-259/759W."

71. June 9, 1975 Abnormal Occurrence Report BFAO-259/756W (Final)
* concerning a fire in the cable tray room.

72. June 9, 1975 letter from J. E. Gilleland to R. A. Purple entitled,
* "In the Matter of TVA" regarding revision of procedures for obtaining and preserving evidence.

73. June 11, 1975 NRC Review Effort (slides).

74. June 12, 1975 memorandum from F. Rosa to H. Sullivan regarding CPB input to Browns Ferry Fire SER.

75. June 12, 1975 memorandum from R. Vollmer to R. Purple entitled, "QA Branch Evaluation on TVAs QA Plan (Part XIII, Section A, dated May 28, 2975) for the Restoration of Browns Ferry Units 1 & 2."

76. June 13, 1975 memorandum from T. R. Wilson to T. A. Ippolito entitled, "Comments on Draft Document Entitled, NRC Evaluation of the Browns Ferry Fire."

77. June 16, 1975 memorandum from F. Schroeder to T. R. Wilson entitled, "Comments-Preliminary Draft of Schedule of Investigation and Evaluation of Fire at Browns Ferry Station."

78. June 17, 1975 letter from R. Purple to TVA requesting Operating License
* No. DPR-33 and 52 for BF be amended.

79. June 18, 1975 letter from E. F. Thomas TVA to B. Rusche entitled, "Tennessee
* Valley Authority - Browns Ferry Nuclear Plant Unit 2 - Docket No. 50-260 - Facility Operating License DPR-52 - Abnormal Occurrence Report BFAO-50-260/758W.

80. June 24, 1975 report from J. Gilleland to B. Rusche entitled, "12th
* Revision to Plan for Evaluation, Repair and Return to Service of Browns Ferry Units 1 & 2."

81. June 24, 1975 report entitled, "Browns Ferry Fire-Status-NRR Evaluation."

82. June 24, 1975 report entitled, "Browns Ferry Fire-Status-NRR Evaluation."

83. June 25, 1975 memorandum from L. Gossick to all Directors entitled, "Browns Ferry Review Group."

84. June 26, 1975 memorandum from F. Schroeder to Martin Biles entitled, "Inter-Agency Loan -- Andrew J. Pryor."

85. June 27, 1975 memorandum from B. Rusche to L. Gossick entitled, "NRR Evaluation Plan - Browns Ferry Fire."

86. August 4, 1975, Letter to David Comey from E. G. Case, in response to Mr. Comey's ltr of June 16, 1975 re: Indian Pt and Dresden 1.

87. June 30, 1975 letter from J. E. Gilleland to A. Giambusso entitled,
* "In the Matter of the Tennessee Valley Authority Docket Nos. 50-259 and 50-260."

88. June 30, 1975 letter from J. E. Gilleland to A. Giambusso entitled,
* "In the Matter of the Tennessee Valley Authority Docket No. 50-296."

89. July 1, 1975 report entitled, "Protection Against Fire - Objective: One Division of Shutdown Cooling Available."

90. July 2, 1975 memorandum from S. H. Hanauer to F. Schroeder, V. Stello, and R. Tedesco entitled, "Conflicting Design Requirements."

91. July 3, 1975 memorandum from P. Collins to F. J. Williams entitled, "Response to Review of the Browns Ferry Fire Report."

92. July 3, 1975 memorandum from B. H. Grier to K. Goller entitled, "Fire Protection Requirements for Nuclear Power Plants."

93. June 27, 1975 memorandum from A. Schwencer to distribution entitled, "Comey Letter on Browns Ferry Fire."

94. July 9, 1975 letter from J. E. Gilleland to B. Rusche entitled, "In the Matter of Tennessee Valley Authority Docket Nos. 50-259 and 50-260. Enclosure: "Plan for Evaluation, Repair, and Return to Service of Browns Ferry Units 1 & 2."

95. July 9, 1975 letter with enclosure from J. E. Gilleland to A. Giambusso entitled, "In the Matter of the Tennessee Valley Authority Docket Nos. 50-259 and 50-260.".

96. July 10, 1975 memorandum from S. Hanauer to Browns Ferry File entitled, "Meeting with Nelpia, June 26, 1975."

97. July 15, 1975 letter from R. Purple to TVA with corrected pages 54 and 108 of Change No. 11 to Technical Specifications Appendix A of Facility Operating Licenses.

98. July 16, 1975 summary of meeting held on July 1, 1975 at NRC to discuss design modifications at BFNP.

99. July 18, 1975 letter from E. F. Thomas to B. Rusche with report on broken second stage stem in main steam relief valves.

100. July 21, 1975 summary of meetings held on May 20 and May 21, 1975 at NRC offices to discuss TVA's Safety Analysis Report and Technical Specifications for unloading fuel from Units 1 and 2 prior to restoration work.

101. July 21, 1975 from T. Wambach to FACILITY: BFNP Units 1 & 2 summary of meeting held on July 11, 1975.

102. July 22, 1975 letter from J. E. Gilleland to B. Rusche entitled, "In the Matter of Tennessee Valley Authority Docket Nos. 50-259 and 50-260 enclosure: "Plan for Evaluation, Repair and Return to Service of Browns Ferry Units 1 and 2... regarding the 3-22-75 fire..."

103. July 22, 1975 memorandum from T. Wambach to Files entitled, "NRC Approvals of Restoration and Return to Operation of Browns Ferry Units 1 and 2."

104 July 22, 1975 TELEX from TVA entitled "Criteria for Splicing Cables Recovery Plan for Browns Ferry Units 1 and 2."

105. July 24, 1975 copy of "Questions Developed during Development Mini Draft for Browns Ferry Review Group Report."

106. July 25, 1975 letter from J. Gilleland to B. Rusche concerning, • "15th Revision to Plan for Evaluation, Repair, and Return to Service of Browns Ferry Units 1 & 2 (March 22, 1975 Fire).

107. July 29, 1975 note from V. Moore to T. Ippolito entitled "Questions on the Browns Ferry Fire."

108. July 30, 1975 memorandum from V. Moore to T. Ippolito entitled, "Questions on the Browns Ferry Fire."

109. July 31, 1975 final draft entitled, "TVA - Browns Ferry Nuclear Plant Preoperational Retest Program for Units 1 and 2 - Part XI Section C Recovery Plan."

110. August 4, 1975 letter from F. J. Williams to V. Stello entitled,"Status Report - Browns Ferry Fire Task Force."

111. August 15, 1975 from J. E. Gilleland to B. Rusche with enclosure of 16th • Revision of document entitled "Plan for Evaluation, Repair, and Return to Service of Browns Ferry Units 1 and 2."

112. August 15, 1975 letter from R. Purple to TVA with enclosure of Amendments • No. 13 and 10 to Facility Licenses No. DPR-33 and 52 for BFNP.

113. August 18, 1975 memorandum from T. Ippolito to R. Vollmer entitled, "Browns Ferry Recovery Plan Final Draft Part XI Section C."

114. August 21, 1975 information report entitled, "The National Observer Article on Browns Ferry Fire."

115. August 20, 1975 letter from J. Gilleland to B. Rusche with enclosure of • 18th Revision to document entitled, "Plan for Evaluation, Repair and Return to Service of Browns Ferry Units 1 and 2." -

116. August 20, 1975 memorandum from F. J. Williams to File entitled, "Browns Ferry Fire - Alternate Cooling Methods During Incident."

117. August 21, 1975 memorandum from F. J. Williams to S. Hanauer entitled, "Browns Ferry Restoration - One Hour Divisional Separation."

118. August 25, 1975 memorandum from V. Stello to R. Maccary entitled, "Browns Ferry Restoration.

119. August 25, 1975 memorandum from R. Higgins to facility entitled, "Summary of Meeting on 8/19 - 20."

120. August 27, 1975 letter from J. Gilleland to B. Rusche transmitting Revision 19 to Plan for Evaluation and Repair.

121. August 28, 1975 letter from B. Rusche to TVA regarding: Restoration work.

122. August 28, 1975 memorandum from C. Long to F. Williams transmitting Fire Protection Engineering Consultant Report.

123. September 2, 1975 (originally 8/18) letter from J. Gilleland to B. Rusche, "TVA Responses to NRC Questions of August 13, 1975."

124. September 2, 1975 letter from J. Gilleland to B. Rusche regarding: Revision 1 to TVA Design Criteria.

125. September 2, 1975 letter from J. Gilleland to B. Rusche regarding: 20th Revision to Plan for Evaluation, Repair and Return to Service.

APPENDIX C

DOCUMENTS RELATING TO BROWN'S FERRY FIRE AND
ITS IMPLICATIONS FOR OTHER PLANTS

1. Undated, Memorandum from John Davis to Steve Hanauer, "Interim Plan for Browns Ferry Investigation."

2. Undated, Memorandum from T. R. Wilson to John Davis, "Preparations for the JCAE Hearing on Browns Ferry Nuclear Plant Fire."

3. Undated, Chart - NRC Investigation of the Browns Ferry (BF) Fire Objectives.

4. Undated, Journal Article (Chemical Engineering Dept - Washington State Univ.) "Fireproof Insulating Material from Cenospheres" R. A. V. Raff, H. F. Austin, and W. Long.

5. March 24, 1975, Notes on Trip to Browns Ferry to Investigate Fire in Cable Trays - Steve Hanauer

6. March 24, 1975, Memo to Licensees from J. P. O'Reilly, "Cable Fire at Browns Ferry Nuclear Power Station."

7. March 24, 1975, Press Release (Contact Frank Ingram) re Note to Editors and Correspondents: Information on Fire at Browns Ferry

8. March 26, 1975, Notification of Incident or Occurrence (I&E) Fire in Electrical Control Cables (Tennessee Valley Authority - Browns Ferry Nuclear Power Plant)

9. March 27, 1975, Memorandum from Harry Thornburg to Regional (I, III, IV, V) Directors, "Cable Fire at Browns Ferry on March 22, 1975."

10. March 27, 1975, Press Release (Contact Joseph Fouchard) re NRC Statement on Browns Ferry Fire

11. March 28, 1975, Note from W. M. Morrison to Steve Hanauer "Sandia - Cable Tray Fire Tests."

12. March 28, 1975, Review Group for Browns Ferry Fire - Interim Work Outline.

12a. March 28, 1975, Levine, Relationship of Browns Ferry to Rasmussen Study Results and Methodology.

13. March 31, 1975, Memorandum from Aubrey Godwin, Director, Div. of Radiological Health, State of Alabama Dept. of Public Health, to Radiation Advisory Board of Health, Subject: March 22, 1975, Fire at Browns Ferry Nuclear Plant.

14. April 1, 1975, Newspaper Article, Kalamazoo Gazette, Tom Stersic, "Palisades' Safeguards Unlike TVA's N-Plant."

15. April 1, 1975, Letter from E. F. Thomas, TVA, to E. Case "TVA - BROWNS FERRY NUCLEAR PLANT UNIT 1 - DOCKET NO. 250-259 - FACILITY OPERATING LICENSE DPR-33 - ABNORMAL OCCURRENCE REPORT BFAO-50-259/756W

16. April 3, 1975, Press Release "NRC Staff Directs Broadened Review of Nuclear Power Plants Following Fire at Browns Ferry" (Contact Frank Ingram.

17. April 7, 1975, Memo Routing Slip from V. Panciera to Browns Ferry Special Review Group - Information Related to Sandia Testing of Cable Tray Fires.

18. April 15, 1975, Letter from H. C. Parris, TVA, to A. Giambusso, "Requested Amendment to Licenses DPR-33 and DPR-52 to change Technical Specifications on Browns Ferry."

19. April 15, 1975, Memorandum from Harold Collins to Stephen Hanauer "Emergency Preparedness Interest Areas (Browns Ferry Fire)."

20. April 15, 1975, Letter from J. Gilleland to A. Giambusso, "50-259 and 50-260 - Changes to Tech Specs - TVA - Browns Ferry."

21. April 16, 1975, Report, Review Group for Browns Ferry Work Plan - Draft

22. April 16, 1975, Memorandum from Frank Schroeder to W. Cornelius Hall, Chemtree Corp., "Fire at Browns Ferry and Control of Combustible Material in nuclear power plants."

23. April 21, 1975, Note from Voss Moore to S. Hanauer "Work Plan for Browns Ferry Review Group."

24. April 17, 1975, Memorandum (TVA) from J. R. Calhoun to Work Group - Nuclear Regulatory Commission - Commitments - Browns Ferry Nuclear Plant Fire Restoration.

25. April 23, 1975, Memorandum from S. Hanauer to E. G. Case, J. G. Davis, "Protection of Fire Evidence in Restoration of Browns Ferry Station."

26. April 24, 1975, Memorandum from Harold Collins to John Harris thru J. D. Lafleur "Your Note to Herbert H. Brown, DR:ISP Dated April 22, 1975 Concerning the Browns Ferry Fire."

27. April 28, 1975, Memo Routing Slip from W. Minners to Browns Ferry Review Group, transmitting Letter from F. Schroeder to Ralph Harding, Pres., The Society of Plastics Industry, Inc. (2/28/75).

28. April 29, 1975, Testimoney of Norman Rasmussen before the House Committee on Interior and Insular Affairs Subcommittee On Energy and the Environment.

29. May 5, 1975, Memorandum L. V. Gossick to T. R. Wilson, "Temporary Assignment."

30. May 6, 1975, Memorandum from John Davis to S. Hanauer, E. G. Case and F. Schroeder, "Browns Ferry Units 1 and 2 - Status of Systems as of April 30, 1975."

31. May 12, 1975, Memorandum from John Davis to Boyce Grier, "Generic Concern and Utilities' Electrical Fire Fighting Capabilities"

32. May 15, 1975, Memo Routing Slip from S. Hanauer to Browns Ferry Review Group "Meeting with T. R. Wilson"

33. May 19, 1975, Federal Register Notice "TVA - Issuance of Amendments to Facility Operating License.

34. May 23, 1975, Browns Ferry - Major Milestone Dates.

35. May 30, 1975, Memorandum from S. Hanauer to Browns Ferry Review Group - Notification of PRESS RELEASE BY TVA.

36. May 30, 1975, Letter from J. E. Gilleland to A. Giambusso "Proposed Changes to Browns Ferry Tech Specs."

37. June 3, 1975, Note from L. V. Gossick to B. C. Rusche and S. Hanauer, "Schedule for Review and Evaluation of the Browns Ferry Fire."

38. June 5, 1975, Memorandum from J. G. Davis to S. Hanauer, E. Case, F. Schroeder, T. R. Wilson, and K. Seyfrit, "Request for Comments on Draft Information Relative to the Fire at Browns Ferry."

39. June 6, 1975, Note from S. Hanauer to Browns Ferry Review Group "Schedule Revisions"

40. June 6, 1975, Memorandum from B. Grimes to C. Long, "Browns Ferry Recovery Plan Safety Evaluation."

41. June 16, 1975, Note from E. Case to L. V. Gossick, "Redundancy Available for Shut Down and Cool Down of Browns Ferry Unit 1 During Fire."

42. June 18, 1975, Memorandum from Harold Collins to Stephen Hanauer "Additional Information Request: Browns Ferry Fire Incident."

43. July 1, 1975, Note to Files from S. Hanauer, "Telephone Discussion with Mr. W. A. Riehl, Marshall Space Flight Center, March 26, 1975.

44. July 15, 1975, Note from Karl Coller to B. Rusche Thru A. Giambusso, "Electrical Cable Fire Stops."

45. July 15, 1975, Note from Sheldon Schwartz Thru Herbert Brown to L. V. Gossick, "Contact with State of Alabama on Browns Ferry Investigation."

46. July 18, 1975, Part of Press Release on Browns Ferry.

47. July 19, 1975, S. Hanauer Handwritten notes on Browns Ferry Separation

48. July 21, 1975, Meeting Summary from Thomas Wambach on Meeting Held on July 11, 1975, at NRC Offices to Discuss The Changes to Administrative Controls and Procedures Resulting From TVA's Evaluation of March 22nd Fire.

49. July 30, 1975, Statement by TVA General Manager Lynn Seeber in Response to the NRC Field Investigation Report on the Browns Ferry Nuclear Plant Fire on March 22, 1975.

50. August 1, 1975, Memo Routing Slip from T. Ippolito to S. Hanauer, Information on Fires at Beaver Valley 1; Nine Mile Point 1 and Salem Nuclear Generating Station.

51. August 5, 1975, Memorandum from T. Ippolito to S. Hanauer "Electrical Fires Aboard the Naval Aircraft Carrier, USS Saratoga."

52. August 8, 1975, Note from S. Hanauer to Mike Williams "Splicing in Browns Ferry and Elsewhere."

53. August 8, 1975, Memorandum from Paul Collins to D. J. Skovholt, "OLB Report of Browns Ferry Fire."

54. August 13, 1975, Letter from Ralph Nader to Chairman Anders, transmitting NELPIA Investigation of Browns Ferry Fire.

55. August 14, 1975, Letter from W. J. Dircks to J. Gilleland (TVA) transmitting NELPIA Investigation and Nader Letter to Gilleland.

56. August 26, 1975, Browns Ferry Review Group Meeting - August 21, 1975.

Appendix D

Documents Relating to Cable Tray Fires

1. February 3, 1965, Teletype to AEC from Philadelphia Electric Co., re: a fire at Peach Bottom Atomic Reactor Power Plant.

2. February 5, 1965, Report to Director of Regulation from Director of Compliance Report to Commission re Peach Bottom Fire of 2/3/65.

3. February 11, 1965, Note to B. Grier from J. P. O'Reilly transmitting two newclips concerning the Peach Bottom fire.

4. February 18, 1965, Memo to L. Kornblith from J. R. Sears - "Inspection of Philadelphia Electric Company (Peach Bottom Reactor)," Inspection Report.

5. February 18, 1965, Copy of Inspection Report transmitted by 4. above.

6. March 17, 1965, Ltr from H. L. Price to John T. Conway - Information on Peach Bottom Fire.

7. March 1, 1965, Letter to Honorable Nuiman Craley, Jr. from Harold L. Price, Summary of Info on Peach Bottom Fire.

8. February 24, 1975, Information on Peach Bottom Fire.

9. April 9, 1965, Memo to L. Kornblith from J. R. Sears "Inspection of Philadelphia Electric Company (Peach Bottom Reactor)" Inspection Report.

10. May 7, 1965, Memo to E. G. Case from L. Kornblith - Report from Field Inspector on recent visits to Peach Bottom Reactor.

11. November 24, 1965, Memo to D. M. Gardiner from D. F. Hayes "Fire at Peach Bottom Reactor on February 3, 1965"

12. July 26, 1965, Note to N. J. Palladino, W. D. Manly, Completion Date Extension

13. November 24, 1965, Memo D. M. Gardiner from D. F. Hayes Provides Information to J. McNamara on Peach Bottom Fire.

14. April 17, 1967 , Teletype to AEC from Philadelphia Electric Co. transmitting report to Director of Regulation.

15. February 9, 1968, Memo to D. J. Skovholt from J. P. O'Reilly, INQUIRY MEMO - Fire in Cable Penetration - San Onofre.

16. February 9, 1968, Memo, to J P O'Reilly from G. S. Spencer, Compliance Inquiry Memorandum - San Onofre fire.

17. February 21, 1968, Memo to J. P. O'Reilly from G. S. Spencer, Inspection Report of San Onofre fire.

18. March 8, 1975, Memo to Multiple addressees from J. P. O'Reilly - Investigation Report - San Onofre Fire.

19. March 13, 1968, Teletype to J. P. O'Reilly from G. S. Spencer re: Fire in Switchgear Room No. 2, San Onofre.

20. March 13, 1968, Memo to D. Skovholt from J. P. O'Reilly, transmits copy of teletype on San Ouofre fire.

21. March 15, 1968, Note to Dave Low, from R. Engelken. Report of telecon with Captain Bauser re San Ohofre fire.

22. March 15, 1968, Memo to the Commission from J. A. Harris - "Licensee's Announcement of San Onofre Shutdown.

23. March 19, 1968, Note to the Commission from W. B. McCool, transmitting Report to the Director of Regulation on San Onofre Fiie.

24. March 21, 1968, Memo to Files, from Marvin M. Mann - re: telecon with J. B. Moore, So. California Edison Co.

25. March 22, 1968, Memo to J. P. O'Reilly from G. S. Spencer, Field Report from visit on March 12-13.

26. March 28, 1968, Memo to F. A. Gifford, Report to ACRS on San Onofre Fire.

27. April 4, 1968, Letter to John T. Conway from Harold Price, re: San Onofre Fire.

28. April 15, 1968, Memo to Multiple Addressees from J. P. O'Reilly, Handouts from April ACRS meeting re: San Onofre Fire.

29. April 22, 1968, Memo to J. P. O'Reilly, from G. S. Spencer - Field Report on Site Visit March 28-29, 1968.

30. April 29, 1968, Memo to J. P. O'Reilly from G. S Spencer - Field Report on visit April 11-12, 1968.

31. May 3, 1968, Memo to Multiple Addressees from J. P. O'Reilly, transmits a copy of field report on inspection visit of March 28-29.

32. May 10, 1968, Note to Dr. Mann from R. H. Engelken, "Questions Raised Concerning the San Onofre Investigation Report (Compliance Report No. 50-206/68-3).

33. May 15, 1968, Memo to D. J. Skovholt from R. H. Engelken, Report of Inspection Visit April 11-12 on San Onofre.

34. May 24, 1968, Memo J. P. O'Reilly from G. S. Spencer - "Meeting with San Francisco Office Bechtel Personnel"

35. May 24, 1968, Memo to J. P. O'Reilly from G. S.Spencer - Report of Investigation from visit on May 2-3, 1968.

36. June 3, 1968, Note to Commission from W. B. McCool, Supplement to Secretary's report of March 19, 1968 on San Onofre Fire.

37. June 3, 1968, Letter to John T. Conway from Harold L. Price, Supplemental Information to letter of April 4, 1968 re San Onofre Fire.

38. June 5, 1968, Memo to D. J. Skovholt from R. H. Engelken "Meeting with San Francisco Office Bechtel Personnel"

39. June 12, 1968, Memo to D. J. Skovholt from R. H. Engelken, Inspection Report on visit May 2-3, 1968.

40. June 28, 1968, Memo to D. J. Skovholt from R. H. Engelken, Results of a Meeting with the regulatory staff and So. California Edison on June 20, 1968.

41. July 16, 1968, Memo to R. H. Engelken from Marvin M. Mann - "Return of San Onofre Plant to Operation"

42. July 16, 1968, Memo to J. P. O'Reilly from G. S. Spencer - Inspector's Report of June 11, 12 and 13 visit to San Onofre.

43. July 26, 1968, Memo to D. Skovholt from J. P. O'Reilly - Inspector's Report of June 11 - 13, 1968 visit to San Onofre.

44. August 1, 1968, Memo to Marvin M. Mann from R. H. Engelken, Clarification of several phrases used in memo to Dr. Mann from Mr. Engelken dated 6/28/68.

45. August 5, 1968, Note transmitting two letters to G. S. Spencer from J. P. O'Reilly.

46. August 6, 1968, Letter to Robert N. Coe from Peter A. Morris, Results of June 20, 1968 meeting.

47. August 6, 1968, Memo to D. J. Skovholt from R. H. Engelken - "Special Inspection of the San Onofre Nuclear Generating Station"

48. August 13, 1968, Letter to Peter A. Morris from Robert M. Coe - Request for Inspection date.

49. August 16, 1968, Memo to F. A. Gifford - Report to ACRS on San Onofre Fire.

50. August 16, 1968, Memo to D. Skovholt from R. H. Engelken - "Inspection Schedule - Southern California Edison Company.

51. August 16, 1968, Letter to Robert Coe from R. H. Engelken, "Inspection Schedule"

52. August 23, 1968, Note to H.L. Price from Peter A. Morris - "Resumption of Operations at San Onofre"

53. August 30, 1968, Memo to J. P. O'Reilly from G.S. Spencer, Inspector's Report on visit, August 14-15, 1968.

54. September 10, 1968, Memo to files from D. J. Skovholt, Summary of Telecon with J. B. Moore.

55. September 10, 1968, Memo to J. P. O'Reilly from G. S. Spencer, Inspector's Report of August 27 - 29, 1968 visit.

56. September 17, 1968, Memo to D. J. Skovholt from R. H. Engelken - Report of Second Preoperational Inspection of San Onofre - August 27 - 29, 1968.

57. September 19, 1968, Memo to The Commission from H. L. Price "Resumption of Operations at San Onofre"

58. September 24, 1968, Letter to John T. Conway from Harold L. Price, Supplemental Information to letters of April 4, and June 3, 1968 re San Onofre Fire.

59. September 30, 1968, Letter to Harold L. Price from Edward Bauser, JCAE. Requests additional information on San Onofre Fire.

60. November 18, 1968, Letter to H. G. Mangelsdorf, Report to ACRS of San Onofre Fires

61. November 22, 1968, Letter to Edward Bauser from Harold Price, Response to Mr. Bauser's letter to Mr. Price dated September 30, 1968.

62. March 13, 1969, Letter to H. G. Mangelsdorf, Report to ACRS - San Onofre Semi Annual Operating Report No. 3.

63. August 21, 1970, Memo to J. P. O'Reilly from W. C. Seidle - Report of Fire at Oconee Unit 1.

64. April 30, 1971, Special Report on the Operation of the San Onofre Plant.

65. November 5, 1971, Telegram to J. P. O'Reilly from William E. Caldwell Fire in DPR-26, Indian Point 2.

66. November 11, 1971, Memo to R. H. Engelken from J. P. O'Reilly - Indian Point 2 Fire.

67. November 14, 1971, Letter to Peter A. Morris from William E. Caldwell - Report of Indian Point 2 Fire.

68. November 15, 1971, Note to L. M. Muntzing from Lawrence D. Low - Con Ed Fire

69. November 15, 1971, Teletype to J. Fouchard from R. T. Carlson - Report of Indian Point 2 Fire.

70. November 16, 1971, Memo to R. C. DeYoung from R. H. Engelken - Preliminary Report of Indian Point Fire.

71. November 17, 1971, Note to L. M. Muntzing from Lawrence D. Low - Con Ed Fire.

72. November 17, 1971, Teletype transmitting previous teletype dated November 5, 1971 to Multiple Addressees from R. T. Carlson.

73. November 18, 1971, Memo to J. B. Henderson from E. M. Howard - Indian Point 2 Fire.

74. November 26, 1971, Memo to Files from J.B. Henderson, Documentation of Telecon with D.R. Muller on November 18 re Recovery from Recent Fire At Indian Point 2.

75. December 1, 1971, Memo to J. B. Henderson from E. M. Howard, Inspection Report

76. December 9, 1971, Memo to R. T. Carlson from E. J. Brunner - Inspection Report

77. December 15, 1971, Memo to D. R. Muller from J. B. Henderson, Inspection Report.

78. January 14, 1972, Memo to R. C. DeYoung from R. H. Engelken, Inspection Report of November 4, 1971 fire at Indian Point.

79. January 24, 1972, Memo to J. B. Henderson from E. M. Howard - Inspection Report

80. January 25, 1972, Inquiry Report prepared by A. F. Ryan re: A recent development believed to be in connection with IP-2 Fire of 11/4/71.

81. January 25, 1972, Memo to R. W. Carlson from E. J. Brunner - Inquiry Report

82. January 26, 1972, Memo to D. R. Muller from J. B. Henderson - Inspection Report conducted on December 29-30, 1971.

83. February 7, 1972, Memo to J. G. Keppler from E. J. Brunner - Inspection Report

84. March 13, 1972, CO Inquiry Report prepared by C. E. Murphy - Electrical Fire At Oconee Unit 1.

85. March 17, 1972, Memo to J. G. Keppler from E. J. Brunner - Indian Point 2 - Arson Indictment

86. March 17, 1972, CO Inquiry Report prepared by A. F. Ryan - Fire - Arson Indictment

87. July 24, 1972, Letter to J. F. O'Leary from F. A. Palmer. Details of Quad Cities Unit 2 fire in electrical cable trays.

88. August 4, 1972, Supplement No. 156 to ACRS Report on Quad Cities Fire

89. August 17, 1972, Memo to J. G. Keppler from D. M. Hunnicutt - Report of Inspections at Quad Cities Unit 2.

90. August 21, 1972, Supplement to ACRS Report - Quad Cities Unit 1 and 2

91. September 26, 1972, Inquiry Report prepared by F. S. Cantrell - re: Fire in Nine Mile Point 1.

92. October 18, 1972, Memo to D. J. Skovholt from R. H. Engelken - July 12 - 21, 1972 Report of Inspection.

93. October 31, 1972, Letter to Lawrence D. Low from William A. Conwell - Incident Report describing a Fire at Beaver Valley Unit 1.

94. December 30, 1972, Summary of Fire - Oconee Nuclear Station Unit 1

95. March 1973 - Report of Fire at Muhleberg Nuclear Power Plant

96. March 12, 1973, Memo to J. G. Keppler from C. E. Murphy re Oil Fire at Oconee Unit 1.

97. May 4, 1973, Letter to Angelo Giambusso from A. C. Thies - Incident Report of Fire at Oconee 1.

98. February 1974, International Guidelines for the Fire Protection of Nuclear Power Plants.

99. May 14, 1974, Letter to Donald F. Knuth from John F. Betz, Cable Tray Fire - Salem Nuclear Generating Station.

100. November 19,1974, Letter to Donald F. Knuth from E. N. Echwalje - Report of Fire - Hatch Containment No. 2.

101. April 1, 1975, Letter to E. G. Case from E. F. Thomas, Report of Details of Fire at Browns Ferry Unit 1.

102. April 4, 1975, - Multiple Addressees - memo transmitting IE Bulletins 75-04 and 75-04A.

103. April 23, 1975, Memo to ACRS Members from M. W. Libarkin - Detailed Sequence of Events during Cable Fire at Browns Ferry on Marcy 22, 1975.

104. April 25, 1975, Memo to ACRS Members from M. W. Libarkin - Report of Tour at Browns Ferry.

105. July, 1975, Article from Nuclear News on Miscellaneous fires in Nuclear Power Plants.

106. Article, "Peach Bottom Fire Spurs Improved Cable Design".

107. Listing of Fires in Nuclear Power Plants.

108. March 28, 1975, Memorandum to ACRS Members - Fire at Browns Ferry Unit 1.

APPENDIX E

Documents Relating to Regulatory Guide 1.75

1. July 5, 1972 Safety Guide "Physical Independence of Safety Related Electric Systems."

2. May 31, 1973 letter from L. Rogers to John C. Russ, Chairman, IEEE Nuclear Power Engineering Committee, General Electric Company.

3. September 7, 1973 letter from L. Rogers to John C. Russ, Chairman, IEEE Nuclear Power Engineering Committee, General Electric Company.

4. September 11, 1973 letter from John C. Russ to L. Rogers -"IEEE Standard on Separation."

5. October 12, 1973 memorandum from Guy A. Arlotto to S. Hanauer, J. Hendrie, R. Boyd, B. Grier, C. Kuhlman - "Regulatory Guide Review Request."

6. October 19, 1973 memorandum from Victor Stello to Guy A. Arlotto - "Response to Regulatory Guide Review Request on 'Physical Independence of Electrical Systems' (TAR-664)."

7. October 23, 1973 memorandum from R. B. Minogue to Thomas A. Nemzek - "Regulatory Guides."

8. October 26, 1973 Draft 1 - Regulatory Guide 1.XX "Physical Independence of Electric Systems."

9. October 26, 1973 memorandum from R. B. Minogue to Raymond F. Fraley - "Regulatory Guides."

10. November 8, 1973 memorandum from Voss A. Moore to R. B. Minogue - "Draft Regulatory Guide on 'Physical Independence of Electrical Systems'."

11. November 30, 1973 memorandum from R. B. Minogue to Raymond F. Fraley - "Regulatory Guide."

12. December 5, 1973 Draft 3 - Regulatory Guide 1.XX "Physical Independence of Electric Systems."

13. December 6, 1973 memorandum from L. Rogers to Frank Karas - "Regulatory Guides."

14. December 6, 1973 memorandum from L. Rogers to Raymond F. Fraley - "Regulatory Guides."

15. February 25, 1974 letter from Romano Salvatori, Manager, Nuclear Safety Department, Westinghouse Electric Corporation to Secretary of the Commission, U.S. Atomic Energy Commission.

16. April 4, 1974 letter from James F. Mallay, Manager, Licensing, Babcock & Wilcox to Secretary of the Commission, U.S. Atomic Energy Commission.

17. April 12, 1974 letter from J. S. Loomis, Head, Nuclear Safeguards & Licensing Division, Sargent & Lundy Engineers - "AEC Regulatory Guide 1.75 - Physical Independence of Electric Systems" to Secretary of the Commission, U.S. Atomic Energy Commission.

18. April 26, 1974 letter from Charles J. Maynard, Director of Project Engineering Services, Consumers Power Company to Secretary of the Commission, U.S. Atomic Energy Commission.

19. April 26, 1974 Draft A - "Draft Regulatory Guide 1.75, Revision 1 - Physical Independence of Electric Systems."

20. May 1, 1974 memorandum from Guy A. Arlotto to S. Hanauer, J. Hendrie, R. Boyd, B. Grier, C. Kuhlman - "Regulatory Guide Review Request."

21. May 8, 1974 memorandum from B. Grier to Guy A. Arlotto - "Draft Regulatory Guide 1.75, Revision 1, Physical Independence of Electric Systems."

22. May 9, 1974 letter from Guy A. Arlotto to L. M. Johnson, Secretary, IEEE - NPEC, Combustion Engineering, Inc. -

23. May 17, 1974 memorandum from R. Boyd to Guy A. Arlotto - "Guide Title: Draft Regulatory Guide 1.75, Revision 1, Physical Independence of Electric Systems."

24. May 21, 1974 memorandum from Victor Stello to Guy A. Arlotto - "Comments on Draft Regulatory Guide 1.75, Revision 1, Physical Independence of Electric Systems (TAR 930)."

25. June 5, 1974 Draft b - " Draft Regulatory Guide 1.75, Revision 1, Physical Independence of Electric Systems."

26. June 6, 1974 letter from Guy A. Arlotto to J. S. Loomis, Head, Nuclear Safeguards and Licensing Division, Sargent and Lundy Engineers.

27. July 10, 1974 letter from John A. Hinds, Manager, Safety and Licensing, General Electric Company to Secretary of the Commission, U.S. Atomic Energy Commission - "Regulatory Guide 1.75, 'Physical Independence of Electric Systems'."

28. July 15, 1974 "Draft Regulatory Guide 1.75, Revision 1, Physical Independence of Electric Systems."

29. August 20, 1974 memorandum from Victor Stello to Guy A. Arlotto - "Revision (July 15, 1974) to Regulatory Guide 1.75."

30. August 21, 1974 letter from R. I. Hayford, Chairman, Ad Hoc Committee 6, IEEE/PES/NPEC.

31. August 22, 1974 Working Paper A - "Draft Regulatory Guide 1.75, Revision 1, Physical Independence of Electric Systems."

32. October 22, 1974 letter from R. B. Minogue to John A. Hinds, Manager, Safety and Licensing, General Electric Company.

33. October 23, 1974 letter from R. B. Minogue to R. I. Hayford, Chairman, IEEE/NPEC AH6, Westinghouse Electric Corporation.

34. March 26, 1975 letter from W. R. Corcoran, Licensing Manager, Combustion Engineering, Inc. to Secretary of the Commission, U.S. Nuclear Regulatory Commission.

35. March 27, 1975 letter from Ivan F. Stuart, Manager, Safety and Licensing, General Electric Company to Secretary of the Commission, U.S. Nuclear Regulatory Commission - "Regulatory Guide 1.75, "Revision 1, 'Physical Independence of Electric Systems'."

36. March 31, 1975 letter from C. Eicheldinger, Manager, Nuclear Safety Department, Westinghouse Electric Corporation to Secretary of the Commission, U.S. Nuclear Regulatory Commission.

37. March 31, 1975 letter from Douglas E. Sahlin, Supervising Engineer - Nuclear Licensing to the Secretary of the Commission, U.S. Nuclear Regulatory Commission - "Regulatory Guide 1.75, Rev. 1 (January 1975) 'Physical Independence of Electric Systems'."

38. April 2, 1975 letter from William J. Cahill to Secretary of the Commission, U.S. Nuclear Regulatory Commission.

39. April 2, 1975 letter from James F. Mallay, Manager, Licensing, Babcock & Wilcox to Secretary of the Commission, U.S. Nuclear Regulatory Commission.

40. April 17, 1975 letter from A. E. Schubert to Secretary of the Commission, U.S. Nuclear Regulatory Commission.

41. May 6, 1975 letter from Sol Burstein, Executive Vice President, Wisconsin Electric Power Company to Secretary of the Commission, U.S. Nuclear Regulatory Commission - "Regulatory Guide 1.75, Physical Independence of Electric Systems."

42. May 21, 1975 memorandum from Guy Arlotto to H. J. Kouts, et al., - "Regulatory Guide Review Request - Physical Independence of Electric Systems."

43. June 2, 1975 memorandum from Stephen Hanauer to D. F. Sullivan, "Comments on Proposed Revision 2 to Regulatory Guide 1.75, 'Physical Independence of Electric Systems,', Dated May 12, 1975."

44. June 6, 1975 memorandum from Roger Boyd to Guy Arlotto, "Guide Title: Physical Independence of Electric Systems."

45. June 6, 1975 memorandum from J. F. Pearson to Guy Arlotto, "Regulatory Guide 1.75, Working Paper "A", Proposed Revision 2, Physical Independence of Electrical Systems, Dated May 12, 1975."

46. June 11, 1975 memorandum from Boyce Grier to Guy Arlotto,"Regulatory Guide 1.75 - Physical Independence of Electric Systems, Revision 2, May 5, 1975."

47. June 16, 1975 memorandum from V. Stello to Guy Arlotto, "Regulatory Guide 1.75 - Physical Independence of Electric Systems - Request for Postponement (TAR-1623)."

48. September 18, 1973, Draft C, Draft Regulatory Guide 1.XX "Physical Independence of Electrical Systems."

Representative HORTON. The other thing which is kind of related to that is this. I assume from the fact that there is a great amount of public attention on this particular subject and because of the controversy that we find nuclear energy and licensing procedures involved in today in America, that you have had to spend extra time.

I assume from your statistics it probably would not have been a type of accident that you would have expected to happen, and yet probably it is a good thing it did happen because it does give you an opportunity to review the procedures and the safeguards and the defenses that are involved in these nuclear plants.

Are you looking at this as a general investigation not only of what happened here, but also pointing toward the possibilities of fire and the attempt to eliminate fire as a hazard in these nuclear plants?

In other words, is there a bigger motive behind this than just the investigation of the Browns Ferry?

Mr. ANDERS. We tended today to focus on the accident investigation report, which I said earlier was really focussed on what went wrong down at Browns Ferry. In my view, the main value to be gained from this very realistic "fire drill" is what flows back up and speaks to inadequacies in our licensing system. Correcting problems found here would have a general impact on all nuclear powerplants, not just the Browns Ferry plant.

So the short answer is yes, it is the generic or broad implications which are the most important.

Dr. Hanauer has briefly gone over these in his oral summary, and his testimony which he submitted for the record goes into those in considerably more detail.

Representative HORTON. Thank you.

Representative YOUNG. Thank you, Mr. Horton.

I want to say in connection with the fun I was having with my friend from New York, there is no more able member of this House than the gentleman from New York and, if anybody can do anything about the paperwork, Mr. Horton can.

Representative HORTON. I have 13 other Commissioners, and we are going to do something.

FLAMMABILITY OF POLYURETHANE

Representative YOUNG. I want to ask one more question, Mr. Chairman, before we get off this, about the polyurethane.

It is undoubtedly highly inflammable. I don't know of anything more impractical than to have that type of material around a heat source such as a nuclear reactor.

With all the emphasis that we are putting on fire-retardent clothing and fire-resistant materials of all kinds, surely there must be some major effort made in trying to have a material that is not flammable, or fire-resistant at least. Are you engaged in that?

Mr. ANDERS. Yes. May we have Dr. Hanauer comment?

Representative YOUNG. Yes.

Dr. HANAUER. The use of polyurethane in Browns Ferry was approved, as Mr. Rusche pointed out, as a result of some testing that

TVA did. They built a model of it, and they, in effect, set fire to it, and showed that that resisted fire in a satisfactory way.

There are several kinds of polyurethane. The kind that was tested and the kind that was originally installed in these seals is flammable, but not extremely so, and, as I say, passed the test and behaved in a satisfactory way when it was tested.

The material that was being used and caught fire and started this fire was not the same material that was tested. The material that was tested is the kind that you mix together and it foams up, and you put it in where you want that to go, and it then hardens up and makes a rather hard material which then, as you recall, was covered with the fireproofing material.

The material that caught fire was a soft polyurethane form, it has the same name, but it is not the same substance, and it does not have the same flammability characteristics. It is more like the seat cushion material which you sit in and it gives as you sit.

It now turns out we have had some flammability tests done in the NASA testing labs.

Representative YOUNG. You might just talk to some of these Metrobus people about that, too.

Dr. HANAUER. Yes; this is the same problem.

So what caught fire was not the material which was tested and which was shown to have at least at the time satisfactory characteristics.

Now, Browns Ferry was designed before 1967. Today it is 1975, and we have some materials today which we didn't have in 1967, which have, as usual, some improved characteristics and some characteristics which are not as favorable as polyurethane.

I would like to address one more part of your question, and that is that these particular seals were not located in a hot area at all.

The building wall, the wall between the two rooms where these seals were placed, was subject just to the ordinary room temperatures and not to any heat.

Representative YOUNG. Apparently eventually they got exposed to considerable heat.

Dr. HANAUER. Yes. If you use this seat cushion material and then light it with a candle, it burns.

Representative YOUNG. I agree with you, the most exact science in the world is the science of hindsight, and I hope that we will draw on that large body of scientific knowledge to see that this does not happen again, if we possibly can, at least to the point of not letting people test one material and putting another one in the plant.

Are there any other questions?

That is all.

I believe that is all from the NRC group. I do thank you, Mr. Chairman, and your very able staff, for being here today before us.

Mr. ANDERS. Thank you very much.

Representative YOUNG. Is Dr. Hanauer still here?

Doctor, just one more question.

Have you and your group now banned that polyurethane that is burnable?

Dr. HANAUER. The study group does not ban anything, Mr. Chairman. We are a study group and we make recommendations. We have advised against its future use.

Mr. RUSCHE. Mr. Chairman, we have advised licensees to look at the use of such material or look at places where they may have used it and remove it, if it is in an unprotected condition, and not to use it in the future.

Representative YOUNG. Well, the answer to that is no, then.

Mr. RUSCHE. "Ban" has the connotation of going back and using it to apply to all the things in the past.

In the future we plan to ban its use.

Representative YOUNG. That is yes, then.

Mr. RUSCHE. Yes.

[Additional questions asked of NRC in writing, and their answers, are provided in appendix 9.]

Representative YOUNG. We will now hear from Mr. Aubrey J. Wagner, Chairman, Board of Directors, TVA.

I believe you have with you Mr. Gilleland, Assistant Manager of Power.

All right, Mr. Chairman, if you will just proceed.

STATEMENTS OF AUBREY J. WAGNER, CHAIRMAN, BOARD OF DIRECTORS, TVA, AND JACK E. GILLELAND, ASSISTANT MANAGER OF POWER, TVA, ACCOMPANIED BY E. G. BEASLEY, HEAD OF THE NUCLEAR ENGINEERING GROUP, DIVISION OF ENGINEERING DESIGN; JACK R. CALHOUN, CHIEF, NUCLEAR GENERATION BRANCH, NUCLEAR PLANT OPERATION AND MAINTENANCE; AND HARRY J. GREEN, NUCLEAR PLANT SUPERINTENDENT, BROWNS FERRY NUCLEAR PLANT

Mr. WAGNER. Thank you.

I have with me, as you indicated, Mr. Jack Gilleland who is our Assistant Manager of Power in TVA and also at the witness table are Mr. E. G. Beasley, Division of Engineering Design; Mr. Jack Calhoun, Chief of the Nuclear Generation Branch which is responsible for nuclear plant operation and maintenance in TVA and Mr. Harry J. Green, the nuclear plant superintendent of the Browns Ferry Nuclear Plant.

In addition to that we have other people backing us up and it might be that depending on the nature of your questions we would like some of them to help us with the responses.

Mr. Chairman, I have a very brief statement which I would like to make and then ask Mr. Gilleland to perhaps summarize his statement since much of what he has in it has already been said, but we are at your pleasure in that respect.

Mr. Chairman, we are pleased to have this opportunity to discuss with you and the other committee members the fire that occurred at TVA's Browns Ferry Nuclear Plant on March 22, 1975.

TVA's major objectives in any incident associated with its nuclear powerplants are to protect the health and safety of the public and plant personnel, and prevent damage to the reactor core and other major plant structures.

I think it is significant that these objectives were achieved in the Browns Ferry incident. This has been recognized in the recent NRC inspection and enforcement report on the fire and I believe it was related this morning and several times this afternoon.

The reactors were shut down safely and maintained in a safe condition. There was no radioactivity released from the plant above the normal fluctuations in the environmental background radiation. In fact, the radiation releases were only small fractions of that permitted by the operating license. No member of the public or plant employee was injured, with the exception of minor smoke inhalation sustained by a few employees who fought the fire.

It should be emphasized that the Browns Ferry incident has demonstrated the soundness of the underlying defense-in-depth design philosophy. The cooling equipment available permitted adequate core cooling, and nuclear fuel in the reactors remained covered by water at all times, even though several systems for cooling were lost due to the electrical fire.

The importance of having highly trained operators thoroughly familiar with the plant was illustrated in the initiative exercised in quickly shutting down and stabilizing the units.

The fact that the plant was safely shut down and stabilized during the fire shows that such unusual events can be coped with. That is the essence of the concern and care that go into a licensee's design and construction effort and the review given that effort by the NRC regulatory staff and licensing boards.

In effect the Browns Ferry fire was a test—although a most unwelcome one—of the ability of a nuclear powerplant to shut down safely under very difficult and extreme conditions. The plant and TVA personnel met that challenge successfully, but it was a costly test. Thus, the concept of defense in depth, based on sound design, engineering, construction, and operation does provide the necessary assurance that the public health and safety will be protected.

We must learn as much as possible from each incident which occurs, in order to improve these plants as the industry evolves. In the case of Browns Ferry much of the design criteria, design and construction involved in the fire was completed several years ago and Dr. Hanauer just stated that.

Improvements have since been made and incorporated into later plants. Nevertheless, as much must be learned as possible from Browns Ferry in order to benefit from the incident if possible. Both TVA and NRC are exerting extra efforts to do so.

TVA does not seek to minimize the seriousness of the Browns Ferry fire. Obviously, Mr. Chairman, we made mistakes or the fire would not have occurred. By the same token, it is a mistake to make the incident appear far more serious than it was by assuming, for example, that a radiological disaster was narrowly averted because a major pipe rupture could have coincided with the fire. Actually, a loss of coolant accident would be unlikely to occur in the full lifetime of such a plant—much less in the brief period during the fire that was required for cooldown after the reactors were quickly and safely shut down.

As a result of the fire at Browns Ferry, we are making changes in some of our work procedures and in some phases of the plant design at Browns Ferry, and I want to say they have been most helpful.

With your permission now, Mr. Chairman, I would ask Jack Gilleland, our Assistant Manager of Power, to give you a little fuller report on the fire and what is being done to avoid the repetition of it.

[Chairman Wagner's prepared statement follows:]

STATEMENT BY AUBREY J. WAGNER, CHAIRMAN, TENNESSEE VALLEY AUTHORITY

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It should be emphasized that the Browns Ferry incident has demonstrated the soundness of the underlying defense-in-depth design philosophy. The cooling equipment available permitted adequate core cooling, and nuclear fuel in the reactors remained covered by water at all times, even though several systems for cooling were lost due to the loss of electrical power.

The importance of having highly-trained operators thoroughly familiar with the plant was illustrated in the initiative exercised in quickly shutting down and stabilizing the units.

The fact that the plant was safely shut down and stabilized during the fire shows that such unusual events can be coped with. That is the essence of the concern and care that go into a licensee's design and construction effort and the review given that effort by the NRC regulatory staff and licensing boards.

In effect the Browns Ferry fire was a test—although a most unwelcome one—of the ability of a nuclear power plant to shut down safely under very difficult and extreme conditions. The plant and TVA personnel met that challenge successfully, but it was a costly "test." Thus the concept of defense-in-depth, based on sound design, engineering, construction, and operation does provide the necessary assurance that the public health and safety will be protected.

We must learn as much as possible from each incident which occurs, in order to improve these plants as the industry evolves. In the case of Browns Ferry much of the design criteria, design, and construction involved in the fire was completed several years ago. Improvements have since been made and incorporated into later plants. Nevertheless, as much must be learned as possible from Browns Ferry in order to benefit from the incident if possible. Both TVA and NRC are exerting extra efforts to do so.

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As a result of the fire at Browns Ferry, we are making changes in some of our work procedures and in some phases of the plant design at Browns Ferry. We are of course coordinating these changes and our repair work at Browns Ferry with the Nuclear Regulatory Commission.

Mr. Gilleland, TVA's Assistant Manager of Power, will describe in greater detail the events surrounding the fire as well as the steps we are taking to prevent any recurrence of the accident.

Representative YOUNG. Before we go into that, Mr. Chairman, I would want to ask one or two questions of you and perhaps Mr. Anderson might have a question to ask.

What monetary loss will TVA sustain as a result of the fire?

Mr. WAGNER. Well, the figures I have seen would indicate we have had about a $3.5 million repair problem and the losses of going to more expensive sources of power in the absence of this plant are perhaps in the neighborhood of $80 million to $100 million.

Representative YOUNG. The next question following that is, did this bring about any loss of electrical service to your customers?

Mr. WAGNER. No, sir, it did not. We have been able to maintain service although, as I have indicated, it is rather expensive.

Representative YOUNG. Here is another question, and it may be that you might want to defer it to subsequent witnesses. Was there anything in the Browns Ferry license or technical specifications or any other regulatory requirement which contributed to the problem associated with the fire?

Mr. WAGNER. I would like to defer to Mr. Gilleland on that.

Representative YOUNG. Mr. Anderson, do you have any questions?

Representative ANDERSON. I just have one question that it might be appropriate to ask this witness.

Some of the opponents of nuclear power have made the statement that after the fire both TVA and the Commission clamped a very tight lid on any news about the fire and issued what they described as a bland release. That is obviously intended, I think, to infer that maybe there was less than the kind of full disclosure that there should have been. What would be your response, Mr. Wagner, to that allegation?

Mr. WAGNER. I would say it is false. As I recall it, as quickly as we could, we opened the plant up to newsmen and they were in and could see what had happened and we had our staff there to answer questions as well as we were able to at the time. News releases were made as factual as we could make them. There were no bland releases. We did not underplay the fire.

At the same time, as I indicated in my testimony, I think the real concern in a case like this should be the safety of the plant and the safety of the public. There was no radioactive release, no one was injured, the plant was shut down and put in a safe condition and maintained that way. If any member of the press felt like he was shut out or denied information, I don't know what the basis for it would have been.

Representative ANDERSON. I appreciate your response because realizing your concern for the avoidance of panic by the public—and that is a justifiable concern—I think those of us who believe in nuclear power and see that it is a necessary alternative source of energy also feel that this whole question of public information is one that has to be handled not in the sense that we try to screen even the bad news from the public but that we fully communicate what does happen in a situation of this kind. I am very pleased to have you reply that there was no effort made then to stop the flow of any news that should have been made available about what happened.

Mr. WAGNER. Quite the contrary. We felt the public should know what was going on and we made every effort that we could to see that they could.

Representative ANDERSON. Thank you.

Thank you, Mr. Chairman.

Representative YOUNG. Do you have any questions, Senator?

Senator BAKER. No, thank you.

Representative YOUNG. Mr. Wagner, think you very much.

Mr. Gilleland, would you proceed, sir?

Mr. GILLELAND. Thank you, sir.

Mr. Chairman, I am Jack E. Gilleland, Assistant Manager of Power for the Tennessee Valley Authority. We are pleased to appear and make a statement on the fire that occurred at our Browns Ferry Nuclear Plant on March 22, 1975. I will discuss how the fire was ignited and extinguished, shutdown of the reactors, the availability of cooling equipment during the occurrence, and changes in design and administrative controls as a result of the fire.

Events leading up to and including the fire are discussed in the final report of TVA's Preliminary Investigating Committee, dated May 7, 1975.[1] The report, a copy of which has been provided for your use, is included as section III A in our overall "Plan for Evaluation, Repair, and Return to Service of Browns Ferry Units 1 and 2—March 22, 1975, Fire." The NRC Office of Inspection and Enforcement has issued a report. TVA filed a response to that report on September 2 pointing out where we agree and disagree.[2] Additional NRC reports remain to be completed.

HOW THE FIRE WAS IGNITED AND EXTINGUISHED

The fire started when an employee using a candle to check for air leaks around electrical cables penetrating the wall between the cable spreading room and the Unit 1 reactor building inadvertently ignited some material being used to seal the leak. Candles were a long accepted and highly effective method of detecting and locating leaks. The sealing operation under way was necessary since additional electrical cables added to the penetration had broken the previous seal.

The fire began to propagate due, in part, to the difference in air pressure between the two areas. The cable spreading room has a positive pressure with respect to the reactor building. The men were working in the spreading room. Thus, when the fire started, the flow of air into the reactor building caused the fire to spread through the penetration to the cables on the reactor building side of the wall.

The men working in the cable spreading room began immediately to fight the fire, using portable fire extinguishers, and some of the men went to the reactor building to fight the fire.

At 12:35 p.m., the plant fire alarm was sounded. At 12:51 p.m., an interval of 16 minutes, the Unit 1 reactor was tripped; at 1 p.m. the Unit 2 reactor was tripped. At about 1:10 p.m. in accordance with previously established arrangements and procedures the Athens, Ala. Fire Department was called to assist, if needed. They arrived at the site at about 1:45 p.m. The fire in the spreading room was successfully extinguished by 4:30 p.m. using dry chemical fire extinguishers and the

[1] See appendix 7.
[2] See appendix 14.

installed CO_2 fire protection system. Excessive smoke in the Unit 1 reactor building required the use of emergency breathing apparatus. The fire in the reactor building was controlled to some extent by using portable CO_2 and dry chemical extinguishers, but eventually it was necessary to spray water on the burning cables to extinguish the fire, using an installed in-plant fire hose. The fire was out by 7:30 p.m. Most of the fire damage was to cables on the reactor building side of the wall and was confined to a small area of the plant approximately 20 by 40 feet.

Once the fire was extinguished, an orderly approach to verifying actual conditions was undertaken and temporary power was reestablished to necessary equipment. The Plant Operations Review Committee met at 8:45 p.m. on March 22, to discuss plant conditions and reexamine and set priorities for restoration activities to be taken during the night.

The TVA radiological emergency plan was activated on March 22 as a precautionary measure, although no radiological emergency existed. A Division of Power Production Emergency Control Center in Chattanooga was established at 3:10 p.m. This management team participated in all major decisions associated with plant operation and firefighting activities. The Central Emergency Control Center—CECC—was staffed and directed from Chattanooga, Tenn., beginning at 3:25 p.m. The CECC performed valuable communications with the NRC—Atlanta—the Alabama Department of Public Health, and the Tennessee Department of Public Health. The CECC was closed at 10:30 p.m. on March 22; however, surveillance in the environment around the plant continued until 2 a.m., March 23, and the Environs Emergency Staff terminated its activities at 4:15 a.m., March 23, following the restoration of necessary equipment. The Division of Power Production Emergency Center was terminated at 2:45 p.m. on March 23, 1975.

Appendix A describes the sequence of events, describes the fire and equipment used in firefighting, and gives the time of various events for both the cable spreading room and the reactor building.

AVAILABILITY OF COOLING SYSTEMS

Mr. Chairman, the next portion of my statement deals with the reactor cooling system which I think has been well covered by the NRC with maybe one exception. I would like to refer to figure 4. I think there may have been a misunderstanding of the information presented on figure 4.[1]

Representative YOUNG. Without objection, Mr. Gilleland, proceed along the lines you have mentioned and we will just have your whole statement made a part of the record.

Mr. GILLELAND. Thank you, sir.

If you will note on figure 4, we show several high pressure systems that were available to insert cooling water into the reactor and this shows that some of the systems were available all of the time. These include the control rod drive system, which consists of three pumps, any one of which could have done the job adequately by itself; and was available for the full time. The reactor core isolation cooling could

[1] See page 148 for fig. 4.

have been made available by nonstandard operation, and the standby liquid control system also could have been made available.

Senator BAKER. Mr. Chairman, could I interrupt just for a second. How could they have been made available?

Mr. GILLELAND. The reactor core isolation cooling system is operated by a steam turbine. During the fire this turbine became isolated from the source of steam from the reactor to drive that turbine.

Now this system is one of the key systems for cooling and preliminary to issuing an operating license, it must be tested. There is an auxiliary steam line from an auxiliary boiler to allow the system to be tested. Because of safety aspects, this line then has a section removed during plant operation. This section of line, which is provided for the purpose, is kept close by and it could have been installed. As a matter of fact, the station auxiliary boiler had already been fired in anticipation of the need to do this.

Senator BAKER. It could have been done by the operator or somebody else getting instructions to install the joint pipe?

Mr. GILLELAND. Yes, sir, it was not needed.

Senator BAKER. The first thing that would happen is somebody would give the order and have the crew available to do it?

Mr. GILLELAND. Yes; sir.

Senator BAKER. Whose responsibility would it be to perceive the problem and to give the instruction?

Mr. GILLELAND. That would be the plant superintendent. The auxiliary boiler had been fired.

Senator BAKER. So then the operator knew that or the plant superintendent knew what the problem was early enough to have given the instruction to have made that system available?

Mr. GILLELAND. Yes; sir.

Senator BAKER. How much time are we talking about?

Mr. GILLELAND. Probably an hour.

Senator BAKER. What about the other system, the standby liquid control system?

Mr. GILLELAND. Those systems would have been made available by operating valves.

Senator BAKER. From the control room?

Mr. GILLELAND. No, sir, manually.

Senator BAKER. Now, just to visualize in my mind what you would have been up against, you would have had to have had the plant superintendent, an operator, someone able to perceive the problem, be aware of it and appreciate that systems were available to cope with it.

Mr. GILLELAND. Yes.

Senator BAKER. Was there any failure of that perception?

Mr. GILLELAND. No; sir.

Senator BAKER. Was there any proof that he was aware of the availability of that system—either the superintendent or the operator?

Mr. GILLELAND. The fact that the auxiliary boiler had been fired is proof that he was aware of it and making plans to use it if he had needed it.

Senator BAKER. Maybe I am asking for too much but in debriefing afterward did the operator or the plant superintendent say: "I thought at the moment that this could have been done and if it were necessary I could have taken those steps"?

Mr. GILLELAND. Mr. Green.

Mr. WAGNER. The plant superintendent is here.

Mr. GREEN. Senator Baker, the only documentation I could show that we thought of it was that evening about 8 o'clock we had a meeting of our Plant Operation Review Committee and the minutes of that meeting will reflect that at that time we were going to put this spool piece in.

Senator BAKER. I am not asking for proof; I am asking for your assurance to me that the level of competence of your operator was such that that system was not only perceived to be available but all you need to do is tell me yes, that is true.

Mr. GREEN. Yes; that we knew it was there.

Senator BAKER. That is not just an afterthought that your engineers thought up later, that had we been quick enough we could have done that. You or your operator were aware of that at the time?

Mr. GREEN. Yes; sir.

Representative YOUNG. Did I understand that the missing joint had been put in or was just available to be put into place?

Mr. GREEN. It was not put in, it is stored in a location adjacent to where it fits.

Representative YOUNG. Any further questions?

Representative ANDERSON. I have one question.

I don't want to plow the same ground again either but to my lay eye it still seems like it took an awfully long time to come to this ultimate management decision to use water on this fire. I am wondering since we have the plant superintendent here, and he is the man that apparently had to give that OK, why it took all of these hours of time before you came to the decision that this was a way to extinguish the fire.

Mr. GREEN. Mr. Anderson, it was a matter of priorities and concern. Now, initially and not by accident the operator took positive steps to trip the reactor. The priorities then went towards cooldown and fire fighting. We fought the fire in the spreading room continuously and I would like to describe the fire so that perhaps you can understand better.

This was not a conflagration that would fill this room, it was in the cable trays and it progressed at the rate of perhaps 1 inch per minute. By using dry chemical, we were able to keep the fire suppressed. Because of the heat in the cables when this was stopped, the fire would flame back. So we protected the control room by fighting the fire in the spreading room and we kept the spreading room from burning. However, when we stopped our efforts it would come back.

Now, one of the things that we were short of was air-breathing equipment. In the plant we had 24 air-breathing units and our people can get between 10 and 15 minutes out of a unit under stress. Now, these were used primarily to aline valves in an attempt to get into a shutdown cooling mode of the reactor, putting the priority there. After perhaps the first 30 or 40 minutes, the equipment loss was almost over and the jeopardy to the plant and to nuclear safety was not increasing much. The fire didn't spread clear through the plant, it stayed confined in one concrete room and many times it was reported out by using dry chemical and then it would flare back again.

After we got the plant depressurized and got a positive source of water going into the reactor, then we turned our attention to fighting the fire in the reactor building.

Now, as has been brought out, I did not authorize the use of water and it was because we had lost so much control equipment and we didn't understand at the time the mechanism whereby we had lost so much. We had lost redundant components that we didn't think you could lose and I was afraid that if we put water on we would short-out additional controls that we needed.

In retrospect some of the circuitry we used in shutting down the reactor did come through the fire zone. After we had the reactor in what I felt was a stable shutdown condition, then I authorized the use of water.

Representative ANDERSON. Thank you.

Representative YOUNG. Mr. Gilleland, would you proceed, please.

Mr. GILLELAND. Yes; I would like to continue my explanation of figure 4.[1] The point I wanted to make in connection with figure 4 is that there are a number of low pressure systems, and the chart shows that the low pressure systems were not available part of the time due to high pressure in the reactor. This does not mean that tho › systems were inoperable. It only means that their pumping head as lower than the pressure in the vessel and, therefore, in order to put those systems back into operation the only thing required was to depressurize the reactor which was done as you will note in the bottom line of the chart. The manual operation of the safety relief valves was reeffected at about 2130 and there was a period of 2 hours from 1930 to 2130 when there were no low pressure systems available, but not because they were inoperable. I felt that from the questions this morning it might have appeared that they were inoperable.

AUTHORITY FOR REACTOR SHUTDOWN

Senator BAKER. Mr. Chairman, let me ask one other question.

When I was at Browns Ferry shortly after the accident I had a good briefing then. I also asked who had to understand the situation and take the initial action. At that time I talked to you, the plant superintendent and to the operator. I have forgotten the operator's name.

Mr. GREEN. Gary McChristian, sir.

Senator BAKER. Is he here?

Mr. GREEN. No, he is not.

Senator BAKER. Was it his first responsibility or your first responsibility to issue the orders that were necessary for the reactor to shutdown and for shutdown cooling to commence?

Mr. GREEN. After my arrival at the plant I am the senior member. The individual operator has authority and responsibility to care for his reactor and to shut it down when his judgment warrants it. His supervisors were also in the station and in fact when he shut it down our plant shift engineer was at his side.

I am not sure I am answering your question.

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[1] See fig. 4, p. 148.

Senator BAKER. I am not sure I put it very well. What I am asking for, is who, if a single person, or names if there was a group, made the decision to do whatever was done? Was it a single operator? Was it the plant superintendent? Who was it?

Mr. GREEN. If I may take you through some of the key steps in sequence. When the time came to trip the reactor, the unit operator and his supervisor jointly agreed it needed to be tripped. The next what I would call a significant decision was to depressurize the reactors. I made that decision. In the control room at that time was our operation supervisor, the assistant operations supervisor and the shift engineer. They also agreed with that decision because it was a mutual discussion of what avenues were open to us.

Now, none of the actions of the afternoon occurred haphazardly. We plotted our course and we went there and it just didn't happen that the reactors got depressurized. This was the mode of shutdown we picked and I will say it worked. We shut down the reactor successfully without injuring a person and without endangering the public.

Senator BAKER. Or without damaging the reactor.

Mr. GREEN. Yes.

Senator BAKER. What I am reaching for really is a statement by you that it was normal and ordinary perception of the situation as it was presented to the operator and the management.

Mr. GREEN. Yes, sir.

Senator BAKER. And it took no particular intuition or lucky guess to take the action that was taken and that the shutdown was done in an orderly way.

Mr. GREEN. I think you will find that the shutdown, even though it was a partial manual operation, paralleled the normal shutdown method of the plant. There is much speculation why didn't you do this and why didn't you do that. I can tell you why we went the way we did because it paralleled a normal shutdown method.

Senator BAKER. I personally think you did a good job but I just wanted to establish the sequence.

Thank you.

Thank you, Mr. Chairman.

Representative YOUNG. Proceed, Mr. Gilleland.

Mr. GILLELAND. Mr. Chairman, I will make a final comment on this review of the core cooling system. I just make the statement that we think this demonstrates that there was indeed no near nuclear disaster.

I would like to go to the portion of my statement on improvements in design and in administrative controls, to show what we are doing to see that this sort of thing does not happen again.

The foregoing discussion illustrates that the Browns Ferry plant utilized an in-depth design philosophy with the flexibility and versatility which permitted plant operators to accommodate even the unique circumstances of this fire. Thus, the public health and safety was fully protected.

Nevertheless, our studies of the incident show that improvements can be made to substantially reduce the likelihood of such a fire in the future and to mitigate further the consequences of any fire that occurs. These measures consist of changes in administrative controls, design and fire protection.

CHANGES TO ADMINISTRATIVE CONTROLS AND PROCEDURES

The changes in administrative controls and procedures can be summarized as follows:

1. We have made procedural changes requiring greater in-depth review of all significant activities prior to their authorization. This procedure will ensure that all items of safety significance are recognized and provided for. In addition we require that all significant activities be reviewed by the plant Quality Assurance staff to verify that the activities are supported by approved instructions necessary for their performance.

2. Procedures have been written establishing controls over cutting, welding, and the use of any open flame in operating plants. This procedure requires a physical survey of the areas where the work is to be performed and the establishment of safeguards including additional firefighting equipment and a fire watch prior to any work involving open flame or welding.

3. TVA's general fire training program is being given increased emphasis. This includes fire reporting and firefighting techniques, including the use of water.

4. Procedures have been issued to require surveillance by engineering personnel of all penetrations under repair or construction. These procedures also provide for instruction of craft personnel engaged in penetration sealing and flameproofing, and provide for inspection by trained engineers. Instructions are being issued to require additional surveillance by firefighting personnel at open penetrations in operating areas.

CHANGES IN DESIGN

Reanalysis of the overall plant fire detection and protection has resulted in design changes in three areas:

1. Browns Ferry has two separate redundant groups, called divisions, of electric power and control circuits for shutdown cooling systems. Increased cable separation of these redundant safety system cables will reduce the probability of fires involving both the divisions. This has involved a redesign of circuits, rerouting of cables and conduits, and installation of additional fire barriers between divisions of cables.

2. Rerouting of selected cables will reduce the probability of events initiated within one unit adversely affecting any other unit.

3. Additional requirements for the control of combustible materials, fire detection, and fire prevention will give early warning of a fire and provide for rapid extinguishing of a fire. This objective will be accomplished by changes of three types: changes to the fire detection system, changes to the fire protection facilities and changes to cable tray penetration materials.

Additional fire detectors will be provided throughout the plant and along cable trays. These detectors will consist of thermal—heat—detectors and products-of-combustion—ionization—detectors. Annunciation of these detectors in the control room will allow plant operators to know the location of any detectors that have been activated.

The fire protection facilities will be upgraded by the addition of fixed water spray extinguishers along specific cable trays, provisions for portable access ladders and platforms for manual firefighting, additions of adapters to assure that hose and nozzle connections will be compatible with the equipment used by the local fire department, additional hose connections and racks, and automatic initiation of the fixed water spray system and of the CO_2 system in the cable spreading room.

TVA will install new material in future penetration seals and in penetration seals that are breached during the restoration activities.

CONCLUSIONS

Implementation of the safety objective of safety shutting down the nuclear reactors in the unlikely event of another fire falls naturally into two major areas of endeavor. First, administrative procedures have a dual rule of eliminating sources of ignition that could arise from personnel actions and then of efficiently fighting any fire that occurred. Second, modifications in the design of the electrical systems and in the fire protection systems will further reduce the effects of a fire on necessary shutdown systems. Both the administrative procedures and design are mutually supportive in this effort.

We believe that a desirable balance has been achieved between design features and personnel actions required to protect the plant against the effects of a fire. For a plant already constructed, the design to which we have committed has been optimized. We reaffirm that the Browns Ferry Nuclear Plant, as modified by the design changes summarized above, is safe.

Mr. Chairman, in closing I would like to say that the regulatory staff has conducted a rigorous review of the Browns Ferry recovery plan. They have worked long and hard to complete the reviews and issue reviews on a timely basis. In no case has the restoration schedule been held up by the review and approval process. We want to express our appreciation for the extra effort which the staff has contributed.

[Prepared statement of Jack E. Gilleland follows:]

STATEMENT OF JACK E. GILLELAND, ASSISTANT MANAGER OF POWER, TENNESSEE VALLEY AUTHORITY

Mr. Chairman, I am Jack E. Gilleland, Assistant Manager of Power for the Tennessee Valley Authority. We are pleased to appear and make a statement on the fire that occurred at our Browns Ferry Nuclear Plant on March 22, 1975. I will discuss how the fire was ignited and extinguished, shutdown of the reactors, the availability of cooling equipment during the occurrence, and changes in design and administrative controls as a result of the fire.

Events leading up to and including the fire are discussed in the final report of TVA's Preliminary Investigating Committee, dated May 7, 1975. The report, a copy of which has been provided for your use, is included as Section III A in our overall "Plan for Evaluation, Repair, and Return to Service of Browns Ferry Units 1 and 2 (March 22, 1975, Fire). The NRC Office of Inspection and Enforcement has issued a report. TVA filed a response to that report on September 2 pointing out where we agree and disagree. Additional NRC reports remain to be completed.

HOW THE FIRE WAS IGNITED AND EXTINGUISHED

The fire started when an employee using a candle to check for air leaks around electrical cables penetrating the wall between the cable spreading room and the unit 1 reactor building inadvertently ignited some material being used to seal the leak. Candles were a long accepted and highly effective method of detecting and locating leaks. The sealing operation underway was necessary since additional electrical cables added to the penetration had broken the previous seal.

The fire began to propagate due, in part, to the difference in air pressure between the two areas. The cable spreading room has a positive pressure with respect to the reactor building. The men were working in the spreading room. Thus when the fire started, the flow of air into the reactor building caused the fire to spread through the penetration to the cables on the reactor building side of the wall.

The men working in the cable spreading room began immediately to fight the fire, using portable fire extinguishers, and some of the men went to the reactor building to fight the fire.

At 12:35 p.m. the plant fire alarm was sounded. At 12:51 p.m. an interval of 16 minutes the unit 1 reactor was tripped and at 1:00 p.m. the unit 2 reactor was tripped. At about 1:10 p.m. in accordance with previously established arrangements and procedures the Athens, Alabama, Fire Department was called to assist, if needed. They arrived at the site at about 1:45 p.m. The fire in the spreading room was successfully extinguished by 4:30 p.m. using dry chemical fire extinguishers and the installed CO_2 fire protection system. Excessive smoke in the unit 1 reactor building required the use of emergency breathing apparatus. The fire in the reactor building was controlled to some extent by using portable CO_2 and dry chemical extinguishers, but eventually it was necessary to spray water on the burning cables to extinguish the fire, using an installed in-plant fire hose. The fire was out by 7:30 p.m. Most of the fire damage was to cables on the reactor building side of the wall and was confined to a small area of the plant approximately 20 x 40 feet.

Once the fire was extinguished, an orderly approach to verifying actual conditions was undertaken and temporary power was reestablished to necessary equipment. The Plant Operations Review Committee met at 8:45 p.m. on March 22, to discuss plant conditions and reexamine and set priorities for restoration activities to be taken during the night.

The TVA Radiological Emergency Plan was activated on March 22 as a precautionary measure, although no radiological emergency existed. A Division of Power Production emergency control center in Chattanooga was established at 3:10 p.m. This management team participated in all major decisions associated with plant operation and firefighting activities. The Central Emergency Control Center (CECC) was staffed and directed from Chattanooga, Tennessee, beginning at 3:25 p.m. The CECC performed valuable communications with the NRC (Atlanta), the Alabama Department of Public Health, and the Tennessee Department of Public Health. The CECC was closed at 10:30 p.m. on March 22; however, surveillance in the environment around the plant continued until 2:00 a.m., March 23, and the Environs Emergency Staff terminated its activities at 4:15 a.m., March 23, following the restoration of necessary equipment. The Division of Power Production Emergency Center was terminated at 2:45 p.m. on March 23, 1975.

Appendix A describes the sequence of events, describes the fire and equipment used in firefighting, and gives the time of various events for both the cable spreading room and the reactor building.

REACTOR SHUTDOWN

At the time of the fire, units 1 and 2 were each producing approximately 1,100,000 kW. The first indication of the affect of the fire on unit 1 operation came about 20 minutes after the fire started, and consisted of anomalous annunciations of various events and automatic starting of some equipment. Soon after this the unit operator tripped the unit 1 reactor. This results in full insertion of the control rods and shuts down the nuclear reaction.

Shortly after unit 1 was tripped, the unit 2 operator observed decreasing reactor power, alarms, and the loss of some indicating lights. Unit 2 was then tripped.

Thus both units 1 and 2 reactors were shutdown shortly after anomalous events began to occur.

AVAILABILITY OF COOLING EQUIPMENT

It would help to place the events in perspective if we look at the cooling equipment that was available during the occurrence, along with that which was lost. This will demonstrate that the plant was not close to a nuclear disaster.

As you know, following the shutdown of a reactor, radioactive material still present in the reactor fuel continues to generate a significant amount of heat, called decay heat, which must be removed to prevent fuel damage. There are several methods available for removing decay heat.

1. By passing steam to the main condensers, and returning water to the reactor to keep the fuel covered at all times.

2. By closing the main steam line isolation valves and allowing the reactor temperature and pressure to increase, causing relief valves to open and close automatically to maintain a safe, relatively constant reactor pressure. The steam discharge from the relief valves passes through pipes to a large pool of water called a suppression pool. For this type of operation, the reactor water level is automatically maintained above the fuel by means of one or more high pressure reactor water makeup systems.

3. By opening one or more relief valves by remote control to discharge steam to the suppression pool to reduce reactor pressure to a relatively low value. This permits the use of low pressure reactor water makeup systems to maintain the reactor water level above the fuel.

4. It is also possible to remove the decay heat from the reactor when it is at low pressure by pumping the reactor water directly through heat exchangers. With this mode of operation the reactor water temperature can be lowered and maintained below the boiling point indefinitely.

After the reactors were shutdown and the control rods fully inserted, decay heat removal was complicated because the fire in the electrical cables caused a number of pieces of equipment to lose some or all of their capabilities.

The fire damaged the control arrangements for the main steam line isolation valves in unit 1 of the plant and the valves closed and could not be reopened. The decay heat was removed for a time using automatic operation of the relief valves with the reactor remaining at high pressure. However, the fire also affected the two primary high pressure reactor water makeup systems provided for maintaining water level in an emergency. Therefore, the operator chose to depressurize the reactor by remote control of the relief valves and use the low pressure reactor water makeup systems which were still available for safe shutdown of unit 1.

HIGH PRESSURE REACTOR WATER MAKEUP SOURCES FOR UNIT 1

The Browns Ferry design includes a number of provisions for supplying makeup water to the reactor when it is at high pressure. (See figure 1.)

The reactor core isolation cooling (RCIC) system started automatically at the beginning of the fire and was manually shutdown by the operator because the extra water was not needed. The system soon became unavailable; however, the RCIC system could have been made available for manual operation throughout the fire by installing a short piece of pipe allowing steam from the plant's auxiliary boiler to drive the RCIC turbine. A special pipe for that purpose was conveniently stored close by.

A high pressure coolant injection (HPCI) system can supply about 5,000 gallons per minute to the reactor at maximum pressure. The system soon became unavailable due to extensive electrical disruptions.

The control rod drive pumps for units 1 and 2 remained in service throughout the fire. The pump on unit 1 was operating and delivering about 130 gallons of makeup water per minute at full reactor pressure during the fire. By opening a bypass valve, the capacity of one pump for delivery of makeup water could have been increased to about 225 gallons per minute which would have been sufficient to keep the core covered indefinitely. With the unit 1 and unit 2 pumps operating in parallel the total makeup flow to unit 1 could have been increased to at least 300 gallons per minute.

The standby liquid control system for each unit can supply high pressure reactor water makeup. Pumps could have been aligned to provide water to the reactor at approximately 100 gallons per minute. This system could have been made available at any time during the fire by the operator performing a manual valve alignment, actuating two valves, and manually restoring power to the pumps.

The reactor feedwater system was available at the beginning of the fire and continued to operate to maintain reactor water level until the main steam isolation valves closed. The system was not available for high pressure makeup thereafter but was available for low pressure feed to the reactor.

REACTOR DEPRESSURIZATION CAPABILITY FOR UNIT 1

Eleven relief valves and two safety valves are provided to protect the reactor against overpressure. In addition, the relief valves serve the function of depressurizing the reactor so as to allow low pressure sources to inject makeup water into the reactor. All of the relief valves can be operated by remote control from the main control room. Any one of the relief valves can so depressurize the reactor. Remote control of each valve requires that compressed air be available which is supplied by the drywell control air system. Early in the fire, damage to electrical wiring disabled the remote controls for seven of the relief valves. Remote control of the remaining four valves was lost for about three hours at a later time due to fire associated loss of electrical power to the drywell control air system.

A main steam line drain pipe connection to the main condenser is also provided which was opened to help reduce reactor pressure and remove a portion of the decay heat from the reactor. The flow capacity of this pipe connection is small compared to a relief valve, but it could have been utilized during the latter portion of the fire in conjunction with available high pressure reactor water makeup sources. The steam line drain was not available for about the first four hours due to a fire associated loss of electrical power to one of the drain valves which was located inside of the primary containment.

AVAILABILITY OF LOW PRESSURE REACTOR WATER MAKEUP SOURCES FOR UNIT 1

The Browns Ferry design also includes a number of provisions for supplying makeup water to the reactor at low pressure. (See figure 2).

The residual heat removal system consists of four large electric motor-driven pumps and associated heat exchangers. During the fire, damage to valve and pump wiring disabled two of the residual heat removal pumps on unit 1 for an indefinite period. Use of the other two pumps was temporarily lost until power was restored to the valves which were required to line up the system for operation.

Four large electric motor driven core spray pumps are provided. During the fire, damage to valve and pump wiring disabled two of the core spray pumps on unit 1 for an indefinite period. Use of the other two pumps was temporarily lost until power was restored to the valves which were required to line up the system for operation.

Condensate and condensate booster pumps can supply low pressure makeup from the main condensers to the reactor. The arrangement consists of three condensate pumps and three condensate booster pumps. Each condensate booster pump is capable of delivering about 10,000 gallons per minute of low pressure makeup and each pump was available throughout the fire. A condensate pump can deliver makeup water to the reactor if the reactor pressure does not exceed approximately 150 pounds per square inch. For higher reactor pressures, it is necessary to operate a condensate pump and a condensate booster pump in series. This series arrangement can provide makeup flow for reactor pressures up to about 435 pounds per square inch and was the method used by the Browns Ferry operators during the fire.

One other source of low pressure reactor water makeup could have been called upon by using a nonstandard system configuration and manual valve alignment. Two residual heat removal pumps in unit 2 could have been aligned to supply reactor water makeup to the unit 1 reactor through a crosstie pipe between the units. This method of low pressure makeup was not needed during the fire.

As an additional backup safety feature, river water can be pumped directly into the reactor from an inplace service water pump connection. A flow of up to 4500 gallons per minute is possible if the reactor pressure is low. The flow cannot be sustained if the reactor pressure exceeds about 185 pounds per square inch. During the fire, availability of this method of reactor water makeup was limited by the loss of electrical wiring to certain valves. Manual operation of the valves could have been performed if the need had developed.

The availability of the core cooling systems is summarized in Figure 4 and clearly demonstrates that there was indeed no "near nuclear disaster."

AVAILABILITY OF REACTOR WATER MAKEUP SOURCES FOR UNIT 2

Alternate sources of reactor makeup water, similar to those available for unit 1, were also available on unit 2; however, the unit was shut down using the normal system alignments which were designed for use following a main steam line valve closure.

Adequate high pressure reactor water makeup was available at all times to keep the fuel covered during the unit 2 shutdown.

AVAILABILITY OF SUPPRESSION POOL COOLING FOR UNITS 1 AND 2

In addition to its ability to supply water at low pressure to the reactor, the residual heat removal (RHR) system is used as the primary means of extracting heat from the suppression pool. The RHR pumps circulate the suppression pool water through heat exchangers where it is cooled by river water. One of the four RHR pumps per unit and its associated heat exchanger can maintain suppression pool water temperature well below the boiling point. On unit 1, two RHR pumps could have been manually aligned throughout the fire for supression pool cooling. Similarly, two pumps were available for remote alignment on unit 2. RHR suppression pool cooling was established before excessive temperature was reached in the suppression pool.

IMPROVEMENTS IN DESIGN AND ADMINISTRATIVE CONTROLS

The foregoing discussion illustrates that the Browns Ferry plant utilized an in-depth design philosophy with the flexibility and versatility which permitted plant operators to accommodate even the unique circumstances of this fire. Thus the public health and safety was fully protected.

Nevertheless, our studies of the incident show that improvements can be made to substantially reduce the likelihood of such a fire in the future and to mitigate further the consequences of any fire that occurs. These measures consist of changes in administrative controls, design, and fire protection.

CHANGES TO ADMINISTRATIVE CONTROLS AND PROCEDURES

The changes in administrative controls and procedures can be summarized as follows.

1. We have made procedural changes requiring greater in-depth review of all significant activities prior to their authorization. This procedure will ensure that all items of safety significance are recognized and provided for. In addition we require that all significant activities be reviewed by the plant Quality Assurance staff to verify that the activities are supported by approved instructions necessary for their performance.
2. Procedures have been written establishing controls over cutting, welding, and the use of any open flame in operating plants. This procedure requires a physical survey of the areas where the work is to be performed and the establishment of safeguards including additional fire fighting equipment and a fire prior to any work involving open flame or welding.
3. TVA's general fire training program is being given increased emphasis. This includes fire reporting and fire fighting techniques, including the use of water.
4. Procedures have been issued to require surveillance by engineering personnel of all penetrations under repair or construction. These procedures also provide for instruction of craft personnel engaged in penetration sealing and flameproofing, and provide for inspection by trained engineers. Instructions are being issued to require additional surveillance by fire fighting personnel at open penetrations in operating areas.

CHANGES IN DESIGN

Reanalysis of the overall plant fire detection and protection has resulted in design changes in three areas:

1. Browns Ferry has two separate redundant groups, called divisions, of electric power and control circuits for shutdown cooling systems. Increased cable separation and these redundant safety system cables will reduce the probability of fires involving both the divisions. This has involved a redesign of circuits, rerouting of cables and conduits, and installation of additional fire barriers between divisions of cables.
2. Rerouting of selected cables will reduce the probability of events initiated within one unit adversely affecting any other unit.

3. Additional requirements for the control of combustible materials, fire detection, and fire prevention will give early warning of a fire and provide for rapid extinguishing of a fire.

This objective will be accomplished by changes of three types: changes to the fire detection system, changes to the fire protection facilities and changes to cable tray penetration materials.

Additional fire detectors will be provided throughout the plant and along cable trays. These detectors will consist of thermal (heat) detectors and products-of-combustion (ionization) detectors. Annunciation of these detectors in the control room will allow plant operators to know the location of any detectors that have been activated.

The fire protection facilities will be upgraded by the addition of fixed water spray extinguishers along specific cable trays, provisions for portable access ladders and platforms for manual fire fighting, additions of adapters to assure that hose and nozzle connections will be compatible with the equipment used by the local fire department, additional hose connections and racks, and automatic initiation of the fixed water spray system and of the CO_2 system in the cable spreading room.

TVA will install new material in future penetration seals and in penetration seals that are breached during the restoration activities.

CONCLUSIONS

Implementation of the safety objective of safely shutting down the nuclear reactors in the unlikely event of another fire falls naturally into two major areas of endeavor. First, *administrative procedures* have a dual role of eliminating sources of ignition that could arise from personnel actions and then of efficiently fighting any fire that occurred. Second, modifications in the *design* of the electrical systems and in the fire protection systems will further reduce the effects of a fire on necessary shutdown systems. Both the administrative procedures and design are mutually supportive in this effort.

We believe that a desirable balance has been achieved between design features and personnel actions required to protect the plant against the effects of a fire. For a plant already constructed, the design to which we have committed has been optimized. We reaffirm that the Browns Ferry Nuclear Plant, as modified by the design changes summarized above, is safe.

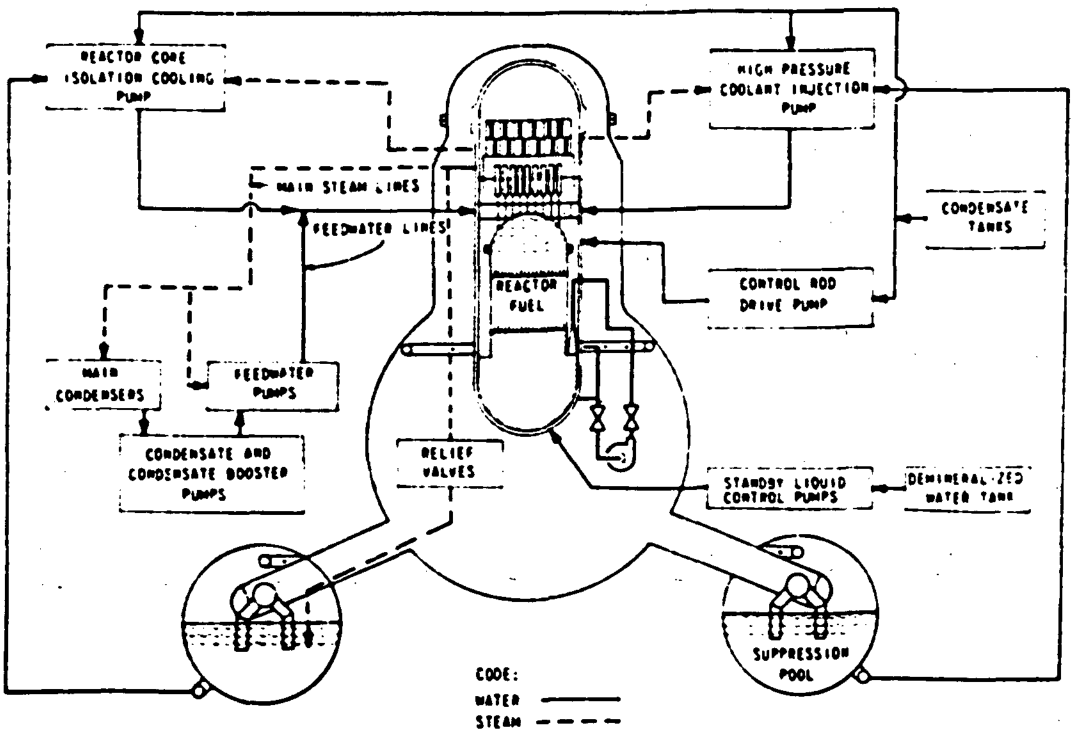


FIGURE 1.—High pressure reactor water makeup sources.

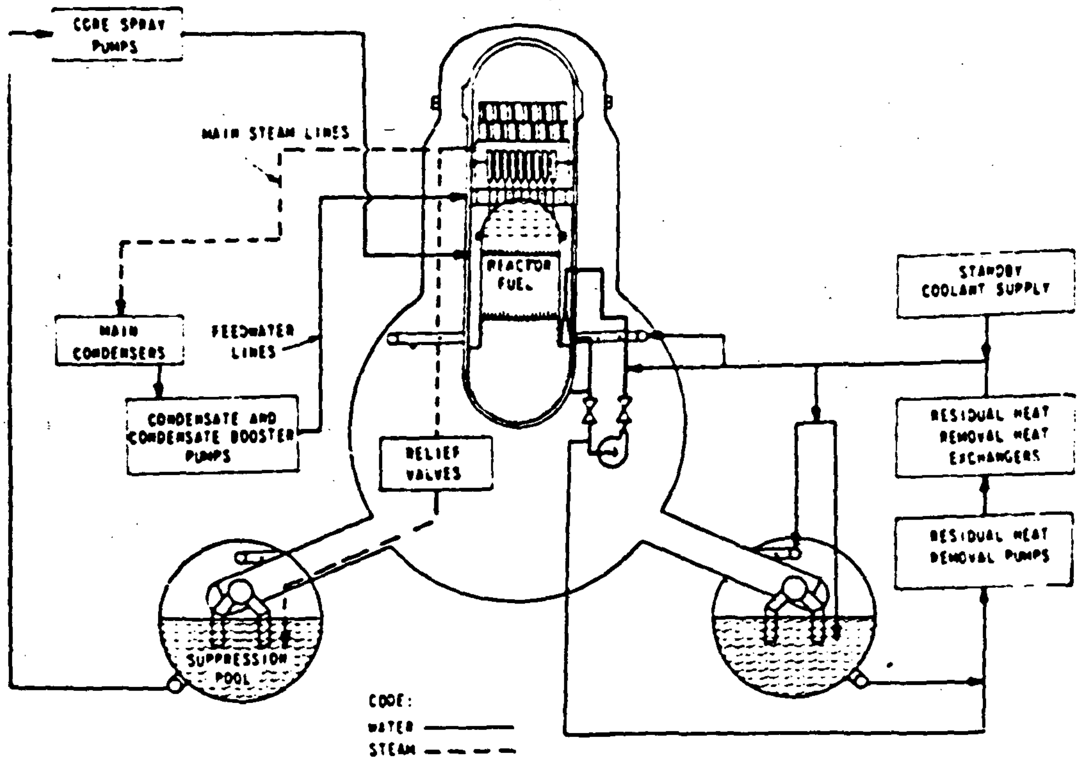


FIGURE 2.—Low pressure reactor water makeup sources.

SUMMARY OF SYSTEM CAPABILITY TO SUPPLY REACTOR WATER MAKEUP DURING THE MARCH 22, 1975, FIRE

	TOTAL NUMBER OF PUMPS IN SYSTEM	NUMBER OF PUMPS AVAILABILITY AFTER 1255 (MAXIMUM/MINIMUM)	CAPACITY PER PUMP (GALLONS PER MINUTE)	MAXIMUM REACTOR PRESSURE FOR SYSTEM OPERATION (POUNDS PER SQUARE INCH)
HIGH PRESSURE SYSTEMS				
REACTOR CORE ISOLATION COOLING	1	1	600	NOT LIMITING
CONTROL ROD DRIVE	3	3	130	NOT LIMITING
STANDBY LIQUID CONTROL	2	2	50	NOT LIMITING
REACTOR FEEDWATER	3	0	10,800	NOT LIMITING
HIGH PRESSURE COOLANT INJECTION	1	0	5,000	NOT LIMITING
LOW PRESSURE SYSTEMS				
RESIDUAL HEAT REMOVAL	4	2/0	10,000	300
CORE SPRAY	4	2/0	3,100	300
CONDENSATE AND CONDENSATE BOOSTER	3	3	10,800	435
CONDENSATE	3	3	10,800	150
STANDBY COOLANT SUPPLY	2	2	4,500	185
RESIDUAL HEAT REMOVAL CROSSTIE	4	2/0	10,000	300

NOTES: AVAILABILITY REQUIRES A NON-STANDARD SYSTEM CONFIGURATION

USING A NON-STANDARD SYSTEM CONFIGURATION, THE CONTROL ROD DRIVE PUMP CAPACITY CAN BE INCREASED TO OVER 200 GALLONS PER MINUTE.

FIGURE 3.—Browns Ferry Nuclear Plant Unit I.

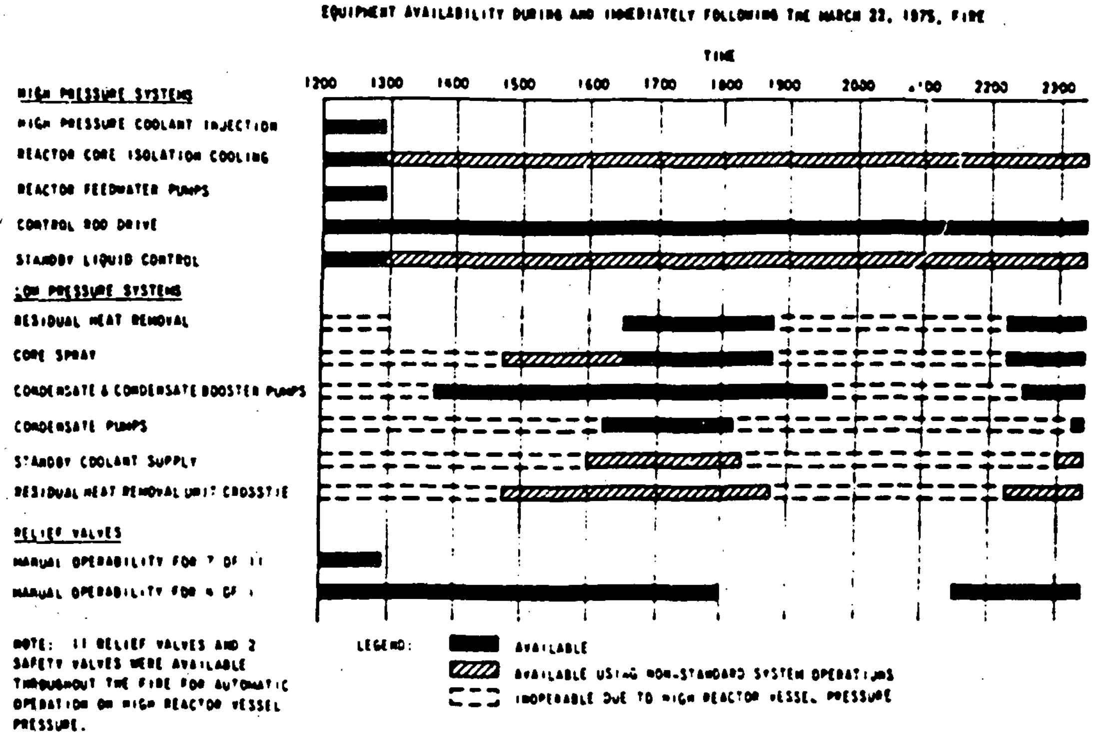


FIGURE 4.—Browns Ferry Nuclear Plant Unit I.

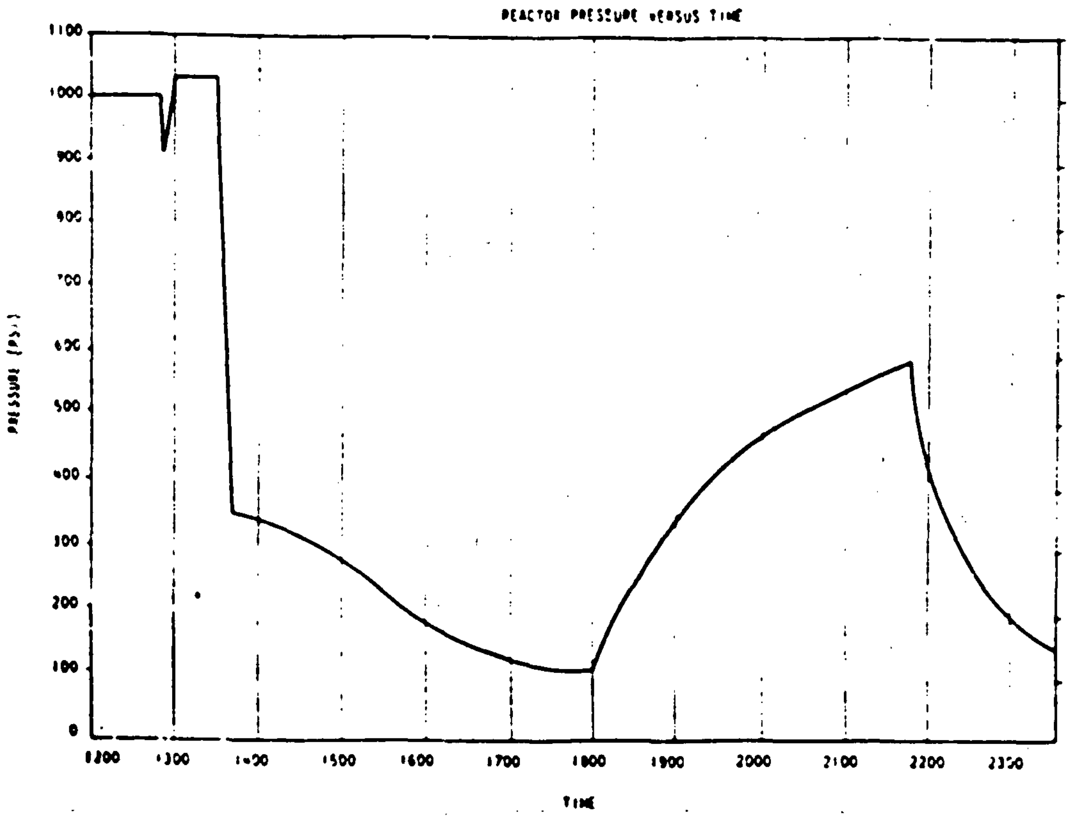


FIGURE 5.—Browns Ferry Nuclear Plant Unit I.

APPENDIX A

[From the Final Report of Preliminary Investigating Committee, May 7, 1975]

D. FIRE

1. Spreading Room Area

a. *Sequence of Events.*—Six men were working in the units 1 and 2 cable spreading room, checking conduit and cable penetrations for air leaks and sealing leaks.

An engineering aide and an electrician were checking cable penetrations through the wall between the spreading room and the unit 1 reactor building, in a window containing 10 cable trays in 2 vertical rows of 5 trays.

The engineering aide was using a candle flame to detect air leaks.

A differential air pressure existed between the spreading room and the reactor building, with the reactor building having a slightly negative pressure and thus causing air to flow from the spreading room through leaks into the reactor building.

The aide detected a strong air leak in the penetration for the second tray from the bottom on the west row.

The leak was caused when additional cables were pulled through the penetration, which resulted in breaching the originally installed air pressure seal and fire stop.

The electrician could not reach the penetration since it was recessed into the wall farther than he could reach.

The aide volunteered to seal the leak for the electrician. The electrician handed the aide two pieces (about 2 inches by 2 inches by 4 inches) of resilient polyurethane foam which the aide inserted into the hole.

After inserting the resilient polyurethane foam into the leak, the aide placed the candle about 1 inch from the resilient polyurethane foam.

The airflow through the leak pulled the candle flame into the resilient polyurethane foam, which sizzled and began to burn.

The aide immediately told the electrician that the candle had started a fire.

The electrician handed the aide a flashlight, which was used to try to beat out the fire with no success.

Another construction worker heard the aide state that there was a fire and gave the aide some rags to use to smother the fire, which was also unsuccessful.

The electrician called for fire extinguishers.

When the rags were pulled away from the penetration, they were smoldering.

Meanwhile, the other worker brought a CO_2 fire extinguisher to the aide.

The fire burned for about 1½ minutes before the first extinguisher arrived.

The entire contents of this CO_2 extinguisher was emptied on the fire. The fire appeared to be out.

About ½ to 1 minute later, the fire started up again.

The aide stated that the fire was now on the reactor building side of the wall.

Two construction workers left the spreading room for the reactor building to fight the fire.

The electrician took two fire extinguishers to the aide who remained in the spreading room. Each extinguisher gave only one good puff.

When the aide received the third extinguisher, he heard a fire extinguisher being discharged on the reactor building side of the wall.

As the aide prepared to discharge the fourth extinguisher, the spreading room CO_2 system alarm was sounded; and all workers evacuated the spreading room.

A plant operator, assistant shift engineer (ASE), after ensuring that no workers were in the spreading room, attempted to initiate the spreading room fixed CO_2 system from outside the west door to the room but was unable to do so because it had been deenergized while workmen were in the spreading room.

The ASE then ran to the east door of the spreading room, where he restored the electrical power and initiated the CO_2 system, which then operated properly.

Another ASE later operated the CO_2 system a second time.

After the CO_2 system had been operated the second time, the first ASE checked the spreading room and found that the fire had restarted.

He then directed the fire brigade in fighting the fire in the spreading room.

At 1310 hours, the ASE in charge of the reactor building fire requested the Athens Fire Department to come to the plant.

Employees from the Athens Fire Department assisted in fighting the spreading room fire.

The spreading room CO_2 system was operated one additional time.

An off-duty shift engineer (SE) arrived about 1500 hours and took charge of firefighting in the spreading room and relieved the ASE.

. The spreading room fire was extinguished between 1600 hours and 1630 hours, primarily by using dry chemicals.

b. *Description of Fire in the Spreading Room.*—The material ignited by the candle flame was resilient polyurethane foam.

Once the foam was ignited, the flame spread very rapidly.

After the first application of the CO_2, the fire had spread through to the reactor building side of the penetration.

Once ignited ,the resilient polyurethane foam splattered as it burned.

After the second extinguisher was applied, there was a roaring sound from the fire and a blowtorch effect due to the airflow through the penetration.

The airflow through the penetration pulled the material from discharging fire extinguishers through the penetration into the reactor building.

Dry chemicals would extinguish flames, but the flame would start back up.

c. *Equipment.*—Portable CO_2 and dry-chemical fire extinguishers were used in the spreading room fire.

The spreading room fixed CO_2 system was activated three times.

Breathing apparatus (air packs) received limited use in the spreading room.

The doors to the spreading room were kept open most of the time to assist in keeping smoke out of the control room.

An inplant fire hose was run from an outlet in the turbine building to the spreading room. This was not used.

The Athens Fire Department made available in the spreading room about 5 gallons of an agent which, when combined with water, forms "light water." This was not used.

Athens Fire Department employees discussed with the SE the possibility of using water on the fire in the spreading room.

No water was used in the spreading room since there was no assurance that the cables were deenergized.

d. *Time of Events* (approximate times shown with ~).—

~1220—Fire started in penetration.
~1230—Two construction workers leave spreading room for reactor building.
1235—Plant fire alarm sounded. Fire logged in SE's log.
~1237—First fire extinguisher discharged in reactor building.
~1240—CO_2 alarm sounded in spreading room: CO_2 system operated.
?—Spreading room CO_2 system operated second time.
?—ASE assumes direction of fire brigade in fighting fire.
?—Spreading room CO_2 system operated third time.
~1500—SE assumes charge of spreading room firefighting.
~1600-1630—Spreading room fire extinguished.

e. *Reporting the Fire.*—Two construction workers left the spreading room at about 1230 hours to go to the reactor building to fight the fire.

One worker stopped at post 8D, a construction portal manned by the Public Safety Service (PSS), and informed the public safety officer on duty that there was a fire in reactor building number 1 and took the fire extinguisher with him to use in fighting the fire.

The officer immediately called the SE and reported a fire in unit 1 reactor building.

The ASE who received the fire report immediately gave the message to the SE and the unit 1 operator and then proceeded to the control room and switched the fire alarm to assure continuous sounding.

The unit operator (UO) immediately began to announce over the PA system that there was a fire in the unit 1 reactor building.

At this time, operators in the control room did not know the exact location of the fire.

An ASE located the fire in the unit 1 reactor building shortly after the construction workers had begun to fight it there. He telephoned the exact location to the operators in the control room.

Shortly thereafter another ASE in the reactor building reported the spreading room fire to the operators in the control room.

2. *Reactor Building Area*

a. *Sequence of Events.*—When workers in the spreading room saw that the fire had spread into the reactor building, two construction workers left the spreading room and proceeded to the reactor building to fight the fire.

One worker told the public safety officer at post 8D that there was a fire in the reactor building and took a fire extinguisher with him. The other construction worker proceeded to the reactor building where he met a third worker; each of the three workers took a fire extinguisher to the fire.

All three workers arrived at the fire at about the same time. It was burning in the trays which were 20 feet above the second floor of the reactor building. One moved a ladder, already at the scene, next to the fire. Another worker climbed the ladder and discharged a dry-chemical extinguisher on the fire. This application knocked down the flames, but the fire flared up again.

One of the workers alerted other workers on the second level of the unit 1 reactor building of the fire.

The worker who applied the first extinguisher was affected by the smoke and fumes around the cable trays at the top of the ladder.

The unit 1 control room operator was informed by telephone of the precise location of the fire by a plant operator on the scene.

An ASE then arrived and, along with another operator, discharged a CO_2 and a dry-chemical extinguisher simultaneously on the fire. The ASE assumed charge of firefighting activities. Construction workers were instructed to leave the operating units.

Smoke was becoming so dense that breathing apparatus was required; approximately 5 minutes after it was requested, it was available. Until it arrived, CO_2 was applied to the cable trays from the floor.

After the breathing apparatus (air packs) arrived, it was utilized in fighting the fire until visibility became so bad that the workers could not get near the fire. The smoke backed them up to the area of the reactor building closed cooling water system heat exchangers.

The ASE left the fire to assist in unit shutdown. An assistant unit operator (AUO) assumed charge of firefighting activities. The first floor of the reactor building was also evacuated. The AUO went to the control room due to some ill effects of the smoke. Another ASE assumed charge of firefighting activities.

Power to the elevator was lost. The second floor of the reactor building was then evacuated. Some time was utilized to check 5 floors of the reactor building for the elevator to ensure that no one was trapped on the elevator. A head count was made, and from that point on a count was kept of all personnel leaving and entering the reactor building.

About 1330 hours, lighting was lost in the reactor building.

Limited firefighting was resumed in the reactor building for a period between 1430 hours and 1500 hours. A wire was used to rig a guideline. At this time the fire was still confined to the area in the cable trays near the north wall and had not proceeded very far on the south trays.

At this time, the doors between units 1 and 2 were opened, which improved visibility on the second level of unit 1 to about 5 feet.

At about 1630 hours, the SE who had been directing activities in the spreading room took charge of firefighting in the reactor building in order to concentrate activities there. The SE consulted the plant superintendent frequently during fighting of the reactor building fire.

On inspection of the fire at 1630 hours, the major fire was in the cable trays running south from the penetration, with a smaller fire in the cable trays running west from the penetration.

The SE established a routine of sending 2 to 3 people in at a time to fight the fire, using dry chemicals primarily.

Shortly after 1630 hours, temporary d.c. lighting was strung on the second level of unit 1.

A rope was utilized as a guideline, which assisted employees from the Athens Fire Department in approaching the fire to inspect it. The SE went into the vicinity of the fire between 1730 hours and 1800 hours.

On one of his trips into the second level, the SE laid out the fire hose installed there and checked to ensure that water was available. The plant superintendent authorized the use of water as an emergency backup, for example, in case a worker's clothing caught fire. Otherwise, there was a decision not to use water on the fire due to the electrical shock hazard. The Athens fire chief suggested that water would be the best thing to use on the fire if it could be used.

The SE suggested to the plant superintendent that water be used on the fire. The superintendent made the decision to allow the Athens Fire Department employees to use water on the fire.

Water was initially applied to the trays running west; however, from the floor level, the water would effectively reach only the bottom tray. Athens Fire Department employees attempted to utilize one of their nozzles on the hose, but the thread did not match; and the nozzle came off when pressure was applied.

Water was also applied to the fire in the cable trays along the north wall and successfully extinguished it.

Firefighters began using Chemox respirators as the supply of compressed air for the air packs ran low.

The SE and two other operations workers entered the area of the fire to utilize water to fight the fire. The SE took the hose and climbed within four feet of the fire with assistance of the other two men. He sprayed water on the fire in the south cable trays for approximately 10 seconds, which extinguished the fire.

The fire hose was left stuck in a position so that it continued to apply water to the south cable trays.

The second level was entered again and water reapplied. It was then determined that the fire was out. There were subsequently some reports of sparks, but investigation failed to reveal any further fire.

During the course of the fire, it was noticed that a small diameter station control air line under about 90 pounds of pressure, running along the north wall, had parted. The line was later isolated.

Several fire extinguishers were discharged early in the fire from the third floor through an opening in the floor but all missed the fire in the cable trays since te opening was not directly over the fire.

b. *Description of Fire in Reactor Building.*—The fire was initially observed in the lower cable trays, extending out from the penetration a distance of 2 to 4 feet. Height of the flames varied from a few inches to a few feet, dying down as extinguishing materials were applied and flaring up between applications. The flames were coming straight up.

Some polyurethane foam was flowing from the penetrations into the trays, and bright yellow flames were coming from the penetrations.

The fire did not advance significantly into the south trays until after 1500 hours.

Scaffold boards had been previously placed below the trays in the unit 1 reactor building, near the cable tray penetration where the fire started. These boards were used to work from in pulling cables through the penetration. These boards were charred by the fire. The charring did not extend to the side away from the fire, indicating little influence as fuel for the fire.

c. *Equipment.*—Portable CO_2 and dry-chemical fire extinguishers were used in the reactor building fire.

MSA air packs were used that had a rating of 30 minutes for moderately heavy activity of the user. A cascade system of large air cylinders was available for charging the packs, but the supply was eventually depleted. There are no air compressor facilities at the plant to fully recharge the air packs. The charges in some air packs did not last 30 minutes. Air packs from Athens Fire Department were also used along with their recharging facilities on their truck and at their station in Athens.

MSA Chemox respirators were used. Several users experienced difficulty when using these for very strenuous activity.

The fire hose and nozzle provided in the second level of the reactor building functioned properly and succesfully extinguished the fire.

A nozzle from the Athens fire truck did not fit the threads on the hose on the second floor of the reactor building.

Ladders present on the second level of the reactor building were utilized.

Temporary d.c. lighting was utilized.

A wire and a rope were utilized as guidelines.

A fire hose was laid out on the third floor of the reactor building but was not utilized.

d. *Time of Events.*—

~1230—Two construction workers leave spreading room for reactor building.
~1237—First fire extinguisher discharged in reactor building.
~1240—Unit operator informed of exact location of fire in reactor building.
?—Air packs requested and received.
~1310—ASE requested that Athens Fire Department come to the plant.
~1330—Lighting lost in reactor building.
~1645—Temporary d.c. lighting installed.
~1835—Water applied to fire.
~1930—Fire determined extinguished.

Representative YOUNG. Thank you, Mr. Gilleland.

I would like to come back to the question I had asked Mr. Wagner and he deferred to you about the flexibility of the regulatory program. I believe that the question that I asked you earlier had to do with regulatory technicalities and specifications, and whether that caused you any problems in connection with this fire?

Mr. GILLELAND. I believe the question you asked was whether we saw any deficiency in the regulations or the procedures.

Representative YOUNG. I will read the question to you again, Mr. Gilleland.

Was there anything in the Browns Ferry license or the technical specifications or any other regulatory requirement which contributed to the problems associated with the fire?

Mr. GILLELAND. No, sir.

Representative YOUNG. The answer is no?

Mr. GILLELAND. The answer is no.

Representative YOUNG. I have several questions here but I am only going to ask two of them.

ACTIVATION OF THE CO_2 SYSTEM

Mr. Gilleland, the Nuclear Regulatory Commission people found that the metal plates installed on the manual switch for the CO_2 stations during the plant construction had never been removed which resulted in some delay in providing the CO_2 to the fire. The question is, how could the TVA fire inspectors have possibly overlooked such an obvious deficiency?

Mr. GILLELAND. Mr. Chairman, I think there is some misunderstanding about how the system operates. The manual system is a backup system only and it is much, much slower to activate than the method that was used. The method that was used is a manual electrical system, and that was the fastest system and the use of it did not slow down or impair in any way the activation of the CO_2 system. Now, the existence of the metal plates was documented and was known by all of our operating personnel.

Representative YOUNG. Your answer then is that the plates were in place and were supposed to be there?

Mr. GILLELAND. Yes, sir, they were there to protect against inadvertent operation because we still had work going on in the cable spreading room. CO_2, of course, is very dangerous to personnel and we were concerned that we could have an inadvertent activation of that in the spreading room. So the plates were installed as a safety device.

Representative YOUNG. NELPIA's internal report [1] of the fire made some rather severe comments regarding TVA's performance, and outstanding among these was that the CO_2 system prevented a catastrophic occurrence.

Would you comment on that? In other words, did we nearly have a catastrophe there? Did CO_2 prevent it or not?

Mr. GILLELAND. I think the question that was answered a while ago was that we did not feel that we were near nuclear disaster in any way. Of course, the CO_2 system was installed to control fire in the cable spreading room and that is its job. It performed and did the job.

Representative YOUNG. Mr. Gilleland, I heard the answer given by Mr. Wagner. This question is being put to you. I take it then that

[1] See Appendix 8.

you concur in the answer of Mr. Wagner that there was no near catastrophe involved in this matter?

Mr. GILLELAND. Yes, sir.

Representative YOUNG. "The congestion in the cable spreading room was inexcusable." That is another observation of the insurance report.

Mr. GILLELAND. The cable spreading room was designed to route cables to the control room for two units. It is the common control room for two units and this, of course, is just a matter of plant design. I do not think we consider that charge to be justified.

Representative YOUNG. You don't think that was inexcusably congested?

Mr. GILLELAND. No, sir, we do not.

Representative YOUNG. The last point raised by the report that was prepared worthy of mentioning here is that minimal consideration was given to the obvious burning characteristics of the cable. If you remember the testimony, it was that the cable insulation was tested but a different type of material was used than the type tested. What is your view on that?

Mr. GILLELAND. May I make a comment?

Representative YOUNG. Indeed.

Mr. GILLELAND. The comment that was made I believe earlier by Dr. Hanauer was that the sealing material that was used in the penetration was different from some of the sealing materials that had been tested. I think the design of the——

Representative YOUNG. Excuse me, Mr. Gilleland. I think the testimony was that it was different than the sealing material that was tested.

Mr. GILLELAND. Yes, that is what I meant to say.

The penetration was designed to have the polyurethane to be used as the sealant and then outside the sealant there was put a fire retardant material called flamemastic. The flamemastic was the fire stop, not the sealant itself. There are a number of different kinds of polyurethane. I think it is true that in this particular penetration there was some polyurethane used that was different than we had tested.

Representative YOUNG. And more inflammable apparently than you had tested.

Mr. GILLELAND. Apparently so, yes.

Representative YOUNG. Obviously using hindsight again it would be important, as I see it, that you not only test the material for its fire resistant characteristics but you in fact use it if it proves satisfactory in that respect. I think it is obvious that testing does not do much good unless you use what you have tested.

Mr. GILLELAND. Yes, except the point I was making is that the fire stop material was the flamemastic which we did use and the polyurethane was not intended to be the fire stop but was intended to be the sealant. What happened, of course, was we tested the penetration before the flamemastic was used to cover the polyurethane and the polyurethane did catch fire and that was a mistake.

Mr. WAGNER. I think the answer is we made a mistake using this material, and we will not do it again.

Representative YOUNG. Which would be correct.

Mr. WAGNER. Yes.

Representative YOUNG. Any further questions?

Senator Baker.

Senator BAKER. Mr. Chairman, I thank you for letting me interrupt a minute ago. I have been to another hearing and I have got to go back to that hearing. I just would like to make one general remark that will take 30 seconds.

Representative YOUNG. Please do.

Senator BAKER. Shortly after this fire I had the opportunity to go down there and examine the Browns Ferry plant. I have had an occasion to read and try to understand the observations made by NRC. I know the TVA will have trouble thinking of it this way because it costs a lot of money, I suppose a lot of embarrassment, and certainly a lot of trouble. But really in a way we have learned a lot from this accident and I am pleased with the way TVA has "bellied up to the bar."

There were admitted design errors or mistakes and now they propose to change them in the future. Now, they have dealt honestly and I believe candidly with the situation as it presented itself to the operators and the management. We are mortal and we hope to be mortal for some time. You are going to have mistakes and unanticipated events. I think the rather extraordinary event is that the system worked and it worked in a regime and under circumstances that really it was not designed to work under. I think there is lots to be learned from it, but I think that it will be extraordinarily useful to us in the design and advocation of future nuclear plants. I want to commend management and NRC for treating this realistically.

Mr. Chairman, with that I have to leave. I thank you for letting me interrupt. I thank the TVA witnesses for answering the questions.

Representative YOUNG. Senator, we are most happy to have you here and we appreciate your contribution.

I would not argue with my friend and colleague at all but of course the problem that we have here is that we have a rather substantial segment of our citizenry that expects us not to make any mistakes at any time in any respect with anything nuclear, and indeed we should have that as our goal. But when we do have a mistake—and I commend these gentlemen for admitting to any errors that they might have made—we just have to delve deeply into these things in the hope that again we will be able to employ that infallible science of hindsight and certainly not have anything happen such as we have had happen here with materials and so forth.

Senator BAKER. I make just an interesting commentary that the folks at the TVA service area are sure hoping they will get that plant back in service in a hurry because it has had a direct impact on the cost of electricity.

Representative YOUNG. In that connection I might ask how long will it take to get the plant back into operation?

Mr. WAGNER. We hope it will be in service next January.

Representative YOUNG. We had some testimony that it has not affected the service in the area.

Mr. WAGNER. Not the service, the cost of the service. We are providing the needs from other sources and we have had to purchase considerable amounts of power from outside but it has been at a cost considerably above the cost of generating in those plants. When the plants come back on, the rates will come down in the Tennessee Valley.

Representative YOUNG. I think that would have some effect on the service.

Mr. WAGNER. I misunderstood your question, Mr. Chairman.

Representative YOUNG. I am glad that was clarified.

Thank you.

Any further questions?

Mr. Murphy has a question here, please.

Mr. MURPHY. Thank you, Mr. Chairman.

Mr. Gilleland had mentioned that the metal plate was put on for safety reasons and it seems a little bit at variance with the NRC. All I wanted to do was read a very short two sentences and something from NRC and then ask a question.

In referring to what happened at the precise time at the beginning of the fire the statement is made on page I-6 of the NRC report that an assistant shift engineer attempted to initiate the system at the Unit 1 spreading room control station without success. He found that the power had been shut off at the disabling switch at the Unit 2 entrance so he then went around to the Unit 2 door and turned the power on at this station. The automatic initiation did not appear to be successful and he next attempted to use the manual crank system but found that a metal plate had been installed under the break-out glass. He stated that he later determined the metal plate had been installed on the manual station during the plant construction and that almost all were still installed during the day after the fire.

Continuing on with the same report under the heading of other important factors not directly contributing to the cause of the fire, with regard to its severity or its consequences, the statement is made that manual operation of the installed carbon dioxide system in the cable spreading room was precluded because metal plates installed behind the breakout glass during construction had not been removed. The same statement is made by NRC in a press release dated July 30 which I just read.

My question really is this: NRC seems to feel that the presence of the metal plates had some implications and you feel just the other way. What is your comment?

Mr. GILLELAND. I think the explanation came out in further investigations to determine exactly what happened. The storage system for the CO_2 is some distance from the spreading room because it serves several areas of the plant. It is true that there are two push buttons to activate the system electrically and on one side of the spreading room at one door is the switch. The switch and push button are right there side by side.

Now, the first attempt was at the push button which was not at the same location as the switch, but it was a very short distance to run around to the other side, turn on the switch and push the button. Now, there was a travel time from the central storage of the carbon dioxide to the spreading room. The first impression of the operator was that the system had not operated and before the CO_2 arrived he noticed the metal plate which covered the manual back-up control. Before he could have taken any further actions the carbon dioxide arrived in the spreading room.

So what I am saying is that this system was actually operated electrically. While the operator's first impression was it had not operated, in fact it had and the manual system would have been much, much slower because he has to open two valves, one at the spreading room, and then he would have to travel about, I guess, 10 to 15 minutes to open the other valve which is at the central storage spot in the CO_2.

Mr. MURPHY. So your contention is that in this crucial time it took about 3 minutes to do this; is that it?
Mr. GILLELAND. Yes.
Mr. MURPHY. Instead of the 10 to 15?
Mr. GILLELAND. Yes.
Mr. MURPHY. And what you are essentially saying is that the NRC report in this case is in error?
Mr. GILLELAND. Well, it was because——
Mr. MURPHY. Because the supervisor——
Mr. GILLELAND. Yes.

USE OF WATER IN FIREFIGHTING

Mr. MURPHY. Mr. Chairman, one other question to this plant supervisor.

You very ably described your decision not to use water. You mentioned that equipment cut out that you didn't expect to cut out as you indicate.

The question is, what was it that you were concerned about? What was it you thought might happen if you used water right at that time before the situation softened up a little? What did you think was the worse consequence?

Mr. GREEN. I don't know that but at that time I had put it in a reference that a specific thing could happen if we used water. We had had several functions disappear and then come back which it was not clear why and which of the corrective actions we had taken were the ones that had restored that.

I want to put it in the context of knowing where the fire was. I can't at all say it was out. It was controllable and this weighed against just this unknown that might happen if you started shorting the whole mess of wires out with water.

Mr. MURPHY. With water?

Mr. GREEN. Yes.

Mr. MURPHY. Thank you.

Thank you, Mr. Chairman.

Representative YOUNG. I would presume that the basic instinct is not to use water on an electric fire.

Mr. GREEN. TVA permits the use of water. Our procedures put it in the second place. It says on electrical fires it may be used if other attempts fail. It is true that the fire hoses, for example, which we install in the plants have a nozzle on them of a type that is used on electrical fires. It is not an insulated type, it is a spray or fog type that can be used on higher voltage circuits, and for the most part our first was in very low voltage circuits.

Representative YOUNG. Thank you very much.

Any further questions?

That is all then from the TVA Chairman and staff. We appreciate very much you gentlemen being here today and we appreciate your testimony.

Mr. WAGNER. Mr. Chairman, we appreciate your questioning. I want you to know we have learned something from this. My colleagues remind me that there were questions about fire drills. We have learned to conduct more fire drills. We have been conducting them regularly. I think Mr. Green might like to answer as to fire drills.

Mr. GREEN. The fire drill previous was conducted on August 14, sir.

Representative YOUNG. Thank you very much, gentlemen.
Mr. WAGNER. Thank you.
(Additional questions asked of TVA in writing and their answers are provided in Appendix 10.)
Representative YOUNG. We will now hear from our last witness for the afternoon, Mr. Aubrey V. Godwin, Director of the Division of Radiological Health, Alabama Department of Public Health.

STATEMENT OF AUBREY V. GODWIN, DIRECTOR, DIVISION OF RADIOLOGICAL HEALTH, ALABAMA DEPARTMENT OF PUBLIC HEALTH

Representative YOUNG. If you will follow the suggestion I made previously. Because of the obvious limitation of time, I wonder if you could not summarize your statement and allow us without objection to put the whole statement in the record in this proceeding.
Mr. GODWIN. Yes, sir, Mr. Chairman.
Before I do that there is one correction to be made to the statement. On page 4, the top line, after the period it should read "This also supports our philosophy that the licensee should initially call the"—and it picks up with "most expert in the problem at hand."
Representative YOUNG. Yes, sir. That will be reflected in the record.
Mr. GODWIN. Mr. Chairman and members of the committee, I thank you for giving me this opportunity to participate in this hearing on the Browns Ferry nuclear plant fire. Because of the concern of the Alabama Department of Public Health for matters affecting the health of our people, we feel that we should meet with this committee and provide information regarding our response to this fire so as to aid the committee in its deliberations.
I would like to express the philosophy used in developing the Alabama radiation emergency plan. First, the operator/licensee is to contact the group with the most expertise in the immediate problem. In other words, in case of fire, call the fire department; if radiation is the problem, call the division of radiological health.
Second, the plan sets up an organization which can respond to the emergency as it develops. The organization is flexible and redundant. It goes without saying that inability to contact one individual or organization should not prevent the taking of proper protective actions.
Third, we have provided by memorandum of understanding a mechanism whereby the various agencies that expend monies during the emergency in aiding the public may be reimbursed for those expenditures. This covers only evacuation associated costs, animal feed costs and agency costs above normal operation expenses. We leave to each individual affected to determine and collect monies for losses of property, and use thereof, and such associated costs.
As I have indicated in my prepared remarks, I have included a copy of our Division of Radiological Health's Report to the Alabama Radiation Advisory Board of Health. With your permission, Mr. Chairman, I would like that also included.
Representative YOUNG. Without objection, it will be made a part of the record.
[Material referred to follows.]

ATTACHMENT A

State of Alabama
Department of Public Health
State Office Building
Montgomery, Alabama 36104

IRA L. MYERS, M. D.
STATE HEALTH OFFICER

March 31, 1975

MEMORANDUM

TO: Radiation Advisory Board of Health

FROM: Aubrey V. Godwin, Director AVG
Division of Radiological Health

SUBJECT: March 22, 1975, Fire at Browns Ferry Nuclear Plant

At approximately 12:35p.m., March 22, 1975, a worker using a candle to check for air leakage into the reactor building ignited some polyurethane foam in the Unit 1 spreader room. Over the next 7 hours, this fire burned at least 6 cable trays primarily in Unit 1 reactor building. As these cables were burning, the control room lost the ability to operate several safety functions. Attached is a summary of the incident and a log of my actions. This incident is the first that actually activated the Radiation Emergency Plan involving other state agencies and including notification of Governor Wallace. We did not activate the Southern Radiological Emergency Plan.

The fire did cause the loss of some safety systems, but at all times there was adequate core cooling. In fact, there was at least one backup also available. There has been some speculation about a core meltdown. The core was properly cooled at all times, and even if all the core cooling makeup had been lost we understand it would have taken some 20 hours for melting to occur due to decay heat. There was no breach of the vessel or main steamline.

Some continuous radiation monitoring was lost. The available data did not indicate a significant release; however, we did start our air sampling network around the plant. As you review the attached information, you will note that the downwind sampler was broken. The Division has no standby samplers and when the sampler broke some 3 months ago, there was no replacement. Each air sampler costs approximately $1000.00. We are reevaluating our program in hopes of finding a cheaper type which would be within our budget.

In retrospect we would have liked to have had a continuous readout monitor in operation. Ideally, we could have received reports every 15 minutes from the unit and confirmed the situation immediately instead of having to wait some 3 weeks to get the data as we are now having to do. The equipment which will do this ideal costs $25,000.00 per station.

The same equipment may be set up so it does not provide for readout in Montgomery, but could be read by someone at the station. Cost for this is approximately $10,000.00 per station. Our current budget does not provide for any of this equipment. The proposed fee bill would allow the purchase of this equipment in time. This would be from the fees derived from the reactor operators.

Also enclosed is a copy of the March 24, 1975 Press Conference.

AVG/dm

Enclosure

Brown's Ferry Nuclear Plant Fire of

March 22, 1975

The following is a reconstruction of the events occuring during and after the fire. Many of the times are approximate and the entire reconstruction is based on information available on March 28, 1975. For actual times and scope of events, the individual logs should be consulted. All time are Central Daylight Time, 24 hour clock.

Approximate Time	
1230	BFNP Unit 1, base load 1100 Mw(e). BFNP Unit 2, base load 1104 Mw(e).
1235	Worker using a candle to check for air leakage ignites polyurethane sealant in Unit 1 spreader room below control room. Worker attempted to put out fire.
1247	Fire alarm sounded.
1251	Unit 1 scramed on operator initiated signal in control room. Scram was normal.
1300	Unit 2 scramed on operator initiated signal in control room. Scram normal Athens Fire Department called.
1303	Initial loss of ability to operate motor operated valves, otherwise both units were being brought to a cold shutdown condition normally.

During the next 6.5 hours, the control room lost the ability to operate or monitor at least the following systems of Unit 1:

1. Residual Heat Removal.
2. High Pressure Injection.
3. Low Pressure Injection.
4. Relief valves (note the overpressure portion of the system still worked).
5. Reactor building radiation continuous air monitors.
6. Reactor building radiation vent monitors.
7. Dry well radiation monitors.

Unit 1 was being cooled by one of three condenser booster pumps. An additional method of cooling was the river water from the raw water supply.

The control room lost the ability to monitor Unit 2 reactor building and its vents. Manual monitoring was instituted to determine radiation levels and potential. Additionally, some redundant cooling capacity was lost.

1500	An emergency possibility involving radiation was declared and the TVA Central Emergency Control Center, CECC, was activated.
1520	State of Alabama informed of this incident by Mr. Belvin. (See log of A.V. Godwin).

NRC answering service notified of the incident.

1700 State of Alabama started air samplers.

1735 EPA Laboratory notified of possible request for assistance.

1945 Fire out.

2020 State of Tennessee notified.

2030 Relief valve of Unit 1 opened from control room.

2100 Unit 1 now being cooled through the control rod makeup system.

2120 Governor Wallace notified by Dr. Hyers.

2400 NRC investigating team arrived on site.

<u>March 23, 1975</u>

0400 Unit 1 RHR system reestablished.

<u>March 24, 1975</u>

Arranged for water sample collection.

<u>March 25, 1975</u>

Arranged for milk sample collection and the changing and reading of TLD's. Received air samples.

<u>March 26, 1975</u>

Picked up the water samples.

<u>March 27, 1975</u>

Tried to pick up milk samples. The Dairyman did not keep the sample as requested. He will mail in a March 27, 1975 sample.

Personal Log Aubrey V. Godwin
Brown's Ferry Nuclear Plant Fire

March 22, 1975 Prepared March 28, 1975

This log has been prepared from notes taken during and after the incident. The times are Central Daylight unless otherwise indicated and are within 5 minutes of actual time.

Time (24 Hour Clock)	Action or Event
1520	Received telephone message from Mr. L. Belvin, TVA, Muscle Shoals He informed me that Browns Ferry Nuclear Plant had a fire in the spreader room, both units had been scramed and that monitoring was available for water release. A reading of 1.03 X 10^{-9} uCi/cc unidentified activity in air was given; however I did not record the location of the sample. In Mr. Belvin's opinion, there were no unusual levels of radioactivity outside the plant. Mr. Belvin did indicate that reactor building monitoring was out, and some vent monitoring was out. I then asked Mr. Belvin a question which would confirm that this was a valid call. My conclusion was that a serious situation existed at Browns Ferry Nuclear Plant and the State Emergency Plan should be activated even though radiation did not appear to be a problem. Also, we should consider starting our air samplers.
1530	I informed Mr. W.T. Willis of the above and he asked me to inform Dr. Myers.
1540	Dr. Myers not available according to Mrs. Myers.
1543	So informed Mr. Willis. He concurred in my conclusion and asked I follow through.
1545	Notified Mr. Sam Maple, State Civil Defense Office of this incident and our conclusions.
1555	Notified Dr. Betty Vaughn, Tri-County Health Officer, of this incident and our conclusions.
1558	Notified Dr. John Regnier, Environmental Health Laboratory Director, of this incident and our conclusions. He suggested we reconsider the situation prior to the actual start of the air samplers.
1605	Recontacted Mr. Belvin who suggested that more up to date information was available from Dr. J.Oppold at TVA, CECC, (Central Emergency Control Center).
1610	Contacted Dr. Oppold who indicated: 1. Both reactors scramed 13:00 CDT.

2. Reactor radiation monitors out.
3. Vent monitors operating. He qualified this by saying he could not confirm this.
4. Fire was still burning.
5. CO_2 had been used.
6. Core cooling was continuing but not on primary system.
7. Fire was below control room in spreader room.
8. Athens Fire Department on site.
9. NRC answering service notified. Call terminated to allow Dr. Oppold to perform duties.

My conclusions were that the situation was more serious than first evaluation, and it was imperative to start air monitoring.

1620 Mr. S. Maple called and reported that he had notified the Morgan Civil Defense Deputy Director. When informed that the Athens Fire Department was at the plant, Mr. Maple commented that that was where the Limestone people must be. Mr. Maple indicated that he had gone to the State Civil Defense Office to make his calls. I indicated this was a serious incident, but radiation levels at that time did not warrant any Civil Defense action. We agreed if the movement of people was necessary, then law enforcement personnel would be contacted.

1630 Recontacted Dr. J. Regnier. Indicated my conclusions, and he agreed to start all air samplers.

1645 Recontacted Mr. Willis and advised of the current status. He agreed with my conclusions and agreed we should notify Governor Wallace. Mr. Willis suggested that since no action would be required by the Governor, we should not call directly, but contact Mr. W.A. Jackson or J. Karrh of the Governor's Staff.

Note Mr. Jackson and Karrh were not available and attempts were made to contact them until 2115.

1715 Recontacted Dr. Oppold who indicated:

1. Unit 2 in normal shut-down process.
2. Unit 1 has the fire problem.
3. Fire still burning but under control. Still using powder chemicals and trying not to use water to put out the fire.
4. No injuries reported.
5. No entrance had been made into spreader room.
6. Five engine in building.
7. Stack monitors low (normal).
8. Reactor building vent monitors no reading, probably out.
9. Unit 1, 200 psi gauge.
10. Mr. Murphy, NRC had been informed.
11. They were having trouble contacting me.

On questioning of ability to maintain cooling, Dr. Oppold referred me to a Mr. Coffee. Mr. Coffee indicated for Unit 1:

1. Cooling was being maintained by the condenser booster pumps.
2. The relief valves from the control room was lost.
3. The ability to operate the motor operated valves from the control room was lost.

4. HPCI system was lost, (High Pressure Coolant Injection).
5. RHR system was lost, (Residual Heat Removal).
6. Parts of ECCS lost, (Emergency Core Cooling System).
7. Another source of cooling was available; i.e., river water.
8. Torus was 126° F with no leakage indicated.

My conclusions after reviewing notes were:

1. Cooling systems degraded and must be watched.
2. Some leakage may be unmonitored.
3. Reconfirmed serious situation.
4. Need more radiation data, confirm monitoring ability.
5. Need to confirm scram closed MSLVs, (Main Streamline Valves).
6. There was no need to activate the Southern Radiological Emergency Plan.

1745 Dr. Regnier reported that air samplers had been started at Athens WTP, Athens STP, Hillsboro, and Rogersville. Further, the sampler in Decatur was broken, perhaps in the wind controller section. He had asked them to remove the wind controller and start sampling.

Note: Attempted to contact NRC twice after 1715 and before 2130.

1850 Dr. G..old contacted me and indicated:

1. Environmental radiation measurements around the plant were essentially background.
2. Measurements at the gate-house were essentially background.
3. Measurement at the site boundary was essentially background.
4. Wind was from the northwest at 5 m.p.h. under Pasqual class A mixing.
5. Fire condition had not changed.
6. Gross activity in Unit 2 building was above MPC at 3.56×10^{-7} uCi/cc. Unit 1 was 7.5×10^{-8} uCi/cc.

I suggested that the State of Tennessee should be informed.

Mr. Coffee indicated:

1. MSLV closed on both units.
2. At time of scram, Unit 1 was at 1100 MW(e) and Unit 2, 1104 MW(e).
3. Unit 1 scramed 1254.

1900 Updated Mr. Willis. Conclusions were situation still serious, but radiation was not a significant problem.

1915 Updated Dr. Regnier. Discussed possibility of using Air Pollution Control Commission air sampling equipment.

1925 Called Mr. J. Cooper, Director, Air Pollution Control Commission, about the possibility of using air sampling equipment.

1930 Contacted Dr. Betty Vaughn to inform of latest information.

1940 Contacted Mr. R.L. Woodruff of the Division of Radiological Health. Informed him of status of incident.

1950 Dr. Regnier called and reported that there was no air sampler in Decatur.

1955 Contacted Mr. J. Cooper and asked if an air sampler could be started in Decatur. Mr. Cooper indicated he would attempt to start a sampler that night, but could start one on Sunday in any case.

2045 Mr. B. Graham, Tennessee Department of Public Health, called. We compared notes. The only deviation was that the fire was out. General conclusion situation serious.

2105 Dr. Ira L. Myers, State Health Officer, was informed of the incident with Mr. Willis' and my conclusions and with the specific recommendation that Governor Wallace be informed of this incident.

2115 Called CECC. Talked to Hodges. He indicated:

1. The fire was out at 1945.
2. Water fog was used to put it out.

Dr. Oopold came on the phone and reported that 6 more environmental monitoring samples (air) had been collected and analyzed. The highest result was 8.5×10^{-10} uCi/cc. Dr.Uppold did not know the location. Dr. Oppold further reported:

1. Continuous air monitoring equipment that was operational was indicating a continuous dropping of radioactivity.
2. Vent monitoring was out, but was being done manually, preliminary results indicated no abnormal levels.

Mr. Coffee came on phone and indicated:

1. That Unit 1 relief valve was operating again and pressure was dropping. The unit 1 relief valve lost operating air and vessel pressure built up to 500 psi.
2. Weeks would be required for decay heat removal.
3. Cooling now utilized the control rod makeup water system.
4. Six trays were involved in the fire.

My conclusion was that the situation was still serious, but improving. An additional cooling system was available.

2145 Advised Dr. Myers of the current status including the time cooling would be required. Dr. Myers indicated he had talked to Governor Wallace, who had expressed three areas of concern.

1. Was additional state resources, particularly the National Guard?
2. Was electrical power available in north Alabama?
3. Was sabotage involved?

Dr. Myers apparently indicated in reply:

1. No
2. Yes
3. We had no indication of the cause of the fire.

After Last Entry	Mr. N. Moseley NRC Regional Director, called and indicated:

1. An investigation team would be on site around midnight.
2. The situation was serious, but seemed to be improving.
3. Possibility as many as 10 cable trays involved.
4. Long repair time.
5. Problem mostly in Unit 1.

March 23, 1975

1040 Dr. Uppold called and gave status report:

1. Unit 1 0 psi with respect to Torus, Torus 280° F., 60" water in reactor.
2. Unit 2 0 psi with respect to Torus, Torus 170° F., 60" water in reactor.
3. Unit 1 on normal RHR shutdown cooling.
4. TVA vent monitoring for 3-22-75 results were::

1645	3.2×10^{-4} or 10^4	uCi/cc uCi/sec.	Gross Activity
1810	1.7×10^{-5} 1.7×10^3	uCi/cc or uCi/sec.	Gross Activity
1920	5.8×10^{-5} 5.8×10^3	uCi/cc or uCi/sec.	Gross Activity
1940	$1. \times 10^{-4}$ 7×10^3	uCi/cc or uCi/sec.	Gross Activity

Only identified isotope Rb^{88}
Calculated fenceline dose 1.8 mRem total.

Total estimated dose assuming highest discharge rate and highest concentration, all as Gross activity

17.2 mRem or 8.6 mRem/unit. This Dose was not actually received.

Technical specification gas 1.3×10^3 uCi/sec.
Technical specification particulate 3.3×10^5 uCi/sec.

5. Unit 2 building air now $<10^{-9}$ uCi/cc.
   Unit 1 building air ok.

6. Major problem at one time (2300) was 500 ppm CO.
7. Probable cause worker using a flame to test leakage ignited polyurethane.

Afternoon Informed Mr. Willis of current status.

March 24, 1975

0755	Called Mr. Long, NRC Atlanta, to review general status. Mr. Long indicated an investigation team from Washington would go to BFNP today. Basically, we agreed on seriousness of incident.
0830	Mr. J. Lyon, TVA, called and indicated that a press conference would be held at 1300.
0835	Called Mr. James, Tri-County Health Department, and informed him of press conference. Mr. James indicated that Mr. P.E. Saywell would attend.
0920	Briefed Dr. Myers; Mr. Willis; Mr. Joe Downey, Water Division; Mr. G.R. Wright, Inspection Division; and Mr. H. Henderson, Assistant to Dr. Myers. Agreed to follow situation and to collect a Muscle Shoals raw water sample, a Wheeler dam water sample, and a milk sample.
1030	Mr. R. Woole, Tennessee Department of Health was called. Informed him of press conference and current status. Mr. Wolle offered any resources that they could make available.
	Note: Tennessee is a member of the Southern Radiological Emergency Council.
Morning	Requested Dr. Regnier arrange for me to collect a milk sample of 3-24-75 milk. I would pick up 3-26 or 3-27.
1330	Attempted to contact Mr. E. Belvin. Mr. Hodges only available.
1530	Mr. P.E. Saywell called and indicated that the press conference was "rough". More than expected attended. Prime impression Athens Fire Department provided needed manpower.
1545	Mr. E. Belvin called and indicated: 1. Fire discovered or started 1235. 2. Fire alarm given 1247. 3. Unit 1 scram 1251. 4. Unit 2 scram 1300. 5. Athens F.D. Notified 1300. 6. CECC activated 1500.

March 25, 1975

0905	Cont[illegible]ted by Ms. M. Riley, Penn. Wanted to know if: 1. ECCS needed. 2. ECCS failed. 3. What happened.

I gave details.

Morning — I asked Dr. Regnier to arrange for me to pick up a milk sample.

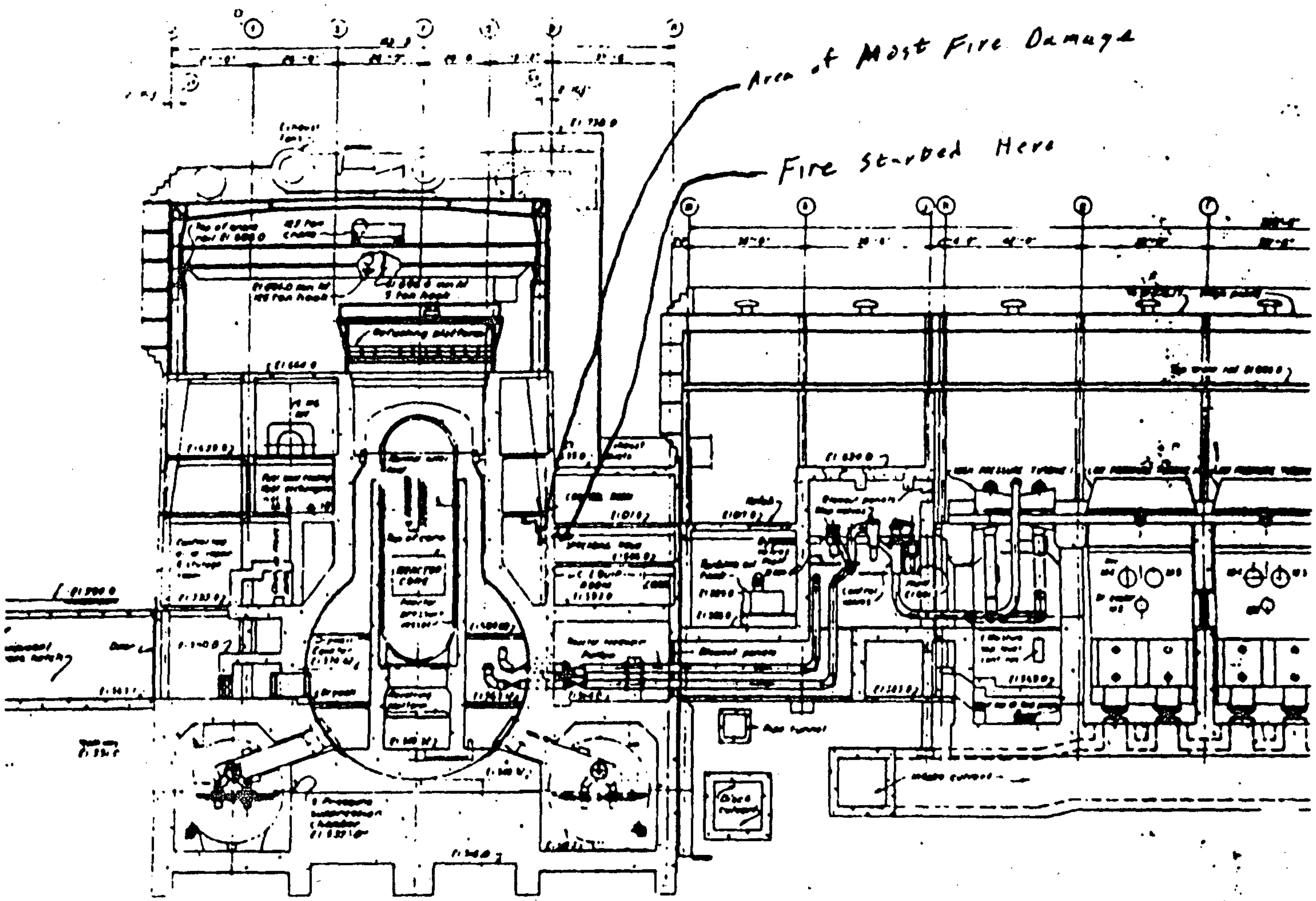
1000 — Called J. Sutherland, NRC. Reviewed our evaluation, also pointed out AEC's publication "Serious Accidents" #336 issued September 19, 1974. Referred to foam fires.

1505 — Mr. J.W. Hufham, NRC , called asked for an item by item review of our actions during the incident. Also asked for long range plans.

I

1. Gave orally my notes.
2. Indicated which samples we would run, (milk, air, and water).

1640 — Called Mr. Hufham at home informed him we will also going to have Eberline read our environmental TLD's as soon as replacement units are in.



SECTION A-A & A1-A1

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
230 PEACHTREE STREET, N. W. SUITE 818
ATLANTA, GEORGIA 30303

MAR 31 1975

No. 75-13
Contact: Ken Clark
Telephone: 404/526-4503

FOR IMMEDIATE RELEASE
(THURSDAY, MARCH 27, 1975)

NRC STATEMENT ON BROWNS FERRY FIRE

The Nuclear Regulatory Commission has received preliminary information from its inspectors who are investigating the cause and implications of the March 22 fire in electrical cabling at Tennessee Valley Authority's Browns Ferry Nuclear Station in Alabama.

The NRC's Executive Director for Operations has reported to the Commission that:

—Thus far, the investigation indicates that the fire resulted in considerable local damage to electrical cables. The reactor core, coolant piping and important structures were not damaged.

—There was no release of radioactivity and there were no injuries.

—The functioning of some in-plant operating and safety systems, including emergency core cooling systems, was impaired due to damage to the cables.

—The two reactors were safely shut down and cooled during the fire. NRC inspectors report that there was redundant cooling equipment available during the reactor cooldown.

—The staff has issued an advisory bulletin relating to construction and safety practices at other nuclear power plants. At present there are no indications that additional immediate action is required for other plants.

NRC inspectors arrived at the Browns Ferry site about 1 a.m. Sunday, March 23, to assure that the two reactors, shut down when the fire started, are maintained in a safe condition and to begin the NRC investigation. In addition, the Commission's inspectors are closely following TVA's own investigation into the fire.

Concurrently, a special group of Commission technical experts, named last Sunday and headed by Dr. Stephen Hanauer of the NRC staff, already has begun a review of the circumstances and implications of the fire. This group is charged with reviewing the circumstances of the incident, and evaluating its origins and consequences. This group will examine technical considerations such as the design criteria of the affected equipment, its materials, and how it was installed and maintained, and whether any modifications of NRC policies, procedures or technical requirements are indicated.

The NRC teams are being assisted by fire experts from the Factory Mutual Research Corporation and the National Aeronautics and Space Administration.

On March 24 the NRC's Office of Inspection and Enforcement advised licensees building nuclear power plants at sites where there also are operating plants to review their overall systems for control of construction activities.

In the advisory bulletin, plant operators were instructed to specifically review the design, installation and testing of seals used where cables penetrate walls between compartments, and to evaluate procedures for the control of ignition sources in areas containing flammable materials.

The licensees are instructed to report to the NRC the results of the review and evaluation. Other licensees operating nuclear power plants also were provided with the information in the bulletin.

TVA reported that the fire was started by a candle which was being used to check for possible air flow through a seal where the cables pass from one compartment to another.

Detailed information on the cause of the fire and the operability of the various reactor systems during the fire is being developed by NRC inspectors. However, information developed thus far indicates:

(1) Although some instrumentation was lost, certain critical instrumentation such as reactor water level, temperature and pressure indicators continued to function and both plants were safely shut down.

(2) The nuclear fuel remained covered by cooling water at all times.

(3) Control of some systems normally used for cooling down the reactors was impaired or lost due to the fire, and alternate methods of cooling, principally the control rod drive pumps and the redundant condensate booster pumps were used. The reactors were depressurized.

(4) On Unit 1, although a loss-of-coolant accident had not occurred, the emergency core cooling system was activated and supplied additional water to the reactor. It was manually shut down to prevent overfilling. Later, during cooldown, when ECCS was called for manually as one of several alternate means of supplying cooling water, it did not activate; the alternate methods had more than sufficient capability to cool the core. The ECCS behavior is under specific investigation.

(5) On Unit 2, although four pumps in the low-pressure emergency core cooling system were inoperable, adequate cooling from ECCS was available since redundant pumps were functional. Emergency core cooling water was provided initially as in Unit 1, but emergency core cooling systems were not called back into operation in order to cool down the reactor. The core was cooled during the first several hours by a high capacity pump. After several hours the high capacity pump was shut off, and Unit 2 was cooled down essentially as was Unit 1.

The investigation is being structured to develop a full and detailed understanding of the exact sequence of events associated with the fire. In addition to the action already taken, the NRC will take such other action as may be indicated as the investigation progresses, and it will make public a complete report when its investigation is completed.

#

(EDITORS: This information has also been released by the NRC in Washington, D.C.).

TVA NEWS

INFORMATION OFFICE
WASHINGTON OFFICE
TENNESSEE VALLEY AUTHORITY

RECEIVED
MAR 18 1975
STATE HEALTH DEPT.
DIV. OF RAD. HEALTH

For immediate release

Fire at Browns Ferry Nuclear Plant
Takes Two Generating Units Out of Service

TVA said Sunday it is investigating a fire among control cables which caused both units at the Browns Ferry Nuclear Plant in northern Alabama to shut down Saturday afternoon.

The agency said the loss of the cables made inoperative the automatic operation of several safety systems, but plant personnel were able to shut down both units so that control of the reactors was maintained. Unit 1 was shut down manually, while Unit 2 shut down automatically.

No one was injured by the fire, and TVA's environmental monitoring program showed that all levels of radioactivity in the surrounding area were normal and have remained well within allowable limits.

The fire started when workmen were checking for air leaks in a room where the cables converge, and which is located beneath the plant's control room. A preliminary investigation indicates that a small candle flame normally used to check for any air flow apparently ignited a sealant being used to plug a hole. The resulting fire spread to cables outside the room and was mostly confined to that area.

The workmen responded immediately by spraying carbon dioxide extinguishers on the fire and by trying to smother the flames with material. Shortly, the insulation on the cables caught fire and the room's CO_2 extinguishing system automatically activated. The workmen had to evacuate the area at this point.

Other plant personnel donned masks and air supply tanks and entered the area with additional extinguishers, but the fire was difficult to put out because of the intense heat and smoke buildup. The heat and inaccessible cable locations caused flareups over several hours.

Plant emergency procedures were activated as previously designed. TVA staffs that deal with emergency situations were called into action in Chattanooga and Knoxville. Also, the Nuclear Regulatory Commission, the states of Alabama and Tennessee, the Athens, Alabama, fire department and other involved agencies were notified.

TVA said the Athens fire department assisted in controlling the blaze and in putting it out.

The plant's operators remained at their stations in the control room throughout the firefighting operation.

TVA will make a detailed investigation of the fire, assess the damage to the plant, and determine when the units might return to service, but the agency said these determinations would not likely be made for several days.

Unit 1 at the plant, located south of Athens, Alabama, began commercial operation on August 1, 1974, and Unit 2 went into commercial operation on March 1 of this year. A third unit is under construction and is expected to begin commercial operation by the first of next year.

# # #

(Mailed March 23, 1975)

Representative YOUNG. Thank you, Mr. Godwin.

Mr. GODWIN. A few other comments, particularly in light of the U.S. Nuclear Regulatory Commission's Office of Inspection and Enforcement report of July 25, 1975.

First, our communications with the Central Emergency Control Center (CECC) TVA, did focus on the plant status. At all times we were informed of the known information regarding the release rates of radioactive materials. We feel that it was imperative to know the plant status in order to protect environmental problems; therefore, we closely followed the plant status during the incident.

Second, we feel that the efforts to contact local officials were adequate. I would point out that no action was required of local officials. The basic question is how long do you attempt to contact someone and tell him he does not have to do anything. In my log at 1620 you will note that Mr. Maple and I discussed and agreed that law enforcement personnel were available if we needed to move people. In fact, subsequent to preparing the report to the Alabama Radiation Advisory Board of Health, I have been informed that the State police on their own went to Browns Ferry Nuclear Plant to offer their assistance.

Third, our environmental data, air samples, milk analysis, and thermoluminescent dosimeters confirmed our original estimate of minimal radiological impact.

Things we learned:

1. We will have to update the telephone directory more often.
2. We will have to document who has a copy of the plan and its revisions.
3. External to the plan we need to establish a rapid information-transfer system for nonradiological incidents.
4. Environmental monitoring is a must during an incident.
5. A clearer definition of State versus local responsibilities for training should be included.

Actions to implement these items have been initiated.

I believe that concludes my short summary, Mr. Chairman.

[Prepared statement of Aubrey V. Godwin follows.]

STATEMENT BY AUBREY V. GODWIN, DIRECTOR, DIVISION OF RADIOLOGICAL HEALTH, ENVIRONMENTAL HEALTH ADMINISTRATION, ALABAMA DEPARTMENT OF PUBLIC HEALTH

Mr. Chairman and Members of the Committee: Thank you for giving me this opportunity to participate in this hearing on the Browns Ferry Nuclear Plant Fire. Because of the concern of the Alabama Department of Public Health for matters affecting the health of our people, we feel that we should meet with this Committee and provide information regarding our response to this fire so as to aid the Committee in its deliberations.

To fully understand our actions, I would like to express the philosophy used in developing the Alabama Radiation Emergency Plan. First, the operator/licensee is to contact the group with the most expertise in the immediate problem. In other words, in case of fire, call the fire department; if radiation is the problem, call the Division of Radiological Health. We feel that this is vital to any plan because it allows another review of the problem and aids in preventing unwarranted responses, either too great or too small.

We feel that this adds to the public confidence in that we are not crying wolf, nor are we hiding a problem. This independent evaluation is important. It does allow us to review in detail the situation and arrive at independent conclusions. We are then in a position to take the protective action we feel necessary to protect the public health and safety.

I should also add that our plan does include guidelines for protective actions not only for airborne exposures but for other routes of exposure also. Further, we have established reentry or release guidelines to aid us in our decision-making processes. But in spite of the availability of such guidance, we cannot overemphasize the great need for a full-time qualified professional in health physics to make a review of the information available before initiating a major response to an incident.

Second, the plan sets up an organization which can respond to the emergency as it develops. The organization is flexible and redundant. It goes without saying that inability to contact one individual or organization should not prevent the taking of proper protective actions. This requires a careful review of the overlapping and concurrent jurisdictions of all state and local agencies.

This overlap in jurisdictions may be worked to tremendous advantage in that if there is a clear emergency and one organization cannot take actions, another organization is available to do so. One further word, state agency to local agency or even to another state agency does not, in general, have a line commnd relationship as in the military; but rather a mutual trust, repect, and concurrent recognition of each others abilities. The recognition of this fact by emergency plan writers together with the recognition that *most* incidents are *not* disasters but require only a limited response by federal, state, and local agencies is vital. What I am saying is that an emergency plan should not address itself to the disaster *only;* but recognize that, by far, most incidents will require an active response by only one or two agencies.

Third, we have provided by memorandum of understanding, a mechanism whereby the various agencies that expend monies during the *emergency* in aiding the public may be reimbursed for those expenditures. This covers only evacuation associated costs, animal feed costs and agency costs above normal operation expenses. We leave to each individual affected to determine and collect moneys for losses of property, and use thereof, and such associated costs.

As Attachment A of this statement, I have included a copy of the Division of Radiological Health's report to the Alabama Radiation Advisory Board of Health. With your permission, Mr. Chairman, I would like for this Attachment to be included in the record as though I had read it. I think this will give the Committee a clear picture of our response to the fire.

I feel that some comments should be made about our response, particularly in light of the U.S. Nuclear Regulatory Commission's Office of Inspection and Enforcement report of July 25, 1975.

First, our communications with the Central Emergency Control Center, (CECC) TVA, did focus on the plant status. At all times we were informed of the known information regarding the release rates of radioactive materials. We feel that it was *imperative* to know the plant status in order to predict environmental problems, therefore, we closely followed the plant status during the incident.

Second, we feel that the efforts to contact local officials were adequate. I would point out that no action was required of local officials. The basic question is how long do you attempt to contact someone and tell him he does not have to do anything. In my log at 1620 you will note that Mr. Maple and I discussed and agreed that law enforcement personnel were available if we needed to move people. In fact, subsequent to preparing the report to the Alabama Radiation Advisory Board of Health, I have been informed that the State Police on their own went to Browns Ferry Nuclear Plant to offier their assistance.

Third, our environmental data, air samples, milk analysis, and thermoluminescent dosimeters confirmed our original estimate of minimal radiological impact. Indeed, our data show no unusual changes in the background levels. This also supports our philosophy that the licensee should initially call the most expert in the problem at hand. There may be some who criticize our sampling plan, location, etc.; but we have very limited funds and are only trying to assure that between the licensee and our program that the environmental effects are properly measured. We are not able to duplicate the licensee's monitoring program.

Things we learned:

1. We will have to update the telephone directory more often.
2. We will have to document who has a copy of the plan and its revisions.
3. External to the plan we need to establish a rapid information-transfer system for non-radiological incidents.
4. Environmental monitoring is a must during an incident.

5. A clearer definition of state vs local responsibilities for training should be included.

Actions to implement these items have been initiated.

We would point out to the Committee that as an Agreement State our Division has other responsibilities, and that responding to this incident was just one of the three incidents we investigated that same week. The others were: 1. A reported 500 REM exposure by x-ray. The investigation later proved this to be false. 2. A possible excessive tritium leakage from a gas chromatograph. This investigation is still in progress.

The Committee should be aware that approximately one man-year was expended by the Division of Radiological Health in developing this plan. The total effort by state and local agencies of Alabama probably reaches four man-years in development. The Committee should be aware that the time expended was in addition to that or in lieu of that required by the normal mission of the respective agencies.

Thank you again for this opportunity to present our ideas and comments to the Committee. I am hopeful that they will be of value in your deliberations.

I will be pleased to answer any questions you may have.

Representative YOUNG. Thank you, Mr. Godwin.

RADIOLOGICAL MONITORING

Mr. Godwin, I have a list of questions here; however, there seems to be one that I think I would like to put to you at this time. The others for the most part you have answered in your summary.

The Nuclear Regulatory Commission investigation report indicates on pages 10 and 11 of section IV that environmental air sampling equipment at Decatur, Ala., the major station of importance, was unavailable for operation, and also indicates on pages 15 and 16 that the State emergency plan was out of date or inoperative.

Would you comment on this situation?

Mr. GODWIN. Taking the first point, the station at Decatur was broken. Due to our funding we do not have sufficient funds to have a spare around so when it broke it is broken until we can get it repaired. It is just that simple.

Regarding the latter, this is the telephone directory information I was referring to. Basically we have had some changes of officials around February and we had not placed them into an updated telephone directory revision. I might point out that the agencies who contacted these officials did have the proper telephone numbers. It just was not in the plan.

Representative YOUNG. Mr. Godwin, this question has been suggested here. If there had been a radiological release of sudden magnitude, what countermeasures or other actions would you have taken? I think you stated earlier the police were there ready to evacuate and so forth.

Mr. GODWIN. Yes, sir. Depending on the levels being released, we could evacuate. We could put restrictions on milk consumption. If it was a liquid release, we could put restrictions on water intakes. We could also restrict the usage of certain crops that might become contaminated.

So we have all of these available in our plan and have guidelines to make the decision on.

Representative YOUNG. I believe you stated that you had asked the dairy to save samples of the milk and when you went back to get it you found out they had forgotten and sold it.

Mr. GODWIN. The dairy just has milk for sale and it sells it. We called him up and said, "How about holding some milk." I didn't get there before the other guy come around buying milk so they sold it to somebody else. He lives rather close to the reactor and I take it that he was not too concerned.

Representative YOUNG. At any rate, you have already stated, and it is your testimony, that there were no unusual or dangerous radiological releases on that occasion.

Mr. GODWIN. We did not detect it.

Representative YOUNG. And if there were any, then you were unaware of it?

Mr. GODWIN. That is right.

Representative YOUNG. Are there any further questions?

Representative ANDERSON. I have just one question.

Mr. Godwin, you make the statement that "In my log at 1620 you will note that Mr. Maple and I discussed and agreed that law enforcement personnel were available." That was, in other words, the moment of your initial contact with the people at the Browns Ferry plant?

Mr. GODWIN. No; sir. I believe I was contacted at 1520.

Representative ANDERSON. An hour earlier?

Mr. GODWIN. An hour earlier.

Representative ANDERSON. Which would be about 3 hours after the fire began or 3 hours and 20 minutes?

Mr. GODWIN. Something in that neighborhood, yes, sir. Then we contacted the State Civil Defense Office, Mr. Maple, and he started contacting the local civil defense coordinator and the local sheriffs.

In addition, I contacted the local health department which is one health department for three countries. Now, we got the local health department which does include three counties. However, in one county you will notice in our report the sheriff apparently does not maintain the 24-hour on-call system and the civil defense.

Representative ANDERSON. Maybe there is not as much crime there as in the District.

[Laughter.]

Mr. GODWIN. I would not express an opinion.

Representative YOUNG. Mr. Godwin, we appreciate very much your statement and your summary and it will be entered in the record along with the other material.

[Letter subsequently received from Mr. Godwin follows:]

STATE OF ALABAMA,
DEPARTMENT OF PUBLIC HEALTH,
Montgomery, Ala., September 26, 1975.

Mr. GEORGE F. MURPHY, Jr.,
Executive Director, Joint Committee on Atomic Energy, Congress of the United States, Washington, D.C.

DEAR MR. MURPHY: In the morning session of the Joint Committee on Atomic Energy's hearing on the Brown's Ferry Nuclear Plant Fire, Senator Case indicated some concern over the notification of the Sheriff in Limestone County. Although it is not as clear as may be possible, I believe that a close review of the evidence will show that attempts were made to contact the Sheriff and the Civil Defense Director in this County. These officials were unavailable; their unavailability would not have prevented the taking of protective actions. You will also note the one Limestone County Official was aware of the situation.

I hope this removes any confusion that may exist on this point.

Sincerely,

AUBREY V. GODWIN,
Director, Division of Radiological Health.

[Additional questions asked of the State of Alabama in writing and their replies are provided in Appendix 11.]

Mr. Young. This concludes today's hearings.

I want to thank all the witnesses who gave testimony.

As we have indicated, the committee intends to print and distribute the record of today's session along with supporting materials as rapidly as possible. Soon after the printed record becomes available, additional hearings will be scheduled where an opportunity will be provided to the public and other interested parties to either present an oral argument or a written statement for the record.

The hearings are recessed subject to the call of the Chair.

[Whereupon, at 3:47 p.m., the hearing was recessed, subject to the call of the Chair.]

[Note. The following information was provided by NRC subsequent to the hearing with respect to cable inspections as discussed on pages 17–18:]

We are enclosing 37 reports [1] of inspections at Browns Ferry that relate to cable installation. NRC inspections determine whether or not the licensee has and is implementing his quality assurance program. In making this determination, the NRC inspection program audits, on a sampling basis, the inspection of systems and components installed and accepted under the licensee's QA program. This sampling is aimed at inspection of plant systems rather than specific plant locations and we have not maintained records of visits to specific plant locations. In our program, the inspection of cable installations systems would take inspectors into the plant areas affected by this fire. We estimate a minimum of about 10 visits to these areas.

It should be noted that at the time of the Browns Ferry construction, there were no specific criteria which established requirements for fire stops, and; accordingly, the existing inspection program had no specific provisions for such inspection activity. Our then existing inspection program, as indicated in material submitted in answer to question 4 of the supplemental JCAE staff questions, placed emphasis on separation of redundant cables and cable tray spacing. These are the areas discussed in the inspection reports provided. Our search of the IE files revealed only one IE memorandum (V. D. Thomas to F. J. Long dated December 22, 1972) relating to fire stops resulting from the inspections of cable installation at Browns Ferry. This memorandum which is included with the related inspection report, discussed the results of a meeting between NRC inspection personnel and TVA, and noted that at the time of the meeting, TVA had not yet determined the areas in the plant that required fire barriers. The memorandum states that TVA indicated orally that plans would immediately be implemented to make this determination. No further NRC action was taken on this matter since, as stated this area was not at that time a part of formalized inspection program. Specific recommendations to, or requirements on, licensees resulting from those inspections are discussed in the inspection reports and associated transmittal correspondence with TVA.

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[1] Retained in Committee files.

APPENDIX 1

JCAE PRESS RELEASES ON BROWNS FERRY FIRE

JOINT COMMITTEE PRESS RELEASE DATED MARCH 26, 1975, ANNOUNCING INTENTION TO HOLD HEARINGS ON BROWNS FERRY NUCLEAR PLANT FIRE.

JOINT COMMITTEE PRESS RELEASE DATED JULY 31, 1975, ANNOUNCING HEARING TO RECEIVE TESTIMONY ON BROWNS FERRY NUCLEAR PLANT FIRE.

JOINT COMMITTEE PRESS RELEASE DATED SEPTEMBER 4, 1975, ANNOUNCING SCHEDULE OF WITNESSES FOR HEARINGS ON BROWNS FERRY NUCLEAR PLANT FIRE

Press Release No. 801
For Immediate Release
March 26, 1975

From the Office of the Joint Congressional Committee on Atomic Energy

ATOMIC ENERGY COMMITTEE ANNOUNCES INTENTION TO HOLD HEARINGS ON BROWNS FERRY NUCLEAR PLANT FIRE

Senator John O. Pastore, Chairman of the Joint Committee on Atomic Energy, announced today the intention of the Joint Committee to hold hearings on the circumstances and the implications, particularly from the standpoint of nuclear safety, of a fire on March 22, 1975, at the Tennessee Valley Authority's Browns Ferry nuclear plant site near Athens, Alabama.

The hearings will be held as soon as the investigatory phase of the Nuclear Regulatory Commission's (NRC) review of the fire is completed. In the meantime, NRC will keep the JCAE "fully and currently" informed on all significant developments in the investigation, pending the completion of the final report.

The hearings will be conducted by the full Committee.

Press Release No. 822
For Release
July 31, 1975

From the Office of the Joint Committee on Atomic Energy

JOINT COMMITTEE ON ATOMIC ENERGY ANNOUNCES HEARING TO RECEIVE TESTIMONY ON BROWNS FERRY NUCLEAR PLANT FIRE

Senator John O. Pastore, Chairman of the Joint Committee on Atomic Energy, announced today that the Joint Committee will hold hearings commencing at 10 a.m. on Tuesday, September 16, 1975, in the Committee's public hearing room (S-407) to receive testimony on the circumstances and implications, particularly from the standpoint of nuclear safety, of a fire which occurred on March 22, 1975, at the Tennessee Valley Authority's Browns Ferry Nuclear Plant located near Athens, Alabama.

The Nuclear Regulatory Commission today issued a report on the fire which covers the field investigation phase of its review. The Joint Committee had earlier announced that it would conduct public hearings on the fire as soon as the investigatory phase of the Nuclear Regulatory Commission's review is completed. The

Committee notes that the Nuclear Regulatory Commission is carrying out two other separate reviews relating to the occurrence of the fire—a review by a special technical group directed towards identifying whether improvements in NRC policies, procedures or technical requirements are indicated and a specific review of the plans for repair and modification of the Browns Ferry Nuclear Power Station. In view of the investigatory information which is now available, the Committee has decided to commence the hearings on September 16. At that hearing testimony will be received from NRC, State, and licensee witnesses. Either on that date or at a later date, interested members of the public will be given an opportunity to be heard or to submit statements. Members of the nuclear industry and interested members of the public who would like to testify or submit statements on the subject matter of this hearing, are requested to inform the Executive Director of the Joint Committee on Atomic Energy in writing by August 22, 1975. Hearing dates in addition to the scheduled September 16 date and the witness list for such future dates will be published at the earliest practical opportunity.

Press Release No. 823
For Immediate Release
September 4, 1975

From the Office of the Joint Committee on Atomic Energy

JOINT COMMITTEE ON ATOMIC ENERGY ANNOUNCES SCHEDULE OF WITNESSES FOR HEARINGS ON BROWNS FERRY NUCLEAR PLANT FIRE

Senator John O. Pastore, Chairman of the Joint Committee on Atomic Energy, today announced a tentative list of witnesses who will present oral testimony on September 16, 1975, at the Committee's initial day of public hearings on the Browns Ferry Nuclear Plant fire.

The Committee will first receive testimony from William A. Anders, Chairman of the Nuclear Regulatory Commission, and senior staff of NRC. This will be followed by testimony from Aubrey J. Wagner, Chairman of the Board of Directors, Tennessee Valley Authority, and senior staff of TVA. Finally, the Committee will hear from Aubrey Godwin, representing the Alabama State Health Department. The detailed list of witnesses is set forth below.

The hearing is scheduled for morning and afternoon sessions beginning at 10 a.m. and 2 p.m. on September 16. It will be held in the Joint Committee's public hearing room (S–407) in the U.S. Capitol. The hearings were announced earlier by Senator Pastore in press releases issued on March 26, 1975 (No. 801) and on July 31, 1975 (No. 822).

The Committee plans to publish the record of the September 16th hearing, along with other pertinent supporting materials, as rapidly as possible. After the record becomes publicly available, the Committee intends to schedule additional hearing sessions on the Browns Ferry fire at which time opportunity will be given to interested members of the public to be heard or to submit statements.

It is also anticipated that the Committee will receive further testimony from the Nuclear Regulatory Commission on two other reviews on the fire that are not yet completed—a review by a special technical group directed towards identifying whether improvements in NRC policies, procedures or technical requirements are indicated and a specific review of the plans for repair, modification and restart of the Browns Ferry Nuclear Power Station. Hearing dates and witness lists for future sessions on the Browns Ferry fire will be published at the earliest practical opportunity.

SCHEDULE OF WITNESSES

Tuesday, September 16, 1975—10 a.m. and 2 p.m.

William A. Anders, Chairman, Nuclear Regulatory Commission.

Dr. Donald F. Knuth, Director of Office of Inspection and Enforcement, NRC.

Benjamin Rusche, Director, Office of Nuclear Reactor Regulation, NRC.

Dr. Stephen [illegible]auer, Technical Advisor to Executive Director for Operations, NRC.

Aubrey J. Wagner, Chairman, Board of Directors, Tennessee Valley Authority.

Jack Gilleland, Assistant to the Manager of Power, Tennessee Valley Authority.

Aubrey Godwin, Division of Radiological Health, State Health Department of Alabama.

APPENDIX 2

NRC PRESS RELEASES ON BROWNS FERRY FIRE

NRC Press Release Dated March 24, 1975 Announcing NRC's Appointment of a Special Group of Technical Experts To Review the Circumstances of the Browns Ferry Fire

NRC Press Release Dated March 27, 1975 Announcing NRC's Statement on Browns Ferry Fire

NRC Press Release Dated April 3, 1975 Announcing That the NRC Staff Directed Broadened Review of Nuclear Power Plants Following Fire at Browns Ferry

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

(11)

No. 75-59
Contact: Frank L. Ingram
Tel. 301/492-7715

FOR IMMEDIATE RELEASE
(Monday, March 24, 1975)

NOTE TO EDITORS & CORRESPONDENTS: The following was issued to the wire services yesterday afternoon, March 23:

The Nuclear Regulatory Commission has organized a special group of technical experts to review the circumstances--and the implications of--a fire yesterday that damaged electrical cables at the Tennessee Valley Authority's Browns Ferry site near Athens, Alabama. The fire resulted in the shutdown of the two operating nuclear power power plants there. A third is under construction.

NRC Chairman William A. Anders said the group will be headed by Dr. Stephen Hanauer, technical advisor to the NRC's Executive Director for Operations, and will include representation from other organizations within the Commission. The group will be assisted by consultants as needed.

TVA reported to the NRC yesterday afternoon that a fire in cable "trays" in an area beneath the control room for Browns Ferry Units 1 and 2 had damaged electrical cables. One reactor was shut down automatically, and the other manually.

A team of inspectors from NRC's Atlanta office is onsite conducting an investigation. TVA reported there were no injuries and no radioactivity was detected offsite.

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

No. 75-69
Contact: Joseph Fouchard
Tel. 301/492-7715

FOR IMMEDIATE RELEASE
(Thursday, March 27, 1975)

NRC STATEMENT ON BROWNS FERRY FIRE

The Nuclear Regulatory Commission has received preliminary information from its inspectors who are investigating the causes and implications of the March 22 fire in electrical cabling at Tennessee Valley Authority's Browns Ferry Nuclear Station in Alabama.

The NRC's Executive Director for Operations has reported to the Commission that:

--Thus far, the investigation indicates that the fire resulted in considerable local damage to electrical cables. The reactor core, coolant piping and important structures were not damaged.

--There was no release of radioactivity and there were no injuries.

--The functioning of some in-plant operating and safety systems, including emergency core cooling systems, was impaired due to damage to the cables.

--The two reactors were safely shut down and cooled during the fire. NRC inspectors report that there was redundant cooling equipment available during the reactor cooldown.

--The staff has issued an advisory bulletin relating to construction and safety practices at other nuclear power plants. At present there are no indications that additional immediate action is required for other plants.

NRC inspectors arrived at the Browns Ferry site about 1 a.m. Sunday, March 23, to assure that the two reactors, shut down when the fire started, are maintained in a safe condition and to begin the NRC investigation. In addition, the Commission's inspectors are closely following TVA's own investigation into the fire.

Concurrently, a special group of Commission technical experts, named last Sunday and headed by Dr. Stephen Hanauer of the NRC staff already has begun a review of the circumstances and implications of the fire. This group is charged with reviewing the circumstances of the incident, and evaluating its origins and consequences. This group will examine technical considerations such as the design criteria of the affected equipment, its materials, and how it was installed and maintained, and whether any modifications of NRC policies, procedures or technical requirements are indicated.

The NRC teams are being assisted by fire experts from the Factory Mutual Research Corporation and the National Aeronautics and Space Administration.

On March 24 the NRC's Office of Inspection and Enforcement advised licensees building nuclear power plants at sites where there also are operating plants to review their overall systems for control of construction activities.

In the advisory bulletin, plant operators were instructed to specifically review the design, installation and testing of seals used where cables penetrate walls between compartments, and to evaluate procedures for the control of ignition sources in areas containing flammable materials.

The licensees are instructed to report to the NRC the results of the review and evaluation. Other licensees operating nuclear power plants also were provided with the information in the bulletin.

TVA reported that the fire was started by a candle which was being used to check for possible air flow through a seal where the cables pass from one compartment to another.

Detailed information on the cause of the fire and the operability of the various reactor systems during the fire is being developed by NRC inspectors. However, information developed thus far indicates:

(1) Although some instrumentation was lost, certain critical instrumentation such as reactor water level, temperature and pressure indicators continued to function and both plants were safely shut down.

(2) The nuclear fuel remained covered by cooling water at all times.

(3) Control of some systems normally used for cooling down the reactors was impaired or lost due to the fire, and alternate methods of cooling, principally the control rod drive pumps and the redundant condensate booster pumps were used. The reactors were depressurized.

(4) On Unit 1, although a loss-of-coolant accident had not occurred, the emergency core cooling system was activated and supplied additional water to the reactor. It was manually shut down to prevent overfilling. Later, during cooldown, when ECCS was called for manually as one of several alternate means of supplying cooling water, it did not activate; the alternate methods had more than sufficient capability to cool the core. The ECCS behavior is under specific investigation.

(5) On Unit 2, although four pumps in the low-pressure emergency core cooling system were inoperable, adequate cooling from ECCS was available since redundant pumps were functional. Emergency core cooling water was provided initially as in Unit 1, but emergency core cooling systems were not called back into operation in order to cool down the reactor. The core was cooled during the first several hours by a high capacity pump. After several hours the high capacity pump was shut off, and Unit 2 was cooled down essentially as was Unit 1.

The investigation is being structured to develop a full and detailed understanding of the exact sequence of events associated with the fire. In addition to the action already taken, the NRC will take such other action as may be indicated as the investigation progresses, and it will make public a complete report when its investigation is completed.

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

No. 75-79
Contact: Frank L. Ingram
Tel. 301/492-7715

FOR IMMEDIATE RELEASE
(Thursday, April 3, 1975)

NRC STAFF DIRECTS BROADENED REVIEW OF NUCLEAR POWER PLANTS FOLLOWING FIRE AT BROWNS FERRY

The Nuclear Regulatory Commission's staff today issued a broadened directive to operators of all nuclear power plants to review procedures for orderly shutdown and cooldown of the reactor should normal and preferred alternative systems be inoperative.

In addition, the plant operators must review and, if necessary, revise procedures and policies for use and control of combustible materials, ignition sources, and fire fighting equipment during construction, plant modification or maintenance.

Each licensee must report to NRC within 20 days the schedule for conducting the review, with results to be reported later.

The NRC's Office of Inspection and Enforcement issued the advisory on the basis of information indicating that the March 22 fire at Tennessee Valley Authority's Browns Ferry Nuclear Station near Athens, Alabama, involved a design modification to existing facilities. Two nuclear units were shut down because of the fire.

Today's directive supplements one issued March 24. The earlier advisory applied to nine plants with operating units adjacent to units under construction. This latest advisory covers all operating units and broadens the licensees' review.

Information developed by NRC inspectors confirms that there were no offsite effects from the fire, other than the loss of generating capacity. Although some safety systems had failed, redundant pumps and cooling systems were available to cool the reactors during and after the fire.

The fire started in the cable spreading room under the control room of Units 1 and 2, both of which were operating. A third unit is under construction.

Workers were resealing a cable penetration through a wall separating the cable spreading room from the building which houses the reactor for Unit 1 following a design modification. They were checking the airflow through a temporary seal with a candle when the material caught fire.

As a result of lower atmospheric pressure in the building housing Unit 1, the fire progressed through the penetration into the cable trays causing significant damage to electrical cabling. Damage to cables in the cable spreading room was limited to a few feet from the penetration. No mechanical equipment was damaged.

An NRC investigation of the causes and circumstances of the fire is continuing.

NOTE TO EDITORS: Attached is information, obtained from TVA personnel, on equipment and systems which did and did not perform during the fire, and a preliminary description of the method of cooling each of the two units. This information was provided to the Advisory Committee on Reactor Safeguards today. The NRC will issue a full public report when its investigation is complete.

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BROWNS FERRY FIRE 3/22/75

A. The Unit 1 reactor was cooled during the incident as follows:

12:51 p.m. - The reactor was scrammed (shutdown) manually. The high pressure coolant injection system (HPCI) and the reactor core isolation cooling system (RCIC) started automatically. The normal water sources, which include 3 feedwater pumps and the control rod drive pumps, were operating. Since the flow of all the normal water sources was more than necessary, all of the above were shutdown except one feedwater pump and the control rod drive pumps.

1:03 p.m. - The main steam isolation valves closed and could not be reopened. These valves shut off steam flow to the feedwater pumps, which are turbine driven, leaving the control rod drive pumps operable. By this time control of the HPCI and RCIC systems had been lost and they could not be restarted. It was necessary to depressurize the reactor in order to use the condensate booster pumps (350 psi output pressure) for core cooling and to maintain reactor level. This was done manually using the safety relief valves, 4 of which were controllable. During the depressurization, the reactor water level dropped from the normal level of about 200 inches above the fuel to 46 inches above the fuel. (The large drop in level was due to the steam released through the relief valves while bringing the system down from operating temperature to the saturation temperature at 350 psi.) When the condensate booster pumps were started the water level was restored to 200 inches above the fuel.

1:30 a.m. - Suppression pool cooling by the residual heat removal system was established.

4:10 a.m. - Normal shutdown cooling of the reactor was established using the residual heat removal system. Part of the residual heat removal system continued to be used to cool the suppression prol as necessary.

B. The Unit 2 reactor was cooled during the incident as follows:

1:00 p.m. - A reactor scram (shutdown) and main steam isolation valve closure occurred simultaneously. Since this shuts off steam to the feedwater turbines, the reactor core isolation cooling (RCIC) system and the high pressure coolant injection system (HPCI) were started manually. (Note: The RCIC system alone can provide the necessary water to maintain the normal level. The HPCI system, which is driven by reactor steam, was used, in this instance, only as a means of venting steam.) There was no difficulty controlling water level in this reactor at any time.

2:00 p.m. - The reactor was depressurized to 350 psi through safety relief valves to the suppression pool. (One of the relief valves was suspected to be stuck open during part of this time.

3:00 p.m. - Suppression pool cooling was started using the residual heat removal system.

4:00 p.m. - A condensate and condensate booster pump was put into use supplying required makeup water to the reactor. Shortly after this, cooling of the reactor using the residual heat removal system was started.

C. Additional means of cooling both reactors were available (if cooling described in A and B above had failed) as follows:

1. Two additional condensate booster pumps were available. If the one was lost, the other two were available for reactor makeup.
2. The operators could have dropped reactor pressure to under 150 psig and used one or more of three low pressure condensate pumps.
3. River water is available by use of two service water pumps. Although control power was lost, the two valves in this system are located in the reactor building in a place where manual operation is possible if needed.

   If offsite power had been lost, the condensate booster pump would be momentarily lost (less than a minute) while manual transfer switches were thrown to the emergency diesel generators.

Three emergency diesel generators were available, and were kept on running standby.

D. The following safety features and critical instrumentation functioned initially and throughout the fire, including the cooldown period for Unit 1.

SAFETY FEATURES

1. All Safety Valves
2. Safety Relief Valves (all operable on automatic, only 4 on manual)
3. Reactor Protection System. (prevents any rod withdrawal)
4. Control Rod Drive Pumps. (Pump water into the reactor vessel)
5. Emergency Electrical Power (3 of 4 diesel generators, on manual control only)
6. Reactor (Primary) Containment
7. Reactor Building Vent System (Exhausts filtered air from reactor building, to the stack.)
8. Two of the 8 Reactor Containment (Drywell) Coolers.
9. Liquid Poison reactor shutdown system.

INSTRUMENTATION

1. Reactor Vessel Pressure
2. Reactor Water Level
3. Radiation monitoring system (Partial)
4. Stack effluent radiation monitor and radwaste effluent radiation monitor.
5. One Half of Rod Position Indication System

E. The following safety features and critical instrumentation function initially and throughout the fire, including the cooldown period for Unit 2.

SAFETY FEATURES

1. All Safety Valves
2. All Safety Relief Valves (auto controls)
3. Reactor Protection System
4. Control Rod Drive Pumps
5. Emergency Electrical Power (3 of 4 diesel generators, on manual control only, these are same as ones listed above)
6. High Pressure Coolant Injection System
7. Reactor Core Isolation Cooling System
8. One of the Two Core Spray Systems (there are 2 pumps in each system)
9. Low Pressure Coolant Injection System (2 of 4 pumps)

10. Reactor (Primary) Containment
11. Reactor Building Vent System
12. 2 of the 4 pumps in the Residual Heat Removal System (one needed for shutdown cooling)
13. Reactor Containment (Drywell) Coolers
14. Liquid Poison Reactor Shutdown System

INSTRUMENTATION

1. Reactor Vessel Pressure
2. Reactor Water Level
3. Rod monitoring System.
4. Stack effluent radiation monitor and radwaste effluent radiation monitor.
5. Rod Position Indication System
6. Nuclear Instrumentation
7. Suppression Chamber level and temperature
8. Drywell pressure and temperature

F. The following features and control instrumentation were inoperative all or part time during the incident for Unit 1.

SAFETY FEATURES

1. All automatic and remote control of the Emergency Core Cooling System
2. Manual control of all except 4 relief valves

INSTRUMENTATION

1. All nuclear instrumentation
2. Suppression chamber level and temperature
3. Drywell pressure
4. One Half of the rod position indication system. (Lost after it had been verified that all rods were in following the scram)
5. Area radiation monitors in the vicinity of the fire
6. Reactor building continuous air monitor and area monitor.
7. Turbine building continuous air monitor.

G. The following safety features were inoperative all or part time during the incident for Unit 2.

1. Two of the four pumps in the residual heat removal system. (The system was functional)
2. One of the two core spray systems (there are two pumps in each system)
3. Manual Control on all relief valves during part of the time.

APPENDIX 3

INSPECTION BULLETINS

NRC INSPECTION BULLETIN 75-04 MARCH 24, 1975

NRC INSPECTION BULLETIN 75-04A APRIL 3, 1975

IE Bulletin No. 75-04.

MARCH 24, 1975.

CABLE FIRE AT BROWNS FERRY NUCLEAR POWER STATION

DESCRIPTION OF CIRCUMSTANCES

Preliminary information from Tennessee Valley Authority regarding a fire which occurred on March 22, 1975, at their Browns Ferry site near Athens, Alabama, indicates that the fire was started as a result of construction activities. The fire resulted in the shutdown of two operating nuclear plants and made several safety systems inoperative, including systems normally used for decay heat removal during shutdown. The workmen were engaged in construction activities on a third unit not yet licensed for operation by NRC.

Initial information indicates that during the installation and testing of cable through-wall penetrations an open flame ignited a flammable material used in the penetration seals.

ACTION TO BE TAKEN BY LICENSEES

The following actions are requested of selected Licensees with operating power reactor facilities and major construction activities at a common site:

1. Review your overall procedures and system for controlling construction activities that interface with reactor repeating activities, with particular attention to the installation and testing of seals for electrical cable between compartments of the reactor building, e.g., control room to cable spreading room.
2. Review the design of floor and wall penetration seals, with particular attention to the flammability of materials.
3. Evaluate your procedures for the control of ignition sources which may be used for leak testing or other purposes in areas containing flammable materials.
4. Report to this office, in writing within 20 days of the date of this Bulletin, the results of your reviews or evaluations regarding items 1 through 3 above.

ACTIONS TO BE TAKEN BY LICENSEES MAY BE REVISED

The actions requested of Licensees above may be revised as additional details of the Browns Ferry occurrence are available and evaluated by the NRC.

IE Bulletin No. 75-04A.

APRIL 3, 1975.

CABLE FIRE AT BROWNS' FERRY NUCLEAR PLANT

The following material supplements and modifies IE Bulletin 75-04.

DESCRIPTION OF CIRCUMSTANCES

Additional, though still preliminary, information has become available related to the fire which occurred at the Browns Ferry Site on March 22, 1975. The fire started in the cable spreading room at a cable penetration through the wall between the cable spreading room and the reactor building for Unit 1. A slight differential pressure is maintained (by design) across this wall, with the higher pressure being on the cable spreading room side. The penetration seal originally present had been breached to install additional cables required by a design modification. Site personnel were resealing the penetration after cable installation and were checking the airflow through a temporary seal with a candle flame prior to installing the permanent sealing material. The temporary sealing material was highly combustible, and caught fire. Efforts were made by the workers to extinguish the fire at its origin, but they apparently did not recognize that the fire, under the influence of the draft through the penetration, was spreading on the reactor building side of the wall. The extent of the fire in the cable spreading room was limited to a few feet from the penetration; however, the presence of the fire on the other side of the wall from the point of ignition was not recognized until significant damage to cables related to the control of Units 1 and 2 had occurred.

Although control circuits for many of the systems which could be used for Unit 1 were ultimately disabled by the fire, the station operating personnel were able to institute alternative measures by which the primary system could be depressurized and adequate cooling water supplied to the reactor vessel. Unit 1 was shut down manually and cooled using remote manual relief valve operation and condensate booster pump, and control rod drive system pumps. Unit 2 was shut down and cooled for the first hour by the RCIC. After depressurization, Unit 2 was placed in the RHR shutdown cooling mode with makeup water available from the condensate booster pump and control rod drive system pump.

ADDITIONAL ACTIONS TO BE TAKEN BY LICENSEES

1. Because the occurrence appears to have resulted from modifications being made to an operating unit, all power reactors with operating licenses should address the actions requested in Bulletin 75-04 as well as the actions described below.

2. Review your policies and procedures relating to construction or maintenance and modification work to assure that activities which might affect the safety of a unit in operation, including the ability to shutdown and cool the unit are properly controlled. Your review should consider particularly your policy on deferring construction, maintenance or modification work on a unit until a shutdown period except for emergency maintenance vital to continued safe operations or safe shutdown of the unit.

3. Review your policies and procedures to assure that for construction or modification and maintenance activities during plant operation, particular attention is given to the following areas:

(a) The degree of safety significance of affected and nearby cabling and piping.

(b) The use and control of combustible materials.

(c) The use and control of equipment that may be an ignition source.

(d) The assignment of personnel, knowledgeable of plant arrangement and plant operations, whose sole temporary responsibility is monitoring the safe performance of construction or maintenance and modification work, including attention to otherwise unattended areas adjacent to the work areas.

(e) Provision of installed or portable equipment to provide the monitoring personnel with prompt communication with the operating staff in the control room.

(f) Provision of adequate fire prevention and fire suppression equipment, installed or portable, for the following locations:

(1) Areas where work is being performed.

(2) Areas where occurrence of a fire has high safety significance, even though the probability of occurrence is relatively small.

(g) Recognition that a fire, even one involving electrical equipment, may if of sufficient intensity require water as the ultimate suppression medium.

4. Review your emergency procedures to assure that consideration for alternate methods for accomplishing an orderly plant shutdown and cooldown are provided in case of loss of normal and preferred alternative shutdown and cooldown systems for any reason (e.g. a fire). In this connection, assure that the minimum information necessary to assist the operators in such shutdown actions,

the minimum protection system actions required (e.g. scram) and the spectrum of alternative paths available to the operators to supply cooling water and remove decay heat dependent on plant conditions are included in your emergency procedures.

5. Report to this office, in writing, within 20 days of the date of this Bulletin, your schedule for review in each of the above areas.

6. Upon completion of your reviews, provide this office with the results of these reviews and the schedule for accomplishment of any revisions to your policies and procedures, and any proposed changes to the facility, and the date by which the changes are scheduled to be completed. If this latter date is more than 30 days after the date of the initial report, provide a monthly summary report detailing your progress in the review and/or proposed procedure or facility modifications. Reports requested by Bulletin 75–04 may be incorporated with the initial response to this Bulletin.

APPENDIX 4

CORRESPONDENCE

Exchange of correspondence among the Joint Committee, the Union of Concerned Scientists, and NRC from March 26, 1975 through April 14, 1975

April 2, 1975.

Mr. Lee V. Gossick,
Executive Director for Operations, Nuclear Regulatory Commission,
Washington, D.C.

Dear Mr. Gossick: It would be appreciated if NRC would provide the Joint Committee with a response to the allegations made in the enclosed letter of March 26, 1975, to Senator Pastore from the Union of Concerned Scientists.

Please reply by no later than April 11, 1975.

Sincerely yours,

George F. Murphy, Jr.
Executive Director.

Enclosure.

Union of Concerned Scientists,
Cambridge, Mass., March 26, 1975.

Senator John O. Pastore,
Chairman, Joint Committee on Atomic Energy,
Washington, D.C.

Dear Senator Pastore: We want to apprise you of certain practices that have been undertaken by the Nuclear Regulatory Commission (NRC) relating to dissemination of information about the Browns Ferry accident of March 22.

NRC sent a bulletin, from its Inspection and Enforcement Division, on March 24, to selected utilities. The bulletin confirmed certain important details about the Browns Ferry accident and directed follow up investigations at certain other nuclear plants involving the possible reoccurrence of the Browns Ferry type of accident at these other facilities. Despite the significance of this NRC bulletin, there was no public announcement of it. That is, the conventional press release or note to editors alerting them to the bulletin was not made.

The conclusions reached in the bulletin, moreover, implied that there was much more firm information about the accident than NRC and TVA were making public.

UCS learned of this bulletin, I should add, because we specifically investigated what NRC regional offices were doing to follow up at other plants on questions raised by the Browns Ferry accident.

The public information practices of NRC might properly be reviewed in the same context as the NRC advertisement that we sent you last week. That is, it ought to be determined whether the NRC information practices reflect goals other than regulation of nuclear energy.

Sincerely yours,

Daniel F. Ford.

U.S. NUCLEAR REGULATORY COMMISSION,
Washington, D.C., April 14, 1975.

Mr. GEORGE F. MURPHY, Jr.,
Executive Director, Joint Committee on Atomic Energy,
Congress of the United States.

DEAR MR. MURPHY: This replies to your April 2 letter concerning Dan Ford's questions concerning how the NRC advisory bulletin of March 24 was publicized. Commissioner Mason received a similar letter from Mr. Ford, and a copy of his reply to Mr. Ford is enclosed.

As Commissioner Mason points out, the Commission did in fact consider an announcement on March 24, but felt it was necessary to obtain firmer information before issuing an announcement providing more details on the fire. The advisory was discussed in a press release of March 27. The advisory was, however, placed in the Public Document Room, and also was called to the attention of newsmen inquiring about the fire. The second advisory of April 3 was the subject of an announcement that same day.

I hope this information is helpful.

Sincerely,

LEE V. GOSSICK,
Executive Director for Operations.

Enclosure:

U.S. NUCLEAR REGULATORY COMMISSION,
Washington, D.C., April 8, 1975.

Mr. DANIEL F. FORD,
Union of Concerned Scientists,
Cambridge, Mass.

DEAR MR. FORD: This replies to your March 28 letter inquiring as to why the first NRC advisory bulletin on the Browns Ferry fire was not publicly announced at the time it was issued.

Actually, such an announcement was considered on March 24, but we believed it was necessary to obtain firmer information from our inspectors at the Browns Ferry site before issuing an announcement providing more details on the fire. Our announcement of March 27 discussed both the advisory bulletin and other preliminary information on the fire. The initial bulletin was placed in our Public Document Room and was called to the attention of news media inquiring about the fire. The second bulletin on the fire was the subject of an NRC public announcement April 3, the same day it was sent to licensees.

I trust this information will be helpful in understanding our desire to provide the public w'th as much information as possible on the Browns Ferry fire, while at the same time assuring that the information we provide is based upon sound data developed during our investigation.

Sincerely,

EDWARD A. MASON, *Commissioner.*

APPENDIX 5

CORRESPONDENCE

LETTER DATED MARCH 27, 1975, GEORGE F. MURPHY, JR., TO WILLIAM A. ANDERS ON THE EMERGENCY CORE COOLING SYSTEM AT BROWNS FERRY. RESPONSE DATED MAY 1, 1975, LEE V. GOSSICK TO GEORGE F. MURPHY, JR.

MARCH 27, 1975.

Hon. WILLIAM A. ANDERS,
Chairman, Nuclear Regulatory Commission,
Washington, D.C.

DEAR MR. ANDERS: The "Wall Street Journal" of March 25, 1975, contains an article on the fire at the Browns Ferry nuclear power plant. Included in the account of the fire is a statement to the effect that the emergency core cooling system failed on Unit No. 1.

I understand, of course, that you have sent your experts to Browns Ferry and are awaiting their report. In this connection, it would be appreciated if you would advise the Joint Committee if the emergency core cooling system did fail and further, would you provide an assessment of the accuracy of the "Wall Street Journal" article.

Sincerely yours,

GEORGE F. MURPHY, Jr.,
Executive Director.

Attachment.

U.S. NUCLEAR REGULATORY COMMISSION,
Washington, D.C., May 1, 1975.

Mr. GEORGE F. MURPHY, Jr.,
Executive Director, Joint Committee on Atomic Energy, Congress of the United States

DEAR MR. MURPHY: This is in reply to your inquiry dated March 27, 1975, regarding the behavior of the emergency core cooling system during the fire at the Browns Ferry Nuclear Power Station on March 22. The Joint Committee and your staff have been advised as information on this incident became available, and the Commission will of course continue this practice. Our most recent communication on this subject was my letter to Senator Pastore dated April 9, 1975, with four enclosures. [Follows this letter.]

Information regarding operability of various core cooling functions, obtained from TVA personnel, was attached to our April 3 press release which was previously sent to you, and an additional copy of which is enclosed. This information includes both normal and emergency equipment capable of supplying cooling water to the core.[1]

The emergency core cooling system includes the high pressure coolant injection system, the low pressure coolant injection system, the core spray systems, and the auto-depressurization system. As can be seen from the enclosure, there were failures in these systems concurrent with the fire. The reactors were cooled by systems not considered to be part of the emergency core cooling systems, and additional means of cooling both reactors were available if they had been needed.

An emergency core cooling system is provided in each light-water reactor to cool the core in the unlikely event the normal coolant is lost via a leak or break in the primary system piping. No such break occurred in the Browns Ferry fire, which involved electrical cables. There was no loss of coolant accident and so the function which was needed and performed was not emergency core cooling.

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[1] See p. 188 for press release referred to.

The coolant was never lost. However, as in all shutdowns, it was necessary to supply make-up water to the reactors and to remove the decay heat from the cores. The condensate booster pumps, portions of the normal feed water system, were used for this purpose.

You also asked in your inquiry for an assessment of the accuracy of the article in the Wall Street Journal dated March 25, 1975. The statement attributed to TVA comports with our state of knowledge at that time and we see no significant inaccuracy in it today. The statements attributed to Mr. Comey and to the AIF presumably represent their points of view. Mr. Comey's requests to the Commission for actions he believes should be taken as a consequence of the fire at Browns Ferry are under consideration.

We will continue to advise the Committee and staff as further information becomes available.

Sincerely,

LEE V. GOSSICK,
Executive Director for Operations.

[Correspondence referred to in above letter follows:]

U.S. NUCLEAR REGULATORY COMMISSION,
Washington, D.C., April 9, 1975.

Hon. JOHN O. PASTORE,
Chairman, Joint Committee on Atomic Energy,
Congress of the United States.

DEAR SENATOR PASTORE: This is to inform you of the current status of the Commission's investigation into the electrical cable fire at the Tennessee Valley Authority's Browns Ferry site on March 22, and of subsequent NRC actions.

The NRC investigations were started shortly after the fire's occurrence and are proceeding rigorously. The scope of these investigations include: (a) review of the origin of the fire and related matters, (b) identification of events subsequent to the fire, (c) causes of interactions between reactor units, and (d) response by the various organizations involved. The investigations will focus upon implications for other operating plants and any required improvements in procedures or equipment. Since all phases of these investigations must be completed before a full assessment can be made, it is currently estimated that we will be able to make a detailed report by the end of June.

As the Committee was previously advised, two advisory bulletins have been issued to all nuclear power plant operators as a result of the fire. The first, on March 24, specifically instructed the nine utilities engaged in construction activity at sites where there are operating reactors to review certain fire prevention procedures. The second advisory on April 3 was substantially broader and directed all nuclear power plant operators to review procedures for shutdown and cooling should normal systems, and preferred alternatives, be inoperative, and procedures for the conduct of maintenance work during plant operations.

On April 3, we issued a public announcement, which was provided to the Committee, reporting on the latest advisory and including as an attachment detailed information submitted by TVA with respect to the functioning of major systems. It is significant, I believe, that although control of some systems normally used for cooling down the reactors was impaired or lost due to the fire, redundant means were available for cooling down both of the Browns Ferry reactors.

We will make public our investigation report as soon as it is available.

Sincerely,

WILLIAM A. ANDERS, *Chairman.*

[Enclosures printed elsewhere in this volume.]

APPENDIX 6

NUCLEAR REGULATORY COMMISSION REPORTS

NRC PRESS RELEASE OF JULY 30, 1975

LETTER OF JULY 28, 1975 TRANSMITTING NOTICE OF VIOLATION AND REGULATORY INVESTIGATION REPORT

REGULATORY INVESTIGATION REPORT

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

No. 75-180
Contact: Frank L. Ingram
Tel. 301/492-7771

ADVANCE FOR USE AT 6:30 PM EDT
WEDNESDAY, JULY 30, 1975

NRC ISSUES REPORT ON MARCH 22 FIRE
AT BROWNS FERRY NUCLEAR PLANT

The Nuclear Regulatory Commission today made public results of the field investigation phase of its review of the March 22, 1975, fire at the Tennessee Valley Authority's Browns Ferry Nuclear Plant near Decatur, Alabama. The field investigation was conducted by the NRC's Office of Inspection and Enforcement and covers in detail the facts surrounding the fire and its aftermath. Two additional aspects of the fire are being addressed by other organizations within the Commission.

As a result of the fire, Units 1 and 2 remain shut down. Unit 3 is still under construction.

NRC Chairman William A. Anders said of the report being released today:

"The field investigation by the Office of Inspection and Enforcement--the first of three interrelated NRC reviews--presents the facts leading to, during and following the fire. Additionally a special technical group is looking in detail at the circumstances and implications of the fire and will determine where improvements in NRC policies, procedures and technical requirements are indicated. Also, an evaluation of the safety of the units during repair and restoration and of proposed modifications to the units is being conducted by our Office of Nuclear Reactor Regulation.

"Thus, all three of the separate but related endeavors will, when completed later this year, constitute a comprehensive review of the incident. It was the desire of the Commission to release the I&E report promptly in order to make as much information as possible available to interested parties."

Dr. Donald F. Knuth, Director of the Office of Inspection and Enforcement, said the report of the field investigation--performed by the Atlanta regional office and involving 280 man-days of work--notes that the radiological consequences to the public and to the environment resulting from this fire were no more than those which would result from the normal shutdown of these reactors and were well within limits established by the NRC.

Dr. Knuth said that the investigation showed that a basic cause of the fire was a failure to recognize the significance of the flammability of the materials involved in the fire. The immediate cause of the fire was ignition of polyurethane foam sheeting used by construction workers during the sealing of penetrations. The men were using a candle to detect air leakage in a cable penetration between the Unit 1 and Unit 2 cable spreading room and the reactor building. The candle ignited the polyurethane foam sheeting.

The investigation report cites alleged infractions of NRC requirements by the utility. These involve failure to meet quality assurance requirements in the areas of taking corrective actions. inspection and audit performance, and review of design changes; noncompliance with the regulations regarding performance of an evaluation relating to a change in the facility design; and failure to follow the utility's own procedures and requirements.

An enforcement letter has been sent by the NRC Atlanta regional office to TVA. The utility has an opportunity to reply to these alleged items of noncompliance, before any final enforcement action is taken.

In addition to the items of noncompliance, the letter lists a number of areas of concern. Dr. Knuth said, "These items deal principally with matters concerning the utility's actions during the fire and its state of readiness to cope with the fire situation." As with the items of alleged noncompliance, an opportunity is afforded for reply to these matters. It should be noted that NRC review and evaluation of this incident has not yet been concluded and other NRC actions may be taken.

The introduction and conclusions of the field investigation report are attached. Copies of the full invdstigation report are available for inspection at the NRC Public Document Room, 1717 H Street, NW, Washington, D.C. 20555, and at the Athens Public Library, Athens, Alabama.

#

Attachment

Introduction

On March 22, 1975, a fire occurred at the Browns Ferry Nuclear Plant, located near Decatur, Alabama. The Browns Ferry Plant is a three-unit electric generating station owned by the Tennessee Valley Authority (TVA). At the time of the fire, Units 1 and 2 were in operation, each producing approximately 1000 megawatts of electrical power. Unit 3 was still in the construction phase. The fire originated from a candle flame used to check for air leakage in an electrical cable penetration between the cable spreading room, located beneath the control room for Units 1 and 2, and the reactor building. The fire, which burned for several hours, spread horizontally and vertically from its point of origin to all ten cable trays within the penetration, into the cable spreading room for several feet, and along the cables through the penetration for about 40 feet into the reactor building. About 2000 cables were damaged causing the shutdown of both Units 1 and 2. Some components normally relied upon for shutdown cooling of the reactors by design are also part of the emergency core cooling systems (ECCS). Because of the fire all normally used shutdown cooling systems and other components which comprise the ECCS for Unit 1 were inoperable for several hours. TVA, however, used other installed equipment and maintained sufficient cooling capability to protect the nuclear fuel. While in this case there was no pipe break accident which the ECCS are designed to accommodate, ECCS can be used to provide shutdown cooling when normal shutdown cooling systems are inoperable. There were no significant problems associated with the shutdown cooling of the Unit 2 reactor. The nuclear fuel from both units has been placed in storage on site.

At approximately 4:00 p.m. (CDT), Tennessee Valley Authority representatives notified the U. S. Nuclear Regulatory Commission (NRC) as required by the Technical Specifications for the plant. The Office of Inspection and Enforcement, NRC, promptly initiated an investigation of the fire from its Atlanta Regional Office under the provisions of 1.124 of Part 1, Title 10, Code of Federal Regulations. The investigation performed by IE of TVA was to ascertain compliance with license provisions and regulations relating to health and safety and to determine the facts associated with the fire. The investigation covered the events under the control of TVA leading to the fire and the subsequent fire fighting efforts, the sequence of operational events, the interactions between the two operating units, and the reaction of TVA and local and state agencies to the fire.

The IE investigation, the results of which are reported herein, emphasized those conditions and licensee actions which may have caused or contributed to the severity of the fire. In addition to this IE investigation, two other specific efforts are underway by NRC:

- The office of Nuclear Reactor Regulation (NRR) has undertaken a review of the safety of the plant under the post-incident conditions and an assessment of the changes in Technical Specifications required to maintain the units in a safe shutdown condition during repair activities and to return the plant to operation.

- A Special Review Group, appointed on March 26, 1975, is reviewing the circumstances of the incident and evaluating its origin and consequences from both the technical and procedural viewpoints. As a consequence of its review, the Special Review Group will recommend appropriate improvements in NRC policies, procedures, and technical requirements.

The IE investigation report does not contain the results of the NRR nor of the Special Review Group studies. These will be reported separately. For a more complete understanding of the fire and its consequences, reference should be made to the reports of all three efforts - IE, NRR, and the Special Review Group.

Scope of Investigation

The scope of this investigation included: (a) Events leading to the fire at the Tennessee Valley Authority's Browns Ferry Nuclear Plant on March 22, 1975, and the subsequent fire fighting efforts; (b) the sequence of operational events and the problems experienced with the Units 1 and 2 nuclear steam supply systems; (c) the interactions between the two operating units; and (d) the response of the TVA groups and the various state and local governmental bodies following receipt of notification of the fire. The investigation consisted of private interviews of personnel, reviews of documentation and observations by the investigators and NRC consultants. Copies of signed statements resulting from the interviews conducted by the investigators are included as Exhibit 1. In addition, flammability tests were made of the penetration sealants and cable insulation by NRC consultants.

Conclusions

A. General

1. The radiological impact on the public, plant personnel or the environment resulting from this fire was no more significant than from the routine shutdown of the reactors.

2. Both Unit 1 and Unit 2 cores remained adequately cooled throughout this occurrence. Although some systems were rendered inoperable, plant operators were successful in bringing into operation alternate means to cool the reactors.

3. Some minor injuries occurred as a result of the fire.

B. Direct Cause of the Fire, its Severity or its Consequences

1. The following factors contributed directly to the cause or severity of the fire:

   (a) Failure to evaluate the hazards involved in the sealing operation and to prepare and implement controlling procedures.

   (b) Failure of workers to report numerous small fires experienced previously during the sealing operations, and the failure of supervisory personnel to recognize the significance of those fires which were reported, and to take appropriate corrective actions.

   (c) Use of an open flame without fire precautions specific to this activi

   (d) Ineffectual leadership by TVA in fire fighting activities.

   (e) Inadequate training of TVA personnel in fire fighting procedures and equipment.

   (f) Delay in application of water in fighting the fire.

   (g) Dificulties encountered in use of self-contained breathing apparatus, caused by inadequate training of personnel in its use, inadequate maintenance and inability to recharge air bottles fully.

(h) Inaccessibility of the initial ignition location due to the position of the penetration and testing of seals prior to flameproofing.

(i) Lack of familiarity with the performance characteristics of the fire nozzle provided for such fires and ultimately used successfully to fight the fire.

2. The following factors contributed to difficulties experienced during the post-shutdown cooling of the reactors:

(a) For Unit 1, fire rendered inoperable for a significant period of time portions of the control and power circuit for components use in the normal cooldown of the reactor and in emergency core cooling systems.

(b) Lack of initiative on the part of onsite supervisory personnel to coordinate shutdown activities until after the first manual initiation of depressurization of Unit 1.

(c) Lack of knowledge of location and severity of the fire on the part of control room personnel and delay in rapid shutdown of the reactor when there were anomalous instrumentation indications.

(d) Reduction of the options for cooling available during the period of repressurization of the Unit 1 reactor cooling system.

(e) Failure to establish priorities for restoration of equipment. This resulted in uncoordinated and, in some cases, counter-productive or duplicative attempts to restore key equipment to service.

3. The following factors contributed to the adverse interaction between safety-related systems of Unit 1 and Unit 2:

(a) Two identified cases of failure to follow the TVA cable separation criteria described in the FSAR.

(b) Some Unit 1 and Unit 2 ECCS equipment is supplied from common 4 KV boards in accordance with the FSAR. The failure of any of these boards causes a loss of powe. to the connected equipment in both units.

C. Other Important Factors not Directly Contributing to the Cause of the Fire, its Severity or its Consequences

1. Rapid manual operation of the installed carbon dioxide system in the cable spreading room was precluded because metal plates installed behind break-out glass during construction had not been removed.

2. Operation of TVA's Central Emergency Control Center (CEEC) was not well coordinated; information sometimes was not exchanged internally. CECC communications with personnel at the Browns Ferry site were not effective in keeping the CECC currently informed concerning the status of the plant or of recovery activities. Consequently, communications with other agencies led to misunderstanding of plant status by those agencies.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
230 PEACHTREE STREET, N. W. SUITE 818
ATLANTA, GEORGIA 30303

JUL 28 1975

In Reply Refer To:
IE:II:CEM
50-259/75-1
50-260/75-1

Tennessee Valley Authority
ATTN: Mr. J. E. Watson
Manager of Power
818 Power Building
Chattanooga, Tennessee 37401

Gentlemen:

This refers to the investigation conducted by a team supervised by Mr. C. E. Murphy of this office on March 22 to June 10, 1975, of the activities authorized by NRC Operating License Nos. DPR-33 and DPR-52 for Browns Ferry Nuclear Plant, Units 1 and 2, as they relate to the fire of March 22, 1975, and to the discussion of our findings held by Mr. Murphy and myself on May 1 and July 8, 1975, with you and members of your staff.

The areas examined during this investigation and our findings are discussed in the enclosed investigation report. Within these areas, the investigation consisted of selective examinations of procedures and representative records, private interviews with personnel, and observations by the investigators.

From information developed during this investigation, it appears that certain of your activities were not conducted in full compliance with NRC requirements as set forth in the Notice of Violation, enclosed herewith as Appendix A. These items of noncompliance have been categorized into the levels as described in our correspondence to you dated December 31, 1974.

This notice is sent to you pursuant to the provisions of Section 2.201 of the NRC's "Rules of Practice," Part 2, Title 10, Code of Federal Regulations. Section 2.201 requires you to submit to this office, within 20 days of your receipt of this notice, a written statement or explanation in reply including: (1) corrective steps which have been taken by you, and the results achieved; (2) corrective steps which will be taken to avoid further noncompliance; and (3) the date when full compliance will be achieved.

In addition to the the need for corrective action to avoid further noncompliances of the types identified in the enclosed report, we are concerned about certain other items identified in Appendix B that also raise questions as to the effectiveness of your management control system. Consequently, in your reply to this letter, you should describe those actions taken or planned to improve the effectiveness of your management control system to assure that activities affecting quality and safety are properly implemented.

As you are aware from the "Criteria for Determining Enforcement Action" which was provided to you by letter dated December 31, 1974, the enforcement actions available to the Commission in the exercise of its regulatory responsibilities include administrative actions in the form of written notices of violation, civil monetary penalties and orders pertaining to the modification, suspension of revocation of a license. We wish to reiterate the necessity for taking prompt management action to assure full compliance with NRC requirements in the future and to correct items of noncompliance identified during the recent investigation. We plan to continue to conduct unannounced inspections to ascertain whether adequate corrective action has been taken. Our inspection findings and your reply to this letter will provide bases for us to determine whether any further enforcement action is called for, such as civil monetary penalties, or suspension, or revocation of your license.

It should be noted that NRC review and evaluation of this incident has not yet been concluded. Consequently, in addition to this letter, further NRC action may be forthcoming.

Very truly yours,

Norman C. Moseley

Norman C. Moseley
Director

Enclosures:
Appendix A, Notice of Violation
Appendix B, Areas of Concern

Docket No. 50-259 (DPR-33)
Docket No. 50-260 (DPR-52)

APPENDIX A
NOTICE OF VIOLATION

Based on the results of the NRC investigation conducted on March 22 - June 10, 1975, it appears that certain of your activities were not conducted in full compliance with conditions of your NRC Facility License Nos. DPR-33 and DPR-52 as indicated below:

1. Failure to Comply with 10 CFR 50.59

Items appearing to be in noncompliance with 10 CFR 50.59, "Changes, Tests and Experiments," as indicated below:

a. 10 CFR 50.59, requires, in part, that records be maintained of changes to the facility to the extent that such changes constitute changes to the facility as described in the Safety Analysis Report. It further requires that these records shall include a written safety evaluation which provides the bases for the determination that the change does not involve an unreviewed safety question. The Browns Ferry FSAR Section 5.3.3.5 specifies, in part, that all electrical penetrations are sealed with sealant around conductors.

Contrary to this requirement, a safety evaluation was not made of the "change to the facility as described in the Safety Analysis Report" which was constituted by operation of the reactor with containment penetrations unsealed while concurrently sealing and testing the penetrations.

This infraction had the potential for causing or contributing to an occurrence related to health and safety.

2. Failure to Comply with Technical Specifications

Items appearing to be in noncompliance with the facility Technical Specifications, as indicated below:

a. The Technical Specifications, Sections 6.3.A and 6.3.B state, in part:

"A. Detailed written procedures, including applicable check-off lists covering items listed below shall be prepared, approved and adhered to. . . .'

"4. Emergency conditions involving potential or actual release of radioactivity. . . .'

"5. Preventive or corrective maintenance operations which could have an effect on the safety of the reactor.'

"6. Surveillance and testing requirements. . . .'

"B. Written procedures pertaining to those items listed above shall be reviewed by PORC and approved by the plant superintendent prior to implementation. . . Such changes shall be documented and subsequently reviewed by PORC and approved by the plant superintendent."

(1) Contrary to these requirements, the leak testing, sealing and inspection of the penetrations were being accomplished; but detailed written procedures approved by the plant superintendent and reviewed by PORC had not been developed for the control of this work.

(2) Contrary to these requirements, persons discovering the fires on March 20 and March 22, 1975, did not adhere to the provisions of the Emergency Procedure in that they did not initiate the fire alarm.

(3) Contrary to these requirements, the Browns Ferry Emergency Procedure was not adhered to in that the Shift Engineer did not delegate onscene responsibility for fire fighting to an assistant shift engineer when he departed the fire area.

(4) Contrary to these requirements and the requirements of Browns Ferry Standard Practices Manual which specify, in part, in Standard Practice BFS3 that:

"Plant fire protection systems shall be fully operational at all times. Removal of a plant fire protection system from service for any reason other than as required in a test procedure requires the approval of the plant superintendent. Removal of a system from service for more than seven days requires a review of PORC."

The fire protection system for the cable spreading room was not fully operational in that metal plates had been installed under the glass in the manual stations during the construction of the plant and had not been removed. The approval of the installation of the plates had not been documented prior to or subsequent to the issuance of the operating license and the installation had not been reviewed by PORC. Additionally, the CO2 manual-automatic initiation system had been electrically disabled by the construction workers without documented approval of the Plant Superintendent.

This infraction had the potential for causing or contributing to an occurrence related to safety.

3. Failure to Comply with Appendix B to 10 CFR 50

Items appearing to be in noncompliance with Appendix B to 10 CFR 50, "Quality Assurance Requirements for Nuclear Power Plants and Fuel Reprocessing Plants," as indicated below:

a. Criterion XVI of Appendix B to 10 CFR 50 and the related commitments in the FSAR, Appendix D.4, "Operational Quality Assurance Program Plan," Section D.4.2.4.7 specifies, in part, that measures be established to assure that conditions adverse to quality are promptly identified and corrected; that measures assure that causes of conditions be determined and action taken to preclude repetition; and that the corrective actions are documented and reported to appropriate levels of management.

   Contrary to these requirements, during the penetration sealing operations, the conditions adverse to quality were not promptly identified and corrected; the causes of conditions were not determined and actions taken to preclude repetition; and the required documentation was not supplied in the two instances that fires were reported to management.

   This infraction had the potential for causing or contributing to an occurrence related to safety.

b. Criterion X of Appendix B to 10 CFR 50, requires, in part, that a program for inspection of activities affecting quality be established to verify conformance with documented instructions, procedures, and drawings; and that persons assigned the responsibilities for such inspections shall be independent of individuals directly responsible for work performance. Related commitments are spelled out in the FSAR, Appendix D.4., Sections D.4.2.3.1. and D.4.2.1.1., respectively.

   Contrary to these requirements, inspections of the sealing of cable penetrations were not conducted so as to assure conformance with drawings; and inspectors were involved in the work activities for which they had inspection responsibilities.

This infraction had the potential for causing or contributing to an occurrence related to safety.

c. Criterion XVIII of Appendix B to 10 CFR 50, and the related commitments set forth in the PSAR, Appendix D.4, "Operational Quality Assurance Program Plan," specifies, in part, that a comprehensive system of planned audits be carried out to assure compliance with all aspects of the quality assurance program.

Contrary to these requirements, a review of the records of the audits conducted at Browns Ferry and discussions with responsible individuals indicated that no audits had been conducted of the penetration installation.

This infraction had the potential for causing or contributing to an occurrence related to safety.

d. Criterion III of Appendix B to 10 CFR 50, and the related commitments set forth in the FSAR, Appendix D.2, "TVA Quality Assurance Plan for Design and Construction," Section D.2.4.3.4., specifies, in part, that certain basic design drawings, such as single line diagrams, are reviewed to determine that they meet the design bases, design criteria and other design input requirements.

The FSAR, Amendment 25, "Response to AEC Question 7.5," states, in part, that cables for the Engineered Safeguards Systems are separated into two redundant divisions (Division I or Division II) such that no single credible event could damage the cables of redundant counterparts. This section further states that power cables from the 4160-Volt Shutdown Boards are installed in separate conduits. It further states that the electric circuits of one of the two loops of the Core Spray System including the pump motors and electrically operated valve, are in Division I; and the circuits of the other loop are in Division II. Additionally, it states that the electric circuits associated with pumps A and C, and their valves, of the LPCI system are in Division I; and the electrical circuits of pumps B and D, and their valves are in Division II.

(1) Contrary to this requirement, the power cable supplying 480 Volt Shutdown Board 1B from 4KV Shutdown Board C (Division II) is routed in the same tray as the power cable supplying 480 Volt Shutdown Board 2A from 4KV Shutdown Board B (Division I).

(2) Contrary to this requirement, RHR Pump 1C and Core Spray Pump 1C are supplied from 4KV Shutdown Board B, and their associated valves are supplied from 4KV Shutdown Board C. Shutdown Board B is in Division I and Shutdown Board C is in Division II.

This infraction had the potential for causing or contributing to an occurrence related to safety.

APPENDIX B
Areas of Concern

1. Construction personnel were involved in work using open flame but had not been given fire fighting training nor did they have personnel with this training assigned to their area.

2. Construction personnel working in the plant were not familiar with the plant emergency procedures, and some were not familiar with the plant layout.

3. Operations personnel on shift were not fully aware of the ongoing construction activities.

4. Plant operations personnel and Public Safety Services personnel at the plant were not familiar with the reactor building fire extinguishing equipment in that they did not know that the nozzle on the hose was specifically designed for fighting electrical fires.

5. Emergency breathing apparatus appears not to have been properly maintained.

6. Certain of the operations personnel as well as the construction personnel were not familiar with use of the breathing apparatus.

7. There was apparently no attempt by CECC management to obtain expert advice on methods for fighting the fire.

8. A design review subsequent to the fire revealed two cable trays with cables installed in excess of the design criteria. The practice of the construction forces of abandoning cable in-place could result in additional cable trays being overfilled.

9. The Browns Ferry Emergency Procedures and Standard Practice Manual contain errors and inconsistencies relating to emergency response. The Standard Practice Manual, BFA-34, permits the DPP Coordinator to review work plans for safety significance and does not require further review of his evaluations. This could result in a Technical Specification violation.

10. There was apparently no attempt by the CECC director to direct the operations of the center. The plant superintendent was involved in extensive communications with the center rather than in directing the plant fire fighting and recovery operations. As a consequence, the efforts at both locations were not coordinated, management personnel were not provided with accurate information and their actions were relatively ineffective.

11. Although plant procedures specify that the Unit operator has the authority to shut down the reactor in the event of an emergency, his responsibility to effect a shutdown promptly has not been defined by management.

REGULATORY INVESTIGATION REPORT

OFFICE OF INSPECTION AND ENFORCEMENT
REGION II

Subject: Tennessee Valley Authority
Browns Ferry Units 1 and 2
50-259/75-1 and 50-260/75-1
Fire in Cable Spreading Room and
Reactor Building on March 22, 1975

Period of Investigation: March 22, 1975, through June 10, 1975

Report Prepared by:

Charles E. Murphy, Chief — 7/25/75 Date
Facilities Construction Branch
(Investigator-In-Charge)

Other Investigators:

James W. Devlin, Investigator
John J. Ward, Investigator
Michael V. Annast, Physical Security Inspector
Donald L. Caphton, Senior Reactor Inspector
William S. Little, Reactor Inspector
Floyd S. Cantrell, Reactor Inspector
Stewart D. Ebneter, Reactor Inspector
Howard A. Wilber, Reactor Inspector
Robert D. Martin, Reactor Inspector
Terry L. Harpster, Reactor Inspector
Tolbert Young, Reactor Inspector
Vince D. Thomas, Senior Instrumentation and Controls Engineer
Farris U. Bower, Reactor Inspector
Robert F. Sullivan, Reactor Inspector
James W. Hufham, Radiation Specialist
William W. Peery, Radiation Specialist

Reviewed By:

Norman C. Moseley, Director — 7/25/75 Date
Office of Inspection and Enforcement
Region II

TABLE OF CONTENTS

Introduction

On March 22, 1975, a fire occurred at the Browns Ferry Nuclear Plant, located near Decatur, Alabama. The Browns Ferry Plant is a three-unit electric generating station owned by the Tennessee Valley Authority (TVA). At the time of the fire, Units 1 and 2 were in operation, each producing approximately 1000 megawatts of electrical power. Unit 3 was still in the construction phase. The fire originated from a candle flame used to check for air leakage in an electrical cable penetration between the cable spreading room, located beneath the control room for Units 1 and 2, and the reactor building. The fire, which burned for several hours, spread horizontally and vertically from its point of origin to all ten cable trays within the penetration, into the cable spreading room for several feet, and along the cables through the penetration for about 40 feet into the reactor building. About 2000 cables were damaged causing the shutdown of both Units 1 and 2. Some components normally relied upon for shutdown cooling of the reactors by design are also part of the emergency core cooling systems (ECCS). Because of the fire all normally used shutdown cooling systems and other components which comprise the ECCS for Unit 1 were inoperable for several hours. TVA, however, used other installed equipment and maintained sufficient cooling capability to protect the nuclear fuel. While in this case there was no pipe break accident which the ECCS are designed to accommodate, ECCS can be used to provide shutdown cooling when normal shutdown cooling systems are inoperable. There were no significant problems associated with the shutdown cooling of the Unit 2 reactor. The nuclear fuel from both units has been placed in storage on site.

At approximately 4:00 p.m. (CDT), Tennessee Valley Authority representatives notified the U. S. Nuclear Regulatory Commission (NRC) as required by the Technical Specifications for the plant. The Office of Inspection and Enforcement, NRC, promptly initiated an investigation of the fire from its Atlanta Regional Office under the provisions of 1.124 of Part 1, Title 10, Code of Federal Regulations. The investigation performed by IE of TVA was to ascertain compliance with license provisions and regulations relating to health and safety and to determine the facts associated with the fire. The investigation covered the events under the control of TVA leading to the fire and the subsequent fire fighting efforts, the sequence of operational events, the interactions between the two operating units, and the reaction of TVA and local and state agencies to the fire.

The IE investigation, the results of which are reported herein, emphasized those conditions and licensee actions which may have caused or contributed to the severity of the fire. In addition to this IE investigation, two other specific efforts are underway by NRC:

> The office of Nuclear Reactor Regulation (NRR) has undertaken a review of the safety of the plant under the post-incident conditions and an assessment of the changes in Technical Specifications required to maintain the units in a safe shutdown condition during repair activities and to return the plant to operation.
>
> A Special Review Group, appointed on March 26, 1975, is reviewing the circumstances of the incident and evaluating its origin and consequences from both the technical and procedural viewpointz. As a consequence of its review, the Special Review Group will recommend appropriate improvements in NRC policies, procedures, and technical requirements.

The IE investigation report does not contain the results of the NRR nor of the Special Review Group studies. These will be reported separately. For a more complete understanding of the fire and its consequences, reference should be made to the reports of all three efforts - IE, NRR, and the Special Review Group.

Scope of Investigation

The scope of this investigation included: (a) Events leading to the fire at the Tennessee Valley Authority's Browns Ferry Nuclear Plant on March 22, 1975, and the subsequent fire fighting efforts; (b) the sequence of operational events and the problems experienced with the Units 1 and 2 nuclear steam supply systems; (c) the interactions between the two operating units; and (d) the response of the TVA groups and the various state and local governmental bodies following receipt of notification of the fire. The investigation consisted of private interviews of personnel, reviews of documentation and observations by the investigators and NRC consultants. Copies of signed statements resulting from the interviews conducted by the investigators are included as Exhibit 1. In addition, flammability tests were made of the penetration sealants and cable insulation by NRC consultants.

Conclusions

A. General

1. The radiological impact on the public, plant personnel or the environment resulting from this fire was no more significant than from the routine shutdown of the reactors.

2. Both Unit 1 and Unit 2 cores remained adequately cooled throughout this occurrence. Although some systems were rendered inoperable, plant operators were successful in bringing into operation alternate means to cool the reactors.

3. Some minor injuries occurred as a result of the fire.

B. Direct Cause of the Fire, its Severity or its Consequences

1. The following factors contributed directly to the cause or severity of the fire:

(a) Failure to evaluate the hazards involved in the sealing operation and to prepare and implement controlling procedures.

(b) Failure of workers to report numerous small fires experienced previously during the sealing operations, and the failure of supervisory personnel to recognize the significance of those fires which were reported, and to take appropriate corrective actions.

(c) Use of an open flame without fire precautions specific to this activity.

(d) Ineffectual leadership by TVA in fire fighting activities.

(e) Inadequate training of TVA personnel in fire fighting procedures and equipment.

(f) Delay in application of water in fighting the fire.

(g) Dificulties encountered in use of self-contained breathing apparatus, caused by inadequate training of personnel in its use, inadequate maintenance and inability to recharge air bottles fully.

(h) Inaccessibility of the initial ignition location due to the position of the penetration and testing of seals prior to flameproofing.

(i) Lack of familiarity with the performance characteristics of the fire nozzle provided for such fires and ultimately used successfully to fight the fire.

2. The following factors contributed to difficulties experienced during the post-shutdown cooling of the reactors:

(a) For Unit 1, fire rendered inoperable for a significant period of time portions of the control and power circuit for components use in the normal cooldown of the reactor and in emergency core cooling systems.

(b) Lack of initiative on the part of onsite supervisory personnel to coordinate shutdown activities until after the first manual initiation of depressurization of Unit 1.

(c) Lack of knowledge of location and severity of the fire on the part of control room personnel and delay in rapid shutdown of the reactor when there were anomalous instrumentation indications.

(d) Reduction of the options for cooling available during the period of repressurization of the Unit 1 reactor cooling system.

(e) Failure to establish priorities for restoration of equipment. This resulted in uncoordinated and, in some cases, counter-productive or duplicative attempts to restore key equipment to service.

3. The following factors contributed to the adverse interaction between safety-related systems of Unit 1 and Unit 2:

(a) Two identified cases of failure to follow the TVA cable separation criteria described in the FSAR.

(b) Some Unit 1 and Unit 2 ECCS equipment is supplied from common 4 KV boards in accordance with the FSAR. The failure of any of these boards causes a loss of power to the connected equipment in both units.

C. Other Important Factors not Directly Contributing to the Cause of the Fire, its Severity or its Consequences

1. Rapid manual operation of the installed carbon dioxide system in the cable spreading room was precluded because metal plates installed behind break-out glass during construction had not been removed.

2. Operation of TVA's Central Emergency Control Center (CEEC) was not well coordinated; information sometimes was not exchanged internally. CECC communications with personnel at the Browns Ferry site were not effective in keeping the CECC currently informed concerning the status of the plant or of recovery activities. Consequently, communications with other agencies led to misunderstanding of plant status by those agencies.

Summary of Findings of Facts

A. Events Leading to the Fire and Fire Fighting Efforts

During the investigation of events leading to the fire and the fire fighting efforts, the following facts were disclosed:

1. The immediate cause of the fire was the ignition of polyurethane which is used at BFNP as a cable penetration sealing material. Construction workers were checking for leaks in a penetration connecting the cable spreading room with the reactor building. Personnel performing the checks utilized a candle flame to detect air flow from the cable spreading room to the reactor building. The candle flame ignited the polyurethane. (Details I, page 3)

2. There was no approved written procedure provided to control the inspecting, sealing, and testing of cable penetrations. (Details I, page 11)

3. Personnel responsible for inspecting the penetration seals were also involved in making the seal repairs. (Details I, page 11)

4. Some penetration seals were not properly completed prior to operation of Units 1 and 2, nor were they properly maintained subsequent to operation. (Details I, page 9)

5. The location of the fire barrier in the penetration did not conform to the approved design drawings. (Details I, page 9)

6. The electrical design specified the installation of cable in trays in excess of that permitted by the TVA design criteria in at least two cases. Other trays could contain excess cable in that cable was abandoned in place when designs were changed. (Details I, page 12)

7. Although not required by license conditions, laboratory tests had been performed by TVA to verify the ability of the penetration seals to perform as a fire barrier. These tests, however, did not simulate the conditions existing at the time of this fire. For example, the tests were not conducted with a pressure differential across the seal, the polyurethane was not exposed to the flame nor was a leakage path established through the seal. Additionally, the test did not include the sheet type polyurethane in use in the cable penetrations. The use of this sheet type foam had not been approved by the TVA design department. (Details I, page 10)

8. Construction personnel were utilizing an open flame in an area which is vulnerable to fire, but specific precautions and trained fire fighters were not present at the time. (Details I, page 12)

9. Previous fires in the polyurethane foam materials had not always been reported to the appropriate levels of management, and, on the occasions when reported, no action was taken to prevent recurrence. (Details I, page 11)

10. The various procedures for responding to fires contained conflicting information. For example, the BFNP Emergency Procedure lists two different telephone numbers to be used in reporting a fire, one in a table of emergency numbers and the second in the test of the procedure. The appropriate number is the one in the test; dialing this number automatically sounds the fire alarm and rings the Unit 1 operator's telephone. (Details I, page 4)

11. The BFNP Emergency Procedure was not followed by those involved when reporting the fire. The construction workers first attempted to extinguish the fire, whereas the procedure specifies that the fire alarm be sounded first. The guard reporting the fire telephoned the Shift Engineer's office rather than calling either of the numbers listed in the procedure. (Details I, page 4)

12. During construction, metal plates were installed under the breakout glass in the CO2 system manual crank stations located at the entrances to the cable spreading room. These metal plates zwere not removed prior to plant operation, thus preventing manual initiation of the CO2 system had it been necessary. The CO2 system was successfully initiated automatically. (Details I, page 3)

13. TVA does not have outside agencies inspect their fire protection equipment and systems. Internal inspections of fire protection equipment had not revealed the presence of the metal plates. (Details I, page 6)

14. Contrary to good safety practices, the plant procedures do not restrict the use of elevators during fires; both operations and construction personnel used the plant elevators while the fire was in progress. (Exhibit 1, Elect.-D, AUO-Q et al.)

15. The practice at TVA's facilities has been that manual application of water on electrical fires be avoided for personal safety reasons. CO_2 and dry chemical equipment is provided for use by personnel fighting such fires. TVA's Fire Safety Manual permits the use of water on electrical fires and a nozzle designed for manual application of water on electrical fires had been provided at the hose station in the vicinity of the fire area. (Details I, page 13)

16. The BFNP fire hoses and nozzles were not interchangeable with the Athens Fire Department equipment because different types of threads were used. (Details I, pages 8 and 13)

17. Some TVA personnel experienced difficulty in using the self-contained breathing apparatus. (Details I, pages 6, 9 and 13)

18. There were no organized efforts to extinguish the fire in the reactor building for about three and one half hours. Sporadic individual efforts may have been made during this period. (Details I, page 7)

19. Although the responsibility for handling this occurrence rested with the Plant Superintendent, he was hesitant in authorizing the use of water on the fire; during discussions with his supervisor he requested permission to use water on the fire. (Details I, page 8 and Details IV, page 1)

20. Only minor injuries to personnel were sustained during the fire. (Details I, page 18)

21. The fire fighting and the shutdown of Units 1 and 2 were accomplished without overexposure of personnel to radiation and without the release of radioactive effluents from the plant in excess of license limits. (Details I, pages 13-18)

B. Operational Events and Problems Experienced During Fire and Until Shutdown Cooling Established

During investigation of operational events and problems experienced during the fire and until shutdown cooling was established, the following facts were disclosed:

1. The nuclear cores for both units were adequately cooled during and subsequent to the fire. The minimum water level (Unit 1) was about forty-eight inches above the top of the active fuel. Maintaining reactor water level was never a problem in Unit 2. (Details II, page 7)

2. The control room was manned at all times although some smoke and fumes entered the room for a brief period. (Details II, page 5)

3. The ability to use the emergency core cooling system (ECCS) 1/ as backup to normal cooling was lost for Unit 1. The capability to operate the Unit 1 Standby Liquid Control System was lost for at least three hours. (Details II, pages 4 and 8)

4. Loss of the air supply system for relief valve control resulted from the loss of electrical power to a solenoid valve in the air supply to a diaphram type isolation valve. This led to the repressurization of the Unit 1 reactor from about 6:40 p.m. to 10:20 p.m. (Details II, page 9)

5. The drywell vent valves for Unit 1 were wired open to prevent drywell pressure buildup. (Details II, page 9)

6. The spare control rod drive (CRD) pump was inoperable from approximately 1:30 p.m. on. The Unit 2 CRD pump could have been valved to supply Unit 1 if required. (Details II, page 4)

C. Effects on Each Unit and Interactions Between Units

During the investigation of the effects on each unit and interactions between units, the following facts were disclosed:

1. The failure of a single 4 KV Shutdown Board results in the loss of power to the ECCS equipment supplied from the board for both Units 1 and 2. This feature is inherent in the design of the electrical systems for Units 1 and 2 and conforms to the FSAR commitments. Unit 3 electrical systems are not shared with Units 1 and 2. (Details III, page 4)

2. The coordination of the power supplies for the ECCS is such that motors for the systems pumps are not in every case supplied from the same power source as their associated valves. In six cases the pumps and valves are supplied from power sources in the same division, but in two cases the valves are supplied from one division and the pumps from another, in violation of the separations criteria defined in the FSAR. (Details III, page 8)

1/ The ECCS consists of the Automatic Depressurization System, the High Pressure Injection System, the Core Spray System, and the Low Pressure Coolant Injection Mode of the Residual Heat Removal System.

3. The power cable supplying 480 Volt Shutdown Board 1B from 4 KV Shutdown Board C and the power cable supplying 480 Volt Shutdown Board 2A from 4 KV Shutdown Board B were both routed in the same tray. In that 4 KV Shutdown Board B is in Division I and 4 KV Shutdown Board C is in Division II, this routing constitutes a violation of the TVA separations criteria as defined in the FSAR. (Details III, page 9)

4. Power was lost to the common inlet valve for the core standby coolant (raw water) supply for Units 1 and 2. This valve could have been operated manually. (Details III, page 5)

5. Power from the Unit 2 preferred power bus was lost for a period of time because an operator connected it to the Unit 1 preferred bus which sustained faults. A second operator subsequently opened the tie breaker and restored power to the Unit 2 bus. (Details III, page 5)

D. The Response of TVA Groups and Various Governmental Bodies

During the investigation the following facts were established concerning the response of TVA groups and various state and local governmental bodies:

1. The Athens Fire Department (AFD) responded promptly to the notification of the fire and was prepared to assist within approximately 35 minutes of receiving the alarm. Athens is about ten miles from the site and, upon arrival, the firemen had to be issued personnel radiation monitoring devices prior to entry. (Details IV, page 1)

2. The AFD fire chief initially made the recommendation to use water on the fire at about 2:00 p.m.; however, permission to use water was not given by the Plant Superintendent to the fire fighters until approximately 6:40 p.m. and it was about 7:00 p.m. to 7:20 p.m. when water was actually used. (Details IV, page 1)

3. The individual who would normally function as TVA Director, Central Emergency Control Center (CECC), was not immediately located. The first alternate was not aware of this and had been at the CECC approximately thirty minutes before realizing that he should have been functioning as Director. (Details IV, page 2)

4. Logs kept by individuals at the CECC did not always indicate the times of the events.

5. Information provided by the CECC to others was not always accurate or current. For example, at about 5:00 p.m. an individual at CECC advised NRC that the fire was confined to the spreading room. (Details IV, pages 2 and 3)

6. The Director of the CECC was not aware that the fire had been extinguished until approximately 8:45 p.m. when in fact the fire was extinguished prior to 7:45 p.m. (Details IV, page 4)

7. Communications by the CECC with state and local agencies focused on plant operating status rather than offsite radiological releases which is the prime responsibility of these agencies. (Details IV, page 8)

8. The Alabama equipment for the downwind air sampling was not available for service. Sampling was initiated at Decatur station at approximately 9:00 p.m. on March 22 using a sampler obtained from the Alabama Pollution Control Commission. (Details IV, page 10)

9. The State of Alabama emergency plan was out of date, not available to certain responsible officials and some officials were not cognizant of their individual responsibilities. (Details IV, pages 15, 16 and 19)

10. Attempts to contact some local officials by state organizations was minimal. (Details IV, pages 15 and 16)

Enforcement Items

The following apparent items of noncompliance were identified during the investigation:

1. Failure to Comply with 10 CFR 50.59

   Items appearing to be in noncompliance with 10 CFR 50.59, "Changes, Tests and Experiments," as indicated below:

   a. 10 CFR 50.59, requires, in part, that records be maintained of changes to the facility to the extent that such changes constitute changes to the facility as described in the Safety Analysis Report. It further requires that these records shall include a written safety evaluation which provides the bases for the determination that the change does not involve an unreviewed safety question. The Browns Ferry FSAR Section 5.3.3.5 specifies, in part, that all electrical penetrations are sealed with sealant around conductors.

      Contrary to this requirement, a safety evaluation was not made of the "change to the facility as described in the Safety Analysis Report" which was constituted by operation of the reactor with containment penetrations unsealed while concurrently sealing and testing the penetrations.

      This infraction had the potential for causing or contributing to an occurrence related to health and safety.

2. Failure to Comply with Technical Specifications

   Items appearing to be in noncompliance with the facility Technical Specifications, as indicated below:

   a. The Technical Specifications, Sections 6.3.A and 6.3.B state, in part:

      "A. Detailed written procedures, including applicable check-off lists covering items listed below shall be prepared, approved and adhered to. . . .'

         "4. Emergency conditions involving potential or actual release of radioactivity. . . .'

"5. Preventive or corrective maintenance operations which could have an effect on the safety of the reactor.'

"6. Surveillance and testing requirements. . . .'

"B. Written procedures pertaining to those items listed above shall be reviewed by PORC and approved by the plant superintendent prior to implementation. . . Such changes shall be documented and subsequently reviewed by PORC and approved by the plant superintendent."

(1) Contrary to these requirements, the leak testing, sealing and inspection of the penetrations were being accomplished; but detailed written procedures approved by the plant superintendent and reviewed by PORC had not been developed for the control of this work.

(2) Contrary to these requirements, persons discovering the fires on March 20 and March 22, 1975, did not adhere to the provisions of the Emergency Procedure in that they did not initiate the fire alarm.

(3) Contrary to these requirements, the Browns Ferry Emergency Procedure was not adhered to in that the Shift Engineer did not delegate onscene responsibility for fire fighting to an assistant shift engineer when he departed the fire area.

(4) Contrary to these requirements and the requirements of Browns Ferry Standard Practices Manual which specify, in part, in Standard Practice BFS3 that:

> "Plant fire protection systems shall be fully operational at all times. Removal of a plant fire protection system from service for any reason other than as required in a test procedure requires the approval of the plant superintendent. Removal of a system from service for more than seven days requires a review of PORC."

The fire protection system for the cable spreading room was not fully operational in that metal plates had been installed under the glass in the manual stations during the construction of the plant and had not been removed. The approval of the installation of the plates had not been documented prior to or subsequent to the issuance of the operating license and the installation had not been reviewed by PORC. Additionally, the CO_2 manual-automatic initiation system had been electrically disabled by the construction workers without documented approval of the Plant Superintendent.

This infraction had the potential for causing or contributing to an occurrence related to safety.

3. <u>Failure to Comply with Appendix B to 10 CFR 50</u>

Items appearing to be in noncompliance with Appendix B to 10 CFR 50, "Quality Assurance Requirements for Nuclear Power Plants and Fuel Reprocessing Plants," as indicated below:

a. Criterion XVI of Appendix B to 10 CFR 50 and the related commitments in the FSAR, Appendix D.4, "Operational Quality Assurance Program Plan," Section D.4.2.4.7 specifies, in part, that measures be established to assure that conditions adverse to quality are promptly identified and corrected; that measures assure that causes of conditions be determined and action taken to preclude repetition; and that the corrective actions are documented and reported to appropriate levels of management.

Contrary to these requirements, during the penetration sealing operations, the conditions adverse to quality were not promptly identified and corrected; the causes of conditions were not determined and actions taken to preclude repetition; and the required documentation was not supplied in the two instances that fires were reported to management.

This infraction had the potential for causing or contributing to an occurrence related to safety.

b. Criterion X of Appendix B to 10 CFR 50, requires, in part, that a program for inspection of activities affecting quality be established to verify conformance with documented instructions, procedures, and drawings; and that persons assigned the responsibilities for such inspections shall be independent of individuals directly responsible for work performance. Related commitments are spelled out in the FSAR, Appendix D.4., Sections D.4.2.3.1. and D.4.2.1.1., respectively.

Contrary to these requirements, inspections of the sealing of cable penetrations were not conducted so as to assure conformance with drawings; and inspectors were involved in the work activities for which they had inspection responsibilities.

This infraction had the potential for causing or contributing to an occurrence related to safety.

c. Criterion XVIII of Appendix B to 10 CFR 50, and the related commitments set forth in the FSAR, Appendix D.4, "Operational Quality Assurance Program Plan," specifies, in part, that a comprehensive system of planned audits be carried out to assure compliance with all aspects of the quality assurance program.

Contrary to these requirements, a review of the records of the audits conducted at Browns Ferry and discussions with responsible individuals indicated that no audits had been conducted of the penetration installation.

This infraction had the potential for causing or contributing to an occurrence related to safety.

d. Criterion III of Appendix B to 10 CFR 50, and the related commitments set forth in the FSAR, Appendix D.2, "TVA Quality Assurance Plan for Design and Construction," Section D.2.4.3.4., specifies, in part, that certain basic design drawings, such as single line diagrams, are reviewed to determine that they meet the design bases, design criteria and other design input requirements.

The FSAR, Amendment 25, "Response to AEC Question 7.5," states, in part, that cables for the Engineered Safeguards Systems are separated into two redundant divisions (Division I or Division II) such that no single credible event could damage the cables of redundant counterparts. This section further states that power cables from the 4160-Volt Shutdown Boards are installed in separate conduits. It further states that the electric circuits of one of the two loops of the Core Spray System including the pump motors and electrically operated valve, are in Division I; and the circuits of the other loop are in Division II. Additionally, it states that the electric circuits associated with pumps A and C, and their valves, of the LPCI system are in Division I; and the electrical circuits of pumps B and D, and their valves are in Division II.

(1) Contrary to this requirement, the power cable supplying 480 Volt Shutdown Board 1B from 4KV Shutdown Board C (Division II) is routed in the same tray as the power cable supplying 480 Volt Shutdown Board 2A from 4KV Shutdown Board B (Division I).

(2) Contrary to this requirement, RHR Pump 1C and Core Spray Pump 1C are supplied from 4KV Shutdown Board B, and their associated valves are supplied from 4KV Shutdown Board C. Shutdown Board B is in Division I and Shutdown Board C is in Division II.

This infraction had the potential for causing or contributing to an occurrence related to safety.

DETAILS I

Events Leading to the Fire and Fire Fighting Efforts

A. Introduction

The fire began in the cable spreading room, elevation 606, in a bank of cable trays where the trays pass through a penetration in the common wall between the cable spreading room and the reactor building. The bank of trays on the spreading room, or north, side of the wall consists of two stacks of five trays each. The trays are approximately four inches deep and are spaced vertically on nine inch centers. On the reactor building side of the wall the trays connect with a complex system of trays, some of which continue southward, others extend vertically, and others are oriented in an east-west direction (Exhibit A1, page 1 indicates the general plant arrangement. Page 2 of this exhibit provides greater detail of the fire area). On the reactor building side there are approximately 110 conduits in the area of the penetration. Although the fire originated in the cable spreading room, it spread into the reactor building and most of the damaged cables were located in this area. The fire extended in the trays and conduit approximately thirty feet to forty feet from the penetration on the reactor building side of the wall; but only two or three feet on the spreading room side. This was primarily due to the fact that the reactor building is maintained at a negative pressure of 0.25 inch of H_2O with respect to the spreading room and the air flow tended to sweep the fire away from the spreading room. The installed CO_2 system probably aided in restricting the area of fire damage in the cable spreading room but the fire was extinguished in the cable spreading room with portable CO_2 and dry chemical extinguishers, and in the reactor building by water. Approximately twenty-six sections of cable tray were involved and some 2000 cables were damaged.

The immediate cause of the fire was ignition of polyurethane which was used as a cable penetration sealing material. At the time the fire started construction workers were checking for leaks in a cable penetration connecting the cable spreading room with the reactor building. The personnel performing the checks were utilizing a candle flame to detect the flow of air from the cable spreading room to the reactor building. Heat from the candle ignited the polyurethane. Several independent, but interrelated factors, contributed to the cause of the fire and increased the severity of the damage.

B. Activities Preceding Fire and Fire Fighting Efforts

Construction activities related to Browns Ferry Unit 3 had progressed to the point where a temporary barrier wall separating Units 2 and 3 at the refueling floor elevation of the reactor building required removal. In that this wall served as a part of the boundary of the secondary containment for Units 1 and 2, its removal would have increased the number of potential leakage paths as well as the total volume of the secondary containment. The secondary containment integrity could have been violated if the air leakage through the penetrations into the reactor building was not minimized prior to the removal of the wall.

On March 7, 1975, a secondary containment leak rate test was conducted to determine the actual leak rate. The measured rate was 7000 cfm at 0.25 to 0.27 inch of water negative pressure. The Technical Specifications require that a negative pressure of 0.25 inch of water be maintained in the secondary containment with an air flow through the Standby Gas Treatment System (SBGTS) not exceeding 9000 cfm.

The installation of the cable tray penetration sealant and fire retardant coating specified on the design drawings had not been completed in either the Unit 1 or Unit 2 area prior to the operation of the units. In addition, Division of Engineering Construction (DEC) personnel had removed sealant and fire retardant coating from conduit and cable trays over a period of months to permit installation of Unit 3 cables and cables required by design modifications to Units 1 and 2. (Operating the facility with the penetrations unsealed is in noncompliance with the requirements of the Technical Specifications, Section 5.4.B.). Subsequently, DEC implemented Work Plan No. 2892 (Exhibit A2) on March 7, 1975, to test the individual penetrations entering the reactor building and to seal those penetrations identified as leaking. This work plan did not provide detailed written procedures and it had not received Plant Operations Review Committee review nor had it been approved by the Plant Superintendent as is required for corrective maintenance operations by Technical Specification Sections 6.3.A.5 and 6.3.B.

The work under Work Plan No. 2892 had continued on a priority basis until March 22, 1975. On that day, Electrical Engineering Supervisor A directed Engineer B to coordinate the efforts of three teams, each consisting of an engineering aide and an electrician, in this work. Work proceeded that day without incident until approximately 12:15 p.m.[1/] Engineering Aide C, was leak testing, from the spreading room side, a tray penetration connecting the spreading room and the Unit 1 reactor building (Exhibit A3). He observed a hole which he estimated to be two inches by four inches and which contained three cables. Engineering Aide C stated that he passed a lighted

1/ Exact times could not be determined for many of the events described in this report. Estimated times are based upon the statements obtained from the individuals involved, and, when available, the recorded times of concurrent or related events. All times are CDT.

candle by the hole and the cable flame blew horizontally into the hole, indicating to him a significant leakage path into the containment. The hole was approximately twenty inches back into the penetration from the face of the concrete wall. The entire penetration was congested with cable trays and the location of the hole was difficult to reach. Engineering Aide C stated that Electrician D, handed him two pieces of sheet polyurethane foam. He then stuffed the foam into the hole. At approximately 12:15 to 12:25 p.m., Engineering Aide C relit the three to four inch candle and checked the opening to determine if it were now sealed. Engineering Aide C emphasized the difficulty he had reaching into the penetration, (even with his relatively long arms). Engineering Aide C stated that the flame again went horizontal, indicating a large air flow and leakage path. He saw that the foam had caught fire and there was a low red glow. Engineering Aide C yelled "fire."

Electrician D stated that he handed Engineering Aide C his flashlight and told him to beat the fire out. Engineering Aide C stated that he used the flashlight and that the lens was burned trying to put the fire out.

Electrician E, who was also working in the cable spreading room, stated that he heard someone call for a fire extinguisher and passed them some rags. Engineering Aide C stated that he stuffed three or four rags in the hole trying to snuff the fire out. He then removed the rags, but the fire continued to glow. Individual F, a third electrician working in the cable spreading room, stated that the rags were smoldering when removed.

Electrician E stated that he crawled out of the cable spreading room to get a fire extinguisher. He then returned with a CO_2 fire extinguisher. Electrician E gave the extinguisher nozzle to Engineering Aide C and discharged the extinguisher twice. Engineering Aide C stated that the CO_2 blew right through the hole without putting the fire out. The fire, by this time, had gotten further back into the wall penetration. Engineering Aide C stated that he then discharged two dry chemical fire extinguishers into the hole, but the fire continued. Electrician D stated that he called for someone to notify the shift engineer that there was a fire in the cable spreading room. Engineering Aide C stated that after approximately 15 minutes of fighting the fire, there was very little heat and no smoke on the cable spreading room side of the penetration. He could still put his hand close to the fire and the draft was moving the fire further into the hole, away from him. Engineering Aide C stated that he heard someone discharge a fire extinguisher on the other side of the penetration. Someone handed Engineering Aide C another fire extinguisher (CO_2), but before he could discharge this extinguisher, the cable spreading room evacuation siren alarmed. Engineering Aide C and Electrician D crawled to the exit door at the Unit 1 end of the cable spreading room. Prior to this time the other two teams had also evacuated the cable spreading room. These teams consisted of Engineering Aide H, and Electricians J, E and F. Assistant Shift Engineer G then shut the cable spreading room Unit 1 door in preparation for triggering the permanently installed CO_2 Cardox fire extinguishing system. (The system may be actuated only from the control stations at the Unit 1 and Unit 2 doors to the cable spreading room).

Electrician E stated that upon exiting the cable spreading room prior to the triggering of the Cardox system he proceeded to the reactor building to fight the fire in that area (See Exhibit A1, page 1).

C. Instructions for Reporting a Fire

The BFNP Emer :ncy Procedure, a one page document dated February 11, 1972, specifies, in part, that the person discovering the fire will:

"1. Sound fire alarm and report condition to control room operator. (Dial 299 and wait for an operator to take information.)'

"2. Take immediate corrective action (use fire extinguisher, etc.) If not sure of correct action, wait for instructions."

Dialing 299 activates the plant fire alarm and rings the telephone in the Unit 1 control room.

The listing of emergency numbers in a table at the bottom of the emergency procedure does not list 299 as the fire number, but rather gives 235, the Public Safety Service number. The emergency procedure, however, defines the role of the Public Safety Service as a support function to the operations personnel. The emergency procedure is posted in various areas of the plant.

The Browns Ferry Standard Practice Manual, in Standard Practice BFS3, dated February 24, 1975, specifies that fires be reported immediately to the construction fire department telephone 235, whether in the construction area or an area for which the Division of Power Produc ion (DPP) is responsible. This procedure contains no mention of the plant fire alarm number (299) or the need to notify operations personnel.

D. Reporting This Fire

The BFNP Emergency Procedure was not followed in reporting the fire by either the construction personnel or the Public Safety Service. According to the construction personnel involved in the sealing operations and the project construction manager, the construction workers were not familiar with the plant emergency procedures. Some, in fact, did not recognize the fire alarm. The plant operations personnel first became aware of the fire after Electrician F went to the guard post in the turbine building at the entrance portal on the El. 565 level near the temporary wall between Units 2 and 3. According to Electrician F and the Guard K, Electrician F took a fire extinguisher from the post back to the fire and the guard called telephone number 213 and reported the fire to the operations man (Assistant Shift Engineer L) who answered his telephone call. Assistant Shift Engineer L activated the fire alarm by dialing 299 and advised Operator M and Shift Engineer R of the fire. Assistant Shift Engineer L then sealed in the fire alarm. Operator M and Shift Engineer R were advised of the fire at 12:34 p.m. CDT.

Operator M then announced the fire over the public address system stating, "Fire in reactor building, location unknown." Electrician F met Apprentice N while on his way to the fire area in the reactor building. Apprentice N ran to get another fire extinguisher and met two other construction workers, Pipefitter O and Steamfitter P. All three returned to the reactor building. Electrician F and Apprentice N located the fire in a cable tray on the northeast side of the reactor building where they met Electrician E. A ladder was placed against some temporary wood scaffolding which was adjacent to the cable trays. Apprentice N went to the top of the ladder with a dry chemical fire extinguisher. He stated that at this time the flames were about eight inches high and the fire was about two feet from the wall in the bottom tray. There was a general red glow and yellow flame. The cables appeared to be burning and there was white smoke curling off the cable but there was no swelling. There were no cracking or popping sounds. He had some difficulty discharging the extinguisher and was forced to leave when he could not breathe. The people on the floor thought the fire was out. Electrician F had climbed the ladder behind Apprentice N and he stated the packing in the penetration appeared to flow out as it melted.

Pipefitter O stated that when he arrived at the fire he called the shift engineer and notified him of the fire. (This call would have been subsequent to the call made by Guard K.) He also assisted Apprentice N to get fresh air when he came off the ladder. Electrician E stated that he dialed the fire number to get some respirators and to make sure the fire was out. (At that time, he thought that it was out.) Electrician E stated that someone arrived wearing a mask or respirator and emptied a CO_2 extinguisher on the burned cables. Operators then arrived at the fire. Electrician E then went to the 565 level and told the guard that the fire might feed back into the spreading room.

After activating the fire alarm at about 12:34 p.m., Assistant Shift Engineer L went to the reactor building fire area with a dry chemical fire extinguisher and discharged it on the fire and the fire appeared to go out. He then took a CO_2 fire extinguisher up the ladder and discharged it on the fire. Assistant Shift Engineer G had arrived at the fire by this time and he and Assistant Shift Engineer L called the control room to request breathing apparatus. (The dense smoke and the lack of satisfactory breathing apparatus were considered by those interviewed by the investigators to be the major deterrents to the fire fighting efforts.) Operator M advised Assistant Shift Engineer L that he was having trouble with erratic operation of Unit 1 pumps and the indicating lights on panel 9-3. This information was relayed by Assistant Shift Engineer L to Shift Engineer R who was at the fire and Shift Engineer R and Assistant Shift Engineer G left at approximately the same time for the control room to assist Operator M. (The estimated time of Assistant Shift Engineer L's call was 12:40 p.m. to 12:43 p.m.)

Assistant Shift Engineer L stated that he then observed that the fire was coming out of the bottom tray close to the wall and was extending up about four trays. Assistant Shift Engineer L did not know that the fire was initiated in the spreading room and he called Operator M to initiate the spreading room evacuation alarm and have Assistant Shift Engineer G initiate the spreading room CO_2 (Cardox) system. Assistant Shift Engineer L stated that after talking to the Unit 1 operator, the lights went out, he ordered all bystanders in the fire area to go to the reactor building first floor because of the smoke and he returned to the control room. Assistant Shift Engineer L called the Athens Fire Department (AFD) at 1:09 p.m. The AFD fire chief stated that their truck arrived at BFNP at approximately 1:30 p.m. and that by 1:45 p.m. the AFD firemen had been admitted to the plant and were prepared to assist in the fire fighting efforts.

Assistant Shift Engineer G stated that he initiated the evacuation alarm and went to the Unit 1 door of the spreading room. He saw Engineering Aide C and Electrician D leaving the room and determined from them that to their knowledge, no one else was in the room. He also called out loudly and looked under the trays. He stated that he then attempted to initiate the Cardox system at the Unit 1 spreading room control station without success. He found that the power had been shut off at the Disable Switch at the Unit 2 entrance so he then went around to the Unit 2 door and turned the power on at this station. The automatic initiation did not appear to be successful and he next attempted to use the manual crank system but found that a metal plate had been installed under the breakout glass. (He stated that he later determined that metal plates had been installed on the manual stations during plant construction and that almost all were still installed the day after the fire.) 1/ About three minutes

1/ TVA does not have outside agency inspection of their fire protection equipment. Annual inspection is provided by TVA's Public Safety Service, Chattanooga office. An inspection had been conducted in June 1974, and another had been scheduled for the week following the fire.

after initiating the automatic controls the CO_2 system began discharging into the cable spreading room from the automatic initiation. Other fire fighting efforts appear to have been temporarily suspended at approximate 1:00 to 1:10 p.m., at the time Assistant Shift Engineer L departed from t reactor building.

E. Further Fire Fighting Efforts

According to Shift Engineer R he had not delegated anyone to take over direction of the fire fighting activities when he returned to the control room but had assumed that Assistant Shift Engineer L would take charge since Assistant Shift Engineer L was the senior person remaining at the fire. The emergency procedures specify that the shift engineer, Individual R in this case, has the responsibility for directing fire fighting efforts; but, that this responsibility may be delegated to other Shift Engineer R further stated that he was not in the control room when the Plant Superintendent arrived, but assumed that the Plant Superintendent had assumed responsiblity for all activities. Shift Engineer R could not remember whether or not he had discussed the status of the plant with the Plant Superintendent when he saw him.

Although individual operations personnel may have entered the reactor building subsequent to the withdrawal of personnel by order of Assistant Shift Engineer L, at approximately 1:00 p.m., there appears to have been no central organized direction of the fire fighting efforts in this area until approximately 4:30 p.m. Shift Engineer U, who arrived at the site at 3:00 p.m., stated that he discussed the need to fight the fire with Shift Engineer V and then with the Plant Superintendent, Shift Engineer R and Assistant Operations Supervisor W. He first went to the cable spreading room and found that Assistant Unit Operator X and some public safety people were putting dry chemicals on the fire. Individual G had also returned to the spreading room during the period from approximately 2:00 p.m. to 4:30 p.m. to aid in fighting the fire. An Athens fireman was in the group fighting the fire. Shift Engineer U stated that the CO_2 flooding system had previously been set off three different times, and at approximately 4:00 p.m. the fire seemed to be contained and was reported extinguished at 4:20 p.m.

Shift Engineer U stated that he then went to the reactor building and, after some discussions with Athens Fire Department personnel, entered the reactor building through the air lock behind the "B" 4KV shutdown room at about 4:30 p.m. He and Assistant Shift Engineer C directed the setting up of direct current lights both inside and outside the reactor building. He and two others entered the reactor building and found the fire going strong in two places and smoldering in several other places. He stated that the hottest fire was in the trays thirty feet straight out from the penetration. He directed several people including Assistant Shift Engineer L and Y, and Assistant Unit Operators, Z, AA and BB in fighting the fire with dry chemicals. Some of the people strung a life line into the area. Public Safety Officer CC and members of the Athens Fire Department assisted in these operations but did not actually use the extinguishers.

During this time (approximately 4:30 p.m.) it was necessary to use breathing apparatus even though the ventilation system was operating. Fire fighters operated in groups of three using the life line strung into the area. The AFD fire chief at one point went to the area of the fire with Shift Engineer U. Shift Engineer U also made periodic reports to Operations Supervisor DD and to the Plant Superintendent. The Plant Superintendent also came down to the area of the shutdown room. The AFD chief had recommended the use of water on the fire and Shift Engineer U discussed it with the Plant Superintendent.

At approximately 5:30 p.m., the Plant Superintendent received permission from his management in Chattanooga to use water on the fire, but to use extreme caution. Shift Engineer U stated that at about 5:00 p.m., they still considered the use of water as being too risky. Breathing was still a problem at this time since the Scott Air Packs only had from ten to fifteen minutes service because there was no way to fully charge them. The AFD Fire Chief advised the investigators, that he again recommended the use of water at about 6:00 p.m. According to the AFD Fire Chief, the Plant Superintendent agreed to the use of water on the fire at approximately 7:00 p.m. The use of water was contrary to the recommendations of Public Safety Officer CC. The initial effort to use water was made by Shift Engineer U, the Athens Fire Chief and another operator. The investigators were advised that first attempts were unsuccessful since the nozzle was a cascade type and would only reach the bottom tray. It is also probable that at this time the hose had not been completely removed from the hose rack and that full water pressure did not reach the nozzle. Manufacturer's literature indicates that the nozzle has a reach much in excess of this distance. A nozzle supplied by the Athens Fire Department was tried but it had incorrect type threads and would not stay on the hose. Shift Engineer V stated that at approximately 6:30 p.m. he obtained sets of Chem-Ox breathing apparatus from the chlorine building and the health physics office. Shift Engineer V stated that at about 7:00 p.m., he decided to use water on the fire. He, Shift Engineer U and Assistant Unit Operator AA then went to the area of the fire. Shift Engineer V had put the original nozzle back on the hose. He climbed up the scaffolding with Assistant Unit Operator AA feeding the hose to him. According to both

Shift Engineer U and Assistant Unit Operator AA, when the water was used on the fire, it steamed but there was no indication of electrical shorts. Because of difficulty with breathing, Shift Engineer V jammed the nozzle in the tray so that it would continue putting water on the fire area, climbed off the ladder, and left the reactor building. Shift Engineer J and Assistant Unit Operator AA also left the building but returned at about 7:15 p.m., and found the fire to be out. Shift Engineers V and U and Assistant Unit Operator AA then sprayed the area additionally. The Chem-Ox was considered by some who used it to be highly effective and permitted them to remain in the fire area for the time necessary to get the fire under control. Others, however, experienced problems with the air demand limitations of these units.

The fire was declared to be out at 7:45 p.m.

F. Design, Construction and Testing of Penetrations

The reactor building (RB) at Browns Ferry functions as the secondary containment for the three nuclear steam supply systems. The building is required to be maintained at a negative pressure in relation to the remainder of the plant and to the outside environment in order to preclude uncontrolled and unmonitored releases of airborne radioactivity. The Ventilation and the Standby Gas Treatment Systems are designed to maintain the negative pressure. Conduit connecting the reactor building with the remainder of the plant is sealed to minimize inleakage of air. Cable trays are routed through wall penetrations and these penetrations are also required to be sealed after installation of the cable so as to minimize inleakage.

The penetrations where the fire occurred were located in the cable spreading room, elevation 606, in the Unit 1 area immediately east of the Unit 2 area. The cable spreading room and the reactor building in this area is approximately twenty-six inches thick (See Exhibit A3). The penetration is four feet by four feet and contains ten horizontal trays arranged in two parallel stacks of five trays. The trays in each stack are spaced on nine inch centers vertically. The trays pass through openings in a vertical steel plate bulkhead located approximately twenty-three inches from the face of the spreading room wall. This is contrary to the design drawings which specify that the plate location was to be midway between the faces of the wall. The licensee committed in Amendment 24 to the FSAR to sealing the penetrations to minimize the inleakage of air to the reactor building. Failure to construct the penetration barrier according to the design drawings is contrary to commitments made in the FSAR. The failure to complete the penetration seals prior to licensing Units 1 and 2 and to maintain the seals during operation is contrary to the FSAR commitment and the requirements of Technical Specification, Section 5.4.B.

The design drawings for the conduit and cable tray system show polyurethane to be the sealing material but do not otherwise provide sufficient details of the method of sealing these openings to assure proper sealing. The intent of the design, however, was that after the installation of the cable in the conduit, the penetrations were to be sealed with polyurethane foam applied by pouring or as a spray. The pourable foam was Pittsburgh Chemical Selectrofoam (or similar type) and the sprayable foam was Instafoam. (The spray version is reported by the manufacturer as self-extinguishing according to ASTM-1692-59T.) After completing the sealing operations, the penetrations were to have been coated with a fire retardant, Flamastic 71.

Tests were conducted of flammability of the foam with Flamastic fire retardant by the TVA Electrical Engineering Design Section and the results were reported in a memorandum dated June 4, 1973 (Exhibit A4). This test did not duplicate the conditions existing at Browns Ferry at the time of the fire in that the test was performed with a complete coverage of the polyurethane by the fire retardant, a leakage path did not exist through the foam, and a differential pressure did not exist as it did in the actual conditions at the time of the fire as discussed below.

An examination made of the penetrations by the investigators and construction personnel in Units 1 and 2 after the fire indicated that for the most part, the Flamastic coating either had not been applied or that its integrit had been violated by modifications made after the installation of the Flamastic coating (Exhibit A5). The work in progress at the time of the fire was directed toward identifying those penetrations that had not been properly sealed to minimize air flow and those whose integrity had been violated after their seals had been completed.

Electrical Engineer FF stated that he had conducted a test of RTV-102 silicone rubber used in sealing penetrations. He had authorized the use of sheet polyurethane foam that had been used by DEC as a dam to retain the spray and pourable types of foam. The sheet foam had not been tested to his knowledge. Electrical Engineer FF further stated that the spray and pourable foams had been tested and these tests had demonstrated that this foam was not flammable when coated with Flamastic. Tests conducted by the NRC consultant subsequent to the fire (Exhibit A6) demonstrated that the sheet foam, identified as Aire-Lux in the report, is extremely flammable and that the other two types of foam are also flammable under the conditions existing at the time of the fire. The tests conducted by the consultant demonstrate that the foam is relatively nonflammable when completely coated with Flamastic.

Interviews conducted by the investigators with the personnel inspecting and testing the penetrations at the time of the fire indicate that the personnel performing the installation of sealing materials were also responsible for the tests and inspections. Inspections by personnel independent from those doing the work, therefore, had not been performed. Reviews of the reports of the QA audits conducted by the auditors from the Office of Engineering Design and Construction and discussions by the investigators with responsible individuals, revealed that the penetration installations had not been the subject of any audits conducted by that group.

The inspectors also questioned DEC personnel with regard to the use of procedures in installing the penetration seals. None of those questioned were aware of the existence of procedures, but advised the investigators that they had been previously told or shown how to do the work and conduct the tests. A DEC craft supervisor, JJ, advised the investigators that a procedure had been developed for this work and supplied the investigators with a copy of this document (Exhibit A7). This document had not received the reviews and approvals required of procedures by TVA; the construction management and the individuals doing the work were unaware of its existence and it did not include the essential elements of a procedure; e.g., precautions, prerequisites, acceptance criteria, test details, and approvals, prescribed by TVA Administrative Procedures. Electrical Engineer FF stated that there was no procedure that describes the air leak test.

Interviews with DEC personnel also revealed that it was not unusual for the sheet foam material to ignite. The usual method of extinguishing the fire was for the personnel to pinch it out with their fingers or smother it with cloth. Electrical Engineer FF was aware of these fires and it was reported to the investigators that Electrical Engineering Supervisor A was also aware of them in the spray and pour type foams. He said that he was aware of one fire in the sheet foam which occurred on March 20, 1975, but was not aware that the sheet foam was used in the permanent installation. Two fires had occurred on Thursday, March 20, 1975, in the spreading room, one of which required the use of dry chemicals to extinguish it. This latter fire was reported in the Shift Engineer's Log on March 20, 1975 (Exhibit A8) and was discussed in the operators' meeting on March 21, 1975. Assistant Superintendent GG stated he was aware of this occurrence. Assistant Operations Supervisor W had directed that the maintenance electricians examine the cables in these trays for damage and this was done. No other action was taken relative to either of these fires.

No action to prevent recurrence of fires could be identified by the investigators.

G. Cable Tray, Conduit and Cable Installation

The investigators reviewed the installation of the conduit and tray systems located in the fire zone. This review did not detect any deviations from the FSAR commitments for separation in the installation of these systems.

Subsequent to the fire, TVA calculated the cable fill at eight checkpoints in the zone of influence. Two trays were determined to exceed the allowable fill requirements. Tray MD at checkpoint 131 had a calculated design fill of 55.693 square inches as opposed to the design critiera of 43.2 square inches maximum. Tray KT has a fill of 50.104 square inches as opposed to the maximum allowable fill of 43.2 square inches. Since TVA construction did not remove cables once installed in the trays, design changes issued subsequent to the start of cable installation resulted in cables being abandoned in place. These abandoned cables are not included in the design fill calculations and it is possible that other trays were also physically overloaded. The actual fill will be determined at the time that the damaged cables are removed from the trays.

H. Training of Personnel in Fire Fighting

The DEC personnel involved in the fire fighting efforts had not received training in fire fighting. Three of the Division of Power Production (DPP) shift engineers had received training to varying degrees. One had extensive navy training and a one day refresher course. Two had received short state directed courses. Two of the assistant shift engineers had received training including refresher courses at Browns Ferry. One unit operator had received approximately seven weeks training. Very few of the operations personnel had participated in fire drills. The public safety officers receive formal training and fourteen were engaged in a fire drill at the time of the fire. These officers provide both guard and primary fire fighting services for DEC and guard service for operations. They also provide backup fire fighting service for operations, but have no direct responsibility and were not utilized in a coordinated manner in this instance. (See Exhibit A9 for the NRC consultant's report of the fire fighting and related activities.)

I. Conduct of Fire Fighting Activities and Hazards Involved

The initial fire fighting efforts were by the DEC personnel who had been leak testing and sealing the penetrations. These individuals had not received training in fire fighting and few, if any, had participated in fire drills. In addition, the work plan did not require that there be anyone assigned to standby with equipment in the event of a fire.

Shift Engineer R, who had received training in fire fighting, participated in the fire fighting during the first short period following the reporting of the fire. He returned to the control room because of the difficulties being experienced with the cool down of Unit 1. The senior operations man remaining, Assistant Shift Engineer L, had not had previous experience in fire fighting. Assistant Shift Engineer L did much to coordinate the fire fighting efforts until the smoke became too dense. According to those fighting the fire, this smoke, coupled with the lack of effective breathing apparatus, forced a cessation of fire fighting activities. Assistant Shift Engineer L, did not receive direction from his supervisor and his activities in the reactor building were not successful since neither CO_2 nor dry chemicals were effective. The Public Safety Fire Brigade Leader was not utilized in the capacity of brigade leader but as a member. He recommended against the use of water on the fire, although the TVA Fire Protection Manual, Section XI, establishes water as an acceptable agent to be used for electrical fires and the nozzle installed on the fire hose in the reactor building was specifically designed for electrical fires (Exhibit A10).

The investigators were advised by some individuals that breathing apparatus was in short supply and not all of the Scott packs were servicable. Some did not have facemasks and others were not fully charged at the time of the start of the fire. The breathing apparatus was recharged from pre-charged bulk cylinders by pressure equalization. As the pressure in the bulk cylinders decreased, the resulting pressure decrease in the Scott packs limited the length of the time that the personnel could remain at the scene of the fire. The air demand limitations of the Chem-Ox units posed a problem with some of the personnel. In addition, the breathing apparatus interferred with vision and prevented voice communication. The bulkiness of the Scott packs hindered fire fighting activities in confined spaces, such as the cable spreading room, and on the ladder. The exact number available to the firefighters was not determined.

Only after Shift Engineer U assumed direction of the fire fighting in the reactor building was a coordinated effect achieved, and it was not until water was used that the fire was extinguished.

J. Plant and Personnel Radiological Surveys

1. General Discussion

As a result of the fire at the Browns Ferry Nuclear Plant, the reactor building ventilation system was inoperable from approximately 12:45 p.m. until 4:00 p.m.; however, there was flow through the vents induced by natural drafts. The Unit 1 reactor building ventilation duct monitor was inoperable during the entire period. The Unit 2 reactor building vent monitor was inoperable from approximately 2:00 p.m. until it was restored to service at approximately 9:00 p.m. During the fire and the time the building duct monitors were out of service, grab samples were collected approximately every hour to determine concentrations of radioactivity being released from the reactor building. All other building ventilation monitors except Unit 1 and Unit 2 reactor building monitors were operable. Gamma spectrum analyses of samples collected both inside the plant and in the reactor building ventilation ducts indicated that the only isotope present was rubidium-88, a daughter product of krypton-88 with a half-life of 17 minutes. No surface contamination was found in smear samples taken in the Unit 1 Reactor Building. The Health Physics Supervisor F stated that the airborne activity level for Rubidium reached a maximum level of 35% MPC but after ventilation was reestablished the activity began to decrease and by 9:00 p.m. on March 22, 1975, it was less than 5% MPC. This was the only activity detected. These spectra were provided by the radiochemistry laboratory at the BFNP. Utilizing the reactor building ventilation grab sample results, data from the other operable building vent monitors, and the stack monitoring data, dose estimates were calculated. Meterological data collected from the BFNP meterological tower during the period were also utilized. Calculations performed utilizing these combined data indicated the maximum estimated dose in any one sector to be 1.8 millirem at the site boundary. These calculations are conservative since factors such as radioactive decay of the isotope during travel downwind were not considered. Based on actual measurements and collected data, calculations indicated that during the fire at the BFNP the amount of radionuclides released to the environment was below the plant Technical Specification limits.

Health Physics Supervisor F stated that no radiological overexposure to personnel occurred as a result of the fire.

Following the BFNP fire the health physics technicians completed a list of individuals who would have been the most likely individuals to receive an internal exposure. This list was based on the assumption that the individuals had been in the Unit 1 reactor building during the fire and had a considerable amount of soot on their clothing and body. Whole body counts were performed on all individuals on the list. Results of the whole body counts were received on March 24, 1975, and they showed no indication of the deposition of radioactive material. In addition to the plant personnel, two construction personnel had been in Unit 3 during the evening shift on the day of the fire, and the results of the whole body counts on these men indicated no internal deposition.

On the day of the fire, the plant personnel were provided with personnel monitoring using the monthly film badges and a pair of indirect reading dosimeters. Each of the Athens Fire Department personnel were issued visitor film badges and a pair of indirect reading dosimeters when they arrived onsite. The film badges and dosimeter results indicated that no plant personnel exceeded the Browns Ferry daily radiation limit of 50 millirems and the Athens Fire Department personnel received no detectable radiation exposure (10 millirem is the minimum detectable exposure for film badges.) The film badges recorded the exposure of the plant personnel for the entire month and although two readings exceeded 50 millirem for the month these were attributed to other work assignments. The number of film badge readings that exceeded 10 millirems for the month of March was essentially the same as the previous month.

Radiation surveys performed during the period of the fire revealed no unusual dose rates. No plant personnel entered any high radiation areas and a study of the situation indicated no reason for any entry.

2. Health Physics Log

The Health Physics Log provided information as follows: Fire was reported in the cable spreading room at 12:40 p.m. on March 22, 1975. Several area monitors alarmed. Radiation surveys revealed no abnormal levels. The area monitors had apparently malfunctioned due to loss of power. An air sample at the Unit 1 control room corridor at 2:30 p.m. showed 1.03×10^{-9} microcuries per milliliter gross beta-gamma. All personnel fighting the fire were in self-contained respirators. An air sample in the Unit 1 and Unit 2 control room at 3:25 p.m. showed 1.5×10^{-9} uCi/ml beta-gamma. Unit 2 constant air monitors at the 617 ft. elevation showed 8000 counts per minute during the period 1:45 to 2:00 p.m. decreasing to 2000 to 4000 c/m at 4:00 p.m.. Thirteen (13) air samples taken from 2:53 to 6:30 p.m., in various areas including the area of the fire, showed levels of 1.34×10^{-9} uCi/ml to 7.5×10^{-8} uCi/ml. Later counts of these samples after decay showed decreased by a factor of 600 to a 1000 indicating the presence predominantly of rubidium-88. This was proven to be true by isotopic identification. An air sample on March 26, 1975, at the 593 ft. elevation in Unit 1, location of the fire, showed 8.1×10^{-10} uCi/ml beta-gamma on a particulate filter and 2.5×10^{-10} uCi/ml on a charcoal gaseous filter. (12,000 c/m on the constant air monitors corresponds to 3×10^{-9} uCi/ml. The maximum permissible concentration for rubidium-88 in 10 CFR 20 is 1×10^{-6} uCi/ml.)

3. Radiation and Contamination Surveys

Records of radiation and contamination surveys showed no abnormal radiation readings and no abnormal contamination levels. Recorded levels were less than 100 d/m/100 cm^2 in areas of concern. The maximum air sample result for samples collected during the entire incident was 3.56×10^{-7} uCi/ml and this was identified as rubidium-88.

4. Whole Body Counts

Whole body counts were performed on fourteen individuals. Review of records of these counts revealed no evidence of any radioactivity uptake.

5. Reactor Zone Stack Samples

(a) The continuous gas monitor and the fan for this stack became inoperative during the fire and hourly manual gas sampling was commenced. The results were documented in plant survey records as follows:

March 22, 1975

Time	uCi/sec
1.00 p.m.	86
2.00 p.m.	33
3:00 p.m.	24.5
4:00 p.m.	2833
5:00 p.m.	2814
6:00 p.m.	2195
7:00 p.m.	4352
8:00 p.m.	9019
9:00 p.m.	8044
10:00 p.m.	4971
11:00 p.m.	4982
12:00 p.m.	7175

March 23, 1975

Time	uCi/sec
1:00 a.m.	8536
2:00 a.m.	9640
3:00 a.m.	6401
4:00 a.m.	2700
5:00 a.m.	2054
6:00 a.m.	154
7:00 a.m.	263
8:00 a.m.	245
9:00 a.m.	607
10:00 a.m.	358
11:00 a.m.	411
12:00 Noon	624
1:00 p.m.	301
2:00 p.m.	558

3:00 p.m.	3554
4:00 p.m.	6708
5:00 p.m.	3183
6:00 p.m.	62

These values may be compared to the Technical Specification limit of 130,000 uCi/sec. At 4:00 p.m. on March 22, 1975, the fan for the reactor zone stack was returned to service. At 11:00 a.m. on March 23, 1975, the monitor for the stack was returned to service.

(b) Particulate and charcoal samples were also analysed from the reactor zone stack. The results were as follows:

<u>March 22, 1975</u> (1 sampling)

Particulate - 1.8 X 10^{-8} uCi/ml, beta-gamma
Charcoal - 3.7 X 10^{-9} uCi/ml, iodine

<u>March 23, 1975</u> (4 samplings)

Particulates - range of 2.1 X 10^{-8} to 5.46 X 10^{-8} uCi/ml beta-gamma
Charcoal - range of 3.7 X 10^{-9} to 5.09 X 10^{-9} uCi/ml iodine

6. <u>Stack Monitor</u>

The control room chart for the stack gas monitor was reviewed. Between 12:45 and 4:30 p.m. on March 22, 1975, the reading dropped from 700 counts per second down to 10 counts per second. There was no input to the stack during this period. The reading remained at 10 counts/sec from 4:30 to 6:20 p.m. At 6:30 p.m. the reading increased to 1000 counts/sec and remained at this reading until 10:00 p.m. This increase occurred because input to the stack had been resumed and in effect the holdup pipe was now being purged to the stack. During the period from 10:00 to 10:30 p.m. the reading dropped to 100 counts/sec and thereafter continued a gradual decrease on back down to 10 counts/sec ambient. The 700 counts/sec would be a normal reading during reactor operations and the 10 counts/sec a normal reading while not operating.

7. <u>Stack Particulate and Charcoal Samples</u>

The stack particulate and charcoal samples in operation from midnight on March 22, 1975, to 2:50 p.m. on March 26, 1975, gave results as follows:

Particulate	Charcoal
1.8×10^{-5} uCi/sec beta	I-131 3.8×10^{-2} uCi/sec
7.5×10^{-6} uCi/sec gamma	

The values are below the Technical Specification release limits.

8. Environmental Samples

Special environmental air particulate samples were started by the licensee in the environs around BFNP at 5:02 p.m. on March 22, 1975, and continued until 11:50 p.m. Twenty-one particulate air samples were taken in the sectors surrounding the plant site with at least two samples in the prevailing wind direction from the site. The sample results showed a range of 1.5×10^{-11} uCi/ml to 8.5×10^{-10} uCi/ml gross beta. Four additional particulate air samples, which were collected from permanent environmental samples operating from March 17-24, 1975, showed 1.1×10^{-13} uCi/ml to 1.3×10^{-13} uCi/ml beta and 0.57×10^{-14} uCi/ml to 0.60×10^{-14} uCi/ml iodine. Two of the four samples were being collected during and after the fire at BFNP in the predominant wind direction at 0.98 and 1.7 miles, respectively, from the plant stack. These values do not differ greatly from routine environmental sample results and approximate background levels.

9. Reactor Water Isotopic Analysis

Licensee records of the results of reactor water isotopic analysis were reviewed. Exhibit A11 provides results for samples collected on millipore filter paper, cation filter paper and anion filter paper, respectively. The eighteen isotopes identified and quantified from the anion filter paper did not show changes that would indicate increased or excessive fuel leakage. The ten isotopes identified and quantified from the anion filter paper similarly showed increases that would not indicate major fuel damage. Reactor water iodines showed increases not too different from the last previous scram shutdown.

K. Injuries Resulting from Fire

Only minor injuries were sustained during the fire. One person received a possible hairline fracture of the wrist and ten people were treated for smoke inhalation by the plant and construction medical services and were released. Other injuries were minor and required no treatments.

Details II

Operational Events and Problems Experienced Until Shutdown Cooling Established

A. Effect of Fire on Unit 1 Operation

At 12:00 noon March 22, 1975, Unit 1 was generating 1098 Mwe. Shift Engineer R was in charge of Units 1, 2 and 3, Assistant Shift Engineer G and Operator M were assigned to the Unit 1 control room. Shift Engineer R and Assistant Shift Engineer G have senior reactor operator (SRO) licenses and Operator M has a reactor operator license.

Operator M received the telephone call on the fire telephone from Assistant Shift Engineer L and Operator M began announcing over the public address (PA) system that a fire existed in the reactor building at an unknown location.

Operator M stated to the investigators that as he was making the fire announcement over the PA system, he thought that the fire might affect Unit 1 so he began "walking" the control console looking for abnormalities. Several minutes (possibly 5 to 6) after 12:35 p.m., the first alarm was received on Panel 9-3 which contains the controls and associate instruments for the Emergency Core Cooling Systems.[1/] This alarm indicated "Reactor Low Level Auto Blowdown Permissive." Operator M checked Panel 9-5 which contains instruments indicating vital reactor parameters (water level, pressure, steamflow, feedwater flow, power level, neutron monitors, etc.) and found all parameters normal.

The second alarm received was "Core Spray (CS), Residual Heat Removal (RHR) Pumps Running." Operator M stated that he then scanned Panel 9-3, the panel containing controls for the ECCS, and found that there was no indication that the CS or RHR pumps were running.

The third alarm received was "Core Cooling System Diesel Generator Initiate." Operator M called an operator assigned to Unit 2, Operator KK, and asked him to determine if the diesels were running. Assistant Shift Engineer L called Operator M asking for fire fighting help and Operator M sent Operator LL to help fight the fire. Operator M asked Assistant

[1/] The ECCS consists of the Automatic Depressurization System, The High Pressure Injection System, the Core Spray System, and the Low Pressure Coolant Injection mode of the Residual Heat Removal System.

Shift Engineer L for help in the Unit 1 control room. Assistant Shift Engineer L passed this information to Shift Engineer R and Assistant Shift Engineer G who were in the reactor building and Assistant Shift Engineer G immediately went to help. Operator M also called Unit 3 and asked Operator MM to come to Unit 1 control room to help him.

Operator M stated that alarms from shutdown panels came in indicating that RHR, CS, high pressure coolant injection (HPCI) and reactor core isolation cooling (RCIC) pumps were all running. Operator M, observing that Panel 9-5 still showed normal reactor parameters, tripped these pumps.

Assistant Shift Engineer G arrived in the Unit 1 control room and observed that RCIC, HPCI, LPCI and CS were automatically initiated, automatic depressurization system (ADS) alarms were annunciating and the ADS timers were running.

Both Operator M and Assistant Shift Engineer G then observed that the recirculating pumps were "running back" causing a reactor power decrease as indicated by the average power range monitors (APRM's) decreasing.

Shift Engineer R arrived in the Unit 1 control room at about this time. At approximately 12:48 p.m., RHR, CS and HPCI initiated again and random lights on Panel 9-3 began getting abnormally bright and then going dim or out. Assistant Shift Engineer G and Operator M then tried to shut down the RHR and CS pumps. Operator M decided to scram Unit 1 and this was discussed briefly with Shift Engineer R and Assistant Shift Engineer G. Operator M zeroed the master manual control for the recirculation pumps and had begun to zero the individual controllers when the recirculating pumps tripped.

Operator M immediately manually scrammed the reactor at 12:51 p.m. from a power level of 704 Mwe.

At approximately 12:53 p.m., Assistant Shift Engineer G tripped the turbine and Operator M tripped two feedwater pumps and ran the level setpoint back to zero. Operator M confirmed that all control rods had inserted and started driving in the intermediate range monitors (IRM's) and source range monitors (SRM's).

Operator M stated that the reactor water level dropped to +5 inches and returned to +47 inches at which time he adjusted the level setpoint to the normal +33 inches. One feedwater pump was kept running to control reactor water level and a turbine bypass valve was opened to provide a path to the condenser for a heat sink.

The HPCI and RCIC had started automatically on low reactor water level immediately after the scram. Since reactor water level subsequently increased and was high (+47 inches) both HPCI and RCIC were manually tripped. Operator MM had arrived in the control room at approximately 12:51 p.m. and tripped RCIC at Operator M's request. Assistant Shift Engineer G tripped HPCI. Operator MM checked the diesels and found "C" and "D" running and tied to the shutdown boards and "A" and "B" idling and ready to tie in. At 1:00 p.m., Unit 2 was scrammed and Operator MM left Unit 1 to assist on Unit 2.

Assistant Shift Engineer G stated that the 250 Volt DC control boards and MOV boards, unit preferred system, and both reactor protection buses tripped.

At 1:03 p.m., the main steam isolation valves (MSIV's) closed. Operator M informed the investigators that he thought the reactor feed pump was running until the MSIV's closed, but the Sequence of Events Recorder Printout shows the feed pump tripped at 12:56 p.m.

Prior to the MSIV's closing, power to the following electrical boards had been lost:

1A and 1B 250 V MOV Boards

1A, 1B, and IE 480 V Reactor MOV Boards

1A and 1B 480 V Shutdown Boards

120 V Unit Preferred Power

Shutdown Bus No. 1

The TVA "Sequence of Significant Operational Events Report," (Exhibit B1) contains an entry indicating that at 12:56 a.m., the only energized board remaining was the 1C 250 V Reactor MOV board which provides control power to four relief valves (RV's).

Assistant Shift Engineer G stated that the RCIC lights on Panel 9-3 (ECCS control panel) were still "on" when the MSIV's closed but the valves would not function and RCIC could not be started. Lamps on Panel 9-3 indicated that HPCI was not operable.

When the MSIV's closed, reactor pressure increased and RV's opened automatically to control reactor pressure. (The lowest pressure at which a valve is set to open is approximately 1080 psig.) The recorder chart indicates that a pressure of 1060 psig was reached. Each nuclear steam supply system is provided with two safety valves and eleven relief valves which are self actuated on high steam pressure. The eleven relief valves are also pneumatically actuated and may be manually controlled from the control room. Six of the eleven valves form the ADS and may be automatically controlled as a part of the ECCS system. These six valves are provided with an air accumulator to assure that they can be operated and held open on the loss of the air supply. The accumulators are sized for a minimum of five valve operations.

The reactor coolant system had one remaining source of high pressure water, the control rod drive (CRD) system. Operator M increased the CRD pump output to its maximum by adjusting the flow controller. Although the control room instrument has a maximum scale reading of 100 gpm, Operator M stated that he knew that the CRD pump was pumping greater than 100 gpm but he does not know how much was being pumped. Operator M advised the investigators that he did not think that starting the spare CRD pump would have resulted in significantly greater injection flow, and he recalls that the spare CRD pump was not always operable during the shutdown. (Tests of the motor of the spare pump later indicated low resistance.) Operator M considered using the standby liquid control system (SLCS) as a method of reactor shutdown but did not since he had indication that the rods were driven in and did not require the use of poison for shutdown. He did not consider the system as a source of high pressure water to aid in cooldown. (Information developed by the investigators reveals that power was lost to the SLCS pumps and to the explosive actuated valves for a period of approximately three hours.)

Prior to 1:00 p.m., Assistant Shift Engineer L called and stated that the fire was in the cable spreading room. Assistant Shift Engineer G initiated the "Cardox" alarm to clear personnel from the cable spreading room and left the Unit 1 control room to initiate the "Cardox" system.

Between approximately 12:55 and 1:15 p.m. Operator M observed that the nuclear instrumentation and approximately one-half of the CRD position indications were inoperable. At this time, remote manual control was effective for only four RV's. He recalls observing that the condensate and condensate booster pumps were operable at this time.

The Plant Superintendent stated that the first steps of the Browns Ferry Emergency Plan were implemented at approximately 12:55 p.m. when he was called. He stated that he made the calls to fully implement the plan. The Plant Superintendent notified the Director, Division of Power Production (DPP) of the plant situation at approximately 2:40 p.m. The Nuclear Operations Coordinator, DPP, called and he and the Plant Superintendent agreed that the TVA Emergency Plan should be fully activated. At 3:10 p.m. the TVA Central Emergency Control Center in Chattanooga, Tennessee, was manned and U.S.N.R.C. and the State of Alabama were subsequently notified.

Operations Supervisor DD arrived at the plant at approximately 1:20 p.m. The Plant Superintendent arrived between approximately 1:20 p.m. and 1:35 p.m. Assistant Operations Supervisor W arrived at approximately 1:35 p.m. Operations Supervisor DD, Assistant Operations Supervisor W and the Plant Superintendent each went immediately to the Unit 1 control room upon arrival.

The control room is on the floor above the cable spreading room and when the "Cardox" system dumped CO_2 into the cable spreading room, fumes and smoke were forced into the control room through unsealed penetrations in the floor between the two rooms. Operators M and MM put on Scott air packs for about five minutes during this period. Other personnel went through this period in the control room without using breathing apparatus. An air hose was later brought into the control room and discharged into the room to help provide fresh air.

Numerous attempts were made to restore electrical boards so as to restore the normal capability for providing reactor and torus cooling:.

(1) Assistant Shift Engineer G states that after initiating the "Cardox" he went to the 480 V shutdown boards in the control room where he reset the breakers and heard them humming from a heavy load, but the breakers did not trip.

(2) Assistant Shift Engineer G stated that he went to the 1B reactor MOV board with electrician foreman to try to get power to a needed RCIC valve FCV 71-2,but the valve had a "dead fault" and could not be operated electrically. (They were unaware that Shift Engineer R had previously tried to do the same thing.)

(3) Assistant Shift Engineer OO stated that at approximately 2:00 p.m., Unit 2 lost preferred power and he found that someone had tied Units 1 and 2 preferred power boards together. He separated the two buses and tied the Unit 2 bus to the transformer.

(4) Assistant Shift Engineer OO stated that at approximately 2:00 p.m., the "C" 4160 V shutdown bus was dead. He tried to tie the "C" diesel to the bus but could not. The problem appeared to be in transformer TS-1B which serves 480 volt Shutdown Board 1B.

Operator M stated that at approximately 1:30 p.m. he knew that reactor water level could not be maintained with the CRD pump and was convinced that he must blowdown (reduce pressure) to approximately 350 psig in order to pump water into the reactor using the condensate booster pumps. Operator M discussed this briefly with Assistant Shift Engineer L (who was assigned to Unit 2). Assistant Shift Engineer L concurred that Unit 1 should be blown down. Assistant Operations Supervisor stated that he conferred with Operations Supervisor DD and the Plant Superintendent and that they agreed that blow down to 350 psig should be initiated and to use the condensate booster pumps to keep the core covered.

The Plant Superintendent stated that the decision to blowdown was his and was concurred in by Operations Supervisor DD. Operations Supervisor DD stated that he ordered the operator to blowdown. (He advised the investigators; however, that the order was actually given to Assistant Operations Supervisor W and not Operator M.) Assistant Operations Supervisor W stated that Operations Supervisor DD had told him in Operator M's presence to continue to blow reactor pressure down.

In addition, Assistant Operations Supervisor W, Operations Supervisor DD, and the Plant Superintendent discussed the use of river water as a back up water source in the event that the condensate pumps failed. Operations Supervisor DD stated that the RHR service water system could provide water to the reactor at a pressure of approximately 150 psig, and that two valves would have to be manually opened. These two valves are in the reactor building near the elevator shaft in an area where the smoke was not so dense as to prevent access.

Operator M stated that in order to pump from the condensate system into the reactor he had to open feedwater heater isolation valves which he could do from the control room (Later at 2:15 p.m., Operator NN opened the breakers to these valves to make certain that the valves would not close). The condensate and condensate booster pumps were operable from the control room. Operator M had an operator open the bypass line around the condensate demineralizers and Operator M opened the bypass around the feedwater heaters. Operator M determined that two condensate pumps (out of three) and one condensate booster pump (out of three) were running. After discussing the desirability of blowing down with Assistant Shift Engineer L, Operator M at approximately 1:40 p.m., initiated blowdown using the four RV's that could be manually actuated from the control room.

Although Operator M did not receive direction from his supervisors to initiate blowdown nor did he request their concurrence, the Plant Superintendent as well as Assistant Operations Supervisor W and Operations Supervisor DD were present when blowdown was initiated and, consistent with their discussion, permitted the blowdown.

Operator M watched the indicated reactor water level on Yarway Instrument LI-3-46A&B and Operator MM operated the RV's. Operations Supervisor DD as well as others stated that they were observing the reactor water level during this period since there was much concern about the possibility of the water dropping below the top of active fuel (TAF). Operator M recalled that when blowdown began the reactor water level was at -20 to -30 inches (approximately 148 inches above TAF). Operator M recalled that he was encouraged when only five inches decrease in indicated level was seen after depressurizing approximately 200 psig. Depressurization to approximately 260 psig took about 20 minutes. The minimum indicated reactor water level observed was -100 to -120 inches (about 48 inches above TAF). Operations Supervisor DD and Assistant Operations Supervisor W recalled the water level responding such that it appeared that water was being added to the vessel when pressure reached approximately 350 psig. The water level increased to normal (+33 inches as indicated on the level instrumentation or about 200 inches above TAF), but during this time control of the feedwater pump bypass valve, FCV-3-53, was lost and water level increased to greater than +60 inches. According to Assistant Operations Supervisor W the water temperature entering the vessel was 70 to 80°F. It is not known how high the level rose in the vessel during the period between approximately 2:00 and 3:15 p.m. since full scale on LI-3-46A is +60 inches, and this recorder chart indicates that at approximately 3:15 p.m. level was back "on scale." Operator NN was sent to the turbine building to partially close FCV-3-53 manually in order to control the addition of the water. An Assistant Unit Operator was stationed at FCV-3-53 to make valve adjustments as directed by Operator M in the control room. The relief valves were kept open and the reactor water level was controlled above the normal setpoint of +33 inches.

After condensate flow to the reactor was established the major concern was to establish torus cooling and shutdown cooling using the RHR as quickly as possible.

Operator M stated that he made a list of RHR valves needed to obtain torus cooling. He further stated that at approximately 3:15 p.m. all torus temperature and level instrumentation was inoperable. A survey by GE representatives (Exhibit B2) at 2:45 p.m. indicated the following:

Reactor Pressure: 300 psig

Reactor Water Level: +60 inches (full scale instrument reading)

Equipment Inoperable: All ECCS; MSIV's; Seven RV's; Reactor Building Closed Cooling Water (RBCCW); Reactor Water Cleanup (RWCU); Standby Gas Treatment (SBGT) Train "B"; Eight of Ten Dry Well Blowers; Diesel "C."

Instruments Inoperable: torus temperature and level; drywell temperature; jet pump flow; reactor flange temperature; all neutron instruments; reactor protection instrument system, one of two buses; GE/MAC level, one of three; CRD instrument panel; computer; main steamline radiation monitors.

From about 2:00 p.m. until the fire was extinguished several attempts were made to enter the reactor building and manually align the RHR for torus cooling and shut down cooling modes:

(1) Between approximately 2:00 p.m. and 4:30 p.m., Assistant Shift Engineer G directed several efforts resulting in opening RHR valves 74-73 and 74-71, required for torus cooling.

(2) Operator MM and Assistant Operator AA worked to line up the RHR drain pumps to pump from the torus to the condenser hotwell. They had discharge valve 74-62 open at approximately 6:30 p.m.

(3) Between approximately 2:30 and 3:30 p.m., Assistant Shift Engineer EE sent two operators to open a suction valve on an RHR pump. Three entries were made without success.

(4) At about 4:00 p.m., Assistant Shift Engineer EE got MOV Board 1A restored to service and restored RPS MG Set 1A to service between 4:30 p.m. and 6:30 p.m.

(5) At approximately 4:30 p.m., Assistant Unit Operator Z and another operator made three entries before partially opening a RHR service water valve supplying coolant to a RHR heat exchanger. Power was restored to the valve at about this time.

None of these attempts resulted in establishing torus or reactor shutdown cooling. The attempts were severely limited by dense smoke and inadequate breathing apparatus.

Until about 6:00 p.m. reactor water level was being controlled at about +50 inches using the condensate and condensate booster pumps with a RV open to bleed steam from the reactor. Reactor pressure was 150-300 psig. At this time remote manual control of the last four RV's was lost. The investigators were advised that with the reactor water cleanup system inoperable there were no other means of manually bleeding steam or water from the reactor vessel. (The main steam line drain valves were not operable from the control room and, because of the dense smoke, were inaccessible for manual operation.) Reactor pressure increased as follows:

6:40 pm - Pressure = 300 psig
6:55 pm - Pressure = 400 psig
7:00 pm - Pressure = 420 psig
7:30 pm - Pressure = 460 psig
8:20 pm - Pressure = 540 psig
9:30 pm - Pressure = 600 psig

As pressure increased above 350 psig the condensate booster pumps could not inject water into the vessel and only the CRD pump was adding water. (In the event that the Unit 1 CRD pump became inoperable the Unit 2 CRD pump could have been valved to supply Unit 1, but the operators were unaware of this fact.) Reactor vessel leve remained at about +50 inches during the pressure buildup. Assistant Unit Operator Q and an electrician checked to determine if control power to the RV's had been lost but found that it had not been. At about 7:00 p.m. Assistant Shift Engineer QQ, Operator NN and Assistant Unit Operator RR were sent to open the valve to vent the drywell to the gas treatment system so as to relieve drywell pressure. They wired the valves open. Assistant Operations Supervisor W stated that the 2" valve was opened at 8:40 p.m.

Shift Engineer PP was sent to check the air supply to the RV's. He found that a solenoid valve had failed closed due to the fire damage to the electrical cables. This solenoid closure cut off control air supply to the drywell air compressor flow control valve FCV-32-27 (FSAR identification) causing the valve to close and in turn cutting off the air supply to the relief valves. (An alternate source of air to the relief valves could not be utilized since it feeds through the same control valve as the normal supply.) With the aid of an electrician, Shift Engineer PP bypassed the solenoid by connecting the control air directly to the flow control valve and opened it so that air could be supplied to the RV's.

Assistant Operations Supervisor W reported that at about 9:50 p.m. control of the relief valves was restored. The reactor was then depressurized slowly from about 600 psig to less than 350 psig in about 30 minutes at which time the condensate booster pumps again pumped water into the reactor.

During depressurization the reactor water level as indicated by LI-3-46 dropped below 0 inches (approximately 168 inches above TAF) for about 10 minutes after which it was raised to +30 inches in about 4 minutes. Due to the lower decay heat and resulting boiloff rate at this time it is concluded that the reactor water level did not drop as far as it did during the initial depressurization.

The fire was declared "out" at 7:45 p.m. As the smoke cleared and the reliance on breathing apparatus lessened, a more orderly approach to obtaining RHR torus and reactor shutdown cooling was taken: actual valve conditions (opened or closed) were determined and control power to motor operators, pump controls, etc., was established using temporary jumpers. The Electrical Maintenance Supervisor stated that as wires were lifted or contacts jumpered the steps were written down and later put in a log as required by the plant Standard Practice BFA-25 Operator M stated that control of the reactor water cleanup system valves was restored at about 8:00 p.m. Torus cooling was established at 1:30 a.m. and shutdown cooling established at 4:10 a.m. on March 23, 1975.

At 1:00 a.m. on March 23, 1975, source range nuclear instrument indications were established locally in the reactor building. These were calibrated by an instrument engineer and indicated 10 counts per second. An operator was stationed at the local readout in telephone contact with the control room with instructions to notify the control room of any increase in counts.

What was known about drywell temperature and pressures and torus temperatures and pressures is described in Exhibits B1 and B2. The drywell had been vented to the SGTS at 8:30 p.m. on March 22, 1975.

B. Effect of Fire on Unit 2 Operation

At 12:00 noon on March 22, 1975, Unit 2 was generating 1098 Mwe Assistant Shift Engineer L and Operator KK were assigned to Unit 2. Assistant Shift Engineer L has a SRO license and Operator KK has a reactor operators license.

Operator KK heard the fire alarm at 12:35 p.m. and the fire announcements on the PA system that followed. Unit 1 and 2 control panels are in the same room and throughout the period he could see and hear the activities at the Unit 1 control panels.

At approximately 12:50 p.m. two annunciations were received on the 9-7 (Turbine Control) panel: "Steam Jet Gas Ejector-Offgas Filter delta-P High" and "Offgas Dilution Air Flow Low." Operator KK believed these alarms to be erroneous. Operator KK knew when Unit 1 was scrammed.

At approximately 1:00 p.m. Operator KK observed numerous annunciators associated with DC power failure and that a one-half scram existed on the reactor protection system (RPS) due to an M-G set failure. Operator KK walked over to scram the reactor, but he is not certain whether it scrammed automatically or if it scrammed when he pushed the scram button and turned the mode switch to shutdown.

An instrument engineer arrived at the Unit 2 control panels a few minutes before it scrammed and recalled noticing malfunctions on ECCS Panel 9-3 and the feedwater panels. Immediately before the scram, Operator KK asked him to switch the reactor building fans to "low" which he did and returned to the control panels about the time Unit 2 scrammed.

Operator KK stated that he immediately confirmed that all rods inserted and the control rod indicators appeared normal for the scrammed condition. Assistant Shift Engineer G came from Unit 1 and tripped the turbine at approximately 1:01 p.m. The reactor water level behaved normally: dropping to about 0 inches (about 160 inches above TAF) on the GE/MAC instrument. Yarway level indicator LI-3-52 was inoperable, but Yarway LI-3-62 appeared normal before and after the scram. The sequence of events recorded indicates that HPCI and RCIC may have initiated automatically on low level and tripped at 1:03 p.m. by high water level.

At approximately 1:03 p.m. the reactor water level increased to +40 inches (about 200 inches above TAF) and Operator KK tripped all three feedwater pumps. Operator MM came to Unit 2 from Unit 1 and offered his assistance. After the scram the instrument engineer observed that RPS was not operating.

At 1:03 p.m. the MSIV's closed. The instrument engineer thought the isolation was initiated by a high steamline tunnel temperature but the Sequence of Events Recorder printout indicates the isolation occurred when the other one-half of the RPS malfunctioned.

At approximately 1:10 p.m. RCIC was initiated manually and HPCI was initiated manually in the recirculation mode (no injection into the reactor). HPCI was initiated in this mode to relieve steam from the reactor. There was no problem in controlling water level with RCIC. The CRD pump was operating.

At this time the RV's were operating automatically and could be actuated manually. At approximately 1:20 p.m. manual actuation capability of the RV's was lost, but automatic operation continued at approximately 1020 psig. RCIC and the CRD pump continued to pump water into the reactor to control water level.

At approximately 1:20 p.m. diesel generator "D" tripped. This coupled with the loss of power to a 480 V shutdown board resulted in the loss of power to all 480 V shutdown and reactor MOV boards for approximately 45 minutes. Diesel Generator D could not be operated from the control room. At approximately 10:30 p.m. Assistant Shift Engineer EE got the diesel started and loaded from 4KV shutdown Board D.

Assistant Shift Engineer Y and others worked on restoration of RV manual actuation capability. It was found that isolation valves in the drywell control air supply line had isolated due to the loss of instrument air control bus "A." Shift Engineer V arrived onsite at approximately 1:40 p.m. and directed his full attention to Unit 2 problems. The air supply to the RV's was restored by approximately 2:15 p.m.

At approximately [illegible]:10 p.m., the reactor began to depressurize, apparently because a RV had stuck open. When manual actuation of the RV's was restored at approximately 2:15 p.m. Shift Engineer V told Operator KK to continue to depressurize while they still had RCIC, and before Unit 2 systems degraded further. Shift Engineer V stated that the Plant Superintendent asked him several times if he had the unit under control.

At approximately 2:30 p.m., all reactor level indication except LI-3-62 was lost for approximately ten minutes.

At approximately 2:30 p.m. RHR pump "D" was placed in the torus cooling mode.

At approximately 3:00 p.m. the RHR drain pump was initiated to control torus water level.

After approximately 3:00 p.m. the reactor pressure was less than 200 psig. The condensate booster pump was used prior to 4:10 p.m. to pump water into the reactor with the same system alignment as used Unit 1. At approximately 4:00 p.m. a main steam drain line was opened to the main condenser. Difficulties were encountered getting the mechanical vacuum pumps on, but this was accomplished by about 7:00 p.m.

At approximately 6:30 p.m., Unit 2 conditions were stabilized and Shift Engineer V then directed his attention to fire fighting. At about 10:45 p.m. RHR shutdown cooling was established using RHR pump B.

Exhibits B1 and B2 provide additional detail developed by TVA and GE personnel concerning events which took place on Unit 2 including details of systems lost during the incident.

DETAILS III

Effects on Each Unit and Interactions Between Units

A. Introduction

Prior to the fire Units 1 and 2 were each generating 1098 MWe. The electrical systems were normal with the unit common power buses for each unit being fed from their respective station service transformers. There were no known problems with any of the electrical systems at the time.

Although the fire began in a cable tray penetration connecting the cable spreading room with the Unit 1 area of the reactor building and was confined to the penetration area and the Unit 1 reactor building, cables associated with both Units 1 and 2 in the conduit and trays located in the fire area were destroyed. The fire caused the loss of the ability to operate all Unit 1 Emergency Core Cooling Systems (ECCS)[1] and the partial loss of Unit 2 residual heat removal (RHR) and core spray (CS) systems. Other systems were available for Unit 1 cooldown and were used.

Exhibit C1 describes the extent of the damage to the electrical cables in the fire. Four hundred eighty-two safety related cables for Unit 1, 22 safety related cables for Unit 2, and 114 safety related cables common to both units were damaged.

Exhibit C2 is a tabulation of the electrical boards and their loads for Units 1 and 2. Figure 1 of Exhibit C2 is a single line schematic diagram of the electrical systems required for a unit shutdown. The power to the 4KV unit boards is supplied from the Units 1 and 2 main turbine generators through the Unit Station Service Transformers TUSS 1 and TUSS 2, respectively, or from the 161KV offsite power sources through Common Station Service Transformers TCSSA and TCSSB. There is an automatic high speed transfer from the Unit Station Service Transformers to the Common Station Service Transformers when the main turbine generator trips or when the safety system logic circuitry output indicates an accident condition.

The Engineered Safeguards Systems (ESS) are supplied from four 4KV Shutdown Boards through two shutdown buses. Shutdown Bus 1 normally receives power from Unit Board 1A; Shutdown Bus 2 normally receives power from Unit Board 2A.

The Engineered Safeguards Systems (ESS) are separated into two divisions, I and II, as follows:

1. The Automatic Depressurization System (ADS) electrical circuits are in Division I while the HPCI system circuits are in Division II.

2. The "A" and "C" pumps and associated valving in the CS and RHR systems controls are in Division I and the "B" and "D" units are in Division II.

1/ The ECCS includes the Automatic Depressurization System (ADS), the High Pressure Coolant Injection System (HPCI), the Low Pressure Coolant Injection (LPCI) mode of the Residual Heat Removal System (RHR) and the Core Spray System (CS).

3. The "A" MOV Board loads are in Division I and the "B" MOV Board loads are supplied from Division II. The "C" MOV Boards have automatic transfers so they may be fed from either Division.

This manner of separation and load assignments results in 4KV Shutdown Boards A and B supplying power to Division I and 4KV Shutdown Boards C and D to Division II of the Engineered Safeguards Systems. Each of the 4KV Shutdown Boards supply power to equipment for both Units 1 and 2. The loss of a single board, therefore, affects both units and the partial loss of Unit 2 RHR and CS systems is attributed to this fact.

The distribution of the various loads on the 4KV Shutdown Boards, the 480 Volt Shutdown Boards, and the 480 volt Reactor Building Motor Operated Valve (MOV) boards are given in Tables I through III of Exhibit C2. Table IV of Exhibit C2 is a listing of the 250 VDC boards. Table V of this exhibit is a summary of the auxiliary power supplies and bus transfer schemes.

The interaction of the cable fire on the operation of Units 1 and 2 is summarized based on reports by TVA personnel, the Electrical Sequence Events printout, the operations logs for March 22 and discussions with TVA personnel. The descriptions of certain of the conditions are repeated in the evaluations of both Units 1 and 2 because there were effects on both units from single events or conditions.

B. Effects of Cable Fire on Unit 1 Operation

The reactor was manually scrammed at 12:51 p.m. (Details II). Power from the Unit 1 preferred bus was lost at 12:55 p.m. due to fire damage.[1/] The unit preferred bus provides a source of power to instrumentation and controls that require a transient-free source of power. Transient-free power is accomplished through the use of a motor-motor-generator (MMG) set with a flywheel. The generator is a 240 VAC single phase unit and is normally driven by a 480 VAC motor. When the AC power source is lost, the flywheel inertia is used to maintain the generator speed until the backup motor, a 250 VDC unit, is automatically energized and drives the generator. Typical loads on the unit preferred system are the plant computer, rodworth minimizer, feedwater control, and rod position indication. At approximately 12:55 p.m., power from Instrumentation and Control (I&C) bus 1B was also lost due to fire damage. (Although cables to the computer were destroyed, the computer apparently escaped damage.) One consequence of the failure of the preferred power buses was a false high reactor water level signal to the Reactor Feed Pump (RFP) control. RFP "A" was tripped from this false high level signal. RFP "B" and "C" had been manually tripped at the time the reactor was scrammed. The HPCI system was not operable because of fire damage to 250 VDC MOV Board 1A which supplies power to HPCI valve controls. Reactor Core Injection Cooling (RCIC)

1/ A tabulation of the major Unit 1 systems lost, together with the probable cause, is contained in Exhibit C3.

system was not operable because of power failures on 480 Volt Reactor MOV Board 1B which supplies power to the RCIC valves. When RFP "A" tripped, the Control Rod Drive (CRD) system was the only remaining operable system capable of delivering water to the reactor pressure vessel at a pressure greater than 350 psig; therefore, it was necessary to reduce reactor system pressure to permit coolant to be delivered by other systems.

At approximately 1:13 p.m., a cable fault occurred in the power feed from 4 KV Shutdown Board C to transformers for 480 Volt Shutdown Board 1B and 2A. This cable was in Tray AX which was in the fire area. The consequence of this power failure on Unit 1 was a loss of power to the following equipment and systems:

1. Loads on 480 V Reactor MOV Board 1B (See Table III, Exhibit C2)

2. Standby Liquid Control (SLC) Pump 1B

3. I&C Bus B transformer (There had been earlier individual failures on this I&C Bus)

4. Unit preferred transformer

5. Other loads on 480 V Shutdown Board 1B as shown in Table II, Exhibit C2

At 1:22 p.m., 480 V Shutdown Board 1A lost power because the trip circuit red light cable was damaged in the fire. A short in this cable energized the breaker trip coil. All Unit 1 480 V Reactor MOV Boards were then without a power source. With the power lost to the MOV boards, the power to the valves in both divisions of ESS for Unit 1 was lost. The Unit 1 Core Spray, the RHR, the HPCI, and the RCIC systems were not operable from the control room. The Standby Liquid Control (SLC) System did not have power to the pumps or squib valves, and both MG sets for the RPS were tripped. Only the 250 VDC Reactor MOV Board 1C remained energized and this power was used to operate four main steam relief valves to decrease the reactor pressure and thereby permit an increased core cooling capability using the condensate booster pumps.

At approximately 1:55 p.m., the unit operator determined that he could not control the reactor feed pump air operated bypass valve from the control room. The valve was found to be wide open and an assistant unit operator was stationed at the valve to open or close it as required to maintain the reactor vessel water level.

Information contained on the electrical sequence printout does not show the restoration of power to any of Unit 1 480 Volt Shutdown Boards prior to 4:30 p.m. (The electric sequence printer tape terminated at this time and it apparently was not replaced until approximately 2:00 p.m. on March 23.) However, the assistant shift engineer's log reports that 480 Volt Reactor MOV Board 1A was re-energized at about 4:30 p.m. and RPS M-G set 1A was returned to service. The restoration of MOV Board 1A would require power to 480 Volt Shutdown Board 1A or to 480 Volt Shutdown Board 1B plus a manual transfer of MOV Board 1A to 480 Volt Shutdown Board 1B.

At 6:00 p.m., the manual control of the four available relief valves was lost because there was no control air supply to the relief valves. (Automatic operation at 1080 psig was still possible.) The control air was lost because a solenoid valve that controls the air supply to the drywell flow control valve (FCV 32-27) closed due to cable damage. The closure of the solenoid valve caused a closure of FCV 32-27 which in turn cut off the air supply to the relief valves. FCV 32-27 was made operable by bypassing the solenoid valve and control air was restored to the relief valves at 9:30 p.m. The reactor pressure was then reduced using the four operable relief valves.

C. Effects of Cable Fire on Unit 2 Operations

4KV Shutdown Bus No. 2 was de-energized at approximately 1:00 p.m. by the action of the bus differential relaying which sustained fire damage in the control wiring. The normal feed breaker was tripped open and the closure of the alternate breaker was blocked because the trip had been initiated by the differential relays. The effect of the loss of power on Bus No. 2 was to de-energize 4 KV Shutdown Boards C and D and 480 Volt Shutdown Boards 1B and 2B. The normal transfer of the power source to Shutdown Boards C and D from Shutdown Bus No. 2 to No. 1 was blocked because fire damage to Unit 1 Selection logic circuitry was such that an accident condition was indicated. (See Table V, Exhibit C2).

Diesel Generators C and D picked up the loads of 4KV Shutdown Boards C and D respectively. The pickup of Shutdown Board D by Diesel Generator D was not fast enough to prevent the automatic shutdown of Safety System M-G Set 2B which receives power from 480 Volt MOV Board 2B. The unavailibility of M-G set 2B resulted in a 1/2 scram on the reactor. The momentary power interruption from Shutdown Board 2B also caused an interruption to the I&C Bus B, the power source for the Feedwater Control, and initiated a recirculation pump runback. The unit was manually scrammed and a normal shutdown initiated.

At about 1:08 p.m., fire damage caused a cable failure in the normal power feed from 4 KV Shutdown Board B to 480 Volt Shutdown Board 2A. This power feed is in Tray AX which was in the fire area. The loss of power from 480 Volt Shutdown Board 2A resulted in the trip of Reactor Protection System M-G set 2A. The unavailability of M-G set 2A completed the requirements for the isolation action of the Main Steam Isolation Valves (MSIV's) and the main steam lines could not be used to dump steam to the Unit 2 condenser.

At approximately 1:12 p.m., a cable fault occurred in the power feed from 4 KV Shutdown Board C to 480 Volt Shutdown Board 1B and 480 Volt Shutdown Board 2A. This feed is also located in Tray AX. The consequences of this cable fault on Unit 2 was a loss of power for the units and systems itemized below:

1. Loads on 480 Volt Reactor MOV Board 2A (See Table III, Exhibit C2)

2. Standby Liquid Control Pump 2A

3. AC Feed to Unit preferred MG Set 2

4. Instrument and Control Bus 2A Transformer

5. Other loads on 480 Volt Shutdown Board 2A as shown in Table II, Exhibit C2

At this point, the "B" and "D" systems of CS, RHR and RHR Service Water for Unit 2 were still functional as well as Standby Liquid Control System 2B and RCIC. There was an additional impact on the ECCS systems for Unit 2 since the valve in the common raw water supply to the Unit 1 and 2 Core Standby Coolant System receives power from 480 Volt Reactor MOV Board 1B; and without Unit 3 MOV Board 3B in service, there would have been no means of introducing Service Water to Unit 2 vessel without manually operating this valve. The deenergization of the I&C Bus 2A caused an isolation of the drywell control air supply and the manual operation of the relief valves on Unit 2 was achieved by the air stored in the accumulators of six of the relief valves. At approximately 1:40 p.m., the relief valves could no longer be operated from the control room; but they could and did relieve pressure when they reached the relief setpoints of approximately 1080 psi.

At approximately 1:20 p.m., 4 KV Shutdown Bus 1 was de-energized but this had no significant effect on Unit 2 since all the 4 KV Shutdown Boards had diesel generator power available at this time. 4 KV Shutdown Bus 2 was reenergized at 1:35 p.m. and D. G.'s were not needed since all 4 KV Shutdown Boards except "C" were supplied from Shutdown Bus 2 through either normal or alternate feeds. The supply from Shutdown Bus 2 to 4 KV Shutdown Board C was not restored for reasons unknown at this time. Diesel Generator C was still supplying power to the board.

From approximately 1:35 until 4:36 p.m. 4 KV Shutdown Board C was energized and de-energized several times, therefore, power was not always available for CS Pump 2B, RHR Pump 2B, and RHR SW Pump 2B. Since control of the valves for the "A" and "C" systems of CS and RHR had been lost earlier, only the "D" systems of Unit 2 CS and RHR were available during portions of the cooldown.

The DC power for Unit 2 valves was reported to be available at all times. There had been earlier indications that 4 KV Shutdown Board D had been de-energized but this was later determined to be a failure of indicating lights in the control room and not a loss of power.

At approximately 2:00 p.m., the unit preferred Bus No. 2 output power was lost after Assistant Shift Engineer L tied the Unit 1 and 2 buses together in an attempt to restore preferred power to Unit 1. Another member of the operating staff found that the unit preferred power bus from Unit 1 had been tied to Unit 2. He separated the two buses and tied Unit 2 bus to the transformer that is powered from 480 Volt Shutdown Board 2B.

D. Instrumentation

1. Unit 1

The following is the sequence of failures of the instrumentation for Unit 1 as a consequence of the fire. The times given are estimates and are presented only to indicate the order of events.

At 12:55 p.m. all rod position indication was inoperable apparently because Unit 1 preferred power source was lost. The operator confirmed that all rods were inserted by putting the reactor mode switch in "Refueling" and observing the white permissive light for one rod withdrawal. This white light is illuminated only when all rods are in. The total unavailability of Unit 1 preferred power would have caused the process computer to be inoperable, however, the computer continued to function.

Between approximately 12:55 p.m. and 1:15 p.m. all nuclear monitoring systems (NMS) were observed to be inoperative. These systems included the Average Power Range Monitors (APRM), the Intermediate Range Monitors (IRM) and the Source Range Monitors (SRM). (At 1:00 a.m. on March 23, two of the SRM indicators were relocated to the reactor building and connected by temporary cable to their chambers).

Shortly after 2:00 p.m. on March 22, the indication of the torus temperature and torus level became inoperative. At 11:00 p.m., torus temperature readings were taken locally. (At about 2:00 a.m. on March 23, the torus level instrumentation was returned to service).

At 2:15 p.m. on March 22, the drywell temperature recorder indication went full scale (300°F) and remained there until about 4:30 p.m. when the recorded temperature started decreasing to reach 85°F at midnight. Subsequent investigation by TVA indicates that the thermocouples were damaged in the fire and were measuring temperatures in the fire zone rather than drywell temperatures. (At 5:30 a.m. on March 23, the drywell temperature instrumentation was still out of service but readings were being taken with temporary instruments in the reactor building).

The radiation monitors that measure the effluent from the reactor building vent were found to be inoperable at about 3:30 p.m. on March 22. (They were restored to service at approximately 4:00 p.m. on March 23). Hourly "grab" samples were taken starting at 4:45 p.m. on March 22 to monitor the radioactivity of the building exhaust. These monitors receive their power from the reactor protection system bus.

A one-inch copper line in the fire zone separated at a sweated joint because of heat from the fire. This line supplied control air to valves in the Unit 1 Reactor Water Cleanup (RWCU) Demineralizer system, to valves on the closed cooling water side of the Non Regenerative Heat Exchangers of the RWCU and to valves in the suction line from the refueling floor to the Standby Gas Treatment System.

2. Unit 2

The portion of the instrumentation that was confirmed to be inoperable on Unit 2 is listed below:

a. LI-3-46A Reactor Level Yarway on panel 9-5

b. LI-3-46B Reactor Level Yarway on panel 9-5

c. LI-3-52A Reactor Level Yarway on panel 9-3

d. FR-3-32A Reactor Feedwater Flow

The above instruments failed or were erratic due to interruptions to I&C Bus A.

e. FR-1-81 Turbine Steam Flow

f. PR-3-59 Reactor Pressure Narrow Range

The above instruments were erratic due to interruptions to I&C Bus 2B.

g. LR-3-53 Reactor Water Level

h. PR-64-50 Torus and Drywell Pressures

i. LI-64-54A Torus Water Level

The above instruments failed due to the loss of the Unit 2 preferred bus.

In addition to the instruments listed above, the vent monitor for Unit 2 was inoperable from about 3:30 p.m. until 7:00 p.m. Hourly "grab" samples of the exhaust gases were taken just as they were in Unit 1.

E. Restoration of Systems

Several attempts were made to restore systems during the early hours of the incident. (Some of these are documented in Details II of this report). Information received by the investigators from the plant superintendent indicates that a detailed record of breaker operations and fuse replacements was not maintained. In addition, individual operators attempted to operate components and to restore systems. In that there was no central coordination, on occasion multiple attempts were made to perform the same tasks. On at least one occasion, an operator opened a valve manually only to find that others had restored power to the valve.

Immediately after the fire, a program was started to make temporary repairs to restore power to selected systems. Individual operators could and did request temporary changes to systems and these changes were made. (Since no priorities were established and there was no central coordination, there was no assurance that systems were restored in accordance with their significance to safety of the facility). This program is outlined in the attached Temporary Cable Log (Exhibit C4). Since the restoration work was not well coordinated, exact times were not maintained of this work, and the listing of the cables does not represent the sequence in which they were installed. The time of the restoration of many of the individual components has not been determined.

F. Adherence to Design Criteria

1. Separation of Power Sources

The BFNP FSAR contains the following commitments in Sections 6.4.3 and 6.4.4:

"The core spray pumps for each unit receives power from the plant 4160 V shutdown boards. Each core spray pump motor and the associated automatic motor valves for one unit receive a-c power from different buses. Similarly, control power for each loop of the Core Spray System for one unit comes from different d-c buses. This arrangement satisfies design basis 5."

"LPCIS operation includes using associated valves, controls, instrumentation, and pump accessories. The LPCIS pumps for each unit receive power from the plant 4160 volt shutdown boards. Each LPCIS pump motor and the associated automatic motor valves for each unit receive a-c power from different buses. This arrangement satisfies safety design basis 5."

The investigators compared the actual installation with these commitments. The comparison of the normal power supply for the pump motors and the associated valves is given below:

System	Motor Power Supply	Normal Valve Power Supply
C.S. 1A	4 KV Shutdown (S.D.) Board A	4 KV S.D. Board A thru MOV Board 1A
RHR 1A	4 KV S.D. Board A	4 KV S.D. Board A thru MOV Board 1A
*C.S. 2A	4 KV S.D. Board A	4 KV S.D. Board B thru MOV Board 2A
*RHR 2A	4 KV S.D. Board A	4 KV S.D. Board B thru MOV Board 2A
*C.S. 1C	4 KV S.D. Board B	4 KV S.D. Board C thru MOV Board 1B
*RHR 1C	4 KV S.D. Board B	4 KV S.D. Board C thru MOV Board 1B
C.S. 2C	4 KV S.D. Board B	4 KV S.D. Board B thru MOV Board 2A
RHR 2C	4 KV S.D. Board B	4 KV S.D. Board B thru MOV Board 2A
C.S. 1B	4 KV S.D. Board C	4 KV S.D. Board C thru MOV Board 1B
RHR 1B	4 KV S.D. Board C	4 KV S.D. Board C thru MOV Board 1B
*C.S. 2B	4 KV S.D. Board C	4 KV S.D. Board D thru MOV Board 2B
*RHR 2B	4 KV S.D. Board C	4 KV S.D. Board D thru MOV Board 2B
*C.S.1D	4 KV S.D. Board D	4 KV S.D. Board C thru MOV Board 1B
*RHR 1D	4 KV S.D. Board D	4 KV S.D. Board C thru MOV Board 1B
C.S. 2D	4 KV S.D. Board D	4 KV S.D. Board D thru MOV Board 2B
RHR 2D	4 KV S.D. Board D	4 KV S.D. Board D thru MOV Board 2B

In eight cases, the CS and RHR pump motors were not supplied from the same boards which powered their associated valves. (These are indicated by an asterisk.) In six of these cases, the design was in compliance with the separation criteria in that the boards are in the same safety division. In two cases, the design does not comply with the separation criteria. Specifically RHR Pump 1C and Core Spray Pump 1C are supplied from 4 KV Shutdown Board B and their associated valves are supplied from 4 KV Shutdown Board C. Shutdown Board B is in Division I and Board C is in Division II.

2. Routing of Cable

The investigators determined that the power cable supplying 480 Volt Shutdown Board 1B from 4 KV Shutdown Board C and the power cable supplying 480 Volt Shutdown Board 2A from 4 KV Shutdown Board B were both routed in Cable Tray AX. In that 4 KV Shutdown Board B is in Division I and 4 KV Shutdown Board C is in Division II, this routing constitutes a violation of the separation criteria.

DETAILS IV

Response of TVA Groups and Various Governmental Bodies

A. Administrative Management of Incident Response
Notification and Response of Individuals and Agencies

1. Notification and Response by the Browns Ferry Nuclear Plant Individuals

Assistant Shift Engineer L was given permission by Shift Engineer R to contact the Athens Fire Department at 1:09 p.m. The Plant Superintendent was notified at about the same time and arrived at the site at about 1:30 p.m. He instructed others to start calling supervisors to the plant. At this time the plant emergency plan had not been fully implemented as there was no release of radioactivity but some portions of the plan, such as the recall of personnel, were used. The Plant Superintendent notified the Director, Division of Power Production, about the situation at approximately 2 p.m.

The Athens Fire Department personnel had arrived on the plant site at about 1:30 p.m. They were issued film badges and dosimeters and admitted to the plant. They were ready to assist by about 1:45 p.m. The Athens Fire Chief examined the fire area and about 2 p.m. he initially recommended the use of water. The Plant Superintendent stated that upon the advice of Public Safety Officer CC, and as instructed by the TVA Fire Safety Manual, he decided not to use water, because, in his judgment this would be too dangerous. Furthermore, he stated that his main attention was focused on the core cooling problems and not on the fire.

At about 2:40 p.m., the Chief Nuclear Generation Branch (NGB) notified the Plant Superintendent that he was at the TVA CECC and that the center would be activated. The Plant Superintendent contacted the CECC at about 5:30 p.m. and was advised by the Chief, NGB, that water could be used; however, he was cautioned not to use an excessive amount. At approximately 6 p.m. the Athens Fire Chief again recommended the use of water and offered the services of his own men and equipment. He stated at this time in his opinion, that if water were not used, the fire would continue to burn until it reached the end of the cables. The Plant Superintendent stated that the permission to use water had been given reluctantly and he did not direct that water be used until approximately 6:40 p.m. Some time was required in preparation, but about 7 p.m. to 7:20 p.m. plant personnel, using protective gloves, started using water and the fire in the reactor room was declared extinguished at about 7:45 p.m. The Athens Fire Department cleaned their equipment and departed the plant at about 9:50 p.m.

2. Notification and Response by TVA Central Emergency Control Center (CECC)

a. The TVA Load Dispatcher at Chattanooga was notified of the Browns Ferry Nuclear Plant fire at 12:54 p.m. March 22, 1975. The Load Dispatcher notified the Assistant Chief, Nuclear Generation Branch after receiving information from BFNP that both units had been scrammed. The Assistant Chief, NGB, tried to contact the Chief, NGB, at 1:23 p.m. but was not able to locate him at home. The Chief, NGB, returned the Assistant Chief's call at 2:45 p.m. and stated that the radiation emergency plan (REP) should be activated. The Assistant Chief, NGB, notified several key people, among them individual AC, who arrived at the CECC at 3:35 p.m. Individual AC, according to the REP, is the first alternate director of the CECC in the event that the person normally assigned this position is not available. Individual AC stated that he was not advised that the person designated as the director had not been located and that he, Individual AC, was the director. He advised the investigators that at about 4:05 p.m. be realized that he was in charge of the CECC.

The Chief, NGB, had contacted the BFNP Plant Superintendent at about 2:40 p.m. and advised the Plant Superintendent that the CECC would be activated. The Chief, NGB, arrived at the CECC at 3:40 p.m. and headed the Division of Power Production (DPP) effort.

b. The TVA-BFNP Emergency Plan provides that the CECC be decentralized into the Division of Power Production (DPP) Emergency Section in Chattanooga, Tennessee; the Environs Emergency Section at Muscle Shoals, Alabama; and the Site Emergency Section at BFNP. During an emergency each of these sections are responsible for providing the Director of the CECC in Chattanooga, with up-to-date and accurate information. The investigators determined that during the fire incident, the director was not always aware of the exact status of the reactors because he was not periodically briefed by the DPP. Additionally, it appears that the DPP emergency section was not always abreast of the latest plant status. Consequently, incorrect interpretations and information were forwarded by the Director to principal support groups. An example of this incorrect information was apparent in the director's notification to the State of Tennessee at 8:15 p.m. He informed the Tennessee officials that "Unit 1 and Unit 2 had been wiped out" and the plant had only one alternate core cooling system available and that was the river water. After receiving this information, the Tennessee officials became quite alarmed and notified primary support agencies and staff personnel. Also, they subsequently questioned the NRC investigators about the possibility of a "meltdown" at BFNP. This information to the State of Tennessee was especially confusing because it was presented thirty-five minutes after the fire had been extinguished, thus, indicating that the CECC Director did not have access to the most current information concerning the fire.

c. Although the DPP section of the CECC was apparently kept advised of the status of the nuclear steam supply systems, a review of the transcripts of the CECC conversations indicates that information relating to the fire and fire fighting activities was inaccurate and incomplete. As late as approximately 5 p.m. DPP was advising others that the fire was confined to the cable spreading room and was apparently unaware that the major fire damage was in the reactor building and that the fire in the cable spreading room had been reported extinguished at 4:20 p.m. Members of the DPP section were also advising other groups during this same time period that preparations were being made to put foam on the fire and apparently were not aware that the use of foam had been rejected since it would readily conduct electricity.

d. The Chief, NGB, talked to the Plant Superintendent at 3:40 p.m. and issued instructions that all communications with the plant be taped. Later, a review of the tapes revealed mechanical problems with the tape recorders and only partial transcripts were obtained. (See Exhibits D1 and D2). At about 5:00 p.m. the Plant Superintendent contacted the Chief, NGB, and requested permission to use water on the fire. At 5:20 p.m., the Chief, NGB, approved the use of water under "controlled conditions." According to the Plant Superintendent, this approval was given reluctantly. At 6:09 p.m. the BFNP Superintendent notified the Chief, NGB, that the fire kept re-igniting from the heat and he had been advised by the Athens Fire Chief to use water to cool the wiring but that he was still reluctant and wanted to hold off for another 30 minutes. According to information given the investigators, the decision to delay using water was made primarily because the Plant Superintendent wanted to keep the cables functional as long as possible and, although the Chief, NGB, had the authority to order the use of water, he did not do so because he felt that the plant supervisory staff should have the final judgment. The investigators were told by the

Plant Superintendent that the main concern, throughout, was the safe shutdown of both reactors and the providing of core cooling. During this time (about 6:10 p.m.) CECC also received the first indication that a candle, used to check leaks in the reactor building cable penetrations, had caused the fire. At 7:47 p.m. the Chief, NGB, received information that the fire was out. The Director, of CECC, Individual AC, apparently was not made aware of that fact until sometime after 8:45 p.m. At 10:15 p.m. the Chief, NGB, again contacted the U. S. NRC, Region II, and furnished a status report. The Director, CECC, at about this time secured the CECC until 7:30 a.m., March 23, 1975, but two DPP personnel were instructed to remain on duty during the night. The BFNP personnel reported the plant status at regular intervals throughout the night. The CECC was reactivated at 8:10 a.m., March 23, and the log, kept by the Chief, NGB, is attached as Exhibit D3. (As evidenced by the Summary of CECC Activities (Exhibit D4) there was no centralized direction of the overall effort. The role of the CECC was, in fact, minimal during the whole period).

3. Notification and Response by the TVA-Environs Emergency Center (EEC)

At 3:05 p.m. on March 22, 1975, Individual AD was notified by the Director, CECC, that a fire had been reported at the Browns Ferry Nuclear Plant at 12:35 p.m. Following the receipt of this notification, Individual AD with himself as Director, established the EEC, in Muscle Shoals, Alabama, at 3:15 p.m. Immediately upon establishing the center, he began efforts to determine if any environmental release had occurred. All of the information that was available to the EEC at this time indicated that no release had occurred but that activity in the reactor building had increased to 1.03 X 10^{-9} microcuries/cc. Since the EEC had worked with the State of Alabama on all emergency planning, the Director, EEC, notified the State at 3:22 p.m. and informed the Director of Radiological Health (DRH) on all of the pertinent information concerning the fire. The Director, EEC, recontacted the CECC at 3:25 p.m. and received the following status report:

(1) Both units had been tripped

(2) Circuitry was lost to the Unit 2 relief valves

(3) The fire was limited to the Unit 1 cable spreading room.

Under the direction of the EEC, the Site Emergency Center (SEC) was activated at the meterological tower at 4:15 p.m. Constant communications were established between the SEC and the EEC. The EEC received information from the SEC at 4:15 p.m. that all area, air particulate, and effluent monitors at the plant were inoperable. Due to the losg of this monitoring capability, the Director, EEC, questioned the status of the reactor vent system and requested that air samples be collected as close to the exclusion area as possible. At this time, the instrumentation in the meteorological tower indicated that the wind speed was five mph from the northwest at 300 degrees.

The first air sample results that were taken from within the plant were completed at 4:40 p.m. and the activity levels in the reactor plant were verified as follows:

(1) Unit 1 (Control Room) - 1 X 10^{-9} microcuries/cc

(2) Unit 2 (565 Level - 3 X 10^{-7} microcuries/cc

(3) Unit 3 (565 Level) - 1 X 10^{-9} microcuries/cc

(4) Unit 3 (refueling floor) - 9 X 10^{-10} microcuries/cc

At 5:05 p.m., the Director, EEC, directed that additional environmental air samples be obtained from stations located at the gatehouse and the southeast fence. At this time individuals in the SEC observed smoke emanating from the reactor building and the decision was made to evacuate the meterological tower.

Environmental air sample results received at 5:15 p.m. indicated the following concentrations:

(1) Gatehouse - 5 X 10^{-10} microcuries/cc

(2) Southeast fence - 9 X 10^{-10} microcuries/cc

(3) Met Tower - 8 X 10^{-10} microcuries/cc

After receiving the environmental air sample data, the decision was made at 5:34 p.m. to man the meterological tower again. Upon reoccupying the tower, the meterological instrumentation indicated that the wind speed was approximately three mph and still from the northwest direction between 240 degrees and 300 degrees.

At 5:25 p.m. the EEC received information that carbon monoxide (CO) levels were increasing inside the reactor building and assistance was requested. The Director, EEC, notified four industrial hygienists, who were dispatched to the plant. At 5:45 p.m. the EEC was informed that the reactor building ventilation system was inoperable.

More environmental air sample results were received at 6 p.m. which indicated relatively little change in radiation levels surrounding the plant. The samples indicated the following results:

(1) Site boundary - 9 X 10^{-10} microcuries/cc

(2) Gatehouse - 5.62 X 10^{-10} microcuries/cc

(3) Met tower - 8 X 10^{-10} microcuries/cc

Based on these results, the Director, EEC, requested addi..ional environmental air sampling further from the site boundary.

At 6:15 p.m. the EEC received an updated report on the radiation levels within the plant indicating the following:

(1) Unit 1 (reactor building) - 7.5 X 10^{-8} uCi/cc

(2) Unit 1 (control room) - 6.9 X 10^{-8} uCi/cc

(3) Unit 2 (reactor building) - 3.56 X 10^{-7} uCi/cc

(4) Unit 2 (refueling floor) - 2.9 X 10^{-9} uCi/cc

At 6:35 p.m. Health Physicist AG reported to the EEC that the radiation levels in the Unit 1 control room were increasing and that some of the individuals in the room did not have respirators. Health Physicist AG requested that the EEC record the conditions that existed at this time in the control room and that whole body counting be performed at a later date. He also related that all personnel had been checked for external contamination and that none had been found. The Director, EEC was recontacted at 6:50 p.m. and advised that the turbine building activity levels were increasing. He was informed that the results from the analyses of air samples that had been collected in the turbine building area were as follows:

(1) TB deck (617 level) - 2.5 X 10^{-7} uCi/cc

(2) TB deck (586 level) - 4.8 X 10^{-3} uCi/cc

(3) OG monitor (565 level) - 6.6 X 10^{-9} uCi/cc

(4) TB deck (565 level) - 2.7 X 10^{-8} uCi/cc

Air sample results from the health physics laboratory indicated an activity of 9.9 X 10^{-9} uCi/cc.

The EEC was informed at 7:35 p.m. that the reactor building ventilation system was reactivated but that the Unit 2 fan would not operate. At 8 p.m. the EEC made a site boundary dose rate calculation based on a 2000 uCi/sec release rate and the calculations indicated a dose of 1.16 millirem/hour in the northwest section.

At 8:37 p.m. a member of the environment staff made an attempt to telephone the gatehouse by using a public telephone to inform the security guards that the warning lights on the plant stack were not operating. Since the gatehouse could not be reached, the environmental representative telephoned the EEC and explained the condition. The Director, EEC, directed the information to the plant because of the need to contact FAA authorities immediately.

The CECC informed the EEC that the fire had been extinguished at 7:45 p.m. but that entry into the reactor building still required respiratory equipment. The CECC related that the radiation levels had dropped below mask requirements but that carbon monoxide levels were in excess of safe limits.

At 9:30 p.m. Health Physicist AG reported from the plant that the Unit 2 radiation monitors were operable and were indicating essentially no release from Unit 2. He also related that "grab" samples were being taken from Unit 1 but that no results were available. The results of the "grab" samples were made available at 1:45 a.m. on March 23, 1975. They indicated the following concentrations from the Unit 1 reactor building ventilation duct between 4:45 p.m. - 7:40 p.m. CDT on March 23, 1975:

(1) 4:45 p.m. - 3.2 X 10^{-4} uCi/cc

(2) 6:10 p.m. - 1.7 X 10^{-5} uCi/cc

(3) 7:20 p.m. - 5.8 X 10^{-5} uCi/cc

(4) 7:40 p.m. - 1 X 10^{-4} uCi/cc

The CECC requested that both the EEC and the site boundary station be manned throughout the night of March 22, 1975, because the CECC would do the same. At 10:30 p.m. the EEC received a final updating of the incident from the CECC that elaborated on the following areas:

(1) All cooling (RHR, HPCI, Core Spray) to the reactor was inoperable

(2) Feedwater was being provided through the control rod drive mechanism

(3) Relief Valves were being used to reduce pressure in the reactor

The EEC continued to review air sample results throughout the night, The meteorological tower was secured at 3 a.m. on March 23, 1975, and the EEC was closed at 5:15 a.m.

4. Notification and Responses by States and Local Support Agencies

(a) Notification and Response By The State of Alabama Department of Public Health

On March 22, 1975, at 3:20 p.m. the Director of Radiological Health (DRH) for the State of Alabama Department of Public Health was notified by the Director, EEC, Muscle Shoals, Alabama, that the Browns Ferry Nuclear Plant had a fire in the cable spreading room and both reactor units had been scrammed. DRH immediately notified the Director of Environmental Health Administration who requested that the State of Alabama Health Officer be informed. An attempt was made to notify the health officer at 3:40 p.m. but this attempt was not successful.

At 3:45, DRH informed the State of Alabama Civil Defense Department that a fire had occurred at the Browns Ferry Nuclear Plant but that there had been no radiation release. After notifying the Civil Defense, the Tri-County Health Officer was informed of the fire. (The tri-counties, consist of Lawrence, Limestone and Morgan Counties). At 3:58 p.m. the State of Alabama Environmental Health Laboratory was notified, and the laboratory director suggested that the situation be carefully reviewed prior to actually initiating environmental air sampling around the site.

DRH recontacted the Director, EEC, at 4 p.m. to obtain more information concerning the status of the reactor. At this time the Director, EEC, requested that the State obtain all future information from the CECC in Chattanooga, Tennessee. DRH contacted the Director, CECC, at 4:05 p.m. and received the following information:

(1) Both reactors were scrammed at approximately 1 p.m.

(2) Reactor radiation monitors were out

(3) Vent monitors were thought to be operating but this situation had not been confirmed.

(4) The fire was still burning.

(5) CO_2 fire extinguishers had been used.

(6) Core cooling was continuing but not on the primary system.

(7) The fire was below the control room in the cable spreading room.

(8) Athens (Alabama) Fire Department was on site.

(9) The U. S. NRC - Region II answering service had been notified.

The DRH advised the investigators that after receiving this information he concluded that the situation was more serious than the initial evaluation and it was imperative to start environmental air sampling at this time.

The Director of Environmental Health Administration was recontacted at 4:45 p.m. and he agreed with DRH that the Governor of the State of Alabama should be notified. Since the directors determined that no immediate action would be required by the Governor, an attempt was made to notify the Governor's staff. This attempt was not successful.

The Director, CECC, was recontacted at 5:15 p.m. and the following information from the CECC was received by DRH:

(1) Unit 2 was in normal shutdown process.

(2) Unit 1 had the fire problem.

(3) The fire was still burning but was under control.

(4) Powder chemicals were still being used to extinguish the fire. No water had been used to put out the fire.

(5) No injuries had been reported.

(6) No entrance had been made into the spreading room.

(7) Fire engine was in the building.

(8) Stack monitor indications were low (normal).

(9) Unit 1 was at a pressure of 200 psi.

(10) The U. S. NRC - Region II had been informed of the fire.

(11) The CECC was having trouble contacting the State of Alabama.

At this point, DRH questioned the ability to maintain core cooling, and for confirmation to the question the director was referred to the Assistant Chief, NGB, who gave the following information:

(1) Core cooling was being maintained by the condensate booster pumps.

(2) Control of the relief valves from the control room had been lost.

(3) The ability to operate the motor operated valves from the control room had been lost.

(4) The high pressure coolant injection (HPCI) system was inoperable.

(5) The residual heat removal (RHR) system was inoperable.

(6) Part of the emergency core cooling system (ECCS) lost.

(7) Core cooling using river water was available.

(8) The torus temperature was 126°F with no leakage indicated.

DRH stated that after receiving the above information he concluded that the core cooling system was degraded and must be watched. He also concluded that the situation was serious, the monitoring capability of leakage was questionable, and confirmation was needed to verify that scram closed main steam isolation valves (MSIV's)

The health laboratory director reported at 5:45 p.m. that environmental air sampling had been started at the Athens Water Treatment Plant, the Athens Sewage Treatment Plant, Hillsboro, and Rogersville, Alabama. The sampler at Decatur, Alabama, was thought to be inoperable possibly due to the wind directional control system but the laboratory director was asked to investigate the problem. The laboratory director reported to DRH at 7:50 p.m. that no air sampler was available at Decatur. This station would have been the major air station of importance because Decatur, Alabama, is located in the southeast direction from the site and the wind direction at the time of the fire was from the northwest section. Arrangements were made with the State of Alabama Air Pollution Control Commission for using one of their samplers at the Decatur station. Air sampling was initiated at this station at approximately 9 p.m., CDT, on March 22, 1975.

The Director, CECC, recontacted DRH at 6:50 p.m. and stated the following:

(1) Environmental radiation measurements around the plant were essentially background.

(2) Radiation measurements at the Browns Ferry Nuclear Plant gatehouse were essentially background.

(3) Radiation measurements at the Browns Ferry Nuclear Plant site boundary were essentially background.

(4) The wind direction was from the Northwest at 5 mph under the Pasqual Class A condition.

(5) The fire condition had not changed.

(6) Gross radioactivity in the Unit 2 reactor building was above the restricted area maximum permissible concentration (MPC) of 3.56 X 10-7 uCi/cc.

(7) Gross radioactivity in the Unit 1 reactor building was 7.5 X 10-8 uCi/cc.

During this conversation it was recommended by DRH that the State of Tennessee be notified.

At 9:15 p.m. the DRH recontacted the CECC and received the following information:

(1) The fire was out at approximately 7:45 p.m.

(2) Water fog had been used to extinguish the fire.

(3) Six environmental air samples had been collected and analyzed and the highest result was 8.5×10^{-10} uCi/cc. The exact location that the sample had been collected was not known by the CECC.

(4) Continuous air monitoring equipment that was operational at this time indicated a continuous drop in radioactivity.

(5) Vent monitoring was "out" but monitoring was being done manually and preliminary results indicated no abnormal radioactivity levels.

(6) Unit 1 relief valves were operating again and pressure was dropping. The Unit 1 relief valve lost operating air and vessel pressure increased to 500 psi.

(7) Weeks would be required for decay heat removal.

(8) Core cooling now utilized the control rod makeup water system.

(9) Six cable trays were involved in the fire.

At 9:45 p.m. DRH notified the State Health Officer of the current status of the fire. In this conversation the State Health Officer confirmed that he had talked to the Governor of Alabama. He advised DRH that the Governor wanted to know the following facts:

(1) Were additional State resources required, especially the National Guard?

(2) Was adequate electrical power available in North Alabama?

(3) Was sabotage possibly involved?

The State Health Officer assured the Governor that no additional resources were required, there was adequate electrical power and there was no indication as to the cause of the fire at this time.

DRH received the following information from the CECC at 10:45 a.m. CDT on March 23, 1975:

(1) Units 1 and 2 had established shutdown cooling.

(2) TVA vent monitoring results were provided with the only identifiable isotope as rubidium-88.

(3) The calculated fenceline dose was total of 1.8 millirem. The total estimated dose, assuming the highest discharge concentration and rate as gross radioactivity, was 17.2 millirem.

(4) The major problem at one time during the fire was 500 ppm carbon monoxide.

(5) The probable cause of the fire was attributed to a worker using an open flame to test leakage and igniting polyurethane.

On March 24, 1975, DRH continued to review the situation with the U. S. NRC Region II, State of Alabama personnel, and the State of Tennessee. On this date the State made the decision to take "grab" water samples below the site at Wheeler Dam and Muscle Shoals, Alabama, and collect milk samples from designated milk sampling stations around the site. TLD environmental monitors from the TLD monitoring stations would be collected. The results of the BFNP TLD's would be compared to other TLD stations from throughout the State.

DRH stated that after considering all aspects of the fire the following future plans were considered:

(1) No news release would be issued concerning the activities of the State of Alabama relative to the BFNP fire.

(2) Environmental surveillance activities around all existing nuclear reactor sites would be carefully reviewed.

(3) Additional appropriations from the State legislature would be requested for surveillance activities around reactor sites.

(4) Members of the Governor's cabinet would be briefed on the issues concerning the fire.

(5) Emergency notification procedures from TVA to the State of Alabama would be reviewed.

(6) The State of Alabama Radiation Advisory Board and possibly the State of Alabama Board of Health would be briefed on the Browns Ferry situation.

(b) Notification and Response by the State of Alabama Civil Defense Department

On March 22, 1975, the DRH telephoned, a "duty" representative for the State of Alabama, Civil Defense Department. The CD representative was informed that a fire had been reported at Browns Ferry Nuclear Plant but radiation levels were not above permissible limits at this time. He was advised that notification procedures should be carried out. The CD representative proceeded to the Civil Defense office for communication facilities.

At 4:04 p.m. the CD representative telephoned the Morgan County Civil Defense office but no answer was received.

He then contacted the Morgan County Sheriff's Office at 4:05 p.m. but requested no action. The representative continued to attempt to notify the Civil Defense Directors and the county sheriffs of Limestone and Lawrence Counties. He was only successful in contacting the Civil Defense Director of Lawrence County and the Morgan County Civil Defense Coordinator, who at this time was at BFNP. The CD representative made attempts to notify the State of Alabama Civil Defense Director and his assistant but could not locate them. The representative discontinued all notifications at 4:40 p.m. CDT and returned to his home.

The State of Alabama Civil Defense Coordinator and the notification representative advised the investigators that all aspects of emergency notification between the State of Alabama Health Department and the State of Alabama Civil Defense Department should be reviewed. They also confirmed that the notification procedures within the State of Alabama Civil Defense structure would be evaluated.

(c) Notification and Responses by the Morgan County Civil Defense

The Morgan County Civil Defense Coordinator advised the investigators that he received official notification of the Browns Ferry fire from the State of Alabama Civil Defense Department in Montgomery, Alabama, at 4:05 p.m. on March 22, 1975. The coordinator was at BFNP when he received the official notification because he had learned of the fire approximately thirty minutes after it started from a local police radio system. No action was taken by the coordinator upon receipt of the unofficial notification to contact the State of Alabama Civil Defense Department or the State of Alabama Department of Public Health.

(d) Notificaton and Responses by the Morgan County Sheriff

The Morgan County Sheriff's Office was officially notified by the State of Alabama Civil Defense Department at 4:05 p.m. on March 22, 1975. No specific action was requested of the Sheriff's Office except that he not inform the public in order to avoid alarming the population. The Sheriff advised the investigators that he had begun his first term as Sheriff of Morgan County on January 20, 1975, and since assuming the responsibilities of Sheriff he had not been briefed in the State of Alabama Emergency Plan for BFNP and that he did not have a copy of the plan. He further stated that the principal support agencies in Morgan County should meet with the State of Alabama Department of Public Health and define the emergency responsiblities of each agency and update the plan.

(e) Notification and Responses by the Limestone County Civil Defense

The State of Alabama Civil Defense Duty officer could not locate the Limestone County Civil Defense Coordinator on the day of the fire. The coordinator indicated that he received the information of the fire on the morning of March 24, 1975. He related that he would personally investigate the notification procedures of the State of Alabama Civil Defense Department and determine who was responsible for his emergency notification. The coordinator indicated that his copy of the State of Alabama Emergency Plan for the BFNP was not updated and that he had not received any information concerning the plan in several years.

(f) Notification and Responses by the Limestone County Sheriff

The Limestone County Sheriff stated that he was never officially notified of the BFNP fire but that he did receive some information after it was extinguished. The State of Alabama Civil Defense Department has responsibility for notifying the Limestone County Sheriff and their notification records verify that an attempt was made to contact the Sheriff at 4:08 p.m. on March 22, 1975, but that no answer was received. The

Sheriff related that he did not have a copy of the State of Alabama Emergency Plan for BFNP and that he had received very little information concerning his emergency responsibilities in the past two years.

(g) <u>Notification and Responses by the Lawrence County Civil Defense</u>

The Lawrence County Civil Defense Coordinator was officially notified of the fire by the State of Alabama Civil Defense Department at 4:10 p.m. Pertinent information concerning the fire was forwarded to the coordinator but no specific action was requested.

(h) <u>Notification and Responses by the Lawrence County Sheriff</u>

State of Alabama Civil Defense Department notification records for March 22, 1975, verify that an attempt was made to notify the Lawrence County Sheriff at 4:08 p.m. but no answer was received. Only one attempt was made to locate the Sheriff.

(i) <u>Notification and Responses by Other Support Agencies</u>

(1) <u>Tri-County Health Department</u>

The Tri-County Health Officer advised the investigators that she was notified by DRH on March 22, 1975, at 3:55 p.m. DRH informed the officer of the status of the reactor and the conclusions that had been determined by the State. No action was required of the Tri-County Health Department.

(2) <u>State of Alabama Highway Patrol</u>

No official notification was made to the State of Alabama Highway Patrol by the State of Alabama Department of Public Health or by TVA. However, a representative of the highway patrol who was assigned to the Limestone County District on the day of the fire received information of the conditions at the plant from the local police radio system. The representative drove to the site and offered his assistance to the security guards. No action was requested of the highway patrol representative at anytime during the fire.

(3) Colonial Manor Hospital

The Colonial Manor Hospital located in Athens, Alabama, is designated by the BFNP Emergency Plan as the offsite medical treatment facility. The facility is designed to provide decontamination and minor treatment of individuals involved in a radiation accident at the plant. Since no one was contaminated or seriously injured, the medical facility was not notified during the fire.

(j) Notification and Responses by the State of Tennessee Department of Public Health

The Tennessee Assistant Director of Radiological Health advised the investigators that he received a call from the CECC at 8:15 p.m. on March 22, 1975, reporting a fire at BFNP. He stated that he was informed that Unit 1 and Unit 2 were scrammed and that most of the emergency core cooling system (ECCS) was inoperable due to the fire and the two units were being cooled by booster pumps. He was also advised that radiation monitors located around the site had detected no increase in radioactivity. Further conversation revealed that there was a fire in the cable tray room which had "wiped-out Units 1 and 2" and that the first and second alternates for the core cooling were gone and the third alternate was considered. He advised the investigators that the individual from CECC stated that the one alternate for the core cooling system left was to pump river water through the reactors and circulate it to and from some ditches for cooling. He further advised the investigators that he was told that smoke was everywhere. During the interview he asked the investigators about the possibility of "meltdown" occurring at BFNP. The following additional details were provided to the State of Tennessee:

(1) Unit 1 was operating at a power level of 1000 megawatts (electric) at the time of the scram.

(2) Both units have their initial fuel loading.

(3) The radioactivity and pressure in the containment structures was unknown due to radiation monitors not being functional.

(4) Stack releases were normal.

(5) Radioactivity levels were related as:

turbine room - 2.5×10^{-7} uCi/cc

control (Unit 1) room - 7.5×10^{-8} uCi/cc

control room (Unit 2) - 3.56×10^{-7} uCi/cc

The Tennessee Assistant Director of Radiological Health also advised the investigators that at 8:35 p.m. he contacted the Alabama Director of Radiological Health. The latter stated that, to his knowledge, the fire was not out as of the last communication with the CECC. He also informed the Tennessee Director of his conclusions and exchanged technical information.

The Tennessee Assistant Director of Radiological Health contacted his staff and the Tennessee Civil Defense Department. These individuals were alerted to standby in the event that the fire became worse or that radioactivity was released.

B. Implementation of Existing Emergency Plans and the Adequacy of Prior Emergency Drills

As a result of the Browns Ferry Nuclear Plant fire on March 22, 1975, portions of both the TVA-Browns Ferry Radiation Emergency Plan and the State of Alabama Emergency Plan for the Browns Ferry Nuclear Plant were implemented. Both emergency plans specifically clarify classifications, emergency actions by support agencies, and evacuation procedures for plant personnel and members of the general public. Additionally, both plans have been reviewed and approved by various federal and state agencies, and have been used as references for other radiation emergency plans.

On the day of the fire, the entire TVA-Browns Ferry Radiation Emergency Plan was implemented at approximately 2:40 p.m. and portions of the plan remained implemented until the following day (Sunday, March 23, 1975). The implementation of the plan activated the Central Emergency Control Center (CECC) in Chattanooga, Tennessee, a site boundary station at the plant, and an Environs Emergency Center (EEC) in Muscle Shoals Alabama. TVA individuals who participated in the implementation of the plan expressed satisfaction with the prompt notification procedures and the ability of everyone to quickly assemble at prescribed emergency stations.

The State of Alabama Emergency Plan for the Browns Ferry Nuclear Plant was implemented at 3:20 p.m. to the extent that notifications were made to designated State personnel and principal support agencies. No action was required by anyone contacted except for the initiation of environmental air sampling around the site by the State of Alabama Environmental Health Laboratory. Since no radiation emergency existed at the time that the State made notifications, only one attempt was made to contact principal support agencies that were located in counties surrounding the site regardless of whether the agency was contacted or not. The notification process was discontinued at 4:40 p.m. The investigators commented to the DRII that, due to the uncertainty relating to the status of the reactors from 12:30 p.m. - 7:45 p.m., the implementation of the State plan indicated that a "standby" classification was necessary that would have required continuous notifications and recommendations to be made to support agencies until the reactor was verified to be in a safe condition. Additionally, some agency officials related that they did not have a copy of the State plan or the plan that they had needed updating. Other officials indicated that they had received very little information concerning their defined responsibilities relating to an emergency at the plant. Almost all of the support agencies expressed a desire for additional training, annual briefings, and a continuous updating of the State plan.

The State of Alabama and BFNP personnel have participated in emergency drills to test the effectiveness of their emergency plans for the past several years. Participation in the drills by the State has involved the verification of notification procedures and the time required to travel to the site to perform environmental sampling. Browns Ferry personnel have participated in the drills to the extent that all notifications systems have been tested and an environmental sampling program has been employed.

50-259/75-1
50-260/75-1

BROWNS FERRY NUCLEAR PLANT

FIRE INVESTIGATION

EXHIBIT 1

STATEMENTS OF INDIVIDUALS

INTERVIEWED BY INVESTIGATORS

BROWNS FERRY FIRE INVESTIGATION

KEY TO JOB TITLES OF TVA AND OTHER AGENCY PERSONNEL

Key	Title
EES-A	Electrical Engineering Supervisor
EE-B	Electrical Engineer
EA-C	Engineering Aide
Elec.-D	Electrician
Elec.-E	Electrician
Elec.-F	Electrician
ASE-G	Assistant Shift Engineer
EA-H	Engineering Aide
Elec.-J	Electrician
PSO-K	Public Safety Officer
ASE-L	Assistant Shift Engineer
O-M	Operator
AP-N	Apprentice Pipefitter
P-O	Pipefitter
SF-P	Steamfitter-Welder
AUO-Q	Assistant Unit Operator
SE-R	Shift Engineer
CE-S	Chemical Engineer
HPS-T	Health Physics Supervisor
SE-U	Shift Engineer
SE-V	Shift Engineer
AOS-W	Assistant Operations Supervisor
AUO-X	Assistant Unit Operator

Key to Job Titles of TVA and Other Agency Personnel

Browns Ferry Fire Investigation -2-

Key	Title
ASE-Y	Assistant Shift Engineer
AUO-Z	Assistant Unit Operator
AUO-AA	Assistant Unit Operator
AUO-BB	Assistant Unit Operator
PSO-CC	Public Safety Officer
OS-DD	Operations Supervisor
ASE-EE	Assistant Shift Engineer
EE-FF	Electrical Engineer
AS-GG	Assistant Superintendent
AES-JJ	Assistant Electrical Superintendent
O-KK	Operator
O-LL	Operator
O-MM	Operator
O-NN	Operator
ASE-OO	Assistant Shift Engineer
SE-PP	Shift Engineer
ASE-QQ	Assistant Shift Engineer
AUO-RR	Assistant Unit Operator
AUO-SS	Assistant Unit Operator
AUO-TT	Assistant Unit Operator
EE-UU	Electrical Engineer
ASE-VV	Assistant Shift Engineer

Key to Job Titles of TVA and Other Agency Personnel

Browns Ferry Fire Investigation -3-

Key	Title
AGCS-WW	Assistant General Construction Superintendent
O-XX	Operator
ASE-YY	Assistant Shift Engineer
O-ZZ	Operator
O-AAA	Operator
AUO-BBB	Assistant Unit Operator
EE-CCC	Electrical Engineer
EF-DDD	Electrician Foreman
PRS-EEE	Plant Results Supervisor
ASE-FFF	Assistant Shift Engineer
MMF-GGG	Maintenance Machinist Foreman
Elec.-HHH	Electrician
IE-JJJ	Instrument Engineer
GEF-KKK	General Electrician Foreman
EA-LLL	Electrician Aide
Elec.-MMM	Electrician
Elec.-NNN	Electrician
EA-OOO	Electrical Engineering Aide
MS-PPP	Maintenance Supervisor
AUO-QQQ	Assistant Unit Operator
MMS-RRR	Maintenance Mechanical Engineer
AUO-SSS	Assistant Unit Operator

Key to Job Titles of TVA and Other Agency Personnel

Browns Ferry Fire Investigation -4-

Key	Title
O-TTT	Operator
EE-UUU	Electrical Engineer
Elec.-VVV	Electrician
AUO-WWW	Assistant Unit Operator
EE-XXX	Electrical Engineer
Elec.-YYY	Electrician
AUO-ZZZ	Assistant Unit Operator
AUO-AAAA	Assistant Unit Operator
AUO-BBBB	Assistant Unit Operator
AUO-CCCC	Assistant Unit Operator
QASS-DDDD	QA Staff Supervisor
Elec.-EEEE	Electrician
APMS-FFFF	Assistant Plant Maintenance Supervisot
ME-GGGG	Mechanical Engineer
AUO-HHHH	Assistant Unit Operator
IE-JJJJ	Instrument Engineer
O-KKKK	Operator
EA-LLLL	Engineering Aide
AUO-MMMM	Assistant Unit Operator
ASE-NNNN	Assistant Shift Engineer
ASE-OOOO	Assistant Shift Engineer
ASE-PPPP	Assistant Shift Engineer
EF-QQQQ	Electrician Foreman

Key to Job Titles of TVA and Other Agency Personnel

Browns Ferry Fire Investigation -5-

Key	Title
ASE-RRRR	Assistant Shift Engineer
SE-SSSS	Shift Engineer
EE-TTTT	Electricial Engineer
AUO-UUUU	Assistant Unit Operator
CNGB-AB	Chief, Nuclear Generation Branch
DCECC-AC	Director, Central Emergency Control Center
DEEC-AD	Director, Environs Emergency Center
ACNGB-AE	Assistant Chief, Nuclear Generation Branch
CECC-AF	Central Emergency Control Center Representative
HP-AG	Health Physicist
CDC-AH	Civil Defense Coordinator
AFC-AJ	Athens Fire Chief
SLC-AK	Sheriff of Limestone County
CDC-AL	Civil Defense Coordinator
SMC-AM	Sheriff of Morgan County
DRH-AN	Director, Radiological Health (Alabama)
ADDRH-AO	Assistant Director, Division of Radiological Health (Tennessee)
AAO-AP	Assistant Administration Officer
ASR-AQ	Answering Service Representative

Key to Job Titles of TVA and Other Agency Personnel

Browns Ferry Fire Investigation -6-

Key	Title
HP-AR	Health Physicist
IH-AS	Industrial Hygienist
PHY-AT	Physicist
DMS-AU	Director of Medical Services
IE-AV	Instrument Engineer
IE-AW	Instrument Engineer
IM-AX	Instrument Mechanic
IM-AY	Instrument Mechanic
IM-AZ	Instrument Mechanic
IM-BA	Instrument Mechanic
AUO-BC	Auxiliary Unit Operator
ACS-BD	Assistant Construction Engineer
ADDPP-BE	Assistant to the Director Division of Power Production
EF-BF	Electrician Foreman
FM-BG	Foreman
HDB-BH	Hydraulic Data Branch
HP-BJ	Health Physicist
NE-BK	Nuclear Engineer
PSO-BL	Public Safety Officer
DPPC-BM	DPP Coordinator
FM-BN	Foreman
EE-BO	Electrical Engineer
PHY-BP	Physicist
Elec.-BQ	Electrician

Key to Job Titles of TVA and Other Agency Personnel

Browns Ferry Fire Investigation -7-

Key	Title
PFM-BR	Pipefitter Foreman
DEH-BS	Director, Environmental Health (Alabama)
CDDD-BT	Civil Defense Duty Officer (Alabama)
HD-BU	Health Officer
DEH-BV	Director, Environmental Health Laboratory
DPP-BW	Director, Division Power Production
DRH-BX	Director, Division of Radiological Health (Tennessee)
DCD-BY	Director, Civil Defense (Tennessee)
PHY-BZ	Physicist
PHY-CA	Physicist
PHY-CB	Physicist
PHY-CD	Physicist
SHO-CE	State Health Officer
PHY-CF	Physicist
Elec.-CG	Electrician
CDMC-CH	Chief Deputy of Morgan County
FM-CJ	Foreman
DPPP-CK	DPP Painter
HP-CL	Health Physicist
Elec.-CM	Electrician
Elec.-CN	Electrician
EA-CO	Engineering Aide

Key to Job Titles of TVA and Other Agency Personnel

Browns Ferry Fire Investigation -8-

Key	Title
PSO-CP	Public Safety Officer
PSO-CQ	Public Safety Officer
PSO-CR	Public Safety Officer
ACS-CS	Assistant Construction Superintendent
PSO-CT	Public Safety Officer
PSO-CU	Public Safety Officer
FM-CV	Foreman
TK-CW	Timekeeper
GCS-CX	General Construction Superintendent
CE-CY	Construction Engineer
GEIE-CZ	GE Instrument Engineer
GE-DA	GE Representative
ASR-DB	Answering Service Representative
PSO-DC	Public Safety Officer
CPM	Construction Project Manager
P-Supt.	Plant Superintendent

I, EES-A, voluntarily give the following information to Howard A. Wilber, Tolbert Young, D. Caphton and J. Devlin, who have identified themselves to me as representatives of the U.S. Nuclear Regulatory Commission.

I am employed by the Tennessee Valley Authority as the Electrical Engineering Unit, Supervisor. I have been employed by TVA since 1951 and at Browns Ferry since 1968.

I was working on Saturday when the fire occurred and I heard the announcement of the fire over the paging system. As the ranking engineering member of DEC on site I responded for DEC to DPP concerning engineering personnel. It was approximately 1:00 PM when I heard the announcement and alarm

The next hour was consumed supervising the mustering of all DEC engineering people working for DEC that were on site. There were approximately 50-55 DEC engineering personnel working. On Friday March 21, 1975, I was advised by EE-B that he had been checking a penetration in Unit 1 reactor building on elev. 565 when a small flame up occurred which he quickly extinguished by rubbing out with his hand. This occurred on March 20, 1975. This penetration contained poly-foam that had initially been coated with flamestic. This is not the incident in which a fire extinguisher was used to extinguish the flame. I did not know of that incident until after March 22, 1975.

I do not know of any tests of the poly-foam. Originally I understood the poly-foam was a temporary measure that was used by electricians to seal penetrations until they could apply sealant and flamestic. But I now know it is used as permanent fix.

The urethane is purchased from the same manufacturer, it is therefore assumed that it is always the same material. It is not batch tested.

The use of candles to check air leaks has been an acceptable method. It has been regularly used in Units 1 and 2. Other methods of checking air leaks were considered before any checking was done on Unit 1. I do not believe any other method was considered at the time the leak checks were started concerning Unit 3 in early March, 1975.

Later at the request of DEC management and DPP I started to line up DEC electrical engineering personnel to assist DPP in recovering or stabilizing temporary back up systems that could be used if required. Electrical engineers and electricians were assigned to work with DPP under authorized work plan to lift cables and install jumpers, where required.

At about 3:00 PM I directed EE-B to have EA-H and EA-C, Engineering Aids document chronologically the events of the fire. CPE-CY requested this, the statements were sent to him.

The penetrations that were being sealed were the result of cable pulls, work done under other work plans. These cables installations were not considered design changes, alterations or modifications and did not require design review. These were considered and included under original designs at outset as future needs not initially completed.

10% or more of all critical cables are traced by engineering personnel when installed. Several audits have been made by Site Quality Assurance Audit Group. The installations are subject to audits by the OEDC - Quality Assurance Group in Knoxville.

Browns Ferry procedures are coordinated and processed through the Engineering Planning and Coordinating Unit. The procedures must then be approved by the construction engineer and the project manager. The procedures must also be approved by the OEDC quality assurance manager in Knoxville.

Frost Pak is a spray on material that is also used. I am not sure who originally ordered the poly-foam. Froth-Pax spray on material was also approved by Design (DED).

/s/ C. E. Murphy	/s/ EES-A	April 17, 1975
Witnessed	Signature	Date

I, EE-B, make the following free and voluntary statement to John J. Ward, who has identified himself to me as a representative of the U. S. Nuclear Regulatory Commission.

I am an electrical engineer for TVA at the Browns Ferry plant. On March 22, 1975, I had four engineering aides working under me who were checking for air leaks around cable penetrations in the wall between the cable spreading room and the reactor building, at 606' elev. and other reactor elevations. As these leaks were found, two aides (EA-C and EA-H), working with electricians, were sealing up the leaks from the cable spreading room side. One other aide was on the third floor and another (EA-LLLL) was on the fourth floor of the reactor building.

Shortly after noon I had just checked EA-C's work and was on the reactor side to check with EA-LLLL when I heard all the fire alarms go off. There was no fire in my area. I went back around the reactor and met a security guard who went on around the reactor and up to SLC level, we than back down with me. We then saw two DPP men on the 621 level who could see fire through a floor opening. One of them discharged a CO2 extinguisher at this fire which appeared to be in a cable tray about five feet below this opening. An announcement was made for all personnel to evacuate from the reactor. I went back to tell EA-LLLLto get out. I went down to 565 back to the spreader room side at the #2 stairs. I saw ASE-G , an operator, outside the spreader room at the #2 door trying to set off the CO2 (Cardox) system. I asked if all persons were out of the spreader room, opened door, called into the darkened room, and receiving no answer and since the #1 door was closed assumed that my men were out. The operator then successfully activated the Cardox system by using the emergency switch. I than went to the Unit 1 stairs at the spreader room, saw EA-C and told him get out. I then returned to the reactor building to look for EA-H. I found him on the stairway in the reactor building. He was leaving the reactor building. He had gone to the reactor side, attempting to put out the fire, after he had first brought a fire extinguisher over to EA-C to use. We both then went to the assembly area.

I understand that the fire started in some polyurethane foam rubber packing that EA-C had inserted in the space around one of the cable penetrations. The flame from the candle EA-C had used to check for the air flow through the leak had apparently ingited the foam rubber packing. I had had a similar thing happen on March 20 when I was checking a leak with a candle.

The urethane foam had caught fire. I had put the fire out with my hands and reported the matter to EE-FF. Also on March 20, on a later shift, I understand another such fire occurred when no engineers were present. Two electricians who had been using the candle had put out the fire and the incident had been reported in the shift log.

I have read the statement summarized above, which is true and correct.

/s/ John J. Ward	/s/ EE-B	4-2-75
Witness	Signature	Date

I, EA-C, make the following free and voluntary statement to John J. Ward, who has identified himself to me as a representative of the U. S. Nuclear Regulatory Commission:

I am an engineering aide, SE-2, for TVA at the Browns Ferry plant. On March 22, 1975, EA-H and I, with electricians Elec.-D, Elec.-J, and Elec.-E, were checking seals for air leaks in the cable spreader room. after lunch, Elec.-D and I had started checking the wall on the Unit #1 side. We found a 2 x 4" opening in a penetration window in a VE tray with three or four cables going through it. The candle flame was pulled out horizontal showing a strong draft. Elec.-D tore off two pieces of foam sheet for packing into the hole, but he could not reach the opening. I inserted them as far as I could into the hole. I rechecked the hole with the candle. The draft sucked the flame into the hole and ignited the foam which started to smolder and glow. Elec.-D handed me his flashlight with which I tried to knock out the fire. This did not work and then I tried to smother the fire with rags stuffed in the hole. This also did not work and we removed the rags. Someone passed me a CO_2 extinghisher with a horn which blew right through the hole without putting out the fire, which had gotten back into the wall. I then used a dry chemical extinguisher, and then another, neither of which put out the fire. At about the time I started to use the second dry chemical extinguisher, I believe I heard another extinguisher on the other side of the wall. I was about to use another CO_2 extinguisher, but before I released it, the alarms went off for the CO_2 release. As I went to the door at the Unit 1 end the operator was shutting the door. Someone hollered at the other end. I ran outside into the hall. Elec.-D had come out with me.

In the past on three or four occasions I have had fires started by the candle, but this has been only a little blaze of the RTV or silicone coating, which is readily extinguished. EE-P was the one who instructed me in the use of a candle to detect air leaks. The foam rubber used for packing is obtained by the electricians from the warehouse.

I have read the statement summarized above, which is true and correct.

/s/ John J. Ward	/s/ EA-C	4-2-75
WITNESS	SIGNATURE	DATE

I, Elec.-D, voluntarily give the following information to Howard A. Wilber, Tolbert Young, and James Devlin, who have identified themselves to me as representatives of the U. S. Nuclear Regulatory Commission:

I am a Journeyman Electrician, employed by the Tennessee Valley Authority, Division of Engineering Construction. I have been working at Browns Ferry Nuclear Plant for past five years. I have worked on Units 1, 2, and 3.

On Saturday, March 22, 1975, I was working in Units 1 and 2 under a Work Plan, number unknown, that covers the sealing of air leaks in penetrations around cable trays and conduits in the walls and floors around the reactor containment Units 1 and 2. The job started on the previous Thursday morning.

I was working with EA-C, Engineering Aid, in the cable spreader room. After lunch we were compleging a second group of 10 trays, two wide and five deep. We were in among the trays, a difficult place to work and a difficult place to get to because of the configuration of the trays amd lack of room. To get to the cable tray window, that we were working on, we had to stand on an air duct and squeeze in between the trays. EA-C was on the right side and I was on the left. In the middle tray, left side, was a hole maybe 2" x 4".

Because the wall is about 30 inches thick and the opening deep, I could not reach in far enough, so EA-C asked me for the styrafoam (I call the spongy material, styrafoam) and he stuffed it into the hole. The styrafoam is in sheet form, it is a plastic about 2" thick, that we use as a backing material before th RTV is put into the penetration.

Before putting in the RTV, EA-C checked for an air leak using the lighted candle. He reached into the cable tray window and the styrafoam was ignited. He said he had a fire and tried to reach in to knock it out. He couldn't reach it, so I gave him my flashlight to use to knock down the fire but this also didn't work.

I believe the blown foam material, and styrafoam was producing its own ox.

I yelled that we have a fire and to notify the Shift Engineer that we got a fire. We then used two dry chemical extinguishers but they didn't do any good. We tried to use another CO_2 extinguisher but then the 20-second siren went off. This siren sounds before the CO_2 is released that floods the room. I knew that we had to get out of the room in a hurry, before the CO_2 is released.

As we were coming out of the room, we met an employee of DPP who asked me if everybody was out fo the room. I told him I was the last one.

I met FM-CJ, Foreman, who ordered us out of the plant. We left by the elevator in the control bay and the lights were still on there.

I thought that everybody knew that the material we were using to seal our leaks in penetrations would burn. We have been using it and using a candle as a detector since Unit 1. It works good, but I never did like it. I do not remember using any other method. I personally had not used the candle before on this day. The inspectors used the candle on this day. I had used the candles previously.

I have seen the RTV ignite several times but it was easy to rub out. I heard that a fire occurred on Thursday in a penetration and the same type of material burned. I don't know what group was involved but they were doing the same type work, under the same Work Plan.

I have often thought about the flammable properties of this material. I felt it could be trouble not during installation, but at a future time. I questioned whether a fire would start if the insulation on the cable deteriorated and the cable sparked and heated up, would the plastic material burn from within?

I knew the bell is used as the fire alarm, it is used everywhere by TVA. I have not been in any fire drills but I have been involved in two emergency evacuation drills.

/s/ C. E. Murphy	/s/ Elec.-D	16 April 75
WITNESS	SIGNATURE	DATE

Browns Ferry Nuclear Plant March 22, 1975

I, Elec.-E, an electrician employed by TVA will give the following account of how I saw the events before and during the fire which occurred on this date.

At 7:AM we reported to work at Elec.-CM's meeting place in unit #3. We were assigned to the Electrical Engineer EE-B for checkout work in unit 1. We checked in by the guard desk to unit 1. EE-B sent to the tool room for about 6 caulking guns and a carton or two of RTV caulking compound and a box of candles. These were to be used for testing and caulking air leaks in the reactor building. This was the method of checking air leaks we were instructed to use. This method had been used for testing air leaks for at least a year at this plant. EE-B called the group together and warned us how hazardous this method was. Why just the other day, EE-B said, (in effect) I caught some of that foam on fire and put it out with my bare hands, burning them in the process.

EE-B, the electrical engineer, said, wait here while I take these two men and show them where to check. Later he assigned Elec.-F and Elec.-D to the spreader room to assist EA-CO an engineer aid and Elec.-D an electrician in checking the air leaks and stopping the holes in the spreader room. Elec.-F and I went to start work where Elec.-CN and his partner were working, as they were being assigned to another area.

We checked along this wall and ceiling for some distance finding visible leaks in the ceiling of this corridor on the south wall of the spreader room. We stopped these small visible leaks with RTV caulking. We came upon one hole in the ceiling about 12"x12". This I stopped with sponge as a temporary measure to improve the air seal. The pressure was much greater in the control room above, for the air was blowing down like a fan.

I noticed two workers coming east along the wall with a lit candle checking leaks in the wall. A little later I heard one of them call for someone to get a fire extinguisher as they had a fire. I, Elec.-E, who was about 20 ft. east of those workers crawled under the trays and handed them some rags to smother out the tiny flame while I crawled under the trays and out of the spreader room where I found a CO_2 extinguisher just outside. I rushed back sliding the extinguisher ahead of me and crawled to the spot and handed them the nozzle and told them to remove the rags. The rags were smoldering but not aflame. I gave the extinguisher a couple of puffs and the fire was out on this side. He said I think it has gone through. I gave the extinguisher a few more puffs in that direction. On close examination I could see a gleam of fire through a hole about the size of a pencil eraser. I gave the pair the extinguisher trigger at their request as Elec.-F and I headed for the interlock. Elec.-F and I decided he would go ahead as I alerted the guard about the fire and told him where it was. I grabbed a CO_2 near the desk and headed for Elev 593. I first ran up the mezzanine steps in search of the fire on the north wall of the reactor. Someone hollered its back the other way as I turned I looked back I could see a tray on fire. I ran back down the mezzanine stairs and noticed a ladder against the last wall of the reactor. I ran up the ladder and cut the tie wires and moved it to the tray that was on fire. Just as I placed the ladder in place AP-N, a pipe fitter apprentice, ran up with a chemical fire extinguisher and opened it on the fire. (which at this time appeared to be 2 or 3 feet along the tray) The fire blaze was immediately snuffed out it appeared. The resulting fog enveloped the tray and the upper part of the ladder I was holding. The fire extinguisher was dropped and AP-N made it down the ladder and ran a few feet out of the fog and fell to the floor gasping for breath. Two pipefitters helped him further out of the area.

Seeing the chemical and smoke was too heavy to get near the tray I went back to the phone near the elevator and called the fire No. 299 and asked if the fire (location) had been reported and requested air line respirators.

AES-JJ came up and asked where the fire was. I directed him to the ladder. The smoke was pretty heavy and AES-JJ said lets get out of this smoke.

One operator arrived with an air tank respirator. He was helped into the respirator and it was adjusted for breathing. Then he went back and directed a CO_2 extinguisher toward the trays. I not having an air respirator had to move back from the chemical fog and smoke. I went toward the stairs and directed other operators to the fire.

We looked for more fire extinguishers on 2nd floor someone must have already carried them to the 3rd floor. I took the stairs down to 565 el. I met 2 wagon loads of air tank respirators and several operators with fire extinguishers. They did not wait for the elevator but ran up the stairs with them. One operator came up a few seconds behind them with a fire extinguisher. I said they have plenty help on 2nd lets go to third on elevator and see if the fire is under control there. We found several extinguishers laying around and lots of smoke but no fire had broken through to third. The operator asked if I had a flash light. I said no. Neither have I he said. He said, we better get out of here before the lights go off. Seeing everything on 3rd was o.k. I agreed and we left immediately.

Seeing there were many trained operators and air-tank respirators on 2nd. I figured everything was under control and went to the guard desk.

There all construction personnel was ordered out of the building which is routine in case of an alarm. I started out and happened to think of the spreader room where the fire had begun. I turned to go back about 10 ft. They were waving me back out but I said I must tell the guard to have some operator check on the spreader room in case the fire back feed into that area.

We went outside and grouped in craft to account for all construction personnel. Some were missing, so was Elec.-J. I thought the last time I remember seeing him was in the spreader room. When the alarm is sounded you have 20 seconds to clear the spreader room. They sent someone (Maybe AES-JJ) to look for Elec.-J. After a long wait they found him and he came to the assembly area just outside Unit 3 on the east side of the building.

After a while one of the men got up on the stairs and announced it was 2:15 PM and that we would knock off at 2:30 PM this Sat. and Sun. They said they would announce on radio about Mon.

We were not permitted to go back in the building to get our lunch kits. The foremen were told not to go back in for their time cards, that they could make them out Mon.

/s/ C. E. Murphy	/s/ Elec.-E	April 17, 1975
Witnessed	Signed	Date

I, Elec-F, make the following free and voluntary statement to C. E. Murphy, who has identified himself to me as a representative of the U. S. Nuclear Regulatory Commission:

I am an electrician. On March 22, I was working along the south wall of the cable spreader room, using RTV, when I heard the other crew say there was a burning back in the place where they were working. Elec.-E handed them some rags and Elec.-F passed up a broom handle so that they could wedge them in tight. After that try, the rags were jerked out and a rag was on fire. Elec.-E got a fire extinguisher. After it was used the fire was still going. Elec.-E said he would turn in the alarm. I then went around to the reactor side and met a pipe fitter who went with me (AP-N). A ladder was moved over to where the fire was. AP-N went up the ladder with a fire extinguisher with me right behind him on the ladder. AP-N had trouble getting the extinguisher started as he did not realize he had to first pull the pin.

I have read the statement summarized above, which is true and correct.

/s/ C. E. Murphy	/s/ Elec-F	5/6/75
WITNESS	SIGNATURE	DATE

2:55 pm
4/7/75

I, ASE-G make the following free and voluntary statement to Michael V. Annast, who has identified himself to me as a representative of the U. S. Nuclear Regulatory Commission:

I am an Assistant Shift Engineer at the Browns Ferry Plant. I was working inspecting the Turbine Building of Unit #1 when I heard the fire alarm at around 12:30 pm. I was by condensate pumps and did not hear the paging system. I called the control room and was told the fire was in the Reactor Building. I went through the "A" Shutdown Board Room Entrance on third floor of Unit #1. There was no fire or smoke so I went to the second floor. A couple of electricians were running and I followed them. The fire was in the cable trays. ASE-L and O-LL were already there.
ASE-L went up the ladder and set off an extinguisher. I went to "B" Shutdown Board Room door and got a CO_2 and a dry chemical extinguisher. I went back and gave one to ASE-L who set it off. I called the control room for breathing apparatus. ASE-L came down the ladder covered with CO_2 and also requested breathing apparatus. The operator requested help so ASE-L and I decided that I would go up to help out. When I got to the control room, I saw that RCIC, HPCI, LPCI and core spray was automatically initiated because of cable shorting. ADS annuciators were in and ADS timers were running. The recirc pump started ramping back. O-M the operator, and I tried to shutdown RHR and core spray pumps. The ECCS was out. We talked to SE-R and we all agreed to scram the reactor. I assisted the operator in shutting down the turbine and trip exciter field breaker. I disconnected the motor operatored disconnects and opened the turbine drains. I shut down the HPCI. We were in pretty good shape when we lost the 250 volt DC boards and the RX MOV boards and 480 shutdown boards. Also lost the unit preferred and both reactor protection systems we had isolated. The only thing left was the RCIC lights but one valve FCV 71-2 was out. I went to B RX MOV board and both sources were dead so went back through control room. ASE-L called and stated that the fire was in the cable spreading room. He suggested setting off the cardox. I set off the alarm siren to clear the room and went down to make sure that everyone was out of the spreader room. I went inside and cleared two electricians out then I tried to set off the Cardox and discovered the power was off. I ran over to the Unit #2 side and put the power on. There was no noise of discharge, so I thought that power had failed. I tried to use the manual crank system and discovered that it had a metal construction plate on under the glass and I tried to remove it. This was difficult without a screwdriver. About this time I heard the CO_2 come down the pipes. I did not realize that it would take several minutes to arrive. The next day, I checked other manual Cardox initiators and found that almost all of them had these construction plates attached.

After initiating the Cardox, I came up by Unit Ø2 side. Just then Unit Ø2 also scrammed and I helped the operator, O-KK, to shutdown the turbine. I went to the 480 volt shutdown boards. I reset the feeder breakers and hotted up the boards and heard them humming under heavy load but they did not trip. I then went to "B" MOV board where I met EF-BF and together we tried to hot up board to get the single RCIC valve 71-2 - It was a dead fault. I told EF-BF to do anything to restore power to that valve. I went back to Unit Ø1 where the operator had started to depressurize using the 4 ADS valves. AUO-Q and I put on air packs to go to the top of the torus to open valves manually to get in torus cooling mode. We opened valve 74-73 and then had to leave because we ran out of air. AUO-Q passed out at the turbine building door. After I got my breath back, I went back to the control room and told AOS-W about the one open valve.

I drew a map of the route to the valve on top of the torus for AUO-UUUU and AUO-MMMM who were going in next to open the other valve. I went back to the cable spreading room and saw that the fire was still going. I got some AUOS and an Athens fireman with an airpack to fight the fire and we passed CO_2 and dry chemicals to him, then other people with air packs fought fire. The fire was finally driven back to the reactor side. I believe that during this effort on the spreader room side there was no one fighting the fire on the reactor side. There was still a lot of shorting and sparking in the cable trays. We left two men to watch the situation and I went back to the control room; was relieved by ASE-EE and went with SE-U to try and restore DC boards. We also ran some lights into the fire area and tried to get a lifeline in, but ran out of air. I worked with SE-R and restored one (1) RX protecting MG set, but went back periodically, to check on the cable spreading room. I was asked to tape my recollections of the day's activities and then, about 10:30 pm, I went home.

I have received some fire fighting instructions, but was not very familiar with the Scott Air Pack.

/s/ Michael V. Annast	/s/ ASE-G	4-25-75
Witness	Signature	Date

I, EA-H, voluntarily give the following information to and answered questions for Howard A. Wilber, Tolbert Young and James Devlin, who have identified themselves to me as representatives of the U.S. Nuclear Regulatory Commission.

I am employed by the Tennessee Valley Authority as an Engineering Aide, SE-2 assigned to the Browns Ferry Nuclear Plant. I am in the Division of Engineering and Construction, Conduit and Grounding Group. I have been so employed since 12/3/74.

On Saturday, March 22, 1975, I was working in Units I and II checking air leaks around penetrations in the walls. EE-B, Electrical Engineer was my supervisor. Three other Engineering Aids and four or five electricians were also assigned to this task. The job consists of checking for air leaks using a lighted candle held close to the penetration that had been sealed by the electricians using foam packing as a backing and RTV (silicon sealant). The material is pressed into the hole by the electrician, poly foam, RTV or urethane, and we check for drafts with a candle.

I was working with Elec.-J, electrician. In the morning we did two windows in the Unit II side and then went to lunch. After lunch at 11:30 AM or 11:45 AM EE-B brought us up to the cable spreader room, 593 elevation. EA-C, Engineering Aid and Elec.-D, electrician were working in the cable spreader room.

A little after twelve thirty while I was on top of a stack of cable trays ten to fifteen feet away from EA-C and Elec.-D I heard one of the two, say that something was on fire. I kept working but when I heard someone hollering for a fire extinguisher I started to climb down. By the time I got a CO_2 fire extinguisher was being used. Another extinguisher was brought in, I brought in a dry chemical extinguisher.

It sounded like they couldn't put the fire out and EA-C said it was burning in the other side of the wall.

I only worked in Unit I & II on one occasion so I didn't know my way around. I asked Elec.-J if he knew how to get to the other side of the wall. Elec.-J did know, we took off.

We saw a person from DPP with a detector and two other persons who had extinguishers. We saw the fire coming through the wall. The smoke was beginning to get heavy, we could hardly see. After the CO_2 was put on the fire we thought the fire was out. Because of the smoke the group began backing away from the fire area. One fellow with a Scott Air Pack walked into the fire area and soon came running out and went to the phone. He said the fire was still burning.

Elec.-J and I went to the first floor. We were told by a person from DPP to leave the Dry Powder extinguishers on the 593 level at the site of the fire and carry the CO_2 extinguishers to the 621 level. On the 621 level we met a person wearing an air pack. Heavy smoke was coming through the floor but we could nto see any fire or flames.

Elec.-J and I went back to the near stairwell but the smoke was too heavy, we tried the elevator but it was not running. In the stairwell next to the elevator the smoke was not too bad. When we reached the second level the lights went out, about then we were met by EE-B and told to go outside.

The first time I used this procedure, that is, testing for air leaks with a candle, was Tuesday of that week. I never tested with the foam exposed. I inspect after the electrician seals the leak. On at least two occasions the RTV fused but it was easily pinched out.

This procedure was taught to me on Tuesday, March 18, 1975, by Elec.-CG. The candle has to be held close, right next to the material. EE-B told us that the sealing material would burn and to be careful not to set it on fire. I did not report that the silicon (RTV) had fused on me because everyone knew that it would.

I never saw a fire drill at Browns Ferry. The electricians knew that the bell was the fire alarm, but I did not know this.

I estimate that it was approximately eight (8) minutes from the time I heard there was a fire to the time I got to the reactor side, at the fire.

There were no emergency lights on the stairwells. In other parts of the plant, the lights were on.

At 12:30 p.m., I noticed the time, the fire was shortly after.

I have read the above statement, consisting of six pages. It is correct to the best of my knowledge.

/s/ EA-H	/s/ C. E. Murphy	April 16, 1975
SIGNED	WITNESS	DATE

I, EA-H, make the following free and voluntary statement to W. S. Little who has identified himself to me as a representative of the U. S. Nuclear Regulatory Commission:

I am an engineering ai[illegible] grade SE-2, employed by TVA. On Tuesday, March 18, I was shown t[illegible] first time how to test conduit and cable tray windows for air leaks [illegible] how electricians seal them, by Elec.-CG. On Tuesday I worked on this with two electricians. The electricians would seal any leaks I would find.

On Saturday afternoon, March 22, I started working in the cable spreading room checking for leaks with an electrician, Elec.-J. We were working on checking conduits which came through the wall and dropped into the rows of cable trays about 10 to 15 feet back from the Reactor Building wall. Where I was at, I was not able to see EA-C and Elec.-D where they were working. I was at the top of a stack of trays and was checking the conduits coming into a junction box, when I heard somebody say something about "Fire." Then I heard somebody ask for a fire extinguisher. I started climbing down from the trays. When I got down, I was passed an extinguisher by somebody at the door and I passed it on. Just before the alarm went off I heard someone say "smother it with rags". Then I heard someone say "It's burning in the wall."

I had only been in the Unit 2 area one time prior to Saturday and I ask the electrician with me if he knew how to get to the other side of the wall. We took an elevator to elevation 565 and then took another elevator to the second floor of the Reactor Building and got over to the area of the fire. There were two guys there - I don't know who they were. One was up high spraying a fire extinguisher, and the other was down low. I could see flames coming out of an overhead cable tray. We backed out of the area because the smoke was starting to fill the area. The guys came down and said nobody else was back in there.

As I started to leave, I met a man putting on a mask and air tank and being helped by another. He went to check the fire. He quickly came back and said the fire was burning. He called somebody on the phone. The electrician and I started hunting for other extinguishers, and 4 or 5 other guys showed up. We took a few extinguishers to the third floor (smoke was starting to get heavy up there) we looked around for fire but I didn't see any. We went to go down the stairs but the smoke was coming up heavy. We went to the alternate stairway because the elevator shut off by now. The lights went out at about the second floor, and I decided to go all the way down. Then I ran into EE-B on the stairs and he told me to go outside.

/s/ William S. Little	/s/ EA-H	April 16, 1975
Witness	Signature	Date

I, Elec.-J, make the following free and voluntary statement to C. E. Murphy who has identified himself to me as a representative of the U. S. Nuclear Regulatory Commission.

I am an electrician employed by TVA. On March 22, I was working with EA-H in the cable spreading room checking leakage from conduit and junction boxes. I was about 25 feet from the area where the fire started. Somebody yelled that there was a fire. I think they thought they put it out but then it started again. Some other electricians and I started to go around to the other side of the wall. Elec.-F was the first one to get to the Reactor Building side. We were all carrying fire extinguishers. Some of the others went up the ladder to put out the fire I thought it was out 2 or 3 times but it kept flaming back up again.

It got so smokey in the area, I couldn't take it anymore so I left. On my way out, I passed the cable spreading room on the way to the control room. I looked into the room - it was dark and the CO_2 had been set off. When I got to the control room, one of the foreman that was there told me to go to the assembly area. I did not return to the plant until Monday.

/s/ C. E. Murphy	/s/ Elec.-J	7-16-75
Witness	Signature	Date

I, PSO-K, make the following free and voluntary statement to C. E. Murphy, who has identified himself to me as a representative of the U. S. Nuclear Regulatory Commission:

I am a Public Safety Officer.

I was stationed on post at the construction portal to the turbine building when a construction man came by and took my fire extinguisher off the wall. I asked him what he was doing and the man said there was a fire in No. 1. I picked up the pax phone and called the shift engineer to tell him there was a possible fire in No. 1. The time was 12:20 or 12:30. I heard the fire alarm go off. I pulled the cards on the persons who had come through the post--to account for them as they came out. I stayed at my post until 2:55 p.m. No equipment was brought in prior to that time. I let the DFP back in. I was relieved by PSO-DC. I reported to PSO-CC and was assigned to help tote fire extinguishers.
I worked until 5:30 or 5:45. I took, in all, about 15 extinguishers up to the cable spreader room. There was one Athens fireman in the room to whom they had to slide the extinguishers under the trays.

I have read the statement summarized above, which is true and correct.

/s/ C. E. Murphy	/s/ PSO-K	4-30-75
WITNESS	SIGNATURE	DATE

I, ASE-L, made the following free statement to representatives of the U. S. Nuclear Regulatory Commission:

I was an assistant shift engineer normally assigned to Unit 2.

On March 22 while I was in the shift engineer's office I received a call from Officer PSO-K. A construction man had just passed him (PSO-K) at the construction entrance portal between Units 2 and 3, and said there was a fire in the Unit 1 Reactor building on the north side. I then dialed 299 (which automatically sets off the fire alarm) which rings the shift engineer's office, Units 1 and 2 and electrical control desk. The operator on Unit 1 answered on the first ring. I identified myself and told him that a fire had been reported on the north side of Unit 1 reactor building. I then told the shift engineer, and upon leaving the shift engineer's office I went to the electrical control desk and "sealed in" the fire alarm system. From here I went through the 4 KV Shutdown Board room "A" and into the reactor building via the emergency entrance elev. 621. I checked out the north side of elev. 621, (3rd floor) and found smoke in the north east corner. There was no smoke present at the SLC level.

I then went down to the second floor. As I was passing the reactor building closed cooling water heat exchangers a man passed me carrying a fire extinguisher. He was coming from behind the reactor. Two other men went around with me, one was AUO-Q, the other a construction worker.

I went around to where the fire was. There was a ladder already in place at the fire area. I went up the ladder. At this time the fire was coming through the reactor building wall and was around the four cable trays nearest me. I used a dry chemical extinguisher which appeared to knock down the fire. I went down the ladder and someone gave me a CO_2 extinguisher. I went up and used it. The fire appeared to go out but after a few seconds it would flare back up. I then went down and called the control room for breathing apparatus. With the help of other personnel I went back up the ladder a few more times. Then some of them started going up as I went back to call the unit one operator, O-M. When I talked to O-M he told me that he was having problems with Panel 9-3 with shorts, lights burning brightly and going out, and different alarms, coming in. I told the Shift Engineer, SE-R, and Unit 1 Assistant Shift Engineer, ASE-G, about the problems and they returned to the control room. By this time the air packs had been brought in so we put them on and went back to fighting the fire. The fire was then 6" to 8" out into the cable trays.

The smoke turned from light to black almost instantly. There was not much air movement in the area. I called O-M and instructed him to have ASE-G set off the CO_2 system in the spreading room. At this time I was told that Unit 2 operator O-KK was having problems and had scrammed Unit 2.

By now it was very difficult to see; so I asked all personnel not fighting the fire to go down to the 1st floor and wait there. It was mostly instrument mechanics and 8 to 10 construction workers. About this time the light went out. I went back to the Shift Engineer's office and told him of the problems we were having and called the Athens Fire Department to come out and be on stand-by which is part of the emergency procedure. It was a little past 1310 at this time.

I checked with O-KK on Unit 2 and he was not having any problems so I went to Unit 1 and worked with O-M for a few minutes then I checked out the 161 KV and 500 KV electrical systems. Both were normal. I then checked the 4 KV shutdown boards and found all four Diesel generators running and tied onto the S.D. board. I reenergized 4 KV shutdown bus 2 from Unit 2. I then transferred the shutdown board from the Diesel Generator back to the #2 shutdown bus, except "C" which would not stay on the normal feed. So it was left on the D/G feed.

I went back to Unit 1 and worked with O-M again on the RHR system. I tried to start "A" RHR pump from the shutdown board. but it tripped out. I then started "B" RHR from shutdown board "C". It ran for a few minutes and was taken off because there was no minimum flow.

I talked to O-M about the condition of Unit 1. So between us we decided to blow down the unit. I instructed O-M to open the four ADS valves which were operable and to leave them open. The water level at this time was > -40". It dropped to > -120" before the condensate booster pumps started putting water into the vessel.

I returned to Unit 2 and conditions were deteriorating so I told O-KK to start depressurizing Unit 2. It was at this time we lost control of the ADS valves.

I called the electricans and instrument mechanics for assistance and then went to the back-up control panel to try to operate the ADS valves but they would not operate from here either.

The drywell control air system was checked and was normal.

I then went to 1-C elevation to check the 120V Pref. system. I got Unit 2 restored to its normal feed then fed Unit 1 from Unit 2. While I was working on Unit 2 Reactor Protection System, SE-R and ASE-G were working on Unit 1, ASE-Y was working with me. The fire in the spreading room had begun to start back so I set off the CO_2 system for about 1 minute. I went back to the Unit 2 control room and the operator informed me that he was now able to depressurize. I again went to the electrical control panel and had to realign part of the 4 KV shutdown boards.

At about 1500 SE-V was in Unit 2 control room so I went back to the reactor building via shutdown board room "B", accompanied by

ASE-Y to start fighting the fire. We strung life lines and lights then started organizing groups to fight the fire as well as going in ourselves. I took a 150# dry chemical extinguisher to the fire and used it. We ran out of air for the Air Packs about 1830 and started using Chem-ox units.

Water was used about 1900 and the fire was extinguished.

Two days later I heard of the fire which had occurred on the 20th of March.

The summarized statement above is true and correct to the best of my recollection of the incidents which occurred on March 26, 1975.

/s/ BA	/s/ ASE-L	May 17, 1975
WITNESS	SIGNATURE	
NOTARY PUBLIC		

Summary of Interview with O-M

I,O-M, voluntarily give the following information to Don Caphton, Tolbert Young, Jr. and James Devlin, who have identified themselves to me as representatives of the U. S. Nuclear Regulatory Commission. I am a Unit Operator and I have been employed in the Division of Power Production, TVA since June 1972 at Browns Ferry Nuclear Plant.

On Saturday March 22, 1975, I was assigned as Unit Operator (UO) of Unit 1 on the day shift. At 12:35 I received a call on the fire phone from ASE-L, Assistant Shift Engineer (ASE) (a call on this phone automatically initiates the fire alarm through out the plant) who stated that "there was a fire in Unit 1 reactor building, backside." I made an announcement over the PA system "Fire in reactor building, unknown location." ASE-L then went to the reactor building to locate the fire. Assistant Unit operator (ASU) AUO-Q called and stated the exact location to the second floor of the reactor building in a cable tray. This announcement was made over the PA system.

The first alarm came in "Reactor low level auto blowdown permissive" on panel 9-3. I checked panel 9-5 to confirm but everything on panel 9-5 was ok. The second alarm was Core Spray (CS), Reactor Heat Removal (RHR) pumps running" I checked but the pumps were not running. A 3rd alarm came in "Core Cooling system diesel generator initiate. I called O-KK told him to check and see if the diesels were running.

ASE-L then called for additional help and some Scott Air Paks to fight the fire. I sent O-LL to help. I told ASE-L that help was needed in Unit I control room as conditions were unstable. ASE-G was sent to help me. I called O-MM in Unit III to come to Unit 1 and help.

Alarms from the shutdown panels then came in, "RHR, CS, HPCI and RCIC" all running. More alarms came in. I checked panel 9-5, indications from panel 9-5 were that everything was normal. I then tripped all emergency pumps.

I then noticed that the APRM's were decreasing, reactor water level showed a slight increase. The recirc pumps started running back. About this time ASE-G and SE-R came into the control room to help.

At 12:48 the RHR, CS and HPCI initiated again and I realized that the unit was too unstable. The lights on panel 9-3 began getting bright and then getting dim. These were more unknown conditions.

I told SE-R "lets scram the unit." SE-R said ok.

I then zeroed the master manual controller for the recirc. pumps and then began to zero the individual controllers but the recirc. pumps tripped. I immediately tripped the reactor and put the mode switch in shutdown. I noted the time was 12:51 and that the Mwe meter read 704 Mwe.

At about 12:53 the turbine was tripped by ASE-G. I then tripped two of the feedwater (FW) pumps and ran the level stepoint back to zero.

Selected and started driving in the IRM's and SRM's. Checked that all rods were in and that I had the single rod permissive light.

Reactor water level caught at 5 inches and returned to 47 inches. I then adjusted the level setpoint to 33 inches. The steam dump bypass valve was open with normal control.

At about 1:10 the outboard MSIV's closed. Reactor pressure started increasing and the relief valves began opening.

I checked and found that the only water supply to the reactor at this time was the control rod drive (CRD) pumps so I increased its output to max. I then manually opened one relief valve and blewdown to about 850 psi. About this time we lost AC lighting and the DC lights came on. The pressure started right back up. At least five relief valves started popping again.

At about 1:15 I lost my nuclear instrumentation. I only had control of four relief valves. I noticed that the condensate and the booster pumps were ok.

At about 1:30 the Operations and the Assistant Operations Supervisor arrived, shortly thereafter the Plant Superintendent arrived.

At this time I knew that the reactor water level could not be maintained and I was concerned about uncovering the core.

ASE-L came back to the control room and I told him that I needed to blowdown. ASE-L said "do it."

I started blowing down with four relief valves. The pressure was down to 300 psi in about 20 min. and the min. water level was - 120 in. on the Yarway when water started entering the vessel from the booster pumps. The level increased to normal but I lost auto signal to the FW pumps bypass valve. Level increased above 50 in. I then sent an AUO to manually isolate the bypass line. When the level came back on scale, I had the AUO manually adjust the flow until level was maintained.

During this time I made a list of RHR valves needed in order to obtain torus cooling.

About 7 p.m. we finally got lined up to discharge from the torus to radwaste.

The reactor cleanup system was isolated until about 8 p.m.

At about 5 p.m. I lost air to the four relief valves that I had control of manually. Pressure rose to 600 psi before I got control of the valves again.

About 2 p.m. I put on a scott air pak for about 10 min.

The max. number of people in the Unit I end of the control room at any one time was about 20 people.

I was relieved at 8 p.m. The conditions of the plant at that time were: relief valve open, booster pump on, one CRD pump on, reactor water level stable. RHR pumps still out. No torus temperature or level indication and no torus cooling.

/s/ O-M 4/13/75

Witnessed: /s/ Willaim S. Little

I, AP-N, make the following free and voluntary statement to Michael V. Annast who has identified himself to me as a representative of the U. S. Nuclear Regulatory Commission:

I am an apprentice-pipefitter, welder for TVA at the Browns Ferry plant. On March 22, 1975, shortly after noon, I was coming down in the elevator from the second floor of the reactor building. As I came out of the elevator on the first floor I met a construction man (who was later identified to me as Elec.-F) who had a fire extinguisher in his hand and who was very nervous. I asked him what was the trouble and he said, "There is a fire on the second floor". Elec.-F entered the elevator to go to the second floor. I ran up the stairs back up to the second floor and met Elec.-F as he was coming out of the elevator there. I ran to the south wall and got a dry chemical fire extinguisher from there. As I did this, I passed by the welding crew I had been working with, SW-P and P-O, who asked me what was happening. I told them there was a fire at the opposite side of the reactor. I then went around back of the reactor with Elec.-F, and P-O and SW-P followed. Elec.-F did not know exactly where the fire was and we went up on the grating (or mezzanine) looking for it. I saw the fire then on the bottom cable tray on the east side away from the mezzanine. Elec.-E was moving a ladder ove to within about five feet from the tray. I could see the flames which extended about 8 inches above the tray and extending out from the wall about two feet into a semi-circle in the tray. There was no material dripping from the tray and no other trays were on fire. Elec.-F and I got down from the mezzanine. I went up to the top of the ladder and Elec.-F was behind me at bottom of ladder. There was a scaffolding between me and the tray, but I was able to reach out with the extinguisher and discharge the dry chemical directly on the fire. The fire went out, the flames were gone and the glow was gone. Elec.-F and the others said the fire was out. I was in a cloud of smoke from the fire and the extinguisher, and I had difficulty breathing so I came down from the ladder fast as I was about to pass out. As I came down O-KK and SW-P carried me around to the other side of reactor so I could recover. I did not go back to the fire area. I did not know that the fire had started up again after I had put it out, until later. The smoke that I observed as coming from the flames was white in color and it was rolling up from the tray. There was no

(continued)

Page 2 - Interview of AP-N

crackling or popping. I could not tell whether the cable insulation was on fire. I did not observe anything in the tray which appeared to be melting or flowing into the tray from the hole through the wall.

I have read the statement summarized above, which is true and correct.

/s/ Michael V. Annast
Witness

/s/ AP-N

4-25-75
Date

I, P-O, make the following free and voluntary statement to C. E. Murphy, who has identified himself to me as a representative of the U. S. Nuclear Regulatory Commission:

I am a pipefitter. On March 22 when the fire occurred I was working on the second floor of the reactor building with SW-P and AP-N. My first knowledge of the fire was when I saw AP-N come by with a fire extinguisher in his hand. I could then see smoke and I called the shift engineer to report the fire. I watched as AP-N went up the ladder to fight the fire The smoke coming from the cable tray was rust colored. AP-N fell as he was coming down off the ladder. The flames were gone, but the tray was still smoking when AP-N came down. It was five minutes before the operations people got there. SW-P and I carried AP-N to the south side of the reactor when he dropped down again. We then went outside for several minutes--this was about 1:20. The elevator was not working. The radiation monitor went off. The air monitor went off. A tank of CO_2 was brought in on wheels by men wearing masks and I heard it go off.

I have read the statement summarized above, which is true and correct.

/s/ C. E. Murphy	/s/ P-O	5-6-75
WITNESS	SIGNATURE	DATE

I, SW-P, voluntarily give the following information to H. Wilber, T. Young and J. Devlin, who have identified themselves to me as representatives of the U. S. Nuclear Regulatory Commission. I am a steamfitter-welder. I have been hired for work by the TVA - DEC at Browns Ferry off and on for past two years. I reside at Athens, Ala.

On Saturday I was working in Unit I on the 593 elevation, second floor, welding in the southwest corner of the reactor building, is on the opposite side of the reactor from where the fire occurred. P-O steamfitter told us there was a fire on the other side of the reactor.

We went around the reactor and located the fire. It was high up in the cable trays. We needed a ladder which we brought over. AP-N climbed the ladder with an extinguisher and tried to put out the fire with the CO_2.

P-O went to the phone and called the Shift Engineer and reported the fire. As far as I know this was the first report.

AP-N was told of the fire by an electrician, believe his name is Elec.-F.

The smoke was getting heavy, soon some DPP people arrived, two then three at a time. AP-N came down off the ladder and could hardly breathe because of the smoke. We had to take him outside.

When we first arrived at the fire the flames were not big, it was just getting started.

Shortly after we moved out the DPP people also had to leave because of smoke. They got Scott Air Paks. The smoke at first was white but it soon turned black.

I guess it was about 30 minutes from the time we saw the fire to the time we left the reactor building.

AP-N told us he thought he got the fire out but after taking AP-N outside I returned and the DPP people told me it was really going. We went in to gather up our equipment but the smoke was getting too thick, particularly along the floor.

We did not know that the bells we heard was the fire alarm. We did not pay any attention to it because we are always hearing this code call. We did not hear a siren which we associate with an evacuation.

AP-N had discharged one CO_2 extinguisher, another was available that the electricians brought in, but I do not know if it was used.

There were about 15 people at the fire, some were members of the fire brigade, I don't know who the others were.

The lights did not go out, the elevator was not working.

It did not appear to me that DPP was organized for this type of thing.

The work we were doing was on the EECW (Emergency Cooling Water).

When outside we reported to our Foreman F-CV Steamfitters Welders. TK-CW checked us off and sent us home.

/s/ C. E. Murphy	/s/ SW-P	4-16-75
Witness	Signature	Date

7:05 a.m.
4/11/75

I, AUO-Q, make the following statement freely and voluntarily to Michael V. Annast, who has identified himself to me as a representative of the U. S. Nuclear Regulatory Commission:

I am an assistant unit operator and was working in the radwaste building on the day shift, on March 22, 1975, at the Browns Ferry plant. At about 12:40 p.m., I heard the fire alarm and called the unit operator who told me that the fire was in the reactor building I got a CO2 and a dry chemical extinguisher and went to the reactor building. On the way, I met two others and we decided that each of us would check one floor. When I got off the elevator on the second floor, I smelled smoke. I went behind the reactor and saw 3-4 others there. One man was on the ladder putting CO2 on a relatively small fire - maybe a foot-high flame. The flame looked like a torch because of the draft from the other side and it was difficult to put chemical on. At this time I called the unit operator and told him the exact location of the fire. Then there was an electrical arc and the man on the ladder fell down. ASE-L, who had gotten there at the same time as I did, told me to go and get some Scott air-packs because the smoke was getting very bad. The cables were arcing and the insulation seemed to be bubbling. I believe that at this point, we could have put the fire out, had we been able to get at it and had there been no draft caused by the negative pressure. I went to the HP Lab and got some airpacks. I asked maintenance people, who were standing around, to help me carry them to the fire area. About half of the airpacks were unusable because they were marked "not charged" or "no mask." (*See Note #1) I put on an air pack and went up he ladder. Something that looked like tar was melting and running down e wall. The smoke was very heavy. I used two more extinguishers and ran out of air. We fell back to the far end of the heat exchangers. ASE-L told me they were bringing both units down and he left for a few minutes to help with the shutdown. We all withdrew to the first floor. I put on a fresh airpack and went back to make sure that no one else remained behind. On the way back, I had to use the stairs because the elevator had stopped. I then called the control room for another ASE to direct the fire fighting. ASE-OO arrived and about this time one of the units scrammed. I went to the control and talked to O-MM, who was trying to get cooldown. He asked if I knew where the exact location of valves 74-71 and 74-73 on top of the torus was. I told him that I was familiar with the valves and he asked me to get them open. I went back to the reactor building and noticed that the smoke was on the first floor. Everyone had fallen back behind the double doors on the Unit #2 side. I put on an air pack and went down the ladder on top of the torus. I got #74-71 partially open and ran out of air. I then changed air packs and, together with ASE-G , got 74-71 fully open. We started on 74-73 when we heard the safeties lifting. ASE-G indicated that we should go.

*Note #1: In the statement marked #1 "About half of the airpacks were unusable, because they were marked "Not charged" or "No mask." "This is very misleading. The air packs that were brought to the fire sight by some of the maintenance people were unusable due to marking. They simply failed to notice the marking. We had enough air packs at the sight but a lot of the air packs brought to the fire sight were not charged or had the mask in them.

I hit my mask against something and got a leak. We were also out of air. I got quite a bit of smoke and passed out just outside the watertight doors. I remember reviving on top of a table in the lunchroom. I don't know how I got there. I rested a bit and, since the first aid room was locked, (I heard that it was opened at 4:30) went back to the control room. At this time I saw the Athens Fire Department enter the plant. SE-R and SE-PP asked me what the conditions were in the fire area. We decided to try to pull some smoke from the reactor building. I went with EF-BF , an electrician foreman, to the top of the reactor building where we jumpered a fan. When we first tried it, there was no smoke, so I went to the refueling floor and manually opened the dampers which had isolated. I ran the fan for 5 minutes and was told on the radio to shut it off. SE-PP then told me to turn it back on for 15 minutes and turn it off. I continued operating the fan 4-5 times on direction by radio from the control room. At about 10:30 p.m., SE-PP told me to come back to the control room. Things seemed quite normal by then and I was told to go home.

/s/ Michael V. Annast	/s/ AUO-Q	4-25-75
WITNESS	SIGNATURE	DATE

I, 'SE-R , make the following free and voluntary statement to C. E. Murphy who has identified himself to me as a representative of the U. S. Nuclear Regulatory Commission:

I am a shift engineer at BFNP. On March 22. I was on the 7-3 shift with ASE ASE-G on Unit 1, ASE-L on Unit 2, ASE-OO on Unit 3. At 12:35 a call came in on the S.E. office phone. ASE-L answered the call, which was from Public Safety Officer PSO-K who reported that a construction worker had said that there was a fire on the back side of Unit 1 reactor. ASE-L relayed or repeated the message to me and then initiated the fire alarm (Pax 299) which was answered by the Unit 1 control room operator (O-M) who was given the information we had received and then continued with the fire alarm procedure. I told ASE-L to locate the fire and call me back, however, before receiving a report on the fire I proceeded to the Unit 1 reactor building. I met ASE-L on the second floor and he described the fire and its location as being a cable tray penetration to the spreading room. We returned to the fire and ASE-L made a second attempt to control the flames with a (20 lb.) dry chemical unit while standing on a ladder that was already in place, however, the effect on the fire was very slight, and the dense smoke in the area made further efforts without breathing equipment near impossible.

Maintenance personnel and other operators were reporting to the fire area with additional portable dry and CO_2 units. ASE-L called Unit 1 control room for an operator to bring breathing equipment. At this time the control room operator reported receiving some false alarms apparently related to the fire. I returned to the control room while ASE-L remained at the fire. The additional personnel were sent after breathing gear and to return to the fire. When I arrived in the control room the Unit 1 had shortly before experienced an initiation of the ECCS. The operator, assisted by (ASE-G) Unit 1 ASE secured the HPCI, RCIC, RHR, CS pumps as they were not required, however, the D/G's (4) which had started were left running and available. In conjunction with the ECCS starts were indications of electrical faults occuring in the ECCS motor oper. valve controls. I asked the operator to drop the core flow and manual scram the reactor, which he did. The turbine was tripped, RPV pressure remained near normal with some spill thru the bypass valves to the condenser. The operator reduced the one (1) feed pump with RPC level near +47". All control rods were indicated full in. This condition existed for only a few minutes, due to losses of both 4800 shutdown bds., 4800 and 250V DC Rx MOV bds., we lost RPS, unit preferred and I&C 120V sources as well, which prevented further operation of the feed pump, and reduced the number of manually operable relief valves to four (4) which were powered from the 250V DC Rx MOV bd. 1C that apparently was not being affected by the fire. The operator controlled the pressure below the safety relief valve settings (1080-1100 psi) by manual operation of these valves.

At approximately 1300 hrs we scrammed Unit 2 due to upset 480V shutdown bds., resulting from fire damage to the 4 KV feed to a 4800 transformer. Unit 2 ECCS required after the scram operated normally. A loss of control air to the relief valves later prevented manual control until air was restored.

The continuing loss of level in Unit 1 reactor and the difficulty and indefinite delays in restoring the RCIC to an operable condition made the choice of depressurizing by manual operation of relief valves available to approximately 350 psi to enable restoring level with the running condensate and condensate booster pumps, the only choice available at the time.

Both Unit 1 and Unit 2 were depressurized to the suppression pool following MSIV closure. Operators and electricians continued to restore needed power supplies where possible.

The fire in the reactor building was finally controlled by the use of a large cart type dry chemical unit and water hose from the HP fire system.

The CO_2 deluge system for the cable spreading room was used on three (3) occasions following the reactor scrams in an attempt to help control the fire, however, the fire spread most rapidly from the penetration into the reactor building with the air flow.

Concerning the March 20 fire I reported of similar origin, it was first reported to me by the foreman of the crew doing the work. I told ASE-EE to make a visual check, and I made a journal entry giving some details, including location and identification of the tray involved. Elec. Dept. was reminded of the occurrence as is routine in such cases.

Plant management personnel and the Athens Fire Department were notified between 1310 and 1330 hours following our determination of the critical nature of the fire and its potential effects to plant safety systems and the maintaining of safe reactor conditions, etc.

I received my most recent fire training at the training station located at Murfreesboro, Tennessee. This training included fundamentals of fire prevention, firefighting with dry chemical, CO_2, and some use of water.

Concerning fire drills, I am not sure when the last one was held or if Athens Fire Department responded. The use of air-pacs is primarily related to HP drills and used in airborne areas. During and following the fire, operators and electricians were the most needed skills for reestablishing safe reactor conditions. According to ASE-L statement to me there was a period of about two hours during which very little was being done to control the fire. This was primarily due to a shortage of airpacks, Chem-Ox units, lack of lights in the reactor building and lack of large cart type dry chemical units on the second elevation. The use of raw water was considered a personnel risk during the earlier phase of the fire due to energized circuits in the area of the burn.

As all areas of the Unit 1 reactor building required breathing equipment to enter, the demand for this equipment eventually exceeded the supply available.

I have read the statement above, which is true and correct.

/s/ C. E. Murphy	/s/ SE-R	5-8-75
Witness	Signature	Date

I, CE-S, make the following free and voluntary statement to Howard A. Wilber who has identified himself to me as a representative of the U. S. Nuclear Regulatory Commission:

I am a chemical engineer at BFNP. On March 22 I was at home when I was called about 1:30 or 2:00 pm and notified of the fire. I called and notified PRS-EEE, my supervisor, and stood by until aboiut 3:00 p.m. when I received a call to come out to the plant. I arrived at the plant at approximately 3:30 p.m. and reported to P-Supt. and PRS-EEE. PRS-EEE asked me to evaluate the plant effluent releases, particularly gaseous, and take any appropriate actions deemed necessary by the circumstances.

By observing the recorders associated with the continuous air monitors located in the control room I noted that the following monitors were out of service: 1-RM-90-250 and 2-RM-90-250. These instruments measure and record the airborne activity released from the plant via the turbine and reactor building ventilation systems for Units 1 and 2 respectively. Since there was no power available in the reactor building the normal backup sampling program utilizing a vacuum pump and Marinelli containers could not be used. Therefore, gaseous samples were taken by filling one liter polyethylene bottles with water and then inverting them in the airstream being sampled. The water would pour out of the bottle and would be displaced by air. The bottles containing the air samples would then be sealed and taken to the radio chemical laboratory for analysis. Each air sample was quantitatively analyzed for its concentrations of gamma-emitting isotopes. This sample program was started ~ 1645 hours 3/22 and continued until ~ 1600 hours on 3/23 when 1-RM-90-250 was returned to service. Grab sample frequency was ~ 1 per hour. Upon analysis, all values determined by this method were found to be below technical specification limits.

A small gasoline driven sample pump was used to sample periodically for iodine and particulates.

The Unit 1 reactor building exhaust fan was returned to service at ~ 1645 hours 3/22 coincidental with the start of the grab sampling program. The standby gas treatment system was in service during the incident. The stack monitor remained operable throughout the incident.

The primary coolant is monitored routinely for conductivity, chloride concentration, and isotopic content in compliance with the technical specification requirements. The continuous conductivity monitors on Units 1 and 2 were lost as a result of the fire. A grab sampling program was begun to monitor the primary coolant for conductivity and chlorides. Normal backup sample point and techniques were used for this program. Grab samples were taken ~ once every 4 hours. I am not aware of any instance when technical specification limits for primary coolant parameters were exceeded.

There were no detectable chlorides (less than 50 pph) in Units 1 or 2's primary coolant and the conductivities have stabilized well below 1.0 umho/cm at 25oC.

I have read the statement summarized above, which is true & correct.

/s/ CE-S

/s/ Howard A. Wilber
Witness

4/24/75
Date

I, HPS-T, voluntarily give the following information to Floyd Cantrell, Tolbert Young, and James Devlin, who have identified themselves to me as representatives of the U. S. Nuclear Regulatory Commission. I am the Health Physics Supervisor and have been employed at BFNP since October 1974.

I learned of the fire at about 1:30 p.m. At the time of the fire there was one health physics technician on duty. At 2:30 p.m., two more health physics technician personnel arrived onsite and one of them called me, told me of the seriousness of the fire and passed on the plant superintendent's need for three more health physics personnel. I arranged for these three personnel to report to the site by 3:30 p.m I and HP-CL, a health physicist, arrived onsite at 4:00 p.m. HP-BJ, the former BFNP HP supervisor arrived onsite at 4:15 p.m.

The airborne activity in the reactor building at the time of my arrival was 35% of MPC of Rb-88. No other isotopes were identified. At about 5 PM the same isotope at about 25% of MPC was identified in the turbine building.

The ventilation in Unit 2 reactor building was out during the incident until 2 AM, 3-23-75, and Unit 1 ventilation was out for about three hours between 2 and 5 PM. After the ventilation for Unit 1 reactor building was turned back on, the activity began to clear up. A downward trend was noted on the air monitor at about 7:30 p.m.

We took samples in the reactor building, the turbine building, and also in Unit 3. There were no significant problems with radiation or contamination throughout the incident. There were about 35 air samples taken between the time of the fire and 4 AM the next morning.

All personnel and their clothing were checked before they left the plant. Between 12 and 15 men were whole body counted on 3/24/75, with negative results.

/s/ C. E. Murphy	/s/ HPS-T	May 6, 1975
WITNESS	SIGNATURE	DATE

I, SE-U, voluntarily give the following information to Donald Caphton, Tolbert Young and James Devlin, who have identified themselves to me as members of the Nuclear Regulatory Commission.

I am a Shift Engineer and have been employed by the Division of Power Production, TVA since 1971 at the Browns Ferry Nuclear Plant.

I was called at 2 pm on 3/22/75 and told about the fire. I arrived on site at 3 pm, went to the control room and talked to SE-V. We agreed that I should go and fight the fire. I talked to AOS-W, OS-DD, and P-Supt. then went to the cable spreading room. The CO? flooding system had been set off three different times. AUO-X and some public safety people were in the spreader room putting dry chemicals on the fire. The fire in the spreader room seemed to be contained.

Shortly after 4 pm I went to the reactor building. I had some discussions with the Athens FD personnel.

At about 4:30 we opened the air lock behind the "B" 4KV Shutdown room. To my knowledge this was the first entry after the evacuation. The reactor building was completely filled with smoke. We generally went in in threes. There was no tracks on the floor. When the air lock was opened there was a negative pressure in the reactor building because the air flowed inward. We set up some DC lights inside and outside of the reactor building then three men went in.

We found the fire going strong in two places and smoldering in two other places (2). The hottest fire was in the cable trays straight out of the penetration about 30 ft. Flames were coming out of each of five trays. In the tray to the west was a lesser fire and there was smoldering in (2) other trays. We concentrated on the hottest fire at time of fighting.

I sent in several men with dry chemicals. However we weren't making much headway.

I then took the Athens Fire Chief in and reported back to P-Supt. on the fire while men were stringing in a rope life line.

We discussed situation with P-Supt. I had already pulled the standpipe water hose out of the rack and tested it for water pressure.

It was around 5 pm and we still considered the use of water too risky. So we continued to fight the fire with dry chemicals. P-Supt. authorized the use of water if I thought it necessary. Also at this time we discussed the conditions of both reactors. Both were shutdown, both cores covered with water and as good shape as could be expected. All of this was discussed in preparation of using water.

P-Supt. suggested taking in the Athens firemen and letting them man the hoses.

The fire chief, maybe one other man and I went in to try the water. The nozzle would give spray but not much force. We were able to hit the bottom of the bottom tray. We went out and the fire chief stated that we needed a different type of nozzle. He obtained one from his truck but we found that it would not fit the hose threads. We put the old nozzle back on and went out to hold discussions. SE-V came down, then he, AUO-AA, and I went in.

SE-V took hose up a scaffold and put some water on the fire knocking down the fire pretty good. He then got into some trouble with his air. He came down pretty fast, dropping the hose which hung onto some scaffolding. We came down checking on SE-V and looked all the way back to the door and found him in the shutdown panel room. AUO-AA and I went back in, the hose was still on and we sprayed water on the fire and wet the whole area down. Some other people went in and put out the other fires and wet the area down again.

I got back in about 6:15 and all of the fires were out. We put more water on wetting and cooling down. At about 6:35 I reported the fires out.

Smoke never came out of air lock.

I had one man on the door counting heads in and out of the reactor building.

I made reports and assigned fire watches. I had construction install some scaffolding and went home about 3 am 3/23/75.

I had three days of fire training last year and I had a fire drill on my shift last year.

/s/ Michael V. Annast	/s/ SE-U	4-23-75
WITNESS	SIGNATURE	DATE

I, SEV, voluntarily give the following information to Don Capthon, Tolbert Young, Jr., and James Devlin, who have identified themselves to me as representatives of the U. S. Nuclear Regulatory Commission. I am a Shift Engineer (SE) and I have been employed in the Division of Power Production, TVA since 1953,

I was called at 12:55 p.m. on 3/22/75 and told about the fire. I arrived on site about 1:40 p.m., and I went to Unit 1 control room. I noted that Panels 9-3 and 9-4 were out and part of Panel 9-5 was out. The pressure was low and the condensate booster pump was making up water to the reactor. I talked to the operator and he told me that he had things under control. I then went to Unit 2.

Reactor pressure 1000 psi and having trouble getting relief valves open. Then Unit 2 isolated from the steam tunnel, hi temperature. About five minutes later we got two relief valves open. We had CRD pumps and RCIC with normal water level. One relief valve was popping and closing.

After the valves were open, I told the operators to keep on this path depressurizing.

I think one valve stuck open because after pressure fell to below 600 psi, we didn't have to open relief valves any more.

We could not get out of the isolated condition.

We had no more problems with level and temperature in the reactor. We then worked on getting the condensate and booster pumps in operation. Shortly after 3 PM, both were running. We later got the main steam drain valve open so we could use the main condenser as a heat sink. Stayed with Unit 2 until about 6:30 stabilizing conditions.

I then went to "B" 4160 shutdown panel and met with SE-U, obtain three Chem-ox masks. Then, SE-U, AUO-AA, and I signed into the reactor building.

At about 7 PM, I decided to use water on the fire. The hose was out of the rack on the floor (some water was on the floor). The fire was still going pretty good.

I then tested the nozzle against the wall, went up the ladder with the hose under my arm, and climbed up on a wooden scaffold underneath fire and sprayed water on the fire. Smoke was getting to me so I hooked the nozzle in the tray with the water spraying and left the room. At about 7:20, we went back in and sprayed the whole area down with water and the fire was declared out.

I knew of the fire on Thursday night, 3/20/75, discussed it with my ASE's and SE-PP (S.E.) who said it had been reported. P-Supt. asked me if I had control of Unit 2 and if everything was O.K. almost continuously after about 20 minutes of my arrival in the control room.

/s/ C. E. Murphy	/s/ SE-V	5-7-75
WITNESS	SIGNATURE	DATE

I, AOS-W, make the following free and voluntary statement to C. E. Murphy, who has identified himself to me as a representative of the U. S. Nuclear Regulatory Commission:

I am the Assistant Operations Supervisor. I was at home on Saturday, March 22, when I was called to the plant by the shift engineer, SE-R, at approximately 1:00 p.m. I was told there was a fire. I arrived at the plant at approximately 1:35 and went straight to the control room. It was full of smoke and the operators were wearing Scott air packs. I saw Messrs. P-Supt. and OS-DD on Unit 1 (the superintendent and operations supervisor). Unit 1 had scrammed. The operator said that all the control rods had gone in when the reactor scrammed. I looked over to panel 9-3 and noted no emergency cooling was available (all the lights were out). The relief valves were lifting. There was no means to get the water in. P-Supt., OS-DD, and I talked about the following actions:

1. Blowing the pressure down to 350 psi so that they could use the condensate pumps and condensate booster pumps to bring water in to keep the core covered.

2. Possibility of the use of river water.

OS-DD told me in the presence of O-M, to continue to blow reactor pressure down to where the condensate booster pump could pump water into reactor vessel. When H_2O from the condensate booster pump started entering the vessel, water level at this time was minus 120 inches in the vessel, or about 4 feet above the fuel. Two yarways were on panel 9-5. They left the relief valves open. I saw the flow pick up on the condensate pump--the level recorder started to respond. At about 33" which is normal water level, I told the operator to come up to approximately 45", but at that point, they lost control of pneumatic valve operation. The unit operator sent the assistant unit operator to close the isolation valve. We had trouble in the telephone communication. As a result, the level was stopped at 60". The water temperature entering vessel was 70 to 80 degrees. At about 1545 (3:45 p.m.), everyting looked better. The pressure was at 200#. OS-DD went to unit 2, and reported problems so I went to Unit 2.

At Unit 2, the relief valves were lifting because of pressure which was 1000 psi. The HPCI and the RCIC were tripped off by high water. There was trouble with the loop 1 RHR pump. The A and C core spray pumps were not available. We decided to blow the pressure down to keep the core covered. We tried to open the relief valves electrically, but could not get them open. The level had started coming down because of the lifting of relief valves. I told the operator to try to get the HPCI on - no luck - I told the operator to put the RCIC on. The water level was approximately normal.

We could not get (RCIC) to operate on manual, but got it on automatic. About this time, the assistant unit operator got the relief valves to open and unit operator got the RCIC on. The level on Unit 2 reactor never got below normal. The extra shift engineer, SE-V, came in I told him to continue to blow down Unit 2, to where he could get water in through the condensate booster pump.

At 1545, the status of the two units was as follows:

Unit 1: (SE-PP, Shift Engineer, assigned to unit)

The pressure in the reactor was 200# - level normal.
There was no reactor protective system.
There was no neutron monitoring system.
There was no HPCI or RCIC.
The emergency core cooling valves were inoperable.
The turbine was off of the turning gear.
The torus level was unknown.
The drywell pressure was at about 2.2 lbs.
The drywell temperature was erratic.
Two drywell blowers were on.

Unit 2: (SE-V, Shift Engineer, assigned to unit)

The main turbine was on turning gear
Reactor water level normal
Reactor pressure at about 100# psi
RCIC available and running
HPCI was out
Torus level at minus 2
Torus temperature 132°F.
(Torus cooling was on when I first got there)
Drywell temperature 140 degrees
Drywell pressure normal (atmospheric)
Both drywells (units 1 and 2) inerted
No A or C RHR or core spray pumps

Sometime earlier SE-U, another shift engineer, had come in. I had put him to work on the fire. SE-SSSS was also assigned to help SE-U with the fire.

At 1630, we got Unit 1 main turbine back on turning gear.

At about 1800, the pressure started coming back up on Unit 1. We decided that we had either lost voltage or control air to the relief valves. I got it checked and found that we had control voltage. At that time, SE-PP went to check the air at the drywell - the isolation solenoid operated valve was disabled. He got the electricians to bypass it to get air. This took until 2150. By this time, the pressure on reactor had increased to 500-600 psi and level dropped to minus approximately 100 inches.

At 1930, SE-U and SE-V reported that the fire had been put out.

At 2010, Unit 2 started pumping from the torus to the hotwell. At 2040, they got the 2" vent from the drywell to the standby gas treatment system open on Unit 1. I observed stack monitor radiation level at this time.

At about 2040, they were flushing the RHR on Unit 2. At 2150, they got air back on pressure relief valves Unit 1. At 2240, Unit 2 was in shutdown cooling. At 2300 torus temperature on Unit 1 was 145 degrees. The flange temperature went off scale. They assumed that the thermocouple wire was shorting. The drywell also showed a sudden increase for possibly the same reason.

At 2400, the Unit 1 reactor pressure was 100? psi and 305 degrees. Torus temperature was 145 degrees. With no torus level indication. Drywell pressure was unknown, but was vented through the 2" line through SGTS. One relief valve was open. Lining up RHR System II to flush for shutdown cooling.

At 2400, the Unit 2 reactor pressure was zero, the moderator temperature was 180 degrees, torus temperature 118 degrees, torus level at a minus 4, in shutdown cooling. Drywell pressure was plus 0.3 psi.

At 0100, two SRM's were temporarily hooked up just outside Unit 1 drywell elevation 565 for Unit 1. They registered 10 counts per second on each. They were being continously monitored and attended by a licensed reactor operator at all times. I was informed by IE-JJJJ that he had calibrated them.

0130 - Unit 1 "C" RHR pump in torus cooling mode. Torus temperature was 177 degrees.

0220 - There was an indication on Unit 1 that the torus level was restored. The drywell pressure was restored - 2.3 psig. We had restored some MOV boards. We tested the valves and the A and C core spray pumps on Unit 1.

0410 - Unit 1 put into shutdown cooling using System II of RHR.

0530 - <u>Unit 1</u> zero reactor pressure
Moderator temperature was 208 degrees
Water level was plus 60
Torus temperature was 153 degrees
Torus level was plus 5
Drywell temperature (no reading)
Drywell pressure zero
Torus cooling and shutdown cooling was in progress

0530 - Unit 2 pressure was zero

Moderator temperature was 165 degrees
Normal water level
Torus level plus 1.5"
Torus temperature 122 degrees
Drywell temperature 100 degrees
Drywell pressure zero
In shutdown cooling

0630 - I asked an instrument engineer, IE-AV, to check the drywell temperature on the cable at the drywell. It was 110 degrees.

I had not been aware of the use of candles to detect leaks prior to the fire. The only other fire I was aware of was the one logged in shift engineer March 20, 1975, at 2230 and a previous fire when rags caught on fire from welding.

I have read the statement summarized above, which is true and correct to the best of my recollection. I willingly sign this statement only after making known in this document the fact that I was not informed, prior to this interview, that I would be expected to endorse this statement as a legal document.

/s/ C. E. Murphy	/s/ AOS-W	5/7/75
WITNESS	SIGNATURE	DATE

I, ASE-Y, voluntarily give the following information to Donald Caphton, Tolbert Young and James Devlin, who have identified themselves to me as representatives of the U. S. Nuclear Regulatory Commission. I am an Assistant Shift Engineer and have been employed by the Division of Power Production, TVA since September 1970 at the Browns Ferry Nuclear Plant.

I was called in between 1 and 1:30 the day of the fire. I reported in between 1:30 and 2 pm. At the time of my arrival both units were scrammed and since I'm assigned to Unit II, I went to the Unit II console. For the first hour I worked on getting relief valves open, checked some fuses then went to put in the aux instrument room the RPS M-G sets.

ASE-L the assigned ASE on Unit II was in and out of the control room. I then ran into PSO-CC in the control room. At this time I thought the fire was out in the spreader room since it had been flooded with CO2. PSO-CC wanted someone to enter the spreader rm with him, to check fire site. I went into the spreader room with PSO-CC and found the fire still glowing. We put some CO2 and dry chemicals on the fire and got some more people to do the same.

At the time I left the spreader room the fire was not out but was under control.

ASE-L told me that the fire was still going in the reactor building. We went in to string some emergency lights. When we got the fire we sprayed some on CO2 and/or dry chemicals.

At about 6 or 6:30 the decision was made to put water on fire. I went with one man from Athens FD to put water on fire. He got water on one spot but couldn't get water on largest fire. A different kind of nozzle was supplied from by the Athens FD but it wouldn't work.

The ventilation system was off a good part of the time and between 4:30 and 5 pm there was some smoke in the shutdown board room.

I went home about 9 pm.

I participated in one fire drill about a year ago and had fire training eight years ago.

The max number of people in the control room any one time I guess to be about 50-75.

At my first entry into the reactor building, I was with ASE-L, then SE-U showed up and took charge. The decision to use water seemed to come from the control room. When we used the hose the first time, the hose was already layed out. The spray was more like a shower head rather than fog.

/s/ C. E. Murphy — Witness   /s/ ASE-Y — Signature   22 April 75 — Date

I, AUO-Z, make the following free and foluntary statement to M. V. Annast, who has identified himself to me as a representative of the U. S. Nuclear Regulatory Commission:

I came in on regular shift at 3:00 p.m. on 3/22. I was stopped by construction workers at the gate and told of the fire (They were there to turn construction workers back as they arrived). I reported to the control room and found out what was happening. I had been working in the Unit 1 turbine building (on a 10 week assignment), but on 3/22 I was assigned by ASE-RRRR to do other duties. I carried empty air pack tanks and extinguishers from the control bay to the service bay for refill. I did this for about one hour. I brought the refills back to the control bay. At about 4:00 p.m. they were going to send a man into the spreader room to use an extinguisher to discharge it on the fire on the reactor side through the hole in the wall from the spreader room side. O-AAA was going to do this. I was supposed to stand by the Cardox push button at the Unit 2 side door to prevent someone from pushing it. I had been told that it had been set off before. The glass in the door of the trigger button box was missing. There was no sign of breakage. There was no sign of a metal plate or cover around--or bent metal. I could not see O-AAA fighting the fire. It took about 30 minutes and I saw O-AAA come out out of the spreader room.

At about 4:30 p.m. the operator wanted the RHR service water valve open, so he sent AUO-BC and I into the Unit 1 reactor building to try to open one. We had air packs on and got one valve partially open, had to come out--out of air--we took a 20 minute break because AUO-BC had smoke in his mask. We got fresh tanks and made another entry. After three entries we had not gotten the valve all the way open. We returned to the control room and learned that they had gotten the power back to the valve. There was no need to go back. I then went down to the shutdown board room where they were making entry to the reactor (same level as the fire). I was told to bring up spare cannisters and Chem-ox units. I brought them back and stocked them in the shutdown board room. Several persons were going to fight the fire and put water on it. ASE-L told me to stand at the entrance and take the names of people entering--he did not relieve anyone else. SE-V, AUO-AA, and a couple of Athens firemen and SE-U were all fighting the fire. The had fire hose and were using water. They were there about an hour and all came out about 7:30 p.m. when the fire was put out. At about 8:30 AUO-AA and I started an hourly patrol of the reactor building, reporting to ASE-PPPP. I was through about 11:30 and went home.

I have read the statement summarized above, which is true and correct.

/s/ Michael V. Annast	/s/ AUO-Z	4-23-75
WITNESS	SIGNATURE	DATE

I, AUO-AA, voluntarily give the following information to Howard Wilber, Tolbert Young and James Devlin, who have identified themselves to me as representatives of the U. S. Nuclear Regulatory Commission. I am employed by the TVA as an Assistant Unit Operator. I have been employed by the TVA for four years and assigned to Browns Ferry for the past 3½ years.

On March 22, 1975 I was assigned to the third shift, Unit III. Upon arriving I was told to go to Unit I for assignment. ASE-OO told me to go to Unit 2 condensate booster pump and check the auxiliary oil pumps to determine if it was running and if not, get it running. The condensate booster pump is located below elevation 565 turbine building on a sub-level. I called back to Unit 2 and told the UO that the pumps were OK. This was about 3:10 PM. Unit II had already scrammed. I returned to the Control Room.

ASE-L ASE, told me to go to the air lock at elevation 565 and bring back Scott Air Paks. I met another AUO who was also gathering air paks. I helped him for a while and returned to the control room.

O-MM, U.O., asked me to help him line up the RHR drain pump to pump from torus to hot well on Unit 1. We entered Unit I through Unit 2 on elevation 519, we could not see anything because of smoke. My air ran out and I had to leave. Shortly, O-MM came out and in a few minutes we went back in. This time we got the system lined up. O-MM manually started the pumps.

We then went to the torus to open 74-62 discharge valve. This took quite a while because we had to come out for recharged air tanks three times. This job took until approximately 6:30 PM.

ASE-L told me to bring air pacs to the 1-C air lock. SE-V, SE, SE-U, SE, and me went into the Rx, the fire was still burning. SE-V went in and climbed the ladder with the hose and he went down the trays again. I left at about 7:15 - 7:30 PM after the fire was considered out.

I was then assigned to the Rx Building fire watch. We checked each floor once an hour and made a report to the ASE after each round.

I left the plant at about 11:30 PM and returned for my regular shift at 7:30 AM Sunday.

At the times I was in the control room their were from 12 to 15 people there.

/s/ Michael V. Annast	/s/ AUO-AA	4/23/75
WITNESS	SIGNATURE	DATE

I, AUO-BB, make the following free and voluntary statement to Charles E. Murphy who has identified himself to me as a representative of the U. S. Nuclear Regulatory Commission:

I am an Assistant Unit Operator at the Browns Ferry Plant. On March 22, 1975, I was assigned to the turbine building. When I heard the fire alarm. I called the control room and was told that the fire was in the reactor building. I grabbed a fire extinguisher and went on to the elevator to the second floor of the reactor building. Some people were leaving when I got there. I then went to the fire area. A ladder was in place and people were fighting the fire in the trays at the penetration. I then went to the third floor of the reactor building with a DPP painter (His name is DPPP-CK.) I did not know his last name then. I heard the reactor scram and saw that it was 12:55 p.m. by my watch. There are about four pipe penetrations in the floor above the fire area. We discharged about six bottles down onto the fire. Smoke finally ran us out and I returned to the control room. The Unit Operator sent me to open up the minimum valves on the feedwater pumps. I then went back to the control room and next went back to the third floor of the reactor building to retrieve two good extinguishers. Then, I went to the spreading room to confirm that the CO_2 system was still operating and the room was still full of CO_2 and/or status of room. The CO_2 had already been operated. I returned to the control room bay area. Then I tried to go to the second floor of the reactor building by going down the stairs from the third floor, AUO-ZZZ was with me and we had a life line. AUO-ZZZ's bell went off signalling his Scott Air Pack was low. My diaphram on my air pack came off about then. (It was probably knocked off by my air line.) I went to the nurses station for smoke inhalation and was there about thirty minutes. It was then about 5:00 p.m. When I left the nurses station I went to get a rain suit and entered the reactor building through the second floor. I helped fight the fire with water for about fifteen minutes. Then I got an OBS and returned to the fire and fought it for about 30 minutes. I was with SE-U both times that I was fighting with water. The Athens Fire Department people asssisted us. I used their lamp and one aided me in putting on a breathing apparatus during the period of the fire.

/s/ C. E. Murphy	/s/ AUO-BB	4/29/75
WITNESS	SIGNATURE	DATE

I, PSO-CC make the following free and voluntary statement to C. E. Murphy who has identified himself to me as a representative of the U. S. Nuclear Regulatory Commission:

On the March 22, 1975, at about 12:45 p.m., Officer PSO-K called PSO-CP at headquarters about a fire reported in Unit No. 1. PSO-CP notified me on the project at the fire hall. PSO-CR called me also and said that maybe I should come down and check it out. I was the Public Safety Service Supervisor on duty. PSO-CQ was called and reported to Browns Ferry shortly thereafter. We were having a fire drill with 14 Public Safety Officers at the time. I immediately went to the plant in a Public Safety Service car and had the Public Safety Service fire engine follow. I went to the Units 1 and 2 control room to check with the shift engineer. He told me that the fire was in the spreader room and that they had released the CO_2 system. I asked what I and Public Safety Service could do. He did not respond to my request. The control room was filling with thick smoke and fumes. The shift engineer and others were choking and coughing on the smoke. It was obvious the control room would have to be evacuated in a very short time unless ventilation was provided. I contacted ACS-CS, Assistant Construction Superintendent, to take care of this matter. I checked back with the shift engineer to see if the power could be cut off in the fire area. He did not know. I then went to the spreader room.

The CO_2 in the spreader room may have slowed down the fire but did not put it out. After the CO_2 cleared up, we opened the doors for air, as the smoke in the whole area had become dense and sickening. A DPP employee and I each donned a breathing apparatus and went into the spreader room to locate the fire. The spreader room is filled with trays of energized cables to within 30 or so inches of the floor. We used hand lamps for illumination, but they penetrated the smoke only a few inches. We crawled under the trays searching for the fire, and finally found it in cable trays in the wall between the spreader room and Unit No. 1 reactor. The neoprene covers on the cables were burning giving off dense black smoke and sickening fumes. Air was forcing much of the fire outward through the portal in the wall toward the No. 1 reactor room. There was a lot of blue arcing and popping of electricity, and the fire was slowly backing down the cable tray into the spreader room. We came out, and I went to the control room and asked the shift engineer and another DPP employee there if they knew the exact point at which the cable trays came out into the Unit No. 1 reactor from the spreader room. Neither of them knew.

I gave Public Safety Officers instructions to begin bringing CO_2 and dry chemical extinguishers to the fire areas and to back up DPP employees and fire brigade members in fire fighting. I went back into the spreader room wearing a Scott Air Pak and mask and carrying a fire estinugisher. I had to crawl under the trays. The air pak cylinder was too cumbersome to wear on my back so I took it off and slid it and the fire extinguisher under the trays about 30 feet to the fire which was continuing to inch back into the spreader room. An unidentified employee, DPP, I believe, went in with me. I do not know how many trips I made into the spreader room, but there were several. We used a number of extinguishers and masks. I swapped the Scott for an MSA with a front mounted cannister because it was easier to crawl with.

After a while, the fire area in the No. 1 reactor room was located in the trays along the ceiling. While I was taking a breathing spell from the spreader room I helped get Construction to put emergency lighting and ventilation, and a life line in the reactor room area.

It was impossible to not swallow some smoke. I got sick several times. On one trip into the spreader room, I was fighting the fire at the portal to the reactor room when the building trembled and concrete chunks or something fell from the ceiling. An air pak cylinder fell off a cable tray, struck my head, and knocked my mask loose. I got the mask back on, however, for a time afterwards I think I was addled. I do not recall the exact passage of time, but I worked between the spreader room and the reactor room. I sent Officers PSO-CT and PSO-CU into the spreader room to continue fire fighting efforts as needed.

I went to the control room. There I found a DPP employee talking with P-Supt. about setting up a fire brigade. I suggested to P-Supt. and the other employeee since we now knew where the fire was located in Unit No. 1 reactor room, that we use a 125-pound Purple K dry chemical extinguisher on the cable trays to put a coating of dry power as insulation on the cables and then bank water fog off the ceiling onto the trays. I then left the area for a few minutes because I was sick from smoke inhalation. When I returned, the fire brigade had used the 125-pound dry chemical and also some water fot to cool down the cable trays.

I went back into the Unit No. 1 reactor fire area with Shift Engineer XE-V and another Power employee. We finished putting the fire out with water fog. I talked with Shift Engineer SE-U about getting a tall ladder so that we could reach the ceiling trays and have someone finish cooling them down.

I left the fire area of the reactor room and returned to the control room. I talked with General Construction Superintendent GCS-CX and Power Plant Superintendent P-Supt. to tell them the fire was out. Then I left the DPP area and returned to Public Safety Headquarters at about 8:30 p.m.

During the fire I conferred several times with PSO-CQ and P-Supt. I tried to assist in any way I could. I made a suggestion or two to P-Supt. regarding fire suppression methods which he accepted. I was not given any instructions by anyone in Power.

To my personal knowledge, several officers were involved and all officers carried out my instructions. I particularly recall PSO-CT and PSO-CU working in the cable spreader woom. The did a good, if not heroic, job.

/s/ C. E. Murphy	/s/ PSO-CC	5-5-75
WITNESS	SIGNATURE	DATE

I, AES-JJ, make the following free and voluntary statement to Michael V. Annast, who has identified himself to me as a representative of the U. S. Nuclear Regulatory Commission:

I am an Assistant Electrical Superintendent at the Browns Ferry plant. At approximately 12:40 p.m., March 22, 1975, I was laying out a job to two of my foremen in our print room when we got the report of a fire in #1 reactor building. I immediately went to 565' elevation.

There were several used CO_2 extinguishers around and DPP personnel were trying to get into the area and were putting on air masks. Prior to about 1:00 p.m., I met Elec.-D and EA-C coming from the 606' elevation to the Unit #2 control bay and told me they were checking the containment sealing in that area with a candle when the fire started in a cable tray opening from 606' to 593' elevation. The CO_2 system had been set off the cable spreading room and we could nto get in that area. I met my foreman, FM-CJ, and helped him to locate and account for his men. It was still impossible to get into the reactor building because of the smoke so I started helping account for all of our people who had been in the reactor area.

/s/ C. E. Murphy	/s/ AES-JJ	5/6/75
WITNESS	SIGNATURE	DATE

I do not know of any requirement to make written reports of small fires.

I was not aware that the poly foam (polyurethane) was combustible. I understood that tests had been run on the polyurethane and it was determined that it was a fire retardant. This influenced my thinking regarding my level of concern.

I also have significant concern regarding the lapse of time between the start of the fire, the time the construction people spent fighting it and the delay in reporting it to the Shift Engineer.

/s/ AS-GG 4/12/75

/s/ Witnessed: William S. Little

I, EE-FF, voluntarily give the following information to Howard Wilber, Tolbert Young and James Devlin, who have identified themselves to me as representatives of the U. S. Nuclear Regulatory Commission.

I am employed by the Tennessee Valley Authority and have been so employed for 19½ years and at Browns Ferry since August 1968. I am an Electrical Engineer, Group Leader, Conduit and Grounding, Division of Engineering and Construction.

I was not working on Saturday March 22, 1975 when the fire occurred. I was called in at 6:15 PM. I worked with EE-B and DPP analyzing damage and coordinating work. We worked until Sunday morning. EE-B left the Project about 11 p.m., 3/22 and returned about 11 a.m. Sunday 3/23.

The practice of using RTV-102 and sheet foam to seal air leaks has been the practice for two or three years. We believed that the urethane would not sustain a fire. Urethane samples had been tested several years ago and it needed a flame for 20 min. to sustain fire. I have a file note from Knoxville on urethane test.

On Friday afternoon March 21, EE-B and I tested the RTV-102 and found that it was self extinguishing.

I did not know about the fire that occurred on Thursday March 20, 1975 until after the fire on Saturday, March 22, 1975.

The sheet foam that we were using was not fire tested or given a safety evaluation. The use of sheet foam for this application was a "field fix" that I approved. It is purchased by DEC as an off the shelf item. I do not know of any specification covering this material.

I am not aware of any instructions issued to Browns Ferry DEC describing the bell fire alarm or fire evacuation procedures. DEC standard fire alarm is a constant pulse on the code call system. The BFNP radiation evacuation alarm is a siren. To my knowledge there has not been any DPP fire drills.

There is no written procedure that describes the air leak tests as performed by the Engineering Aids. The Engineering Aids should perform the test after the electrician has completely sealed the penetration.

/s/ EE-FF

Witnessed: /s/ William S. Little 4/16/75

I, AS-GG, voluntarily give the following information to H. Wilber, T. Young, D. Caphton and J. Devlin, who have identified themselves to me as representatives of the U. S. Nuclear Regulatory Commission. I am employed by the Tennessee Valley Authority as Ass't. Superintendent of the Browns Ferry Nuclear Plant and have a SRO license.

I learned of the fire and resultant events from ASF-QQ at 1:45 a.m. Sunday when he called me. I arrived at the plant at 3:00 a.m. Sunday morning.

I went to the control room to check the status of the plant. I found that both units had zero pressure and both had cooling water. I spoke with the Assistant Operations Supervisors, read all the key logs and surveyed the fire areas. There was no smoke but a strong smell of smoke was prevalent.

I was most concerned with the capability and continued reliability of the ECCS, the three operating and one down diesel generators and any possible health hazards. I wanted assurance that there was no question of the reliability of the station diesel generators to supply power as a dependable backup to off site power. The decision was that the primary recovery effort early Sunday morning should be made toward making the diesels a reliable source of emergency power.

At this time analysis showed about 20 Mwt decay heat and the one RHR pump and one RHR service water pump could carry away the heat.

I later reviewed technical specifications since changes may be required.

On Sunday or Monday directed that the in-out power to control rods be disconnected, disarmed drivers.

I had no direct involvement on Sunday with lifting leads and installing jumpers, the Ass't. Operating Supervisor was providing overall operational direction early Sunday morning. I directed that SE be assigned to the JLWR log.

I was aware of the fire that occurred on March 20, 1975 on elevation 565 but not in the proper perspective. I read it in the log after the fire of March 22, 1975 and it was mentioned on Friday morning by OS-DD in my presence. It was light and casual it was considered to be of no consequences. The log stated that it was put out with one extinguisher. I understood that construction ignited insulation and mention was made of the sealer but not specifically described as involved with the Work Plan, sealing air leaks.

I was not aware of the procedure employed to check air leaks, i.e. using lighted candles. There have been greater than 2892 work plans issued, can't know details of each.

I, ASE-EE, make the following statement freely and voluntarily to Michael V. Annast, who has identified himself to me as a representative of the U. S. Nuclear Regulatory Commission:

I am an assistant shift engineer assigned to Unit 1 at the Browns Ferry plant. I came to work about 3:00 p.m. on March 22, 1975. As I passed by the lunchroom, I saw AUO-O lying on one of the tables People informed me that a fire was in progress in the cable spreader room. I went to the control room where ASE-G briefly told me what had happened. AOS-W was in charge and was sending people to try to open the shutdown cooling valves manually. I went to MOV board 1A, which was dead, after SE-PP came in. I got the board reenergized about 4:00 p.m. and operated some valves. At about 4:30 p.m. I assisted in getting the turbine back on turning gear and thereafter restored RPS MG set 1A. Shortly after 6:00 AOS-W told me that the relief valves had quit. The reactor pressure was rising. The phones were out, so I worked on them and got them working about 7:00 p.m. After the fire was put out at around 7:30, we were still sweating out the relief valves. I was asked to go and secure the reactor building doors where I posted a man. "D" diesel could not be operated from the control room and I was asked to go and get it going. I got the diesel started and loaded at about 10:30 p.m. from "D" 4 KV shutdown board.

I was working Thursday, March 20, 1975, when a fire report was received from the cable spreading room. I had not been aware that workmen were using candles to check for leaks until this time. SE-R told me that a construction electrician's foreman had called and reported a fire in the sealing around the cable penetrations and that they had extinguished it. I should get a fire extinguisher they used and get it refilled. When I got there, the construction people had left and there was no fire or indications of damage. I reported this to the shift engineer.

/s/ Michael V. Annast	/s/ ASE-EE	4/23/75
WITNESS	SIGNATURE	DATE

I, OS-DD, make the following free and voluntary statement to C. E. Murphy, who has identified himself to me as a representative of the U. S. Nuclear Regulatory Commission:

I am the Operations Supervisor. On March 22, 1975, I was at home when I was telephoned about 1250 at home and told by the shift engineer a fire was in Unit 1. P-Supt. also called me at home. I arrived at the plant at 1:20 p.m., and went to the shift engineer's office and asked him to call AOS-W and others. The Athens Fire Department had been notified. Both units scrammed at that time. All core spray and safeguard equipment was wiped out. The HPCI and RCIC were both out. The MSIV was closed. There were no feed pumps available. The water level was high (above normal). The safety relief valves were being opened because of pressure.

We discussed the level situation and the operator was ordered to blow down the reactor pressure to allow water introduction by the condensate booster pumps.

There was no information on core reactivity.

P-Supt. and the Athens fire chief, and I went into the cable spreader room, and looked at the hole where the fire started, to establish whether the fire was out. The fire had moved back in the cable tray. We discussed the use of water.

The March 20 fire was discussed in the 8:30 meeting on the 21st. I was told that it had been started by a propane torch.

I have read the statement summarized above, which is true and correct to the best of my recollection. I sign this statement only after making known in this document the fact that I was not informed, prior to this interview, that I would be expected to endorse this statement as a legal document.

/s/ C. E. Murphy	/s/ OS-DD	5/7/75
WITNESS	SIGNATURE	DATE

4/3/75

I, O-KK, voluntarily give the following information to H. A. Wilber, Tolbert Young, and J. Devlin, who have identified themsleves to me as representative of the U. S. Nuclear Regulatory Commission.

I am employed by the Tennessee Valley Authority as a Unit Operator. I have been employed by TVA for seven years and assigned to Browns Ferry for four years. I hold an RO license issued in January 1974.

On Saturday, March 22, 1975, I was the duty operator on Unit II when the fire occurred.

At approximately 12:35 p.m., I heard the fire alarm, Unit I and the paging, announcing a fire in the Unit I Rx building.

At approximately 12:50 p.m., two annunications showed up on 9-7 panel indicating 1. steam jet gas ejector - off gas filter DELTA, p high and 2. off-gas dilution air flow low. I felt that these annunciators had to be erroneous. Unit I was also getting annunciations that were strange and unexplainable.

At 1300 I saw that on both "A" and "B" channels, there appeared to be an annunciator DC power failure, the megawatt meters dropped and I lost Rx protection power in at least 1/2 Rx protection systems (scrams). I went over to scram the Rx but I believe it scrammed automatically. I put the mode switch to "shutdown" and pushed both scram buttons.

A few seconds later I checked all rods. All were in, all indications were green. All of these things occurred almost simultaneously.

At about 1301 the turbine was tropped by ASE-G. I was at 9-5 panel and saw that the Rx H_2O level dropped to approximately zero level on GE/MAC, which is in the lower mid range, appeared normal before and after scram.

At 1303 water level increased to 40". I tripped all three Rx feedwater pumps. About 4 to 5 minutes later the main steam isolation valves isolated (closed.) The steam pressure was about 930 psi. RCIC manually started to inject H_2O to the Rx, HPCI also started manually in the recirc mode.

I started blowdown through relief valves. I initiated four non-ADS valves in sequence for next 45 minutes to one hour, all valves manually lifted.

Sometime between 1320 - 1400, lost ability to lift the relief valves manually. The Rx pressure began to rise and some of the relief valve bumped open at 1020 psi. The instrumentation indicated no water level rise or pressure drop. One relief valve appeared to have stuck open. The Rx didn't depressurize over the cool down rate.

At 1415 I thought we had relief valve power back. The water level appeared to change when valves actuated. ASE-L and AUO-QQQ, AUO, verified water level from the 24-32 panel. These appeared accurate. Lost all Rx water level indications except L1-362. At this time power was lost to several charts and indicators.

At 1430 torus level indication went down scale. Several times when running HPCI and RCIC lost power to controller for manual mode, left it in automatic mode. Also at approximately this time I lined up RHR "D" shutdown cooling mode to cool torus. I also lost A & C, RHR pumps and A&C core spray pumps and 1B and 2B RHR service water pumps.

No record of these events are in the computer because the computer was out of service for re-programming.

I went ahead depressurizing with the stuck open relief valves while making up water with the RCIC system.

I left the board at 2000.

Some time prior to 3:00 p.m. smoke and CO_2 was blown into the control room. It was not bad at Unit II board, worse at Unit I.

/s/ Howard A. Wilber	/s/ O-KK	May 2, 1975
WITNESS	SIGNATURE	DATE

I, O-LL, voluntarily give the following information to H. A. Wilber, T. Young and J. Devlin, who have identified themselves to me as representatives of the U. S. Nuclear Regulatory Commission. I am employed by the Tennessee Valley Authority and have been employed since 1951. I am a Unit Operator but I cannot obtain a RO license because of health reasons. I have been working at Browns Ferry since January 1971.

On Saturday, March 22, 1975, I was working on Unit 1, O-M was the RO licensed Unit Operator. I was assigned to logs and keys.

I was originally assigned to Unit 3 but I was transferred to Unit 1 for this shift because the work scheduled at Unit 3, was too strenuous, core spray pump checks which require manually aligning valves.

I first knew of the fire when I heard the fire bell at about 12:30 pm. I was in the control room. I heard the Unit Operator, O-M, try on two or three times to get the information regarding the exact location of the fire. All he could learn was, it was behind the reactor but did not know what floor.

As soon as everybody realized it was a fire alarm the phone began to ring. Other DPP personnel at other stations were calling the control room, according to procedures. I answered the phone, O-M used the paging system. He was announcing, "fire in the Unit 1 reactor building unknown location."

I was in the control room for two or three minutes. I asked O-M if I was needed or should I go to find the fire. He told me to stay by the Shift Engineer's office and pick-up a "cutie pie" and go to the fire. While in the Shift Engineer's office the SE was talking to someone by phone and was told that the fire was on the second floor in the reactor building. I took the "cutie pie" and went to the second floor by stairs.

There was no smoke except at the fire. When I got back of the reactor I saw the flames in the cable tray about 20' above the floor. Someone was on a ladder or scaffold using a fire extinguisher. Several people were standing around.

The man on the ladder came down and we all backed out of the area because the smoke began to get very heavy. I heard equipment running behind me and from the sounds I realized or believed that the Unit 1 had tripped. I met SE-R , SE and told him I thought the Unit 1 had tripped. He had come into the Rx building through the A Kv shutdown and battery board rooms. These doors are kept locked but as the SE on duty he had the control keys. He told some of the people that were there to get Scott Air Paks.

Also there, at this time, were ASE -L and ASE-G. I called O-M by phone and asked if he needed help. He said he did and I returned to the control room. When I arrived, O-MM, RO licensed Unit 3, Unit Operator, and ASE-OO, Unit 3 unlicensed SRO ASE were in the Unit 1 control room assisting. O-M was on Panel 9-5 and the others were moving between the other panels. In a few minutes ASE-G, ASE returned to the control room. About five people were in the control room at the Unit 1 side. On the Unit 2 side was only the Unit Operator. SE-R was at the scene of the fire.

I answered the phone, it was ASE-L. I gave the phone to ASE-G. ASE-L told ASE-G to flood the spreader cable room with CO_2. ASE-G hit the siren button and went to the spreader room to manually set off the Cardox System. He told me, as he left, to announce over the paging system, "to clear the cable spreader room." I announced this several times and went to the cable spreader room. The door was closed and the room flooded with CO_2.

I returned to the control room and noticed that Unit 2 had scrammed. The scram lights on the rod drive display panel were blue.

I answered the phone and the chem lab wanted to give a report of the water chemistry. I told them to disregard since Unit 1 had scrammed and we had a fire. I do not believe that the chem lab people on duty knew that the reactors had scrammed or that there was a serious fire. (This may have occurred before going to the spreader room.)

The personnel in the control room were about evenly divided between Units 1 and 2.

I spent the remaining time in the control room doing what I could. I noticed H_2O level in Rx low on recorder paper strip, 0-60 scale, but the Yarways showed OK. RCIC was on so someone said the water will come back. I do not know what condition existed with HPCI.

Panel lights were changing color, going on and off. I noticed the annunciators on the Diesel Generator Control Circuit. It showed Diesel Generator "C" start failure alarm and on all four showed control circuit ground alarm. I notified the SE of this condition and said I didn't think they would start.

I was sent to bring back flashlights, the AC power failed and the DC lights were on. When I returned there was smoke in the control room. The Public Safety Officers were in the control room wearing Scott Air Pacs. They were told to give the breathing equipment to the Unit Operators. The Athens fire department people were in the hall outside the control room.

About this time the next shift took over. It was about 3:00 PM. There were about eight people in the control room and a larger crowd outside of the gate. Smoke was coming through conduit in the floor.

I believe the regular ventilation system was shut down but the control room ventilation system was running. This was about 3:20 PM.

I was sent to the Turbine Bldg., number one turbine had stopped. I attempted to start the turning gear but the motor was out. The oil pumps were on but the lift pumps were not running. I returned to the SE office to get additional help. Another AUO was sent with me. We started the lift pumps but could not manually turn the shaft. I talked to the General Electric representative and then remembered the procedure. Number 2 turbine ran longer than Unit 1 but was stopped by this time. We called the control room and No. 2 was put on the turning gear from the control room. After assistance from an electrician No. 1 turbine was put on turning gear.

I do not know exactly what time the fire was extinguished. I left at about 8:40 PM.

Witnessed: /s/ C. E. Murphy   /s/ O-LL Signed

April 14, 1975 Date

I, O-MM make the following free and voluntary statement to Michael V. Annast, who has identified himself to me as a representative of the U. S. Nuclear Regulatory Commission.

I was the Unit #3 operator when the fire alarm went off. I called Unit #1 operator who told me that they needed CO_2 breathing apparatus. I sent two AUO's with the equipment and went to Unit #1 control room. At this time Unit #1 scrammed and I assisted O-M with RCIC shutdown at his request. I placed RCIC controls on manual. "C" and "D" diesels were running tied to their respective boards. "A" and "B" diesels were idling, ready to tie in. At this time, about 12:45, Unit #2 scrammed. I went to assist Unit #2 operator with shutdown. Pressure on the vessel rose to max of 1100 psi. The relief valves were popping. Outboard MSIV's closed. Lost the ECCS. All relief valves, except four, were lost. I tripped the "B" RHR pump which I had started earlier in the torus cooling mode at about 12:55.

ASE-G, the ASE, tried to get the board room to restore power to the valves that isolated but it was too smokey. OS-DD arrived at the control room at about 1:10 p.m. He and O-M decided to blow down Unit #1. O-M watched the water level and I operated the four operable relief valves at his direction. When the pressure reached about 350 psi, the condensate booster pump was started to restore reactor water level. I believe the water level never went below -100 inches on safeguard Yarway. During this time, the CO_2 from the spreader room caused fumes in the control room. O-M and I put on air packs for about 5 minutes. An air hose was brought in from Unit #3 to clear the air. The turbine was off turning gear and I sent some people to get it back on, but there was no power. I went out and checked at 3:10 p.m. and found both turbines off turning gear. ASE-G and AUO-Q opened the test valve for torus cooling mode on top of the torus. AUO-AA and I then got valves lined up and finally got the "B" RHR pump running (pumping to the hotwell). We made four trips to open valves on top of the torus. This took some time because we kept running out of air. The torus temperature was not uncomfortably hot to the touch but the air supply only lasted 12-15 minutes.

The shift engineer told me to go home at 8:30 p.m.

/s/ C. E. Murphy	/s/ O-MM	4-29-75
WITNESS	SIGNATURE	DATE

I,O-NN, voluntarily give the following information to Floyd Cantrell, Tolbert Young and James Devlin, who have identified themselves to me as representatives of the U. S. Nuclear Regulatory Commission. I am a Unit Operator and have been employed by the Division of Power Production, TVA since May 1970.

I was called at 1:15 on 3/22/75 and arrived on site at 1:35. I went to the Unit I control room and found the Unit I operator with a Scott Air Pak on. O-M told me to open the breakers on a string of feedwater heaters. I finished this about 2:15 and went back to the control room.

The turbine turning gear motor was out and I was sent to try and turn the turbine shaft manually. We could not and about 4 pm power was restored and we got both Units I and II turbines on their turning gears.

I was then sent to manually open and close the feedwater pump bypass valve on Unit I until about 7 pm.

SE-PP then sent me into the reactor building to line up some valves on the standby gas treatment system. I was not familiar with this system so I went along with O-AAA and ASE-QQ but we could not find the valves for some time.

I have had some fire training but never participated in a fire drill.

The avg. number of people in the control room at any one time was about 20.

I went home about 9:30. Came back on Sunday and was assigned to keeping logs on Unit II.

There was quite a bit of confusion but didn't notice anyone overly excited.

/s/ C. E. Murphy	/s/ O-NN	4/22/75
Witness	Signature	Date

I, ASE-OO, voluntarily give the following information to Tolbert Young and James Devlin, who have identified themselves to me as representatives of the Nuclear Regulatory Commission. I am an Assistant Shift Engineer and have been employed in the Division of Power Production, TVA since 1971 at Browns Ferry Nuclear Plant.

I was on duty in Unit III at the time of the fire. I heard the fire alarm about 1 PM and went to Unit II control room. I started assisting the Unit II operator and stayed there for 15 or 20 minutes. We had a group 1 isolation and could not reset it. I then went down to check RPS generator. The generator was running so I got the instrument department to work on the problem. Some fuses were blown.

About 2 PM we lost Unit II 120V preferred power. I found that UI 120V unit preferred was tied to U-II M/G set and the U-II M/G set had a loss of power. I then tied U-II to its transformer supply, at that time U-I was without 120V unit preferred. During this time we lost Shutdown 4KV Board No. C. The diesels were running but found a fault on TS-1B transformer. I tied MOV 2A on to its alternate supply and got some electricians to work on the fault so we could get the shutdown board back.

About 3 PM started having problems with the torus level. I started the 2-B RHR drain pump and pumped the excess water to the U-2 hotwell.

Sometime after 4 PM I did some work to get air supply back to the U-2 relief valves. I got some electricians on this problem.

About 6 PM the electricians had removed the fault from shutdown bus C. I got the board on to its alternate power supply. I was relieved by ASE-VV sometime between 6 and 7 PM. I made a tape recording of my participation about 8 PM and went home about 10:15.

Sunday I worked on Unit I with crew making out J1WR log and tags.

The max no. of people I saw in the control room at any one time was about 15.

The smoke was very bad in the control room around 3 PM. At one point we discussed going to Unit II backup control board.

I never participated in a fire drill at Browns Ferry.

/s/ C. E. Murphy	/s/ ASE-OO	4/15/75
WITNESS	SIGNATURE	DATE

I, SE-PP, make the following free and voluntary statement to Michael V. Annast, who has identified himself to me as a representative of the U.S. Nuclear Regulatory Commission:

I am a shift engineer at the Browns Ferry Plant. I was called at home about the fire and arrived at the plant at 2:00 p.m. I reported to the control room where I noticed the DC lights on and the ECCS 9-3 panel out. AOS-W told me to take over responsibility for Unit #1 which had been blown down. Normal reactor water level was reestablished, reactor pressure was about 200 psi and water was being added by a condensate booster pump. Drywell temperature was indicating 375°F. Three drywell blowers were on with cooling water to the coolers. Dry well temperature dropped to indicated 90°F on panel 9-3. Drywell pressure indicated 2.2 psi. Drywell pressure was lost for a short time and when it came back on it indicated 2.7 psi.

About 5:00 I noticed reactor pressure increase from 150 to 200 psi. At that time we had 4 relief valves indicating open. I put on a Scott airpack and went in the reactor building to check the drywell air supply to cut the plant control air supply into the drywell relief valve air supply. I started the second drywell air compressor but the suction valve to both compressors was closed. Also, the discharge valve to the dry well air supply was closed.[2] With an electrician, we bypassed the isolation valve solenoid in the air supply to the drywell. This reestablished air to operate relief valves. During this period, reactor pressure reached 600 psi. Water level was normal. Pressure returned to 50 psi when relief valves opened.

I started working on torus cooling and RHR shutdown cooling to establish normal conditions. I sent men to the reactor building to manually verify the positions of valves in the cooling systems. These valves were not operable from the control room or the back-up control center. Had some problems in determining the exact of valves and their position due to heavy smoke. AOS-W directed me to flush the loop prior to establishing shutdown cooling. Torus cooling was established at about 9:30 p.m. Maximum torus temperature was reached at about 10:00 - 11:00 p.m. indicating +75°F. Shutdown cooling was established at about 11:00 p.m. I remained on the 11:00 p.m. to 7:00 a.m. shift and tried reestablish the reactor clean-up system.[1]

The main problems encountered were inadequate respiratory equipment for fire-fighting and the loss of air supply to operate the relief valves. The latter appears to be a design problem.

/s/	Michael V. Annast	/s/	SE-PP	4/15/75
	Witness		Signature	Date

1. There was also a concentrated effort to establish neutron monitoring which was accomplished early on the midnight shift at a local temporary monitor in the Reactor building.

2. This statement is misleading in that only the suction valves operate on isolation signal. The drywell supply valve closed on loss of power.

3. All times are very rough estimates (±2 hours accuracy) not being the duty shift engineer I was not keeping a log.

I, ASE-QQ, make the following free and voluntary statement to Michael V. Annast, who has identified himself to me as a representative of the U. S. Nuclear Regulatory Commission:

I am an assistant shift engineer at the Browns Ferry plant. I was called at home by ASE-RRRR at 2:35 p.m. and told that they had a fire. I arrived at the plant at 3:15 p.m. and went to the Control Room. SE-V told me to help ASE-L fight the fire. I had to wait for an air pack. All of the air packs were short of air. The RPS for Unit 1 was dead and the turbine could not be put back on turning gear. For a while I assisted with carrying fire extinguishers and air bottles. I suggested to AOS-W to get a fan going on top of the reactor building. As I got there at about 5:00 p.m. someone had already started one fan. At about 7:00 p.m. I was asked to go into the reactor building to try to open valves which would vent drywell into gas treatment system, to relieve drywell pressure. O-AAA and I put on air packs and located the valves. We used pipe wrenches and tied them off with wire to keep the valves open. I reported back to the control room at 8:25 p.m. I assisted with various tasks until 11:00 p.m.

I have not received fire training.

/s/ C. E. Murphy	/s/ ASE-QQ	4-22-75
WITNESS	SIGNATURE	DATE

I, AUO-RR, voluntarily give the following information to H. Wilber, T. Young, and J. Devlin, who have identified themselves to me as representatives of the U. S. Nuclear Regulatory Commission:

I am employed by the Tennessee Valley Authority as an Assistant Unit Operator. I have been employed by TVA for four years and I have been at at the Browns Ferry site for three years.

On Saturday I was assigned to work the second shift. I arrived at 2:40 pm. I went directly to Rad Waste. The second shift operator was not there and the filters were shut out. Rad Waste was my regular assignment.

I went to the control room and was told to put on a Scott Air Pac and go with AUO-CCCC to help open the suction valves to RHR cooling pumps located on the 519 elevation. We made three tries but could not get to the valves. Our breathing equipment could only supply 18 minutes of air per tank, which was not sufficient to enable us to get to the valves and back out of the area. The air tanks were being recharged but the pressure in the main tanks was not strong enough to fill the tanks to the normal 30 pounds air supply. After the third attempt we went back to the control room and told the ASE of the problem and that we needed different equipment or fully charged tanks to succeed.

ASE-L told us to disregard this assignment and go to the cable spreader room and relieve the men that were there. This was about 3:30 pm.

AUO-X and O-AAA were in the cable spreader room. They were worn out we took their place on top of the ventilator duct in the middle of the room adjacent to the burning cable trays. There was smoke around the ceiling but we did not need breathing equipment. We used CO_2 and dry chemical extinguishers on the cable tray.

Negative pressure was still being maintained; therefore the smoke was being pulled into the Rx building. At this time nobody was fighting be fire on the other side of the penetration. On the spreader room side the fire was considered to be out.

ASE-NNNN, ASE, came in and I went back to Rad Waste. O-AAA stayed and also a public safety man was helping us.

ASE-L sent me and O-ZZ, another AUO, with a couple of other employees to get Chemox breathing equipment and bring them to 1-C. This was sometime after 4:00 p.m. I did not go into the Rx building; the fire was still burning.

ASE-EE, ASE and my regular supervisor on the Rad Waste assignment, sent me to the 1-C motor operated valve (MOV) board to reset two breakers on the Rx H_2O clean up system (RWCU). The breakers didn't reset, one sparked so I left it alone. I reported back to ASE-EE and then went back to 1-C level of control building. SE-U was in charge at the entrance to the reactor through the control room building.

Sometime after 5:00 p.m. I helped SE-U another AUO, an SE and a member of the Athens Fire Department, all of who were wearing Chemox breathing apparatus. They were ready to spray the cable trays with water.

No Health Physicists were in this area.

After the change of shift at 11:00 pm I was assigned to fire watch in the Rx building. Every hour we checked the Rx building, each level. We were allowed to be inside for only 20 minutes out of each hour. I was relieved and went home at 6:45 am.

/s/ C. E. Murphy	/s/ AUO-RR	5/6/75
WITNESS	SIGNATURE	DATE

I, AUO-SS, voluntarily give the following information to Floyd Cantrell, Tolbert Young, and James Delvin, who have identified themselves to me as representatives of the U. S. Nuclear Regulatory Commission:

I am employed by the Division of Power Production, TVA as an Assistant Unit Operator and have been at the Browns Ferry plant since January 1971.

I was assigned Unit 1 of the fire I was at the stack running a routine instrument check. At about I received a report from the control room of the fire in the reactor building. I went back to the control room and the SE gave me some battey power lights and told me to take them to the shutdown panel room. After this UO instructed me to open the bypass valve around the full flow deminerilizers.

I was then sent to stand by the auxiliary boiler and I stayed there the rest of the shift.

I got off at 11 p.m. that night.

The maximum number of people at any one time in the control room was about 10 people.

I have participated in a couple fire drills since 1971.

/s/ Michael Annast　　　/s/ AUC-SS　　　4/25/75

I, AUO-TT, make the following free and voluntary statement to Howard A. Wilber, who has identified himself to me as a representative of the U. S. Nuclear Regulatory Commission:

I am an AUO. I work in the Unit 3 building. On March 22 I was in Unit 3-Cond Demin Area talking by phone to the Unit 3 operator at about 12:30 when I learned from him there was a fire on the second floor of the reactor building. I went back to the 3rd floor of Unit 3, picked up my film badge and dosimeter. I was asked to get Scott Air Packs to the people who were fighting the fire. When I got there the smoke was so bad everybody was backing out. About this time lights went out. I went back to the control room and later went back to the reactor building 3rd floor with AUO-BB. I was with ASE-L when the latter released the Cardox in the spreader room. I also made another Reactor building entry with ASE-L at which time we took in a safety line and tried to pinpoint the fire. I also made an entry on 1st floor of Rx bldg. to retrieve breathing apparatus in vicinity of CRD modules. This was taken to IC and hooked to service air.

I have read the statement summarized above, which is true and correct.

/s/ Howard A. Wilber	/s/ AUO-TT	4-22-75
WITNESS	SIGNATURE	DATE

I, EE-UU, voluntarily give the following information to Howard A. Wilber, Tolbert Young, and James Devlin, who have identified themselves to me as representatives of the U. S. Nuclear Regulatory Commission:

I am an Electrical Engineer and one of the Division of Engineering and Construction Coordinators. I have been employed with the TVA for three years.

When the fire occurred I was working in Unit 3. I was not involved with the fire therefore I do not have any first-hand knowledge of events. I was sent home at 2:30 p.m.

EES-A called me back to work Saturday at 6:00 p.m. I was working on cable lists of the cables in the damaged trays. Later ACS-DD gave us a list of equipment that DPP wanted back in service. Work Plan 2920 was issued.

I again returned to work on Sunday at 11:00 a.m. We started to plan and identify where temporary cables would be needed when we received an order from DPP to DEC to stop this task because it was to be handled by Design. We went back to Work Plan 2920.

/s/ C. E. Murphy	/s/ EE-UU	4/16/75
WITNESS	SIGNATURE	DATE

I, ASE-VV, make the following free and voluntary statement to C. E. Murphy, who has identified himself to me as a representative of the U. S. Nuclear Regulatory Commission:

I am an assistant shift engineer. I normally work in Unit 2. On 3/22 I did not know of the fire when I came to work on my regular shift at 3:00 p.m. As I came in I saw construction people at the gate and a public safety cruiser out in the roadway. I thought at first that there might be a strike or something. I saw in the switchyard that the MOD's for each unit was up in the air. As I came into the parking lot I saw the fire engine. I went directly to the control bay where I was told about the fire and was directed to check diesel shutdown boards in the control room. I saw there was no power on the A shutdown board from Shutdown Bux 2. I checked Unit 2 in the control room and it locked O.K. On Unit 1, they had lost power on the 480 volt MOV board B--I started working with electricians to clear this up.

At about 7:00 p.m., I went to Unit 2 for extended work. I needed the mechanical vacuum pumps on so I could get the bypass valves open. So I started working with the electricains again--this took a couple of hours. At about 9:00 to 9:30 they got the vacuum pumps' going. They had also had valve malfunctions on Unit 2. The RHR minimal flow valve was cycling on and off. The RHR exchangers (on the service water side) A and C loop outlet valves were also cycling (as determined by lights on the control board). I did not fight the fire. I stayed at Unit 2 until 11:00, end of shift, and went home. I came back in at 7:00 a.m. for a 16-hour shift. On Sunday I was on Unit 1 from 7 to 3 hanging tags on leads as they were disconnected by the electricians (according to JIWR procedure). I went back to Unit 2 at 3:00 p.m. for routine work on cold shutdown.

I have had fire fighting and refresher.

I knew of the 3/20 fire at the time it occurred. The 3/20 fire had already been put out before I knew about it. It occurred on my shift and was reported to ASE-EE who made an entry in the log and reported it to the shift engineer. As a result of this incident, ASE-EE had to go down and get the extinguisher which was used to refill it. I had heard about the use of a candle, but I did not know what they did with it. I did not know anything about the sealant.

I have read the statement above, which is true and correct.

/s/ C. E. Murphy	/s/ ASE-VV	4-22-75
WITNESS	SIGNATURE	DATE

I, ACCS-WW, voluntarily give the following information to H. Wilber, T. Young, and J. Delvin who have identified themselves to me as representatives of the U. S. Nuclear Regulatory Commission:

I am employed by the Tennessee Valley Authority as an Assistant General Construction Superintendent, Second Shift. I have been employed by TVA for about 20 years and I have been working at Browns Ferry since September 12, 1966.

I arrived on the site at 2:00 p.m. and learned of the fire. I went out to the gate and turned around second shift employes except a truck driver, four pump operators, one crane operator, and four pipe fitters. I reported back to GCS-CS and then met with DEC and DPP personnel to analyze damage and formulate a course of action.

I went into the fire area at about 5:30 p.m. The area was smokey and small particles of fire were falling. I stayed on a few minutes. I advised PSO-CC of this and showed him where the sparks came from. The fireman returned and wet down the area.

I returned a second time with PSO-CC and the trays were wet down again.

GEF-KKK , General Electrical Foreman, ran temporary lighting, after the fire was out Saturday night.

I know about the fire that occurred on Thursday, it was reported to the Shift Engineer and placed in his log. The fire was reported to me by GEF-KKK.

The fire and the reactor shutdown was controlled by the DPP supervisors located in the Control Room. Everything was orderly and under control. Everybody knew what they were doing.

/s/ C. E. Murphy	/s/ ACCS-WW	4/22/75
WITNESS	SIGNATURE	DATE

I, O-XX, make the following free and voluntary statement to C. E. Murphy, who has identified himself to me as a representative of the U. S. Nuclear Regulatory Commission:

I an a unit operator. At 3:00 p.m. I came out to the plant without knowing there was a fire. I did not realize until I got to the control room and saw people in air packs there that there was a fire. They had already organized fire fighters. I went to Unit 2 and got a lot of flashlights. I was told then that the fire was in the cable spreader room. I found out from the Unit 2 operator that fire was in the reactor building. In a few minutes the unit operator sent me to check the condensate booster pumps. The oil lines had air in them and a fitter was going to bleed them. When I got down there the ASE was there with the fitter. After a few minutes, I came back up by the condensate demineralizers to the control room. I worked in the control room for a while. SE-V sent me out to check on a breaker on the clean up system. The breaker had not tripped, it had not thermaled--the fuses were O.K. (visual). I reported back that the breaker was O.K. I was then sent to check the flush valve for loop 2 RHR. I went into the reactor building Unit 2 to the 3rd floor down to the Unit 1 second floor and reported this. Later they decided to use the 3rd floor valve. I checked on the auxiliary boiler system. I sent back down there and put the demineralizers in. I took a break at about 4:30 p.m. The AUO's were fighting the fire. I helped the assistant engineer to roll an extinguisher up to them on the Unit 1 side. I then went back to the control room and was sent back to the 3rd floor. I then went back to the control room to answer phones, reset alarms, etc. The smoke was getting bad. The flush water was on then on the Unit 2 valve. At about 6:00 I went back to the control room when some meals were brought in. The fire chief, and AOS-W were talking together in the control room. The fire was put out between 7 and 8 p.m. They sent me down to work with the lab and get a sample of water from the RHR system. Although the fire was out I was trying to find out it airpacks were needed. I waited for an hour to about 9:00 p.m. The lab wanted water samples from both units and a RHR water sample. We all put Scott air packs on. I waited down there while the lab people got the sample and returned. I went back to the HP and left the air pack. At about 10:30 I talked to ASE-RRRR and asked if I was needed or if I could go home. At 10:55 I left for home. I was sick on Sunday and did not come in.

I received fire fighting training for one week early '72 at Colbert steam plant. About 2 months ago, I observed smoke and fire at the Unit 1 condensate pump pit and turned in the alarm. The fire was in some rags in the pit set on fire by welder's sparks.

I have read the statement summarized above, which is true and correct.

/s/ C. E. Murphy	/s/ O-XX	4/22/75
WITNESS	SIGNATURE	DATE

I, ASE-YY, voluntarily give the following information to D. Caphton, T. Young, and J. Devlin, who have identified themselves to me as representatives of the U. S. Nuclear Regulatory Commission:

I am employed by the Tennessee Valley Authority as an Assistant Shift Engineer. I hold an SRO license issued in 1974. I have been employed by the TVA since 1966.

I was called into work on Saturday March 22, 1975, and arrived at 4:00 p.m.

I reported to the control room and AOS-W told me to assist SE-V who was on Unit 2. SE-V told me to check fuses on RHR pumps. The fuses for the three pumps I checked were good; 2-C pump had a ground on the normal control circuit, I suspect. I returned to the control room. I then went to 1-C elevation behind "B" shutdown board which was the controlled entry into the Rx Building. I helped to change breathing air bottles. The fire fighters had to frequently come out of the Rx. Building because the tanks could only be charged to 1200 psi which lasted for only about 10 minutes. The low pressure was due to low pressure in the tanks at the charging station.

At approximately 5:30 p.m. I helped to set an air line that permitted the men to stay longer in the Rx building. About this time, the ventilation system was started and a lot of smoke was removed.

SE-SSSS, SE was, or appeared to be, in charge of this operation. He told me to rig the life line. SE-U and SE-V were making entries to the Rx building fighting the fire. At about 7:00 p.m., I heard that the decision was made to use water on the fire. Shortly thereafter I returned to the control room. AOS-W told me to check out and try to open relief valves. I worked on this for some time checking the electrical control circuits. Some of the valves had no power due to the board being deenergized or the ACB for the control circuit being open. The valves that I checked were good electrically as far as I could tell because the indicating lights changed normally, both on the normal and alternate controls.

I got EF-BF, an electrician, with a meter and we checked several valves. The meter checks were good from the backup control center.

At approximately 8:30 p.m., the drywell air compressors were restored. They were needed for the relief valves. SE-PP got these back on and told AOS-W that the air pressure was O.K. SE-PP was working on the drywell control air isolation valve. When the air was restored to the drywell, two of the valves opened.

Between 8:30 - 9:00 p.m. the pressure in the vessel was brought down to approximately 400 psi. O-ZZ and O-KKKK were there during the blowdown.

The second blowdown brought the pressure down so that H_2O could be added now with feedwater system. SE-V and O-ZZ were in Unit 1 control room, GE-DA was also there.

AOS-W told me to write up a procedure for flushing RHR with equipment that was out. The procedure was checked by AUO SSS and AOS-W.

I heard of the fire that occurred on Thursday night when I came in on the first shift Friday. I met ASE-EE ASE with an empty fire extinguisher and he told me about the fire. He had looked at it and said thre was no damage, the only evidence of a fire was the dry chemical residue. He requested that the electrical department check it the next day. On this same evening, we discussed among the group, SE-R, ASE-EE, AUO-SS, and myself, the procedure of using lighted candles to check for air leaks. Our conclusion was that the procedure should be stopped. The log entry in the Shift Engineer's log was noted.

/s/ ASE-YY

4/16/75

Witnessed: /s/ William S. Little

9:00 am
4/9/75

I, O-ZZ, make the following free and voluntary statement to Michael V. Annast, who has identified himself to me as a representative of the U. S. Nuclear Regulatory Commission:

I am a licensed reactor operator assigned to Unit #1 at the Browns Ferry plant. I arrived at the plant about 2:30 p.m., March 22, 1975 to start work on the 3-11 shift. A PSS officer told me about the fire. I reported to O-M, the Unit #1 operator, who was watching the water level. He told me the fire was in the cable spreading room and on the second floor of the reactor building. I noticed that many of the lights were out on the 9-3 panel. The reactor pressure was about 200 psi. Four relief valves were working. My main objective was to get shutdown cooling going. I got a crew of AUOs (AUO-CCCC, AUO-RR, AUO-Z, BC and others) and gave them a list of valves, which were needed. Teams of two men each went out to the reactor building to try to get the valves aligned. The men reported they could not get any of the valves open due the short time the air packs had on them and the lack of visibility in trying to get to the valves. SE-PP, AOS-W and others were told about the air problem. I suggested a life-line since no ropes were available I told the men to use a small chain, which was rigged up. The men finally got a cooling valve FCV 23-40 open but then lack of air caused a suspension of work at about 4:30 p.m. I went and opened the doors between the reactor buildings for more ventilation and it helped somewhat. I was coughing badly at this time, due to smoke inhalation and got some cough syrup, which seemed to help. I assisted in hooking up an air line and breathing apparatus for entry into the Reactor Bldg. on the second floor from the diesel gen. bldg. The whole firefighting effort was hindered due to lack of air. The smoke was very dense from 3:00 p.m. to about 6:00 p.m. They got the exhaust fan turned on and density was down to about 50% at 6:30 p.m. They had started using water by this time to fight the fire. At about 7:00 p.m. one of the fires was out and the smoke density was about 10%. The visibility was reasonably good because the overhead cable trays could be seen with flashlights. At about 7:40 p.m. the visibility was normal and the fire was out with the exception of occasional wisps of smoke. I requested a break at about 8:30 p.m., took a shower and went back to the control room. I stayed until 7:00 a.m., assisting with the establishment of torus cooling and getting shutdown cooling lined up.

In my opinion, the priority was the fire in the cable spreading room. The main problems in the reactor building were visibility and lack of air.

/s/ C. E. Murphy	/s/ O-ZZ	4/17/75
Witness	Signature	Date

I, O-AAA, make the following free and voluntary statement to Michael V. Annast, who has identified himself to me as a representative of the U. S. Nuclear Regulatory Commission:

I am a unit operator who is not yet licensed. I came in at the regular time, 3:00. I had called in to the plant ahead of time to my buddy in Unit 3 to tell him I would be a few minutes late because I wanted to see the end of the Kentucky basketball game on TV. ASE-RRRR answered and said they had a fire and had lost both units.

After coming in as scheduled I saw O-MM (who I was supposed to reliev) on the 1st floor turbine building. O-MM said Unit 3 was shut down and all AUO's were to report to the control room. I did this and ASE-G asked me to go to the spreader room with him. We both had air packs and went into the spreader room. I could see the fire. All the smoke was boing toward Unit 2 and the visibility was good. The flames were about 8" high and extended, in the cable tray, to about one foot into the cable spreader room. They were mostly in the top 2 or 3 trays. All the insulation which had been stuffed in the opening had been burned away. We took two fire extinguishers apiece with us and discharged all four. Because of the air packs and restricted clearance we had to discharge at a distance of about 4 feet which was not too effective. We pulled a cart in with a CO_2 tank and used this to better advantage. I found I could do without the air pack and took it off. I then was able to lay it right on the fire. AUO-CCCC and AUO-RR also used extinguishers in a continuous succession and thus we got the fire out. However, you could tell it was still burning on the reactor side. We could heat the crackling of the fire on the other side (not electrical sparking). We left AUO-X and AUO-BB at about 5:00 p.m., to stay behind and watch to make sure the fire would not flare up again. We went back to the control room. P-Supt. was there. I am not sure who instructed ASE-QQ to get the exhaust fans on. ASE-QQ and I went up there and found that AUO-Q was already there--had already taken the dampers off and had got the fans going. This took up about 30 to 35 minutes. We went back to the control room again. ASE-QQ and I and O-NN were assigned to open the valves at the third floor (top of torus). PFM-BR took the actuator off and used the wrench to open the valve. We go it open about 8:00 or 8:30. I then finished the shift and went home.

I have had the fire fighting training and refresher.

I have read the statement summarized above, which is true and correct.

/s/ Michael V. Annast
WITNESS

/s/ O-AAA
SIGNATURE

4-23-75
DATE

I, AUO-BBB, voluntarily give the following information to H. Wilber, T. Young, and J. Devlin, who have identified themselves to me as representatives of the U. S. Nuclear Regulatory Commission:

I am employed by the Tennessee Valley Authority as an Assistant Unit Operator. I have been employed by TVA for 3 1/2 years and assigned to Browns Ferry for the past 2 years.

On Saturday I was assigned to the third shift. I arrived at the plant at about 3:00 p.m. My assignment was keys and logs, Unit 1. and I was supposed to relieve AUO-TT. Because I have a broken arm that is in a cast, I was not able to do too much.

I spent a few minutes in the control room. Later I was sent to the turbine building to put seal water into the mechanical vacuum pump. The rest of the day I gathered empty fire extinguishers and Scott Air Pacs.

Sometime about 5:00 - 5:30 p.m. AUO-X was sent with me to the cable spreader room as fire watch. We stayed there for the next six hours.

We saw that there might still be a fire among the cables because we saw wisps of smoke. We found a short wooden handle that was part of a mop. We used it to spread apart the cables. When we saw a spark or a glow we hit it with dry chemical. The sparks indicated that some of the cables were still energized. At one time the cables began to flame up. We put it out with the dry chemical. We had available a large 150 lb. or 125 lb. bottle of dry chemical.

I reported back to work on Sunday 3rd shift and was assigned to keys and logs.

/s/ Michael V. Annast	/./ AUO-BBB	4-23-75
WITNESS	SIGNATURE	DATE

I, EE-CCC, make the following free and voluntary statement to C. E. Murphy who has identified himself to me as a representative of the U. S. Nuclear Regulatory Commission:

I am an electrical engineer with TVA. On March 22, I was in Unit 3 at the 4 KV shutdown boards when I was notified of the fire. I was told to leave the building. I left the building via the Unit 3 diesel generators exit and reported to warehouse 14A. I left the site at 2:30 p.m. I was called at approximately 6:00 p.m. by EE-XXX and asked to come in at midnight. I did this and worked with EE-TTTT and EA-OOO. EE-XXX had been given a list of approximately 22 valves for which we were to find out how they could be hooked up and put back in service. We were to do this by means of lifting wires and adding jumper cables to bypass the burned out circuitry. We began by tagging each wire and terminal with its identity as we disconnected them. We were not able to complete the task (work plan) on that shift.

/s/ C. E. Murphy	/s/ EE-CCC	April 16, 1975
WITNESS	SIGNATURE	DATE

I, EF-DDD, make the following free and voluntary statement to C.E. Murphy who has identified himslef to me as a representative of the U. S. Nuclear Regulatory Commission:

I am an electrical foreman with TVA at Browns Ferry. I was in Unit 3 when the fire occured and was not involved in fighting the fire. On Saturday night I was called at home and asked to come in on Sunday morning to direct a crew in recabling some water clean-up valves (system 69). I directed this work as requested. I did not enter the fire area that day but did at a later time. I observed that the fire damage appeared to be confined to the cable trays on the Unit 1 reactor side of the spreader room about 8 to 10 feet from the ceiling.

/s/ C. E. Murphy	/s/ EF-DDD	4-16-75
WITNESS	SIGNATURE	DATE

I, PRS-EEE, make the following free and voluntary statement to W. S. Little, who has identified himself to me as a representative of the U. S. Nuclear Regulatory Commission:

I am the power plant results supervisor.

I was at home on Saturday, March 22, 1975, when about 1315 hours I received a phone call from the plant chemical engineer, CE-S, who informed me that a fire had occurred at the plant. I immediately called the shift engineer's office and ASE-RRRR relayed the plant superintendent's instructions to report to the plant. I arrived at the plant about 1345 hours and reported to P-Supt., who briefly described the incident and acquainted me with the current status. Initial observations indicated that all SRM monitors and most of the CAM units were inoperable, as well as the reactor building fans. I requested the plant instrument engineer, IE-JJJ, to get at least two out of the four SRM's operating as soon as possible. I called the shift chemical laboratory analyst and had him call the plant chemical engineer to have him come out to the plant to assist in evaluating plant effluents. According to CE-S, the reactor building fans were restarted during the collection of the initial grab sample. Analytical results reported by the chemical engineer indicated plant gaseous effluents were well below technical specification limits. I initiated and coordinated the delivery of SRM cable from TVA Construction to facilitate pulling temporary SRM wiring.

I did not fight the fire directly, since this appeared to be adequately covered by others. I noted that Athens Fire Department personnel were also present, assisting in the fire fighting effort.

I continued my surveillance on both units, reporting key observations to both the plant superintendent and the operating supervisor. In general, my observations involved the reactor water level, neutron monitoring, relief valves and drywell temperatures and plant effluents.

When P-Supt. initiated the Radiological Emergency Plan (REP), I took the necessary steps to determine how many results section people were on site and had this information passed on to the Assistant Administration Officer, AAO-AP. P-Supt. requested that I maintain the REP director's log book and exercise access control to the plant while he made some phone calls. I attended a PORC meeting from 2045 hours to about 2330 hours and left the plant at about 2400 hours.

I have read the statement summarized above, which is true and correct.

/s/ William S. Little	/s/ PRS-EEE	4/25/75
WITNESS	SIGNATURE	DATE

I, ASE-FFF, voluntarily give the following information to Floyd Cantrell, Tolbert Young, and James Devlin, who have identified themselves to me as representatives of the U. S. Nuclear Regulatory Commission.

I am an Assistant Shift Engineer and have been employed by the Division of Power Production, TVA, since 1969.

I am assigned to Unit 3.

I arrived on site about 3 p.m. the day of the fire and was sent to work in the control room Unit 2; relieved UO so he could fill out his log. At the time I arrived in the control room, the reactor was shut down, the MSIV's were closed, pressure was being blown down and reactor water was normal and being maintained by the RCIC.

At about 4:30 I helped trying to get the vacuum pump on the line but we could not get it on at this time.

We were having trouble with the feedwater pump bypass valve and I went out to check on it. The reactor pressure was at 90 psi; the FW pump bypass valve could not be opened. We could not use the condensate pumps to feed the vessel. However, by turning on the booster pump we found that the added pressure caused enough leakage through the FW pump bypass valve to maintain the vessel level.

About 7 p.m. AOS-W told me that there was a CAM alarm on Unit 3 and to check it out along with the exhaust dampers on the reactor building and the turbine building of Unit 3. The alarm was clear when I arrived and I finished the other checks about 9 p.m.

SE-R sent me to work with the electricans on diesel generator No. "C." The generator was tagged out by me. The electricans lifted the ground straps and meggared the generator, found no problems. The ground straps were replaced and when the logic power switch was closed the diesel generator automatically started. We shut it back down.

I went home at 11 p.m.

I came in on my regular shift on Sunday and worked 16 hours hanging JIWR tags and cards.

I have had fire training but never participated in a fire drill at Browns Ferry.

The maximum number of people I saw in the control room at any time was about 8.

AOS-W seemed to be in charge.

/s/ James W. Hufham	/s/ ASE-FFF	4/23/75
WITNESS	SIGNATURE	DATE

I, MMF-GGG, make the following free and voluntary statement to C. E. Murphy, who has identified himself to me as a representative of the U. S. Nuclear Regulatory Commision:

I am a maintenance machinist foreman.

On March 22, I was in the shop when the alarm went off. I was in the boiler shop and went out into the assembly area. (This was specified in the Browns Ferry procedures.) I also saw some instrument mechanics come in. About the same time an operator was coming in with a couple of Scott Air Packs. Another man and I went to the second floor reactor building north side. We saw smoke. We started getting fire extinquishers into that area. We brought all that we could. We stayed in the vicinity 10 or 15 minutes. We sent back and got air packs in the machine shop. By then people had moved to the first floor of the reactor. After we got downstairs we refilled Scott Air Packs with air in the service building. I left the plant at 11:00 p.m.

I have read the statement summarized above, which is true and correct.

/s/ C. E. Murphy	/s/ MMF-GGG	4-22-75
WITNESS	SIGNATURE	DATE

I, Elec.-HHH, make the following free and voluntary statement to C. E. Murphy who had identified himself to me as a representative of the U. S. Nuclear Regulatory Commission:

I am an electrician employed by TVA. On March 22, I was working in Unit 3 getting some brackets welded to a junction box when the power went off and the welding machine stopped operating. My foreman came by and said that there was a fire in Unit 1. I was on the 621 level at that time. I continued to work for about 10 minutes until my working buddy returned to get me to leave the area.

I never heard the fire alarm until I got near the Unit 3 control room area. I left the building at about 1:45 p.m. and went home about 2:30 p.m.

I returned Sunday morning at 11:00 a.m. to work with EF-DDD to get three valves operating. We continued to work until 2:30 a.m. Monday morning. I returned to work on Tuesday morning at my regular hours and worked on the 565 level of Unit 1 putting coaxial connectors on cables. I noticed some soot in this work area but it was not as extensive as on the 593 level.

I have performed cable sealing operations using RTV, mixed foam (two part solution), and pressurized foam. I have had some experience with RTV igniting before the RTV has cured. I was always able to put it out with my hand.

/s/ C. E. Murphy	/s/ Elec.-HHH	4-16-75
WITNESS	SIGNATURE	DATE

I, IE-JJJ, make the following free and voluntary statement to Michael V. Annast, who has identified himself to me as a representative of the U. S. Nuclear Regulatory Commission:

I am an Instrument Engineer.

I was at home on March 22 when I received a call from IE-JJJJ at about 1:30 and was advised a fire was in progress. I picked up IE-AV and came to the plant. I arrived at 1400 and was met by IE-JJJJ. First, I went to the assembly area to head count all instrument mechanics-then, I went to the control room. IE-AV went to the control room with a walkie-talkie. I suited up with another individual and entered the reactor through the 621 emergency access hatch. Hard to see and enter there, so I decided to enter at 586. Found it impossible to proceed to instrument panel 25-5. The smoke was too thick; so we left the reactor building and returned to the control room. The reactor water level was still good. Then we went down to Unit 2 and helped the operator reset the scram. IE-AV, IE-JJJJ, and IE-AW were looking at various problems. I helped clear the main steam isolation system on Unit 2. Several instrument mechanics were in the plant taking temperatures--getting air to the control room. At about 1800 I sat down to decide future coverage--use of IM's for the weekend. Engineers were assigned two on each shift. The first priority was neutron monitoring, so I took two drawers from the control panel to see if operational--and to hook up the SRM's. This was done at 2300 on 3/22. The SRM's were in local service at 0100 on 3/23. I went home at midnight. According to the log at 0220 on 3/23, the water level was within 1" of normal and the drywell pressure was normal (2.3 psi). These were readouts in main control room. Temporary power was run over from Unit 2.

At 0615 we got a drywell temperature reading by cutting the thermocouple wire just outside the drywell to obtain the local reading. Again, we got local readings off the thermocouples for the vessel temperature (at 1730 on 3/23).

At 2100 3/23 again, by cutting thermocouple cable, we got the local temperature reading of the moderator.

At 0030 on 3/24, we pulled temporary cable for the vessel moderator (hooked up to recorder in the control room).

We performed a functional test on the recorders; the torus temperature was taken locally.

IM's who participated in fire fighting were IM-AZ, IM-BA, and IM-AX. IM-AY pulled the air hose into the control room.

I have read the statement summarized above, which is true and correct.

/s/ Michael V. Annast	/s/ IE-JJJ	4-23-75
WITNESS	SIGNATURE	DATE

I, GEF-KKK, make the following free and voluntary statement to Michael V. Annast who has identified himself to me as a representative of the U. S. Nuclear Regulatory Commission:

I am an electrical supervisor with TVA. I was called in and reported to the plant at 2:30 p.m. on the 22nd. I assigned my foreman and four electricians to pull cable to get power on some temporary fans and to provide temporary lighting. The fans were needed to pull smoke out of the cable spreader room. I remained at the plant until 6:00 a.m. Sunday morning, and returned to the plant at 2:00 p.m. Sunday afternoon. I then had 20 electricians available under my supervision. A major activity at that time was to replace damaged cable to establish a feed from Unit 2 to the Unit 1 lighting board in order to get the permanent lights back on. These crews returned to normal working hours on Monday.

I am aware of the fire that occurred on March 20. I understand that it had been written up in the shift log and that it involved an electrician checking for leaks with a candle. I understand that either the RTV or the packing material ignited and the electrician had put it out.

I also understand that the shift engineer had been notified.

/s/ Michael V. Annast	/s/ GEF-KKK	4-21-75
WITNESS	SIGNATURE	DATE

I, EA-LLL, voluntarily give the following information to Donald Caphton, Tolbert Young, and James Devlin, who have identified themselves to me as representatives of the U. S. Nuclear Regulatory Commission:

I am employed as an Electrician Aid in the Conduit and Grounding Group, Tennessee Valley Authority. I have been employed by TVA for one year, ten months in mail room and two months assigned to the Conduit and Grounding Group.

On Saturday March 22, 1975, I was working on Units 1 and 2 on the second shift, 7:00 a.m. to 3:00 p.m. checking for air leaks around conduits. In the morning I worked two hours on elevation 593, below the shutdown Board Room. Later I went to elevation 621 to help EA-LLL. I stayed with him for approximately 15 minutes, and then went to the opposite side of the wall directly opposite the conduit that EA-LLL was working on. This was on the Unit 2 side. I was there for possibly one half hour when the fire alarm went off. I did not know it was a fire alarm until the electrician working with me told me it was the fire alarm.

With me on the 621 elevation were three electricians. We continued to work after the fire alarm sounded. An employee of DPP came in and took a fire extinquisher. He told us it was a fire in Unit 1 and not to worry about it.

I then climbed up to the top of a cable concrete house to check a conduit. Some 30 to 45 minutes later another person from DPP told us to leave the building. We did not hear the fire announced over the PA.

When we got to the stairwell the lights were out and the air heavy with smoke. The elevator was out, everybody was using the stairs.

I check for air leaks with a candle. The candle is held close to the conduit that had been packed with foam and sealed with RTV. I do not normally check for air leaks in a conduit that has only the foam, I check after the sealant is placed on the penetration. The RTV has ignited on at least three to five times, the slender strings or runs catch on fire. I have been told by EE-B that the foam and RTV are flammable. The candles are supplied by the electricians.

I have been working on this site for twelve months and have never been given any instructions in emergency procedures.

/s/ C. E. Murphy	/s/ EA-LLL	4/16/76
WITNESS	SIGNATURE	DATE

I, Elec.-MMM, make the following free and voluntary statement to Michael V. Annast, who has identified himself to me as a representative of the U. S. Nuclear Regulatory Commission:

I am an electrician.

I was not at the plant on Saturday, March 22, 1975, when the fire started. I came in on Sunday morning to work with the electrical engineers at getting the valves operating. On March 20, when I was sealing penetration leaks in Ready Room I and checking the draft with a candle, I set fire to the RTV sealant, but I was able to snuff it out with my gloved hand. I reported to my foreman FM-BN I have had fires occur before from the RTV catching fire from the candle flame.

I have read the statement summarized above, which is true and correct.

/s/ Michael V. Annast	/s/ Elec.-MMM	4/23/75
WITNESS	SIGNATURE	DATE

I, Elec.-NNN, make the following free and voluntary statement to C. E. Murphy, who has identified himself to me as a representative of the U. S. Nuclear Regulatory Commission:

I am an electrician with TVA. I was in the warehouse getting material when the fire alarm sounded. I went to the assembly point near the general foreman's office and later went home. I was called back at 1:30 a.m. on Sunday and reported back to the plant at 1:00 p.m. I worked with EF-DDD in Unit 1, hooking up three valves. I had to suit out in protective clothing at least four times when going in to work.

/s/ C. E. Murphy	/s/ Elec-NNN	4-16-75
WITNESS	SIGNATURE	DATE

I, EA-000, voluntarily give the following information to H. Wilber, T. Young and J. Devlin, who have identified themselves to me as representatives of the U. S. Nuclear Regulatory Commission: They showed me cards which stated that they were USAEC employees.

I am employed by the Tennessee Valley Authority as an Electrical Engineering Aid, assigned to the Reactor Wiring Group. I have been employed by TVA for three years.

On Saturday March 22, 1975 I was working in Unit 3. I heard over the PA system "Fire behind Unit 1, reactor," and also heard the code call fire alarm.

I evacuated on the code call and on the way out heard a siren but I do not know where it came from. Only about 20 people evacuated with me; later all the construction employees came out of Unit 3.

The only emergency drill I remember was one where the siren was used, I do not know if it was a fire or radiation drill.

I know about the fire alarm bell but I do not remember who told me.

I was called back to work Sunday at 1:00 a.m. to help finish writing Work Plan 2920.

Shortly thereafter I worked with a crew and we began to lift cables per the approved Work Plant 2920.

/s/ C. E. Murphy	/s/ EEA-000	4-16-75
WITNESS	SIGNATURE	DATE

I, MS-PPP, make the following free and voluntary statement to C. E. Murphy, identifying himself to me as a representative of the U. S. Nuclear Regulatory Commission:

I am a maintenance supervisor. I was at home when an aide called me about noon on March 22, 1975. Upon arrival at the plant I immediately reported to the shift engineer's office and to the plant superintendent. I supervised maintenance personnel transporting extinguishers and breathing air to the fire fighting team. I supervised the establishment of a locale for empty cylinders separate from full for both breathing air and extinguishers. I established a breathing cylinder filling operation using 255 cu. ft. cylinders on site; and a shuttle service with the Athens Fire Department for refilling of cylinders. I requested TVA-Chattanooga personnel to obtain full 255 cu. ft. breathing air cylinders and arranged receipt actions. I am not aware of any period in which breathing air wasn't available. I arranged with the plant storeroom for a CO_2 truck to stand by to refill the Cardox system. I issued instructions to the maintenance supervisor (electrical) to pursue some problems during the fire. At 8:50 p.m. on March 22, 1975, I attended a Plant Operations Review Committee (PORC) meeting. I lined up crews for Sunday work. Cleanup operations were started as soon as possible.

I have read the statement summarized above, which is true and correct.

/s/ C. E. Murphy	/s/ MS-PPP	5-8-75
WITNESS	SIGNATURE	DATE

I, AUO-QQQ, make the following free and voluntary statement to Michael V. Annast, who has identified himself to me as a representative of the U. S. Nuclear Regulatory Commission:

I was on duty as an assistant unit operator at the Browns Ferry plant on 3/22/75. I heard the alarm in the NO. 2 turbine building. I called O-KK and was told that the fire was on the second floor of the reactor building. I went there and saw smoke. ASE-L and SE-R were dumping CO_2 on the fire. The flames were about a foot high-three feet from the wall. ASE-G had come in and he and SE-R went back to the control room. I continued fighting the fire with ASE-L. We put on breathing apparatus. The fire went out and kept relighting. The visibility got worse. ASE-L ordered me to take all the people to the first floor. The lights went out and the elevator was stuck. I told IE-JJJJ to clear the building. I needed additional air packs and flashlights. I found two air packs from the shift engineer's office. ASE-L asked me to come to the control room. I went there with AUO-TT and AUO-Q. I was sent to assist O-KK on Unit 2. Equipment was malfunctioning. Releif valves were inoperable even from the backup control panel. There was intermittent loss of instrumentation. RCIC operated in the automatic mode. HPCI was not operating. I assisted in the turbine building 9:30-10:00 p.m.

/s/ C. E. Murphy	/s/ AUO-QQQ	4/29/75
WITNESS	SIGNATURE	DATE

I, MMS-RRR, make the following free and voluntary statement to C. E. Murphy, who has identified himself to me as a representative of the U. S. Nuclear Regulatory Commission:

I am a maintenance mechanical engineer. I was in the mechanical maintenance office in the service building working on paperwork after lunch at about 12:30 on March 22 when I heard the auto-call alarm. I called the shift engineer (SE-R) who said the fire was on the first or second floor of reactor building, Unit 1. I went up to the second floor and saw ASE-L and several operators. They all had fire extinguishers. I walked around to about the middle of the containment and could not go further because of black, toxic smoke. I saw PFM-BR, the pipe fitter foreman. The operators then said that they couldn't go in without air breathing apparatus. PFM-BR and I helped put Scott Air Packs on the operators. Five or ten minutes went by—eventually the smoke began to engulf the whole second floor. Then ASE-L said all who did not have breathing apparatus should leave the floor which they did. Just as I reached the first floor I heard all the noises of scramming. The lights flickered and then went out. I got out of the reactor building. I tried to get extinguishers and extra bottles of air to the entrance of the service building. I called MS-PPP my supervisor, at home, but MS-PPP had already gone to the plant. When MS-PPP arrived he had me stay in the office and accept calls and continue to aid in keeping a supply of air and fire extinguishers. There were two or three large air bottles that were shuttled back and forth to an air compressor for filling at the Athens Fire Department. This continued all afternoon. 15 to 30 bottles of air arrived from Chattanooga about 8 or 9 o'clock that night. I knew of no period when nobody was fighting the fire. I stayed on the job until 11 p.m.

I have read the statement summarized above, which is true and correct.

/s/ C. E. Murphy	/s/ MMS-RRR	4-22-75
WITNESS	SIGNATURE	DATE

I, AUO-SSS, voluntarily give the following information to Tolbert Young and James Devlin, who have identified themselves to me as representatives of the U. S. Nuclear Regulatory Commission. I am an Assistant Unit Operator and have been employed in the Division of Power Production, TVA, since 1971 at Browns Ferry Nuclear Plant.

I was on duty at the time of the fire. My duty station was AUO at the intake structure. I heard the fire alarm and the fire pumps start. I called Unit 3 control room and was told that the fire was in Unit 1 reactor building. I remained on my duty station and was relieved at 3 p.m.

/s/ Howard A. Wilber	/s/ AUO-SS	4-27-75
WITNESS	SIGNATURE	DATE

I, O-TTT, make the following free and voluntary statement to C. E. Murphy, who has identified himself to me as a representative of the U. S. Nuclear Regulatory Commission:

I am a unit operator who normally is on Unit 2. On March 22 I came on duty at the regular time 3:00 p.m. I knew both units were down from my observation of the switchyard gear and I saw a fire truck at the gate. I reported directly to the control room. I was sent to Unit 2 to relieve O-KK who was bringing it down to shutdown cooling. I stayed there the rest of the shift and for another 8 hours until 7:00 a.m. Sunday. SE-V was the shift engineer.

I have had the fire training and refresher.

I have read the statement summarized above, which is true and correct.

/s/ C. E. Murphy	/s/ O-TTT	4-22-75
WITNESS	SIGNATURE	DATE

I, EE-UUU, voluntarily give the following information to Howard A. Wilber, Tolbert Young, and James Devlin, who have identified themselves to me as representatives of the U. S. Nuclear Regulatory Commission:

I am an Electrical Engineer and I have been employed in the Division of Engineering and Construction, TVA, for the past two and one half (2-1/2) years at Browns Ferry Nuclear Plant.

On Saturday I was working in the diesel generator building, Unit 3 when I received a telephone call and told of the fire. After assembling in warehouse No. 14 we were told to go home. This was 2:30 p.m. I did not go into Unit 1 and do not have any knowledge of the fire.

I was called back to work at 6:00 p.m. Saturday, I was assigned to formulate a plan for DPP to get certain valves and a pump back into service, Unit 1. Work Plan 2920 was written.

On Sunday I came into work at 11:30 a.m. I supervised the lifting of cables and installation of jumper cables ASE-QQ and two electricians, Elec.-BQ, and the other I do not know. EE-XX was the senior Electrical Engineer or Group Leader supervising the job. No power lines were lifted. The cables were deenergized probably because the fuses were blown. The fuses were removed and replaced by the ASE.

I have particpated in one or more fire drills at Browns Ferry and I have been in one or more radiation evacuation drills over the past two and one-half years.

/s/ William S. Little	/s/ EE-UUU	April 16, 1975
WITNESS	SIGNATURE	DATE

I, Elec. -VVV, make the following free and voluntary statement to C. E. Murphy who has identified himself to me as a representative of the U. S. Nuclear Regulatory Commission:

I am an electrician with TVA and I was in the warehouse getting some junction boxes when I heard the fire alarm. I finished getting my material and went to the tool room outside at the assembly point. My supervisor told me and another man to stand at the east end of the turbine building and to take a head count to make sure everybody got out. Then, when this was finished I was sent home.

On Sunday March 23, I reported back to the plant at 3:30 p.m. and worked on removing the generator bus ties for about two hours. Then I worked with EF-DDD pulling cable from the shutdown board to valves 69-02, 69-12, and 69-16. When I was working on the 593 foot level during this job, all the smoke had cleared out but we still wore face masks because of the odor.

/s/ C. E. Murphy	/s/ Elect.-VVV	4-16-75
WITNESS	SIGNATURE	DATE

I, AUO-WWW, make the following free and voluntary statement to Michael V. Annast, who has identified himself to me as a representative of the U. S. Nuclear Regulatory Commission:

I am an AUO. On March 22 I came on duty on the 3 to 11 shift. I arrived at the plant and went directly to my normal work location at the permanent intake building at about 2:40. The guard informed me there was a fire on Unit 1. I spent about 10 minutes in the intake structure and then received a call from O-ZZ, the Unit 1 operator, who said for me to go to the control room. O-ZZ told me on arrival that I should join three other AUO's already at work on the RHR valves. As I got there they were coming out and said the work was finished. I went back to control. ASE-NNNN told me to go to Unit 3 - get extinguishers and bring them back. I did this for about an hour. Then I checked out the Unit 2 feed pumps to insure valves in proper (closed) position. After I had reported to Unit 2, SE-V told me to go back to the intake structure. I spent the rest of the shift there till 11:00 p.m.

I have had the fire training and refresher.

I have read the statement summarized above, which is true and correct.

/s/ Michael V. Annast	/s/ AUO-WWW	4/23/75
WITNESS	SIGNATURE	DATE

I, EE-XXX , voluntarily give the following information to Howard A. Wilber, Tolbert Young and James Devlin, who have identified themselves to me as representatives of the U. S. Nuclear Regulatory Commission:

I am an Electrical Engineer, Group Leader, Division of Engineering and Construction, Tennessee Valley Commission. I have been employed by the TVA since October 1971. On Saturday March 22, 1975, I was working in the Unit 3 Diesel Generator Building.* At about 1:00 p.m. I was told by EE-BO, Electrical Engineer, by telephone of the fire and to evacuate Unit 3. At this time I returned to Warehouse 14A. I left the site at 2:30 pm and was later called by EES-A and told to return to the plant at 6:00 pm.

I was placed in charge of a group to analyze cable tray damage, to analyze cables involved to determine what was necessary to get equipment back on line. Sometime between 6:00 p.m. to 8:30 p.m. we were told that the Division of Engineering Design (DED) was working on the same project and it was thought that they could identify the cables much quicker. Up to this time no work was being done by DEC personnel to get equipment back on line. We continued to prepare Work Plan 2920. The tasks identified on Work Plan 2920 were initiated on March 23, 1975.

/s/ C. E. Murphy	/s/ EE-XXX	4/16/75
WITNESS	SIGNATURE	DATE

I, Elec.-YYY, voluntarily give the following information to Howard Wilber, Tolbert Young, and J. Devlin, who have identified themselves to me as representatives of the U. S. Nuclear Regulatory Commission:

I am an Electrician and I reside at . I have been hired and have worked for the Tennessee Valley authority for about five years, at different times.

On Saturday I was working in Unit 3 when the fire started. I was working on the 621 elevation with Elec.-HHH. I just heard of the fire from a person hurrying through the area who said there was a fire. Shortly thereafter I heard the paging system "Fire in Unit 1, elevation unknown." I went to the Auxiliary Instrument Room Unit 3 593 elevation for a piece of equipment. About this time I heard the code call and a siren. It was rather faint and I did not know where it was coming from. I went out the diesel generator building to the east side. Nobody was there. I asked about the fire when outside and was told it was in Unit 1, cable spreader room.

I then went back inside Unit 3 to get Elec.-HHH, who was still on the 621 elevation.

On Sunday, Foreman EF-DDD called me and I went in to work on Work Plan 2920. Worked on valves 69-2, 69-12, 69-16 (RWCU-Unit 1) from prints that I obtained from our print rack in Unit 3. The electrician's shop in Unit 3 have a complete set of prints for Units 1, 2, and 3. I identified the cables and located terminations from the prints.

I was later assigned to pull temporary cables to the above valves, both power and control lines from motor control to valve. I worked with Elec.-EEEE and Elec.-NNN. We did not have an ASE or SE with us, we worked from the prints.

Temporary cables were run whenever possible or convenient. We used waterpipe chases, accessible conduits or whatever. We kept a record of the work we done but the new temporary cables were not tagged by us. DPP later checked and tagged before the power put on them.

/s/ C. E. Murphy	/s/ Elec.-YYY	4-16-75
WITNESS	SIGNATURE	DATE

I, AUO-ZZZ, make the following free and voluntary statement to Michael V. Annast, who has identified himself to me as a representative of the U. S. Nuclear Regulatory Commission:

I am an assistant unit operator at the Browns Ferry plant. I was near the air compressors when the fire alarm sounded. I called the operator and was told the fire was in the reactor building. I went there with AUO-QQQ and assisted ASE-L by passing extinguishers up the ladder.

ASE-L requested breathing appartus. I went back to SE-R's office and got two Scott Air Packs. I gave one to ASE-L and kept the other. ASE-L went up the ladder. The smoke was extremely thick and ASE-L fell off the ladder on his way down. I went to the third floor and ran out of air. I came down and the smoke kept building up. ASE-L told everyone, not directly involved with fire fighting, to go to the first floor. I went to the first floor elevator landing and the lights went out. I waited for 20 minutes and no one went up during this time. I changed out air pack bottles and then went up to the control room. ASE-G told me to get Unit 1 turbine on turning gear. There was no motor power and I returned to the control room. I was handling and refilling air bottles. I was involved in trying to free various valves in the RHR system, with O-MM and AUO-AA but we kept running out of air. I was involved with the RHR drain pump for a couple of hours and got it running by 7:30 p.m. I went home at approximately 8:30.

/s/ Howard A. Wilber	/s/ AUO-ZZZ	4-22-75
WITNESS	SIGNATURE	DATE

I, AUO-AAAA, voluntarily give the following information to Tolbert Young and James Devlin, who have identified themselves to me as representatives of the U. S. Nuclear Regulatory Commission:

I am an Assistant Unit Operator and have been employed in the Division of Power Production, TVA since 1972 at Browns Ferry Nuclear Plant.

I came in to work on Sunday 3/23/75 at 11 a.m. Worked under the direction of Unit 2 operator in the reactor building.

We were informed that the level in the torus was increasing. I found that someone had left the condensate transfer system lined up to Unit 2 torus. We corrected problem and pumped down torus.

Reactor water cleanup system was out of service. We worked on this until 3 p.m. and got "B" system in service. I then went to my regular routine shift duties until 11 p.m.

At 11 p.m. I was held over on overtime to work on Unit 1's reactor water cleanup system. I went home at 4 a.m.

I have never participated in a fire drill at Browns Ferry.

There was a rag fire in the condensate pump pit about three weeks ago. The fire alarm was set off on the day shift. ASE-NNNN put out the fire.

/s/ Michael V. Annast	/s/ AUO-AAAA	4/22/75
WITNESS	SIGNATURE	DATE

I, AUO-BBBB, voluntarily give the following information to Donald Caphton, Tolbert Young, and James Devlin, who have identified themselves to me as representatives of the U. S. Nuclear Regulatory Commission.

I am an Assistant Unit Operator and have been employed by the Division of Power Production, TVA, since 1970 at the Browns Ferry Nuclear Plant.

I was called at 1:15 and arrived on site at 2:30, 3/22/75. I worked with other operators for an hour trying to close a valve on top of the torus.

I then worked on getting service air to the Reactor building entry.

I helped at reactor building entry with air packs, etc., I went home at 4:30 a.m. 3/23/75.

/s/ AUO-BBBB　　4/12/75
SIGNATURE　　DATE

/s/ C. E. Murphy
WITNESS

/s/ SE-PP
WITNESS

4/12/75
DATE

I, AUO-CCCC, voluntarily give the following information to Howard A. Wilber, Tolbert Young, and James W. Devlin, who have identified themselves to me as representatives of the U. S. Nuclear Regulatory Commission.

I am employed by the Tennessee Valley Authority as an Assistant Unit Operator. I have been employed by TVA for over five years and at Browns Ferry since November 1971

On Saturday, March 22, 1975, I was assigned to the second shift, Unit 1. I arrived at 2:35 p.m. and was told of the fire.

ASE-EE sent me with AUO-RR, another AUO to elevation 565 to manually open the suction side valve, RHR pumps to establish shutdown cooling. We tried three times but did not have enough air in the breathing apparatus tanks.

We reported back to ASE-EE. We thought we could get to the valves by going through the shutdown board room. We would have tried but ASE-L stopped us and told us to go to the cable spreader room.

We relieved AUO-X and O-AAA. Two public safety officers, one was wearing a Scott Air Pac, were moving fire extinguishers into the room. This was approximately 3:30 p.m. The cables inside the penetration window were still glowing and everytime we stopped spraying the CO_2 the cables would immediately start to glow again. Somewhere around 4:15 to 4:30 p.m. the cables stopped glowing and the fire appeared out.

I then went to the roof with ASE-QQ to open up the ventilation dampers. A wrench was needed. One exhaust fan was running. I did not know if the supply fans were running. I left ASE-QQ, AUO-Q, and an electrician on the roof. They had a walkie-talkie and were communicating with the control room. The electrician had jumped the power lines on the other fan. This was about 5:20p.m.

I returned to the control room. There were about eight people there, AOS-W and SE-R were directing the activity.

I was sent with AUO-RR to check the breakers on two valves in the RWCU that were needed to dump water. We tried the breakers, one didn't work, the other sparked. We didn't try any further.

The smoke was heavy when we went to the MOV room but by the time we were returning the smoke had cleared enough to enable us to see several feet. We left by the lower level air lock. I reported back to ASE-EE. At about 11 p.m. Sunday these valves were put into service.

At approximately 6:30 p.m. ASE-EE sent me to the Rx building level to assist at the scene of the fire. ASE-Y. SE-V, SE-U, and the Athens Fire Department Chief and I believe another fireman were there. The decision was made to use water on the fire. SE-U put the fire out with water. At about 7:30 p.m. after again spraying the fire with water, it was declared out.

AUO-RR and me were assigned to fire watch the rest of the night. I went home at about 6:15 a.m. Sunday.

/s/ C. E. Murphy	/s/ AUO-CCCC	4/16/75
WITNESS	SIGNATURE	DATE

I, QASS-DDDD, make the following free and voluntary statement to Howard A. Wilber who has identified himself to me as a representative of the U. S. Nuclear Regulatory Commission:

I am the QA staff supervisor for the BFNP, operations. On March 22, 1975, at approximately 1:00 p.m. I received a telephone call from AOS-W. AOS-W said that there was a fire in the cable trays at the plant and that a safety system may be involved. I arrived at the site at approximately 2:00 p.m. A fire truck (assumed it to be the Athens Fire Department) was outside the fence at the gate house. Upon entering the lobby, I saw IE-JJJJ, who briefed me about the fire. He advised me not to go to the control room because of smoke. Shortly I saw two instrument mechanics who said that the instrument shop area was fairly free of smoke. So I went up to that area where I saw SE-R trying to restore equipment. I decided to stay and work with an operator in an effort to restore equipment. Shortly, P-Supt. came in. I asked him if I was needed e'sewhere, he said no. Shortly after additional personnel came in, particularly instrument mechanics, I left and went to the control room. There I saw several people including AOS-W, OS-DD, PRS-EE, O-MM, APMS-FFFF, and P-Supt. (Some of these were in and out of the control). Specifically, I noticed that AOS-W appeared to be in direct charge of Unit 1 reactor operation, OS-DD moving from unit to unit, and P-Supt. in general being the focal point of control.

While in the control room, I observed the condition of both reactors. Later I decided to go to the spreader room area. There I assisted in bringing dry chemical fire extinquishers to the spreader room area. After many extinguishers were brought and manpower appeared adequate to fight the fire, I left for the control room. The extent of the fire in the spreader room appeared only to be a glow in the penetration. It would extinquish after dry chemical was used and then about a minute later apparently reignite. A glow is all I would see. I returned to the control room and was instructed to go with ACS-BD to start identifying the affected cables. This was in an effort get rebuild started as soon as possible. (The magnitude of the fire damage was not know at this time) After I returned to the plant, approximately about 6 p.m. I went to the spreader room. The fire was out. I visually looked at the penetration that was affected. Some time later, probably about midnight, I, ADDPP-BE, and P-Supt. went into the reactor building to look at the damage. Prior to entering, we were advised of the industrial and radiological hazards associated and dressed accordingly. Damage to the cable and cable trays was observed. Sometime later I met with Bill Little and Bob Sullivan and with P-Supt. About 3:00 a.m. on March 23, 1975, I left the plant.

On March 20, 1975, at the daily coordination meeting, someone, I think OS-DD, stated that there was a fire last night or yesterday. It was associated with the sealing material used for cables. A little discussion was held about the fire, but no specific details such as how it started and where it was, was discussed. I assumed it to be in Unit 3 while construction was sealing cables that passed through the floor. I was not aware prior to this time of the process to seal fire barriers in penetration. The next day more detail, such as the sealant was sprayed in and was flammable while drying, was discussed. Again here I don't think the location, or if it were in a tray, was discussed.

I dismissed the occurrence as minor because of the nature of the discussion concerning the fire. Again I assume it to be in Unit 3 during construction activities, not in Unit 1 spreader room.

I have read the statement summarized above, which is true and correct.

/s/ Howard A. Wilber	/s/ QASS-DDDD	April 25, 1975
WITNESS	SIGNATURE	DATE

I, Elec-EEEE, make the following free and voluntary statement to C. E. Murphy, who has identified himself to me as a representative of the U. S. Nuclear Regulatory Commission:

I am an electrician employed by TVA. On March 22, 1975, I was working on the second floor of the control bay in the Unit 3 area. I was working with another electrician installing an inverter. I heard the alarm followed by a P.A. announcement which identified the fire and alarm. I continued to work for approximately 15 minutes but I kept hearing the P.A. announcements. I really was not sure whether or not I should evacuate. I walked to the far end of the Unit 3 area where I saw a foreman who said I should leave. I left the area about 1:00 or 1:15. In the areas I was in I saw no smoke, no fire, nor did I smell any odor.

I was called back to work on Sunday March 23, 1975, to work with Sub-foreman EF-DDD in hooking up power to 3 valves; 2 of these being in the heat exchanger room (69 series valves). In doing this work we had to follow all necessary clothing requirements. We did our work with flashlights because no temporary lights had been installed in the heat exchanger room. When I finished, my dosimeter almost read 100.

WITNESS ____________ SIGNATURE ____________ DATE ____________

Elec-EEEE refused to sign or to provide signed statement.

/s/ C. E. Murphy

I, APMS-FFFF, make the following free and voluntary statement to C. E. Murphy, who has identified himself to me as a representative of the U. S. Nuclear Regulatory Commission:

I am the Assistant Plant Maintenance Supervisor, Electrical. I was at home on March 22 when I was called to come to the plant at about 1:00 p.m. I arrived at the plant at about 1:30 or 1:40. I was not involved in the fire fighting. I found numerous boards were out, and I worked with operations to restore power to the equipment. In clearing the trouble to restore power, as anything was lifted or jumpered it was written down, and later put in BFA-25 (JIWR). I worked all over the plant supervising electricians doing this work. I was relieved by FM-BG at 11:00 that night. On Saturday afternoon the power (lighting) was off in the reactor building. On Sunday, temporary lights had been installed. Work was done on verbal request on Saturday and the JIWR was backfit by FM-BG on Sunday. I was back in the plant at 0700 on Sunday.

Concerning the incident of March 20, AOS-W had asked me on March 20 to look at the cable tray where the fire had occurred to determine if any damage to the cable. AOS-W supplied a man to point out the location. My foreman, EF-BF, looked at the cable and reported back to me that there was no damage to the cable. I did not make any report about this inspection. I have been at Browns Ferry since 9/20/69. This is the first fire I have been involved with here. My personnel have received local fire training, and indoctrination in various types of alarms. Procedures are periodically reviewed in random safety talks or safety newsletters.

I have read the statement summarized above, which is true and correct.

/s/ C. E. Murphy	/s/ APMS-FFFF	5/6/75
WITNESS	SIGNATURE	DATE

I,ME-GCGG, voluntarily give the following information to Howard A. Wilber, Tolbert Young, Donald Caphton, and James W. Devlin, who have identified themselves to me as representatives of the U. S. Nuclear Regulatory Commission:

I am employed by the Tennessee Valley Authority as a Mechanical Engineer Coordinator, Division of Engineering and Construction. I have been employed by TVA since January 1965 and was transferred from DPP to DEC on July 25, 1971.

The work being performed by the electricians and engineering aides at the time the fire occurred was under Work Plan 2892. This plan covered the sealing of air leaks in penetrations in the secondary containment and was issued as a result of a test of the Standby Gas Treatment System. Technical Specifications were met in this test. The leak rate test was performed in anticipation of the removal of a barrier between Units 2 and 3 on the fuel storage floor and refueling floor.

This Work Plan did not spell out procedures to follow, only the prescribed work to be performed. I did not know the method used to detect air leaks, i.e., using a lighted candle; these are details and procedures, not part of a Work Plan. The job was in progress for two weeks.

A number of cables were pulled through the penetrations in conjunction with previous modifications and/or alterations under other Work Plans. The penetrations should have been sealed when the cables were pulled and were specified on each Work Plan to pull cable.

In 1973 and 1974 the same type of work was done using the same type materials, polyurethane and liquid rubber and transite boards.

The Work Plans go to the DPP Coordinator, DPPC-BM, and are issued to the Shift Supervisor and filed with the Shift Engineer.

Following completion of Work Plan 2892 it would be normal to conduct another leak rate test of the Standby Gas Treatment System under another Work Plan (Work Plan No. 2837).

On Sunday I was called back in to work on a Work Plan for emergency repairs and stabilizing air lines, vessel thermocouples RHR instrumentation, drywell temperature instrumentation and SRM and other equipment not yet running.

It was determined by both DPPC-BM and myself that 10 CFR Part 50.59 review was not necessary on the Work Plan for sealing of penetrations since it was a followup to the leak test work plan.

/s/ C. E. Murphy	/s/ NE-GGGG	April 16, 1975
WITNESS	SIGNATURE	DATE

I, AUO-IIIIIII, make the following free and voluntary statement to C. E. Murphy, who has identified himself to me as a representative of the U. S. Nuclear Regulatory Commission:

I am an assistant unit operator.

I live nearby and was at home. My wife heard the fire sirens when I was asleep. I called in and was told to come back to the plant. I went in about 2:45 p.m. and reported to the ASE in the shift engineer's office. AUO-MCCM and I were trying to open valves on the torus. Our attempts were unsuccessful since the visibility was so bad. I was assigned to keeping track of people and how much time they had on their air packs at the first floor reactor building entrance. Between 6:00 and 6:30, I took the Athens fire chief to the spreader room. We could see the glow of the fire on the other side through the opening. We did not go to the fire on the reactor side.

I had no knowledge of a prior fire. I had been off on Thursday and Friday.

I have had fire fighting training--1 week at a coal plant and a refresher since then.

I have read the statement summarized above, which is true and correct.

/s/ C. E. Murphy	/s/ AUO-HHHH	
WITNESS	SIGNATURE	DATE

I, IE-JJJ, make the following statement, freely and voluntarily, to Michael V. Annast who has identified himself to me as a representative of the U. S. Nuclear Regulatory Commission:

I am an Instrument Engineer and was working in my office at the Browns Ferry plant on March 22, 1975.

The fire alarm came in at 12:35 p.m. After I was sure the fire alarm was serious, I went up to the control room. When I entered the control room, I noticed that the RHR and core spray pumps started by observing annunciation on panel 9-3. I discussed this anomaly with O-M and suggested he back off on the load. By using RECIRC, he initiated it with the concurrence of the ASE. We were getting numerous alarms on panel 9-3 and at about 12:50 p.m. O-M manually scrammed Unit 1. I then walked over to Unit 2 to see if it was being influenced by the abnormal occurrences observed on Unit #1. At that time, I noted none of the phenomena occurring. O-KK, the Unit #2 operator, told me to stay on Unit #2 side to help if necessary. He then asked me to put the Unit #2 reactor building fans from high to low speed. When I returned, I saw Unit #2 scrammed; I think manually but I am not sure. O-KK enabled the Cardox and I heard it go off below in the spreading room. I noticed that one half of the RPS power was lost on Unit #2. I went back to Unit #1 and noticed several instrumentation problems, specifically on panel 9-3. SE-R asked me to call P-Supt., IE-JJJ, and AS-GG at about 1:10 p.m. I stayed in the Control Room. At about 3:00 p.m., I was alerted to smoke emanating from the back of panel 9-5, Unit #1. We used CO_2 on this and removed the leads from the smoking transformer. I felt very concerned at this time because the control room was getting quite smokey. Service air was brought in from Unit #3 by an instrument mechanic. GEIE-CZ GE Instrument Engineer, arrived at about 3:30 p.m. I was called to Unit #2 to help investigate a PCIS isolation on the Unit #2 MSIV's. Two temperature switches in the main steam tunnel were tripped. One switch reset while I was investigating, one remained tripped. These switches had caused problems prior to the fire, due to conservatism in trip set points. Steam tunnel temperature was 150-160°F at 2:30 - 3:00 p.m. I want to state that I jumpered the tripped switch on order of the Unit #2 ASE and registered the jumper wire record on the SE's blackboard. I found GEIE-CZ and together we suggested that two Unit #1 SRM's be removed from the control board of Unit #1 and be made ready for local installation at the preamp since the cables appeared damaged. Unit #2 was having HPCI problems. The suction valve from the torus kept opening and the suction valve from the condensate storage tank kept closing. The Unit #2 operator stated he had plenty of water in the storage tank and wanted to take HPCI suction from it. He had me hold the valve handswitch in the storage tank suction mode while HPCI continued to automatically start and then trip on low suction pressure. The pump would speed up and pump about 2000 gpm and trip after about 5 seconds. Water level was being maintained manually using what appeared to be a FW pump. I was told that some feed level indication was lost so that the recorder could not be used on Unit #1. I found the "B" column inverter fuse blown and replaced it.

I left the plant at 6:00 p.m. and returned at 11:30 p.m.

I feel that the fire fighting effort was inadequate due to improper equipment and confusion. The hand-held CO_2 extinguishers were not adequate as they are not designed for high ceilings. The fire fighting organization was poor; there seemed to be a lack of leadership and no one knew exactly what was burning. I heard that the air supplies were hard to find and did not fit well together.

During the test and startup period of Unit #1, I demonstrated the flammability of the sealing material to P-Supt., the plant superintendent. In the presence of IE-JJJ, I burned the material in P-Supt. office. P-Supt. immediately called someone with Construction, apparently CPM and they discussed the situation. I was later (the next day) told by PRS-ELE, Results Supervisor, that a flameproofing material was to be sprayed over the flammable material. I feel P-Supt. did all that was immediately possible to investigate the situation as it appeared DED was not going to change the material. GEIE-CZ, GE, also sent a sample of the above material to his office in San Jose,. California. I have heard that the GE people tested the material and sent a report to TVA.

/s/ Michael Annast	/s/ IE-JJJJ	4/28/75
WITNESS	SIGNATURE	DATE

I, O-KKKK, make the following free and voluntary statement to Michael V. Annast, who has identified himself to me as a representative of the U. S. Nuclear Regulatory Commission:

I am a Unit Operator at the Browns Ferry plant. I was called at about 1:00 p.m. because they had lost both units and had a fire at a hard-to-get to location. I arrived at 3:00 p.m. and went to Unit 2 where O-KK was the Operator. AOS-W assigned me the personnel access hatch to the reactor building on the first floor. I was posted to insure the integrity of the interlock since the red light was out. After one hour I asked AUO-SS, who was at the auxiliary boilers, to watch the door and went up to Unit 1 Control Room. I assisted with operation of relief valves to lower Rx press to get water into the reactor with condensate booster pumps. The control room was hot and, at times, smokey. Air packs had insufficient supply of air and it was difficult to get valves open manually. I saw some white smoke emanating from the back of the 9-5 panel. Someone put CO_2 on it and it stopped. Electricians were working around the floor penetrations to keep the CO_2 in the spreader room below. I went off shift at 11:00 p.m.

/s/ C. E. Murphy	/s/ O-KKKK	5/6/75
WITNESS	SIGNATURE	DATE

I, EA-LLLL, make the following free and voluntary statement to C.E. Murphy, who has identified himself to me as a representative of the U.S. Nuclear Regulatory Commission:

I am engineering aide, Grade SE-2, employed by TVA. Afternoon on March 22, I was working on the 639 level in the Reactor Building (Unit 2). There were four big pumps in the room which were making a lot of noise and I did not hear any fire alarm bell. I was checking conduit and cable tray seals and working with some electricians. We were using RTV sealant with foam material as a backing. I became aware of the fire when I saw a DPP man looking for a fire extinguisher who told me there was a fire in Unit 1.

I started to leave the building and on the way down the stairs I ran into EE-B and EA-H (on the second floor). We were going down the stairs because the elevator was out of service. EE-B was in the building, he said, to check that his people were getting out.

When I left the building, I went to the assembly area and then went to the Conduit and Grounding Offices and stayed there until about 2:30 p.m. when I was told to go home.

The first time I worked on sealing and checking cable seals was on Thursday night when I was shown how to do it by EE-B. On Thursday night, I worked on the reactor sid of the cable spreading room; on Friday night I worked with EA-C in the Cable Spreading room; and on Saturday I worked mainly on the Reactor Side. On both Thursday and Friday I had experienced fire starting with the RTV from the candle flame used for checking the seals. The flames were always easy for me to extinguish with my gloved hand.

When I first saw the DPP man on March 22, when he told me of the fire, I quickly went back to the area where I had been working to check and make sure that I had not caused any fires.

/s/ C.E. Murphy	/s/ EA-LLLL	4-16-75
WITNESS	SIGNATURE	DATE

I, AUO-MMMM, make the following free and voluntary statement to Michael V. Annast, who has identified himself to me as a representative of the U. S. Nuclear Regulatory Commission:

I am an AUO and I rotate between units every 10 weeks. I was at home on 3/22 when I was called to come back to the plant at about 1:45 p.m. I arrived at the plant at about 2:15 p.m. ASE-RRRR had telephoned me to say that they had lost both units and had a fire. I went directly to the control room. I saw the situation was bad. AUO-UUUU and I worked to do the shutdown cooling for Unit 1. There were few lights in the control room and it was smokey. I talked to ASE-G who had been on thr torus with an air pack, and who said they lacked one valve of getting shurdown cooling on Unit 1. AUO-UUUU and I got air packs on and went down on top of the torus. We could not operate the valve easily without a "cheater". We used one but it still took an hour or more for both of us to get the valve open. We went to shutdown cooling shortly after that. This was about 4:30. ASE-G had been trying to open this same valve sometime before we got there. After we had informed the unit operator the valve was open, AOS-W told us (AUO-UUUU and me) to work with SE-U at fighting the fire. SE-U was in the control room hall--shift engineer's office talking to the Athens fire chief and P-Aupt. We went down to the spreader room. We then went to fire area in reactor building. At first, I kept check at the door (while SE-U was inside using water. I then went in with O-ZZ and found that the hose was lined up with the fire but the hose was hung up on the rack. We got the hose off the rack and O-ZZ started putting water on the fire. The fire was on only one tray which I believe was the top one. Shortly after this O-ZZ ran out of air and came out. I had 10 minutes left and continued with the water hose. I was working alone for about 5 minutes when an Athens fireman came in and said he did not want me to work by myself. I came out with him. I had spent about 10 minutes fighting the fire. It was a smolder type fire with some flames. After my exit about 3 or 4 people put on Chem-ox units and went in to fight the fire.

AUO-UUUU and I went to eat at about 6:30. When we got back upstairs after 7:00, we learned that the fire had been put out.

ASE-PPPP told me and AUO-UUUU that he wanted us to go with him on a reactor inspection every hour. We did this until 5:30 Sunday morning. I went home then.

The only fire I had known of before was a welding machine fire a couple of months ago.

I had one week training at a coal fired plant in fire fighting, and we had one day refresher here.

During the Afternoon of March 22, I knew of no time when the fire was not being fought. We were told that day to use water as a last resort. When doing the hourly reactor inspections, I reported them to ASE-PPPP who logged them. During our tours which began about 9:00 p.m. the smoke gradually diminished. There was no evidence of any fire glow or embers during these tours. We had never lost the lights in Unit 2, but had lost them in Unit 1. Normal lighting got back in use after our inspection tours. Every hour we went over to where the fire had been to verify no fire. We noticed burned insulation.

I have read the statement summarized above, which is true and correct.

/s/ Michael V. Annast	/s/ AUO-MMMM	April 28, 1975
WITNESS	SIGNATURE	DATE

I, ASE-NNNN, make the following free and voluntary statement to MICHAEL V. ANNAST, who has identified himself to me as a representative of the U. S. Nuclear Regulatory Commission. Some of this statement is in response to direct questions of the NRC investigators. I have tried to show where things that I saw and heard were an expression of my opinion and not necessarily the facts as they occurred. This statement is correct to the best of my memory.

I am an Assistant Shift Engineer at the Browns Ferry Plant. I received a phone call from the plant at approximately 1315 and was told by ASE-RRRR that Units 1 and 2 had been lost (tripped) due to a fire and they were in danger of losing the plant. I arrived at the plant at approximately 1325, saw the Athens fire truck at the gate and I could still hear the fire alarm sounding. There was no visible outward signs of the fire coming from the plant.

I entered the plant via the office wing, first floor, stopped in the lunchroom and called the Unit 1 Control Room. I talked with (SE) SE-R and asked the location of the fire and where I was needed. SE-R instructed me to come directly to the control room. When I arrived at the 3C control bay elevation, the AC lighting was off and only DC lighting on. I saw ASE-RRRR and AAO-AP in the Shift Engineer's office on the telephones. In the Control Room I saw the operator, O-M, at panel 9-5 and Unit 1 panel 9-3 was dead. I saw SE-R and ASE-L at panel 9-23 and went directly to them and asked what was needed at approximately 1330.

I saw that shutdown Bus 1 was dead, "A" D/G was on "A" shutdown board, "B" shutdown board and "C" shutdown board were both on Shutdown Bus 2, "D" D/G was on "D" shutdown board.

ASE-L said he could not get any indication on B&C D/G's and directed me to check out the diesels. I proceeded via the stairwell, which was very heavy with the smell of burnt insulation, and "B" shutdown board room to the diesel building. I went first to "B" D/G which was running at half speed. I checked all relays and switches and could find nothing wrong, so I tried to take manual control and increase to normal speed in an effort to place in service. The "B" D/G would not respond and I returned the diesel to its normal operational mode. The only alarm was that of low air pressure.

I then went to "C" D/G which was running at rated speed with 4.1 or 4.2 KV. As I left the diesel building I saw Electrician EF-BF working at 480V Reactor MOV board 1B and he asked if I could help him. I cleared all tripped breakers with EF-BF's help and tried to reclose the normal and alternate feeders with no success. I told EF-BF that I had to go and would return or send help and to do what he could to get the board back. As I left the board room ASE-G was coming in and said he was going to work with EF-BF.

I then went to the "C" shutdown board with a Sync. switch and tried to gain control of the diesel but all diesel instruments and controls were dead.

I returned to the Control Room and reported to the SE-R what I had found.

ASE-Y and I left the Control Room to try and restore the RPS MG sets on Units 1 and 2. We tried first Unit 1 RPS MG sets, but they had no power. Then we got Unit 2 RPS MG sets started but they tripped. Then they were restarted and placed in service. I called Unit 2 Control Room to tell them they had the RPS MG back and was told they were unable to control the relief valves. APMS-FFFF came up just then and ASE-Y, APMS-FFFF, and I went to "C" shutdown board room to the backup control center to try and operate the Unit 2 R.V.s, but could not. We checked 2A 250V DC Reactor MOV Board but found no trouble. Then we entered Unit 2 Reactor Building via the shutdown board room. Unit 2 Reactor Building was clear of smoke and all lights were on. ASE-Y checked the drywell air compressors and said they looked normal. We re-entered the control bay via "D" shutdown board room to check out 2B 250V DC Reactor MOV board, which was hot.

We left the shutdown board room and went to the 2C elevation hallway where we met ASE-L and discussed the problem briefly. It appeared to me that the drywell control air isolation valves must be closed but I had not seen this. Someone in this group (I don't remember who) said he would check on it. ASE-Y and I left the group.

I returned to the control room. When I arrived on the 3C elevation I heard an air line blowing. The control room was hazy and the air difficult to breathe. The operaters on the units were in air packs. I saw SE-V directing operations on Unit 2 and AOS-W directing operations on Unit 1. P-Supt. was consulting with different people and directing the overall effort it appeared. I saw people forcing rags in holes under the electrical operator's desk. CO_2 was coming through them.

I went to the hallway to try and find an air pack, which I had seen earlier in the day. There were two Chem-ox masks in their cases on the floor and ASE-Y and PSO-CC were trying to get them ready to use. I saw they were having problems and tried to help. The unit that PSO-CC had would not puncture the cannister and I don't remember the problem with the other unit. But the Chem-ox mask units did look to be in sad shape in my opinion. PSO-CC got one of the Chem-ox masks working and ASE-Y and PSO-CC entered the west end of the Spreader Room. I followed them to the door and saw a glow against the south wall. ASE-Y and PSO-CC returned to the doorway and I handed PSO-CC a fire entinguisher that was at hand and then I left to find more extinguishers.

I found 6 dry chemical extinguishers outside the SE office and had an AUO help me carry them back to the Spreader Room door and instructed him to stay and help other men who were arriving to fight the fire.

I went back to the 3C elevation to get some help in locating fire extinguishers. I saw an air pack and a cart of air bottles. I went into the control room and saw P-Supt. and told him the fire was not out in the Spreader Room, but we were fighting it. P-Supt. asked me if I knew which cable trays were involved. I told him I did not know, but would find out. I went back to the air mask, put on a fresh bottle and returned to the Spreader Room. PSO-CC was coming out as I arrived and said he was out of air and was going for more help. An AUO and I entered the Spreader Room with dry chemical and CO_2 which we used on the fire.

We used the CO2 to cool the fire and knock down a small flame. Then we dumped the dry chemical in the tray around the burned ends.

A member of the plant PSO group and 2 more people crawled up with more dry chemical and I instructed them to build a fire block in the cable tray with the dry chemical. I made a mental note of all the cable tray identity letters and left the Spreader Room and reported to P-Supt. and a group of Construction and DPP people with him what I had found.

I returned to the Spreader Room to direct the fire fighting effort. A wheeled dry chemical cart had been brought to the Spreader Room, but its nozzle was broken off at the bottle and I told some of the men to get it out of there and find another unit. I stopped 2 AUO's who were coming by with air packs and asked if they were assigned. They said no. They had just finished a job. I told them to go into the Spreader Room and relieve the men in there who were low on air.

A large number of empty CO_2 and dry chemical bottles had accumulated and I instructed some men to carry them to the Machine Shop where they could be refilled when chemicals were available. I re-entered the Spreader Room to check the progress that was being made. The fire was not staying out, but could be knocked down each time it flared up. After some period of time the other people present and I were able to contain the fire in the Spreader Room and confine it to the Reactor Building wall. I instructed the AUO's and PSO people present to stay on top of this fire area and keep using CO_2 and dry chemical to keep the fire confined to the penetration in the Reactor Building wall.

I went back to the Control Room and told P-Supt. that the fire in the Spreader Room had been contained but was still burning in the wall penetration and Reactor Building and that I had people staying with it.

I was instructed by P-Supt. to show the Athens Fire Chief the Spreader Room, which I did. The Athens Fire Chief said he had some ABC Class extinguishers which we could use and I went with him to get them. I returned to the Spreader Room with the extinguishers and the Chief. A new dry chemical wheeled cart had arrived at the Spreader Room and was being used to contain the fire at the wall.

ASE-L came by and said he and ASE-Y needed help in the Reactor Building fire. I replaced my exhausted air bottle and went to the "B" shutdown board room where I saw an electrician standing in the doorway to the Reactor Building and asked him what he was doing. He said that ASE-L and ASE-Y had gone into the Reactor Building to carry a light and cord in so we could see. The Reactor Building lighting was out and the smoke was extremely heavy and black in color. I tried to see into the Reactor Building but could not see beyond the west end of the drywell air compressor.

Presently, ASE-Y and ASE-L returned and I asked the situation. Then I picked up a dry chemical and CO_2 extinguisher and entered the Reactor building behind ASE-L. I could not see around me, but had to walk on ASE-L heels just to see the glow from his flashlight. When we arrived in the fire area, the extension light looked like a white dot on a black field. I could see the flame from the fire and found a ladder leaning against the North wall. I climbed the ladder to the fire area. Using the CO_2, I put out the flame and cooled off the cabling, then went back down for the dry chemical extinguisher. When I attempted to set the dry chemical down on a pipe to activate it, I received a shock and had to move it off the pipe. When the dry chemical was put on the fire it looked like it was out, but in a few seconds it would erupt again. ASE-L and I left the Reactor Building to get more extinguishers and returned and used them on the fire.

We found a wheeled dry chemical extinguisher in the Reactor Building and took it into the fire area and used it on the fire. We saw a new location on the fire. It was burning south toward the North wall of the Reactor WCU Regen and Non-regen heat exchanger room. We put dry chemical on it until we ran low on air. We left the Reactor Building again to find more air.

When we reached the "B" shutdown board room we met (SE) SE-U and told him what the situation was and we were going for more air and extinguishers. When I got to the Service Building, I saw a large number of new fire extinguishers and members of the Maintenance Department filling air bottles. I instructed an AUO to help replace the empty air cylinders with filled ones and bring some to "B" shutdown board room. I then replaced my air tank, loaded a cart with about a dozen CO_2 and dry chemical extinguishers and returned to "B" shutdown board room. SE-SSSS was present along with other men. I asked who was in the Reactor Building and was told SE-U and someone else was in there. I heard an alarm bell from one of their air packs, so I started in and met SE-U and an AUO coming out. I proceeded to the fire area and put dry chemical on some burning cables, then returned to the board room when the extinguisher was empty.

(SE) SE-V, SE-U, and SE-SSSS were present and talking with ASE-L. I saw used equipment laying around and instructed an AUO to remove it from the board room and I helped an Athens Fireman get into a Chem-ox mask. I stayed in the board room and assisted various people with fire fighting equipment until the fire was out. I then went back into the Reactor Building to check the fire out and set a fire watch with 2 AUOs.

I returned to 3C elevation and met SE-PP and he said he wanted someone to go with him into No. 1 Reactor Building to check on the air supply isolation valves to Unit 1 relief valves. We put on air packs and went into Unit 1 Reactor Building via 565 elevation. We could see the power was off or at least the position indicating lights were out on the isolation valves. Then we went around to the west side of the Reactor Building to check 250V DC Reactor MOV Board 1C. It was hot. We then looked around for general conditions and saw nothing further except the heavy coating of soot. Then we left the Reactor Building and met AUO's on their way in.

I returned to the control bay and checked the Spreader Room.

I was feeling sick because of the fumes and was exhausted. I talked with SE-V and told him I was sick and would like to leave, but that I would return if needed at any time. SE-V agreed and I departed the plant at approximately 2030.

/s/ William S. Little	/s/ ASE-NNNN	5/?/75
WITNESS	SIGNATURE	DATE

I, ASE-0000, make the following free and voluntary statement to C. E. Murphy, who has identified himself to me as a representative of the U. S. Nuclear Regulatory Commission:

I am an ASE and senior licensed. On March 22, I was at home and received a call to come in at 2:30 p.m. I arrived at the plant at about 4:00 p.m., reported in, and went to the reactor side. I was at the shutdown board at the 1C elevation. People were going into the reactor by the drywell compressors. I went in one time using an air pack. I also put on an air pack to go into the fire area. SE-U was in there using water from a fire hose. Later, O-ZZ replaced him. O-ZZ and ASE-0000 came out together when their alarms went off. SE-U was getting the hose ready to start using water when ASE-0000 first got there. SE-U and ASE-0000 used it. O-ZZ and ASE-0000 used it. At about 6:00 or 6:30 they closed down the hose and came out and saw a team putting on Chem-ox units to go back in with--the Scotts were not lasting long enough. There was dense smoke. There was a faint glow that looked like it was confined to the trays. Initially, SE-U had told me what direction the fire was. Just after 6:30 when the Chem-ox units were used the fire was put out. Meals were brought in at 6:30. Afterwards, I worked in the control room verifying positions of valves on the RHR system for Unit 1 until I left at 11:00. During the time I was fighting the fire, AOS-W was at the control desk. ASE-L and ASE-Y were fighting the fire when I arrived. SE-U and SE-V were both fighting the fire.

I had training in the fire fighting in a training program about 11 years ago. I have had retraining in two separate weeks in fire fighting since coming to Browns Ferry. I had not known about the fire which occurred on 3/20. I had never seen the sealing operation or knew about the use of candles.

I have read the statement summarized above, which is true and correct.

/s/ C. E. Murphy	/s/ ASE-0000	4-28-75
WITNESS	SIGNATURE	DATE

I, ASE-PPPP, make the following free and voluntary statement to C. E. Murphy, who has identified himself to me as a representative of the U. S. Nuclear Regulatory Commission:

I am an assistant shift engineer. On 3/22 I was at home when I received a call at 1:15 p.m. saying there was an emergency fire and to come in.

I arrived at 1:50 and reported to SE-R who told me to help the boys get the turbine on the turning gear. I met O-LL and O-NN at the turbine building, who told me there was no power on the motor to turn the turbine. I then contacted APMS-FFFF to get the power back on. I stayed with the electricians until they go the boards energized on Unit 1. At about 3 to 5 p.m. SE-V needed mechanical vaccum pumps on Unit 2. I again worked with with the electricians until about 8:00 p.m. on this. One leg of the 250 volt DC was out (which furnished power to the board). When this was fixed then ASO-W had me get two people each for an hourly fire watch patrol on Units 1 and 2. I formed the patrols (one was AUO-UUUU and AUO-MMMM the other was AUO-AA and AUO-Z) and had them report to me after each patrol at the electrician's control desk. I logged each report and turned log into the shift engineer. At about 9:30 p.m. I entered the fire area with air pack to check on smoke presence.

I have had the fire training and refresher.

I have read the statement summarized above, which is true and correct.

/s/ C. E. Murphy	/s/ ASE-PPPP	April 22, 1975
WITNESS	SIGNATURE	DATE

I, EF-QQQQ, make the following free and voluntary statement to C. E. Murphy who has identified himself to me as a representative of the U. S. Nuclear Regulatory Commission:

I am an Electrician Foreman with TVA. I was in the warehouse when the fire alarm sounded on March 22. When the alarm went off I went to the Reactor Building, Unit 3, to check on my other men. When I got there, they had already assembled at the east side of the building. I got Elect.-XXX and Elect.-VVV from the warehouse and then I went home.

On Monday, Tuesday, and Wednesday, I supervised a crew working on Work Plan 2063. We were involved in inventorying grounds on various battery boards for Unit 1 and 2. All grounds that were found were related to Unit 1. Even when grounded circuits were found, we reconnected the cable leads to the battery board. On March 25th (Tuesday) we also worked on Work Plan 2920 which was related to removing cables from the 1B and 1C 480 volt Boards.

The battery boards are located on the 593' level of the Control Bay. While I was in those rooms I saw little evidence of smoke. The rooms had normal ventilation and normal lighting. I saw nothing to lead me to believe that the battery rooms or the equipment inside had sustained any damage.

I returned to my normal duties on Thursday.

/s/ C. E. Murphy	/s/ EF-QQQQ	4/16/75
WITNESS	SIGNATURE	DATE

I, ASE-RRRR, make the following free and voluntary statement to C. E. Murphy, who has identified himself to me as a representaive of the U. S. Nuclear Regulatory Commission:

I am an Assistant Shift Engineer and am presently in charge of Rad-Waste operations.

On March 22 I was in the tower cooling equipment when I called the Unit #2 operator who told me they had lost Unit #1 and had a fire in the control cables. I returned to the plant to see if I could help and reported to the shift engineer. I was assigned to call in, by phone, all supervisors and other plant personnel. It took me more than two hours to complete this. There were a lot of people in the control room--maybe up to 50 but they weren't in the way. There was a smokey haze in the room and office area. I left the plant then and came back on regular shift Monday. On Monday, we started with real short runs on the filters which I assume had been loaded up with soot. We back washed the filters. We received clean up water from the reactor building in 55 gallon drums. One or two of the drums registered above background.

I have read the statement summarized above, which is true and correct.

/s/ C. E. Murphy	/s/ ASE-RRRR	4/22/75
WITNESS	SIGNATURE	DATE

I, SE-SSSS, make the following free and voluntary statement to Howard A. Wilber, who has identified himself to me as a representative of the U. S. Nuclear Regulatory Commission:

I am a Shift Engineer, but my current duty is training coordinator. I first learned of the fire at home when the maintenance supervisor's wife called to ask what was going on - at about 3:30 p.m. on 3/22/75. I got to the plant about 4:00 p.m. and went to the control bay. SE-R was on shift. SE-V was on Unit 2, SE-PP was on Unit 1 and SE-U was fighting the fire. I attempted to assist in checking status of unit which was at ~200 psig at that time and blowing down to hold pressure. Unit appeared to be under control at that time. I left the control room and went to the area where fire was being fought to see if I could help SE-U, ASE-L and ASE-Y. On talking with SE-U and others we decided to get fresh air face units and Chem-ox units brought up to use in fighting fire.

I returned to control room and became aware of loss of air to relief valves and that they had closed. I went with SE-PP (in air packs) to look at Unit 1 drywell control air containment valve to see if we could get plant control air to valves. Containment valve prevented this. We went back up to control bay to electrical boards to get control bay air compressors on, but due to valve this didn't do any good. We then went to SE office and SE-PP got electrical print out to find containment valve power supply. APMS-FFFF came in and we requested his assistance in getting air to relief valves. Told him if necessary, run air line to diaphram to open valve.

I returned to control room and my supervisor asked me to go to Unit 2 to get it into shutdown cooling mode, which was accomplished at 10:45 p.m.

I left plant at 11:00 p.m. and returned Sunday morning at 7:00 a.m. I worked all day Sunday on lifting wires on RHR and CS systems MOV's which removed control from the control room bench board and also prevented any inadvertent operation. We used the approved JWIR procedure.

I left plant at 3:00 p.m. Sunday and returned at 7:00 a.m. Monday. On Monday and the rest of the week myself with two ASE's and two unit operators wrote procedures for each loop of RHR, C.S., DG's, and water level control and incorporated changes made by temporary connections, etc.

We also worked up a status book on Unit 1 systems.

I heard of a small fire in spreader room 3/20/75 on 3/21/75 but didn't know the details.

I know the leak test procedure used on penetrations but did not know until later that the sealing material was flammable.

I have read the statement summarized above, which is true and correct.

/s/ Howard A. Wilber	/s/ SE-SSSS	4/24/75
WITNESS	SIGNATURE	DATE

I,EE-TTTT, voluntarily give the following information to Howard A. Wilber, Tolbert Young, and James Devlin, who have identified themselves to me as representatives of the U. S. Nuclear Regulatory Commission:

I am an Electrical Engineer. I have been employed by the Tennessee Valley Authority since July 1972, assigned to the Browns Ferry Nuclear Plant, Division of Engineering Construction (DEC).

On Saturday 3/22/75, when the fire occurred, I was not working. At approximately 8:00 p.m. I was called at my home and told to report to work on the next shift. I was told that there was a fire in the cable trays which caused a lot of damage.

At 11:40 p.m. I came on the job and under the direction of EE-XXX Group Leader, began working up a "Work Plan". This "Work Plan," numbered 2920, is a list of the damaged cables, by identification number and valve number that would have to be lifted or by-passed with jumpers. Working on this job were EE-CCC, Electrical Engineer and EEA-000, Engineering Aide. After completing the Work Plan, we started to work the Work Plan.

Our purpose was to give local control and stablize systems, particularly to critical valves, RHR, core spray that the Shift Engineers decided they needed. Jumpers were also installed to give local control to motor operated valves without a backup control system on the MOV board. The work was in the 480 VAC MOV board 1-B, 1-C and the 25 VDC MOV board 1-B.

Work was started on four core spray valves 75-39, 75-30, 75-51 and 75-53. I worked until 12 noon Sunday.

The work was done according to a DPP procedure and the cables lifted were recorded by an ASE into the JIWR log. We tagged each terminal and lifted wire and the ASE also tagged the lifted wires and terminal points associated with each lifted wire with JIWR tags.

We had three crews out but only one (1) ASE, Assistant Shift Supervisor, and the work was going too slow. Later another ASE was assigned.

On Tuesday we completed lifting cables; the last items were completed between 3:00 p.m. to 6:30 p.m.

/s/ William S. Little	/s/ EE-TTTT	4/16/75
WITNESS	SIGNATURE	DATE

I, AUO-UUUU, voluntarily give the following information to Donald Caphton, Tolbert Young, and James Delvin, who have identified themselves to me as representatives of the Nuclear Regulatory Commission:

I am an Assistant Unit Operator and have been employed by TVA since May 1970. Since October 1970 I have been employed by DPP, and have been at Browns Ferry since 4/29/75.

I was called to work the evening shift at 12 noon. I arrived on site at 2:30 p. m. and went to the control room. I was given a number of RHR valves to be opened or closed. I worked on this about 4 hours.

AOS-W sent me to work for SE-U at the reactor building entry. Worked there for an hour helping with air bottles, etc. Worked on fire watch the rest of the night with other AUO.

The maximum number of people I saw in the control room was about 20 to 30.

/s/ William S. Little	/s/ AUO-UUUU	4/12/75
WITNESS	SIGNATURE	DATE

I, CNGB-AB, make the following statement freely and voluntarily for Him Hufham who has identified himself to me as a representative of the U. S. Nuclear Regulatory Commission:

I am Chief, Nuclear Generation Branch, for the Tennessee Valley Authority. At about 2:40 p.m. (CDT) on March 22, 1975, I arrived at my house and received a message about a fire at the Browns Ferry Nuclear Plant (BFNP). I immediately called P-Supt., Plant Superintendent at BFNP, and was briefed on the status and shutdown cooling problem involving both reactors. At 2:45 p.m. (CDT) I called my assistant, ACNGB-AE, at the Edney Building and informed him that I had decided to establish the Central Emergency Control Center (CECC) and to activitate the Radiological Emergency Plan (REP). I also asked ACNGB-AE to begin notifying certain key individuals to report to the CECC. I departed my house at 2:50 p.m. (CDT) and arrived at the CECC at 3:10 p.m. Several people started arriving at the CECD and among them, DCECC-AC who reported at 3:35 p.m. (CDT) and became the Director of the CECC. At 3:40 p.m. (CDT) P-Supt. reported that the fire was out on the cable spreading room building had no area monitoring. He also related that the breathing air was in short supply; therefore, I ordered 50 cylinders from the Selax Company in Chattanooga to be delivered to the site. At 4:33 p.m. (CDT), P-Supt. reported that the fire was out on the cable spreading room side and gave some indication that the natural draft, caused by the negative pressure in the reactor building, was aggravating the fire. I made initial report of plant conditions to Long (NRC) at 4:45 p.m. At 5:10 p.m. (CDT), P-Supt. called and asked permission to use water on the fire. He stated that the fire had been determined to be an "A" type and the fire chief from Athens, Alabama, had advised him to use water. I called P-Supt. back at 5:20 p.m. (CDT) and informed him that it was okay to use water but under controlled conditions. At 6:06 p.m. (CDT)I talked with SE-V, a shift engineer at BFNP, who told me that enough heat had been generated in the cable trays to cause recombustion of the fire even though it could be put out temporarily with CO_2 and dry chemicals. At 6:09 p.m. (CDT), P-Supt. reported that he had not used water as yet and stated that he wanted to hold off another thirty minutes before using water. At this time, he also indicated that "construction had candled the penetration and it flashed". I could have ordered the BFNP personnel to use water but I felt that the Plant Superintendent and his people on site were in a better position to evaluate the situation. The plant considered the cables extremely important, and the use of water might short circuit wiring in the fire zone and deprive the reactor of even more controls that would be required for shutdown cooling. Additionally, the Plant Superintendent did not wish to use the services of the Athens Fire Department because he was afraid that injuries might occur. At 6:50 p.m. (CDT), P-Supt. notified me that he had created a task force to determine the cables involved and a course of action to be taken. He also stated that he was now using water to fight the fire (6:50 p.m. CDT). At 7:47 p.m. (CDT), P-Supt. informed me that the fire was out. At 8:27 p.m. (CDT), P-Supt. informed me that they were quite sure that a candle, used to check penetration leaks, had ignited the polyurethane sealant and started the fire. At 9:00 p.m. (CDT), I again called F. Long, (NRC) and informed him of the plant status. The conditions at the plant improved as necessary systems were reactivated. I reported these improved conditions to F. Long (NRC) at 10:15 p.m. (CDT). At 11:00 p.m. (CDT), I left the CECC for the night. Two DPP personnel were instructed to remain in the DPP Emergency Center for the rest of the night and to notify me if anything serious or unusual occurred. The CECC was reactivated at approximately 8:00 a.m., March 23, 1975, and remained manned continuously from about 3:10 p.m. March 22, 1975, to about 5:00 p.m. on March 23, 1975.

Signature /s/ CNGB-AB

Witness /s/ BAH

Date 5/19/75

I, DCECC-AC, make the following statement freely and voluntarily to James W. Hufham who has identified himself to me as a representative of the U. S. Nuclear Regulatory Commission:

I am the Assistant to the Director of the Division of Environmental Planning for the Tennessee Valley Authority. I was notified at my home on March 22, 1975, at 4:05 p.m. (EDT) that a fire had been reported at the Browns Ferry Nuclear Plant. ACNGB-AE was the individual who initially informed me of the fire and of the activation of the Central Emergency Control Center (CECC). From my home I immediately called DEEC-AD and instructed him to call the State of Alabama Department of Public Health. I arrived at the CECC at 4:25 p.m. (EDT), and upon my arrival I was briefed on the status of the reactors by CNGB-AB. At this time there was no indication of a release and the main problem was controlling the reactors and putting the fire out. I recontacted the Environs Emergency Center at 4:44 p.m. (EDT) and related the general information to HP-AG. In this conversation I repeated again to alert the State of Alabama but to indicate there had been no release. At 5:03 p.m. (EDT) I called the U. S. NRC, Region II, and informed the answering service about the fire problem. I stated my name and the name of CNGB-AB to ASR-AQ the answering service representative, who stated that the U. S. NRC would return a call in ten minutes. I provided ASR-AQ with the number (615) 755-2495 for the return call. After completing the attempt to notify the U. S. NRC, I realized that I was the Director of CECC and assumed the responsibilities of the position. I talked with DRH-AN at 5:12 p.m. (EDT) and reported the general information to him. In this conversation with DRH-AN, he informed me that he had notified the State of Alabama Civil Defense Department at 3:45 p.m. (CDT) and requested that the agency stand alert. DRH-AN also indicated that the State may initiate environmental air monitoring around the site. At 5:25 p.m. (EDT) I made another attempt to notify the U. S. NRC, Region II, and in this attempt I talked with ASR-DB of the NRC answering service. I related the fire problem again to the answering service representative, who stated that she would call Mr. Charles E. Murphy, NRC, Region II, Atlanta, At 6:30 p.m. (recontacted DEEC-AD and requested that an industrial hygienist be dispatched to the plant. DEEC-AD also reported air concentrations at the gatehouse and elaborated that levels were approximately equal to background. DEEC-AD continued the conversation and stated that the environmental staff decided to evacuate the environmental station and proceed to the gatehouse because of an odor. The environmental staff manned the environmental station again after an evacuation period of approximately thirty minutes. I returned a call to the Environs Emergency Center at 6:55 p.m. (EDT) and received the information from HP-AR that the meterorological conditions were Class A (stable) and any release should come down. DEEC-AD reported the results of air samples that had been collected and analyzed from inside the plant, and confirmed that the reactor ventilation system was off. I continued the conversation by instructing IE-AS, Industrial Hygienist, of my concern for workers breathing toxic vapors and fumes. AT 7:07 p.m. (EDT) DEEC-AD informed me that no suspected employee was contaminated with radioactivity and respirators were being used. He also elaborated that the environmental staff was dispatched to monitor sites farther away from the plant. At 7:55 p.m. (EDT) I received a call from DRH-AN to whom I reported the latest environmental conditions. DRH-AN questioned why he was not informed for two hours after the fire. I related to him that he would not be notified for just any fire but only when activating the REP. AT 8:30 p.m. (EDT) DEEC-AD informed me that the ventilation system for Unit 1 and possibly Unit 2 was on and that manual/random checks would be made to determine any release. At a time that was not recorded, I was informed that the fire was out. Somewhat after 8:45 p.m. I received a call from IE-AS who confirmed there was no phosgene gas in the control room and the level of carbon monoxide was not hazardous. I reported the fire incident to ADDRH-AO. Director of Radiological Health for the State of Tennessee, at 9:05 p.m. (EDT) At 10:30 p.m. (EDT) DEEC-AD reported to me that all constant air monitors for Unit 2 were operable but the ones for Unit 1 were not. At 11:20 p.m. (EDT) IE-AS reported from the plant that carbon monoxide in reactor buildings 1 and 2 was 500 ppm and I directed him to remain until carbon monoxide dropped to a safe level. At 11:30 p.m. (EDT) I secured the CECC for the night. Two representatives of the Divison of Power Production took charge of their Emergency Center. All CECC participants left with these two individuals their telephone numbers where each could be reached during the night.

On March 23, 1975, I returned to the CECC at 8:10 a.m. (EDT) and was briefed by CNGB-AB and ACNGB-AE on the activities that had occurred at the plant during the night. At 9:05 a.m. DEEC-AD reported the maximum release concentration that had been reported during the fire and provided me with a summary of dose calculations. From 9:11 a.m. until approximately 10:50 a.m. (EDT) I continued to receive chemical, radiological, and environmental data from the Environs Emergency Center. At 10:53 a.m. (EDT) I contacted DMS-AU, Director of Medical Services, about the construction workers who participated in fire fighting but were not examined by the medical staff on March 22. At 11:34 a.m. (EDT) I talked with DRH-AN who related that the Governor of Alabama had questioned any sabotage indications. There were communications between the CECC and TVA personnel for the remaining afternoon. I departed the CECC at 4:15 p.m. (EDT).

TVA's Radiological Emergency Plan and the activation of the CECC were effectively implemented on March 22 and 23.

/s/ James W. Hufham	/s/ CECC-AC	May 22, 1975
WITNESS	SIGNATURE	DATE

I, DEEC-AD, make the following statement freely and voluntarily to James W. Hufham, who has identified himself to me as a representative of the U. S. Nuclear Regulatory Commission:

I am Supervisor, Health Physics Staff for the Tennessee Valley Authority. I was notified by DCECC-AC at 3:00 p.m. (CDT on March 22, 1975, that a fire had been reported at 12:35 p.m. (CDT) at the Browns Ferry Nuclear Plant and an emergency had been declared. I notified CECC-AF at 3:05 p.m. (CDT, and at 3:15 p.m. (CDT) as Environs Emergency Director I manned the Environs Emergency Control Center in the River Oaks Building at Muscle Shoals, Alabama. I called DRH-AN at 3:22 p.m. (CDT) and informed him of the current status of the Browns Ferry Nuclear Plant. After notifying the State, I recontacted the Central Emergency Control Center and talked with ACNGB-AE and informed him that the State had been notified. He related that both reactors had been scrammed, the fire was limited to the Unit 1 cable spreading room, and circuitry was lost to Unit 2 relief valves. I telephoned HDB-BH Hydraulic Data Branch, at 3:45 p.m. (CDT) who later manned the meterological tower (environmental station) at 4:15 p.m. (CDT). I contacted HPS-T at 4:20 p.m. (CDT) who related to me that all area, air particulate, and effluent monitors for the Unit 1 and Unit 2 reactor buildings had been lost. At 4:35 p.m. (CDT) I requested that air samples be taken as close to the exclusion area fence as possible in the downwind direction. Also at this time the met tower instrumentation indicated wind conditions of 5 mph at 300° NW. I was informed at 5:05 p.m. (CDT) that a visible smoke release was coming from the reactor building. It was indicated that the smoke was moving directly toward the met tower and the met tower facility was being evacuated until such time that it could be determined whether a problem existed. The met tower facility remained unmanned until 5:34 p.m. (CDT). At 5:25 p.m. (CDT) I telephoned four industrial hygienists to assist with the carbon monoxide problem that had developed in the plant. I received the information at 5:45 p.m. (CDT) that the Units 1 and 2 reactor building ventilation systems were lost. At 6:05 p.m. (CDT) I requested that the environmental monitoring team move away from the plant and collect air samples at distances further downwind. At 6:30 p.m. (CDT) I received a call from CECC-AF that informed me that the wind direction at this time was 2 mph, east of north. Soon after 6:30 p.m. (CDT) I received a call from HP-BJ who indicated that the radioactivity in the control room was increasing and some of the individuals in the control room did not have respirators. HP-BJ also stated that no one working in the Units 1 and 2 reactor buildings appeared to be externally contaminated but that everyone was covered in soot. In this conversation, HP-BJ requested that whole body counting be performed on individuals after the fire was over. At 6:50 p.m. (CDT) I received a call from HP-BJ who related that the turbine building air activity was increasing and provided me with a summary of results from air samples taken in the turbine building. In this conversation, HP-BJ related that the release rate from the reactor building roof vent from Unit 1 was less than 2000 microcuries/second and that the equivalent concentration was less than the sensitivity of the analytical instrumentation in the chemical laboratory. At 7:35 p.m. (CDT) HP-BJ informed me that the reactor building ventilation systems for Unit 1 and Unit 2 were to be activated at 7:40 p.m (CDT), but he later reported that Unit 2 vent fans would not run. At 8:00 p.m. (CDT) NE-BK calculated a site boundary dose of 1.16 millirem/hour in the NW sector using the maximized release rate of 2000 microcuries/second as the release rate. At 8:20 p.m. (CDT) HP-BJ reported that all building CAM's were beginning to go down. PSO-BL contacted me at 8:39 p.m. (CDT) and reported that he could not contact the gatehouse to inform the security guards that the warning lights were out on the plant stack and that FAA should be notified. I informed him that I would forward this information to the plant and request that it be submitted to the CECC. At 9:52 p.m. (CDT) I reported to the environmental staff at the plant that the fire had been extinguished at 8:45 p.m. (CDT). At 9:30 p.m. (CDT) HP-BJ contacted me and stated that the Unit 2 vent monitor had indicated a 6000 cpm spike when the monitor was returned to service but later returned to 600 cpm which was essentially the background count before the fire. I talked with CECC-AF at 10:20 p.m. (CDT) and informed him that Unit 2 reactor vent was operating and indicating essentially no release. I also informed CECC-AF that grab samples were being taken from Unit 1 vent, but I did not have any results at this time. At 10:35 p.m. (CDT) I recontacted CECC-AF and informed him that the CECC had requested that the met tower be manned throughout the night. At 10:58 p.m. (CDT) I telephoned CECC-AF and provided information on the present status of the reactor that I had received from the CECC. I informed him that all cooling to the reactor was lost because the RHR, HPCI, and core spray were inoperable; however, this had not been confirmed. I also informed him that feedwater was being provided through the control rod drive mechanism and the relief valves were being used to relieve pressure. AT 3:00 a.m. (CDT) on March 23, 1975, the met tower was secured, and at 4:10 a.m. (CDT) I was informed that the RHR system was in operation. At 4:15 a.m. (CDT) I secured the Environs Emergency Center and left the center at 5:15 a.m. (CDT). The Environs Emergency Center was manned again at 8:05 a.m. (CDT) on March 23 and contact made with the CECC staff. It was decided that the Environs Emergency Staff was no longer needed, and the Environs Emergency Center was secured at 10:00 a.m (CDT) on March 23.

/s/ James W. Hufham	/s/ DEEC-AD	5-23-75
WITNESS	SIGNATURE	DATE

I, CDC-AH, make the following free and voluntary statement to Michael V. Annast, who has identified himself to me as a representative of the U. S. Nuclear Regulatory Commission:

I am the Civil Defense Coordinator for Limestone County, Alabama, and was acting in that capacity on March 22, 1975.

I heard about the fire at Browns Ferry on the morning of Monday, March 24, 1975. To the best of my knowledge, no one in the Civil Defense system notified me or attempted to do so. I feel that our county should have been the first to be notified since we are the closest to the plant and it is located in our county, about 10 miles from Athens, Alabama. In my opinion, the emergency plan needs updating and revision.

/s/ Michael V. Annast	/s/ CDC-AH	4-29-75
WITNESS	SIGNATURE	DATE

I, AFC-AJ, make the following free and voluntary statement to Michael V. Annast, who has identified himself to me as a representative of the U. S. Nuclear Regulatory Commission:

I am the Fire Chief of Athens, Alabama, and was acting in this capacity on March 22, 1975. At 1:09 p.m., the Athens Fire Department received a call from the Browns Ferry plant requesting assistance to put out a fire inside the plant. I received a call at my house at about the same time and proceeded immediately to the plant site in my car. I arrived there at about 1:25 and observed the fire truck arriving from Ashens, right behind me. We were promptly processed in and were ready to fight the fire shortly after 1:30 p.m.

My department has 14 paid firemen. During the fire, I had six men and one fully-equipped 1,000-gallon pumper truck on the plant site at all times. I was aware that my effort was in support of, and under the direction of Browns Ferry personnel, but I did recommend, after I saw the fire in the cable spreading room, to put water on it. P-Supt., Plant Superintendent, was not receptive to my ideas because he felt that water should not be used on electrical wires and they could not determine which wires carried current. I informed them that this was not an electrical fire and that water could and should be used because the CO_2 and dry chemical were not proper for this type of fire. The problem was to cool the hot wires to prevent recurring combustion. CO_2 and dry chemical were not capable of providing the required cooling. Throughout the afternoon, I continued to recommend the use of water to P-Supt. He consulted with people over the phone, but apparently, was told to continue to use CO_2 and dry chemical. Around 6:00 p.m., I again suggested the use of water and offered the services of my department to include special gloves used for high voltage and rubber boots. P-Supt. finally agreed and his men put out the fire in about 20 minutes. As I recall, the fire was completely out by 7:30 p.m. My men and equipment cleared the plant site at 9:51 p.m.

During the fire, my department furnished five MSA masks and shuttled air supplies in cascade tanks, which were continuously being filled at our air compressor. We did this because the plant had a shortage of air supplies for their air packs. We also furnished one A-B-C portable extinguisher to plant fire fighting personnel.

I wish to state that the men fighting the fire worked very hard, they were calm and orderly and were following proper procedures. The only thing that was wrong, in my opinion, was the fact that they were using type B and C extinguishers on a type A fire and that the use of water would have immediately put the fire out.

/s/ Michael V. Annast

/s/ LHM	/s/ AFC-AJ	4-28-75
WITNESS	SIGNATURE	DATE

I, SLC-AK, make the following free and voluntary statement to Michael V. Annast, who has identified himself to me as a representative of the U. S. Nuclear Regulatory Commission:

I am the Sheriff of Limestone County and I was acting in that capacity on March 22, 1975.

I heard about the fire at the Browns Ferry plant after it was over. We did not receive a call for assistance, although, I understand that the Athens Fire Department was called. I am aware of my responsibilities in the case that evacuation would be required. My department has outside speakers on the patrol cars and is familiar with the evacuation area and procedures. I have not had any updating of procedures proposed to me since the initial plan was outlined in 1972. I do not have a copy of the emergency plan, however, the Civil Defense Coordinator does have a copy available.

/s/ Michael V. Annast	/s/ SLC-AK	4-29-75
Witness	Signature	Date

I, CDC-AL, make the following free and voluntary statement to Michael V. Annast who has identified himself to me as a representative of the U. S. Nuclear Regulatory Commission:

I am the Civil Defense Coordinator for Morgan County, Alabama, and was acting in that capacity on March 22, 1975. I heard about the fire at the Browns Ferry plant at about 1:00 p.m. over the police radio. I ascertained that there was no release of radioactive materials and stayed in contact to keep abreast of the events. At about 4:05 p.m., CDDO-BT, State Civil Defense Duty Officer, called me and asked me what was going on. I told him that there was no radiation and he gave me his home phone number, should he need to be notified.

It is apparent from the times reported that the notification did not work as outlined in the plan. The plan requires revision and updating of phone numbers. I feel that the State Health Department could be more active in this area. The State Civil Defense people had been invited to the Tri-County area for the purpose of coordination and updating the response capability.

The people of Morgan County are concerned about the fire and possible future incidents. I feel that a planning session is necessary to insure that prompt notification is received and that the efforts of all agencies are coordinated as required by the plan.

/s/ Michael V. Annast	/s/ CDC-AL	4/29/75
WITNESS	SIGNATURE	DATE

I, SMC-AM, make the following statement freely and voluntarily to Michael V. Annast, who has identified himself to me as a representative of the U. S. Nuclear Regulatory Commission:

I am the Sheriff of Morgan County, Alabama, and was acting in that capacity on March 22, 1975.

I heard about the fire at about 4:50 p.m. from CDMCCH. I was asked to keep quiet about the incident to avoid any panic. I was elected to my present office in January 1975. I do not have a copy of the emergency plan and I have had no contact with any agency for coordination. I wish to state that I stand ready to assist any evacuation efforts with all available men and vehicles. I feel that any notification should come directly to me and not through other local agencies.

/s/ Michael V. Annast	/s/ SMC-AM	4/29/75
WITNESS	SIGNATURE	DATE

I, DRH-AN, make the following free and voluntary statement to James W. Hufham, who has identified himself tome as a representative of the U. S. Nuclear Regulatory Commission:

I am the Director of the Division of Radiological Health for the State of Alabama Department of Public Health. I received a telephone call at 3:20 p.m. March 22, 1975, DEEC-AD, who at the time, was at the TVA Environs Emergency Center in Muscle Shoals, Alabama. DEEC-AD told me that there was a fire at the Browns Ferry Nuclear Plant (BFNP) and provided me with reactor status and existing monitoring capabilities. I concluded that the situation was serious and felt that the State of Alabama Emergency Plan should be activated. Although there was no immediate radiation problem, I felt that air sampling in the vicinity of the plant should be started. I immediately notified DEH-BS, Director of Environmental Health Administration, State of Alabama Department of Public Health, who asked me to inform the State Health Officer, SHO-CE. I tried to contact SHO-CE at 3:40 p.m., but he was not available. At 3:45 p.m., I telephoned CDDO-BT, State of Alabama Civil Defense Duty Officer, and informed him of the fire with the conclusion that there would be no action required at this time. I then notified HO-BU, the Tri-Counties Health Officer in Decatur, Alabama, and DEH-BV, Environmental Health Laboratory Director for State of Alabama. During my discussion with DEH-BV, we consideration activation of air samplers around the plant site but he suggested that we reconsider the situation prior to actual start of the samplers. At 4:10 p.m., I contacted CECC-AC, Director of the Central Emergency Control Center for TVA in Chattanooga, Tennessee, who told me that both reactors had been scrammed, many reactor indication monitors were out and the status of the vent monitors was uncertain. He also related that the fire was still burning, CO_2 extinghishers were being used and core cooling was being provided by alternate means. The location of the fire was below the control room in the cable spreading room; and the Athens (Alabama) Fire Department was on site. Based on this information, I concluded that the situation was more serious than I had initially believed and that air sampling was imperative. CPPO-BT, Civil Defense, reported to me at 4:20 p.m. that he had notified the Morgan County Civil Defense Coordinator. I informed AUO-BBB that the situation was serious but that radiation levels did not warrant Civil Defense action at this time. We agreed that, should evacuation become necessary, the law enforcement personnel would be contacted. I called the Laboratory Director at 4:30 p.m. and he agreed to start air sampling around the site. I also called DEH-BS back and advised him of the current status. He agreed with my conclusion and decided to notify the Governor of Alabama, by contacting his staff. Attempts were made to contact two staff members at 4:45 p.m. and additional attempts until 9:15 p.m. however, all were in vain. I called DCECC-AC at 5:15 p.m., and he told me that Unit 2 reactor was in normal shutdown process but Unit 1 had fire problems. He also stated that the fire was burning but was under control and water had not been used. No injuries had been reported, the cable spreading room had not been entered and no reading was available from the reactor vent monitors. I questioned the ability of the remaining systems to provide adequate core cooling and DCECC-AC frferred me to ACNGB-AE who explained that core cooling for

Unit 1 was being maintained by the condensate booster pumps but the ability to operate relief valves from the control room was lost. Further, the HPCI, RHR and some of the ECCS was lost, the torus temperature was 126°F with no leakage indicated. I concluded that the cooling system required attention, more indication data was needed to confirm monitoring ability and main steam line valves condition. I felt that, although serious, the situation did not warrant activation of the Southern Radiological Emergency Plan. At 5:45 p.m., the laboratory director reported that all air samplers, except in Decatur, had started. The latter was inoperable, possibly due to a defective wind directional control system. I requested that the control system be removed and the sampler started. I attempted to contact NRC, Region II at approximately 5:15 p.m. and later prior to 9:30 p.m. but on each occasion talked with the answering service. At 6:50 p.m., DCECC-AC called me and stated that the environmental indication measurements around the plant, at the gate house and the site boundary were essentially background. The wind direction was from the northeast at 5 MPH. The fire condition had not changed. I recommended to DCECC-AC that the State of Tennessee be notified of the situation. At 7:50 p.m., the laboratory director reported that no nuclear air sampler was available in Decatur. I immediately contacted the State Air Pollution Control Commission who activated an air pollution sampler in Decatur at approximately 9:00 p.m. I called the TVA CECC at 9:15 p.m. and was informed that the fire had been extinguished at approximately 7:45 p.m. using water fog. The air monitoring equipment was operational and was indicating a decrease in radioactivity. The vent monitoring was being done manually. Unit 1 relief valves were operating again and pressure was dropping but weeks would be required for decay heat removal. I felt at this time that the situation had improved somewhat and talked to the State Health Officer at 9:45 p.m. He told me that the Governor had been informed of the situation. N. C. Moseley, Director, Region II, U. S. NRC called me and also indicated that the situation seemed to be improving. On March 23, 1975, I continued receiving air sampling results from DCECC-AC. On March 24, 1975, I remained on contact with Alabama, Tennessee, and TVA personnel and kept them informed. On this date, I also made the decision to collect water samples below the BFNP site and milk samples from selected dairies. I also requested that TLD's be collected from around the site and their results be compared with those stationed throughout the State. Since the fire, I have briefed members of the Governor's staff, the State Radiation Advisory Board, and the State Board of Health.

Signature: /s/ DRH-AN

Witness: /s/ James W. Hufham

Date: May 1975

I, AO, make the following statement, freely and voluntarily, to James W. Hufham, who has identified himself to me as a representative of the U. S. Nuclear Regulatory Commission:

I am Assistant Director, Division of Occupational and Radiolgoical Health for the State of Tennessee Department of Public Health. At 8:15 p.m. Saturday night, March 22, I received a call from CECC-AF, Central Emergency Center, TVA Edney Building, Chattanooga, reporting a fire at the Browns Ferry Nuclear Plant in Alabama. He reported that Units 1 and 2 were scrammed and that most of the ECCS was lost due to the fire and that the units were being cooled by booster pumps. Also that monitors located around the site had detected no loss of radioactivity. DCECC-AC came on the phone shortly and said that there was a fire in the cable tray room, which had "wiped out Units 1 and 2." Also that the first and second alternates to core cooling were gone and they were presently on the third alternate. There is one alternate left which is to bring in river water and circulate it to and from some ditches for cooling. He related that smoke was everywhere. The following details were also provided:

(1) Reactor number one was operating at a power level of 1100 MWe at the time of the scram.

(2) Both reactors have their initial fuel loading.

(3) The U. S. NRC and State of Alabama had been notified of the fire which had begun around 1:30 p.m. CDT on March 22.

(4) When questioned about radioactivity and pressure in the containment structures, he said that this was unknown due to monitors not being functional.

(5) Wind at the time of his call was blowing from NW to SE and was of stable Class A condition.

(6) Stack releases were normal.

(7) Radioactivity levels were quoted as follows: Turbine room 2.5×10^{-7} microcurie per cc. Control room number one 7.5×10^{-8} microcurie per cc; Control room number two 3.56×10^{-7} microcurie per cc.

At 8:35 p.m., I placed a call to the home of DRH-AN, Director of Radiological Health in Alabama, phone number . He related that the fire was not out, at least 40 minutes ago, and that he had alerted his staff and other governmental officials in Alabama. He provided the following additional information:

(1) Unit 1 went critical August 1973 and into commercial production in January 1974.

(2) Unit 2 has been critical since October 1974 and is still being tested.

(3) The main problem at present time is toxic fumes.

(4) That the situation was condition D as far as radiation is concerned.

After talking to DRH-AN and securing all the facts available, I called DRH-BX at 8:50 p.m. and briefed him on the situation. Also, our staff and State Civil Defense were contacted with out staff alerted to stand by for possible action in case the situation worsened by the release of radioactive material to off-site areas; 9:15 p.m. contacted DCD-BY, State Director of Civil Defense; 9:25 p.m. PHY-BZ of Chattanooga; 9:30 p.m. PHY-BP, Oak Ridge, instructing him to inform PHY-CA; 9:35 p.m. PHY-CF instructing him to inform his staff; 9:45 p.m., DRH-BX called me saying he had been in contact with DCECC-AC, TVA, and that we were to continue to stand by until further notice; 9:55 p.m. PHY-CB was not at home; 10:15 p.m. PHY-CD; 10:20 p.m. AT.

Signature: /s/ AO

Witness: /s/ DVB

Date: June 2, 1975

I, P-Supt., voluntarily give the following information to D. Caphton, T. Young, and J. Devlin, who have identified themselves to me as representatives of the U. S. Nuclear Regulatory Commission

I have been employed by the Tennessee Valley Authority since 1961 and assigned to Browns Ferry Nuclear Plant as Plant Superintendent for over five years.

On Saturday, March 22, 1975, at approximately 12:51 to 12:55 p.m., I was notified by phone of the fire and that Unit 1 had been tripped. By the time I arrived, Unit 2 had also been tripped. As senior member of management on site, I assumed overall direction and control. This does not mean that I directed each specific action on each unit, although the decision to blow down the reactors was mine and had the concurrence of the Operations Supervisor, OS-DD. Administrative Officer, AP, and two female clerical employees were in the plant. They assisted initially by calling in additional personnel; later in the day, they assisted in conducting a personnel accountability check.

Being unable to reach my supervisor, CNGB-AB, Chief, Nuclear Generation Branch, I called DPP-BW, Division Director, and advised him of the situation shortly after my arrival in the plant. At approximately 2:40 p.m., CNGB-AB called the plant and we determined that the TVA Emergency Plan should be activated. At 3:10 p.m., the emergency center was manned; and, subsequently, radiation monitors dispatched, NRC and the State of Alabama notified.

Initial efforts to extinguish the fire relied entirely on the use of dry chemical and CO_2 fire extinguishing equipment. This is consistent with the TVA training and practice concerning fire in or near energized electrical equipment. Primary considerations were safe reactor shutdown and personnel safety. It must be remembered that we did not know exactly the power or control circuits involved in the fire; and, as it turned out, several of the circuits we used in the shutdown came through the fire zone. Sometime during the afternoon, AFC-AJ, Athens Fire Department, recommended to me the use of water. I called the CECC and discussed this recommendation and was reluctantly told that it was my decision if I considered it necessary. I reviewed this with SE-U, Shift Engineer, who was in charge of the fire fighting effort at that time. He advised that water was not necessary since the fire had been put out. Together, we crawled into the spreading room and looking through the penetration could see that the fire still burned. At that time, we determined to try dry chemical and CO_2 one more time. If the efforts were not successful, we would use water.

The control room was never abandoned. Following use of CO_2 in the spreader room, smoke entered the control room, but it was never severe enough to cause discomfort to me. Although there was a haze, visibility was such that I could see the length of the control room quite clearly. Someone brought an air hose into the control room. I concurred with AGCS-WW (TVA Construction) suggestion to set up a fan in the control bay entrance. I do not know if it was used, and I do not specifically know if the control room ventilation system was on.

PSO-CC came to the control room several times and talked with me and others. He is the Public Safety Officer Fire Marshall and was assisting in fighting the fire. I do not know his specific actions at all times. When the decision was made to use water on the fire, it was contrary to his recommendations.

At the time of the Saturday fire, I was not aware of a fire involving the same or a similar material which occurred on the previous Thursday. The event had been logged by the shift engineer, and I understand it had been brought to the attention of the Assistant Plant Superintendent, AS-GG. I am aware of the practice of using candles to test for air leaks such as around condensers, but it never occurred to me that these penetrations were being tested in this manner. I was aware that polyurethane was flammable but was under the impression that neither completed penetrations nor cables in trays would propagate a fire. At no time was I concerned that we would have a criticality accident or that there was loose radiation which, if released, would endanger the public.

There were no reported injuries to personnel except a few cases of minor smoke inhalation.

The work plan under which employees were working at this time had been approved by DPPC-BM, a Senior Reactor Operator, who serves as coordinator with construction activitites. He is authorized to make the decision and approve non-safety-related work plans. The work plan itself was misleading in that it required checking a number of penetrations for leaks and did not appear to be of safety significance.

The fire was reported out in the spreading room at 4:20 p.m., and all fires were reported out at 7:45 p.m.

/s/ C. E. Murphy	/s/ P-Supt.	5/6/75
WITNESS	SIGNATURE	DATE

BROWNS FERRY NUCLEAR GENERATING PLANT

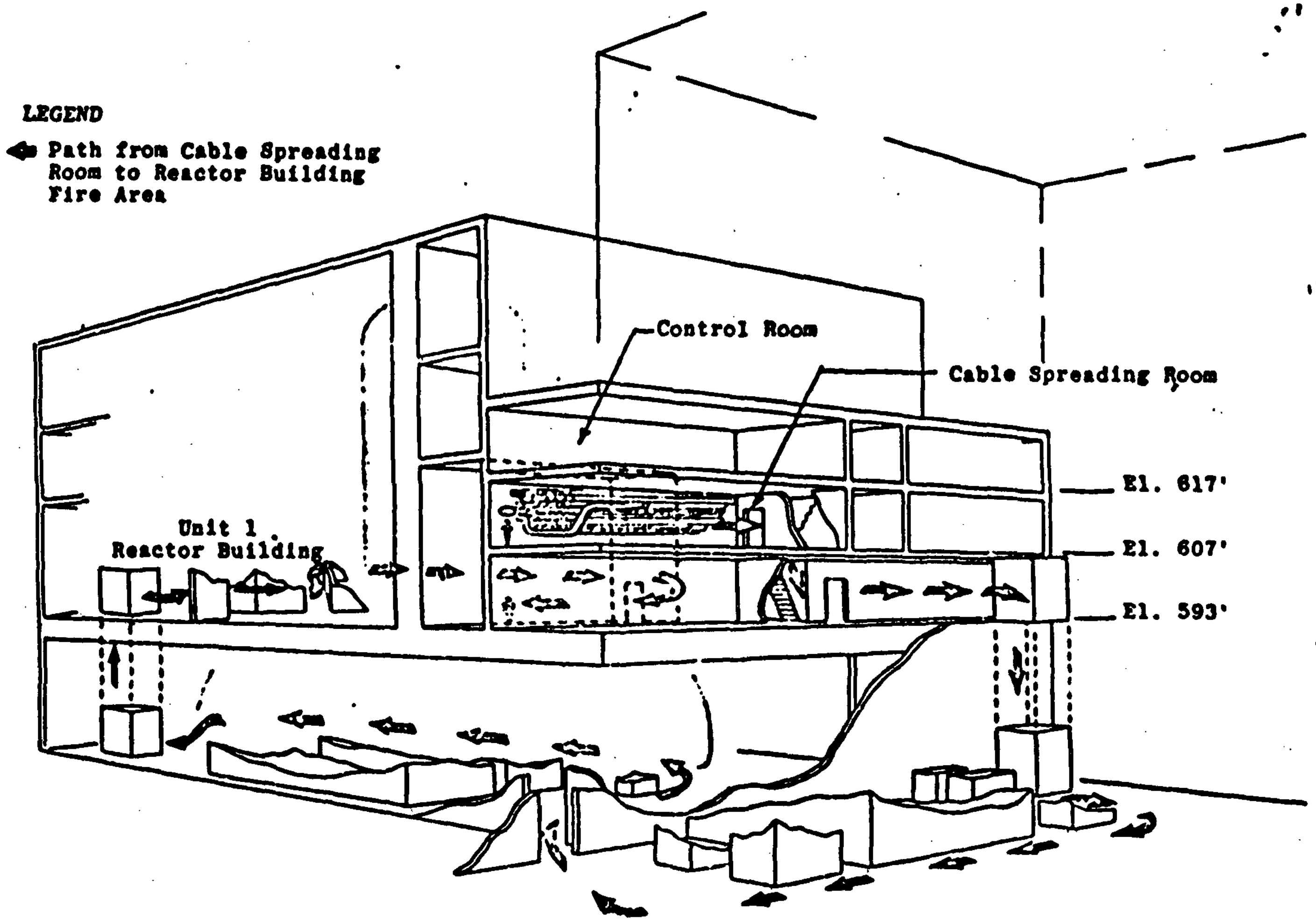

Exhibit A1
Page 1 of 2

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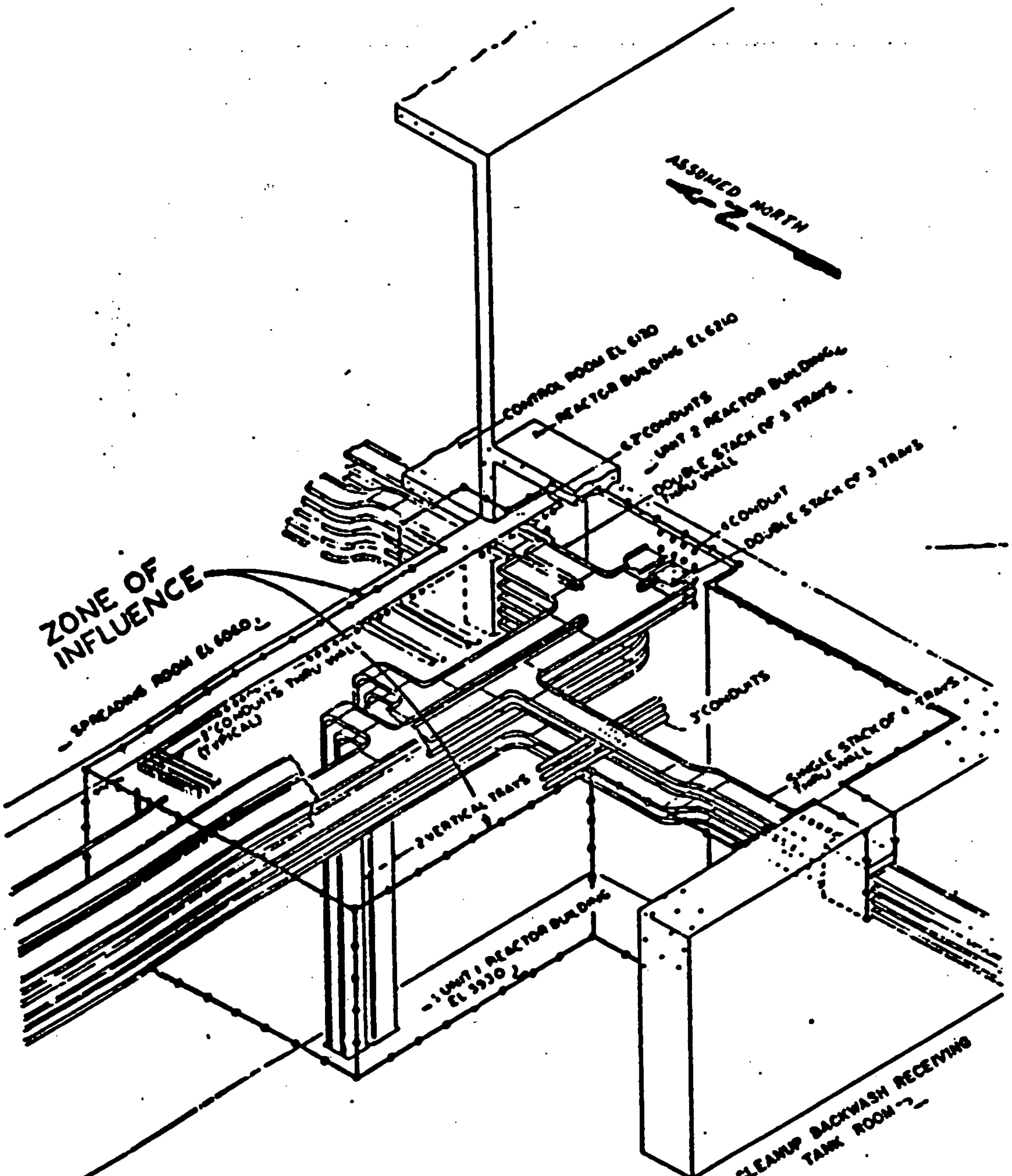


FIGURE 6

Exhibit A1
Page 2 of 2

WP-104, Rev. 2
Attachment 1

BROWNS FERRY NUCLEAR PLANT
WORK PLAN

Requested By: [illegible signature] Work Item No. N/A Work Plan No. 3592
Unit No. 1&2
Date: 3/7/75
Equipment or System: [illegible]
Drawings Associated with This Work: N/A

System Transferred: Yes ✓ No ___ Retest Required: Yes ✓ No See WP 3537
ECN No.: N/A FDI or FDDR No.: N/A MCR No.: N/A Other: [illegible]
Work Location: Reactor Bldg
Description of Work: Check Electrical & Mechanical Sealing for secondary containment (1) Make a punch list of pipes & cable penetrations that require sealing (2) Complete sealing (3) Verify & sign off areas that need [illegible]
Estimated Duration: 7 days Proposed Date: Begin 3/7 End [illegible] 3/15/75
Clearances Required: None

DDC Coordinator: [illegible signature]

*Authorized by the Plant Superintendent, Modification Coordinator:
Precautions:
DPP Requirements & Comments:
DPP Coordinator: Gary L. Denton Date: 3/7/75

Verifications:
Field Work Complete: Bil Pass Jr. Date: 3/26
Retest Complete (If Required): Date:
Drawing Requirements ___ Yes ___ No
Submitted to DCC w/Changes
Submitted to DCC No Changes
Signed Complete: Date:
(Cognizant Engineer)

Engineers Required:
Frank [illegible]
John Byrd
[illegible]
Larry Haggett
Elliot

Responsible Craft:
Electricians, [illegible]
CARPENTERS
PIPE-FITTERS
No. of Craft Required: [illegible] 12

EXHIBIT A2
PAGE 1 of 15

This page revised and effective November 4, 1974
[illegible] as dictated by [illegible].

SECONDARY CONTAINMENT AIR LEAKS

MECHANICAL

REFUELING FLOOR (ELEV. 664')

1. DOOR 707 TO CLOTHING CHANGE ROOM
2. CRACKS IN WALL TO CLOTHING CHANGE ROOM
3. DAMPER ON SOUTH WALL OF CLOTHING CHANGE ROOM NEEDS TO BE SEALED AROUND
4. DOOR 705 IN SW CORNER OF REFUELING FLOOR
5. DOOR 704 IN SW CORNER OF REFUELING FLOOR
6. DOORS 702 & 703 TO STAIRS
7. DOORS 712 & 714 TO STAIRS
8. UNIT 2 TO UNIT 3 TEMPORARY AIR LOCK DOOR, SEALING, & [illegible]
9. SEAL AT LEND S/S ON SOUTH WALL NEXT TO R-3 COLUMN
10. SEAL AT FIRE PROTECTION AT R-7 & P COLUMN
11. RADIATION DETECTION INSTRUMENTATION AT NORTH WALL EAST OF R-[illegible]
12. WATER LINE ON TEMPORARY WALL AT R-14
13. SIDING BETWEEN R-13 & R-14 ON TOP OFF NORTH WALL (4 HOLES)
14. DOOR 130 'ELEVATOR MACHINERY ROOM' AND DAMPER IN SOUTH BLOCK WALL. SEVERAL 1" HOLES THRU BLOCKS IN WALLS ON SOUTH & WEST WALLS.
15. FUEL POOL COOLING VALVE ACCESS HATCH NEAR SOUTH EAST SIDE OF SPENT FUEL POOL — INSTALL NEW PLATE IN BOTTOM TO REPLACE TAPED PLATE.
16. DOOR 710 TO TOILET

UNIT 1 ELEV. 639'

1. DOORS FROM ELEVATOR AREA TO MG SET AREA
2. FIRE EXTINGUISHER SUPPORT BRACKET ON BLOCK WALL
3. 4" PIPE IN NORTH WEST CORNER AT HANGER (RECW SURGE TANK)
4. BEHIND PANEL 25-213 — FIRE PROTECTION SLEEVE IN NORTH WALL (SLC ROOM)

UNIT 1 ELEV. 621'

1. DOOR 664 TO ELECTRICAL BOARD ROOM
2. 6" VENT DUCT IN NW CORNER APPROX. 15' ABOVE FLOOR

EXHIBIT A2
PAGE 2 OF 15

MECHANICAL

UNIT #1 593' ELEVATION
1. DOOR 588 TO ELECTRICAL BOARD ROOM
2. DOOR 490 TO ELEVATOR AREA.

UNIT #1 565' ELEV.
1. DOORS TO ELEVATOR AREA (DOOR 228 ALSO)
2. SLEEVE 6D ON NORTH WALL AT NE CORNER NEAR CS LIN.
3. SLEEVE NEAR VALVE 1-12-776 ON NORTH WALL IN NW CORNE SLEEVE 4G AND LINE UNDERNEATH.
4. EECW SLEEVE THRU NORTH WALL TO AIR COMPRESSOR (N.W. CORNER SLEEVE 4A)
5. POTABLE WATER LINE 11B SLEEVE ON WEST WALL. LEAKI AT CHECK VALVE FROM INSULATION.
6. EECW SOUTH HEADER SLEEVE 11A. LEAK AT NORTH & SOUTH SIDE OF SLEEVE.
7. RHRSW DISCHARGE AT SW CORNER
8. EECW DISCHARGE AT SOUTH WALL EAST SIDE
9. SMALL POTABLE WATER LINE NEAR DOOR 231, LINE GOES INTO EQPT ACCESS LOCK. LEAKING AROUND CONDUIT ALSO.

UNIT #2 ELEV. 639'
1. DOOR 673 TO UNIT #3 AND ALL DOORS TO ELEVAT. AREA.

UNIT #2 ELEV. 621'
1. ALL DOORS

UNIT #2 ELEV 593'
1. EECW VACUUM BREAKER AT SOUTH WALL, SW SIDE. TO BE WORKED ON DIFFERENT WORK PLAN.
2. DOOR TO BATTERY ROOM HAS SLIGHT LEAK

EXHIBIT A2
PAGE 3 of 15

UNIT # 2 ELEV. 565'

1. 1½" PIPE NEAR FLOOR AT R-9 & U COLUMN ON SOUTH WALL NEEDS TO BE CANNED.
2. RHRSW DISCHARGE AT SOUTH WALL SOUTH WEST SIDE.
3. RHRSW "2A" SUPPLY HAS SLIGHT LEAK AT SLEEVE.
4. INSTRUMENTATION LINE JUST BELOW BLDG HEATING LINES ON SOUTH WALL SOUTH EAST SIDE.
5. RCW DISCHARGE ON SOUTH WALL SE SIDE.
6. UNIT 2 & 3 PERSONNEL ACCESS AIR LOCK
7. RCW LINE AT NORTH WALL IN NE CORNER NEAR TIP ROOM. (WEST SIDE OF PIPE AT SLEEVE.)

EXHIBIT A2
PAGE 4 of 15

ELECTRICAL QR1 – Leaks around trays MAZ-I & 1V-I where trays Penet. wall.

QR1 – Box 3098
Conduit ES2195-I3 leaking where conduit Penet. R1 wall.

QR1 – Box 959
Conduit PP497
Leaks where conduit Penet. wall.

NR2 – Leaks where trays AT-I, AU, EL-I and KO-I Penet. "N" line wall.

NR2 – Pull Box near tray openings in wall.
Conduit ES87-I & ES100-I, leaking.

RR1 – Box 3151
3 conduits leaking where they Penet wall.

RR1 – Box 3133
Top left and bottom right conduits leaking where they Penet. wall.

RR1 – Box 1507
Left conduit leaks where it Penet. wall.

RR1 – Box 3029
Right bottom conduit leaks.

RR1 – Box 3030
Left conduit leaks.

EXHIBIT A2
PAGE 5 of 15

Air leaks Unit #1

ELECTRICAL

Chiller Room "A"

(PF2) el 596

Conduit ES1373-I leaking at tray FV-I.

NOTE: ADDITIONAL ELECTRICAL AIR LEAKS WILL FOLL.

Unit #2

El. 565

UR12:

Conduit 2PL1161 leaking. (located over 480V Vent Bd. 2B.)

NR12 Tray ground not terminated where trays Penet. "N" wall. Also loose tray support.

NR12 Leaks around trays BP, BF, EP, KX, AQ-II, EV-II and KW-II where they Penet. "N" wall.

PR14 leaks where 24" tray Penet wall.

NR12 Box 3506
Conduit ES2609-II leaks inside box.

El. 606 (Spreading Room)

EXHIBIT A2

PR4 Conduit leaks at box (l[illegible]) 6 of 15
Near conduit L501 (Right side [illegible])

Unit # 1
(Above CRD Pumps)

GR1 — 1ES 232-I — W.P. 2892

1ES 232-I

1A 2673 — Continuation of

1A 2099 — Electrical Air

MC 231 — Leaks More

1A 1039 — To Come

ES 33-I

1ES 3261-II

1PL 336 — Conduits Leaking At

1ES 3290-II — Tray End.

1PL 335

1ES 231-I

1ES 3292-II

1ES 3291-II

1ES 3260-II

1ES 3291-II

1PL 336 (Seal at LB)

1PL 355 (Seal at Box)

1PL 335 (at tray)

1A 1066
PP 466 — Conduits Leak At Tray End.
PP 465

ES 2051-1B
ES 2037-1B — Leaking Inside Pull Box

ES 918 Leaking At Tray End. Also Needs Tagging.

ES 2019-IA Leaking At Tray End.

4" Conduit Not tagged. Leaks At Tray EN-ES I

Note: Some Conduits appear more than once, because they are common carriers to Shutdown Bds.

EXHIBIT A2
PAGE 7 of 15

Unit #2

El. 565 (over CRD Pumps)

W.P. 2892

RR12 — ES157
2PL5455
2ES3233
2ES3241
2PL5695
2PL461
} Conduits Leaking At Tray end.

RR12 — 3" Conduit located between conduits 2PL461 and 2PL330 Leaking at "LB".

RR12 — 2PL330
PP421
2A1066
2PL391
} Conduits Leaking At Tray End.

RR12 — 2ES3241
2ES3240
} Conduits Leaking At Tray End.

RR12 — Box 3367
2ES4562
2ES4591
2ES4558
} Conduits Leak At Box. (I...

QP12 — 2ES340
ES3417
ES3415
} Conduit Leak At Tray End.

Note: Some Conduits appear more than once because they are common carrying to Shutdown Bds.

EXHIBIT A-2
PAGE 8 of 15

Air Leaks
El. 606 (Spreading room "A" & "B") File

Attachment to WP 2892

NR7 — Leaks around trays [illegible]
where trays Penetrate wall.

NR8 — Leaks around trays KKF-I, [illegible], TF
where trays Penetrate wall.

NR9 - NR10 — Leaks around trays PQ, [illegible]
where trays Penetrate wall.

NR10 - NR11 — Leaks around trays NG & [illegible]
where trays Penetrate wall.

NR4 - NR5 — Leaks around trays VY, [illegible]
where trays Penetrate wall.

PR9 — Box 1382
Conduit 2NM115 leakin at box.

PR9 — Box 1381
Conduit 2NM122 leaking at box.

PR9 — Box 4512
Conduits 2P3750-I, 2P3959-II [illegible]

PR9 — Conduit 2K140 leaks at wall.

EXHIBIT A2
PAGE 9 of 15

(Smallest part of box)

Attachment to NP 2092 – Unit 2 – El 593 [illegible]
5-12-[illegible]

Unit # 2 El 593

Over Roof of Shtdn Bd Rm "D"

conduits

2PL 375	to	Bd	Rm	"C"	SR13
PP 903	"	"	"	"	"
2PL 2449	"	"	"	"	"
2PL 2474	"	"	"	"	"
ES4513	"	"	"	"	RR13
ES3422	"	"	"	D	"
2ES2513	"	"	"	C	QR13
PP916	"	"	"	D	"
2ES3234	"	"	"	D	"
2ES 3242	"	"	"	D	"
2ES 3229	"	"	"	C	PR13
2ES 2511	"	"	"	C	PR13
ES 2268	SR14	at tray			
2LS 304	QR13				
2ES 3035	QR13				
2PL 440	QR13				
A1298	Box	at QR14			
2LS 304	Box	at QR14			
3PP191	Wall	at PR14			
2ES4536	"	"	"		
2ES4522	"	"	"		
2LS 386	Box 3201	QR14			
1A1053	RR14				

2LS 535 }
2LS 586 } RR14 ... leaking around grouting of conduits where they penetrate ceiling.

EXHIBIT A2
PAGE 10 of 15

PR4 — Box 2415 conduit 5591
PR4 — LB over Lighting
PR4 — Box 2415 5311S, 335, 1899
Unit 1 El 593 172
5-12-85

Unit #1 El 593
Over Roof of Shtdn. Bd. Rm "B"

Conduits

PP893	conduit goes to Bd. Rm.	"B"
ES922-I	" " " " "	"B"
ES917-I	" " " " "	B
ES918-I	" " " " "	B
ES903-I	" " " " "	A
PP877	" " " " "	A
1ES238	" " " " "	A
ES2009	" " " " "	A
1ES3212	" " " " "	A
ES905-I	" " " " "	A
ES906-I	" " " " "	A
PP895	" " " " "	B
PP884	" " " " "	B
1ES3202	" " " " "	B

1LS 323-A1 Al troy KFD-I
Conduit not tagged (North of ES2013) to Bd. Rm "A"
1ES236 To Bd. Rm "A"
1ES236 (E) (RR1)
1ES235 (RR1) " " " "A"
1ES3202 (RR1) " " " "B"
1PC124 (RR1) " " " "B"
1ES236 (1) (RR1)

EXHIBIT A2
PAGE 11 of 15

1ES3357	At Box 2054		
LS401-A2	" " "		
1ES236	QRL	to Bd. Rm. "A"	
1ES3212-2	at Box 2611 (PRI)		
1PL455	to Bd. Rm. "B"	at seal-off	
1A1062	to Bd. Rm. "B"	at seal-off	
New conduit (not tagged) ??? not sealed at seal off			
1LS333-A1	to Bd. Rm. "B"	at seal off	
1ES525-I	" " " "B"	" " "	
1LS491-A2	" " " "B"	" " "	

EXHIBIT A2
PAGE 12 of 15

Air Leaks
Unit #1 El. 621

1 of 1
3-1...
W.P. 2892

SR2 - 1V... } seal at seal..... bell
1V2977 } bell rm.

SR2 - 1A2791 seal at LB on wall (Near ...

SR2 - Seal conduits coming from box on outside wall
of bell rm near door 634.

SR2 - 1A2611 seal at box (Near door ...)

RR2 - LB on roof of bell rm not sealed

SR2 - PF474 conduit

SR2 - 1P3163 conduit

TR2 - Box 1336 conduit 1PL467

PR5 - 1PL50 }
1PL 843 } conduits
PL1210 }

Unit #1 El. 630

PR5 - 1A2464 } conduits
PR5 - 1R386 }

EXHIBIT A2
PAGE 13 of 15

3-15-79

AIR LEAKS
Unit #2 El. 621

SR13 — Conduit 2[illegible] ([illegible])

SR13 — 1" conduit (directly over [illegible]) ([illegible])

SR13 — Box 4025 (1" conduit top right of box)

TR14 — Conduit PP137-IE

SR12 — 2PL375 (At [illegible]) ([illegible])

PP10 — 2T7 conduit

PP10 -- Conduit 2RM 26

EXHIBIT A2
PAGE 14 of 15

1-1
5-1-75

Air Leaks

Unit 1+2 El. 639

SR13	PP656	conduits	U.P. 2892
	PP638		
	PP639		

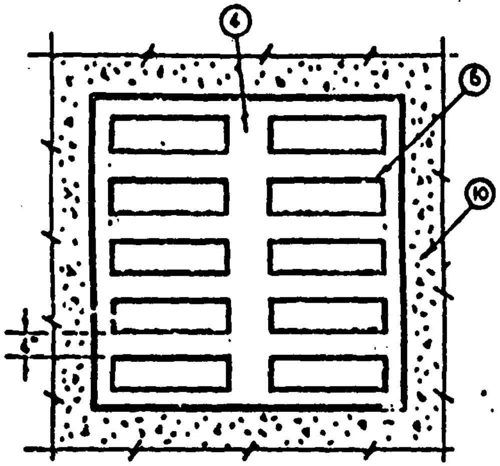
El. 664 Unit 1+2

PXR4 – 1A2464 seal at seal-off.
PXR4 – PL525
PXR5 – 1K33
PXR5 – 1T9
QXR6 – 1A1249
QXR6 – 1PL2740
PXR6 – 1PL5165
PR12 – 2ES1853
PYR12 – Box 3489 Conduits 2PL5100 + 2PL5101
PR10 – Below L13 Cab. LD2 seal open plug
for temporary cable.
URX14 – Box 320 conduit 1A2279
PR14 – 4" Conduit in wall between Unit 2+3.
– leaking around cable that is stubbed off.
UXR12 – 2C210 conduit on right bottom
UXR6 3/4 – 1PL5194
TR6 – 1PL5192

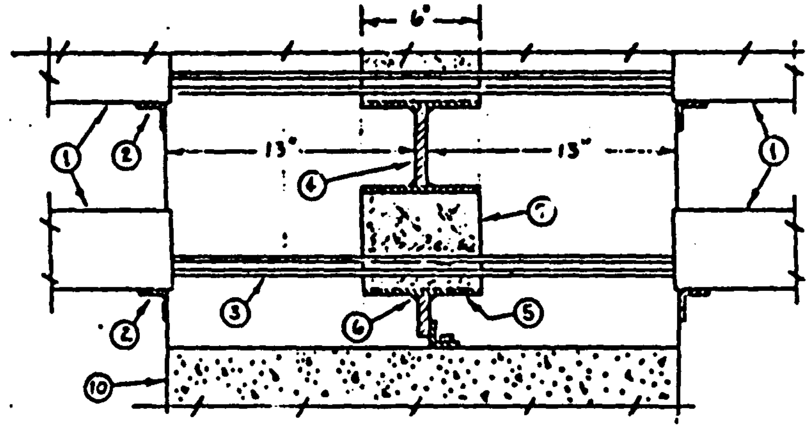
EXHIBIT A2
PAGE 15 OF 15

– J-Box on clothes changing room

C.E. ALDERSON 6-7-75

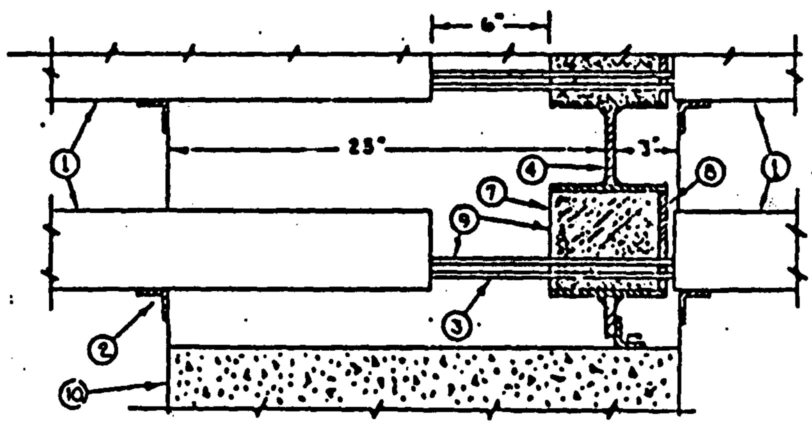


FRONT ELEVATION
(NOT TO SCALE)



AS DESIGNED

1	CABLE TRAYS
2	TRAY SUPPORTS
3	CABLES
4	BULKHEAD - ½" STEEL PLATE
5	SLEEVE - 5" x 18" INSIDE DIMENSION ⅛" THICK
6	SLEEVE WELDED TO BULKHEAD ALL AROUND ON BOTH SIDES
7	POLYURETHANE FOAM FILLED FOR SEAL
8	SHEET POLYURETHANE DAM
9	FACE OF POLYURETHANE & 12" OF CABLE COVERED WITH FLAMEMASTIC
10	FACE OF WALL



AS CONSTRUCTED

TYPICAL CABLE PENETRATION

Exhibit A3

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Electrical Design Files

George Doubleday, Principal Electrical Engineer, 102 DB-K

June 4, 1973

BROWNS FERRY NUCLEAR PLANT - CABLE PENETRATION FIRE STOPS
EVALUATION OF TEST DATA

Reference memorandum to M. N. Sprouse from Gene Farmer dated May 24, 1973, subject, Browns Ferry Nuclear Plant - Cable Penetration Fire Stops Performance Testing.

The purpose of the test was to examine the effectiveness of cable penetration fire stops combined with two types of urethane pressure seals. See Figure I for details of test sample.

The first three burns of 5 minutes each, recorded in the test data, were made to simulate a cable tray fire in which the cable insulation supported combustion. Temperatures of 590, 1190, and 1320 F, respectively, were applied for these burns; the upper T/C's 7, 9, and 11 recorded a constant 80 F. This indicated there was no heat transfer from the lower part of the fire stop to the upper. The fire stop and pressure seal both maintained their integrity.

The fourth burn consisted of the propane-oxygen flame being applied for 15 minutes with a recorded peak flame temperature of 1610 F. The burner was elevated to bottom of cables. This burn was made to simulate an oil fire enveloping the lower part of the fire stop. During this burn large cracks developed in lower Flamemastic coated cables and some small voids were noted in the Flamemastic around the edges of the metal sleeve top and bottom. The postulation that heating the urethane foam contained by the Flamemastic would generate hot gases causing rupture of the coating and spew flaming gas out the top and bottom of the fire stop did not materialize.

The fire stop itself remained intact, although it is assumed the pressure seal failed. The upper T/C's 7, 9, and 11 registered a peak temperature of 120 F; this was only a 40 F rise.

A final burn to failure required 20 minutes of additional flame with a peak temperature of 1740 F. During this burn all the insulation and Flamemastic on the lower end of the fire stop was consumed; the upper part remained intact. Smoke and steam or gas came through the top and around the edges of the seal. The Flamemastic cracked across the sleeve and separated from the foam. The white urethane foam showed superior heat-resistant characteristics.

EXHIBIT A4
PAGE 1 of 3

2

Electrical Design Files
June 4, 1973

BROWNS FERRY NUCLEAR PLANT - CABLE PENETRATION FIRE STOPS
EVALUATION OF TEST DATA

It is my opinion from these test results that the fire stop being installed at Browns Ferry Nuclear Plant will provide a good fire barrier between walls and floors.

The Flamemastic manufacturer's recommendation that the cables should be coated for 6 to 8 feet on both sides of the penetration is not valid. The one-foot coating on each side of the sleeve used in the test proved satisfactory.

It is recommended that pressure and fire test be made on two other type penetrations. The first consisting of filling voids around cables and its sleeve with inorganic thermal insulating wool and then coated with Flamemastic. The second using 1/2-inch-thick Marinite board cut to fit over the cables and then coating the cables and sleeve on both sides with Flamemastic. These types of seals would eliminate the requirements for the urethane foam as a pressure barrier.

George Doubleday

HCR GD:MC
Attachment
CC: F. W. Chandler, 303 UB-K

6-13-73--FWC:FHC
CC (Attachment):
R. N. Hodges
J. E. Holladay
D. R. Patterson, 204 UB-K

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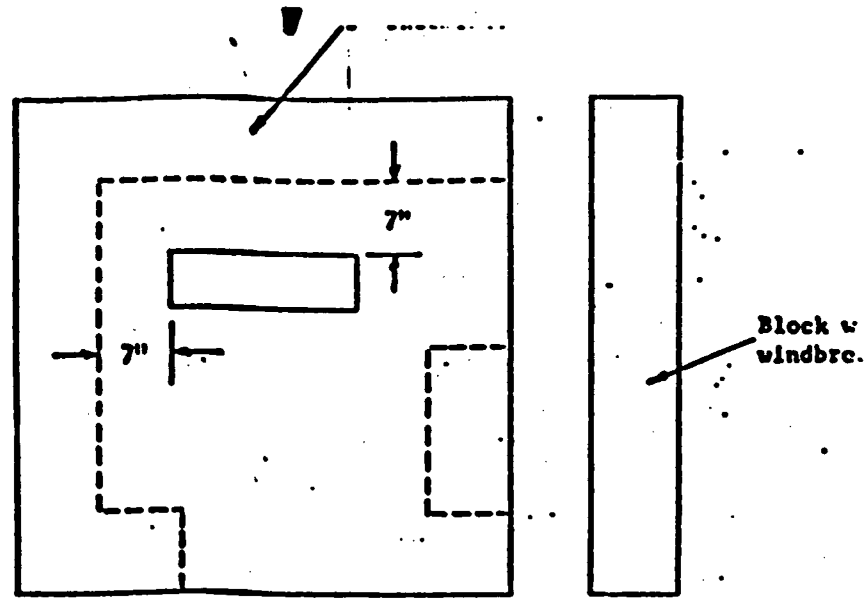
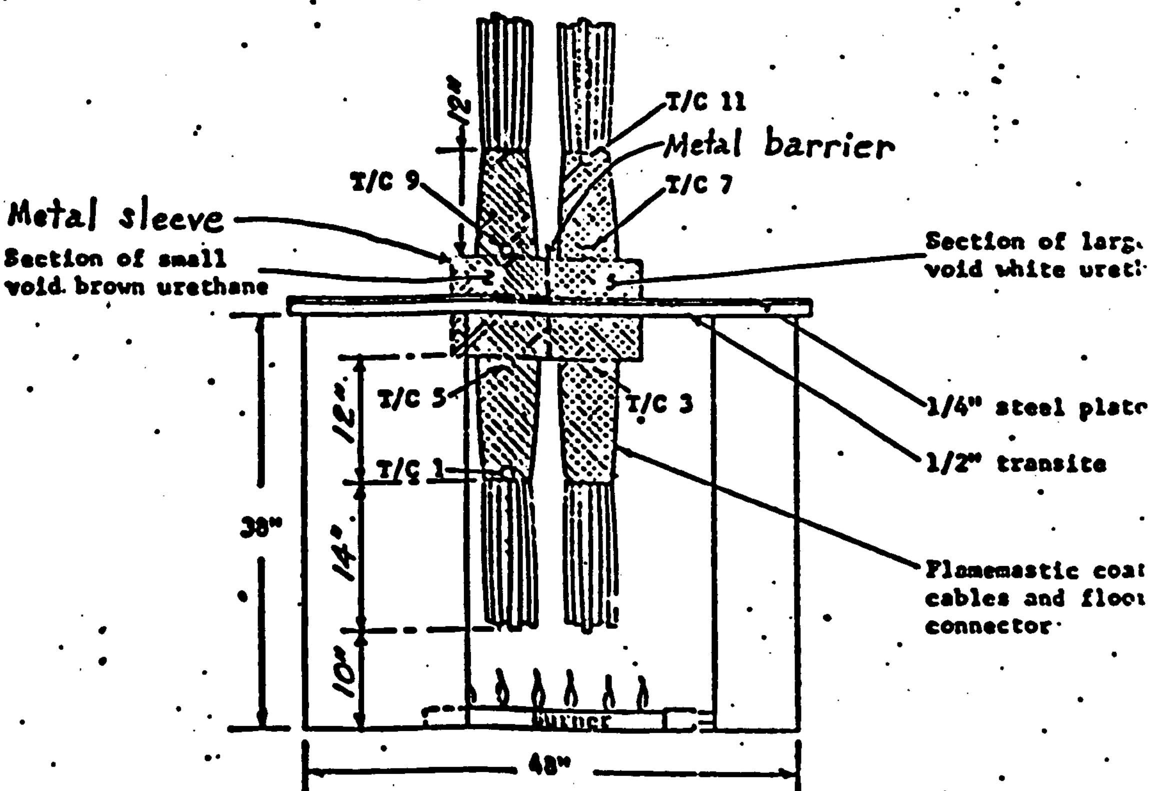
PAGE 2 of 3



Top View



Front View

Figure I

Test of Fire Stop Flamemastic for Cable Sleeve

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PAGE 373

H. C. Russell, Browns Ferry and X17-X20 Designs Projects Manager, 423 [illegible], Knoxville
R. T. Urthcote, Project Manager, Browns Ferry Nuclear Plant, Decatur

April 9, 197[illegible]

BROWNS FERRY NUCLEAR PLANT - CONDITION OF FIRESTOPS IN UNITS 1 AND 2 [illegible]

This is in response to a telephone conversation between M. M. Price at Browns Ferry and Clint Walker, Larry Weber, and Bob Bradley at Knoxville on April 7, 197[illegible]. An investigation of evidence of the application of flamemastic in unit 1 to wall penetrations and unit 2 wall penetrations produced the following observations:

A. On the penetrations above elevation [illegible] between the turbine and reactor buildings, the turbine side showed no evidence of ever having had flamemastic applied. Flamemastic had been applied to the reactor side. Flamemastic had not been reapplied as new cables were added in the penetrations and as resealing was completed.

B. On the penetrations between the reactor and diesel generator buildings, the diesel generator side showed no evidence of flamemastic coating. The reactor building side had an initial flamemastic application. However, as cables were added and resealing occurred, most of the flamemastic was covered or removed. No additional flamemastic treatments were made.

C. The penetrations between the reactor building and the spreading room showed evidence of flamemastic coating on both sides of the penetrations. In most instances this flamemastic material has been covered or removed as new cables were pulled and as resealing occurred.

In addition, the penetrations in the spreading room floor had been flamemastic treated. Most of these had cables added and sealant applied. These additions were not flamemastic treated.

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R. C. Russell
April 9, 1975

BROWNS FERRY NUCLEAR PLANT - CONDITION OF FIRESTOPS IN UNITS 1 AND 2 AREA

D. The penetrations between the spreading room and the turbine building showed evidence of a flamemastic application to both sides. An exception is those trays near the barrier between units 2 and 3 on the unit 2 side. These penetrations were never flamemastic treated on the turbine building side.

E. The penetrations between the auxiliary board rooms and the computer rooms were never flamemastic treated. Some of these penetrations have been broken for pulling additional cable. Spray foam and RTV have been applied.

F. The penetrations in the masonry walls in the spreading rooms show that the cables in the trays, but not the penetrations, have been flamemastic treated.

Summary:

1. Approximately half of the wall penetrations for unit 1 reactor building were flamemastic treated only on one side (secondary containment side).

2. For the majority of penetrations that were flamemastic treated, original flamemastic material was covered or removed when cables were added.

3. Normally, flamemastic was not reapplied when cables were repulled or when resealing occurred.

4. Unit 2 firestops and flamemastic treatment are for the most part non-existent.

R. T. Hathcote

RTH:LI
Attachment
CC: C. S. Walker, 302 UBB, Knoxville
L. D. Weber, 503 UB, Knoxville

EXHIBIT A5
PAGE 2 of 23

BEST AVAILABLE COPY

UNIT - 1 TRAYS PENETRATING WALL
BETWEEN REACTOR Bldg. And D.G.B. #1
El. 565

TRAY	El. & Col.	Side X	Flamastic applied	Side Y	Flamastic applied	[illegible] Cables fire after coating applied	
							As Appl.
IV-I	QRI	RB	Yes	DGB	NO	Yes	NO
MAZ-I	QRI	RB	Yes	DGB	NO	Yes	NO
SAE	QRI	RB	Yes	DGB	NO	Yes	NO

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REACTOR BLDG & TURBINE

UNIT #1 El. 565

TRAY	El. + Col.	[illegible] X	[illegible]	[illegible] Y	[illegible]	[illegible]	Re Appl.
AT-I	NR2	RB	Yes	TB	NO	Yes	NO
AU	NR2	RB	Yes	TB	NO	Yes	NO
EL-I	NR2	RB	Yes	TB	NO	Yes	NO
KO-I	NR2	RB	Yes	TB	NO	Yes	NO
EG	NR6	RB	Yes	TB	NO	NO	N/C
KG	NR6	RB	Yes	TB	NO	Yes	NC
EF-II	NR6	RB	Yes	TB	NO	Yes	NO
KE-II	NR6	RB	Yes	TB	NO	Yes	[illegible]

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Trays Penetrating Wall Between
Reactor Bldg. + Spreading
Unit #1 El. 593 + 606 Computed ____ Date ____

Tray	El. + Col.	Side X	Flamastic Applied	Side Y	Flamastic Applied	Cables Put Above Flamastic Applied
AF	PR3	RB	Yes	Spreading El. 606	Yes	NO
SAB	PR3		Yes		Yes	Yes
FJ	PR3		Yes		Yes	Yes
AG	PR3		Yes		Yes	NO
BA	PR3		Yes		Yes	NO
ML	PR3		Yes		Yes	Yes
KR	PR4		Yes		Yes	Yes
MA	PR4		Yes		Yes	NO
TQ	PR4		Yes		Yes	Yes
MK-I	PR4		Yes		Yes	Yes
LZ	PR4		Yes		Yes	Yes
VN	PR4		Yes		Yes	Yes
UE-I	PR5		Yes		Yes	Yes
NA	PR5		Yes		Yes	Yes
VY	PR5		Yes		Yes	Yes
TU	PR5		Yes		Yes	Yes
ND-I	PR5		Yes		Yes	Yes
NB	PR5		Yes		Yes	Yes
VX	PR5		Yes		Yes	Yes
KEZ	PR5		Yes		Yes	Yes
OQ	PR5½		Yes		Yes	NO
KJA	PR5½		Yes		Yes	Yes
FM	PR6½		Yes		Yes	Yes
MX-II	PR6½		Yes		Yes	Yes
MD	PR6½		Yes		Yes	Yes
VE	PR6½		Yes		Yes	Yes
TE	PR6½		Yes		Yes	Yes
FL	PR6½		Yes			Yes
NW-II	PR6½		Yes		Yes	Yes
LY	PR6½		Yes		Yes	Yes
VK	PR6½		Yes		Yes	Yes
TK	[illegible]		[illegible]		[illegible]	[illegible]

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TRAYS PENETRATING WALLS WITHIN
REACTOR BLDG.
Unit #1 El. 593

TRAY	El. & Col.	Side X	Flamastic coatings	Side Y	Flamastic coatings	Cables Pul. After Flamastic Applied
FO-II	QR6	RB	NO	RB	NO	
ME-II	QR6	RB	NO	RB	NO	
VK	QR6	RB	NO	RB	NO	
TL-II	QR6	RB	NO	RB	NO	Plugs Indicate Seal Only
FO-II	SR6	RB	NO	RB	NO	
ME-II	SR6	RB	NO	RB	NO	
VK	SR6	RB	NO	RB	NO	
TL-II	SR6	RB	NO	RB	NO	

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Trays Penetrating Wall Between
Spreading Room & Turbine Bldg
Unit #1 El. 606

Tray	El. & Col.	Side X	Flamastic Applied	Side Y	Flamastic Applied	Cables Full Above Flamastic Applied	Re App.
AF	NR3	Spread Rm	Yes	T.B. U.586	Yes	NO	-
SAB	NR3		Yes		Yes	Yes	N
FJ	NR3		Yes		Yes	NO	-
KR	NR3		Yes		Yes	NO	-
NA	NR3		Yes		Yes	NO	-
AG	NR3		Yes		Yes	NO	-
BA	NR3		Yes		Yes	NO	-
LW	NR3		Yes		Yes	Yes	N
LZ	NR3		Yes		Yes	NO	-
NP	NR4		Yes		Yes	NO	-
TY	NR4		Yes		Yes	NO	-
NR	NR4		Yes		Yes	Yes	N
WA	NR4		Yes		Yes	Yes	N
FM	NR6		Yes		Yes	NO	-
MC	NR6		Yes		Yes	Yes	N
LN	NR6		Yes		Yes	Yes	N
TJ	NR6		Yes		Yes	Yes	N
FL	NR6		Yes		Yes	Yes	N
MB	NR6		Yes		Yes	Yes	N
LS	NR6		Yes		Yes	Yes	N
VD	NR6		Yes		Yes	Yes	NC

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TRAYS PENETRATING FLOOR BETWEEN
El. 593 & El. 565 REACTOR BLDG.
Unit #1

TRAY	El. & Col.	593 Side X	Flamastic Applied	565 Side Y	Flamastic Applied	Cables Full Above Flamastic Applied	Seal
* EAA-I	PR2	RB	Yes	RB	Yes	Yes	NO
* SAB	PR2	RB	Yes	RB	Yes	Yes	NO
* AY-I	PR2	RB	Yes	RB	Yes	Yes	NO
* EG-I	PR2	RB	Yes	RB	Yes	Yes	NO
* KE-I	PR2	RB	Yes	RB	Yes	Yes	NO
* KG	PR2	RB	Yes	RB	Yes	Yes	N.
OQ	PR5½	RB	Yes	RB	Yes	NO	
KJA	PR5½	RB	Yes	RB	Yes	Yes	
VE	PR6	RB	NO	RB	Yes	Yes	
TE	PR6	RB	Yes	RB	Yes	Yes	
KS-II	PR6	RB	Yes	RB	Yes	Yes	
KT	PR6	RB	Yes	RB	Yes	NO	
NAY-I	RR2	RB	Yes	RB	NO	Yes	
KU-II	RR2	RB	NO	RB	NO	NO	
EW-II	UR5	RB	Yes	RB	Yes	Yes	
LE-II	UR5	RB	Yes	RB	Yes	Yes	
BL	SR5	RB	Yes	RB	Yes	NO	

* Foam Seal has been dug out on 593 Floor and not replaced.

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Unit 1 Reactor Bldg.

TRAY	El. & Col.	Side X (503)	[illegible]	Side Y (621)	[illegible]	Cables [illegible] Above [illegible]	[illegible]
AF	PR2½	RB	Yes	RB	Yes	NO	—
EG	PR2½	RB	Yes	RB	Yes	Yes	N
AG	PR2½	RB	Yes	RB	Yes	NO	—
MM	PR3	RB	Yes	RB	Yes	Yes	N
MN	PR3	RB	Yes	RB	Yes	Yes	N
BB	TR3	RB	NO	RB	Yes	NO	—
BC	TR5	RB	Yes	RB	Yes	NO	—
KFG-I	Pnl. 25-32 QR2	RB	NA	Pnl.	NO	N/A	—
KFF-I	Pnl. 25-31 QR2	RB	NO	Pnl.	NO	N/A	—
HW	Col. 22-A RR2	RB	NO	Pnl.	NO	N/A	—
EW	UR5	RB	Yes	RB	Yes	Yes	NO
LE	↓	↓	Yes	↓	Yes	Yes	NO
TL	↓	↓	Yes	↓	Yes	NO	—
* MU	PR5	RB	Yes	RB	Yes	Yes	NO
* FY	PR5	RB	Yes	RB	Yes	Yes	NO

* Foam Sealant dug out on 621 el. end not replaced

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TRAYS PENETRATING FLOOR BETWEEN
El. 639 & El. 621
Unit #1 REACTOR BLDG

TRAY	El. & Col.	Side X	[illegible]	Side Y	[illegible]	[illegible]	
FU.	SR2	RB	Yes	RB	Yes	Yes	NO
MO.	SR2	RB	Yes	RB	Yes	Yes	NO
NM	SR2	RB	Yes	RB	Yes	NO	
FU	SR6	RB	Yes	RB	Yes	NO	
MO	SR6	RB	Yes	RB	Yes	Yes	[illegible]
BC	TR3	RB	Yes	MG Sol 1A	Yes	NO	
BB	TR4	RB	Yes	MG Sol 1B	Yes	NO	
AG	TR5	RB	Yes	MG Sol 1B	Yes	NO	
NAC	UR4	RB	Yes	[illegible] 23	Yes	NO	
NAB	UR4	RB	Yes	[illegible] 24	Yes	NO	

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Trays Penetrating Floor Between El. 606 (Spreading Rm) & El. 573 (Control Rm)

Unit #1

TRAY	El. & Col.	Side X	[illegible]	Side Y	[illegible]	[illegible]	
KAH		SR	Yes	[illegible]	NO	Yes	NC
KAL-II			Yes		NO	Yes	NC
KAP			Yes		NO	Yes	NC
VAB			Yes		NO	Yes	NC
KAE-II			Yes		NO	Yes	NC
KAI-II			Yes		NO	Yes	NC
KAM			Yes		NO	Yes	NC
KAQ			Yes		NO	NO	—
KAU			Yes		NO	Yes	NC
KAF-I	NR5-R6		Yes		NO	Yes	NO
KAJ-I	PR5-R6		Yes		NO	Yes	NC
KAN			Yes		NO	Yes	N
KAR			Yes		NO	Yes	NC
KAV			NO		NO	N/A	—
KAG-I			Yes		NO	Yes	NC
KAX			Yes		NO	Yes	N
KAK			Yes		NO	NO	—
KAO			Yes		NO	Yes	N
KAS			Yes		NO	Yes	[illegible]
KAW			Yes		NO	NO	—
TAX			Yes		NO	Yes	N
TAY			Yes		NO	Yes	N
TAZ			Yes		NO	Yes	N
TBA			Yes		NO	Yes	N
KAC	PR4	SR	Yes	[illegible]	NO	NO	—
SO	NR4	SR	Yes	[illegible]	NO	Yes	[illegible]
KAT	NR7	SR	Yes	Comm. Rm.	NO	Yes	N
VAF	NR7	SR	Yes	Comm. Rm.	NO	Yes	N
VAG	PR7	SR	Yes	Comm. Rm.	NO	Yes	[illegible]
VAH	PR7	SR	Yes	Comm. Rm.	NO	Yes	[illegible]
NAL	[illegible]	[illegible]	NO	[illegible]	[illegible]	[illegible]	
TRI	[illegible]	RM	NO	[illegible]	NO	[illegible]	—

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Tray Penetrating Wall Between
Spreading Room Stairwell + Mech. Equip. Rm.
Unit 2 El. 606

TRAY	El. + Col.	Side X	[illegible] Method	Side Y	[illegible] Method	[illegible] for other Elements Applied	
A.F.	NR3	M.E.R	Yes	SR	Yes	NO	-
SAB			Yes		Yes	Yes	N.
FJ			Yes		Yes	Yes	N.
AG			Yes		Yes	NO	-
BA			Yes		Yes	NO	-
ML			Yes		Yes	NO	-
KZ	NR4	Stairwell	Yes	SR	Yes	Yes	
MA			Yes		Yes	NO	-
TO			Yes		Yes	Yes	[illegible]
MK-I			Yes		Yes	Yes	[illegible]
LZ			Yes		Yes	Yes	[illegible]
VN			Yes		Yes	Yes	[illegible]

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Control Room #7 Panels — X = SR Ceiling
Y = CR Pnl Floor

Line Lt	El. + Col.	Side X	Flamastic Applied	Side Y	Flamastic Applied	New Cable	Flamastic Added
		SR		CR			
* 9-4		"	YES	"	YES	NO	—
9-5		"		"	"	YES	NO
9-6		"		"	"	"	"
9-7		"		"	"	"	"
9-8		"		"	"	"	"
* 9-2		"		"	"	"	"
9-10		"		"	"	"	"
9-11		"		"	"	"	"
9-12		"		"	"	"	"
9-13		"		"	"	"	"
* 9-14		"		"	"	NO	—
9-34		"		"	"	NO	—
9-53		"		"	"	NO	—
					"		
9-9		"		"	"	YES PNL 6	NO
* 9-25		"		"	"	YES	NO
9-46		"		"	"	NO	—
* 9-47		"		"	"	NO	—
* 9-44		"		"	NO	YES	NO
9-23		"		"	NO	YES	NO
9-20		"			NO	YES	NO
9-21		"			YES	YES	NO
9-22		"		"	NO	YES	NO
9-24		"			YES	YES	NO
9-3		"			YES	YES	NO
CB-1		"		"	YES	YES	NO
CB-2		"		"	"	YES	NO
CB-3		"		"	"	YES	NO
CB-4		"		"	"	YES	NO

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* INDICATES POOR INITIAL FLAMASTIC TREATMENT

Trays [illegible] Between
Reaction Bldg & [illegible] Bldg.
[illegible] El. 51.5

Trays	El. [illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
EU-I	NR9	RB	NO	TB	NO	N/A
KW-I	NR9		NO		NO	N/A
EP	NR9		NO		NO	N/A
KX	NR1		NO		NO	N/A
BP	NR14	RB	NO	TB	NO	N/A
BF	NR14		NO		NO	N/A
EP	NR14		NO		NO	N/A
KX	NR14		NO		NO	N/A
AQ-II	NR14		NO		NO	N/A
EU-II	NR14		NO		NO	N/A
EW-II	NR14		NO		NO	N/A

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Trays Penetrating Wall [illegible]
Reactor Bldg [illegible] Spreading Room
Unit #2. El. 593 + 606

TRAY	El. + Col.	[illegible] X	[illegible]	[illegible] Y	[illegible]	[illegible]
KKF-I	PR8	RB	NO	SR	Yes	Yes
LV-I			NO		Yes	Yes
MT			NO		Yes	Yes
YP			NO		Yes	Yes
TR			NO		Yes	Yes
GB			NO		Yes	NO
KV-I			NO		Yes	Yes
MS			NO		Yes	NO
YO			NO		Yes	Yes
TP			NO		Yes	Yes
PQ	PR9	RB	NO	SR	Yes	Yes
KJG	PR9	RB	NO	SR	NO	N/A
NF-II	PR10	RB	NO	SR	Yes	Yes
NG			NO		Yes	Yes
VZ			NO		Yes	Yes
TW			NO		Yes	Yes
NH-II			NO		Yes	Yes
NJ			NO		Yes	Yes
WA			NO		Yes	Yes
TX			NO		Yes	Yes
JJ	PR11	RB	NO	SR	NO	N/A
XH			NO		NO	N/A
SAN-II			NO		NO	N/A
JK			NO		NO	N/A
XI			NO		NO	N/A
SAO			NO		NO	N/A
CL	PR13	RB	NO	SR	NO	N/A
CR			NO		NO	N/A
HH			NO		NO	N/A
KE			NO		NO	N/A
BW			NO		NO	N/A
CK			NO		NO	N/A

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TRAYS PENETRATING WALLS WITHIN
REACTOR BLDG.
Unit #2 El. 593

COMPUTED ____ DATE

TRAY	El. + C.L.	Side X	[illegible]	Side Y	[illegible]	[illegible]	[illegible]
KKF-I	QR9	RB	NO	RB	NO	N/A	—
MF-I			NO		NO	N/A	—
VL			NO		NO	N/A	—
TM			NO		NO	N/A	—
KKF-I	SR9	RB	NO	RB	NO	N/A	—
MF-I			NO		NO	N/A	—
VL			NO		NO	N/A	—
TM			NO		NO	N/A	—

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TRAYS PENETRATING WALL BETWEEN
SPREADING ROOM & TURBINE BLDG.
Unit #2 El. 663 & El. 586

TRAY	El. + [illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	
KKF	NR9	SR	Yes	TB	Yes	Yes	[illegible]
MY			Yes		Yes	Yes	NO
LU			Yes		Yes	Yes	NO
VS			Yes		Yes	Yes	NO
GE			Yes		Yes	Yes	"
MZ			Yes		Yes	Yes	"
LP			Yes		Yes	Yes	"
TS			Yes		Yes	Yes	"
NS	NR11	SR	Yes	TB	Yes	Yes	"
WB			Yes		Yes	Yes	"
NT			Yes		Yes	Yes	"
TZ			Yes		Yes	Yes	"
CL	NR12	SR	NO	TB	NO	N/A	
CR			NO		NO	N/A	
HY			NO		NO	N/A	
CB			NO		NO	N/A	
HD			NO		NO	N/A	
BW	NR13	SR	NO	TB	NO	N/A	
CK			NO		NO	N/A	
HB			NO		NO	N/A	
PU			NO		NO	N/A	
WN			NO		NO	N/A	
PG			NO		NO	N/A	
PH			NO		NO	N/A	
PJ			NO		NO	N/A	
PK			NO		NO	N/A	
PL	NR14	SR	NO	TB	NO	N/A	
PM			NO		NO	N/A	
PA			NO		NO	N/A	
PC			NO		NO	N/A	
WG			NO		NO	N/A	
PD			NO		NO	N/A	
PF			NO		NO	N/A	

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TRAYS PENETRATING [illegible]
El. 573 + El. 585
Unit #2 REACTOR BLDG. COMPUTED

TRAY	El. + [illegible]	573 [illegible]	[illegible]	5[illegible] [illegible]	[illegible]	[illegible]
NK-I	PR9	RB	NO	RB	NO	N/A
NL			NO		NO	N/A
VE			NO		NO	N/A
TF			NO		NO	N/A
PQ	PR12	RB	NO	RB	NO	N/A
KJC	PR12	RB	NO	RB	NO	N/A
VF	PR12½	RB	NO	RB	NO	N/A
TF			NO		NO	N/A
OG-II			NO		NO	N/A
OF			NO		NO	N/A
EY-I	UR10	RB	NO	RB	NO	N/A
LF-I	UR10	RB	NO	RB	NO	N/A
GU-II	RR13	RB	NO	RB	NO	N/A
BU-II			NO		NO	N/A
SAM-II			NO		NO	N/A
MAJ-I			NO		NO	N/A
BJ	SR10	RB	NO	RB	NO	N/A
BY	TR12	RB	NO	RB	NO	N/A

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[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
GD	[illegible]	[illegible]	[illegible]	RB	[illegible]	N/A
MH	[illegible]	RB	NO	RB	NO	N/A
[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	N/A
[illegible]	[illegible]	[illegible]	[illegible]	[illegible]		N/A
BW	[illegible]	RB	[illegible]	[illegible]	[illegible]	[illegible]
CL	PR13	RB	[illegible]	[illegible]	[illegible]	N/A
GC	PR13	RB	NO	RB	NO	N/A
RP	TK10	RB	[illegible]	RB	[illegible]	N/A
BJ	TK10	RB	[illegible]	RB	[illegible]	N/A
EX	HR10	RB	NO	RB	NO	N/A
LE			NO		NO	N/A
TM			NO		NO	N/A
[illegible]	[illegible]	RB	[illegible]	[illegible]		[illegible]
[illegible]	PR10	RB	NO	[illegible]	[illegible]	N/A
GH	PR13	RB	NO	[illegible]	[illegible]	N/A

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BEST AVAILABLE COPY

TRAYS PENETRATING [illegible]
EL. 659 & El. 631
Unit #2 Reactor B[illegible]

COMPUTED ____ DATE

TRAY	El. & Col.	Side X (631)	[illegible]	Side Y (639)	[illegible]	[illegible]
FU	SR9	RB	NO	RB	NO	N/A
NN	SR9	RB	NO	RB	NO	N/A
FU	SR13	RB	NO	RB	NO	N/A
NN	SR13	RB	NO	RB	NO	N/A
MO	SR13	RB	NO	RB	NO	N/A
BW	SR10	RB	NO	[illegible] 2A	NO	N/A
BY	SR11	RB	NO	[illegible] 3A	NO	N/A
BJ	SR12	RB	NO	[illegible] 2B	NO	N/A
NAD	UR11	R3	NO	[illegible] 25-23	NO	N/A
NAE	UR12	R3	NO	[illegible] 25-24	NO	N/A
IE	UR9	RB	NO	[illegible] 2B	NO	N/A

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EL. 606 (Spreading Room) to EL. 573 (Control Bldg)

Unit # 2

COMPUTED

TRAY	El. & Col.	[illegible]	Flammable [illegible]	[illegible]	[illegible]	Cable Repaired	[illegible]
KEA		SR	NO	Spreading Room	NO	Yes	NO
KEE-II			Yes		NO	Yes	NO
KEI			Yes		NO	Yes	NO
KEM			Yes		NO	Yes	NO
VAC			Yes		NO	Yes	N
KEB-II			Yes		NO	Yes	NO
KEF-II			Yes		NO	Yes	
KEJ			Yes		NO	Yes	
KEN	NR8-NR9		Yes		NO	Yes	
KER	&		Yes		NO	Yes	
KEC-I	PR8-PR9		Yes		NO	Yes	
KEG-I			Yes		NO	Yes	
KEK			Yes		NO	Yes	
KEO			Yes		NO	NO	—
KES			Yes		NO	Yes	
KED-I			Yes		NO	Yes	
KEY			Yes		NO	Yes	
KEH			Yes		NO	Yes	
KEL			Yes		NO	Yes	
KEP			Yes		NO	NO	—
KET			Yes		NO	NO	—
KEA	NR10	SR	Yes	Batt. Bd. No. 2	NO	Yes	
NAL	PR7	Wall Inside Aux Inst Room	NO	Wall Inside Comp. Room	NO	N/A	—
TAM	PR7		NO		NO	N/A	—
KEU	PR12	SR	NO	Relay Cab.	NO	N/A	—
KEV	PR12	SR	NO	Relay Cab	NO	N/A	—
KEW	PR13	SR	NO	Relay Cab	NO	N/A	—
VAE	PR13	SR	NO	Relay Cab	NO	N/A	—

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Trays Penetrating Wall Between
Spreading Room #2 + Spreading Room #3
Unit #2 El. 606

Tray	El. + Col.	Side X	[illegible]	Side Y	[illegible]	Cable Repaired	[illegible]
AH-II	NR11	SR #2	Yes	SR #3	NO	Yes	No
BU			Yes		NO	Yes	No
SU			Yes		NO	Yes	No
SH			Yes		NO	Yes	No
WH			Yes		NO	Yes	No
TO			Yes		NO	Yes	No
[illegible]			Yes		NO	Yes	No
RT			Yes		NO	Yes	No
SC			Yes		NO	Yes	No
SW			Yes		NO	Yes	No
VH			Yes		NO	Yes	No
UL			Yes		NO	Yes	No
PL	NR11	SR #2	Yes	SR #3	NO	Yes	No
PM			Yes		NO	Yes	No
SY			Yes		NO	Yes	No
JG	NR11	SR #2	Yes	SR #3	NO	Yes	No
JH			Yes		NO	Yes	No
JI-I			Yes		NO	Yes	No

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TRAYS [illegible]
PANEL IN CONTROL #2
UNIT #2 — EL 117 + EL 606 — COMPUTED ____ DATE ____

Pnl. No.	El. & Col.	[illegible] X	[illegible]	[illegible] Y	[illegible]	Cable Repulled	[illegible]
* 9-23	NR8	SR	Yes	CR	Yes	NO	—
9-24	NR9		Yes		Yes	NO	—
9-21	NR9		Yes		Yes	Yes	NO
** 9-8	NR10		See Below		Yes	Yes	NO
9-7	NR10		Yes		Yes	Yes	Yes
9-6	NR10		Yes		Yes	Yes	Yes
** 9-5	NR10		See Below		Yes	Yes	N
9-4	PR10		Yes		Yes	Yes	N
9-3	PR10		Yes		Yes	Yes	N
9-2	PR10		Yes		Yes	NO	—
9-10	PR9		Yes		Yes	NO	—
9-11	PR9		Yes		Yes	NO	—
9-12	PR9		Yes		Yes	Yes	N
9-13	PR9		Yes		Yes	NO	—
9-14	PR8		Yes		Yes	NO	—
9-53	PR8		Yes		Yes	NO	—
9-47	NR11		Yes		Yes	Yes	[illegible]
9-46	NR11		Yes		Yes	NO	—
9-25	PR11		Yes		Yes	NO	—
9-9	PR11		Yes		Yes	Yes	[illegible]
9-22	NR9		Yes		Yes	NO	—
9-20	NR9		Yes		Partial	Yes	N
9-44	NR11		Yes		Yes	Yes	Ye

* Cabinet C3-6 not treated with flamastic in S.P. ceiling
** one tray spout not treated with flamastic in S.P. ceiling

In General inside and under Control Room Panels Flamastic application was applied to this.

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INTERIM REPORT

MATERIALS FLAMMABILITY TESTING

FOR

NUCLEAR REGULATORY COMMISSION

April 10, 1975

W. A. RIEHL

Deputy Chief (EH31)
Non-Metallic Materials Division
Materials and Processes Laboratory
Marshall Space Flight Center

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ACKNOWLEDGEMENTS

The ASTM flammability tests herein were conducted by Messrs. John Austin, Lelan Sanders, and Everett Price of the Analytical and Physical Chemistry Branch. Sample preparation was done by Messrs. Bill White and Tom Wood of the Polymers and Composites Branch, and technical consultation was provided by Mr. Frank Key, all of the Non-Metallic Materials Division (EH31, MSFC, NASA).

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I. INTRODUCTION

On March 22, 1975, a fire occurred at the Browns Ferry Nuclear Power Plant of the Tennessee Valley Authority. Pursuant to direction from NASA management, the writer is providing technical expertise and support in the area of materials combustion phenomena to the Nuclear Regulatory Commission on this incident.

It was reported orally to the writer that the fire originated when a construction worker was checking for leaks with a candle and ignited a "plastic foam material" in an electrical cable tray.

The fire location and results were toured on March 24 and 26. Preliminary samples were obtained on March 24 and tested on March 25. Additional material samples were obtained March 28 and tested by ASTM procedures.

Test results, evaluation, and a summary of preliminary findings are described herein.

II. DESCRIPTION OF MATERIALS AND USAGES

Based upon discussions with TVA and NRC personnel, and inspection of both the burn site and an adjacent similar area which was not involved in the fire, the following configuration and materials usages are assumed in this report.

Figure 1 shows an overall simplified depiction of the cable trays which penetrate the concrete wall. Foam rubber apparently was being used to initially stuff the large openings between the cables, tray, and walls, and thus prevent the subsequent pour foam from being sucked through the wall. Either pour or spray foam was then used to provide a thorough and permanent seal. A fire retardant coating was then applied over the pour or spray foam.

In order to check for adequacy of sealing with the stuffing, a candle was used to determine leakage points by the flicker of the flame in the draft. A negative pressure was maintained on the downstream (reactor) side of the wall for safety considerations.

During this leak checking, the candle ignited the stuffing material (a foam rubber) which initiated the ensuing conflagration.

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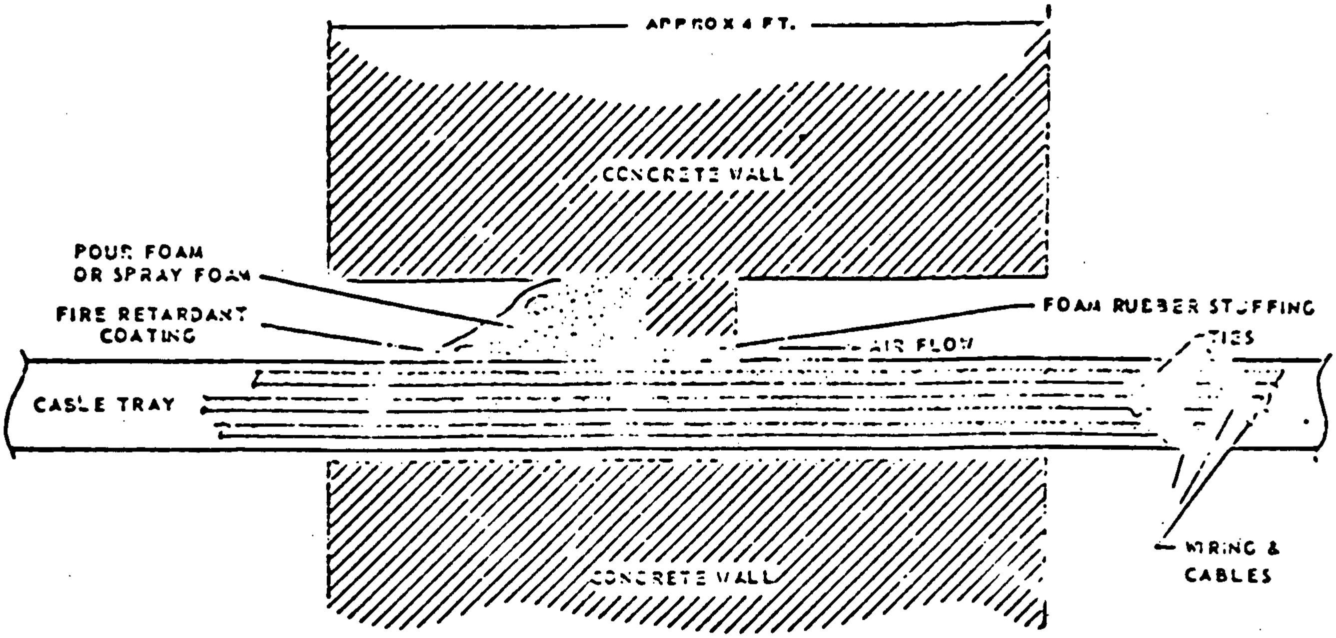
FIGURE 1

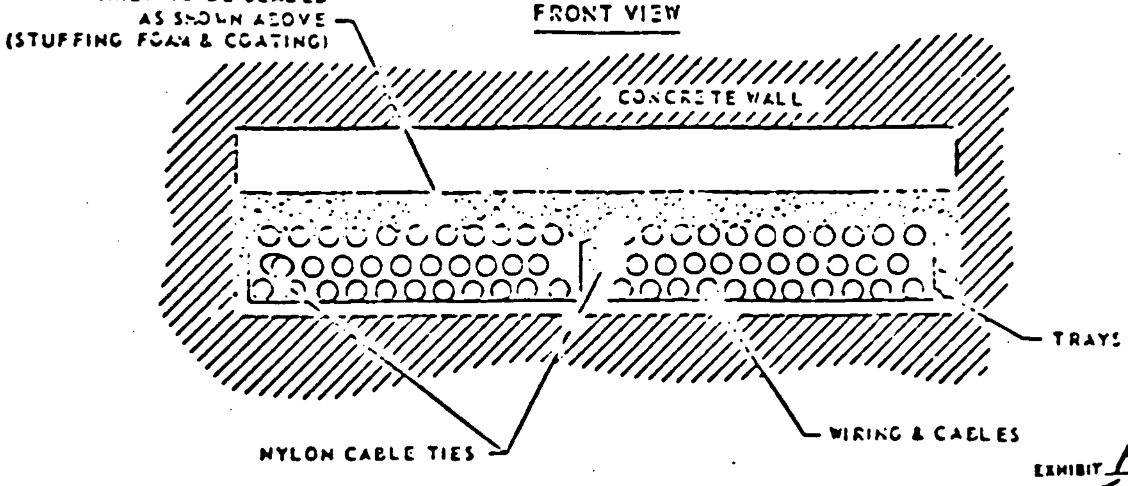
CABLE TRAY MATERIALS & USAGES

(OVERALL SIMPLIFIED DEPICTION – NOT TO SCALE)

SIDE VIEW



FRONT VIEW



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Inspection of the closest similar cable tray penetration through the wall which was not involved in the fire disclosed use of cotton rags as stuffing material, rather than foam rubber, and only sparse coverage of the foam with the fire retardant coating. Other miscellaneous materials of interest include the nylon cable ties, silicone caulking compounds, and a firewall board material.

In another location, the cavity was completely filled with foam, shaved off and then capped with the firewall board, and sealed at cable interfaces with flame retardant coating.

Observation of the cable trays in the cable spreader room also disclosed paper, polystyrene foam plastic pieces, pour foam, cotton and lace slip rags in between the cables in the trays.

It is important to note that three types of polyurethane foam are involved: a foam rubber stuffing, "Aire Lux;" a pourable version (Pittsburgh Chemicals "Selectrofoam"); and a spray-applied version ("Instafoam"). The spray version is reported in the manufacturer's literature as "Self-Extinguishing (according to ASTM-1692-59T)." (Appendix I, page 4)

III. CANDLE IGNITION OF FOAM RUBBER

A sample of foam rubber was obtained from the spreader room on March 24 which, according to a worker therein, was the same type material used for stuffing for initial sealing. A cursory match test on a piece of this disclosed almost instantaneous ignition, very rapid burning, and release of molten flaming drippings.

Subsequently, the sample was cut into two pieces approximately 10" x 2" x 1". A candle was also picked up in the spreader room, by the writer, approximately 15 to 20 feet from the fire initiation site on March 24. This candle was lit and applied to the bottom of the samples, and motion pictures made of both tests. The film was provided to NRC (C. E. Murphy) on March 28. The entire sample was consumed in a few seconds, with flaming drippings.

Subsequently, further tests were done to obtain some indication of ease of ignition and proximities required. Figure 2 shows the test set-up. The candle to sample distance was measured. The flame height varied slightly due to natural flicker, but was roughly up to 2.5 cm from the candle to top of the tip.

Table I shows that ignition occurred almost instantaneously 0.5 cm from the flame tip, and in 2-1/2 seconds at 2.5 cm (approximately 1 inch) away. At greater distances the sample did not ignite, but melted through.

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FIGURE 2

CANDLE IGNITION

TEST SETUP

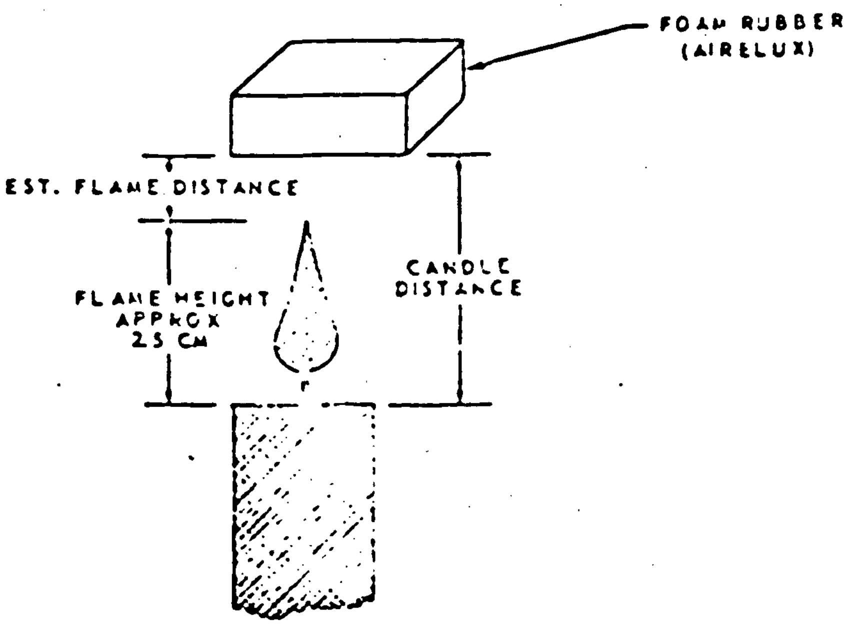

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TABLE 1

CANDLE IGNITION CHARACTERISTICS

DISTANCE (cm)		TIME TO IGNITION (sec.)	COMMENTS
CANDLE	EST. FLAME		
3	0.5	Instantaneous	
5	2.5	2.5	
5.5	3.0	None	Melted through in 45 sec.
6	3.5	None	Melted through in 50 sec.

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Thus, actual contact with the flame was not required, and ignition readily could occur up to 2.5 cm (1 inch) away.

IV. MATERIALS FLAMMABILITY TESTING

A. Initial Cursory Testing - March 25, 1975

At the initial site inspection tour, several material samples were collected on site, most of which were from the closest cable tray penetration through the wall not involved in the fire, but reportedly similar. These included random available samples of the stuffing, pour foam, fire retardant caulking compound, cable ties, fire board, and rags, as listed in Table 2.

In order to obtain a "quick look" assessment of materials flammability characteristics with then available samples, simple match tests were made by holding a small lit match to the bottom of the samples. Results confirmed that most of the materials were readily flammable, and disclosed drip burning of both the polyurethane foam rubber and nylon ties. In these tests, the pour foam was completely consumed (no self-extinguishing).

At first impression, one might be alarmed at the use of cotton rags as stuffing. However, the tests showed that this is a far better choice than the foam rubber; not only was burning much slower, but of lower intensity and without burning droplets being released.

For simple comparative purposes, the writer took a sample of foam rubber from a pillow at his residence and obtained the same results as with that of the stuffing from the cable spreader room.

B. ASTM Tests

The latest available edition (1973) of the Test Methods of the American Society of Testing Materials lists two types of flammability tests for "cellular plastics" such as polyurethane foam materials, i.e., ASTM D-1692 and D-3014.

The most appropriate method for evaluation of the flammability of wiring appears to be a vertical flame test as specified in paragraph 30 of Method D-2633.

Upon obtaining better identified and adequate materials from which the specified samples could be prepared, ASTM tests were initiated on three polyurethane foams, four other construction

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TABLE II

CURSORY* MATERIALS BURNING TESTS - MARCH 25, 1975

MATERIAL			BURNING TESTS*	
USAGE	SOURCE	DESCRIPTION	SAMPLE SIZE	RESULTS
Stuffing	Worker in Cable Spreader Room	Polyurethane Foam Rubber	1" x 1/4" x 1"	Very rapid flash burning with dripping fiery droplets -- sooting.
Pour Foam	Adjacent Cable Tray	Pittsburgh Selectrafoam	1/4" x 3/4" x 1"	Rapid complete burn with heavy soot - no droplets
Fire Retardant Coating	Cable Spreader Room	Flamemastic	1/4" x 1/16" x 1"	No ignition or burning
Caulking	Adjacent Cable Tray	RTV-102 Silicone	1/4" x 1/2" x 1/4"	Very slow but intense burning - self extinguished
Cable Ties	Cable Spreader Room	Nylon	1/8" x 1/32" x 2"	No upward propagation but drips burning droplets
Cotton Rag	Stuffing in Adjacent Cable Tray	Red Denim	1/2" x 1"	Ready burning - no droplets
Pillow	Home Residence	Polyurethane Foam Rubber	1/2" x 1" x 2"	Flash burning - burning droplets
Firewall	Wall Penetration in Stairway	Inorganic Board	1/8" x 1/2" x 1"	No ignition or burning

*Bottom ignition with match in air.

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materials, and thirty wire and cable samples. Those materials found most flammable were then coated with the fire retardant coating and retested.

1. Polyurethane Foam Materials

As mentioned previously, three polyurethane foam materials were associated with this incident, one of which was reported in the manufacturer's literature as "Self-Extinguishing (according to ASTM-D-1692-59T)." That material was the spray foam version (Instafoam) and the most expensive.

ASTM D-1692-59T has been superseded by D-1692-68 (reapproved in 1973). This is essentially a horizontal burning test in which a 6" x 2" x 1/2" sample is ignited and reported as "self-extinguishing" if the samples do not burn past a 5-inch gage mark in a 6-inch length after exposure to a wingtip bunsen burner flame (Figure 3).

Another ASTM test for "Flammability of Rigid Cellular Plastics" is D-3014. This is essentially a vertical burning test in which the samples are ignited at the bottom. Prior experience with flammability testing of spacecraft materials has shown that this type of test (vertical with bottom ignition) is far more severe than horizontal burning tests. In this test 3/4" x 3/4" x 10" specimens are ignited with a gas burner in a 2-1/2 inch diameter chimney (Figure 4). Materials are not reported as "self-extinguishing" by this test, but burning time, extent of burning and other characteristics are reported.

In these tests, the chimney was eliminated in order to allow motion picture filming. It would be expected that the chimney could provide slightly more severe test conditions (upward draft). Material identifications and test results are listed in Table III.

All three of these materials readily burned; thus, samples of all were coated with "Flamemastic 71" and retested in the vertical position (D-3014). Those which did not ignite therein were not tested by D-1692 (the less severe test).

2. Other Construction Materials

ASTM tests on other construction materials were run in the same manner as those for cellular plastics in order to provide a basis of relative comparison (i.e., ASTM-D-1692 - horizontal, and ASTM-D-3014 - vertical, burning tests). Results are provided in Table IV. Motion picture coverage also was made on these tests.

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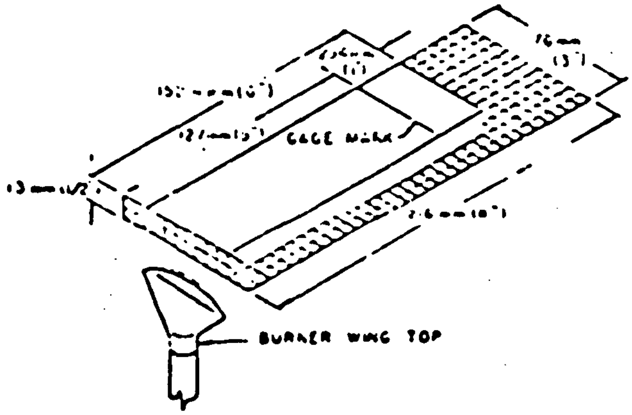

Figure 3 - ASTM D-1692 - Horizontal Test Setup

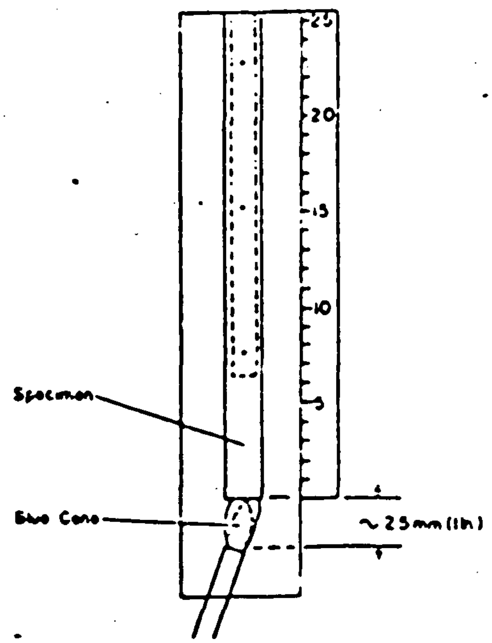

Figure 4 - ASTM D-3014 - Vertical Test Setup

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TABLE III

POLYURETHANE FOAM TESTS

MATERIAL	SOURCE	RESULTS			
		ASTM 1692 HORIZONTAL		ASTM 3014 VERTICAL	
		TEST NR. & TRIALS		TEST NR. & TRIALS	
Foam Rubber	"Aire-Lux" South Aire Co. 1116 So. Canal St. Tupelo, Miss.	592 (2)	Sample burned entirely in 30 sec. - drip burning	604 (2)	Sample burned entirely in 10 sec. - drip burning
Pour Foam	"Selectrafoam - Code 6409" Urethane, Pourable Type (Part A-0293, B-67010) Pittsburgh Paint Co.	590 (3)	Burned to 5", 3", 5" in 40-52 sec., then self extinguishing	598 (3)	Flash burning - 20 - 30 sec. - 8 - 10", then self extinguishing
Spray Foam	"Instafoam 120" Insta-Foam Products, Inc. Joliet, Illinois	591 (3)	Burned to 5", 4-1/2", 5" in 64 - 83 sec. then self ext.	597	Flash burning - 20 sec - 9", then self ext.
Foam Rubber Flamemastic Coated	Prepared by NASA/MSFC		Not necessary to test	603 (2)	No ignition first try; rapid self-extinguishin on 2nd try (both sample
Pour Foam Flamemastic Coated	Prepared by NASA/MSFC		Not necessary to test	599 (2)	No ignition
Spray Foam Flamemastic Coated	Prepared by NASA/MSFC		Not necessary to test	600 (2)	No ignition

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TABLE IV

OTHER CONSTRUCTION MATERIALS TESTS

MATERIAL	SOURCE	RESULTS: ASTM 1692 HORIZONTAL — TEST NR. & TRIALS	ASTM 1692 HORIZONTAL	ASTM 3014 VERTICAL — TEST NR. & TRIALS	ASTM 3014 VERTICAL
Fire Retardant Coating (on Fiberglass cloth) 2" x 6"	"Flame Mastic 71A" Dynatherm Corp. 598 West Ave. 26 Los Angeles, Calif.	594 (2)	No ignition	595 (2)	No ignition
Fire Board	"Marinite #36 - Type B" Johns-Manville Corp.		Not required	601 (3)	No ignition
Nylon Ties	"TY-RAP" Cable Ties TY-525M Thomas & Betts Co.	-	-	602 (3)	Drip burning Self-extinguishing above flame
Caulking Silicone Rubber 1/8" thick-cured	RTV-102 General Electric Co. Waterford, N. Y.	593 (2)	Samples burned with flame) 2 - 3 in. 2 - 3 min.	596 (3)	Self-extinguished in 2 - 3"

515

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3. Wiring

In the ASTM Test (D-2633, p... ...h 30) o.. ...re and cable, approximately a 56 cm (22 in.) lon... ...e is vertically positioned and a gas test flame applied four times for 15 second intervals.

Thirty wires and cables were provided by TVA as representative of the type involved in the incident. Test results are shown in Table V.

Only two wires burned completely and drip burned by this procedure: IWA #12 and IWA #14-AWG-TVA. Only enough of the latter was supplied to test after coating with Flamemastic (Test Nr. 31). (Additional materials have been requested for further tests.)

Over half (18) of the wires did not ignite in this test. The remaining third (10) supported combustion, including flames, charring, and/or severe smoking, but self-extinguished prior to consumption of the entire length of the wire.

V. PRELIMINARY EVALUATION OF RESULTS

Experimental tests clearly verified the ease of ignition of the foam rubber stuffing by the candle. (In fact, actual contact with the flame is not required.) The resulting very rapid, almost flash, burning combined with release of burning droplets constitutes not only an intense local source of ignition, but also a means of propagation of fire over a much larger area, leading easily to a general conflagration with other local combustible materials, especially in an air draft as actually occurred.

Initial cursory tests on materials collected in the cable spreader room confirmed that readily combustible materials were in the vicinity: rags, pour foam, and cable ties.

Interpretation of the ASTM test results must be done with caution. These are intended as relative tests only and done in a draft-free environment in a strictly empirical test procedure.

For example, the manufacturer's claim that the "Instafoam" is "self-extinguishing" was experimentally substantiated by testing in accordance with the referenced ASTM specification (D-1692). However, the data on both the spray and pour foam samples show that the materials do very barely meet the requirements to be rated as "self-extinguishing by this test." Specifically, the requirement is that in this horizontal test no specimens burn past a 5-inch gage mark from the ignited end. Inspection of the data shows burn lengths

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TABLE V

FIRING TESTS

PER ASTM D-2633, PARAGRAPH 30

PASS	TEST NR.	COMMENTS
~~190~~-54753-TVA	1	Supported combustion
WLC-#14-AWG	3	Supported combustion
WMB-2/0-AWG-TVA	4	Supported combustion
WDD-#8-AWG-TVA	5	Supported combustion
WDD-1/0-AWG-TVA	6	No ignition
WVA-#16-AWG-TVA	7	Supported combustion Sparks and smoke
WUB-#16-AWG	8	Supported combustion
[illegible]-TVA	10	No ignition
WUB-#16-AWG	11	No ignition
WVB-#16-AWG-TVA	12	Slow combustion (50-56 sec. @)
WVA-#16-AWG-TVA	13	No ignition
WVA-1-#18-AWG-TVA	14	No ignition
WUB-1-#16-AWG-TVA	15	No ignition
WVB-#16-AWG	16	No ignition
WVA-1-#18-AWG-TVA	17	No ignition
WVC-#16-AWG-TVA	18	No ignition
WVU-1-#18-AWG-TVA	19	No ignition
WGD-#12-AWG-TVA	20	No ignition
WVR-#20-AWG-TVA	21	Smoked (23-31 sec. @)
WLO-#10-AWG	22	No ignition
WVG-#16-AWG-TVA	23	No ignition
PJJ-#12-AWG-TVA	24	No ignition
[illegible]-#14-AWG	25	No ignition
PJJ-#10-AWG	26	No ignition
WVA-1-#18-AWG	27	No ignition
WVI-#16-AWG	28	No ignition
WVJ-#16-AWG-83849-TVA	29	Char burned and smoked - 2 min.
WVJ-#16-AWG-74910-TVA	30	Smoked - 20 sec. @
WMA-#14-AWG-Flamemastic coated	31	No ignition on 3 tries, self extinguished on 4th try

FAILED

WMA #12	2	Drip burned
WMA #14-AWG-TVA	9	Drip burned

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of 5", 3", and 5" for the pour foam and 5", 4-1/2", and 5" for the spray foam. One could infer from these data that the 5-inch limit may have been derived from these type materials, and thus the test was designed to accept such materials. The same inference could be drawn from the ASTM vertical burning test (D-3014) in which a 10-inch long specimen is specified. The data show burn lengths of S to 10 inches.

However, the lead paragraph of both ASTM specifications states: "This method should not be used solely to establish relative burning characteristics and should not be considered or used as a fire hazard classification" and further therein, "Correlation with flammability under use conditions is not implied."

Clearly, both materials are readily ignited, support combustion and exposed surfaces would contribute significantly in a general conflagration.

The data do show that the polyurethane foam rubber burns much faster than the pour or spray foams, and releases burning droplets. Further, these samples of pour foam burn considerably faster than the spray foam. In addition, coating exposed surfaces with Flamemastic was extremely beneficial. In fact, coated pour and spray foam samples did not burn under the test conditions.

The data on the other construction materials indicate that these are a considerably less flammability hazard than the polyurethane foams. The fire preventive type materials, fire board and fire resistant coating, are inherently adequate (did not burn). The silicone caulking burned slowly and self extinguished in much less than 5 inches. However, the nylon ties, as characteristic of this material, do drip-burn, and thus would contribute to spread of a fire.

Only two of the 30 wires and cables tested failed ASTM Test D-2633, paragraph 30. The effectiveness of the Flamemastic coating was again well verified on one of these (other not yet tested). However, another third of the wires supported combustion, and thus would contribute in a general conflagration.

Extrapolation of this ASTM test data on wiring should probably be taken with more caution than discussed previously when attempting to apply to fire hazards in use configurations. In particular, operational use (carrying electricity) would drastically affect flammability characteristics. Further, the ignition source effect may be more pronounced here, i.e., some plastic insulations, once ignited, could serve as more intense initiators than open gas flames.

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Orientation effects also would be expected to be more significant with wiring and cables than with construction materials. Thus, a use configuration type of flammability test is considered of considerably greater merit for electrical wiring than for general construction materials.

VI. INTERIM FINDINGS - SUMMARY

Inspection of the fire site and an adjacent similar area and flammability testing of materials collected in the vicinity and provided subsequently by TVA appears to support the following findings:

1. A candle or match flame easily ignites the foam rubber stuffing material.

2. Ignition of the foam rubber results in very rapid combustion with release of burning droplets, thus constituting an intense local source and means of rapid propagation to other local combustible materials.

3. Other readily combustible materials present in the general area (cable spreader room) included paper, rags, polyurethane foams (spray and pour), and polystyrene foam pieces.

4. Three types of polyurethane materials are of concern: the foam rubber, pour foam, and spray foam. The foam rubber is a considerably greater fire hazard than the other two.

5. Both the pour foam and spray foam would barely pass the ASTM (D-1692) flammability test requirement on these type materials.

6. The pour foam burns more rapidly than the spray foam.

7. The fire retardant coating (Flamemastic) is effective in drastically reducing the flammability of materials.

8. The fire board material (Marinite) also appears adequate for intended purposes.

9. Thirty wire and cable samples were tested by exposure of a vertical specimen to a gas flame in accordance with ASTM D-2633, paragraph 30. Two of the wires (LWA #12 and LWA #14-AWG-TVA) burned completely after removal of the flame, and drip burned.

10. Ten additional wires supported combustion to varying extents.

11. Eighteen wires did not ignite under these conditions.

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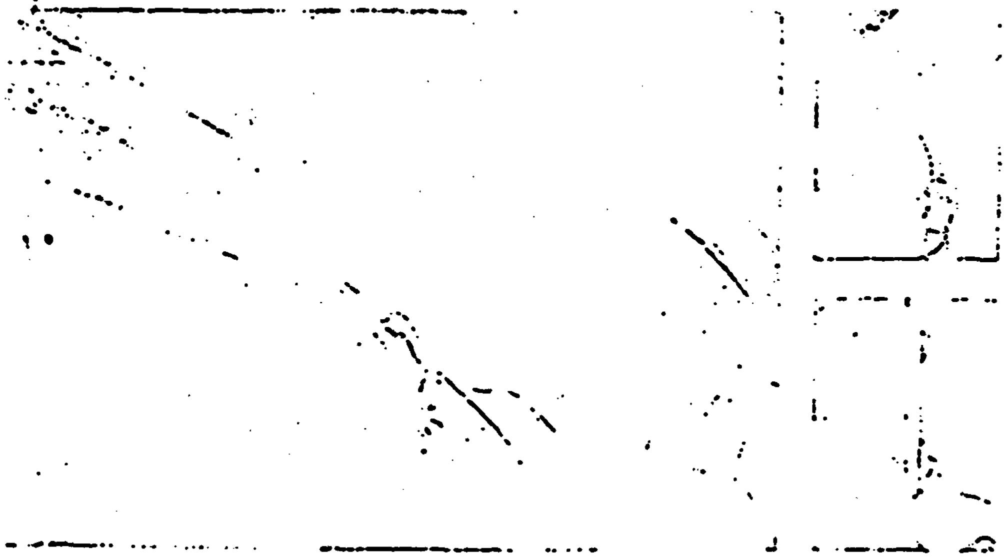
General Information & Operating Instructions

INSTA-FOAM®
FROTH PAK®
"120" - "300" - "500" UNITS

INTRODUCTION: The Insta-Foam FROTH PAK is an aerosol dispensing unit which contains the chemical components for making rigid urethane foam. When the unit is activated, high quality froth foam is dispensed forming a rigid product in less than a minute. The FROTH PAK is the practical and economical solution to the problem of foaming difficult areas and ideal as the small job unit or repair kit. Readily available, it can be used under any weather conditions.



MANY PRACTICAL APPLICATIONS

- Refrigeration & Air Conditioning Insulation
- Refrigerated Trucks & Cold Storage Rooms
- Aircraft Industry for Lightweight Repair and Insulation
- Marine Industry for Flotation Chambers, Boat Insulation and Structural Reinforcement
- Packaging of Delicate Instruments, Tools & Art Objects
- Fill Holes in Structures to Prevent Entry of Rodents or Birds
- In Conjunction with Pre-formed Pipe Insulation for Valves, Elbows and Fittings
- Seal Pipe Openings in Walls
- Repair Duct Work
- Gas-Electric-Telephone Utilities for Maintenance & Insulation
- Cementing Between Sheets of Urethane & Styrene Board Stock
- Insulation Contractors for Curtain Walls, Windows, etc
- Fill Large Holes Preparatory to Plastering or Paneling
- Mobile Home and Camper Repairs
- Display Industry for Design

- PLUS MANY MORE -

HIGH INSULATING EFFICIENCY
MINIMUM WATER ABSORPTION
LONG LIFE
SELF-EXTINGUISHING

HIGH DIMENSIONAL STABILITY
LOW WATER VAPOR PERMEABILITY
SELF-ADHERING
STRUCTURAL STRENGTH

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INSTA-FOAM PRODUCTS, INC. 820 S. FIENE DRIVE. ADDISON, ILL 60101

SET UP INSTRUCTIONS

The [illegible] "120" is self-contained in one carton consisting of two [illegible] cylinders. The larger "300" and "600" units contain cylinders in separate cartons which have [illegible]ccess openings to make the hose connections. All three units produce the same high quality urethane foam and operate identically. The chemicals must be at room temperature (70°) or above for effective application and have a storage life of one year at 100° F. or less. The three units are connected in the same manner and must be left in their shipping cartons when used. To connect the hoses...

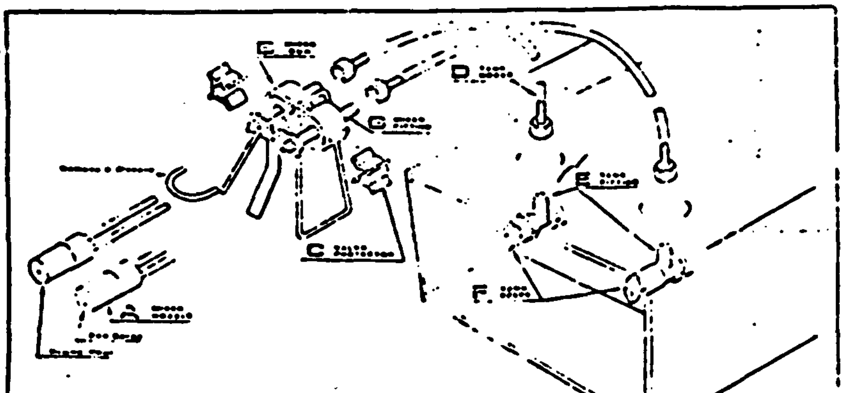
(1) Lay [illegible] unit [illegible] flat surface so Tank Valves (F) point [illegible] Hoses (D) to Tank Valve Fittings [illegible] to Mixer Adapter Fittings (G). Secure connections [illegible] with fingers, then tighten with pliers or wrench.

(2) Raise unit so Tank Valves (F) are at bottom. Open A-component tank valve slowly by turning counter-clockwise until material appears in hose. Stop and [illegible] for [illegible]. If none appear, open valve ALL THE WAY to stop. Repeat the same procedure with the B-component tank valve.

Caution: Do not attempt to open tank valves unless they are pointing downward as this may cause a loss in nitrogen pressure. Also, care must be taken that the valves are not turned in the wrong direction as this could release the safety vent in the tank valves.

(3) Remove U-shaped plastic shipping [illegible] from mixer gun (E) and discard. Detach Winged [illegible] Protectors (C) taking care not to tilt the [illegible] removal. Save for reuse. Before attaching [illegible] Nozzle (A), check the flow of the chemicals thru hoses by pulling trigger of mixer gun (E) ALL THE WAY, allowing material to spray into waste receptacle or onto ground. The flow in both hoses should have the same relative velocity. There should be no attempt to control output with the trigger, it MUST be pulled all the way back to effectively operate the unit.

(4) Select proper Mixer Nozzle (A) required for job. (ROUND POUR for filling holes or cavities... FAN SPRAY for foaming flat Surfaces). Replace Winged Valve Protectors (C) and attach Mixer Nozzle (A) to Mixer Adapter (G) by pushing ALL THE WAY DOWN onto connectors.
Do not attempt to pull trigger before Winged Valve Protectors have been removed. The unit is now ready for use.



FROTH-PAK® "120"

FROTH-PAK® "120"	FROTH-PAK® "[illegible]"	[illegible]OTH-PAK "[illegible]"
10 Cubic feet, 120 [illegible] feet	25 Cubic feet, 300 [illegible]	53 C[illegible] [illegible] Foo[illegible]
Consists [illegible]	Consists of	Consists of
1—Twin Pak carton [illegible]	2— [illegible]	2 [illegible]
2—Pressurized C[illegible] Tanks (1 "A" [illegible])	12 [illegible]	12 [illegible]
6—Mixer Nozzl[illegible] [illegible] (1,3 fan)	2 15' [illegible] I.D. [illegible]	2 [illegible] I.D. [illegible]
2—6' lengths [illegible] w 7, 16"-20 C[illegible]	Connec[illegible]	[illegible]
—Gun type dispe[illegible] Valve Protectors	1—Gun type dispenser [illegible]	1 [illegible] dispenser [illegible]
Carton—Shipping [illegible] 27 lbs.	3 Shipping Cartons	3 [illegible] Cartons—
	Shipping Weight 55 lbs.	Shipp[illegible] Weight 113 lbs.

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OPERATION

The FROTH PAK units are supplied with a gun-type dispenser which operates in the same manner as any spray gun. The valves are properly operated by pulling the gun trigger ALL THE WAY BACK.

The foam dispensing can be started or stopped, as desired, by pulling or releasing the trigger.

(1) Remove Winged Valve Protectors (C) from Mixer Adapter (C). Hold the Mixer Nozzle (A) approximately 12" from area to be coated for surface foaming, or directly into hole or crevice to be filled. For fan spray application, move nozzle rapidly back and forth to obtain even coverage. Always pull the trigger all the way back.

(2) There is a 30 second interval before the foam will start to set in the nozzle. Wipe excess foam from nozzle to prevent any interference with spray pattern.

(3) If use interval exceeds 30 seconds, nozzle must be replaced, or to retain use of existing nozzle, trigger gun into waste container to keep fresh material in the nozzle.

(4) All foam exposed to direct sunlight should be coated after curing.
(Insta-Foam Roof Kote recommended)

DISPOSAL OF EMPTY UNIT

Turn tanks upright so that Valves are at top. Open valves to allow pressure to vent. To avoid danger from incineration, loosen hex bonnet below valve stem which will vent the tank.

COLD WEATHER USE OF THE FROTH PAK

Unlike standard foam machines which are not used in cold weather, the FROTH PAK units will effectively operate outdoors under the following conditions:

(1) The chemicals have been stored at room temperature (70°) or above.

(2) The foam can be sprayed or poured at temperatures as low as -35° F. depending upon moisture or humidity. (Urethane foam will not adhere to a wet or frosty surface). The chemicals can remain outdoors for periods up to 4 hours at +35° F. before they become too cold to react properly.

(3) Insta-Foam may be poured at temperatures as low as -20° F. when used in a confined area where the mass will allow the foam to cure effectively the unit should not be exposed to temperatures below -20° F. for more than one hour.

NOTE: The use of the FROTH PAK at temperatures below +40° F. will result in a longer curing period for the foam. Before any attempt is made to move or cut the foam, ample time must be allowed for curing. Curing has taken place when the foam surface can be touched without any sign of crumbling or cracking. The lower the temperature, the longer the cure.

STORAGE AND REUSE

Contents should be used up within [illegible] valves have been opened.

STORAGE: Replace the Winged Valve Prot[illegible] used Mixer Nozzle in place. The hardened foam in the Mixer Nozzle will [illegible] from [illegible] Close tank valves. Store at room [illegible]

REUSE: Remove used Mixer Nozzle [illegible] [illegible]

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21922

BEST AVAILABLE COPY

TROUBLE SHOOTING GUIDE

Any noticeable color change in the foam is an indication of malfunctioning within the unit. Usually replacing the nozzle will correct this problem. If trouble persists, check the following:

(1) If the foam dispensed remains too soft and/or tacky after 2 minutes when used at room temperatures, a partial blockage of the A-component system is indicated. Remove Mix Nozzle and pull gun trigger directing streams of both chemicals into waste container or onto ground. Note the velocity of the A & B component streams, if approximately equal, the problem was in the nozzle. Merely replace with a new nozzle. If the stream velocities are unequal, the problem is in the gun or the A-component hose. If material foams or turns "creamy" in line, check that A-component valve is fully opened. Close B-component valve and continue to trigger gun with nozzle off in attempt to clear line. If restriction is in gun, close A-component valve and replace gun unit if blockage cannot be cleared.

(2) Before re-starting unit after prolonged shutdown or storage, purge the lines by running unit with nozzle removed until proper velocity is attained. This is the best indication of a properly pressurized clean unit.

(3) Never continue to operate unit if trouble occurs, as a drop in volume of one tank will eliminate any chance of reclaiming the unit.

TECHNICAL SPECIFICATIONS

©

GENERAL: FROTH-PAKS "120", "300", and "600"

All tanks have a baked phenolic coating over an iron phosphate metal base and a white enamel exterior coat. Tanks are fitted with a valve containing a spring loaded safety device which will automatically vent the tank at 425 PSI. Tanks are pressurized with nitrogen to approximately 200 PSI.

FROTH-PAK© "120"

Two 10 lbs. (water capacity) deep drawn steel tanks.
Maximum operating pressure 425 PSI, Minimum burst, 1000 PSI.
Total Chemicals, 17½ lbs.
Hose, two - 6 ft. 3/16" I.D. x ¼" O.D. tubing (250 PSI burst) with 7/16" - 20 connections.

FROTH-PAK© "300"

Two 26½ lb. (water capacity) deep drawn steel tanks.
Maximum operating pressure 425 PSI, minimum burst, 1000 PSI.
Total Chemicals 43½ lbs.
Hose, two 15 ft. ¼" I.D. x ⅜" O.D. tubing (250 PSI burst) with 7/16" - 20 connections.

FROTH-PAK© "600"

Two 51½ lb. (water capacity) deep drawn steel tanks.
Maximum operating pressure 260 PSI, Minimum burst 650 PSI.
Total Chemicals 87½ lbs.
Hose, two 15 ft. ¼" I.D. x ⅜" O.D. tubing (250 PSI burst) with 7/16" - 20 connections.

FROTH-PAK© URETHANE FOAM PROPERTIES AND PHYSICAL CHARACTERISTICS

Self-Extinguishing
(according to [illegible] 591)
Temperature [illegible]
-350° to [illegible]
Core density
1.75 lbs. [illegible]
"K" factor [illegible] .12
Tensile Strength [illegible]
Compressive Strength [illegible] 15 PSI.

Yield [illegible] - 5.3%
[illegible] test (-16°F)
-2 [illegible] 48 hours
Humid aging (158° F, 100% [illegible])
[illegible] after one week
Flex strength 40 [illegible] PSI
Closed Cells (%) - 90% [illegible]
Water vapor transmission - 2.0 perms per inch after two days

EXHIBIT A 6
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DISTRIBUTED BY:

U.F.N.P. - FIRE STOPS

COMPUTED GP DATE [illegible]

NOTES:

1. Distribute cables evenly on tray, fill large voids with inorganic thermal insulating wool or equal.

2. Seal opening between the pieces of marinite board with either Elastomastic CTA, Vimasco Rof [illegible] or equal.

3. Fasten marinite board to floor or wall with 1/4" [illegible] machine screw and cinch anchors, cover heads of screws with 1/8" of flame proofing compound.

4. Apply approx 1/8" to 1/4" flame proofing compound on both sides of marinite or steel sleeve and on the cable for 12" along both sides, top and bottom.

5. Provide mechanical protection to exposed vertical trays using one tray cover length 8' to 10' long; use ventilated covers on power trays.

6. Drill marinite board for conduit, cut out slots for installation then replace, sealing with fire proofing compound.

7. The following tray penetrations of floors and walls should be provide with fire stops
a. All trays entering the class I structures.
b. All tray penetrations of spreading room walls and floors.
c. All vertical tray slots through floors in the class I structure.
d. All other mandatory fire rated walls and floors.
e. All trays between [illegible] in [illegible]

EXHIBIT A7

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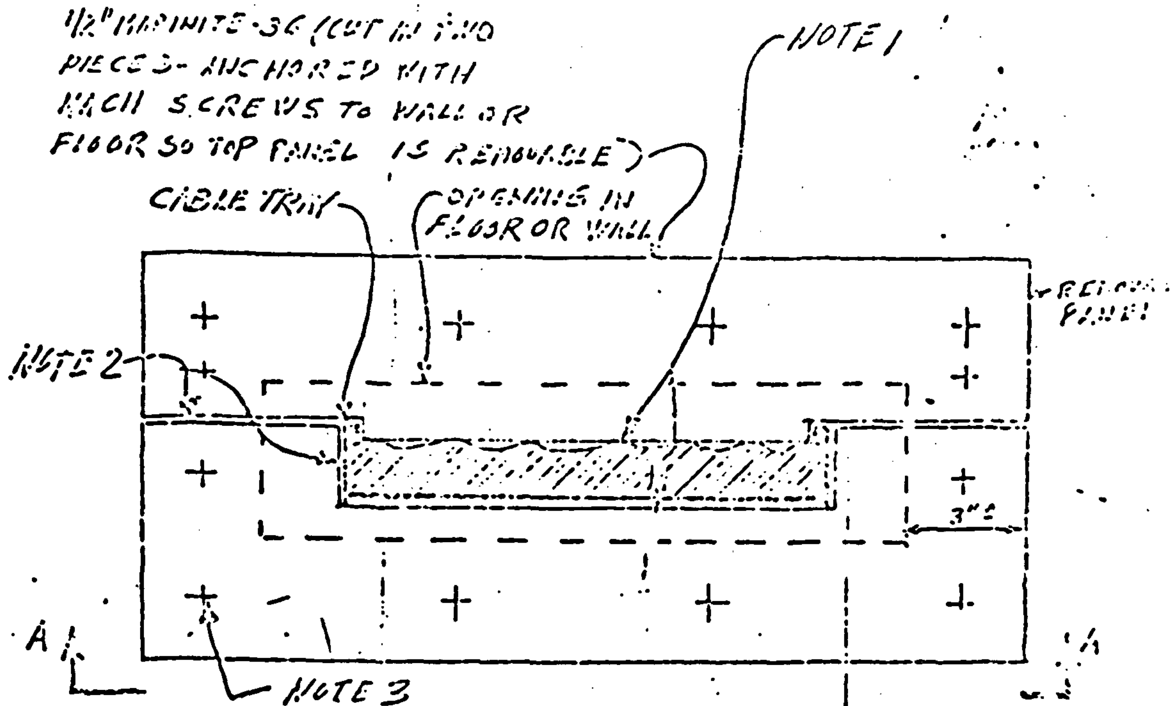
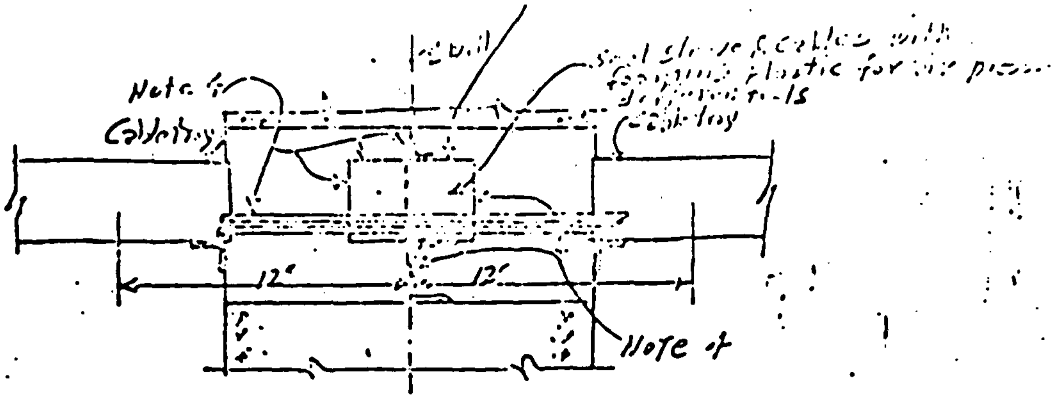


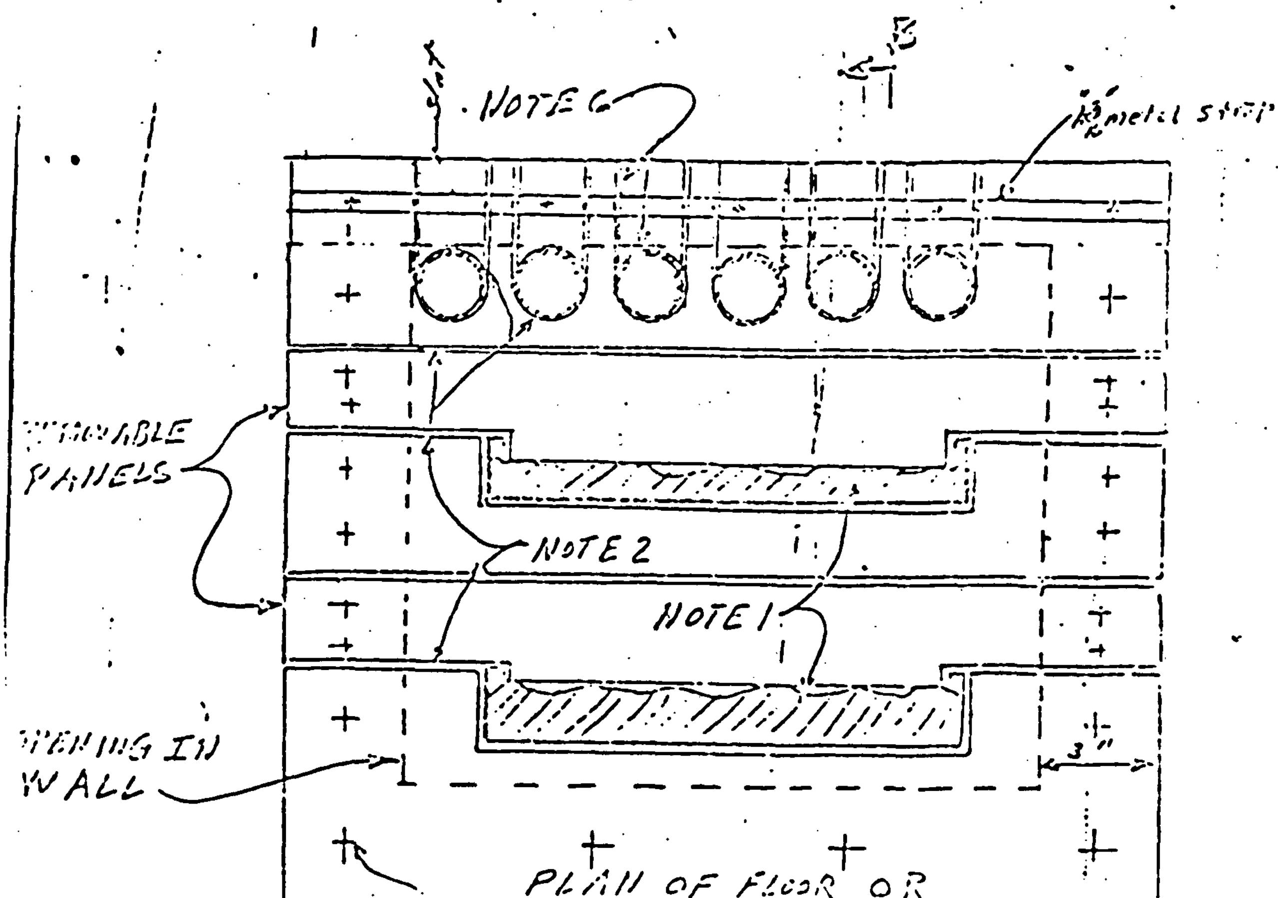
PLAN OR ELEVATION

FIRE STOP FOR CABLE TRAYS
THROUGH FLOOR OR WALL OPENINGS

EXHIBIT A7
PAGE 2 of 6



SIDE VIEW
CABLE SPREADING ROOM
TRAY PENETRATION



PLAN OF FLOOR OR
NOTE 3 / ELEV OF WALL PENETRATION

TYPICAL FOR TWO OR MORE TRAYS
AND CONDUITS

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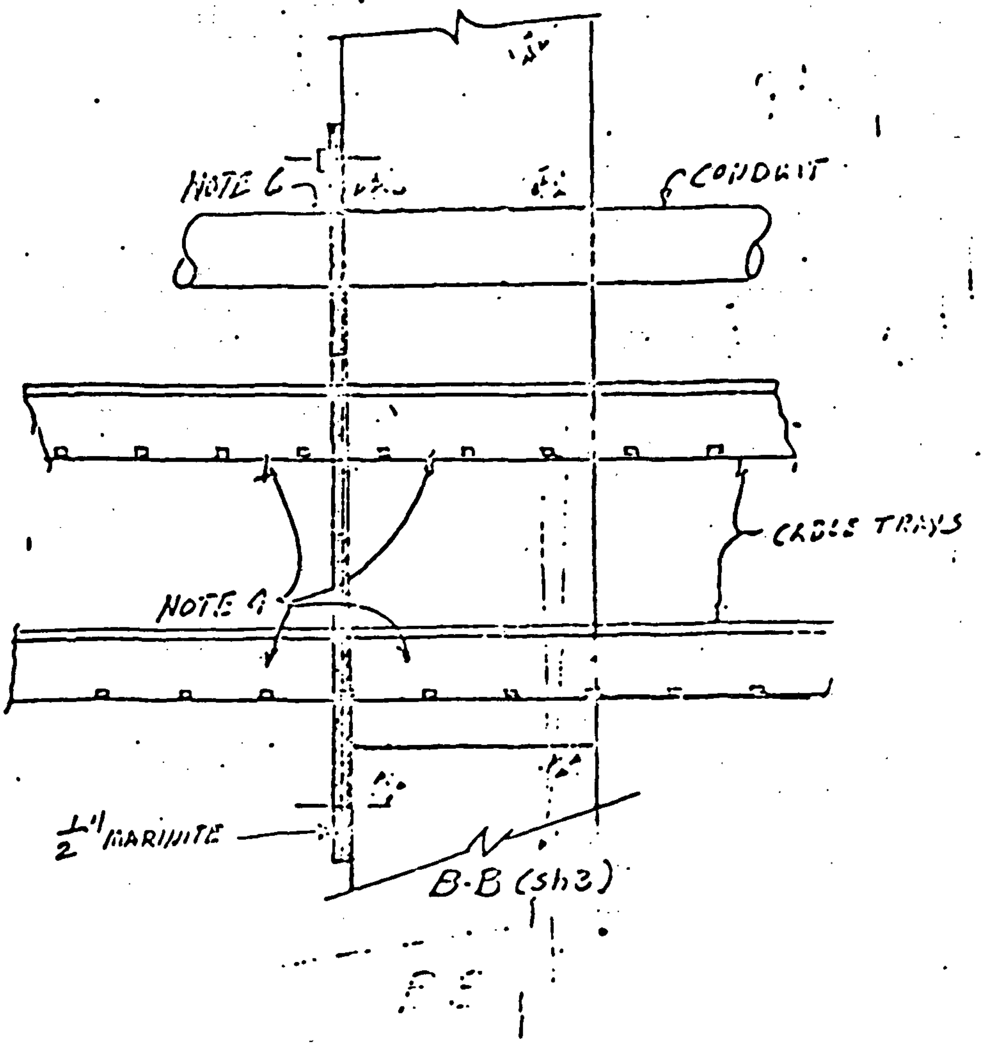
BFNP



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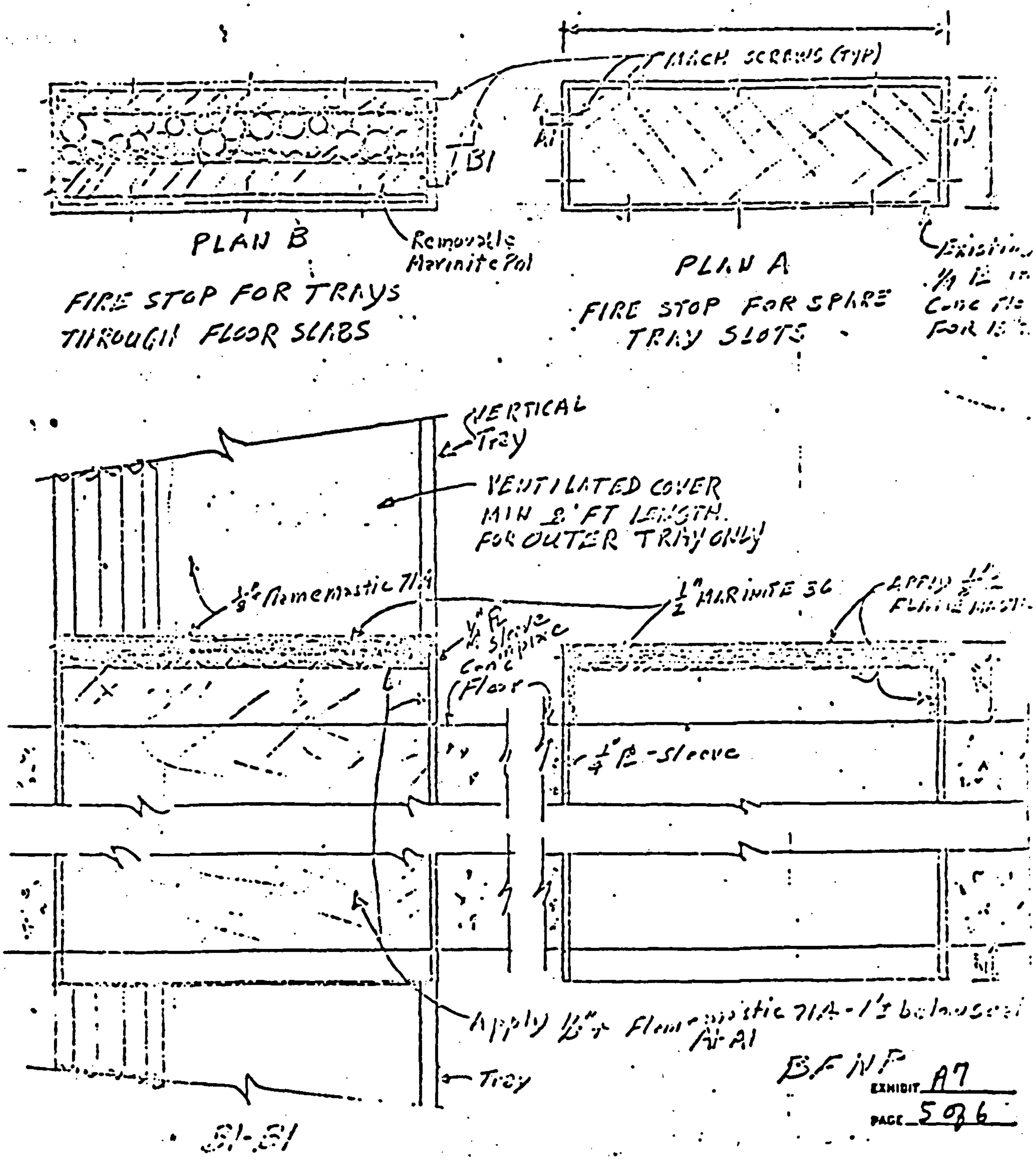

BFNP EXHIBIT A7
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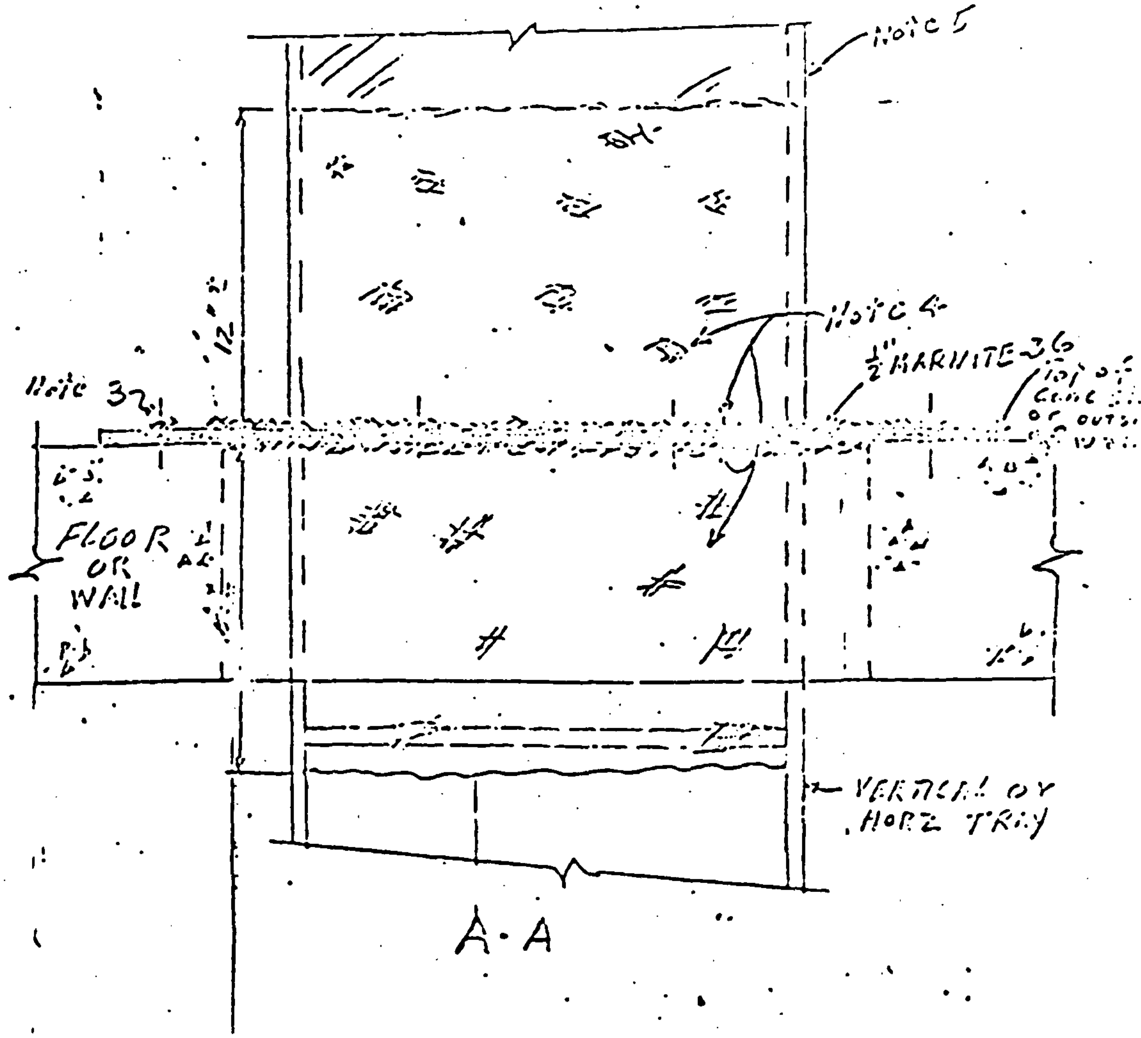

FIRE STOP FOR CABLE TRAYS THROUGH
FLOOR OR WALL OPENINGS

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BFNP

3/20/75 SHIFT ENGINEERS LOG

2230 Const. elect foreman Kimbrough over crew sealing conduits and cable trays U-1 Rx Bldg. reported that a small fire occurred U-1 Rx. Bldg. NW D 565 elev. When test flame ignited some of sealer being used on a cable tray. The fire was extinguished with dry chem Unit R1-14 and no damage was done to cables or other equip in tray (ATESI)

McCrary

EXHIBIT A8

LOSS REPORT

(REVISED)

TENNESSEE VALLEY AUTHORITY

"BROWNS FERRY NUCLEAR PLANT"

ATHENS, ALABAMA

Prepared For

NUCLEAR REGULATORY COMMISSION

INSPECTION & ENFORCEMENT DIVISION

By

FACTORY MUTUAL ENGINEERING ASSOCIATION

NORWOOD, MASS.

On

MARCH 31, 1975

SERIAL NO. 51-315-11

H L Bryant

Approved by H. L. Bryant
V.P. & Director of Inspections

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INDEX

1 Introduction

On March 22, 1975 at approximately 12:30 PM a fire was initiated in a wall opening between the "Spreader Room" in the Control Building and the Reactor Building by construction personnel testing for air leaks at a point where cable trays penetrate a wall opening. Materials initially involved were polyurethane foams used as sealants around the cable trays. The ignition source was a lighted candle. Fire spread in grouped cables and grouped cable trays from the point of higher static air pressure in the "Spreader Room" to the point of lower static pressure in the Reactor Building.

No fire watchers nor ready means of extinguishment were available to construction personnel. The work permit system was oriented to job control and security rather than loss prevention.

No adequate previous investigation had been made of the hazards of materials used for sealants. Fire hazards of grouped electrical cables were not adequately understood.

Firefighting activities took place concurrently with a successful nuclear power reactor shutdown. The time from fire initiation to extinguishment was approximately 6 hours with only portable firefighting equipment being used in an atmosphere obscured by smoke and rendered unsafe by toxic and irritating fumes from burning cable insulation. Loss of breathing air supplies during the course of the fire resulted in a hiatus of approximately two hours during which no effective action was taken. Final extinguishment was accomplished by the use of portable hose from hose stations which were available but not initially used because of a management decision not to use water on a fire involving electrical wiring.

Primary damage was to grouped cables in cable trays, cable trays and conduit. Consequential damage occurred in adjoining areas due to release of corrosive products of combustion from chlorine bearing plastic insulating cable coverings.

II Scope of the Investigation

This investigation and the report are limited to:

1. Firefighting and protection techniques, equipment and materials.
2. Recognition of limitations and response of emergency support groups.
3. Adequacy of training of site personnel in responding to emergencies.

III Firefighting and Protection Techniques, Equipment and Materials

Overall, the plant is protected with manual firefighting equipment consisting of hand extinguishers of the carbon dioxide and dry chemical types, wheeled carbon dioxide and dry chemical extinguishers and small hose connections equipped with nozzles chosen for the variety of hazards. There are no hose standpipes in the Control Building.

There are automatic deluge systems protecting main yard transformers, shunt reactors, hydrogen trailers, carpenter shop in the Service Building, compressed gas cylinder storage rooms and oil storage room. In the Turbine Building, the main turbine oil tanks, reactor feed pump turbine oil tanks, high pressure coolant injection pump turbine oil tanks, turbine head ends and hydrogen oil seal units are protected with fixed systems. Deluge valves not standard to industrial fire protection practice, are utilized and involve systems of three-way solenoid valves, 2-port plug valves and piston operated valves. Since these were not involved in the fire, no investigation was made. Deluge systems inside are arranged so that two of three heat actuating device signals must be transmitted before the deluge valve opens and fire pumps start.

Water supply for the combination fire and mill-use system is derived from raw water service pumps at the Tennessee River water intake backed by three electrically driven vertical automatic starting, 2500 gpm at 300 ft. head fire pumps using strained raw water as a suction supply. Electrical supply to the fire pumps are redundant through individual circuits connected to the unit station service, the common station service and to the diesel emergency generators. Automatic starting for Pump No. 1 is interlocked with a signal from any deluge valve. Pump Nos. 2 and 3 are started on pressure drop to 120 psi with time delays of 15 and 30 seconds respectively. The fire pumps discharge through a single 14 in. main to yard loops. Underground mains are not yet complete but will consist of a double looped system of 10 in., 12 in., and 14 in. pipe with adequate hydrants. Three plant service water pumps rated at 375 gpm at 200 ft. head with two 10,000 gallon surge tanks located on the roof of the Reactor Building pressurize the deluge valve headers.

Service water pumps, surge tank valves and raw water pumps are automatically deactivated on actuation of the fire pumps.

There are push-button stations for remote manual fire pump starting at hose connections, siamese connections, deluge systems and the Unit No. 1 control room. Fire pump operation is monitored at the control room where pressure is also indicated.

There is a central fixed low pressure carbon dioxide system rated at 17 tons. It is located in the Diesel Generator Building. System CO_2 discharge is automatically actuated at the lubricating oil purification unit, paint shop and spray booth, in the four diesel generator rooms and the control board rooms. The automatic systems are actuated by interlocks with rate-of-rise detection devices. Actuation is annunciated at the Unit No. 1 control panel. System discharge is manually actuated in the "spreader rooms", the auxiliary instruments rooms and computer rooms. The manual systems are actuated locally after fire annunciation detected by means of fixed product of combustion detectors. All discharge systems are equipped with time delay relays for safe evacuation of personnel. Conformance with design criteria for carbon dioxide-air concentration was not investigated. All systems are provided with multiple shot capability. The manual systems must be manually reset before they can be actuated again.

IV Support Groups

Emergency support facilities have not yet been fully implemented at this plant which is still under construction. Plant safety responsibility has been assigned to a plant engineering aide in safety with back-up by maintenance personnel in craft areas. A central engineering headquarters staff group provides technical support in fire protection and performs periodic safety audits at all TVA Facilities. No such inspections had yet been made at the operating portions of this plant and their capabilities were not reviewed as part of this investigation.

Fire protective equipment is inspected on a regular basis with written records but with divided responsibility and no management overview. Monthly inspections of fire extinguishers and yard hydrants are made by the engineering aide in safety and by maintenance personnel. There are no inspections of underground main valves, deluge valves, small hose connections or general plant conditions. Extinguishers receive full annual maintenance.

Hydrants are flushed and maintained semi-annually with follow-up by the Maintenance Department.

Fire pumps are operated weekly with records of operating deficiencies and discharge pressure. A tri-annual performance test has been scheduled, but there is no program for flowing the individual yard loop systems.

Annual tests are conducted of interlocks and alarms of the carbon dioxide fixed systems.

Outdoor water spray systems are scheduled for flushing and trip testing on an annual basis by both manual and automatic actuation. Indoor deluge systems are scheduled for tests of the tripping devices and for flushing by bypassing the normal discharge outlet.

Impairments of fire protective equipment are under the direct control of the shift engineer in the Operating Department with follow-up by means of a "log-in" procedure. Impairments must be approved by the plant superintendent or his assistant. Long-term impairments in excess of 7 days must be reviewed by a senior operating committee. Important impairments are treated as plant emergencies subject to overtime work as necessary to restore protection. A work clearance procedure which must be signed by company personnel is used to follow the work to completion. There is no formal tag-out procedure nor written instructions regarding provision of emergency alternate protection or establishment of extra personnel precautions.

Changes and modifications in the operating area and major maintenance operations are controlled by a work permit system, copies of which are furnished to the shift engineer and security force. The permits are prepared by the group involved and outline the nature of the work, personnel involved, clearances needed, and in some cases precautions to be taken. Support groups including quality assurance, operating department and plant superintendent are required to sign the work permits and the Security Department provides necessary entry supervision. The forms are supervised by a work coordinator who follow-up progress in completion of all issued permits. There appears to be no supplemental lock-out system, welding permit system, hot work permit system or other required safety supervision of potentially hazardous operations. One work permit may cover more than one area and may include long time periods. Operating personnel may be aware of the presence of maintenance or construction personnel in the area but are not required to be notified of each day's work plan. A new permit or extension of the existing permit is required if work exceeds the estimated time.

There is a plant operating review committee composed of the plant superintendent and senior department heads. This is a senior staff advisory group. It also functions as an investigatory group reporting to a safety review board. It meets monthly with special meetings to investigate incidents involving safety questions or violations of technical specifications. Among its duties are the review of standard and emergency procedures, review of changes of technical specifications equipment or systems, review of abnormal occurrences, surveillance of plant activities to detect safety hazards, etc. The group makes no regularly scheduled review or audits of the plant from a loss prevention viewpoint, but may make specific visits as required. Major preplanning for emergencies is the responsibility of the plant superintendent with the advice and consultation of this group.

V Emergency Training of Site Personnel

There is an organized plant emergency organization made up wholly of Operating Department personnel. Individual duties of plant organization members are not defined other than to report to the scene of emergencies equipped with extinguishers and under the control of the shift engineer or assistant shift engineer in charge. Support assistance is provided by security forces, all supervisors and their assistants and in special cases, all foremen, dual rate foremen and job stewards. Crafts in maintenance or construction are not members of the plant emergency organization, but stand by for instructions from their foremen.

The shift engineer is designated as the "man-in-charge" with one of two assistant shift engineers as back-up in the event that the shift engineer's presence is needed in the control room. The unit operators who can be relieved of duties at the control room and all assistant unit operators are members of the organization.

The alarm is sounded locally and reported to the unit operator by special telephone code. The unit operator verifies and locks in the plant fire alarm announcing over the plant public address system the nature and location of the emergency. In the event of failure in the public address system, brigade members call a designated number to obtain information. Plant security forces man necessary entrance points and the security officer in charge reports to the scene of the emergency.

Operation of fire pumps and automatic protection systems is monitored from the control panel. There is no special assignment to check fire pump operation, but craft personnel are available on all shifts for assignment by the man-in-charge.

A call-out list of off-duty personnel is available to the shift engineer. The plant superintendent and departmental supervisors are called as necessary. The Athens Fire Department, part paid and part volunteer, is called if in the judgment of the shift engineer, assistance is needed. The nearest fire station is approximately 10 miles from the plant.

All shift engineers receive a special leadership course in fire control at the Tennessee State Fire School as soon as possible after being appointed to the position, with a refresher training course at five year intervals thereafter. Plant emergency organization members are trained at the Tennessee State Fire School and bi-annually thereafter receive refresher training at the plant site from members of the Security Department, the health physics technician and the radiological analyst. There is no established on-site training or drilling in emergency procedures. Weekly safety meetings are provided for craft people but operators (plant emergency organization members) do not attend.

VI <u>Conclusions and Recommendations</u>

The original plant design did not adequately evaluate the fire hazards of grouped electrical cables in trays, grouped cable trays and materials of construction (wall sealants) in accordance with recognized industrial "highly protected risk" criteria.

Preplanning for recognition and control of potentially hazardous operations and for subsequent control of plant emergencies were not fully established by operating management.

Although not specifically a part of this investigation, it is obvious that vital electrical circuitry controlling critical safe shutdown functions and control of more than one production unit were located in an area where normal and redundant controls were susceptible to a single localized accident.

1. Grouped cables in the "spreader rooms", in cable trays, between trays, or between trays and terminal points, and in other areas where electrical cables have combustible insulation and are grouped (i.e. more than one full tray 12 in. wide or wider) should be coated with a Factory Mutual approved cable protecting coating. In areas where there are other combustible materials, automatic sprinkler protection on a wet pipe or pre-action system using closed head sprinklers should be installed. In critical areas such as the "spreader rooms", if the cabling is not coated, the manual carbon dioxide room flooding system should be made automatic in operation with actuation by the products of combustion detectors and should be backed up with a pre-action or multi-cycle automatic sprinkler system.

Comment: The fire hazards and subsequent serious interruptions to production following fire in grouped combustible electrical wiring is well documented in industrial experience. Control of the hazard has been effectively accomplished by use of non-combustible coverings to minimize fuel available to support a spreading fire or by use of automatic extinguishing systems, primarily automatic sprinklers, to control fire.

There are available Factory Mutual approved water base coatings bearing the trade names of "Flamastic 71A" and "Vimasco Cable Coating No. 1" which achieve acceptable levels of fire spread while not requiring derating of the current carrying capacity of the wiring due to heat build-up. The treatment is relatively simple to apply and can be readily replaced when it is necessary to remove it for installation or removal of wiring.

2. A hazard analysis of materials to be used in unprotected and/or critical areas should be made by competent fire protection personnel with hazards tests as needed by approved laboratories before such materials are introduced.

Comment: The unauthorized use of a highly combustible flexible polyurethane foam and the planned use of an allegedly "self extinguishing" material (rigid urethane foam) indicates a serious deficiency in hazard evaluation. The use of such materials in areas of high value, critical importance or where there are no other combustible materials introduces an unnecessary exposure.

3. Where cables pass through wall openings or through floors, the openings should be packed with noncombustible materials such as mineral or unimpregnated glass wool and the cables should be coated for flame retardancy.

Comment: See Comment Nos. 1 and 2 above.

4. Before work permits are issued for areas of high value or having critical or combustible occupancies, a hazard evaluation should be made and necessary precautions such as fire watchers with suitable extinguishment equipment, use of lock-out tags, welding permits, etc., arranged for. Operating supervision should be kept aware of areas in which craft personnel are working. Permits should cover each job and should be renewable on a daily basis with re-evaluation as needed.

Comment: A permit system which does not incorporate a hazard evaluation by operating, construction and safety or fire protection personnel and which does not require regular renewal and reinspection has little value in controlling introduction of sources of ignition. Occupancies may change over a period of time and require different precautionary measures to be taken. Permits covering more than one area complicate the establishment of adequate control procedures.

5. a. Plant emergency organization training should be conducted on a regular basis including classroom lectures and actual field drill in various hazard areas and under a variety of conditions including problems relating to the use of alternate means for extinguishment.

b. Personnel should be assigned to specific duties based on the hazards of individual areas and should include personnel assigned to check fire protective equipment including fire pump operation, sprinkler operation, sprinkler and other water control valves throughout the emergency.

c. Maintenance personnel including pipe fitters and electricians should be included in the plant emergency organization with specific duties assigned.

Comment: While plant personnel functioned well as individuals under difficult conditions, the failure to use available water hose for extinguishment and the exhaustion of the supply of breathing air indicates a lack of adequate training of the emergency organization and a lack of adequate preplanning for emergencies. A well trained organization with understanding of safe methods of fire control would result in prompt control and extinguishment of fire.

6. Management should initiate a Property Conservation Program. A management team should be established to this end. The duties of this group should include evaluation of potential disaster conditions, methods of coping with them, and plans for recovery following a mishap.

Comment: A properly organized program of preplanning for emergencies will not only uncover hazard areas where improvements can be made but will uncover the "incredible" situation in which major incidents can be evaluated and corrective measures planned.

7. The following improvements should be made in the plant self-inspection program.

   a. A single person should be charged with the responsibility of supervising all self-inspection procedures and the forms submitted to management for review and initialing prior to being filed.

   b. Weekly recorded inspections should be made of all valves controlling water supplies to sprinklers in deluge systems and in the yard loop and throughout the entire plant to uncover deficiencies in housekeeping, electrical equipment, arrangement of occupancy and other matters relating to loss prevention.

   c. The existing monthly inspection should be enlarged to include all fire hose racks.

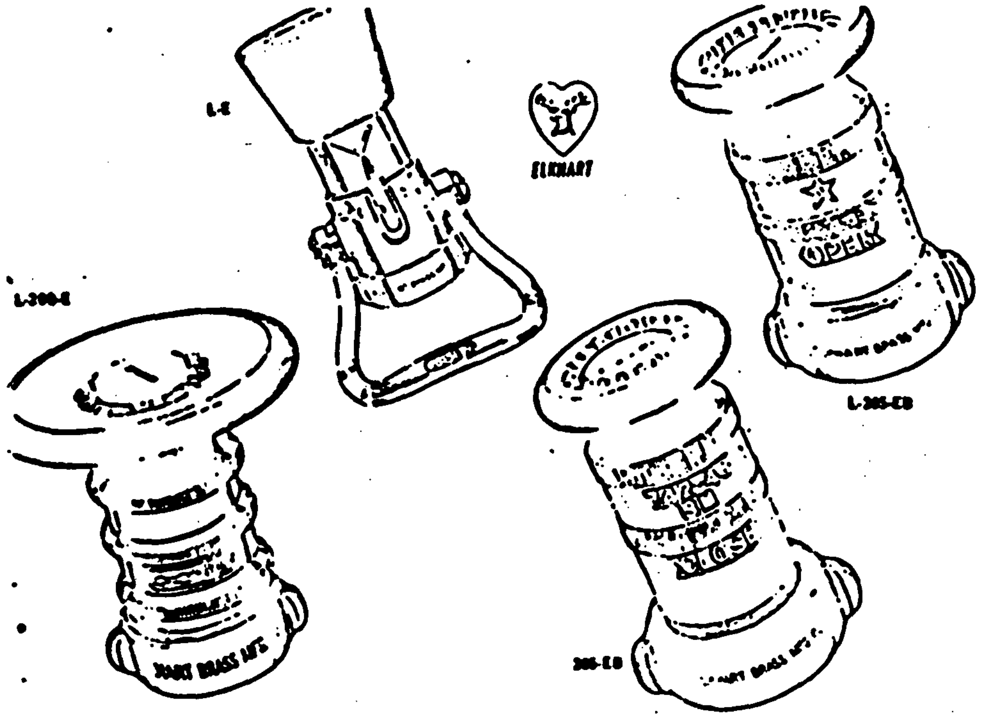
Comment: Although some inspection programs were established and others were planned, responsibility is not well defined and there is no direct management overview and supervision. The inspection procedures do not include some vital factors relating to loss prevention. An adequate self-inspection program will detect and provide information to correct plant deficiencies as well as to establish that all fire protective equipment is in an operating and ready condition.

8. A tag-out system for control and prompt restoration of in-plant protection should be initiated. This should include those factors now in effect as well as notification to plant personnel of impaired protection, provision of temporary alternate protection including fire watchers and extra guard assignments and the possible cessation of operations of a hazardous nature in the impaired area.

Comment: Control of impairments of major fire protective equipment is essential to continued safe plant operation. The existing program does not adequately provide all plant personnel with the knowledge of areas where protection is impaired nor does it include established procedures for providing temporary protection.

9. A re-evaluation should be made of the arrangement of important electrical circuitry and control systems, to establish that safe shutdown controls in the normal and redundant systems are routed in separated and adequately protected areas.

Comment: It is seldom possible to toally eliminate all "bottlenecks" in a production scheme. It is important, however, that such situations be recognized. The scope of the recommended re-evaluation is intended to apply only to those critical functions which could affect maximum shutdown safety.



Elkhart Electrical Fog Nozzles

FOR CLASS "C" ELECTRICAL FIRES

All of the above nozzles are designed so that they are safe for use on fires in electrical equipment up to 250,000 volts at a minimum 10' distance. It is impossible to obtain a straight stream. These nozzles are widely used by power companies and industrial plants which have Class "C" hazards.

EXCLUSIVE STAINLESS STEEL STEMS

To combat today's corrosion and erosion problems, all Elkhart electrical fog nozzles are furnished with stainless steel stems. These nozzles are especially suitable for use in salt water environments.

IMPROVED DESIGN

These nozzles now have a fully machined waterway which reduces friction loss and improves the flow. They now qualify as constant flow nozzles.

LE 1½" Select-O-Stream No-Shock Nozzle

Operates the same as the No. L nozzle except closes and opens in a 30° fog pattern. Standard discharge 95 g.p.m. Optional 60 or 125 g.p.m.

L-205-EB 1½" Adjustable No-Shock Nozzle

The first adjustable fog nozzle listed by Underwriters' Laboratories, Inc., for use on electrical Class "C" fires for voltages up to 250,000 at distances of 10' or more—Standard model opens at a 30° fog—If desired, can be made to open at a 15° fog—For use on 1½" linen or rubber lined hose. Standard discharge 95 g.p.m. Optional 60 or 125 g.p.m.

L-200-E 1½" Mystery No-Shock Nozzle

Operates the same as No. L-200 Mystery Nozzle except closes and opens in a 30° fog pattern. Standard discharge 95 g.p.m. Optional 60 or 125 g.p.m.

205-EB Adjustable No-Shock Nozzle for Playpipe

Same as 1½" No. L-205-EB except 2½" size—Has a 1½" base thread.

Catalog No.	G P M	Size	Length	Wt. lbs.	Threads	Finish
L-205-EB	60, 95, 125	1½"	8"	[illegible]	1½"	Satin Brass or Chrome plated
205-EB	125	2½"	8"	[illegible]	[illegible]	Satin Brass or Chrome plated
L-200-E	60, 95, 125	1½"	6½"	3½	1½"	Satin Brass or Chrome plated
LE	60, 95, 125	1½"	6½"	[illegible]	1½"	Satin Brass or Chrome plated

See Engineering Data beginning page 90.

Exhibit A10

Reactor water isotopic 3/25/75

Millipore filter paper

Isotopes	Unit #1 Act. line (uci/ml)	Unit #2 Act. line (uci/ml)
Cr-51	8.69 E-04	6.48 E-04
Zn-69m	<2.37 E-05	<1.15 E-05
La-140	<5.46 E-05	<2.75 E-05
Ba-140	2.16 E-04	<5.69 E-05
Sb-122	1.83 E-04	5.03 E-05
Nb-97	2.03 E-04	1.51 E-05
W-187	1.02 E-04	6.36 E-05
Zr-95	3.86 E-03	9.90 E-04
Zr-97	<2.01 E-05	<9.33 E-06
Nb-95	6.37 E-04	6.90 E-04
Co-58	2.45 E-04	2.05 E-04
Mn-54	1.00 E-04	2.70 E-05
Mn-56	<2.93 E-05	<9.87 E-06
Zn-65	<1.83 E-05	<1.33 E-05
Co-60	<8.74 E-06	<4.80 E-06
Fe-59	<1.31 E-05	<6.61 E-06
	<4.43 E-03	7.99 E-04

EXHIBIT A 11
PAGE 1 of 3

Reactor Water isotopic 3/25/75

Cation filter paper

Isotopes	Unit #1 Act. line (uci/ml)	Unit #2 Act. line (uci/ml)
Co-57	<1.04 E-03	<1.12 E-05
Cr-51	<2.03 E-02	7.04 E-04
Zn-69m	<2.70 E-03	<3.19 E-05
Ba-140	<1.18 E-02	<1.17 E-04
Cs-134	<3.37 E-03	2.16 E-03
Cs-137	<2.78 E-03	4.31 E-03
Co-58	<2.22 E-03	5.43 E-04
Mn-54	<1.77 E-03	1.96 E-04
Mn-56	<3.79 E-03	<4.56 E-05
Sr-91	<7.62 E-03	<6.53 E-05
Zn-65	2.06 E-02	<2.70 E-05
Co-60	3.91 E-02	<1.23 E-05
Cu-64	1.93 E-01	<1.29 E-03
Na-24	2.42 E-03	<7.17 E-06
Sr-92	<1.36 E-03	3.05 E-05

EXHIBIT A 11
PAGE 2 of 3

Reactor water isotopic 3/25/75

Anion filter paper

Isotopes	Unit #1 Act. Conc. (uci/ml)	Unit #2 Act. Conc. (uci/ml)
Mo/Te-99	3.77 E-02	5.63 E-03
Cr-51	5.41 E-03	8.30 E-04
I-131	1.07 E-01	1.14 E-02
F-18	6.05 E-04	1.31 E-04
I-133	1.62 E-02	1.35 E-03
I-132	<1.70 E-04	4.04 E-04
I-134	2.86 E-03	4.89 E-04
Br-84	<2.91 E-04	<9.15 E-05
I-135	<7.54 E-05	<2.33 E-05
Cl-38	<1.95 E-03	<1.10 E-03

EXHIBIT A11
PAGE 3 of 3.

BROWNS FERRY UNIT 1

SEQUENCE OF
SIGNIFICANT OPERATIONAL EVENTS
AT TIME OF FIRE

Time	Event	Action	Response or Nonresponse
3/22/75 Prior to 1235	Initial Condition	Routine Operation	Unit Load 1,100 MWe
1235	Report of fire received by assistant shift engineer from public safety officer.	Assistant shift engineer set off fire alarm and proceeded to fire. Fire alarm sealed in by unit operator who then used paging system to inform plant personnel of fire location.	Operating personnel fire brigade reported to fire and began fire-fighting activities (describe. elsewhere in investigation report).
~ 1240	Received the following alarms in unit 1 control room: 1. RHR or core spray pumps running/ auto blowdown permissive 2. Reactor level low/auto blowdown permissive 3. Core cooling system/diesel initiate	Unit operator observed control board and determined normal reactor water level and steam pressure, drywell pressure normal at 0.45 psig, and emergency core cooling system (ECCS) equipment aligned in normal standby status. (Reactor water level instrumentation activates the emergency core cooling systems, this being normal, indicated a lack of need for these systems.) (Normal drywell pressure indicated that piping was intact inside the primary containment.)	All diesel generators (D/G's) started from ECCS logic signal which started the core spray pumps.
~ 1242	Residual heat removal (RHR) and core spray (CS) pump running alarm received. High-pressure coolant injection pump (HPCI), reactor core isolation coolant pump (RCIC) started.	Unit operator observed pumps running and RHR aligned to reactor in low-pressure coolant injection (LPCI) mode. Verified reactor water level normal and stopped pumps. Operator attempted to reset alarm. (All four of these systems are ECCS and with normal level were not required.)	Pumps stopped. Alarm would not reset with reactor pressure and level normal.
~ 1244	RHR and core spray pumps restarted with no apparent reason.	Operator observed reactor level normal and attempted to stop RHR and core spray pumps. Pumps could not be stopped from benchboard.	Operator did stop pumps at ~ 1243 from benchboard.
~ 1248	Reactor recirculation pumps ran back for no apparent reason. Began losing electrical boards. Indicating lights over valve and pump control switches on panel 9-3 were glowing brightly, dimming, and going out. (Panel 9-3 is the control board location for all ECCS equipment.) The lights being lost on control circuits for ECCS pumps and valves precluded reliable operation from that control board.	Operator observed reactor power decreasing and average power range monitors (APRM) responding. Also noted reactor level 2 to 3 inches high. Operators observed smoke from control wiring under panel 9-3.	Unit power decreased from 1,100 MWe to 700 MWe.

Exhibit B1

Time	Event	Action	Response or Nonresponse
1248 (Contd.)	Lost 1/2 of reactor protection system (RPS). Lost remote manual control of a number of relief valves. Numerous alarms occurred on all control panels and unit in unstable swing.		
1251	Shift engineer instructed operator to zero recirculating pump loading and scram the reactor.	Operator reduced loading signal to recirculating pumps and manually scrammed the reactor and placed reactor mode switch in the shutdown position. Shift engineer reported plant conditions to supervisors by phone.	Recirculating pumps tripped by unknown cause at approximately 20 percent loading. Reactor scrammed and all rods inserted.
1253	Confirmed that all control rods were fully inserted.	Operator tripped B and C reactor feed pumps (RFP), B and C condensate booster pumps, and C condensate pump. Reduced loading on reactor feed pump subpanel to prevent "over shoot" on reactor level return. RCIC started manually as backup.	Pumps responded to trip signal and reactor level was maintained by reactor feed pump A and RCIC.
1254	Unit conditions indicated need for tripping turbogenerator.	Assistant shift engineer (ASE) initiated turbine trip upon observing generator load at 100 MW. Also opened generator field breaker and motor-operated disconnects (MOD's).	Turbine bypass valves opened to compensate for turbine valve closure and maintain pressure normal. The main-steam isolation valves (MSIV') remain open allowing reactor pressure control through the turbine bypass valves to main condenser heat sink.
		Unit operator inserted source range and intermediate range neutron monitors and observed reactor power decrease.	Neutron monitoring responded normally.
	HPCI started.		HPCI automatically aligned in normal injection mode to reactor vessel.
	Reactor water level restored to approximately normal range.	Operators shut down HPCI and RCIC.	HPCI and RCIC shutdown. Problems incurred upo shutdown with valve operation associated with systems.
1255	Lost 120-V unit preferred power. One of the feeds from this source is the unit control rod position indication on panel 9-5 (reactor control panel).	Operator placed reactor mode switch in "Refuel" mode to verify one rod withdraw permit. (All rods must be fully inserted or the indicating light for one rod withdraw in refuel mode will not illuminate.)	Received white permit light.
	Lost all neutron monitoring.	Operator observed no indication on average power range, intermediate range, or source range monitor.	Capability to monitor core was lost.

TABLE 6

Exhibit B1

Time	Event	Action	Response or Nonresponse
~ 1256	By this time the following electrical boards were lost: 1A 250-V D.C. Reactor MOV board* 1B 250-V D.C. Reactor MOV board 1A 480-V A.C. Reactor MOV board 1B 480-V A.C. Reactor MOV board 1C 480-V A.C. Reactor MOV board 1A 480-V A.C. Shutdown (SD) board 1B 480-V A.C. Shutdown board 120-V A.C. unit preferred	Indication from the unit control room as to electrical sources feeding the various equipment and as verified by ASE as he checked the individual boards.	This caused the loss of vital equipment being fed from these electrical boards. Loss of power to MSIV's caused them to go closed (all 4 outboard valves), placing the unit in isolation from the main condenser heat sink and cutting off the steam supply to the reactor feed pump turbines. All emergency core cooling systems were lost with the exception of 4 relief valves which could be operated from the unit control board.
~ 1258	Reactor pressure rapidly increased to 1,100 psig.	ASE was unsuccessful in opening MSIV's from backup control center.	Relief valves opening and closing to maintain pressure between 1,000 and 1,100 psig. Relieving to the suppression pool (torus).
		Operator manually opened main-steam relief valves; then closed as pressure came back to desired range.	Pressure decreased to 850 psig; then rapidly increased to 1,000 psig.
		Attempts to place RCIC in service were unsuccessful from control room or backup control panel.	Valve 71-2 (steam supply to turbine) was apparently the only valve loss on RCIC but rendered it inoperable...This valve was later opened by use of temporary power. The HPCI was previously rendered inoperable by loss of valve controls.
~ 1259	Reactor water level decreasing due to almost constant blowing down to the torus.	The only water input left with the capability to overcome a pressure above ~ 350 psig was the control rod drive pump; it was increased to the maximum.	
	Torus cooling became essential.	ASE was unsuccessful in placing emergency power on RHR valves at local MOV board. (Those valves required for torus cooling.)	RHR system was unavailable for torus cooling as a result of electrical board losses.
		Shift engineer and two electricians making attempts to restore 480-V 1A and 1B reactor MOV boards and 250-V D.C. boards.	
1300	4-kV SD board C undervoltage shutdown bus 2 undervoltage. (As noted on electrical printer.)		4-kV voltage continued to be supplied to SD boards A and B by shutdown bus 1. Shutdown boards C and D transferred to D/G's C and D.

*MOV - Motor operated valve

TABLE 6
SHEET 3 OF

Exhibit B1

Time	Event	Action	Response or Nonresponse
1320	SD bus 1 undervoltage.		A and B D/G's tied onto their respective board The A and B diesel generators were running and up to speed prior to this but had not received an undervoltage signal to close them onto thei boards until this event. This placed all the 4-kV shutdown board equipment upon the diesel generators.
1321	Process computer lost.		No further printout until ~ 1600 hours.
~ 1330	Decision made to depressurize reactor by blowing down to torus.	Manually opened 4 main-steam line relief valves. Checked condensate booster pumps running. Reactor feed pump bypass valve 3-53 checked opened ready to admit water from condensate system.	Reactor pressure decreased; water level decrea Reactor water level dropped from normal 201 inches above top of active fuel to 43 inches a top of active fuel.
~ 1334	Shutdown bus 2 transferred. (In maintaining a normal configuration on the plant electrical system, the 4-kV shutdown boards will be lined up to feed from the unit or offsite source feed. This allows the diesel generators (D/G's) to be a highly reliable backup, giving two sources of voltage should the need arise.)	Manually initiated by ASE by normal procedure of synchronizing the D/G's with the SD bus; then dropping off D/G feed to the SD board.	Bus 2 energized from unit 2. A,B, and D SD boards transferred to SD bus 2. D/G's remained on running standby.
	Shutdown bus 1 continued to be deenergized.	ASE was unsuccessful in an attempt to manually energize.	
	Shutdown board C remains energized from C D/G.	ASE was unsuccessful in an attempt to manually change C SD board feed from D/G to shutdown bus 2.	Breaker stayed closed for 5 to 10 seconds; then opened. Feed transferred back to D/G.
	D shutdown board deenergized for ~ 5 minutes.	ASE reenergized D board.	D board remained feeding from shutdown bus 2.
~ 1345	Restored unit preferred from unit 2.	ASE manually transferred.	Unit preferred back on both units.
	Reactor steam pressure decreased to 350 psig.	From continued manual operating of relief valves.	Reactor water level increasing as a result of condensate booster pump input.
~ 1355	Water level approaching normal.	Attempted to throttle the feedwater bypass valve 3-53.	No response on feedwater bypass valve 3-53.

Time	Event	Action	Response or Nonresponse
~ 1357	Water level going high.	AUO dispatched to bypass valve. Manually closed down on valve.	Restored level to approximately normal.
	Outgoing (PAX) phones and page inoperable for some time.	All operations requiring control room monitoring set up on a periodic in-call basis.	
~ 1400	Reactor steam pressure at 200 psig.	Operator maintaining reactor steam pressu[illegible] through four relief valves and level control through RFP bypass and CRD pumps. This had to be controlled via phone communication since the paging system was inoperative.	
	480-V SD boards A and B restored.	ASE manually initiated.	Boards appeared heavily loaded as indicated by loud "humming." Boards remained in servic[illegible]
		ASE tried to restore reactor 480 MOV boards A and B and reactor 250 MOV boards A and B.	Initially unsuccessful.
	Off-duty maintenance personnel began reporting.	Electricians and operators working to restore these electrical boards by isolating faulted circuits.	Restored approximately two hours later.
~ 1448	Voltage lost to 4-kV shutdown board C.		There was no control room indication of this condition.
	B D/G found tripped with field breaker open.	ASE closed field breaker on B D/G and brought back to running standby.	
~ 1500	Attempt made to align one RHR system up for torus cooling and the other for SD cooling.	Four AUO's working in pairs using breathing air packs. Made two entries, but insufficient air supply aborted attempts.	
~ 1527	Voltage restored to C shutdown board.	ASE found C D/G running at approximately 1/2 speed. Brought D/G to synchronous speed and closed breaker to board.	C shutdown board was deenergized. C shutdow[illegible] board was also lost from 1545 to 1557. Sever[illegible] at ~ 1630 C D/G was tied onto the board, it[illegible] breaker tripped, and prevented C D/G from being used.
~ 1600	RHR system 1 aligned for torus cooling.	Decision made not to start in this condition since it could not be established that system was charged with water.	This system was subsequently checked for pro[illegible] alignment and charge and placed in service later.

TABLE 6
SHEET 5 OF

Exhibit B1
Page 5 of 9

Time	Event	Action	Response or Nonresponse
1630	480-V reactor MOV board 1A reenergized.	Result of electrician and operator action.	Energized electrical board allowed main turbine to be placed on turning gear and reactor protection system MG set A to be placed in service which energized trip channel A. Restored power to 1/2 of the process monitoring. Restored power to those ECCS valves feeding from that board, etc.
	Valve restoration to ECCS equipment.	By operator interview, it has been established that the following procedure was used in valve operation where valves were not operable from the control room. Placed select switch in emergency position at the electrical board, ran valve to desired position, reopened breaker, immediately returned to control room and established fact that level was not affected by possible electrical fault misalignment. Tagged valve control switch on unit control board showing valve position. This was a safeguard against draining the vessel down.	Level remained normal.
		ABE observation of conditions as appeared on panel 9-3 in control room.	Core spray loop I A and C pumps appear operable from unit control board. All valves and both pumps had indicating lights. Core spray loop II had a few valves that were inoperable. RHR loop II had a few valves available. RHR loop I--same.
1640	Request to start reactor building exhaust fan to remove smoke and fumes.	Started locally from 480-V reactor building vent board.	Fan responded normally. Dampers controlled manually at the damper.
1700	Request to stop reactor building exhaust fan as airflow appeared to aide fire.	Stopped locally by operator.	Fan stopped.
1800	Relief valves inoperable by remote manual control from benchboard due to loss of instrument and control (I&C) voltage to solenoid in air supply to diaphragm valve in air header to primary containment.	Operator observed lights indicated relief valves open. Other indications suggested that valves were closed. Restarted drywell air compressor. Craftsmen bypassed solenoid valve to provide control air supply to primary containment equipment.	Reactor pressure increasing [illegible] 200 psi. The compressor started but discharge isolation prevented airflow to primary containment and relief valve control. Allow relief valve remote manual operation at 2150 hours.

556

TABLE 6
SHEET 6 o

Time	Event	Action	Response or Nonresponse
1900	Decision made to restart the reactor building exhaust fan.	Manually initiated.	Remained in service.
	PAX telephones restored to unit control room.	ASE cleared problem on phones.	Gave control room the capability to call out.
2005	High torus level from earlier blowdown.	Manually aligned and started RHR drain pump to main condenser hotwell.	Torus level decreased. Hotwell level increased.
20-5	Venting drywell via standby gas treatment systems to plant stack. Pressure was about 2.5 psig.	Steamfitters manually opened 2-inch vent to standby gas treatment system.	Drywell pressure decreasing.
2130	Relief valves operable by remote manual control.	Switch left in open position anticipating voltage return. Manually operated relief valves to reduce reactor pressure.	Reactor pressure decreasing from 500 psig maximum.
2200	Secondary containment reestablished.	Operator stationed at reactor building entrance.	Shift engineer approval before entering required. Breathing apparatus required.
2230	ASE made attempt to prove D D/G operable from electrical control board in the control room.		Could not be operated from control room.
		ASE synchronized to D 4-kV shutdown board, locally, picked up load, placed on standby.	Could be used if needed by operating from D shutdown board.
3/23/75 0000	Need for flushing RHR system II prior to placing in shutdown cooling.	Existing procedures could not be used under present circumstance. Two senior reactor operators approved temporary flushing procedure.	System flushed and placed in service at 0415.
0100	Two source range monitors placed in temporary service located on the reactor side of the fire.	Licensed reactor operator stationed at these monitors in the area of unit 1 drywell continuous air monitor unit.	Established capability to monitor core. 10 counts per second reading on monitors.
0130	Torus cooling continues to be a necessity as blowdown continues.	Valves aligned manually by operators and system placed in service.	Decreasing torus temperature.
0212	Torus level instrumentation in service.		Level indicated +1". (Normal level is indicated as 0 with a deviation of + or - 5".)
0245	Restoration of equipment had progressed to the point that A and C core spray pumps could be tested from panel 9-3 in unit control room.	Operator action from unit control room.	Pumps and injection valves operable, thus giving part of the ECCS equipment available if needed.
0-15	Shutdown cooling achieved by normal flow path.	Manually aligned system.	Allowed operator control of vessel temperature.

Exhibit B1
Page 7 of 9

BROWNS FERRY UNIT 2

SEQUENCE OF
SIGNIFICANT OPERATIONAL EVENTS
AT TIME OF FIRE

Time	Event	Action	Response or Nonresponse
3/22/75 Prior to 1300	Initial Condition	Routine Operation	Unit Load 1,100 MWe
~ 1300	4-kV Shutdown bus 2 deenergized (relay action).		Lost reactor protection system (RPS) generator (MG) set 2B; 1/2 scram on [illegible] giving red lights on panel 9-5; react[illegible] recirculation pump automatically decr[illegible] reactor power. Lost voltage to instrument and contr[illegible] Lost indicating lights on system I re[illegible] heat removal (RHR) and system I core [illegible] alarms on RHR and core spray "start," "overcurrent," "pump trip."
	Operator observed decreasing reactor power indication and many scram alarms on control panel.	Operator placed reactor mode switch in "shutdown" and inserted nuclear instrumentation (source and intermediate range).	Reactor scrammed inserting all contr[illegible]
~ 1301	Reactor water level dropped and returned to normal (normal reaction from trip).	Tripped reactor feed pumps A, B, and C - Tripped turbine. Tripped exciter field breaker and opened generator motor-operated disconnects (MOD's).	Equipment response normal.
1308	Main-steam isolation valves (MSIV) closed.	Operator initiated reactor core isolation cooling (RCIC) for level control; initiated high pressure cooling injection (HPCI) for heat sink. Manually initiated relief valves for pressure control.	Equipment response normal. After this start and before ~ 1415 [illegible] and HPCI tripped several times from [illegible] reactor water level. Neither of the[illegible] could be restarted with the controll[illegible] "manual." Operator was unable to ge[illegible] signal from the subpanel control in [illegible] Pumps would start with controller in "automatic." At ~ 1345 HPCI was r[illegible] and brought to ~ 3/4 speed. It he[illegible] about 1 minute. The speed then drop[illegible] with no further response from HPCI; after it was unavailable.

TABLE 7
SHEET 1 OF

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Time	Event	Action	Response or Nonresponse
~ 1320	Lost remote manual operability of relief valves depriving operator of ability to reduce reactor pressure below set point.	Unit 2 assistant shift engineer made an attempt to operate relief valves from backup control panel but was unsuccessful.	Relief valves continued to lift on pressure. Maintaining reactor pressure at 1050 psig and below.
	Torus temperature increasing due to relief valve discharge into torus.	Placed the following pumps in service to establish torus cooling: D RHR pump, D2 RHR service water pump, and D1 emergency equipment cooling water pump.	Torus cooling established at ~ 1320. Torus temperature did not exceed 135° F.
~ 1400	Reactor depressurizing apparently from a relief valve that had lifted on pressure and stuck open.	No indication of coolant leak, and pressure decreasing at desired rate.	Reactor pressure decreasing at desired rate: 150 psig at 1500; 65 psig at 1900; 10 psig at 2010
~ 1415	Pressure starting to decrease.	Placing the condensate system in service.	Anticipating the pressure level that reactor vessel could be supplied from that source.
	Remote manual operability of relief valve restored.	Maintenance and operations personnel working during this period of non-operation checking the instrument and control (I&C) voltage to solenoids, drywell air compressor for proper operation and cutting in the backup control air supply. It is uncertain which of these operations reestablished remote operability of relief valves.	Gave operator discretion on relief valve operation.
~ 1430	Loss of some reactor water level instrumentation.	Determined that level indicator 362 appeared to be reliable and that 2 Yarway level indicators in backup control center corresponded with this indication.	Reactor water level never decreased below 160" above the top of the active fuel. Other level indication began to respond at ~ 1450.
~ 1450	Torus level increasing due to relief valve discharge.	Manually aligned RHR drain pump to transfer torus water to condenser hotwell.	Torus level never increased above + 5".
~ 1557	Voltage restored to C shutdown (SD) board.	Restored power to 480-V SD board 2A by manual operator action. Started D2 RHR service water pump on. Started B RHR pump on. Placed turbine on turning gear (T.C.)	These power supplies allowed turbine to be placed on T.C. and B RHR pump to be tested.
~ 2010	Condenser heat sink available. Allowed use of turbine bypass valves to reduce pressure.	Cleared up electrical trouble with mechanical vacuum pumps and established vacuum in main condenser.	Vacuum above 7" Hg allowing opening of turbine bypass valves for steam admission.
~ 2020	Torus temperature within limits; shut down torus cooling to allow flushing of lines for reactor shutdown cooling.	Aligned valves and flushed system II prior to placing system II in reactor shutdown cooling mode.	
~ 2045	Reactor pressure at 10 psig.	Opened reactor headvents.	
~ 2240	Reactor in shutdown cooling using RHR system II.	Manually aligned system.	Shutdown cooling achieved by normal flow path.

Exhibit B1

GENERAL ELECTRIC PERSONNEL COMMENTS

A. INTRODUCTION

A fire broke out at a cable tray penetration between the Unit I cable spreading room and Unit I reactor building at 1235 hours on March 22, 1975. ... lant personnel and the Athens Fire Department were used to fight the fire. The fire was completely extinguished by 1830 the same day. There were no injuries to personnel and no radiation releases from the plant site boundary.

The majority of the Unit I reactor systems were inoperable during the fire: Unit II was uneffected with a few minor exceptions. Most of the equipment outages were caused by indirect effects of the fire e.g., loss of electrical buses due to overloads caused by electrical shorts and load shedding. Both units were taken to cold shutdown and were in shutdown cooling by 0400 hours on March 23. The lowest reported reactor water level was -120" on Unit I, which is approximately 48" above the active fuel. Reactor pressure never exceeded relief valve level i.e., about 1100 psig. Primary containment was not breached.

At this time, the extent of the equipment damage and the political ramifications are unknown. TVA has assemblied a high level task force at the site on March 24, 1975. NRC held a meeting at their headquarters on March 23 and has a three man task force at the site on March 24, 1975. See attachment IV

B. DISCUSSION

1 Cause

An electrician was sealing a cable tray penetration between the Unit I cable spreading room and the Unit I reactor building. A foam type poly-ureathane is used for the original sealant; when cabl s are added, a hole is drilled in the poly-ureathane, the cable is pulled and sealed with RTV. A common practice has been to use a candle for leaks, i. , to look for drafts. While checking for leaks, the sealant started b..rning and spread out of control. A detailed list of sealant materials are given in attachment III. Both RTV and the poly-ureathane will sustain an open flame when ignited; this was confirmed in a test at the site, after the fire.

2. Extent

The fire burned in the following cable trays in the penetration ·ading from the cable spreading room and i. the Unit I reactor building: FM, MXES2, MX, MD, VE, TE, FL, QQES11, MW. LY, VK, TK, KT, and KS. The length of burned cable ranged up to about forty feet and was confined to the area adjacent to the penetration. Ten trays go through the wall at that location.

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Sequence of Events

Unit I: (Operators Log)

Hour	Event
1235	Fire in Unit I reactor building reported
1300	Manual reactor scram. Offgas filter pressure high and stack offgas dilution air flow low. No other system indications received (computer out of service) prior to scram. MSIV's closed, all rods in, RCIC and HPCI on manually.
1310	Using relief valves
1320	Unable to reset RPS - Loss of power - Loss relief valve ability to blow down reactor - opening on pressure only.
1320	Running HPCI CST to CST for pressure relief
1330	Turbine lift pumps on manually
1400	Relief valve hanging open momentarily
1415	Relief valve power back. Using RCIC for level control.
1430	Lost all reactor level indication except LI 3-62.
1440	Level indications back. Turbine on turning gear. No power on A & C core spray pumps or B2 RHR SW pump. Reactor 550 psig
1500	'A' condensate pump on. 'D' RHR pump in torus cooling. Reactor 250 psig
1600	Reactor level holding. HPCI and RCIC is off. 'C' condensate booster on. Reactor 125 psig
2230	Unit in shutdown cooling

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Unit I

Combination of Operators Log and input from General Electric personnel on site - J.D. Martin, J.H. Cox, and R.T. Thomas.

Hour	Event
1235	ASE received report of fire in Unit I reactor building and called in the fire alarm. Alarm was sealed in and announcing started over plant P.A.
1242	ASE went to reac or building via S.D. Bd. Rm. and met construction worker with CO_2 bottle. Small fire on 2nd floor, north side of reactor building 6 feet from 3rd floor ceiling. Effective visibility in area of fire was lost.
1245	In control room following alarms received on panel 9-3: 1. "Auto Blowdown Permissive" 2. "RHR/Core Spray Valve Opening Permissive" 3. "Standby Diesel Generator Initiate"
--	Verified all four diesels running at this time.
--	RHR System was aligned in LPCI mode with four pumps running. All four core spray also aligned and running. Core Spray and RHR pumps stopped. Normal reactor pressure and water level noticed at this time.
--	Noticed reactor power decreasing on panel 9-5 due to recirculation pump runback. APRM's went downscale as recirc flow decreased.
--	ASE called and requested assistance for Unit I fire. Unit I ASE and SE arrived in control room shortly thereafter.
--	Lights and indications on panel 9-3 glowing brightly or going out in random manner (RHR, CS., HPCI, RCIC).
--	Reactor water level slightly higher (2" - 3") than normal. Reactor pressure observed to be normal.
--	Raw Service Water, RHR and Core Spray Pumps starting and stopping intermittently. No injection valves opened. NOTE: HPCI and RCIC had automatically initiated earlier but not injected to the reactor vessel.
1253	The recirc flow was manually backed all the way down (Recirculation pumps tripped automatically) and the reactor manually scrammed.

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Event

-- ASE tripped turbine when load decreased.

-- Reactor water level decreased and caused HPCI and RCIC to automatically start. When level was restored shutdown HPCI. MSIV's did not isolate.

-- Two RFP's were tripped and reactor water level was controlled with remaining pump. Tripped RCIC. One bypass valve was kept 50% open to provide a heat sink.

-- Problems developed with RCIC steam supply valve 71-2 and other valves associated with this system.

-- Lost the following boards at this time:

1A 250V MOV Board
1B 250V MOV Board
1A & 1B 480V Reactor MOV Boards
1A & 1B 480V S.D. Boards
120V Unit Preferred
S.D. Bus # 1

-- ASE attempted to energize 1B reactor MOV board, but no power supplies available.

-- ASE called C.R. and requested CO_2 be initiated in spreader room. Accomplished by ASE who then went and energized 1A and 1B S.D. boards. Was unsuccessful in energizing 1B 480V reactor MOV board even after opening all outgoing breakers.

-- As conditions in the plant deteriorated, the outboard MSIV's closed and reactor pressure increased and relief valves started opening. Reactor pressure held at approximately 1100 psig and operator opened a relief valve to bring pressure down to 850 psig. Continued to use relief valves to maintain reactor pressure.

-- Lost all ECCS at this point.

NOTE: No method of making up to the reactor vessel, except "A" CRD pump, was available at this time:

RFP lost due to MSIV closure

ASE went to back up control panel and was unsuccessful in opening MSIV's.

1258 With the relief valves in operation, the need for torus cooling became vital. RHR system was unavailable for torus cooling due to loss of 480V reactor MOV boards (no valve control or indication).

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Hour	Event
1258	Control power to RHR pumps was lost momentarily. ASE went to S.D. board and started a RHR pump but it tripped off in 15 seconds.
--	Only 4 relief valves were available (from "C" 250V DC board) at this time due to power loss. As many as three valves were used simultaneously to maintain pressure.
--	Torus water temperatures increased to 99°F before indication lost. Drywell pressure increased slightly after scram (one hour after scram, drywell pressure had increased to 2.25 psig).
--	Attempt was made to put emergency power on RHR valve at local reactor MOV board but was unsuccessful. Also, three seperate attempts were made to manually open the RHR discharge valves, but were unsuccessful due to smoke conditions. Later it was discovered that the RHR pump suction valve was also closed (N.O. Valve).
--	ASE discovered Unit Preferred on both units was out. After restoring Unit Preferred on Unit II, he was able to feed Unit I and II from Unit II MG Set.
1330	Depressurization continued with no make up capability until reactor pressure reached 350 psi. At 350 psig, the condensate booster pump was used to feed through the RFP bypass valve (FCV 3-53). Because water level was so low (~ -120") on safe guards Yarway (L13-46 A & B), the bypass valve was fully opened. When water level reached +30" as shown on panel 9-5, an attempt was made to close down on FCV 3-53. Because of F.W. inverter signal failure which is fed to the feedwater flow controller, the valve failed full open and could not be reclosed.
--	Due to lack of phone communication with AUO, level reached +60" before FCV 3-53 could be throttled.
--	From this time on until 200 psig, reactor level was controlled by the AUO at the manual isolation valve (condensate booster pump). Communication between the AUO and UO was via Pax telephone.
1445 JDM	Pressure 300 psi, level 60". The following were in-op. All ECCS; MSIV's; 7 RV's; RBCCW; Cleanup; recirc; SBGT - B train; EHC; DW cooling 8 of 10 blowers lost; diesel 'C'; diesel 'B' and diesel 'D' local control only.
	The following instruments were in-op. Torus temperature and level, drywell temperature and level, jet-pump flow 10 of 20, reactor flange temperature, APRM's, IRM's, SRM's

9-5

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Hour	Event
1445	ARM's, RPIS 1/2, GE/MAC level 1 of 3, reactor pressure 1 of 3, CRD all, scram discharge voluam, computer, MSL rad monitors.
[illegible]	Fire still out of control. Level 50", pressure 300 psi. Holding level with condensate booster pump. Trying to line up shutdown cooling but smoke in reactor building hampering efforts.
1600	Level 50", pressure 150 psi. Fire still out of control.
1810	Fire still burning. Lost control air, RV's in-op. Reactor starting to pressurize. Started using copious amounts of water on the reactor building fire.
1840	Level 55" press 300 psi & increasing
1850	Fire reported out.
1855	400 psig
1900	420 psig
1930	460 psig holding level with 80 gpm CRD flow (is only water available to the reactor).
2020	540 psig
2130	600 psig - got control air back and opened 2 RV's.
2200	340 psig - have to depressurize slowly, only have 80 gpm going to reactor vessel.
2220	220 psig - holding level with the condensate booster pump.
2250	200 psi, level 36"
2330	130 psig
0010	90 psig
0100	Two SRM's in operation outside the drywell - 10 cps, looks okay.
01[illegible]	RHR loop operable - setting flow by pump amps.
0204	Torus level back - shows +1".
0230	DW pressure back - shows 16.5 psia.
0410	Shutdown cooling on
0930	Cold shutdown, shutdown cooling on. DW temperature 120°F; Torus level +4", temperature 130°F.

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4. Process Variables and Radiation Rates

Obtained from a preliminary TVA report

Unit I

Drywell Temperature

Date	Time	Source	Value
3-22	1600	GE Log	330°F
		TR-80-1	at least 300°F
		TR-64-52	Stopped inking at 370°F

Drywell Pressure

3-22	1600	GE Log	2.3 psi
	1615	GE (PR-76-14)	2.3 psi (went downscale at that value)
	1600	PR-76-14	2.6 psi
3-23	0230	GE (PR-64-50)	1.8 psi

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Reactor Level

Observed to go to -120" on LI-3-46 A & B (46" above TAF) during time reactor was being blown down from ~1060 psi to ~360 psi. All other times was in the range ~ 0 to +70".

Torus Level

Date	Time	Source	Value
3-22	1545		Unknown
	1745	GE (LI-64-54A)	Downscale
	1750	GE (LI-64-54A)	+ 3"
	2400		Unknown
3-23	0204	GE (LI-64-54A)	+ 1"
	0212		+ 1"
	0530		+ 5"
	0940		+ 4"

Torus Temperature

Date	Time	Source	Value
3-22	1500	From 25-32	126°F
	1545		Unknown
	1745	GE (TIS-64-55)	100°F
	1750	GE (TIS-64-55)	110°F
	2300		145°F
	2400		145°F
3-23	0130	("C" RHR on)	177°F
	0530		153°F
	0940		125°F

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Maximum Cooldown Rate of Reactor Coolant (Based on Pressure)

Steady State Conditions 1000 psig - 544°F

m	Pressure	Temperature	Cooldown Rate
1300	1060 psig	551°F	
	26 min	434°F	
	360 psig	117	117°F/hr
1600	450 psig	456°F	
	15 min	377	
	190 psig	79	79°F/hr
2130	580 psig	482°F	
	1 hour		
	250 psig	401	81°F/hr
		81	

Maximum Radiation Level

Airborne Rb^{88} (Appears to be a steam leak in Unit II after the scram)

Unit II Reactor Building	3.5×10^{-7} μci/cc
Unit I Reactor Building	7.5×10^{-8} μci/cc
Control Room	7.0×10^{-8} μci/cc

4. Equipment Damage

At the present time, there is no known equipment damage beyond the cable tray damage. However, there may be some evident as the systems are returned to service. There has not yet been a drywell entry. The following are items which may be indicative of potential equipment damage:

1. Drywell temperature was > 300°F for several hours.
2. RBCCW was off approximately three hours.
3. The main turbine was stopped and off turning gear approximately one hour.
4. CRD temperaures continue to read irratic and are trending up and down.
5. SBGT system B has melted indicator lights in the control room.
6. The majority of the ECCS pumps and turbines and the M/G set - recirc pumps have not yet been started.

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C. CONCLUSIONS

A cable tray fire was experienced at Browns Ferry for a period of six (6) hours and 15 minutes on March 22, 1975. There were no injuries to personnel and no radiation releases from the plant site boundary.

Unit I was effected the most. All of the ECCS and the majority of the other reactor systems were rendered in-operable by the fire. However, primary containment was never breached and the core was never uncovered. At this early stage of the recovery, there is no known equipment damage. Unit II was essentially undamaged.

TVA and NRC have assemblied task forces at the site to investigate the fire and consequences of it.

This report is written two days after the fire; as a result, the facts reported are the best known at this time and should be assumed preliminary.

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ATTACHMENT I

Unit I

In-op equipment <u>during</u> the fire:

- HPCI - Complete
- RHR - Both loops - all valves, pumps A, B, and D.
- CS - Both loops - all valves and pumps
- RCIC - In-board steam isolation valve and flow controller
- MSIV's - All 8 valves
- RV's - Lost 7 valves - lost control air after approximately five (5) hours and all 11 valves in-op for 3 1/2 hours.
- RBCCW - Pump and valves
- Cleanup - All valves
- Recirc - All A loop valves, all lub oil pumps, both field breakers
- Main Turbine - Turning gear
- EHC - Complete system
- SBGT - B train
- Drywell Ventilation - Lost eight of 10 blowers
- Diesels - 'C' inop; B & D local control only

Inop Instruments <u>during</u> the fire:

(Not a complete list)

- Torus Temperature
- Torus Level
- Drywell Temperature
- Drywell Pressure
- Jet Pump Flow - lost 10 of 20
- Reactor flange temperature
- All neutron monitor systems
  APRM IRM
  SRM
- ARM's
- RPIS - 1/2
- B - GE/MAC level
- B - Reactor pressure
- CRD - all
- Scram discharge volume
- Computer
- MSL rad monitors

Unit II

In-op equipment <u>during</u> the fire:

- HPCI - Torus suction valves closed
- RHR (#2) - SDC valve
- ADS - no lights on RV 1-34
  TE on RV 1-31 open
- MSIV - Valve 15A, no lights
- [illegible] (#1) - Pumps A & C, no panel 9-3 control - ok backup panel
- CS (#1) - Pumps A & C, no panel 9-3 control - ok backup panel
- Diesels - 'C' in-op; B & D local control only

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ATTACHMENT II

Unit I

In-op equipment as of March 23, 1975, 1430 hours

- All APRM, IRM
- SRM's in control room OOS (2 SRM's connected locally and being continuously manned)
- Area Rad Monitors 50% OOS, some others are erratic.
- Area Rad Monitors - Refuel floors
- Scram discharge vol vent and drain valves - no power
- Diesel 'C' OOS
- RPIS 50%
- Recirc alternate power control, 1 of 6 lube oil pumps have no power.
- RWCU (pumps have power but not valves) temperature indicator
- DWEDS, DWFDS
- EHC system entirely lost; turbine is on turning gear.
- MSIV's (except for several drain valves)
- Torus temperature readout - 3 points ok, 1 point failed
- ADS - All except one have power. One is in manual open to maintain atmospheric pressure.
- HPCI
- CS II - (B,D pumps have power)
- RHR II - In shutdown cooling (pump B) but have no flow or pressure indication. Most valves are only operable locally.
- RHR I - Pump C in torus cooling with no system pressure or flow indication. A pump OOS
- All RHR and CS area temperatures
- CS I - Appears to be operable.
- RCIC - Flow indicator controller and drywell steam valve.
- ~~RCIC~~
- Fire pump C operable from SDN board only.
- SBGTS B

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ATTACHMENT III

Typical Sealants Used at Cable Tray Penetrations

1) Poly Ureathane

A. "Frothpak"; Insta-Foam Co., Edison, Ill.; ASTM 1692-59T

B. "Selecta-Foam"; Pittsburg Co. 6409/RC266-BC

2) Silicone Rubber

A. "RTV-102" General Electric Company

ATTACHMENT IV

Task Force Personnel as of March 24, 1975

1. NRC

Dr. Steve ~~Hanauer~~ purre	-	Chairman
Norm Mosley	-	Director Region II Compliance, NRC
Dr Don Knuth	-	Director of Inspection and Enforcement, NRC

2. TVA

Harry Fox - Chairman

M.N. Sprouse

F.A. Szczepanski, Jr.

Charles Bonine, Jr.

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TENNESSEE VALLEY AUTHORITY

Division of Engineering Design

Report Number BF-IED(BNP-1)

Physical Damage to Electrical Cables and Raceways Involved in the Browns Ferry Nuclear Plant Fire on March 22, 1975

April 17, 1975

H. C. Russell

H. C. Russell
Browns Ferry Design Project Manager

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TABLE OF CONTENTS

1.0 SUMMARY

This report describes the extent of the physical damage to the cables and the raceway systems involved in the fire at Browns Ferry Nuclear Plant on March 22, 1975. It includes a description of the zone of influence of the fire and a detail listing of all cables, conduits, and cable trays located in this zone. These cables are tabulated with pertinent information and marked off wiring diagrams which clearly define the actual function of each cable. The composition of each type cable is included and the requisitions used to purchase this cable are referenced. These requisitions and other contractual data are not included but are available in permanent contract files maintained by the Electrical Engineering Branch. The design of the secondary containment electrical penetration and the actual materials used in the penetration is defined. Representative cross sections of cable tray tiers are shown with the actual loading and the number of each type cables on each tray is defined. Other detail information, such as applicable portions of conduit and cable tray drawings, definitions and quantities of safety related cables involved, and a listing of all materials feeding the fire is included.

2.0 PURPOSE AND SCOPE

The purpose of this report is to identify the extent of physical damage to the cable tray system, conduit and grounding system, and all cables routed through these raceway systems. This report further shows where damaged cables can be found on electrical wiring diagrams thus identifying their electrical function. This information will be shown by portions of design drawings, sketches, and listings of pertinent cable and raceway data. Excluded from this report are the effects of faults in these cables to mechanical and electrical systems, damage to other equipment resulting from smoke or the chemicals and water used in extinguishing the fire, possible structural damage (other than raceway supports), and possible damage to any equipment located outside the defined zone of influence.

3.0 DESCRIPTION

3.1 Zone of Influence

It has been determined that the fire started when an open flame came into contact with material used in the penetration sealant around the cables where they penetrated through the control bay spreading room wall toward the reactor building of unit 1.

Attachment 9, sheet 1, shows a cross section of trays near the point where fire started. The fire damaged cables and raceways approximately five feet north of the wall penetration inside the spreading room, and propogated along all trays in the reactor building on floor elevation 593 as marked on Attachment 1, sheets 1-4. (Attachment 2 shows affected trays and their intersections in single line representation.) Visible damage in reactor building was observed east along the double stack of 3 trays to the wall between unit 1 and 2, south along the 4 trays to a fire stop approximately 28 feet from wall between the reactor building and control bay, and west along the double stack of 5 trays for a distance of approximately 38 feet from the wall between units 1 and 2. Cables were also damaged on 4 vertical trays from the top about 10 feet down.

Attachment 3, sheets 1-4, shows the zone of influence for all damaged or assumed damaged conduits and grounding systems.

3.2 Identification of Damaged Conduits, Cable Trays, and Cables Routed Through Each Raceway

DED has identified and tabulated 11/ conduits, 8 conduit boxes, 26 cable trays, and 1611 cables routed in these raceways that were damaged or assumed damaged. It is assumed that all supports for the raceway systems were also damaged. Addendum A has an index of damaged or assumed damaged cables and raceways. Blank columns will be completed only for damaged cables at a later date. Attachments 4 and 5 define key letters, prefixes, and suffixes used in the designation of cables listed on Addendum A index. The index contains an alphanumeric listing of cables shown on 204 cable tabulation sheets (Addendum A). These tabulation sheets list pertinent information used by DEC in writing their procedure for the actual verification and removal of damaged cables, including the measured damaged length.

Their procedure will also require that each section of burned cable be identified by its cable number and stored for future examination. Each section of burned cable will include several feet of unburned insulation containing all manufacturer's data stamped on the outer jacket.

Of the total cables identified and listed in Attachment 4, there were 1169 cables for unit 1, 75 for unit 2, 27 for unit 3, and 340 common to plant. It was determined that a total of 628 safety related cables were damaged. These are grouped into categories shown on Attachment 6. Each can be referenced back to tabulation sheets through the index to Addendum A.

The bare ground cable used for grounding the cable tray system was also damaged by the fire. It was routed along the 480V power trays FM, FK, and FO-ESII through the zone of influence.

3.3 Material Feeding the Fire

There were various size conduits (see Attachment 3) made of aluminum and galvanized steel, and only two sizes of ladder type cable trays (18 inches wide by 4 inches deep and 12 inches wide by 4 inches deep). The conduits were purchased per standard TVA specifications (Att. 11 & 12), but the trays were ordered on numerous individual contracts maintained by the Electrical Engineering Branch. However, all trays used in zone of influence were galvanized using a hot dipped process.

Of the 1611 cables, there were 65 different type cables involved in the fire as listed on Attachment 7. Attachment 9, sheets 1, 2 and 3, shows a cross section of the cable trays where the fire started and lists each tray and the type and number of cables found there. Attachment 9, sheets 4-12, lists the number and type of cables at other points along trays as identified by checkpoints on Attachment 2. Types (TVA mark number) WBB through WBF are power and control cables manufactured per TVA standard specifications (Attachments 13 & 14) and are composed of insulating material footnoted in Attachment 7. The remaining types are signal cables which are specified and documented on numerous individual contracts (see Attachment 8) which are in files maintained by the Electrical Engineering Branch. These are composed of insulating material also footnoted in Attachment 7. In all cases, the actual types used will be verified in the removal of cables. The filler materials in these cables and cable ties are included in the listing at the conclusion of this section.

Another "fuel" was the wall penetration pressure sealants between the spreading room and reactor building. A typical penetration is shown in Attachment 10. The sealant material was polyurethane expandable foam, a pressure seal, which is covered with Flamastic 71A fire retardant compound.

The types of urethane foam used at BFNP are as follows. One type is Insta-foam, an aerosol manufactured by Insta-foam Products; there are two pourable types manufactured by North Brothers, one consisting of part number 6403 and 6504, and the other consisting of part numbers 0293A and 67010; and another type is Aire-Lux Foam Rubber manufactured by South-Air Company. We are attempting to secure from appropriate vendors the physical properties and characteristics of the material used as a pressure seal and it will be available at a later date.

Another sealant material which is a possible fuel source would be the silicone rubber compound used in sealing conduits through walls.

The listing below generally summarizes materials acting as fuel sources.

1. Candle
2. Urethane expandable foam
3. Polyethelene (high molecular weight or high density)
4. Nylon
5. Cross-linked polyethelene
6. Polyvinyl-chloride
7. Mylar
8. Aluminum foil and rigid aluminum conduit
9. Polyolefins
10. Chlorosulfonated polyethelene
11. Neoprene
12. Fiberglass
13. Silicone rubber
14. Galvanizing material on raceways
15. Carbon
16. Nonhygroscopic cable filler material
17. Foam rubber
18. Cable ties

4.0 REFERENCES

4.1 Attachments

Attachment 1. Sheets 1-4 are portions of TVA drawings 45N830-6, -16, and -18 showing area of damage along trays.

Attachment 2. Portion of 45W833-2 showing trays and interconnections in single line representation. Cable checkpoints used for logging cables as they are routed are also shown (refer to Attachment 9).

Attachment 3. Portions of 45N804-8, -9, and -19 showing conduits in zone of influence.

Attachment 4. Electrical Design Memorandum 226.01 defining conduit and cable key letters and prefixes.

Attachment 5. Sheets 1-2 defines cable suffixes.

Attachment 6. Listing of safety related cables assumed damaged.

Attachment 7. Sheets 1-2 list cable types and insulating material made from.

Attachment 8. List of requisitions and vendors of cables which were not purchased per a standard TVA specification.

Attachment 9. Sheets 1-12 show cross section where fire started, tray loading at that checkpoint, and tray loading at other checkpoints within zone of influence.

Attachment 10. Sheets 1-2 are portions of 45N830-17 and -27 showing a typical wall penetration.

Attachment 11. TVA Standard Specification 21.000 for rigid aluminum conduit.

Attachment 12. TVA Standard Specification 21.001 for rigid steel conduit.

Attachment 13. TVA Standard Specification 25.013 for polyethelene-insulated wire and cable.

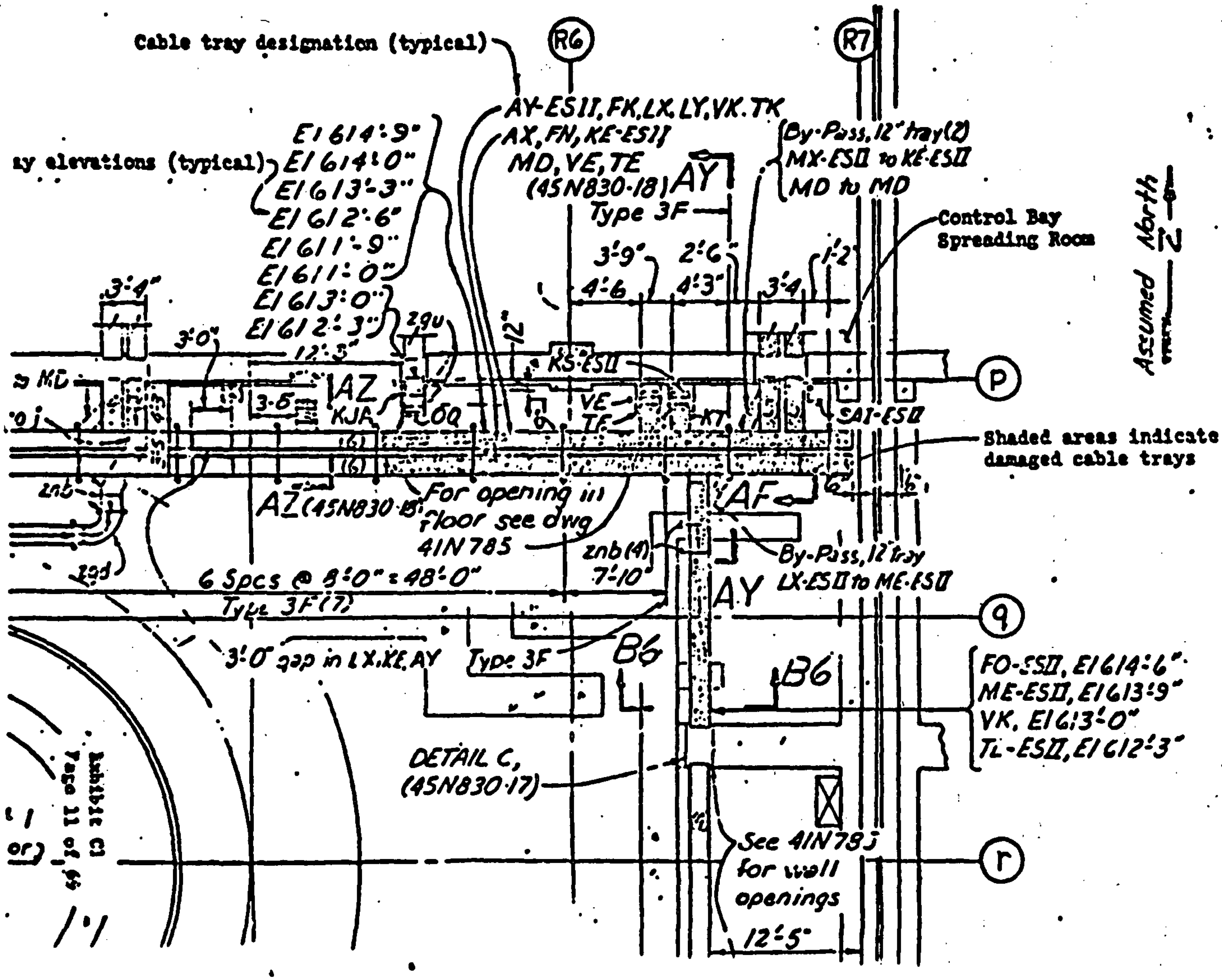
Attachment 14. TVA Standard Specification 25.016 for cross-linked polyethelene-insulated wire and cable.

4.2 Addendums

Addendum A. Includes 204 cable tabulation sheets and an index listing each cable, its purpose, terminations, type (TVA mark number), raceway located in, and electrical drawings used for locating cables as to function.

Addendum B. Includes 315 electrical drawings by vendors and TVA showing where each cable is found per its function.

Addendum C. TVA cable routing checkpoint sheets. See Attachment 2 for location of checkpoints along trays.



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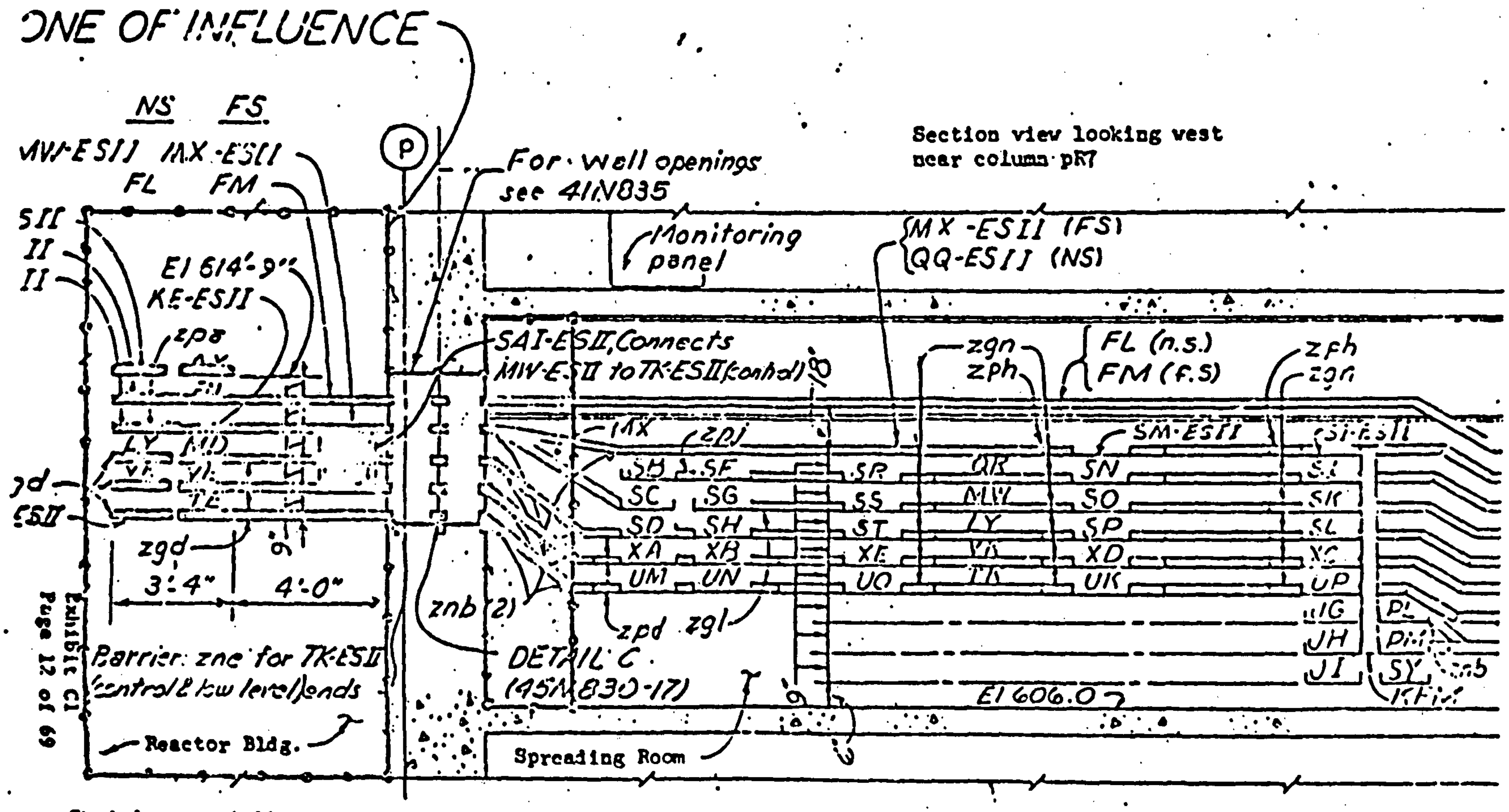
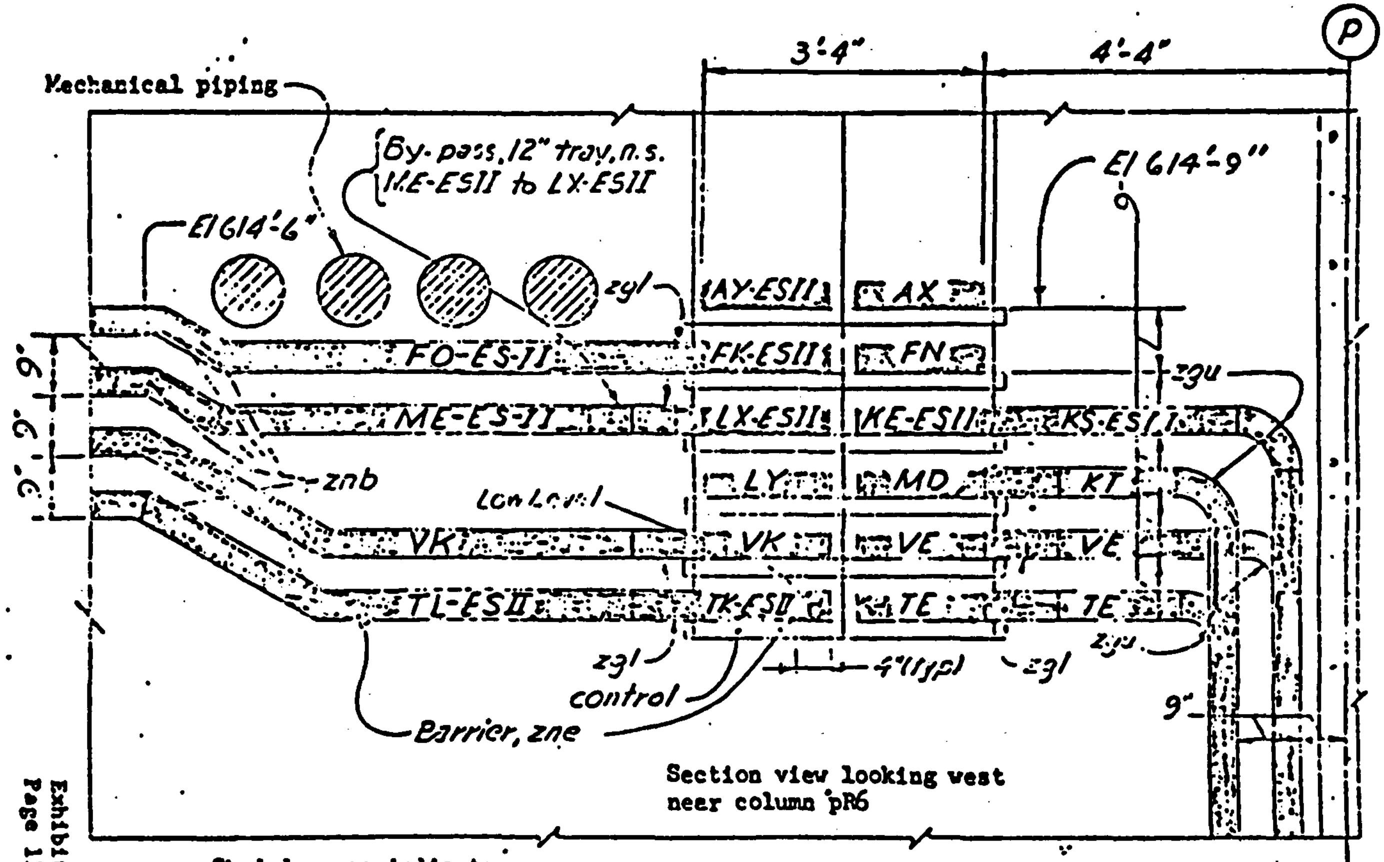


Shaded areas indicate damaged trays

AF - AF

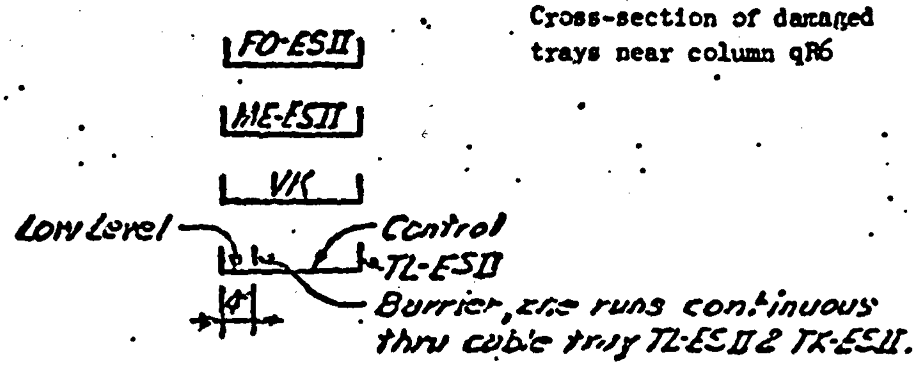
584



Shaded areas indicate damaged trays

AY—AY
(·'5N830-6)

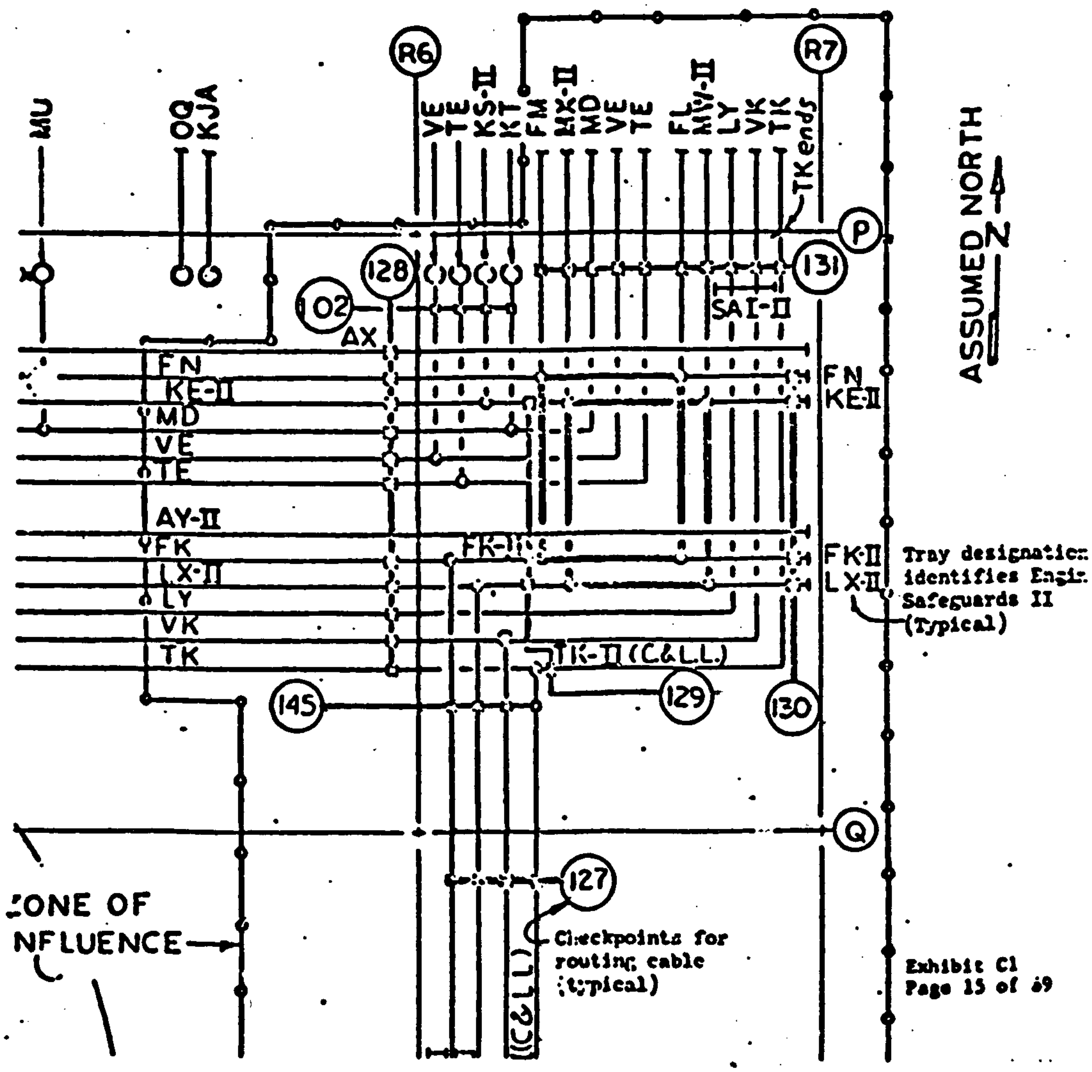
ATTACHMENT 1 SHEET 4



Cross-section of damaged trays near column qR6

SECTION B6-B6
Scale: ½" = 1'-0"

ATTACHMENT 2



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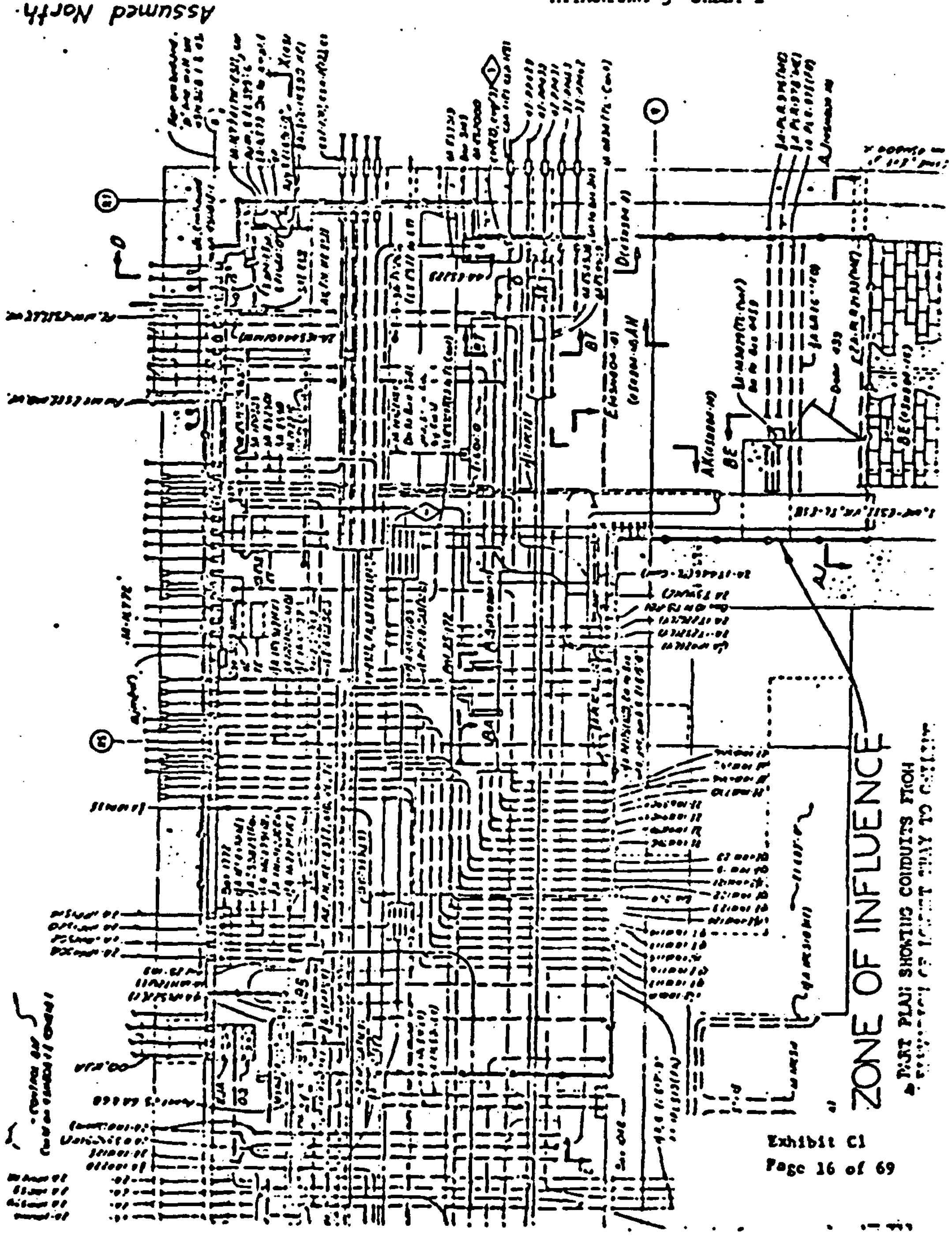
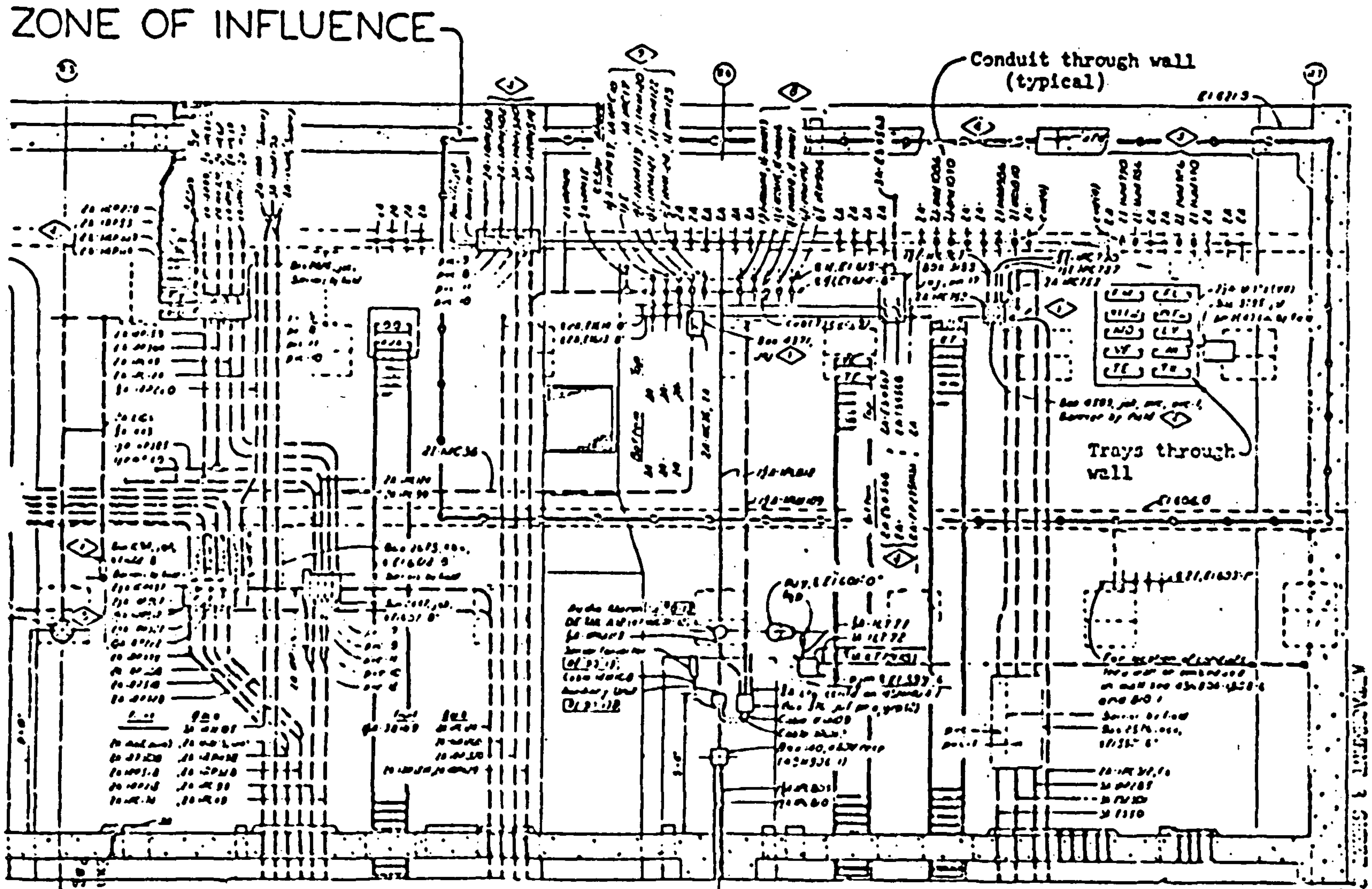
ATTACHMENT 3 SHEET 1

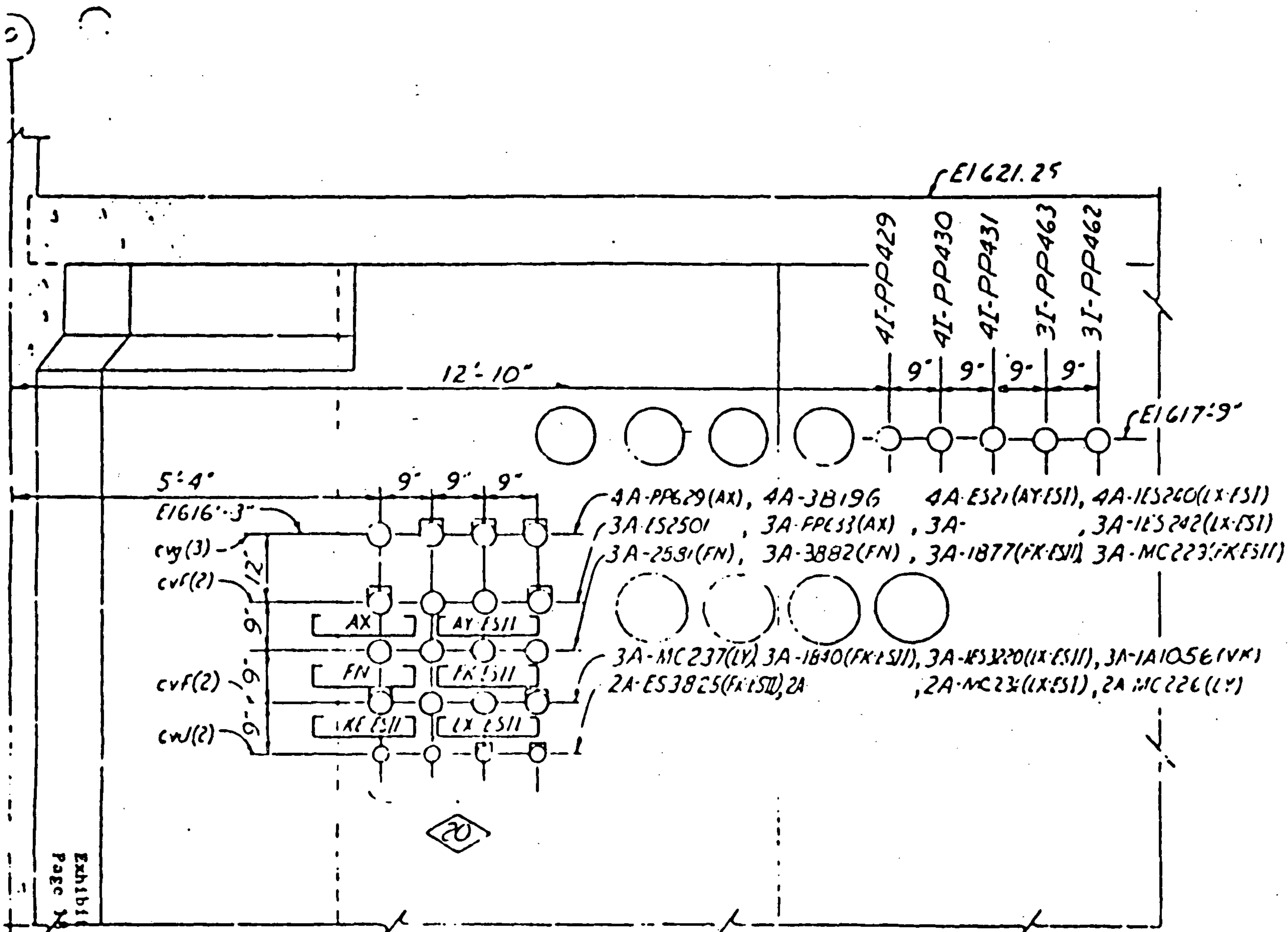


PART PLAN SHOWING CONDUITS FROM

Exhibit C1
Page 16 of 69



ELEVATION VIEW OF WALL BETWEEN REACTOR BUILDING & CONTROL BAY
(LOOKING NORTH)

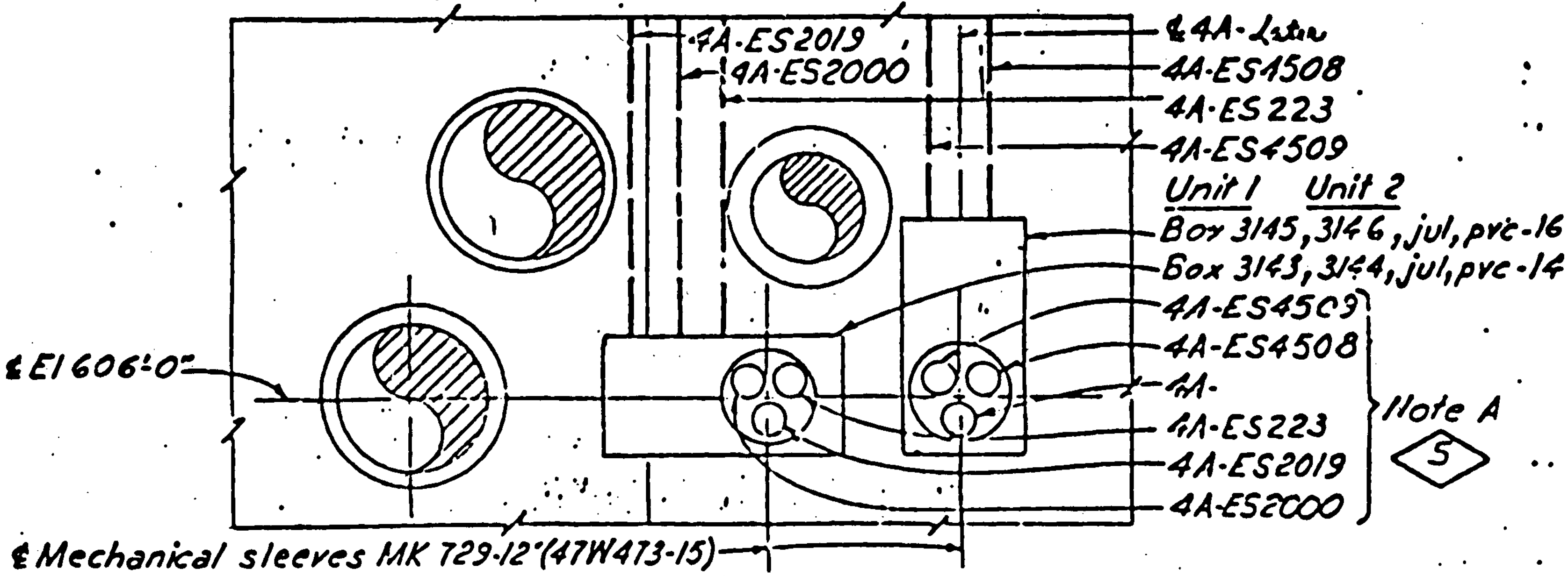


ELEVATION VIEW LOOKING EAST TOWARD UNIT 2
(cable trays run to wall and stop and cables are fed into unit 2 through conduit)

Exhibit C1
Page X of 69

590

ATTACHMENT 3 SHEET 3



SECTION BT-BT & BT1-BT1

BT1-BT1 OPPOSITE HAND

Boxes shown with covers removed.

(45N804-16)

Scale: $\frac{1}{2}$" = 1'-0"

ATTACHMENT 4

226.0181
Page 1
Electrical
Design Memorandum
February 7, 1972

Tennessee Valley Authority
Knoxville, Tennessee

Subject: CONDUIT AND CABLE SCHEDULES
KEY LETTERS AND MATERIAL DESIGNATIONS

Conduits and cables as listed on the schedule are grouped under service headings and assigned numbers for identification on electrical drawings. Each number consists of a key letter (or letters) for the service group, prefixed by a numeral (or other designation as assigned) to denote the unit or particular piece of equipment and followed by a number to show the circuit. Thus, 1G23 indicates circuit 23 for generator 1. The following key letters are used:

Key Letters	Service
A	- Annunciator
AG	- Auxiliary generator
B	- Battery
BBT, BCT, BPT, etc.	- Bus ties, bus sectionalizing, bus PT's, etc. (See below; for exceptions, see Note 1)
C	- Computer (digital inputs)
CC	- Control - Coal-handling logic
CL	- Clock system
CR	- Computer recording (analog inputs)
D	- Data logger
DA	- Data acquisition
DE	- Decontic
E	- Exciter
ES	- Engineering safeguards
FBT, FCT, FPT	- Feeders (see below)
FE	- Fire extinguishers
G	- Generator
GR	- Grounding
J	- Jumper between switchboards
K	- Code-call system
L	- Lighting cabinet and board feeders
LL	- Lock lighting
LS	- Load shedding logic
LT	- Loudspeaker (intercommunication system)
M	- Miscellaneous service
MC	- Conduit for group of nonassociated circuits
MPL	- Power 480V, main 480V switchboard
MS	- Microwave system

Tennessee Valley Authority
Knoxville, Tennessee

Key Letters	Service
NL	- Navigation lights
NM	- Neutron monitoring
P	- Power 240V, auxiliary
PC	- Primary containment
PH	- Power 2400V, auxiliary
PL	- Power 480V, auxiliary
PLA	- Power 480V, common board 1
PLB	- Power 480V, common board 2
PLC	- Power 480V, coal handling
PLL	- Power 480V, lighting boards
PLP	- Power 480V, ash pump boards
PLR	- Power 480V, radiation waste.
PLS	- Power 480V, service and office buildings
PLU	- Power 480V, utility building
PLW	- Power 480V, water supply
PM	- Process miscellaneous
PP	- Power above 2300V, auxiliary
PPC	- Power above 2300V, auxiliary coal handling
PS	- Protection - safeguards
PV	- Power 120V a-c, vital instrument
R	- Recorder
RM	- Radiation monitoring
RP	- Reactor protection
RR	- Reactor control rods
RT	- Radio telephone
S	- Signal system
SB	- Soot blowers
SF	- Fly-ash collection system
SG	- Steam generator
SP	- Spare conduit
SS	- Station service transformers
T	- Telephone system
TBT, TCT, etc.	- Transformers (see below)
TV	- Closed circuit television
V	- Valve control
W	- Carrier current system

Note 1: When a job requires only a few circuits for bus equipment, such circuits may carry the key letter of the main transformer which feeds that bus, i.e., TBT, TCT, TFT, etc.

225.01R1
Page 3
Electrical
Design Memorandum
February 7, 1972

Tennessee Valley Authority
Knoxville, Tennessee

The key letters for bus ties, feeders, and transformer circuits are based on identifying the various voltages. The following letters when prefixed by a B (bus tie), F (feeder), and T (transformer), shall be used for voltages as shown below. In the case of transformers, the letter designating the voltage of the secondary shall always be used:

____ BT - 500 kv	____ FT - 69 kv	____ LT - 11 kv
____ CT - 161 kv	____ GT - 66 kv	____ MT - 6.9 kv
____ DT - 115 kv	____ KT - 13.2 kv	____ ST - 230 kv
		____ UT - 345 kv

Sheet Numbering. The Conduit and Cable Schedule is numbered 45C8CO and is made up of a set of sheets each of which is numbered with the key letter (or letters) of the conduit listed thereon followed by the serial number. Example: Annunciator, first sheet A-1; Annunciator, second sheet A-2, etc.

The index sheets are part of this set and are numbered thus: Ind-1, Ind-2, etc

Conduit Material Designation. For describing the nature of the material for all conduit except that used for branch lighting circuits, add the following designation letters immediately following the numeral indicating the size of the conduit:

A - Aluminum
B - Brass
F - Fiber
FL - Flexible steel
I - Iron (rigid steel)
MT - Metallic tubing (thin wall steel)
T - Transite standard wall
K - Transite thin wall
P - Plastic

Thus, 3I-1C23 indicates 3" iron conduit with designation 1C23 as described above under Key Letters.

This memorandum supersedes Electrical Design Memorandum 225.01 dated July 17, 1957.

(1) JLS
(2) CHS
(3) RAH
(4) MIG:MTC

ATTACHMENT 5 SHEET 1

SUMMARY OF CABLE SUFFIX DEFINITIONS

1. Cables in the PP series with an A or B suffix are to be separated from each other. They are not engineering safeguard cables, but a separate routing is desirable. They involve off-site power.

2. The following suffixes apply to all cable series:

   I - Division I engineering safeguard or Primary Containment Isolation cables

   II - Division II engineering safeguard or Primary Containment Isolation cables

   IA - Diesel generator A shutdown logic cables (may be routed in cable tray with Division I cables)

   IB - Diesel generator B shutdown logic (routed in conduit)

   IIC - Diesel generator C shutdown logic (may be routed in cable tray with Division II cables)

   IID - Diesel generator D shutdown logic cables (routed in conduit)

3. The following suffixes apply to LS series:

   A1 - 480V load shedding logic channel A1: (routed with IA-Diesel A)

   A2 - 480V load shedding logic channel A2: (routed with IB-Diesel B)

   B1 - 480V load shedding logic channel B1: (routed with IIC-Diesel C)

   B2 - 480V load shedding logic channel B2: (routed with IID-Diesel D)

4. The following suffixes apply to RP (Reactor Protection) or NM (Neutron Monitoring) series:

   IA - RPS logic channel A1

   IIA - RPS logic channel A2

   IB - RPS logic channel B1

   IIB - RPS logic channel B2

5. The following suffixes apply to RP (Reactor Protection) series:

   IIIA - RPS manual and back-up scram solenoid channel A

   IIIB - RPS manual and back-up scram solenoid channel B

   A - 120V a-c RPS channels A1, A2, and A3 supply (RPS MG set A)

   B - 120V a-c RPS channels B1, B2, and B3 supply (RPS MG set B)

ATTACHMENT 5 SHEET 2

G2 - RPS scram solenoid Group 2

G3 - RPS scram solenoid Group 3

G4 - RPS scram solenoid Group 4

6. Suffix 1E - Applies to supporting auxiliaries needed for safe shutdown of plant.

ATTACHMENT 6

Number of each class of safety related cables routed in fire zone.

Plant Usage	Number	Safety Classification	Channel or Division*
Common	20	Engineered Safeguard - ECCS	I
	20	Engineered Safeguard - ECCS	II
	13	Engineered Safeguard - Diesel A	IA
	33	Engineered Safeguard - Diesel C	IIC
	5	Engineered Safeguard - Diesel D	IID
	7	Load Shedding - Diesel A	A1
	9	Load Shedding - Diesel C	B1
	7	Supporting Auxiliaries - Electrical	IE
Subtotal	114		
Unit 1	6	Engineered safeguard - ECCS	I
	182	Engineered Safeguard - ECCS	II
	4	Load Shedding - Diesel A	A1
	5	Load Shedding - Diesel C	B1
	1	Load Shedding - Diesel D	B2
	52	Neutron Monitoring (also activates	IA
	52	Neutron Monitoring RPS)	IB
	52	Neutron Monitoring	IIA
	52	Neutron Monitoring	IIB
	14	Primary Containment Isolation	I
	39	Primary Containment Isolation	II
	2	Reactor Protection (control rod	IA
	2	Reactor Protection scram)	IB
	2	Reactor Protection	IIA
	2	Reactor Protection	IIB
	3	Reactor Protection	IIIB
	12	Supporting Auxiliaries - Electrical	IE
Subtotal	482		
Unit 2	15	Engineered Safeguard - ECCS	I
	3	Engineered Safeguard - ECCS	II
	4	Supporting Auxiliaries - Electrical	IE
Subtotal	22		
Unit 3	4	Engineered Safeguards - ECCS	I
	3	Engineered Safeguards - ECCS	II
	3	Supporting Auxiliaries - Electrical	IE
Subtotal	10		
TOTAL	628		

*See Attachment 5 for channel or division definitions.

ATTACHMENT 7 SHEET 1

Summary of cable types involved in fire.

CABLE TYPE (MARK)	DESCRIPTION NO. & SIZE OF CONDUCTORS	DESCRIPTION INSULATED*	NO. CABLES DAMAGED	CABLE TYPE (MARK)	DESCRIPTION NO. & SIZE OF CONDUCTORS	DESCRIPTION INSULATION*	NO. CABLES DAMAGED
WBB	1/c # 12	1	11+	WLB	2/c # 12	4	[illegible]
WCA	1/c # 14	1	2+	WLC	3/c # 12	4	[illegible]
WDD	1/c # 8	2	4+	WLG	7/c # 12	4	[illegible]
WDE	1/c # 6	2	1+	WLN	2/c # 10	4	[illegible]
WDF	1/c # 4	2	7+	WLO	3/c # 10	4	[illegible]
WDG	1/c # 2	2	13+	WLS	7/c # 10	4	[illegible]
WDH	1/c #1/0	2	2+	WNB	1/c #2/0	5	[illegible]
WDI	1/c #2/0	2	2+	WNC	1/c #4/0	5	[illegible]
WDK	1/c #4/0	2	2+	WNF	1/c #500	5	[illegible]
WDN	1/c #300	2	6+	WTA	50 pr #19	6	[illegible]
WDO	1/c #400	2	1+	WTD	12 pr #19	6	[illegible]
WFB	2/c # 10	3	16	WTJ	COAX	7	[illegible]
WFC	3/c # 10	3	1	WTK	COAX	8	[illegible]
WFD	4/c # 10	3	6	WTK-1	COAX	9	[illegible]
WFE	5/c # 10	3	1	WTK-2	COAX	10	[illegible]
WGB	2/c # 12	3	157	WTO	2 pr #18	11	[illegible]
WGC	3/c # 12	3	18	WTR	2 pr #14	12	[illegible]
WGD	4/c # 12	3	37	WUB	Thermocouple	13	[illegible]
WGE	5/c # 12	3	13	WUB-1	Thermocouple	14	[illegible]
WGG	7/c # 12	3	14	WUA	2/c # 16	15	[illegible]
WGI	9/c # 12	3	45	WVA-1	2/c # 18	15	[illegible]
WGK	12/c # 12	3	18	WVB	3/c # 16	15	[illegible]
WGM	16/c # 12	3	2	WVC	4/c # 16	15	[illegible]
WGN	19/c # 12	3	2	WVE	7/c # 16	15	[illegible]
WHB	2/c # 14	3	62	WVG	12/c # 16	15	[illegible]
WHC	3/c # 14	3	15	WVI	27/c # 16	15	[illegible]
WHD	4/c # 14	3	18	WVJ	37/c # 16	15	[illegible]
WHE	5/c # 14	3	13	WVR	29/c # 20	16	[illegible]
WHG	7/c # 14	3	10	WVU	5/c # 18	16	[illegible]
WHH	9/c # 14	3	8	WVU-1	8/c # 18	16	[illegible]
WHJ	9/c # 14	3	7	BELDON 8213	COAX	7	[illegible]
WHL	16/c # 14	3	1	MFR'S TV CABLE	COAX	7	[illegible]
WHT	7/c # 16	3	8	(BELDON 8212)			

+ No. of individual cable designations. Total no. of conductors for each designation appear on Attachment 9 showing cross sections of cables on trays.

* Numbers listed correspond to insulation of cable type as shown below. Refer to standard TVA specifications(Att. 13 & 14)and list of requisitions for signal cable (Attachment 8).

1. Single conductor power or control cable with polyethelene insulation and a nylon jacket over the polyethelene. (Termed "PN" per TVA Specification).

2. Single conductor power or control cable with cross-linked polyethelene insulation and a polyvinyl-chloride insulation jacket over the cross-linked polyethelene. (Termed "CPJ" per TVA Specification).

ATTACHMENT 7 SHEET 2

3. Multiple-conductor cable with a core of the specified number of single conductors as in 1 above covered by a polyvinyl-chloride outer jacket. (Termed "TMJ" per TVA Specification).

4. Multiple-conductor cable with a core of the specified number of single conductors as in 2 above covered by a polyvinyl-chloride outer jacket. (Termed "CPJJ" per TVA Specification).

5. Single conductor high-voltage (5000 volts) power cable with extruded stand and cross-linked polyethene insulation with metallic electrostatic shielding and polyvinyl chloride jacket overall. (Termed "CPSJ" per TVA Specification).

6. Telephone cable with high density polyethelene over each conductor, mylar backed rubber cable tape, aluminum shield, and high density polyethelene jacket overall. Some of these had polyvinyl jacket overall.

7. Coaxial signal cable with both conductor and overall jacket insulated with polyethelene.

8. Coaxial signal cable with conductor insulated with polyethelene and polyvinyl chloride jacket overall.

9. Coaxial signal cable with conductor and overall jacket insulated with irradiated blend of polyolefins and polyethelene and noise free. Some of these types had cross-linked polyethelene over both.

10. Same as 9 but made noise free by a carbon suspension.

11. Same as 6 except without shield.

12. Twisted pair cable with polyethelene over each conductor and polyvinyl chloride jacket overall.

13. Thermocouple cable with high density polyethelene over each conductor, aluminum foil/mylar type laminated shield, and high density polyethelene overall.

14. Thermocouple cable with heat and light stabilized cross-linked polethelene over each conductor, aluminum foil/mylar tape shield, and chlorosulfonated polyethel jacket overall.

15. Signal cable with heat and light stabilized cross-linked polyethelene over each conductor, aluminum foil/mylar tape laminated shield, fiberglass reinforced silicone tape assembly wrap, and chlorosulfonated polyethelene jacket overall.

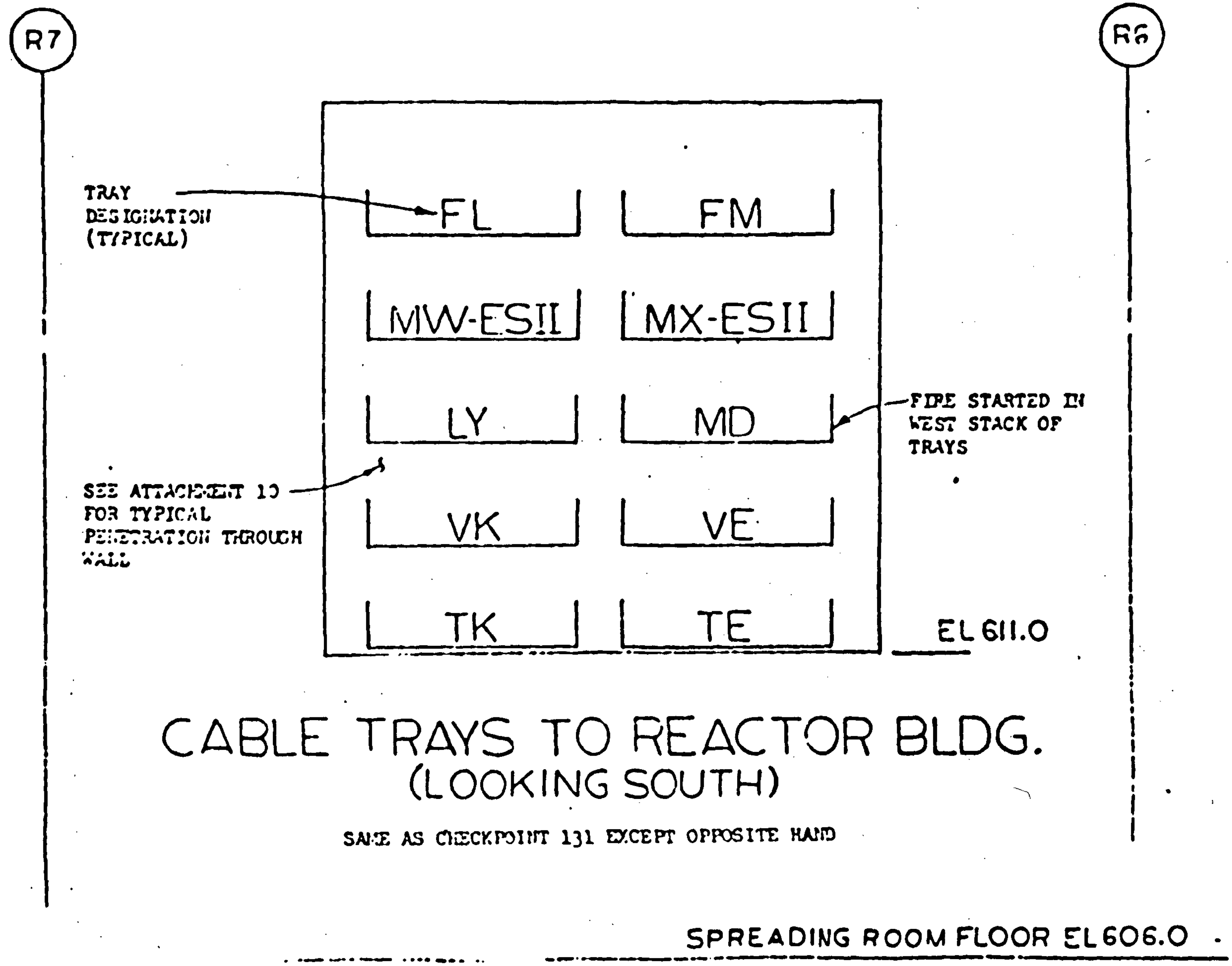
16. Multiple-conductor cable with core of specified number of single conductor cables insulated with cross-linked polyethelene and a neoprene jacket overall.

ATTACHMENT 8
SHEET 1
LIST OF REQUISITIONS FOR CABLE NOT PURCHASED
PER STANDARD TVA SPECIFICATIONS

CABLE TYPE(MARK)	REQUISITION NO.	ITEM	VENDORS
WTA	83451	1	Pwr, Telephone & Supply Company
	91972	1	Anaconda
WTD	75166	2	Anixter
	83451	4	Pwr, Telephone & Supply Company
	85202	3	Plastic Wire & Cable
	86712	1	Phalo Corporation
WTJ	61988	3	Okonite
	83470	2	Bo[illegible]lurant Bros. Company
	84375	1	Dearborn
	84423	1	Alpha Wire
	85488	1	Belden
WTK	61737	4	Continental Wire & Cable
	61988	2	Alpha Wire
	84375	2	Dearborn Wire & Cable
	84453	1	Delco
	85488	2	Belden
WTK-1	61737	1	Brandel-Stephens Company
	75321	1	Boston Wire & Cable
WTK-2	75321	2	Boston Wire & Cable
WTO	83451	7	Pwr, Telephone & Supply Company
	91972	5	Sumitomo Shoji
WTR	64863	12	Jefferson Wire & Cable
	83451	8	Plastic Wire & Cable
WUB	61959	1	Continental Wire & Cable
WUB-1	54336	1	Continental Wire & Cable
	83427	1	Continental Wire & Cable
	85464	1	Continental Wire & Cable
WVA	54335	1	Boston Ins. Wire & Cable
	64863	1	Graybar Electric
	74910	13	Continental Wire & Cable
	83944	1	Continental Wire & Cable
WVA-1	54335	1	Boston Ins. Wire & Cable
	61986	1	Continental Wire & Cable
	74910	2	Continental Wire & Cable
	83944	1	Continental Wire & Cable

ATTACHMENT 8 SHEET 2

CABLE TYPE(MARK)	REQUISITION NO.	ITEM	VENDORS
WVB	54335	3	Boston Ins. Wire & Cable
	64863	2	Graybar Electric
	74910	3	Okonite Wire & Cable
	83849	1	Boston Ins. Wire & Cable
WVC	54335	4	Boston Ins. Wire & Cable
	64863	3	Graybar Electric
	74910	4	Okonite Wire & Cable
	83849	2	Boston Ins. Wire & Cable
WVI	54335	6	Boston Ins. Wire & Cable
	64863	9	Graybar Electric
	74910	10	Okonite Wire & Cable
WVJ	54335	7	Boston Ins. Wire & Cable
	64863	10	Graybar Electric
	74910	11	Okonite
	83849	7	Boston Ins. Wire & Cable
WVR	61844		Sies Electric Supply Company
	85529		ITT Surprenant
WVU	61985	1	General Electric
	75025	1	General Electric
	83426	1	General Electric
WVU-1	61985	2	General Electric
	75025	2	General Electric
	83426	2	General Electric
	85551	1	General Electric


R7
R6
TRAY DESIGNATION (TYPICAL)
FL
FM
MW-ESII
MX-ESII
LY
MD
FIRE STARTED IN WEST STACK OF TRAYS
SEE ATTACHMENT 10 FOR TYPICAL PENETRATION THROUGH WALL
VK
VE
TK
TE
EL 611.0
CABLE TRAYS TO REACTOR BLDG.
(LOOKING SOUTH)
SAME AS CHECKPOINT 131 EXCEPT OPPOSITE HAND
SPREADING ROOM FLOOR EL 606.0

(Looking South)

ATTACHMENT 9 SHEET 2

TRAY DESIG	CABLE TYPE	QTY	CABLE OD	TOTAL AREA	TRAY DESIG	CABLE TYPE	QTY	CABLE OD	TOTAL AREA
[illegible]	WDF	2	.429	.290	ML	WDK	6	.940	4.164
	WDG	12	.485	2.220		WLB	1	.509	.204
	WDK	3	.769	1.395		WLH	1	.559	.245
	WDM	3	.940	2.082		WLO	1	.627	.309
	WDO	4	1.029	3.332					
	WLC	1	.539	.229	TOTAL		9		4.962
TOTAL		25		9.573	MW-ESII	WFB	9	.490	1.701
						WFD	1	.600	.283
MX-ESII	WDD	2	.340	.182		WGB	23	.425	3.266
	WDE	2	.379	.226		WGC	6	.444	.930
	WFB	1	.429	.145		WGD	3	.484	.552
	WFE	2	.659	.684		WGE	2	.559	.492
	WGB	58	.425	8.226		WGI	3	.710	1.183
	WGC	1	.444	.155		WGK	3	.789	1.47
	WGD	3	.484	.552		WHB	14	.384	1.624
	WGE	4	.559	.984		WHE	5	.480	.905
	WGG	4	.660	1.026		WHJ	1	.710	.396
	WGI	37	.710	14.652		WLB	1	.509	.204
	WGK	3	.789	1.47		WTA	1	1.139	1.020
	WGM	1	.874	.600		Belden			
	WHB	26	.384	3.016		8213	1	.405	.129
	WHC	10	.405	1.290	TOTAL		73		11.919
	WHD	4	.439	.608					
	WHE	7	.480	1.267					
	WHG	4	.529	.848					
	WHI	4	.640	1.288					
	WHJ	4	.710	1.584	LY	WDD	2	.340	.182
	WHL	1	.781	.479		WFB	1	.490	.189
	WVU-1	2	.439	.304		WFD	3	.600	.849
	WWN	1	.148	.0172		WGB	8	.425	1.136
						WGC	2	.444	.310
TOTAL		181		38.290		WGD	2	.484	.368
						WGE	5	.559	1.230
MD	WGB	1	.425	.142		WGG	5	.660	1.710
	WGE	1	.559	.246		WGK	1	.789	.490
	WHB	1	.384	.116		WHB	6	.384	.696
	WHD	2	.439	.304		WHC	1	.405	.129
	WHE	1	.480	.181		WHI	3	.640	.966
	WTO	1	.340	.091		WVA-1	2	.333	.174
	WTR	1	.360	.102					
	WVE	1	.461	.167	TOTAL		41		5.783

ATTACHMENT 9 SHEET 3

Checkpoint 131
(Looking South)
(Continued)

TRAY DESIG	CABLE TYPE	QTY	CABLE OD	TOTAL AREA
HD (Continued)	WVG	1	.587	.271
	WVI	11	.834	7.205
	WVJ	16	1.012	12.880
	WVR	101	.650	33.532
	WVU-1	3	.439	.456
TOTAL		141		55.693
VE	WVA	52	.353	5.096
	WVA-1	61	.333	5.307
	WVB	15	.371	1.620
	WVC	10	.401	1.260
	MFRS	1	.242	.046
TOTAL		139		13.329
TE	WUB	37	.231	1.974
	WUB-1	47	.339	4.230
	WVA	3	.353	.2940
TOTAL		87		6.498

TRAY DESIG	CABLE TYPE	QTY	CABLE OD	TOTAL AREA
VK	WVA	41	.353	3.332
	WVA-1	46	.333	4.002
TOTAL		87		7.334
TK	WUB	34	.231	1.427
	WUB-1	25	.339	2.250
	WVA	2	.353	.196
TOTAL		61		3.874

Checkpoint 102
(Looking North)

ATTACHMENT 9 SHEET 4

VE KS-ESII

TE KT

TRAY DESIG	CABLE TYPE	QTY	CABLE OD	TOTAL AREA
VE	WVA	67	.353	6.566
	WVA-1	38	.333	3.306
	WVB	3	.371	.324
	WVC	14	.401	1.764
	MFR	1	.242	.046
TOTAL		123		12.006
TE	WUB	117	.231	4.914
	WUB-1	25	.339	2.250
TOTAL		142		7.164

TRAY DESIG	CABLE TYPE	QTY	CABLE OD	TOTAL AREA
KS-ESII	WFB	1	.490	.18
	WFE	2	.659	.65
	WGB	40	.425	5.68
	WGC	3	.444	.46
	WGD	2	.484	.36
	WGE	1	.559	.24
	WGG	3	.660	1.02
	WGI	1	.710	.39
	WGK	1	.789	.49
	WHB	6	.384	.69
	WHC	2	.405	.25
	WHE	4	.480	.60
	WHG	1	.519	.21
	WHI	3	.640	.96
	WHJ	1	.710	.39
	WLB	1	.509	.20
TOTAL		72		12.
KT	WGB	1	.425	.14
	WGG	1	.660	.34
	WHB	2	.384	.23
	WHD	7	.439	1.06
	WHG	1	.519	.21
	WTO	1	.340	.09
	WTR	1	.360	.10
	WVE	1	.461	.16
	WVG	1	.587	.27
	WVI	11	.834	7.20
	WVJ	8	1.012	6.44
	WVR	101	.650	33.53
	WVU-1	2	.439	.30
TOTAL		138		50.10

ATTACHMENT 9 SHEET 5

Checkpoint 127
(Looking North)

FO-ESII

ME-ESII

VK

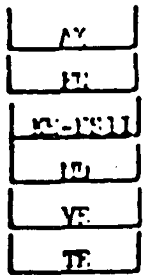
TL-ESIIC | TL-ESIIL

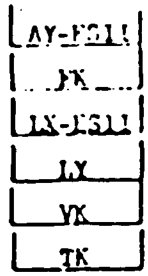
TRAY [illegible]	CABLE TYPE	QTY	CABLE OD	TOTAL AREA
FO-ESII	WDD	3	.340	.273
	WDF	4	.429	.580
	WDG	7	.485	1.295
	WDH	3	.619	.903
	WDI	3	.660	1.026
	WLB	1	.509	.204
	WLC	4	.539	.916
	WLN	1	.559	.245
	WLO	4	.627	1.236
TOTAL		30		6.678
ME-ESII	WDE	2	.379	.226
	WDD	4	.340	.364
	WFB	1	.490	.189
	WGB	27	.425	3.834
	WGD	3	.484	.552
	WGE	4	.559	.984
	WGG	1	.660	.342
	WGI	35	.710	13.860
	WGK	5	.789	2.450
	WGM	1	.874	.600
	WHB	6	.384	.696
	WHC	9	.405	1.161
	WHD	4	.439	.608
	WHE	5	.480	.905
	WHG	4	.519	.848
	WHI	4	.640	1.288
	WHJ	7	.710	2.772
	WHL	1	.781	.479
	WLB	1	.509	.204
	WTO	8	.340	.728
	WTR	1	.360	.102
	WVU-1	2	.439	.304
	WWN	2	.0172	.0344
TOTAL		137		33.530

TRAY [illegible]	CABLE TYPE	QTY	CABLE OD	TOTAL AREA
VK	WVA	8	.353	.78
	WVA-1	20	.333	1.74
	WVB	9	.371	.97
TOTAL		37		3.49
TL-ESIIC	WFB	8	.490	1.51
	WFD	1	.600	.28
	WGB	21	.425	.28
	WGC	5	.444	.77
	WGD	1	.484	.18
	WGE	2	.559	.49
	WGI	2	.710	.79
	WGK	3	.789	.14
	WHB	6	.384	.69
	WHD	1	.439	.15
	WHE	1	.480	.1
	WHG	4	.519	.8
	WTO	9	.340	.8
	WTR	1	.360	.10
	Belden 8213	1	.405	.12
TOTAL		70		7.39
TL-ESIIL	WUB	40	.231	1.6
	WUB-1	4	.339	.3
	WVA	8	.353	.7
TOTAL		52		2.8

Checkpoint 128
(Looking East)

ATTACHMENT 9 SHEET 6





TRAY DESIG	CABLE TYPE	QTY	CABLE OD	TOTAL AREA
AX	WHB	6	.915	3.942
FN	WLC	1	.539	.229
	WLS	1	.817	.524
	WDG	12	.485	2.220
	WDN	13	.940	9.022
	WGD	5	.484	.920
TOTAL		32		12.915
KE-ESII	WIE	2	.480	.362
	WHJ	3	.710	1.198
	WDD	2	.340	.182
	WFE	1	.659	.342
	WGB	4	.425	.568
	WGG	1	.660	.342
	WHB	22	.364	2.552
	WWN	1	.148	.0172
TOTAL		36		5.708
MD	WID	7	.439	1.064
	WIE	1	.480	.181
	WHG	1	.519	.212
	WGB	9	.425	1.278
	WGE	1	.559	.246
	WGG	1	.660	.342
	WHB	2	.384	.232
	WVA-1	2	.333	.174
	WVI	1	.834	.655
	WVJ	8	1.012	5.240
	WVU-1	1	.439	.152
TOTAL		34		9.776

TRAY DESIG	CABLE TYPE	QTY	CABLE OD	TOTAL AREA
AY-ESII	0			
FK	WDF	8	.429	1.160
	WDG	10	.485	1.850
	WDH	5	.619	1.505
	WDI	3	.660	1.026
	WDK	1	.769	.465
	WDN	8	.940	5.552
	WDO	4	1.029	3.332
	WFB	2	.490	.378
TOTAL		41		15.268
LX-ESII	WHC	1	.405	.129
	WIE	1	.480	.181
	WTO	7	.340	.637
	WGC	1	.444	.155
	WGD	1	.484	.184
	WGE	2	.559	.492
	WGG	1	.660	.342
	WGI	1	.710	.396
	WGK	2	.789	.980
TOTAL		17		3.456
LY	WTD	1	.638	.320
	WTO	24	.340	2.184
	WTR	2	.360	.204
	WDD	2	.340	.182
	WFB	1	.490	.189
	WFD	4	.600	1.132
	WGB	10	.425	1.420
	WGC	2	.444	.310
	WGD	1	.484	.184
	WGE	4	.559	.984
	WGG	5	.660	1.710
TOTAL		56		8.819

ATTACHMENT 9 SHEET 7

Checkpoint 128
(Looking East)
(Continued)

TRAY DESIG	CABLE TYPE	QTY	CABLE OD	TOTAL AREA
VE	WVA	94	.353	9.[illegible]12
	WVA-1	85	.333	7.[illegible]95
	WVB	12	.317	1.296
	WVC	4	.401	.504
TOTAL		195		18.407
TE	WUB	80	.231	3.360
	WUB-1	23	.339	2.070
	WVA	3	.353	.294
TOTAL		106		5.724

TRAY DESIG	CABLE TYPE	QTY	CABLE OD	TOTAL AREA
LY	WGK	1	.789	.490
	WHB	6	.384	.696
TOTAL		7		1.186
VK	WVA	29	.353	2.842
	WVA-1	91	.333	7.917
	WVB	9	.371	.972
TOTAL		129		11.731
TK	WUB	16	.231	.672
	WVA	6	.353	.588
TOTAL		22		1.260

Checkpoint 129
(Looking East)

ATTACHMENT 9 SHEET 8

TRAY DESIG	CABLE TYPE	QTY	CABLE OD	TOTAL AREA
AX		0	0	0
FN		0	0	0
KE-ESII	WDD	2	.340	.182
	WFB	1	.490	.189
	WFE	1	.659	.342
	WGB	43	.425	6.106
	WGC	3	.444	.465
	WGD	6	.484	1.104
	WGE	1	.559	.246
	WGG	4	.660	1.368
	WGI	1	.710	.396
	WHB	26	.384	3.016
	WHC	2	.405	.258
	WHE	4	.480	.724
	WHG	2	.519	.424
	WHI	2	.640	.644
	WHJ	4	.710	1.584
	WLB	1	.509	.204
	WWN	1	.0172	.0172
TOTAL		104		18.024
MD		0	0	0
VE		0	0	0
TE		0	0	0

TRAY DESIG	CABLE TYPE	QTY	CABLE OD	TOTAL AREA
AY-ESII		0	0	0
FK-ESII	WDF	10	.429	1.45
	WDG	7	.485	1.29[illegible]
	WDH	2	.619	.602
	WDK	4	.769	1.860
	WDM	8	.940	5.552
	WDO	4	1.029	3.332
	WFB	2	.490	.378
	WHB	1	.384	.116
	WLB	1	.509	.20[illegible]
	WLN	1	.559	.2[illegible]
	WLO	2	.627	.6[illegible]
	WVA	2	.353	.1[illegible]
TOTAL		44		14.5[illegible]
IX-ESII	WDD	4		.364
	WDE	2		.226
	WFB	1	.490	.189
	WGB	25	.425	3.550
	WGD	3	.484	.552
	WGE	5	.559	1.230
	WGG	2	.660	.684
	WGI	36	.710	14.256
	WGK	5	.789	2.450
	WGM	1	.874	.600
	WHB	6	.384	.696
	WHC	9	.405	1.161
	WHD	4	.439	.608
	WHE	5	.480	.905
	WHG	3	.519	.636
	WHI	4	.640	1.288
	WHJ	7	.710	2.872
	WHL	1	.781	.479
	WLB	1	.509	.204
	WTA	1	1.139	1.000
	WVU-1	2	.439	.304
	WWN	2	.0172	.034
TOTAL		123		33.718

ATTACHMENT 9 SHEET 9

Checkpoint 129
(Looking East)
(Continued)

TRAY DESIG	CABLE TYPE	QTY	CABLE OD	TOTAL AREA	TRAY DESIG	CABLE TYPE	QTY	CABLE OD	TOTAL AREA
					TK-ESIIC	WFB	8	.490	1.512
						WFD	1	.600	.283
						WGB	16	.425	2.272
						WGC	6	.444	.930
						WGD	2	.484	.368
						WGE	2	.559	.492
						WGI	2	.710	.792
						WGK	4	.789	1.960
						WHB	4	.384	.464
						WHC	1	.405	.129
						WHD	1	.439	.152
						WHE	2	.480	.362
						WHG	4	.519	.848
						WTO	1	.340	.091
						WTR	1	.360	.102
						Belden 8213	1	.405	.129
					TOTAL		56		10.886
					TK-ESIIL	WHB	1	.384	.116
						WUB	34	.231	1.428
						WUB-1	4	.339	.360
						WVA	4	.353	.392
					TOTAL		43		2.296

Checkpoint 130
(Looking West)

ATTACHMENT 9 SHEET 10

AX
FH
KE-ESII

AY-ESII
FK-ESII
LX-ESII

TRAY DESIG	CABLE TYPE	QTY	CABLE OD	TOTAL AREA
AX		0		
FH	WDH	4	.940	2.576
	WLS	1	.817	.524
TOTAL		5		3.100
KE-ESII	WLB	1	.509	.204
TOTAL		1		.204

TRAY DESIG	CABLE TYPE	QTY	CABLE OD	TOTAL AREA
AY-ESII		0		
FK-ESII	WDF	8	.429	1.160
	WDG	8	.485	1.480
	WDH	2	.619	.602
	WDK	1	.769	.465
	WDH	8	5.552	.940
	WFB	2	.490	.378
	WLO	1	.627	.309
TOTAL		30		5.334
LX-ESII	WGB	4	.425	.568
	WGD	1	.484	.184
	WGG	1	.660	.342
	WGK	2	.789	.980
	WHB	2	.384	.232
TOTAL		10		2.306

Check Point 145

(Looking North)

ATTACHMENT 9 SHEET 11

FO-ESII

ME-ESII

VK

TL-ESIIC TL-ESIIL

TRAY DESIG	CABLE TYPE	QTY	CABLE OD	TOTAL AREA
FO-ESII	WDD	3	.340	.273
	WDF	2	.429	.290
	WDG	6	.485	1.110
	WDH	1	.619	.301
	WDI	1	.660	.342
	WLB	1	.509	.204
	WLC	1	.539	.229
	WLN	1	.559	.245
	WLO	4	.627	1.236
TOTAL		20		4.230
ME-ESII	WDE	2	.379	.226
	WDD	4	.340	.364
	WFD	1	.600	.283
	WGB	27	.425	3.834
	WGD	3	.484	.552
	WGE	4	.559	.984
	WGG	1	.660	.342
	WGI	35	.710	13.860
	WGK	3	.789	.147
	WGM	1	.874	.600
	WHB	5	.384	.580
	WHC	8	.405	1.032
	WHD	4	.439	.608
	WHE	5	.480	.905
	WHG	4	.519	.848
	WHI	4	.640	1.288
	WHJ	7	.710	2.772
	WHL	1	.781	.479
	WLB	1	.509	.204
	WTA	1	1.139	1.020
	WTO	8	.340	.728
	WTR	1	.360	.102
	WVU	2	.376	.222
	WWN	2	.0172	.034
TOTAL		134		32.014

TRAY DESIG	CABLE TYPE	QTY	CABLE OD	TOTAL AREA
VK	WVA	10	.353	.980
	WVA-1	22	.333	1.914
	WVB	9	.371	.972
TOTAL		41		3.866
TL-ESIIC	WFB	8	.490	1.512
	WFD	1	.600	.283
	WGB	29	.425	4.118
	WGC	5	.444	.775
	WGD	1	.484	.184
	WGE	4	.559	.984
	WGI	2	.710	.792
	WGK	3	.789	.147
	WHB	6	.384	.696
	WHC	1	.405	.129
	WHD	1	.439	.152
	WHE	2	.480	.362
	WHG	4	.519	.848
	WTO	8	.340	.728
	WTR	1	.360	.102
	BELDEN 8213	1	.405	.129
TOTAL		77		11.941
TL-ESIIL	WUB	50	.231	2.100
	WUB-1	4	.339	.360
	WVA	8	.353	.784
TOTAL		62		3.244

ATTACHMENT 9 SHEET 12

SAI-ESII

TRAY DESIG	CABLE TYPE	QTY	CABLE OD	TOTAL AREA	TRAY DESIG	CABLE TYPE	QTY	CABLE OD	TO AREA
SAI-ESII	WFB	8	.490	1.512					
	WFD	1	.600	.283					
	WGB	19	.425	2.698					
	WGC	4	.444	.620					
	WGD	2	.484	.368					
	WGE	2	.559	.492					
	WGI	2	.710	.792					
	WGK	3	.789	1.470					
	WHB	6	.384	.696					
	WHC	1	.405	.129					
	WHD	1	.439	.152					
	WHE	2	.480	.362					
	WHG	4	.519	.848					
	Belden 8213	1	.405	.129					
TOTAL		56		10.551					

25.013
TENTATIVE REVISION
OCTOBER 24, 1973

ATTACHMENT 13

TENNESSEE VALLEY AUTHORITY

STANDARD SPECIFICATION

FOR

POLYETHYLENE-INSULATED WIRE AND CABLE

A. SCOPE

*1. This specification covers single- and multiple-conductor wire and cable insulated with polyethylene, for installation in wet or dry locations with conductor temperatures not exceeding 75 C and for use in auxiliary power, control, and lighting circuits at voltages ranging from 600 to 8000 volts, 60 Hz AC, between conductors.

*2. All cable will be installed in conduits, in underground ducts, or on cable trays.

B. APPLICABLE SPECIFICATIONS

*1. The current revision to the following specifications of the issue in effect on date of the invitation to bid shall form a part of this specification:

*a. IPCEA-NEMA Standard S-61-402--NEMA WC5, IPCEA-NEMA Standards Publication, Thermoplastic-Insulated Wire and Cable. Specific references herein are from the Second Edition dated January 1968.

*b. IPCEA-NEMA Standard S-19-81--NEMA WC3, IPCEA-NEMA Standards Publication, Rubber-Insulated Wire and Cable. Specific references herein are from the Fifth Edition dated July 1969.

*c. All references herein are to IPCEA S-61-402 unless S-19-81 is mentioned specifically.

C. GENERAL REQUIREMENTS

*1. TVA type designation, voltage rating, conductor size, stranding if other than class B, special Quality Assurance Procedures if required, and any other specific requirements or exceptions to this specification are stated in the Schedule of Prices.

2. Type designations are TVA designations and are described as follows:

*a. Type PN, single-conductor wire or cable, with polyethylene insulation and a nylon insulation jacket over the polyethylene.

C. GENERAL REQUIREMENTS, continued
C 2, continued

*b. Type PJ, single-conductor wire or cable, with polyethylene insulation and a polyvinyl-chloride insulation jacket over the polyethylene.

c. Type MJ, multiple-conductor cable with a core of the specified number of type PJ single conductors covered by a polyvinyl-chloride outer jacket.

d. Type PJJ, multiple-conductor cable with a core of the specified number of type PJ single conductors covered by a polyvinyl-chloride outer jacket.

*e. Type PSJ, single-conductor high-voltage cable with extruded strand and insulation shielding and with metallic electrostatic shielding.

*3. All outer covering of single-conductor wire or cable No. 8 AWG and larger and the outer jacket of all multiple-conductor cable shall have legible indent printing or legible permanent surface marking to show the following identification on approximately each foot of cable:

a. Manufacturer.

*b. The TVA contract number.

*c. TVA type designation (see section C.2. above) and voltage rating.

d. Size and number of conductors.

*4. Sequential footage marking on single-conductor cable, No. 8 AWG and larger, and on all multiconductor cable shall be supplied. The numerical markings shall be legible indent printing or legible permanent surface marking on the outside surface of the cable at 2-foot intervals and consist of 5-digit numbers increasing sequentially in increments of 2 feet (00020, 00022, 00024, etc.). The sequential markings may begin and end at any two numbers within the range of 00000 to 99998. Sequential footage markings shall be to a tolerance of ±2 percent.

*5. If indent printing is used for section C.3. or C.4. above, the depth of the indentations shall not be greater than 0.007 inch.

C. GENERAL REQUIREMENTS, continued

6. Each bidder shall submit with each copy of his bid complete information regarding the cable he proposes to furnish, including the following:

*a. Type designation and trade name of insulation, insulation jacket, and outer jacket materials.

b. Makeup of multiple-conductor cables.

*c. Performance records of representative installations of the same type and construction of wire and cable offered. If performance records have been submitted in a previous bid or are on file with TVA, an identifying reference shall be acceptable in lieu of a repetition of the data.

7. Color coding of multiple-conductor cable shall be furnished in accordance with IPCEA Section 5.6.3.1.1., Method 1, Table 5-1. All single-conductor wire or cable shall be black.

D. DETAILED REQUIREMENTS

1. Conductors shall be made of soft or annealed copper in conformity with IPCEA Part 2. All conductor sizes shall have concentric class B stranding and may be coated or uncoated.

*2. The insulation over each conductor shall consist of extruded heat-stabilized polyethylene, except that insulation not having a PVC jacket shall also be light-stabilized, applied concentrically about the conductor and shall meet the test requirements of section E.2.d. In-plant repairs of the insulation on shielded cable are prohibited unless specifically agreed to by TVA. All nominal polyethylene thickness shall conform to IPCEA Table 3-2, Column B or Ungrounded Neutral as applicable and except as noted in section D.5.

*3. The nylon insulation jacket covering the insulation on single-conductor cable or single conductors of multiple-conductor cable shall consist of extruded nylon applied concentrically over the polyethylene. The nylon shall have a melting point substantially higher than that of polyethylene. Average thickness of the nylon jacket shall be not less than 4 mils and the minimum thickness shall not be less than 3 mils. No substitution of nylon for an equal thickness of polyethylene insulation shall be permitted.

*4. The polyvinyl chloride insulation jacket covering the insulation on single-conductor cable or single conductors of multiple-conductor cable shall consist of a plasticized, noncontaminating, polyvinyl-chloride compound applied concentrically over the polyethylene. The insulation jacket compound shall meet the test

D. DETAILED REQUIREMENTS, continued
D 4, continued

requirement of section E.2.c. of this specification. The insulation jacket thickness shall meet the requirements of IPCEA Section 4.3.4. as applicable except as noted in section D.5.

*5. Single conductors of multiple-conductor cable with polyethylene insulation and polyvinyl chloride insulation jacket in sizes No. 14 through No. 9 AWG shall conform to IPCEA Section 7.4.3.2.2. The 20-mil polyethylene insulation and the 10-mil polyvinyl chloride insulation jacket shall have minimum thickness of not less than 18 mils and 8 mils respectively and the total thickness shall not be less than 27 mils.

*6. Shielding shall include conductor shielding, insulation shielding, and metallic tape shielding and shall be in conformance with IPCEA Sections 2.4. and 4.1., including test requirements, and the following specific requirements:

*a. Conductor shielding shall be extruded and shall consist of virgin black semiconducting thermoplastic material. The extruded shield shall have an average thickness of not less than 15 mils when measured over top of the strands, and a minimum thickness of not less than 12 mils. The outer surface of the conductor shield shall be cylindrical and shall be firmly bonded to the insulation. A semiconducting tape may be used between the conductor and the extruded semiconducting material if necessary to facilitate separation of shielding from conductor.

*b. Insulation shielding shall be extruded and shall consist of one layer of virgin, black, semiconducting thermoplastic material applied directly over the insulation. The average thickness of the insulation shield shall not be less than 30 mils.

*c. Metallic shielding tape shall be applied over the insulation shield. The metallic tape shall be lapped with minimum lap of 10 percent of tape width.

*7. Assembly of single conductors into multiple-conductor cables shall conform to IPCEA Section 5.2. or 5.3. and the following:

*a. Two-conductor cables may be assembled either parallel or helical

b. Cables with more than two conductors shall be assembled helical.

D. DETAILED REQUIREMENTS, continued
D 7, continued

*c. Thermoplastic nonhygroscopic fillers may be used on parallel assemblies and shall be used on helical assemblies to provide a substantially circular cross section. If necessary, a nonhygroscopic binding tape shall be provided to facilitate the removal of the outer jacket.

*8. An outer jacket of plasticized, noncontaminating, polyvinyl-chloride compound shall be applied concentrically and evenly over the core assembly of multiple-conductor cable or over the shield of shielded cable. The outside surface shall be reasonably smooth and free of irregularities. The jacket material shall meet the test requirements of section E.2.c. of this specification. The outer jacket thickness shall meet the requirements of IPCEA Section 4.3.4. as applicable. The normal calculated core diameter shall govern when manufacturing tolerances create a conflict about the proper diameter to follow. The color of the outer jacket shall be black.

E. INSPECTION AND TESTS

1. Factory Inspection

*a. No material covered by this specification shall be shipped from its point of manufacture before it has been inspected by a TVA Inspector, unless TVA authorizes inspection to be made elsewhere or waived. The TVA Inspector shall have the necessary access and facilities at the Contractor's factory for unrestricted inspection of the work. A schedule of production of TVA's material shall be furnished to the TVA Inspector on his request.

*b. Whether witnessed or not by TVA, the manufacturer shall make the inspection and tests specified herein and shall submit the results in his certified test reports.

c. The release of any material by a TVA Inspector shall in no way relieve the Contractor of any of his responsibility for meeting all of the requirements of this specification, and shall not prevent subsequent rejection if such material is later found to be defective.

2. Factory Tests of Sample Lengths

*a. Unless notified by TVA in writing before factory tests are to begin that a TVA representative will designate the shipping reels or coils from which samples shall be taken, the manufacturer shall take the necessary samples for measuring and testing at random from the shipping reels or coils of each lot of each item of cable he is offering for inspection.

E. INSPECTION AND TESTS, continued
E 2, continued

*b. Test samples for each of the following required tests shall be selected from at least 10 percent of the shipping lengths of each lot of each item offered for inspection. For nominal lengths less than 1800 feet, the number of samples shall be based on 10 percent of the equivalent lengths, as determined by dividing the total footage offered by 1800 feet; however, at least one sample per item shall be selected. Where a fraction results from the use of the 10 percent factor, a fraction less than 0.5 shall be dropped whereas a fraction 0.5 and greater shall require selecting an additional test sample.

*c. Dimensions shall be measured on 6-inch samples at a point free from any distortion or damage. For multiple-conductor cables, the samples shall be taken from the completely assembled cables and all the single conductors measured. Dimensions measured are to be shown in manufacturer's certified test reports and shall include diameter of conductor; thickness of insulation, thickness of extruded and metallic shielding, insulation jacket, and outer jacket; and overall diameter.

(1) Conductors shall be measured with a micrometer caliper reading to 0.001 inch, or equivalent. Acceptance or rejection shall be based on compliance with IPCEA Section 2.5.

*(2) Insulation, insulation jackets, metallic shielding, extruded semiconducting shielding, and outer jackets shall be measured with an Underwriters Laboratories, Incorporated, pin gauge, or equivalent, when fitted with a pin ranging in diameter from 0.025 to 0.020 inch. Acceptance or rejection shall be based on compliance with section D. of this specification.

(3) In case of doubt as to the accuracy of any measurement, the optical method shall be employed utilizing a micrometer microscope that is accurate to at least 0.0005 inch.

*d. Insulation shall be tested in accordance with IPCEA Section 3.9.1. except as follows:

*(1) Cold Bend Test at -35 C ±1 C in accordance with IPCEA Section 6.4.15. is required on all cables rated at 2001 volts and above.

E. INSPECTION AND TESTS, continued
E 2 d, continued

*(2) For multiple-conductor cable samples, select one single conductor for testing when the sample contains 5 conductors or less and an additional single conductor for each 10 conductors or fraction thereof in excess of 5.

*e. Polyvinyl-chloride insulation jackets and outer jackets shall be tested in conformance with IPCEA Section 4.3.1. except as follows:

*(1) No tests other than Unaged Tensile and Elongation in accordance with IPCEA Sections 6.4.11.3. and 6.4.11.4. and (2) below are required on jackets less than 30 mils (0.76 mm) in thickness.

*(2) Physical - IPCEA Section 4.3.1., except the material shall meet the following minimum requirements:

<u>Physical Properties of Unaged Sample</u>

Minimum tensile strength, psi	- 1500
Minimum elongation at rupture, percent	- 250

<u>Physical Properties After Accelerated Aging</u>

After Air Oven Test at 119-121 C for	- 7 days
Tensile strength, minimum percent of unaged strength	- 90
Elongation, minimum percent of unaged sample	- 80

*(3) Flame tests - All completed multiple-conductor cables and single-conductor cables No. 8 AWG and larger shall pass the Vertical Flame Test, IPCEA S-19-81, Section 6.19.6. Both single conductors of multiple-conductor cables and single-conductor cables No. 9 AWG and smaller shall pass the Horizontal Flame Test, IPCEA S-19-81, Section 6.13.2. No flame test is required on nylon-jacketed single-conductor cable or on nylon-jacketed single-conductors of multiple-conductor cables.

*(4) For multiple-conductor cable, the outer jacket and the jacket of only one conductor from the selected sample shall be teste

E. INSPECTION AND TESTS, continued

*3. <u>Factory Tests on Shipping Reel and Coil Lengths</u>

*a. Alternating-current high-voltage tests shall be made on each length of each lot of each item in conformance with IPCEA Section 6.11. and the following requirements:

*(1) A test in water of all completed nonshielded, single-conductor cable shall be made in accordance with IPCEA Sections 6.11.3. and 6.11.4. and Table 3-2, except that for cable rated at 600 volts the minimum AC test voltage shall not be less than 5000 volts for any size conductor.

*(2) On multiple-conductor cable a preliminary test of each single conductor shall be made in accordance with paragraph (1) above. After assembly and final covering, each conductor shall be tested in accordance with IPCEA Section 6.11.4.

*(3) Shielded cable shall be tested in accordance with IPCEA Sections 6.11.3. and 6.11.5. The test voltage shall be applied between conductor and shield and shall be that specified in IPCEA Table 3-2 for cable with ungrounded neutral.

*(4) Shielded cable shall be tested for corona level in accordance with IPCEA Sections 3.9.2.4. and 6.13. The minimum corona extinction level shall be that required for cable with ungrounded neutral.

*b. Insulation resistance tests shall be made on each shipping reel or coil length in conformity with IPCEA Section 6.12. Any cable length which fails to meet the specified insulation resistance shall be rejected.

*c. Direct-current high-voltage tests shall be made on each shipping length of each lot of each item rated at 2001 volts and higher in conformance with IPCEA Sections 3.9.2.3. and 6.11.2. The direct-current test voltage shall be applied between the conductor and the shield. The DC test voltage shall be 3.0 times the AC test voltage as required by IPCEA Table 3-2 for cable with ungrounded neutral.

E. INSPECTION AND TESTS, continued
E 3, continued

*d. Conductor resistance tests shall be made on 10 percent of the shipping lengths of each lot of each item ready at the time of inspection and the results expressed on ohms per 1000 feet at either 20 or 25 C. If the resistance of any length tested exceeds the specified resistance allowed by IPCEA Section 2.6., that length may be retested. If the resistance is still found to be in excess, the length shall be rejected; each length of that lot shall be tested, and any length found to be in excess shall also be rejected.

*e. Shield resistance shall be measured on each shipping length of each lot of each item of shielded cable and expressed in ohms per 1000 feet.

4. Factory Test Reports

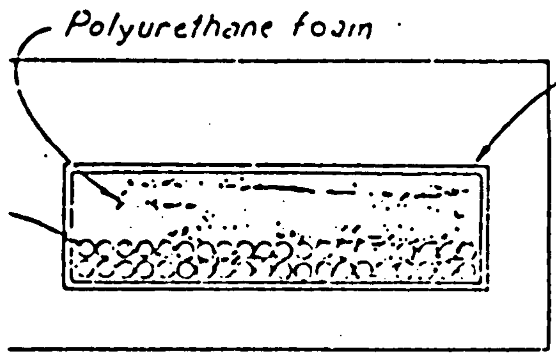
*a. The Contractor shall furnish seven certified copies of each factory test report to TVA promptly after conclusion of tests. Each test report shall be identified by TVA contract number, item number(s), mark number(s), shipping reel(s) number(s), date of test, TVA cable type and voltage rating, and the number and size of conductors. The report shall include the results of all the tests and measurements specified herein.

F. PACKING AND MARKING

*1. The material shall be packed, sealed, and shipped in accordance with ANSI MH5.2.2-1972, Level D classification. Wire or cable shall be shipped on reels or in coils in continuous lengths as called for in the Schedule of Prices. The reels, if wooden, shall be painted or treated in such a way as to resist damage from prolonged exposure to an outdoor environment.

*2. Reels and coils shall be plainly marked with the contract number, item number, reel number, length shipped on reel or coil, number and size of conductors, TVA insulation designation and voltage rating, year of manufacture, and Name of Manufacturer. All labels and tags shall be waterproofed and fastened securely to the reel or coil.

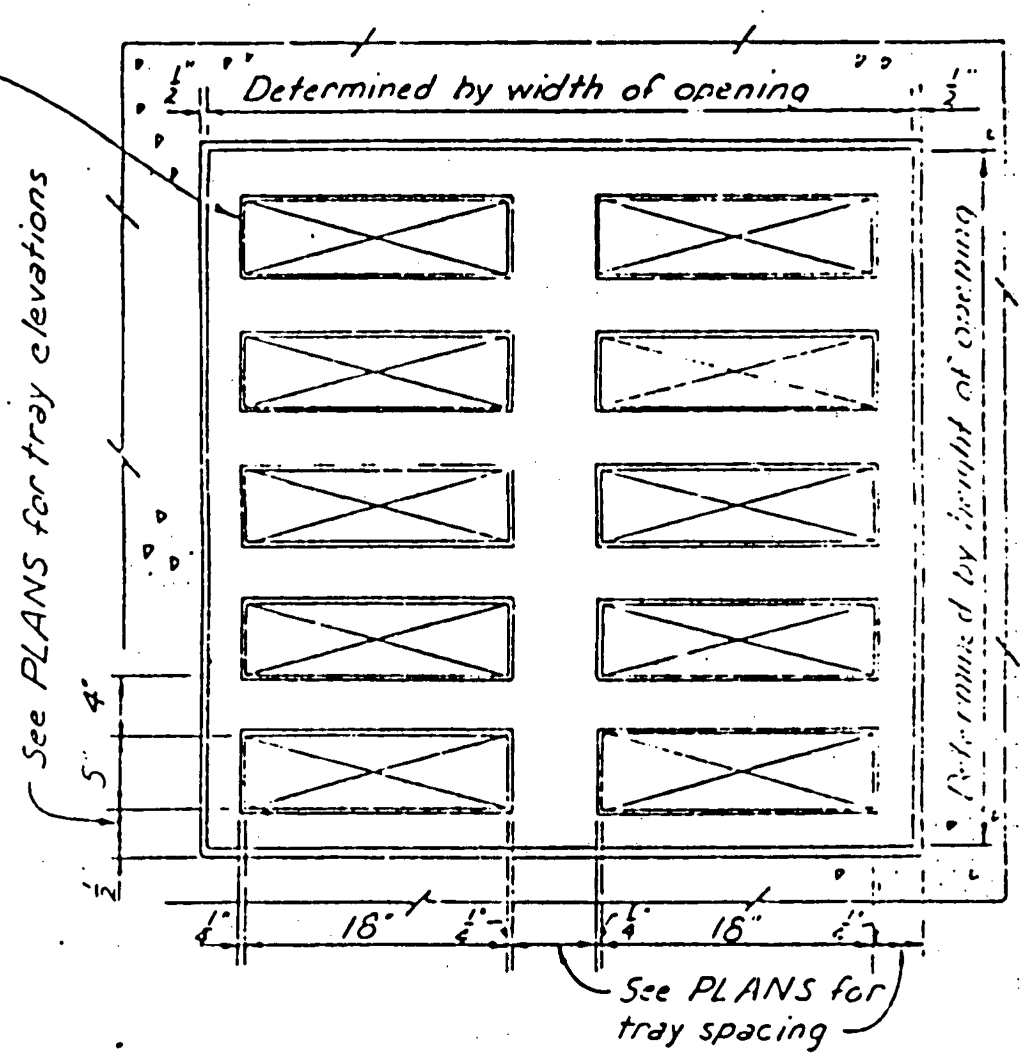
*Revisions since 9-15-70.



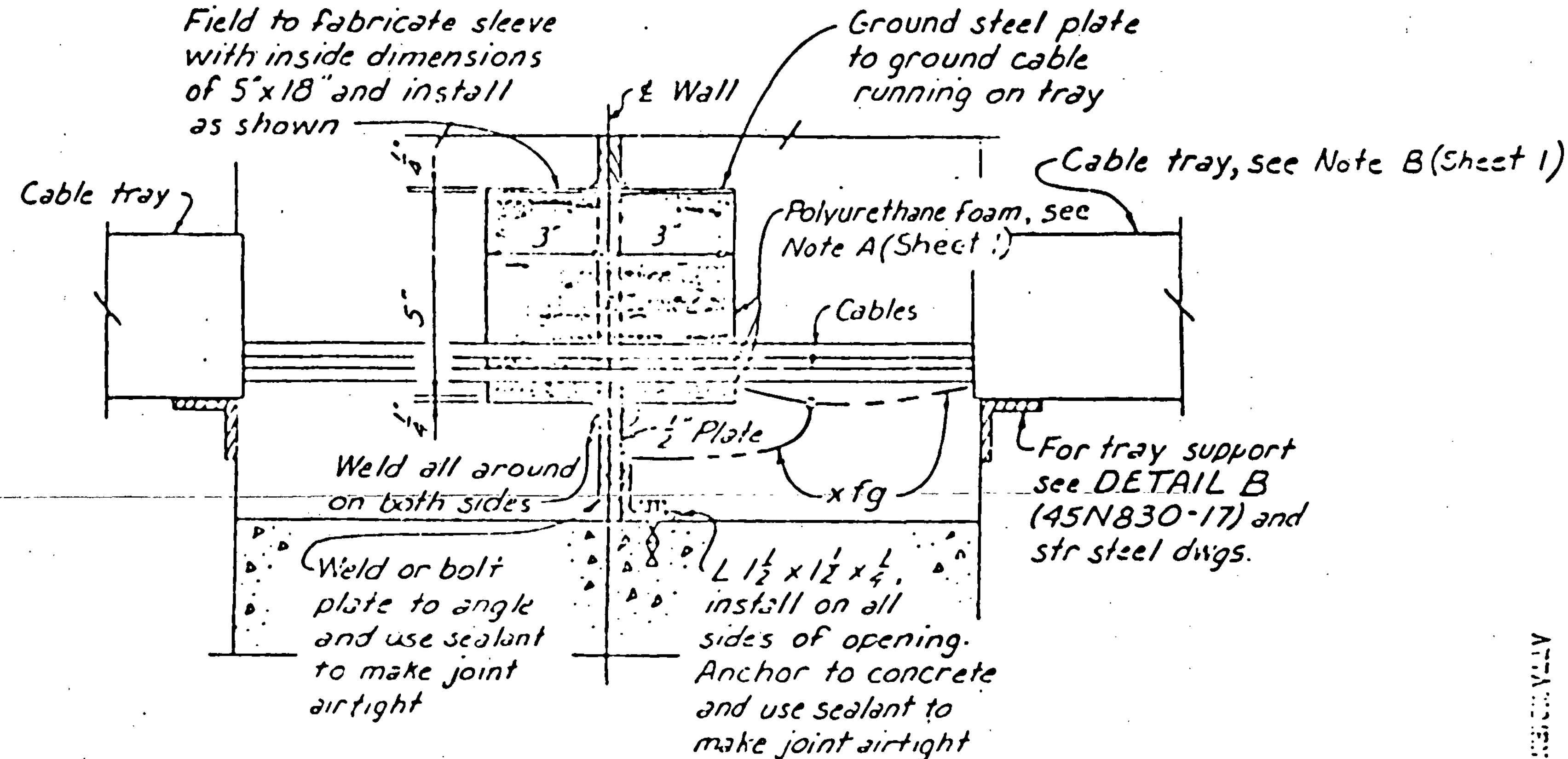
NOTES:

4. When all cables have been installed through cable sleeve, seal the remaining opening and voids with 6" ± of Urethane foam or equal before applying flame proofing compound, Flamastic 71A or equal.

5. Apply approximately 1/8" to 1/4" of flame proofing compound on the steel sleeve and on both top and bottom of the tray and cables for 12" on both sides of the barrier.



FRONT ELEVATION
Scale: 1"=1'-0"



SIDE VIEW
Scale: 3"=1'0"

ATTACHMENT 11

21.000
TENTATIVE REVISION
JANUARY 24, 1975

TENNESSEE VALLEY AUTHORITY

STANDARD SPECIFICATION

FOR

RIGID ALUMINUM CONDUIT

A. SCOPE

*1. This specification covers rigid aluminum conduit and accessories such as couplings, elbows, and fittings for use as metal raceways for wires and cables.

B. APPLICABLE SPECIFICATIONS

*1. The current revision to the following specifications of the issue in effect on date of the invitation to bid shall form a part of this specification:

Underwriters Laboratories, Incorporated,
Standard UL6 - Rigid Metallic Conduit

ANSI C80.5 - Rigid Aluminum Conduit

*ANSI B2.1 - Standard Pipe Threads

C. GENERAL REQUIREMENTS

*1. Size, quantity, special Quality Assurance Procedures if required, and any other specific requirements or exceptions to this specification are stated in the Schedule of Prices.

D. DETAILED REQUIREMENTS

*1. The conduit and accessory items shall be manufactured from 6063 aluminum alloy, T-1 temper with maximum copper content not exceeding four-tenths of 1 percent. Conduit shall be furnished in nominal 10-foot lengths. Each length shall be threaded on both ends with American National Standards Institute tapered pipe thread, ANSI B2.1. One end of each length of conduit shall be fitted with a coupling and the end opposite the coupling shall be fitted with a plastic color-coded thread protector.

*2. Elbows shall be furnished in 6063 aluminum alloy, T-1 temper, and shall be threaded on both ends with American National Standards Institute tapered pipe thread, ANSI B2.1.

*Revisions since 7-18-62.

D. DETAILED REQUIREMENTS, continued

*3. Couplings shall be furnished in 6063 aluminum alloy, T-1 temper, and shall be internally threaded with American National Standards Institute straight pipe thread, ANSI B2.1.

*4. Color-coded thread protectors furnished with aluminum conduit shall be color-coded as follows:

Red - Quarter-inch sizes
Black - Half-inch sizes
Blue - Inch sizes

*5. Specification Compliance

*a. Conduit, elbows, and couplings furnished under this specification shall conform to the requirements of Underwriters Laboratories, Incorporated, Standard UL6; and American National Standards Institute C80.5.

*b. In addition to the manufacturer's name or trademark, as required by Underwriters Laboratories, Incorporated, Standard UL6, each length of conduit and elbow shall have the Underwriters Laboratories, Incorporated, label or an acceptable equivalent, firmly attached.

*6. Conformance labels shall be attached to the conduit with pressure-sensitive adhesive or similar means to provide easy removal without scarring the conduit, or shall be attached to the thread protectors. Stenciling or permanent imprint in the metal, with natural finish, will be accepted.

E. INSPECTION AND TESTS

1. Factory Inspection

*a. No material covered by this specification shall be shipped from its point of manufacture before it has been inspected by a TVA Inspector, unless TVA authorizes inspection to be made elsewhere or waived. The TVA Inspector shall have the necessary access and facilities at the Contractor's factory for unrestricted inspection of the work. A schedule of production of TVA's material shall be furnished to the TVA Inspector on his request.

*Revisions since 7-18-62.

E. INSPECTION AND TESTS, continued
E 1, continued

*b. Whether witnessed or not by TVA, the manufacturer shall make the inspection and tests specified herein and shall submit the results in his certified test reports.

*c. The release of any material by a TVA Inspector shall in no way relieve the Contractor of any of his responsibility for meeting all of the requirements of this specification, and shall not prevent subsequent rejection if such material is later found to be defective.

2. Factory Tests of Sample Lengths

*a. Unless notified by TVA in writing before factory tests are to begin that a TVA representative will designate the shipping bundles or packages from which samples shall be taken, the manufacturer shall take the necessary sample(s) for measuring and testing at random from the shipping bundles or packages of each item of conduit, elbow, or coupling. At least one sample per item shall be taken.

*b. Conduit shall be tested for ductility at a temperature of 72 F ±4 F in accordance with ANSI C80.5, Part 5.

*c. Dimensions of conduit (inside diameter, outside diameter, and wall thickness) shall be measured with a micrometer caliper reading 0.001 inch or equivalent and shall conform to ANSI C80.5, Table 2.

*d. Conduit shall be measured for length, without coupling, and for weight of ten units with one coupling attached to each length and shall conform to ANSI C80.5, Table 2.

*e. Couplings shall be measured for outside diameter and minimum length and shall conform to ANSI C80.5, Table 3.

*f. Elbows shall be measured for radius to center of conduit and length of straight section at each end and shall conform to ANSI C80.5, Table 4.

3. Factory Test Reports

*a. The Contractor shall furnish five certified copies of each factory test report to TVA promptly after conclusion of tests. Each test report shall be identified by TVA contract number, item number, mark number, and date of test. The report shall include the results of all the tests and measurements specified herein.

*Revisions since 7-10-62.

E. INSPECTION AND TESTS, continued

4. TVA, at its discretion, may waive any or all of the provisions for inspection, tests, and test reports, either in the invitation for bids, at time of award of contract, or at any time thereafter.

F. PACKING AND MARKING

1. The conduit and accessories shall be packaged, or bundled, and wrapped to prevent damage and marked in accordance with standard commercial practice so as to ensure acceptance by common carrier for safe transportation at the lowest rate to the point of delivery.

*2. Shipping bundles or packages shall be plainly marked with the contract number, item number, name of material, number and size of items contained therein, and the name of the Contractor. All labels and tags shall be waterproofed and fastened securely to the bundles or packages.

*Revisions since 7-18-62.

ATTACHMENT 12

TENNESSEE VALLEY AUTHORITY

STANDARD SPECIFICATION

21.0001
TENTATIVE REVISION
MAY 23, 1973

FOR

RIGID STEEL CONDUIT (ZINC COATED)

A. SCOPE

*1. This specification covers rigid steel conduit (zinc coated) and accessories such as couplings, elbows, and fittings for use as metal raceways for wires and cables.

B. APPLICABLE SPECIFICATIONS

*1. The current revision to the following specifications of the issue in effect on date of the invitation to bid shall form a part of this specification:

Underwriters Laboratories, Incorporated, Standard - UL6 - Rigid Metal Conduit

ANSI C80.1 - Rigid Steel Conduit (Zinc Coated)
*ANSI B2.1 - Standard Pipe Threads

C. GENERAL REQUIREMENTS

*1. Size, quantity, special Quality Assurance Procedures if required, and any other specific requirements or exceptions to this specificat! are stated in the Schedule of Prices.

D. DETAILED REQUIREMENTS

*1. The conduit and accessory items shall be manufactured from mild steel tube with a zinc coating having a minimum thickness of 0.0008 inch on both inside and outside surfaces. Conduit shall be furnished in nominal 10-foot lengths. Each length shall be threaded on both ends with American National Standards Institute tapered pipe thread, ANSI B2.1., and shall have a uniform coating of zinc applied to the threads. One end of each length of conduit shall be fitted with a coupling and the end opposite the coupling shall be fitted with a plastic color-coded thread protector.

*2. Elbows shall be furnished in mild steel tube with a zinc coating on both inside and outside surfaces and shall be threaded on both ends with American National Standards Institute tapered pipe thread, ANSI B2.1. A UL listed rust preventive coating shall be applied to the threads.

*Revisions since 5-7-76.

D. DETAILED REQUIREMENTS, continued

*3. Couplings shall be furnished in mild steel tube with zinc coating and shall be internally threaded with American National Standards Institute straight-pipe thread, ANSI .2.1. A uniform zinc coating shall be applied to the internal threads.

*4. Color-coded thread protectors furnished with rigid steel conduit shall be color coded as follows:

Red - Quarter-inch sizes
Black - Half-inch sizes
Blue - Inch sizes

*5. Specification Compliance

*a. Conduit, elbows, and couplings furnished under this specification shall conform to the requirements of Underwriters Laboratories, Incorporated, Standard UL6, and American National Standards Institute C80.1.

*b. In addition to the manufacturer's name or trademark, as required by Underwriters Laboratories, Incorporated, Standard UL6, each length of conduit and elbow shall have the Underwriters Laboratories, Incorporated, label or an acceptable equivalent, firmly attached.

*6. Conformance labels shall be attached to the conduit with pressure-sensitive adhesive or similar means to provide easy removal without scarring the conduit, or shall be attached to the thread protectors. Stenciling or permanent imprint in the metal, with natural finish, will be accepted.

E. INSPECTION AND TESTS

1. Factory Inspection

*a. No material covered by this specification shall be shipped from its point of manufacture before it has been inspected by a TVA Inspector, unless TVA authorizes inspection to be made elsewhere or waived. The TVA Inspector shall have the necessary access and facilities at the Contractor's factory for unrestricted inspection of the work. A schedule of production of TVA's material shall be furnished to the TVA Inspector on his request.

E. INSPECTION AND TESTS, continued
E 1, continued

*b. Whether witnessed or not by TVA, the manufacturer shall make the inspection and tests specified herein and shall submit the results in his certified test reports.

*c. The release of any material by a TVA Inspector shall in no way relieve the Contractor of any of his responsibility for meeting all of the requirements of this specification, and shall not prevent subsequent rejection if such material is later found to be defective.

2. Factory Tests of Sample Lengths

*a. Unless notified by TVA in writing before factory tests are to begin that a TVA representative will designate the shipping bundles or packages from which samples shall be taken, the manufacturer shall take the necessary samples(s) for measuring and testing at random from the shipping bundles or packages of each item of conduit, elbow, or coupling. At least one sample per item shall be taken.

*b. Conduit shall be tested for ductility at a temperature of 72 F ±4 F in accordance with ANSI C80.1.

*c. Conduit and accessory items shall be tested for thickness of zinc coating in accordance with ANSI C80.1, Paragraph 5.2, Method 3 (Preece test). The material shall withstand four 1-minute immersions.

*d. Dimensions of conduit (inside diameter, outside diameter, and wall thickness) shall be measured with a micrometer caliper reading 0.001 inch or equivalent and shall conform to ANSI C80.1, Table 2.

*e. Conduit shall be measured for length, without coupling, and for weight of 10 units with one coupling attached to each length and shall conform to ANSI C80.1, Table 2.

*f. Couplings shall be measured for outside diameter and minimum length and shall conform to ANSI C80.1, Table 3.

*g. Elbows shall be measured for radius to center of conduit and length of straight section at each end and shall conform to ANSI C80.1, Table 4.

*Revisions since 5-7-56.

E. INSPECTION AND TESTS, continued

3. Factory Test Reports

*a. The Contractor shall furnish five certified copies of each factory test report to TVA promptly after conclusion of tests. Each test report shall be identified by TVA contract number, item number, mark number, and date of test. The report shall include the results of all the tests and measurements specified herein.

*4. TVA, at its discretion, may waive any or all of the provisions for inspection, tests, and test reports, either in the invitation for bids, at time of award of contract or at any time thereafter.

F. PACKING AND MARKING

1. The conduit and accessories shall be packaged, or bundled, and wrapped to prevent damage and marked in accordance with standard commercial practice so as to ensure acceptance by common carrier for safe transportation at the lowest rate to the point of delivery.

*2. Shipping bundles or packages shall be plainly marked with the contract number, item number, name of material, number and size of items contained therein, and the name of the Contractor. All labels and tags shall be waterproofed and fastened securely to the bundles or packages.

ATTACHMENT 14

25.016
TENTATIVE REVISION
MARCH 16, 1973

TENNESSEE VALLEY AUTHORITY

STANDARD SPECIFICATION

FOR

CROSS-LINKED POLYETHYLENE-INSULATED WIRE AND CABLE

A. SCOPE

*1. This specification covers single- and multiple-conductor wire and cable insulated with cross-linked polyethylene, for installation in wet or dry locations with conductor temperatures not exceeding 90 C and for use in auxiliary power, control, and lighting circuits at voltages ranging from 600 to 15,000 volts, 60 Hz AC, between conductors.

*2. All cable will be installed in conduits, in underground ducts, or on cable trays.

B. APPLICABLE SPECIFICATIONS

*1. The current revision to the following specifications of the issue in effect on date of the invitation to bid shall form a part of this specification:

*a. IPCEA-NEMA Standard S-66-524--NEMA WC7, IPCEA-NEMA Standards Publication, Cross-Linked-Thermosetting-Polyethylene-Insulated Wire and Cable. Specific references herein are from the First Edition dated May 1971.

*b. IPCEA-NEMA Standard S-19-81--NEMA WC3, IPCEA-NEMA Standards Publication, Rubber-Insulated Wire and Cable. Specific references herein are from the Fifth Edition dated July 1969.

*c. All references herein are to IPCEA S-66-524 unless S-19-81 is mentioned specifically.

C. GENERAL REQUIREMENTS

*1. TVA type designation, voltage rating, conductor size, stranding if other than class B, special Quality Assurance Procedures if required, and any other specific requirements or exceptions to this specification are stated in the Schedule of Prices.

2. Type designations are TVA designations and are described as follows:

*a. Type CP, single-conductor wire or cable, with cross-linked polyethylene insulation.

C. GENERAL REQUIREMENTS, continued
C 2, continued

*b. Type CPJ, single-conductor wire or cable, with cross-linked polyethylene insulation and a polyvinyl-chloride insulation jacket over the cross-linked polyethylene.

c. Type CPJJ, multiple-conductor cable with a core of the specified number of type CPJ single conductors covered by a polyvinyl-chloride outer jacket.

d. Type CPNJ, multiple-conductor cable with a core of the specified number of type CP single conductors covered by a polyvinyl-chloride outer jacket.

*e. Type CPSJ, single-conductor high-voltage cable with extruded strand and insulation shielding and with metallic electrostatic shielding.

*3. All outer covering of single-conductor wire or cable No. 8 AWG and larger and the outer jacket of all multiple-conductor cable shall have legible indent printing or legible permanent surface marking to show the following identification on approximately each foot of cable:

a. Manufacturer.

b. Year of manufacture.

*c. TVA type designation (see section C.2. above) and voltage rating.

d. Size and number of conductors.

*4. Sequential footage marking on single-conductor cable, No. 8 AWG and larger, and on all multiconductor cable shall be supplied. The numerical markings shall be legible indent printing or legible permanent surface marking on the outside surface of the cable at 2-foot intervals and consist of 5-digit numbers increasing sequentially in increments of 2 feet (00020, 00022, 00024, etc.). The sequential markings may begin and end at any two numbers within the range of 00000 to 99998. Sequential footage markings shall be to a tolerance of ±2 percent.

*5. If indent printing is used for section C.3. or C.4. above, the depth of the indentations shall not be greater than 0.007 inch.

*6. At the option of the manufacturer, on 15-kv class cable a marker tape under the outer jacket containing information required in sections C.3. and C.4. may be substituted.

C. GENERAL REQUIREMENTS, continued

7. Each bidder shall submit with each copy of his bid complete information regarding the cable he proposes to furnish, including the following:

*a. Type designation and trade name of insulation, insulation jacket, and outer jacket materials.

b. Makeup of multiple-conductor cables.

*c. Performance records of representative installations of the same type and construction of wire and cable offered. If performance records have been submitted in a previous bid or are on file with TVA, an identifying reference shall be acceptable in lieu of a repetition of the data.

*8. Color coding of multiple-conductor cable shall be furnished in accordance with IPCEA S-19-81, Section 5.8.3.1.1., Method 1, Table 5-2. All single-conductor wire or cable shall be black.

D. DETAILED REQUIREMENTS

1. Conductors shall be made of soft or annealed copper in conformity with IPCEA Part 2. All conductor sizes shall have concentric class B stranding and may be coated or uncoated.

*2. The insulation over each conductor shall consist of extruded heat-stabilized cross-linked polyethylene, except that insulation not having a jacket shall also be light-stabilized, applied concentrically about the conductor and shall meet the test requirements of section E.2.d. In-plant repairs of the insulation on shielded cable are prohibited unless specifically agreed to by TVA. All nominal cross-linked polyethylene thickness shall conform to IPCEA Table 3-1 as noted below:

*a. Type CP single-conductor cable and individual conductors of type CPJ multiple-conductor cable shall conform to Column A.

*b. Type CPJ single-conductor cable and individual conductors of type CPJJ multiple-conductor cable shall conform to Column B.

*c. Type CPSJ single-conductor shielded cable shall conform to the column marked Ungrounded Neutral (Shielded).

*3. The insulation jacket covering the insulation on single-conductor cable or single conductors of multiple-conductor cable shall consist of a plasticized, noncontaminating, polyvinyl-chloride compound

D. DETAILED REQUIREMENTS, continued
D 3, continued

applied concentrically over the cross-linked polyethylene. The insulation jacket compound shall meet the test requirements of section E.2.e. of this specification. The insulation jacket thickness shall meet the requirements of IPCEA Section 4.3.3. as applicable.

*4. Shielding shall include conductor shielding, insulation shielding, and metallic tape shielding and shall be in conformance with IPCEA Sections 2.4. and 4.1., including test requirements, and the following specific requirements:

*a. Conductor shielding shall be extruded and shall consist of virgin black semiconducting thermosetting material. The extruded shield shall have an average thickness of not less than 15 mils when measured over top of the strands, and a minimum thickness of not less than 12 mils. The outer surface of the conductor shield shall be cylindrical and shall be firmly bonded to the insulation. A semiconducting tape may be used between the conductor and the extruded semiconducting material if necessary to facilitate separation of shielding from conductor.

*b. Insulation shielding shall be extruded and shall consist of one layer of virgin, black, semiconducting thermosetting material applied directly over the insulation. The average thickness of the insulation shield shall not be less than 30 mils.

*c. Metallic shielding tape shall be applied over the insulation shield. The metallic tape shall be lapped with minimum lap of 10 percent of tape width.

*5. Assembly of single conductors into multiple-conductor cables shall conform to IPCEA Section 5.2. or 5.3. and the following:

*a. Two-conductor cables may be assembled either parallel or helical.

b. Cables with more than two conductors shall be assembled helical.

*c. Thermoplastic nonhygroscopic fillers may be used on parallel assemblies and shall be used on helical assemblies to provide a substantially circular cross section. If necessary, a nonhygroscopic binding tape shall be provided to facilitate the removal of the outer jacket.

D. DETAILED REQUIREMENTS, continued

*6. An outer jacket of plasticized, noncontaminating, polyvinyl-chloride compound shall be applied concentrically and evenly over the core assembly of multiple-conductor cable or over the shield of shielded cable. The outside surface shall be reasonably smooth and free of irregularities. The jacket material shall meet the test requirements of section E.2.e. of this specification. The outer jacket thickness shall meet the requirements of IPCEA Section 4.3.3. as applicable. The normal calculated core diameter shall govern when manufacturing tolerances create a conflict about the proper diameter to follow. The color of the outer jacket shall be black.

E. INSPECTION AND TESTS

1. Factory Inspection

*a. No material covered by this specification shall be shipped from its point of manufacture before it has been inspected by a TVA Inspector, unless TVA authorizes inspection to be made elsewhere or waived. The TVA Inspector shall have the necessary access and facilities at the Contractor's factory for unrestricted inspection of the work. A schedule of production of TVA's material shall be furnished to the TVA Inspector on his request.

*b. Whether witnessed or not by TVA, the manufacturer shall make the inspection and tests specified herein and shall submit the results in his certified test reports.

c. The release of any material by a TVA Inspector shall in no way relieve the Contractor of any of his responsibility for meeting all of the requirements of this specification, and shall not prevent subsequent rejection if such material is later found to be defective.

2. Factory Tests of Sample Lengths

*a. Unless notified by TVA in writing before factory tests are to begin that a TVA representative will designate the shipping reels or coils from which samples shall be taken, the manufacturer shall take the necessary samples for measuring and testing at random from the shipping reels or coils of each lot of each item of cable he is offering for inspection.

*b. Test samples for each of the following required tests shall be selected from at least 10 percent of the shipping lengths of each lot of each item offered for inspection. For nominal lengths less than 1800 feet, the number of samples shall be based on 10 percent of the equivalent lengths, as determined by dividing the total footage offered by 1800 feet; however, at

*Revisions since 12-10-70

E. INSPECTION AND TESTS, continued
E 2 b, continued

least one sample per item shall be selected. Where a fraction results from the use of the 10 percent factor, a fraction less than 0.5 shall be dropped whereas a fraction 0.5 and greater shall require selecting an additional test sample.

*c. Dimensions shall be measured on 6-inch samples at a point free from any distortion or damage. For multiple-conductor cables, the samples shall be taken from the completely assembled cables and all the single conductors measured. Dimensions measured are to be shown in manufacturer's certified test reports and shall include diameter of conductor; thickness of insulation, thickness of extruded and metallic shielding, insulation jacket, and outer jacket; and overall diameter.

(1) Conductors shall be measured with a micrometer caliper reading to 0.001 inch, or equivalent. Acceptance or rejection shall be based on compliance with IPCEA Section 2.5.

*(2) Insulation, insulation jackets, metallic shielding, extruded semiconducting shielding, and outer jackets shall be measured with an Underwriters Laboratories, Incorporated, pin gauge, or equivalent, when fitted with a pin ranging in diameter from 0.025 to 0.020 inch. Acceptance or rejection shall be based on compliance with section D. of this specification.

(3) In case of doubt as to the accuracy of any measurement, the optical method shall be employed utilizing a micrometer microscope that is accurate to at least 0.0005 inch.

*d. Insulation shall be tested in accordance with IPCEA Sections 3.6. and 3.7.1. except as follows:

*(1) Cold Bend Test at -35 C ±1 C in accordance with IPCEA Section 6.10.3 is required on all cables rated at 2001 volts and above.

*(2) For multiple-conductor cable samples, select one single conductor for testing when the sample contains 5 conductors or less and an additional single conductor for each 10 conductors or fraction thereof in excess of 5.

E. INSPECTION AND TESTS, continued
E 2, continued

*e. Polyvinyl-chloride insulation jackets and outer jackets shall be tested in conformance with IPCEA Section 4.3.1. except as follows:

*(1) No tests other than Unaged Tensile and Elongation in accordance with IPCEA Sections 6.4.11.3. and 6.4.11.5. and (2) below are required on jackets less than 30 mils (0.76 mm) in thickness.

*(2) Physical - IPCEA Section 4.3.1., except the material shall meet the following minimum requirements:

<u>Physical Properties of Unaged Sample</u>

Minimum tensile strength, psi - 1500

Minimum elongation at rupture, percent - 250

<u>Physical Properties After Accelerated Aging</u>

After Air Oven Test at 119-121 C for - 7 days

Tensile strength, minimum percent of unaged strength - 90

Elongation, minimum percent of unaged sample - 80

*(3) Flame tests - All completed multiple-conductor cables and single-conductor cables No. 8 AWG and larger shall pass the Vertical Flame Test, IPCEA S-19-81, Section 6.19.6. Both single conductors of multiple-conductor cables and single-conductor cables No. 9 AWG and smaller shall pass the Horizontal Flame Test, IPCEA S-19-81, Section 6.13.2.

*(4) For multiple-conductor cable, the outer jacket and the jacke of only one conductor from the selected sample shall be test

*3. <u>Factory Tests on Shipping Reel and Coil Lengths</u>

*a. Alternating-current high-voltage tests shall be made on each length of each lot of each item in conformance with IPCEA Section 6.14. and the following requirements:

*(1) A test in water of all completed nonshielded, single-conductor cable shall be made in accordance with IPCEA Sections 6.14.3. and 6.14.4. and Table 3-1, except that for cable rated at 600 volts the minimum AC test voltage shall not be less than 5000 volts for any size conductor.

E. INSPECTION AND TESTS, continued
E 3 a, continued

*(2) On multiple-conductor cable a preliminary test of each single conductor shall be made in accordance with paragraph (1) above. After assembly and final covering, each conductor shall be tested in accordance with IPCEA Section 6.14.4.

*(3) Shielded cable shall be tested in accordance with IPCEA Sections 6.14.3. and 6.14.5. The test voltage shall be applied between conductor and shield and shall be that specified in IPCEA Table 3-1 for cable with ungrounded neutral.

*(4) Shielded cable shall be tested for corona level in accordance with IPCEA Sections 3.7.2.4. and 6.16. The minimum corona extinction level shall be that required for cable with ungrounded neutral.

*b. Insulation resistance tests shall be made on each shipping reel or coil length in conformity with IPCEA Section 6.15. Any cable length which fails to meet the specified insulation resistance shall be rejected.

*c. Direct-current high-voltage tests shall be made on each shipping length of each lot of each item rated at 2001 volts and higher in conformance with IPCEA Sections 3.7.2.3. and 6.14.2. The direct-current test voltage shall be applied between the conductor and the shield. The DC test voltage shall be 3.0 times the AC test voltage as required by IPCEA Table 3-1 for cable with ungrounded neutral.

*d. Conductor resistance tests shall be made on 10 percent of the shipping lengths of each lot of each item ready at the time of inspection and the results expressed in ohms per 1000 feet at either 20 or 25 C. If the resistance of any length tested exceeds the specified resistance allowed by IPCEA Section 2.6., that length may be retested. If the resistance is still found to be in excess, the length shall be rejected; each length of that lot shall be tested, and any length found to be in excess shall also be rejected.

*e. Shield resistance shall be measured on each shipping length of each lot of each item of shielded cable and expressed in ohms per 1000 feet.

E. INSPECTION AND TESTS, continued

4. Factory Test Reports

*a. The Contractor shall furnish five certified copies of each factory test report to TVA promptly after conclusion of tests. Each test report shall be identified by TVA contract number, item number(s), mark number(s), shipping reel(s) number(s), date of test, TVA cable type and voltage rating, and the number and size of conductors. The report shall include the results of all the tests and measurements specified herein.

F. PACKING AND MARKING

1. The material shall be packed, sealed, and shipped in accordance with recognized standard practice. Wire or cable shall be shipped on reels or in coils in continuous lengths as called for in the Schedule of Prices.

*2. Reels and coils shall be plainly marked with the contract number, item number, rec number, length shipped on reel or c 'l, number and size of conductors, TVA insulation designation and voltage rating, year of manufacture, and Name of Manufacturer. All labels and tags shall be waterproofed and fastened securely to the reel or coil.

Table I

4 KV Shutdown Boards and Loads

4 KV Shutdown Board A Loads

1. Core Spray (CS) Pump Motors 1A and 2A
2. Residual Heat Removal (RHR) Pump Motors 1A and 2A
3. RHR Service Water Pumps Motors A2 and A3
4. Raw Cooling Water Pump 1D
5. Control Rod Drive (CRD) Pump 1B
6. 480 Volt Shutdown Board 1A Transformer (Normal)
7. 480 Volt Diesel Auxiliary Board A and B Transformer
8. Fire Pump A

4 KV Shutdown Board B Loads

1. CS Pump Motors 1C and 2C
2. RHR Pump Motors 1C and 2C
3. RHR Service Water Pumps Motors C3 and C2
4. 480 Volt Shutdown Board 1A and 1B Transformer (Backup)
5. 480 Volt Diesel Auxiliary Board B Transformer
6. 480 Volt Shutdown Board 2A Transformer (Normal)
7. Fire Pump B

4 KV Shutdown Board C Loads

1. CS Pump Motors 1B and 2B
2. RHR Pump Motors 1B and 2B
3. RHR Service Water Pump Motors B1 and B2
4. 480 Volt Shutdown Board 2A and 2B Transformer (Backup)
5. 480 Volt Shutdown Board 1B Transformer (Normal)
6. Fire Pump C

4 KV Shutdown Board D Loads

1. CS Pump Motors 1D and 2D
2. RHR Pump Motors 1D and 2D
3. RHR Service Water Pump Motors D1 and D2
4. 480 Volt Diesel Auxiliary Board A Transformer
5. 480 Volt Shutdown Board 2B Transformer (Normal)

Table II

480 Volt Shutdown Boards Loads

480 Volt Shutdown Board 1A Loads

1. Unit Preferred M-G Set 1
2. Instrumentation and Control (I&C) Bus A Transfo
3. Control Bay Vent Board A (Emergency)
4. 480 Volt Motor Operated Valve (MOV) Board 1A (Normal)
5. 480 Volt MOV Board 1B (Emergency)
6. 480 Volt MOV Board 1C (Auto throwover) (Emergency)
7. 250 Volt Battery Charger
8. Control and Service Air Compresser 1A
9. Control Bay Water Chiller A
10. Closed Cooling Water (CCW) Pump 1A
11. Reactor Water Cleanup (RWCU) Recirculating Pump 1A
12. Recirculating Pumps M-G Set Oil Pumps
13. Standby Liquid Control (SLC) Pump 1A
14. Fuel Pool Coo[illegible] (FPC) Pump 1A
15. Drywell Blowers 1A-1 and 1B-1

480 Volt Shutdown Board 1B Loads

1. I&C Bus B Transformer
2. Unit Preferred Transformer TUP 1
3. 480 Volt Condensate Demineralizer Board (Emergency)
4. 480 Volt ReactorMOV Board 1A (Emergency)
5. 480 Volt Reactor MOV Boards 1B and 1C (Normal)
6. Turning Gear Oil Pump

7. Control Bay Water Chiller B
8. CCW Pumps 1B and 1C
9. RWCU Recirculating Pump 1B
10. Recirculating Pump M-G Set Oil Pumps
11. SLC Pump 1B
12. FPC Pump 1B
13. Drywell Blowers 1A-2 and 1B-2

480 Volt Shutdown Board 2A Loads

1. Unit Preferred M-G set 2
2. I&C Bus 2A Transformer
3. 480 Volt Diesel Auxiliary Board 3EA (Normal
4. Control Bay Vent Board B (Emergency)
5. 480 Volt Reactor MOV Board 2A (Normal)
6. 480 Volt Reactor MOV Boards 2B and Board 2C (Emergency)
7. 250 Volt Battery Charger 2A
8. Control and Service Air Compressor D
9. CCW Pump 2A
10. RWCU Pump 2A
11. Recirculating Pump M-G Set Oil Pumps
12. SLC Pump 2A
13. FPC Pump 2A
14. Drywell Blowers 2A-1 and 2B-1

480 Volt Shutdown Board 2B Loads

1. I&C Bus B Transformer
2. Unit Preferred Transformer
3. 480 Volt Condensate Demineralizer Board 2 (Emergency)

4. 480 Volt Reactor MOV Board 2A (Emergency)
5. 480 Volt Reactor MOV Board 2B and 2C (Normal)
6. 250 Volt Battery Charter 2B
7. Turning Gear Oil Pump
8. Raw Cooling Water Booster Pump 2A
9. CCW Pump 2B
10. RWCU Recirculating Pump 2B
11. Recirculating Pump M-G set oil pumps 2A-2 and 2B-2
12. SLC Pump 2B
13. FPC Pump 2B
14. Drywell Blowers 2A-2 and 2B-2

Table III

Reactor MOV Board Loads

480 Volt Reactor MOV Board 1A

1. Turbine Building Emergency Lighting Transformer
2. Reactor Building Emergency Lighting Transformer
3. "A" and "C" CS Systems Valves
4. "A" and "C" RHR Systems Valves
5. Drywell Control Air System 1A
6. Reactor Protection System (RPS) M-G set 1A
7. RWCU Isolation Valve FCV 69-1
8. High Pressure Coolant Injection (HPCI) Steam Isolation Valve 73-2
9. SLC Trace Heating (Normal)
10. Drywell Equipment and Floor Drain Sump Pumps 1A
11. Drywell Blowers 1A-3 and 1B-3
12. Turbine Turning Gear Motor
13. Exhaust Fans 1A Shutdown Board Room

480 Volt Reactor MOV Board 2A

1. Loads for Unit 2 that are comparable to those above.

480 Volt Reactor MOV Board 1B

1. Annunciator Battery Chargers
2. Reactor Building Emergency Lighting Transformer
3. RPS M-G Set 1B
4. CS System Valves B and D
5. RHR System Valves B and D
6. Drywell Control Air System 1B

7. CCW Sectionalizing Valve FC70-48

8. CCW Isolation Valve FC70-47

9. RWCU Isolation Valve FC69-12

10. Reactor Core Isolation Cooling (RCIC) System Steam Isolation Valve FCV71-2

11. Units 1 and 2 Standby Coolant Valve

12. RWCU Holding Pump 1B

13. Drywell Equipment and Floor Drain Sump Pumps 1B

14. Drywell Blowers 1A-4 and 1B-4

15. Turbine Bearing Lift Pumps

16. SLC Trace Heating (alternate supply)

17. All cross tie valves in RHR System

18. Various fans and sump pumps

480 Volt Reactor MOV Board 2B

1. Loads for Unit 2 that are comparable to those above.

480 Volt Reactor MOV Board 1C

1. CRD System Valves

2. RWCU System Valves

3. Drywell Blowers 1A-5 and 1B-5

4. Miscellaneous drain valves

5. Miscellaneous sump pumps

6. Cooling water valves for turbine equipment

7. Recirculating System Valves

8. RHR Inboard Valve RCV75-53

480 Volt Reactor MOV Board 2C

1. Loads for Unit 2 that are comparable with those above.

Table IV (From Section 8 of the FSAR)
250-VOLT d-c CONNECTED LOADS

Unit Battery Boards

1. Reactor Motor-Operated Valve Boards (3 per unit, A, B, and C)
2. Turbine Building Distribution Board (1 per unit)
3. Circuit Breaker Board 9-9 (1 per unit)
4. 480-Volt Shutdown Board Controls
5. Emergency d-c Lighting

36| 6. Unit Preferred a-c Motor Generator

7. 4160-Volt Shutdown Board Controls (unit 3)

45| 8. Bus-tie Board

9. Cooling Tower Switchgear

Station Battery Board

1. Main Turbine Emergency Bearing Oil Pumps
2. Main Generator Emergency Seal Oil Pumps
3. Plant Preferred a-c Motor Generator
4. D-c Emergency Lighting at Diesel Generator Building
5. Distribution Board 9-24
6. Diesel Generator Air Compressors

45| 7. Cooling Tower Switchgear

Reactor MOV Board Loads

1. Autodepressurization Relief Valves (MOV Boards A, B and C)
2. Pressure Relief Valves (MOV Boards A, B and C)
3. Main Steam Isolation Valves Solenoids (MOV Boards A and B)
4. Recirculation M-G Set Emergency Oil Pumps (MOV Boards A and B)
5. Backup Scram Valves (MOV Boards A, B and C)
6. RHR Shutdown Isolation Valves (MOV Board B)
7. Division I Engineered Safeguards Logic Power Supply (MOV Board B)
8. Reactor Building Emergency d-c Lighting (MOV Board B)
9. HPCI Turbine Controls and Auxiliaries (MOV Board A)
10. HPCI Valves (MOV Board A)
11. RCIC Turbine Controls & Auxiliaries (MOV Board C)
12. RCIC Valves (MOV Board C)
13. Division II Engineered Safeguards Logic Power Supply (MOV Board A)

35| 250-Volt d-c Control Power Supplies

45| 1. 4160-volt Shutdown Board Controls (units 1 & 2)

Auxiliary Power Supplies and Bus Transfer Schemes

Table V (From Section 8 of the FSAR)

Board and/or Main Bus	Power Sources: Normal	Power Sources: Alternate	Remarks
4160-V Start bd 2 — Start Bus 2B	COM SS TR B, secondary 2, fed from Athens or Trinity 161 kV lines	COM SS TR A, secondary 2, fed from Athens or Trinity 161 kV lines	
4160-V Bus Tie Board	Cooling Tower Transf TCT1	Cooling Tower Transf TCT2	Manual transfer from the normal power source to the alternate power source, or vice versa, is provided by operator; breakers [illegible] and [illegible] from the cooling tower control board in units 1 & 2 control room, control switches for these breakers are also provided on the 4160-V cooling tower switchgear A.

Board and/or Main Bus	Power Sources: Normal	Alternate 1	Alternate 2	Alternate 3	Alternate 4	Remarks
Shutdown Bus 1 (4160-V)	4 kV unit bd 1A or 2B pre-selected on-line unit, fed from secondary 1 of unit SS TR	4 kV unit bd 2B or 1A (that source not pre-selected for "normal")	Same 4 kV unit bd of pre-selected unit, but fed from start bus 1 A or 1 B	Two diesel generators, if required for backfeeding a pre-selected 4 kV unit bd (1A, 2B) See also remarks for items 13, 14, 15, and 16	Bus tie board	The two independent shutdown buses normally supply 4160-V power to assigned 4160-V shutdown boards, with each bus serving as the normal source to two boards and as the alternate source to the two other boards. Of the two possible feeders to each shutdown bus from the two 4 kV unit boards, one feeder is preselected manually as the normal source to that bus. Automatic delayed transfer from the normal to an alternate 1 source is initiated by undervoltage on the normal source. Automatic [illegible] transfer from the normal to an alternate 1 source is initiated when the normal source 4 kV unit board normal source breaker trips. If an alternate 1 source is not available, the transfer is prevented, and the normal source [illegible] alternate 2 source. Automatic transfer is [illegible] after time delay in the presence of an accident signal. Alternate 3 and 4 sources may be selected manually only.
Shutdown Bus 2 (4160-V)	4 kV unit bd 1B or 2A, of pre-selected on-line unit, fed from secondary 1 of units SS TR	4 kV unit bd 2A or 1B (that source not pre-selected for "normal")	Same 4 kV unit bd of pre-selected unit, but fed from start bus 1 A or 1B	Two diesel generators, if required for backfeeding a pre-selected 4 kV unit bd (1B, 2A) See also remarks for items 13, 14, 15, and 16	Bus tie board	

Table V (From Section 8 of the FSAR)

Auxiliary Power Supplies and Bus Transfer Schemes -

Board and/or Main Bus	Power Sources: Normal	Alternate 1	Alternate 2	Alternate 3	Remarks
4 kV Shutdown Board A	Shutdown Bus 1	Shutdown Bus 2	Diesel generator A	Manual, access connection to diesel generator 3A	(See also remarks for Items 5 and 6.)
4 kV Shutdown Board B	Shutdown Bus 1	Shutdown Bus 2	Diesel generator B	Manual, access connection to diesel generator 3B	Automatic delayed transfer from the normal to alternate 1 source is initiated by undervoltage on the normal source, and automatic return is initiated by normal voltage on normal source. These transfers are blocked after time delay in the presence of an accident signal.
4 kV Shutdown Board C	Shutdown Bus 2	Shutdown Bus 1	Diesel generator C	Manual, access connection to diesel generator 3C	All diesel generators are automatically started by an accident signal, loss of start bus voltage (see items [illegible]-4), or by loss of voltage on its shutdown board for 1.5 seconds. After [illegible] seconds without voltage on the shutdown board, all its supply breakers and all its loads except 4160-480-V transformers are automatically tripped. Alternate 2 source is then automatically connected. Manual return to the normal auxiliary power system is permitted if normal auxiliary power system voltage returns and if a unit is not in early stage of accident.
4 kV Shutdown Board D	Shutdown Bus 2	Shutdown Bus 1	Diesel generator D	Manual, access connection to diesel generator 3D	

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Board and/or Main Bus	Power Sources Normal	Power Sources Alternate	Remarks
480-V Shutdown Boards			
Unit 1, 480-V Shutdown Bd 1A	4 kV shutdown bd A	4 kV shutdown bd B	Transfer from the normal to the alternate source is manual. Interlocking is provided to prevent manually transferring to a faulted board and to prevent paralleling two sources.
Unit 1, 480-V Shutdown Bd 1B	4 kV shutdown bd C	4 kV shutdown bd B	
Unit 2, 480-V Shutdown Bd 2A	4 kV shutdown bd B	4 kV shutdown bd C	
Unit 2, 480-V shutdown Bd 2B	4 kV shutdown bd D	4 kV shutdown bd C	
Unit 3, 480-V Shutdown Bd 3A	4 kV shutdown bd C	4 kV shutdown bd D	
Unit 3, 480-V Shutdown Bd 3B	4 kV shutdown bd A	4 kV shutdown bd D	
480-V Reactor MOV Boards			
Unit 1, 480-V Reac MOV Bd 1A	480-V Shutdown Bd 1A	480-V Shutdown Bd 1B	
Unit 1, 480-V Reac MOV Bd 1B	480-V Shutdown Bd 1B	480-V Shutdown Bd 1A	
Unit 1, 480-V Reac MOV Bd 1C	480-V Shutdown Bd 1B	480-V Shutdown Bd 1A	Automatic transfer from the normal to the alternate source is initiated by time-undervoltage on the normal source. Return to the normal source is automatic upon return of voltage to the normal source.
Unit 2, 480-V Reac MOV Bd 2A	480-V Shutdown Bd 2A	480-V Shutdown Bd 2B	Transfer from the normal to the alternate source is manual. Interlocks prevent transferring a fault from one source to another and paralleling sources.
Unit 2, 480-V Reac MOV Bd 2B	480-V Shutdown Bd 2B	480-V Shutdown Bd 2A	
Unit 2, 480-V Reac MOV Bd 2C	480-V Shutdown Bd 2B	480-V Shutdown Bd 2A	Automatic transfer from the normal to the alternate source is initiated by time-undervoltage on the normal source. Return to the normal source is automatic upon

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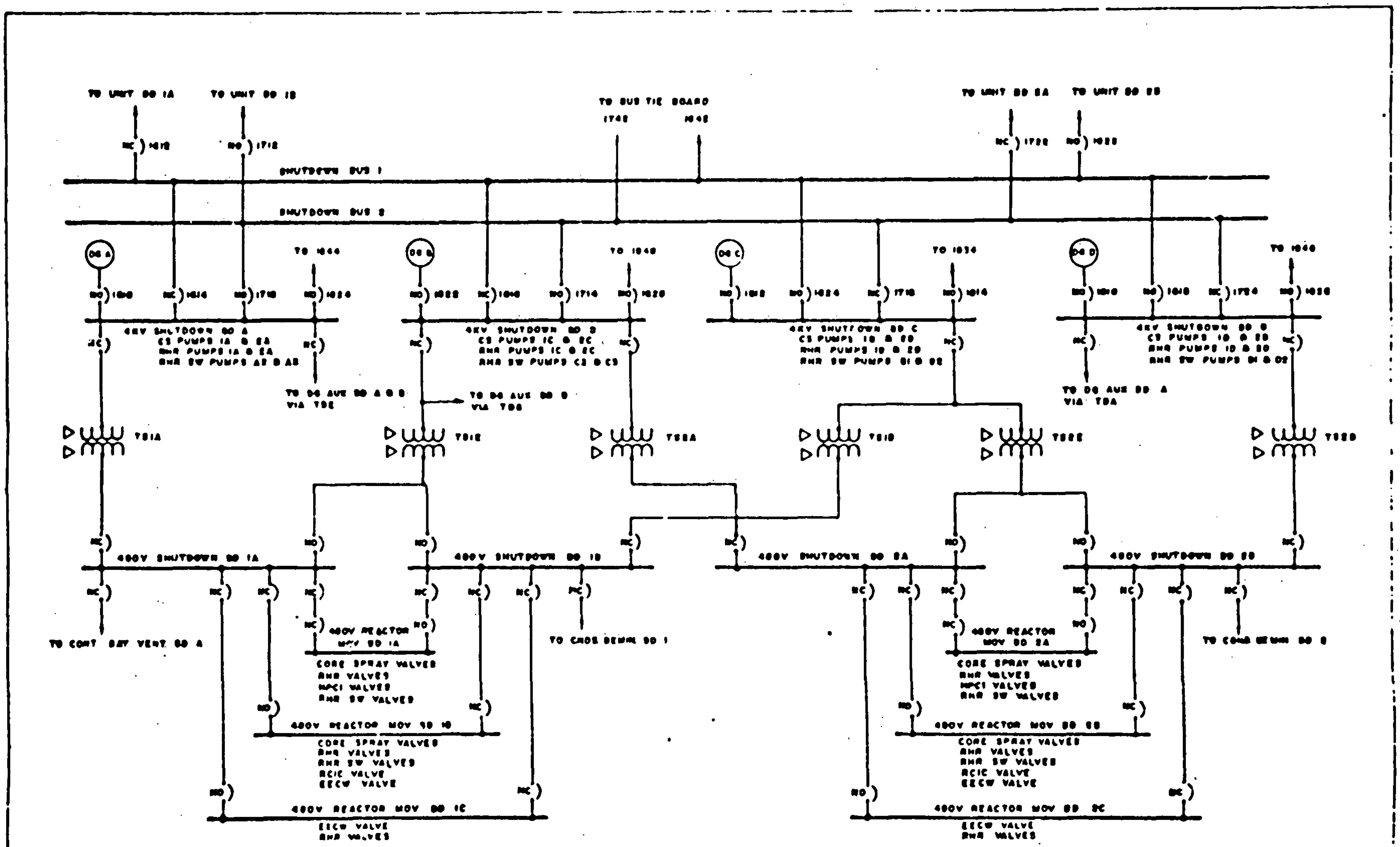

SHUTDOWN AUXILIARY POWER

EXHIBIT C-2 FIGURE 1

Unit 1 Systems Lost During Fire

AUTOMATIC DEPRESSURIZATION SYSTEMS	PROBABLE CAUSE OF LOSS
7 of 11 Relief Valves Inoperable From Control Room	Loss of 250 VDC Boards 1A and 1B
Manual Control of 4 Remaining Valves	Lost in Supply to the Valves

Therefore, manual operation of relief valves lost on Unit 1. Automatic pressure relief still available. Air stored in the valve accumulators provide a discrete number of operations after the loss of the normal air supply.

HIGH PRESSURE COOLANT INJECTION SYSTEM	PROBABLE CAUSE OF LOSS
HPCI	Lost because power to valves lost on 250VDC MOV Board 1A and Power to steam isolation valve lost on 480V MOV Board 1A.

RESIDUAL HEAT REMOVAL SYSTEMS	PROBABLE CAUSE OF LOSS
RHR System 1A	Lost because power to valves lost on 480V Board 1A.
RHR System 1B	Lost because power to valves lost on 480V MOV Board 1B and loss of power to pump motors on 4KV Shutdown Board C.
RHR System 1C	Lost because power to valves lost on 480V MOV Board 1A.
RHR System 1D	Lost because power to valves lost on 480V MOV Board 1B.

Therefore, all RHR systems lost on Unit No. 1.

Exhibit C3
Page 1 of 2

CORE SPRAY SYSTEMS	PROBABLE CAUSE OF LOSS
Core Spray System 1A	Lost because power to valves lost on 480V MOV Board 1A.
Core Spray System 1B	Lost because power to valves lost on 480V MOV Board 1B and power to pump motors lost on 4KV Shutdown Board C.
Core Spray System 1C	Lost because power to valves lost on 480V MOV Board 1A.
Core Spray System 1D	Lost because power to valves lost on 480V MOV Board 1B.

Therefore, all core spray systems lost on Unit 1.

STANDBY LIQUID CONTROL SYSTEM	PROBABLE CAUSE OF LOSS
SLC Pump 1A plus squibb valves.	Lost because power to pump motor and valves lost on 480 V Shutdown 1A.
SLC Pump 1B plus squibb valves.	Lost because power to pump motor and valves lost on 480V Shutdown Board 1B.

Therefore, SLC lost on Unit 1.

REACTOR CORE ISOLATION COOLING SYSTEM	PROBABLE CAUSE OF LOSS
RCIC	Power to the steam isolation valve lost on 480V MOV Board 1B.

REACTOR PROTECTION SYSTEM	PROBABLE CAUSE OF LOSS
RPS MG Set 1A	Lost because power to motor lost on 480V MOV Board 1A.
RPS MG Set 1B	Lost because power to motor lost on 480V MOV Board 1B.
Units 1 and 2 Standby Coolant Valve	Lost because power test valve lost on 480V MOV Board 1B.

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TEMPORARY CABLE LOG

Cable/Wire Number	From	To	Function
1 PL992	480V R. Vent Board 1A	480V T. Vent Board 1A	TB Exhaust Fan 1A Control
1 PL1002	480V R. Vent Board 1A	480V T. Vent Board 1A	TB Exhaust Fan 1B Control
1 PL1187	480V R. Vent	480V T. Vent	TB Exhaust Fan 1B Control
ES 1607-IA	Pnl 25-45A	Pnl 25-45C	Common accident signal to diesel generator
ES 4107-IIC	Pnl 25-45A	Pnl 25-45C	Common accident signal to diesel generator
ES 4393-IIC	Pnl 25-47C	4KV SD Bd C	Diesel Generator C Differential Circuit
ES 4457-IIC	Pnl 25-47C	4KV SD Bd C	4KV Breaker 1812 Close Circuit
ES 4285-IIC	Pnl 25-47C	4KV SD Bd C	Diesel Generator C Protective relay circuit
1ES 4382-II	Pnl 9-36 Unit 1	Pnl 25-219 Unit 1	Plant refuel zone isolation signal
1ES 4385-II	Pnl 9-36 Unit 1	Pnl 25-221 Unit 1	Reactor zone isolation initiation
1ES 4375-II	Pnl 9-36 Unit 1	JB2801	FSV-64-40 Control Reactor Zone exhaust
1ES 4330-II	Pnl 9-36 Unit 1	480V Dsl. Aux. Board B	Relay LRB-6 Control
1ES 4368-II	Pnl 9-36 Unit 1	JB2788	FSV 64-44 Control refuel zone exhaust
1ES 4353-II	Pnl 9-36	JB2801	FSV 64-43 Control Reactor zone exhaust
1B76	250V R. MOV Board 1A	Battery Bd. 1 Pnl 2	250 VDC Reactor MOV. Board 1A Normal Feed

EXHIBIT C4
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Cable/Wire Number	From	To	Function
1B77	250V R. MOV Board 1A	Battery Board 2	250 VDC Reactor MOV Board 1A Emergency Feed
1A2318	Pnl 9-4	JB 3404	Seismic Alarm
1A2319	Pnl 9-4	JB 3404	Seismic Alarm
1A2320	Pnl 9-4	JB 3404	Seismic Alarm
1ES4331-2	Pnl 9-36	JB 2567	Reactor building vent
1R1671	Pnl 9-18	Pnl 25-59	RHR Sys. I Discharge Press Inst. loop 74-51
1R1691	Pnl 9-19	Pnl 25-62	RHR Sys. II Discharge Press Inst. loop 74-65
1R1578	Pnl 9-18	Pnl 25-1	CS Sys. I Press Inst. loop 75-20
1R1588	Pnl 9-19	Pnl 25-60	CS Sys. II Press Inst. loop 75-48
1R1000	Pnl 9-5	Pnl 25-18	CRD Cooling Water ΔP Inst. loop 85-18
1R1031	Pnl 9-19	Pnl 25-18	CRD Cooling Water Hdr. Flow Inst. loop 85-25
1R1032	Pnl 9-19	Pnl 25-18	CRD Cooling Water Hdr. Flow Inst. loop 85-25
1CR950	JB 1043	Computer	Vessel Temperature TE 56-1
1CR955	JB 1043	Computer	Vessel Temperature TE 56-3
1CR960	JB 1043	Computer	Vessel Temperature TE 56-17
1CR965	JB 1043	Computer	Vessel Temperature TE 56-30
1R1681	Pnl 25-62 Unit 1	Pnl 9-19 Unit 1	RHR Instrumentation
1R1686	Pnl 2562 Unit 1	Pnl 9-19 Unit 1	RHR Instrumentation
1R1656	Pnl 25-59 Unit 1	Pnl 9-18 Unit 1	RHR Instrumentation

Cable/Wire Number	From	To	Function
1R1666	Pnl 25-59 Unit 1	Pnl 9-18 Unit 1	RHR Instrumentation
1R2740	Pnl 9-3 Unit 1	Penet. CB RS El 565	Drywell Temp. Recorder
1NM2-IA	Pnl 25-27 Unit 1	Pnl 9-12 Unit 1	SRM A Signal
1NM3-IA	Pnl 25-27 Unit 1	Pnl 9-12 Unit 1	SRM A HV
1NM4-IA	Pnl 25-27 Unit 1	Pnl 9-12 Unit 1	SRM A -15 VDC
1NM18-IB	Pnl 25-27 Unit 1	Pnl 9-12 Unit 1	SRM B Signal
1NM19-IB	Pnl 25-27 Unit 1	Pnl 9-12 Unit 1	SRM B HV
1NM20-IB	Pnl 25-27 Unit 1	Pnl 9-12 Unit 1	SRM B -15 VDC
1NM10-IIA	Pnl 25-61 Unit 1	Pnl 9-12 Unit 1	SRM C Signal
1NM11-IIA	Pnl 25-61 Unit 1	Pnl 9-12 Unit 1	SRM C HV
1NM12-IIA	Pnl 25-61 Unit 1	Pnl 9-12 Unit 1	SRM C -15 VDC
1NM27-IIB	Pnl 25-61 Unit 1	Pnl 9-12 Unit 1	SRM D Signal
1NM28-IIB	Pnl 25-61 Unit 1	Pnl 9-12 Unit 1	SRM D HV
1NM29-IIB	Pnl 25-61 Unit 1	Pnl 9-12 Unit 1	SRM D -15 VDC
1PC518	250V R. MOV Board 1B	Pnl 9-43 Unit 1	FCV 69-2 Control
1PC517	250V R. MOV Board 1B	JB 1088 SR4 B1 593	FCV 69-2 Control
1PC516	250V R. MOV Board 1B	FCV 69-2 Motor	FCV 69-2 Power

Cable/Wire Number	From	To	Function
1PC506	480V R. MOV Board 1B	Pnl 9-43 Unit 1	FCV 69-12 Controi
1PC505	480V R. MOV Board 1B	JB 977	FCV 69-12 Control
1PC504	480V R. MOV Board 1B	FCV 69-12 Motor	FCV 69-12 Power
1V2173	480V R. MOV Board 1C	Pnl 9-4 Unit 1	FCV 69-16 Control
1V2171	480V R. MOV Board 1C	JB 1031	FCV 69-16 Control
1V2170	480V R. MOV Board 1C	FCV 69-16 Motor	FCV 69-16 Power

Calhoun calling Green. Fred Chandler called and said that somebody from BFNP had asked him to get some people out to do a little checking on what cables go in certain cable trays and all.

What I have going is a kind of a task group of Construction and DPP (mostly Construction) looking toward the repair of this and my immediate representative is Jerrold Dewease working with Ben Gant and they have lined out a plan of attack toward repairing this and Jerry just went over with me the plan and the design part was to get Design to identify everything in these trays, so they can have this information ready when we went to taking out circuits to chop them out to place in new circuits. So I think what we are talking about is not just today's problem but toward getting these units back.

Okay. Fred's concern was that they have not gone through you.

But they have gone through me and I've got a representative on this task force just assuming we are going to call on Construction to help to get the unit back.

Okay.

Here's Jerry's notes. It starts out, list all trays that show damage. DEC will complete a list of cables on each tray. DEC will identify where they go and this sort of thing and then the power and control cables can be spliced, the thermocouples coax will be replaced. DED should be notified and I understand they have been. (JRC says yes.) That's what it is--just trying to get together with Construction and looking a little bit ahead.

Okay, Fred wanted to relay this back through here so we wouldn't get fouled up, but I think I can tell him on this particular problem to work directly with that task force. Right?

Yes, work with the task force and I believe Ben Gant is the construction man that is taking the lead. But, of course, Ben might have some other engineers involved, too.

Yes. Now hold on, I've got some other messages for you too.

Recford Arnold and Fred Von Hollen are coming down there to help you.

Okay.

Mr. Thomas has something here. (EFT speaks in background--JRC repeats EFT's question) Is # 1 turbine safely on turning gear?

Yes, they tell me it sounds good.

No troubles with bearings as far as they can tell?

No.

Okay, now Frank Long, Atlanta, called and says he is sending Bob Sullivan, Bill Little, and Bower (electrical man). So they left maybe thirty minutes ago and they will be down there sometime tonight.

Okay.

He asked me did you know whether your control rods were in on unit 1 and I told him that I didn't know.

I've got indication that they are all in.

Okay, good. Now what's the situation on the fire?

Jack, we're fighting it with water now and our men where breathing air is a big thing, but they have gone in twice and had it knocked down with water and had to get out and each time they go back in _______________ and they are in there for another entry right now and I've got hopes that the fire is about under control.

So, they are not using too much water apparently.

No, they are not trying to use too much water.

Do you have any indication of how much of that cable tray outside is involved?

Yes, there is a vertical that runs up and down the wall and two places on horizontal and I din't know just how much, but I got the thing in my mind probably about 15 feet horizontal.

Okay, that's the sort of information we wanted. Now, has there been any change in the torus temperature on unit 1?

No, right now unit 1 has _______. I've lost my unit valves on unit 1, and I know what it is and I know what to do about it. They have burned through an air line, so killed the air and we killed control air while relief valves went shut and I'm trying to establish vacuum and going to use the steamline drain stub to help pull steam off the turbine and I don't know what the absolute torus temperature is. I've been down watching the firefighting efforts, Jack. Just came up.

But, you've got a way that you'll be able to quit but you're not dumping anything in the torus anyway, are you?

Not right now, don't have relief valves. I'll get another rundown here shortly and get back at you.

Yeh, let us know as soon as possible.

Okay.

(Later) Okay, I am reporting that the fire appears to be out. Right now, unit 2 is still down at about 100 pounds and makeup water is the hotwell pump. Okay, unit 1 I've got in a condition where I'm leading steam to the condenser through the drain lines on the steam lines and hold the vacuum with the hoggers and making up water with the control rod drives and I seem to be holding pretty steady at about 500 pounds. The question you asked me: unit 2 torus temperature is 106° and unit 1 torus temperature is unknown.

Okay, you don't have any way to get down to the torus, do you?

I haven't attempted it yet, I have too many other problems.

Okay. Alright. Jim, I guess I didn't have anything to talk to you about. Oh, we've got 35 more breathing air bottles on the way down to you.

Okay, I don't know which came in where, but we got some air in here from some place that we are using to recharge cylinders and we are also using Chemox and that's working pretty well. I got a call that Sullivan, Little, and some other NRC inspector are traveling tonight and will get here some time tonight so all our problems will be over. (JRC laughs).

They will square you away, I am sure.

We probably have a violation. We've kept very poor logs. (JRC laughs)

No doubt.

I wouldn't say we are out of the woods, but I feel better now that the fire is out and we should be able to get in the reactor building in a little while without all this breathing air and mask, you know.

Can you expand a little bit on that air line that burned in two?

Just a minute. Jack, I can't give you the definitive. I'm talking to Joe Mantooth, who was the operator in there, and he says that in that area there is a 6" riser going up. He doesn't think it was the main riser itself, but probably a leg off of it, but that there was an air blow in there blowing into the fire. Keep in mind that it was very poor visibility. With flashlights just a matter of several feet so he's not definitive of it all.

Okay, thank you.

Okay.

Jim has reported that he has the fire out.

Okay.

What was taking him so long, he finally resorted to water, but he wanted to do it with the spray and just controlled amounts of it and what he would do was to put a little bit on it and wait a while and it would flare up and he would put some more on it. Now apparently this is in a space of about 15 feet in this particular cable tray. So he has the fire out. Unit 2 is holding at 100# pressure. And he is feeding water to unit 2 with the hotwell pumps. No problem on unit 2. Now unit 1 since I called you last ran into another problem. They had an air line that was in the vicinity of this fire and they lost air pressure to the relief valves so his relief valves went out on him and now they have the heatsink as the main condenser on unit 1 and taking steam through the main steam drain lines to the main condenser on unit 1. This seems to be stablizing the pressure on unit 1 to 500 psig. Unit 2 torus temperature is 106° F. Unit 1 torus temperature is unknown.

What happened there?

Well, he lost his unit 1.

Measuring that directly before estimating?

He indicated to me at one time that he was getting some indication in the auxiliary shutdown room or on the emergency control room that he got a reading there and that was the one I gave you at what 126°.

On what #1?

Yeh, 126° and I'm pretty sure he told me he was getting that from the emergency control room but now he says he doesn't know what the temperature is on the torus. So, I'm a little bit unclear as to where he got that first reading.

Okay, that just could have been some good educated guess.

I'm pretty sure that he told me he got it from the control room. But I'm not absolutely sure.

Okay, what's the status of main control room? Is that all filled with smoke still?

Oh, no. It never had much smoke in it. The main control room is perfectly okay. They had some smoke coming through a hole close to the control deck in the center of the room. They stuffed some rags into it and got it stopped. But he thinks he will have the smoke out of the reactor building pretty soon. We got a call through some of our people that Mr. Ed Giller was calling from Washington. Do you know Ed Giller? I probably figured that he was their duty man.

Yeh, he's probably their duty officer. Several of them were in the offices up there. Just to see what was going on.

Now, we ended up by asking him to call you. Is that okay?

Yeh. We have some people up there in our headquarters offices. The doggone public news media types will probably drive you out of your mind. Okay, your people did put out a news press release?

Yeh, we put one out about 4:30. Somewhere close to 4:30.

What kind of fire fighting--Did you have any problems with your station fire system?

As far as I know, it worked fine. The CO_2 system in that particular room works okay. But it ends up with a cable fire like this the CO_2 won't put it completely out. Sort of like a mattress smoldering.

Yeh, you can't really get to cool it.

Right? No cooling. So it prevents a conflagration but it won't just finally put it out. Now the Athens fire department got there quickly and I guess I told you we locked them right in to the reactor building with fire trucks. So they were there and acted as advisors and did real well.

Oh, you didn't have to use their equipment?

Yeh, we used some of their equipment too. Principally, this particular (now I'm guessing here now) but Jim was going to use his spray nozzle for this water and I assume that he did use the Athens fire department spray nozzles. So that part of the plan really worked out well. He had the fire department there before I ever got here in the emergency control center here in Chattanooga.

Is that part of your regular fire plan?

That's part of the regular fire plan. See, Jim takes the Athens fire department into the plant and trains them in radiation and also takes them through the plant so they are familiar with the inside of the plant, too. This is all a part of the plan.

Tell me what you know about the cooling thing for unit 1. Give me that again _ _ How you are keeping that temperature.

We are taking steam through the drain lines. These are fairly small drain lines that go to the main condenser. And we're passing steam through those drain lines to the main condenser. And maintaining a vacuum on the main condenser with the hogging pumps.

Do these drain lines have that capacity to take care of that for you?

A 2" drain line off each main steam line. Yeh. So it's holding and it will get better as time goes on. As the decay heat drops.

Does each of your units have two condenser circulating cooling pumps?

There's 3 main circulating condenser pumps. Yeh. Each unit.

Condensate pumps, you've got two or three?

Three, six I guess condensate booster pumps. One for each condenser--Three condensate boosters.

On each unit?

Yeh.

That explains. I was trying to figure whether there was one or two on a plant. You're relying on them. That handled it pretty well, didn't it?

Oh, yeh, we had plenty of cooling water.

It looks like everything is under control.

Things are looking up.

That fire wasn't as extensive as it appeared to be? Pretty localized by 15-foot area length?

The 15-foot area I am talking about is outside the spreading room. The fire went through a penetration the way it looks so I don't know what to say about inside the spreading room.

That was probably pretty extensive in there.

Well, we just don't know. It must have been to affect two units. But we have a feeling now it is nothing like completely wiping out the spreading room. We can say maybe three or four cable trays in there. Just some guess work.

Really don't know?

Really don't know.

Only thing we can say right now is that it could have been a hell of a lot worse.

Oh, yeh.

You know, when you talk about a fire in the spreading room, you've really got problems.

It would affect just about everything.

Yeh, you know everything for those two units come through that one room. It's common to both units, just like the control room is common to both units.

That sorta shoots your redundancy.

We're got a lot of questions to ask about this.

Well, I hope it turns out that it wasn't really as extensive as it seems to have been.

Smoke really fools one and he just doesn't know. Maybe before the night is over, we will have a pretty good feel for the extent and all because he is getting the smoke cleared out now.

Okay. I'll just sit back and wait. I sure do appreciate your calling me, Jack.

Thank you, Frank. Bye.

Exhibit D1
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Fire is essentially out. On the spreading room end, every now and then you see a glow and we got a man standing by like continuously. Now we have set up routine patrol in the drywell and the area where we have fire, you know. Now, #1 reactor building is cleared by the health physicist, but I haven't got it cleared by the industrial hygienist. They are concerned about the CO. So I am still in a mask. Unit 2 is in the best shape and right now we are attempting to get it on shutdown cooling. And it looks like we've lost the (I guess it would be the A & B shutdown) boards because we have lost the A and C pumps, this type of thing you know. And if you'll think that one through, it means that what we have really lost are the controls because where the fire was there were no power cables.

Yeh, okay.

So unit 2 has two RHR pumps, two core spray pumps, what have you. Unit 1 we just got a relief valve open. I think the last time I told you we couldn't get air on a relief valve and we have just got in a relief valve so the pressure is coming down.

Okay.

At the present time I don't have any way to get one in unit 1 except the control rod drives for when I get down to 150 # I can get it in through the condensate system.

Yeh, I understand.

Unit 1: One thing I am going to do is to put in the spool pieces on the RCIC. I think it would be inoperable except the steam isolation valves inside the containment is shut and if I get spool pieces in, it will give me another way to put water using auxiliary steam.

From heating boiler?

Yeh. When I get operators down to it, I can open the valves manually.

Yeh.

And we are starting from nothing to make a procedure to flush the shutdown cooling __________ and put in service on unit 1. There will be a manual alignment, you see. On unit 1 I have most of the CAM's back in service and I have Construction here, and what we plan to do was to take out a component at a time starting with the __________ well, I'm trying to find a way to get core spray and RHR and this thing back in service, you know. And starting with the _________ I'm going to start like a component at a time and say well I need this valve. Ben Gant's people will work with us and we're going to disconnect the control cables from that thing and at the breakers I will put in a local pushbutton that I can operate that valve with and starting a scheme of this nature to try to wind up to get some of this safeguards equipment operable for unit 1 for local control. Now we have established that from here on we are not going to do any of these bastard things without a work plan.

That's good.

We've done some things out of the extreme. We are trying to reestablish control. And Studdard's people are working with Ben Gant's people to identify each of the components that we need and I think we have and could operate this core spray and they are making the proposal what they think they can do to get it in service since the boards hot and apparently all we have lost are controls for it and then we will plan it through. And it looks like most of the night is going to be spent in making procedures to get this thing on shutdown cooling and starting on reviewing procedures and workplans toward getting some of the safeguards equipment back in service.

Okay, Jim, do you know the temperature in the drywell or the torus temperature?

Unit 1 the drywell temperature? I don't know the pressure, I don't know the torus temperature, I don't know the torus level. And we are giving a worklist of priorities to the instrument people and one of the (understand several jobs can run simultaneously) but one of the top priority jobs is that we are going into the drywell and set up a local neutron monitoring station (not in the drywell) in the reactor building _______ set a local neutron monitoring station and we've established that when we set that up we will have to keep an operator in to collect data all the time and then about the No. 2 job will be torus level. This is of particular concern because we don't have any idea where the level is. The side torus temperature is a low priority getting the permanent instrumentation in that will ______ the thermometer to the side of the torus.

Sure.

Somewhere down the list appears then drywell temperature and drywell pressure. We have the small vent open on the drywell that is, the 2" vent; so we know that we are probably not in any overpressure condition or anything like that. And that does vent out through standby gas, you understand that.

Yeh.

Drywell pressure, I feel sure, is probably nothing. But I don't know temperature.

Okay.

__________________. A truck with air just came through the gate.

Okay, good, that should get us out of a bind some on air. I just sent Athens fire department home. And I don't how the hell we ever pay them or anything like that.

Well, they did a good job.

They did us a good job. And they left some of their equipment here that they couldn't find and what have you and I said we'll find it and get it to you or we'll pay you for it.

Okay, sure.

And the only other thing I've been kind of treating them by buying meals. I've bought meals for Athens firemen and everything else.
You're in trouble.
Probably am. That's the biggest __________ decision, you know. And blindly ordered 50 breakfasts.

Jim, unit 1 reactor water level, what's the water level?

I don't know, but there is no problem with it.

Okay. Atlanta asks that ever once in a while.

Well, we haven't had a level problem. Unit 1 we didn't have any way to get steam out so you didn't have to put water in.

Right.

And it's not that ____________ humorous. That situation, unit 1 right now is still not good. The only way we are putting water in is control rod drive. The bright spot _______ we did get a relief valve open and we still do have a condensate system to put water in when we get pressure down.

So the biggest problem is getting cooling to the torus. Isn't it?

Right.

The crosstie doesn't help you any in this case, does it?

No, crosstie don't help any in this case.

Okay, Jim, what are you going to do for yourself?

Well, I'm going to stay here tonight. I'm going to sleep here.

Alright.

Either Studdard or Hunkapillar are going to be here all night.

Allan Qualls is down there some place if he can help you any. Probably in Athens at home.

Oh, hell, Allan Qualls is probably down here visiting his daughter.

Yeh.

It's good to know. If we need him, we will call him. Don't think we are in too bad a shape at that level in that we got some pretty competent shift engineers, you know. If we run into any problem, since we had so many out here today, tat it is going to be a damn long night for them. Mainly, we are going to keep maintenance crew and all crafts here all night long. And Construction has a good many engineers and electricians and craftsmen out tonight just standing by to help us. They have cancelled all construction work, but they did bring in a crew to stand by for any support we need.

Is there anything that we need to do for you in the morning that we could get started tonight?

Jack, the main thing that came to mind that you can do for me and I'm thinking ____________ by April 1 I'm supposed to have this fellow from ABC to take a film report of the plant, you know, some of these tour guide groups, if we can get some of those cancelled out.

That was April 1?

Right.

Well, okay.

That's a little facetious at this time. We can worry about that Monday. But right now I don't know of the support I need from you.

Okay.

You can cancel the relief valve test on unit 2 for awhile.

Yeh, we'll do that.

Jim, Fred Von Hollen and Reeford Arnold, I don't guess they have arrived yet, but they are on their way and if you need any more people out of us in the morning let us know.

Fred Von Hollen and Reeford Arnold are on the way down for any help they can be.

We will keep someone here in the office all night.

Somebody is going to be at this number all night.

Right.

I'll call you if I need you, but I don't know of a specific that you can do for us right now.

Well, I would like to know when you get the RHR restored. If it is sometime during the night, let them know here in the office

On unit 1?

Yeh.

If we get an RHR, I'll let you know.

When you get cooling to the torus. Yeh.

Okay.

Okay.

Log of J. R. Calhoun, Chief, Nuclear Generation Branch, TVA, for March 22, 1975.

J. R. CALHOUN'S LOG

MARCH 22, 1975

2:45 p.m. CDT - JAC called JRC

2:40 - JRC talked to HJG

2:41 - Started calling CECC (Called JEHolmes at 2:45 - not home; called JAC at 2:45; and called EFT at 2:47)

2:50 - Left for CECC office

3:10 - Arrived in CECC office

3:20 - Called CJHodges

3:15 - RCPrice arrived

3:23 - RCPrice called TVA Information Office

3:23 - Called GFStone - no answer

3:25 - Chris Eckl going to Information Office in Knoxville

3:26 - CCFussell says he is on way to CECC office

3:27 - CAMcLaughlin and JAC reported to Division Emergency Center

3:35 - JAOppold reported to take charge of CECC

3:40 - JEWatson notified.

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3:40 - HJGreen reported following information:

Unit 1 - 200 psig + 55 waterlevel in reactor; condensate booster pump in use supplying water to reactor

Unit 2 - + 35" waterlevel; 150 psig boiler pressure; RCIC being used; backup feedwater system is the hotwell pump

Unit 1 has no ECCS valve control

Reactor building has no area radiation monitoring

Apparently fire is in cable tray (TE) in middle of spreading room wall between reactor building and spreader room

Smoke so heavy can't see

Unit 1 torus water up to 126° F.

No torus instrumentation

100 meals ordered for 6 p.m.

Breathing air may be needed

Locking fire truck into building

Unit 2 is on turning gear

Unit 1 sitting - No lift pumps and not on turning gear

News release made to Chris Eckl

4:04 - HJG

Unit 1 is in worst shape - Has a reactor water level of + 55 and 200 psig pressure

Condensate booster pump in service

Unit 1 does not have torus indication

Torus temperature is 126° F.

Unit 1 does not have nuclear instrumentation

Unit 1 does not have core spray, RHR, HPCI, RCIC, etc., probably because MOV boards are dead

Unit 1 has no turning gear or lift pumps

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Jm
4/22/75

Unit 1 drywell temperature 330° F. or 250° F. - Don't know which because indication is erratic.

Pressure in drywell is 2.2 psig

Only have two drywell cooling fans out of ten

Unit 2 has some core spray, some RHR, HPCI, RCIC, and torus cooling

Unit 2 level in torus is 4"

Unit 2 torus temperature is 130° F. and dropping

Unit 2 drywell temperature is 195° F.

Unit 2 turbine is on turning gear

Unit 2 has neutron monitoring

The nursing station is manned - Nobody has been hurt.

4:23 - Tom Fullmer, BFNP storeroom, called

Can't get Selox Company in Chattanooga - BFNP needs breathing air We will get in touch with Selox and ask them to send some breathing air to BFNP

50 cylinders breathing air on TVA Contract No. 70-X53-BPA 58

4:33 - HJGreen

Service Shops leaving with fire extinguishers and an air compressor

Fire is out on spreading room side

Foam was used in pulling cables - Wasn't fireproof and acting as blow torch

Athens fire department can charge our air bottles at BFNP

At 4:30 both turbines on turning gear

Liquid CO_2 truck being ordered because Cardox system is getting low

EXHIBIT D3
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4/28/75

4:55 - Marvin Mathis on way from Chattanooga to help inspect turbine for possible bearing damage

4:58 - Selox Company, Chattanooga, is sending 15 bottles breathing air to BFNP

Frank
4:59 - Operator says to call Mr./Long in Atlanta (404) 633-2426

5:10 - HJGreen wants to decide what to use on fire

Determined to be "A"-type fire - He is being advised to use water on fire

Cable trays MCES2, MD, TE, and VE were involved in fire

Two more immediately near these were also involved

5:20 - Told HJGreen okay to use water on fire but use water under controlled conditions

Norman Moseley in Atlanta (404) 633-4012

5:40 - Fred Von Hollen and Reeford Arnold going to BFNP - Told to do so by RCPrice

6:06 - JDGlover

Enough heat has been generated in cable trays to cause flare up of fire after being put out

Having trouble with reactor building exhaust fans on unit 1

6:09 - HJGreen says hasn't used water yet. Fire keeps coming back from heat but he wants to hold out for another 30 minutes before using water.

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Cable trays extend into reactor building in a vertical and a horizontal direction

Jim says construction "candled penetration and it flashed" while sealing a penetration. Price has written statement from the men.

6:15 - Fifteen breathing air tanks on way from Chattanooga

6:22 - Fred Chandler's people say they are getting people in to check cable runs

6:29 - Frank Long, Atlanta sending Bob Sullivan and Bill Little and Bower to the plant

6:50 - HJGreen - Has created a task force to determine the cables involved in the fire and a course of action

All control rods on both reactors

Now fighting fire with water

Lost relief valves on unit 1 because of control air line failure in vicinity of fire

7:00 - Thirty-five more breathing air bottles on way to BFNP

7:15 - JEWatson called to get more information on problem

7:25 - Godwin Williams says Washington calling him

Ray Sullivan, ERDA, Oak Ridge says Ed Giller in Washington wants to know what happened. Told him to tell Giller to call Frank Long, Atlanta. We are keeping Mr. Long informed

7:47 - HJGreen - Fire appears to be out. Unit 2 reactor at 100 psig. Feedwater is from hotwell pump.

Unit 1 bleeding steam to condenser thru main steam drain lines. Holding at 500 psig.

Jre
4/28/75
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Unit 1 using control rod drive pumps for feedwater

Unit 2 torus 106° F.

Unit 1 torus temperature is unknown.

- Called Frank Long, Atlanta, and gave him a status report

- JACoffey called Fred Chandler to tell him that DED should talk directly to the cable task force on information regarding cable trays

- Charles McBride says Bob Dunnivan with the Huntsville Times heard that a workmen was testing a penetration for air leakage with a candle.

- HJGreen says Milton Price is pretty sure that candle started fire. Planning on PORC meeting to decide work plan for tonight.

Green says cable trays outside spreading room burned for about 25 feet expanse maybe in 10 trays. Spreading room looks okay. Apparently, all fire outside spreading room. Signs of heat on floor in control room.

Pressure on unit 1 has started to climb

9:00 - Called Frank Long and told him about the candle possibly starting the fire

9:20 - Received following information from Henry Russell and Fred Chandler: Polyurethene or styrofoam sealant used to seal penetration after the polyurethene set ups, the workman checks with candle for leaks and then sprays with a fire proofing material. DED says workmen were pulling two new cables thru a penetration when the fire occurred

EXHIBIT D3

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4/25/75

9:30 - Henry Russell says the following cable trays were involved:

FM, MX, MD, VE, TE, FL

Only tray with engineered safeguards was MX.

Pulling cable 3ES43302 from panel 9-36A, unit 3 auxiliary instrument room to 480-volt diesel auxiliary boiler B, panel 11A, elevation 583, diesel generator building

9:42 - Operator on unit 2 says relief valves are now open on unit 1 and the reactor pressure is dropping.

9:45 - EFThomas going home

9:46 - HJGreen says routine patrol established in area where fire occurred

Reactor building atmosphere pronounced okay by Health Physicist

on unit 1, A&B C&D shutdown boards apparently inoperative - controls probably are inoperative

Unit has RHR's and other systems in service

Unit has relief valves opened. Control rod drives are supplying water to unit 1 - 500 psig reactor pressure

Plan to put in spool pieces to connect the RCIC to use heating boiler steam

Unit 1 constant air monitors in service

Unit 1 water level okay

Unit 1 drywell temperature unknown; torus level unknown; torus temperature unknown

Small vent open on drywell containment/venting thru standby gas treatment

Breathing air arrived from Chattanooga

Athens fire department left plant

Jr
4/28/75

EXHIBIT D3

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10:15 - Gave status to Frank Long, NRC, Atlanta.

Frank says Washington wants following information:

Torus temperature and drywell temperature and drywell temperature time sequence when information is available

10:45 - Gave EFThomas latest status

11:00 - JRC left control cente. Willie Webb and George Messer will stay until 8 a.m. CECC members and others left about 10:15 p.m. Key members will congregate about 7 p.m. 3/23/75

JRC 4/22/75

EXHIBIT D3

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Summary of CECC Activities March 22-23, 1975

Larry S. Fox, Assistant to the Director of Power Production, 716 EB, Chattanooga

James A. Oppold, Assistant to the Director of Environmental Planning, 201 401B, Chattanooga

April 28, 1975

SUMMARY OF CECC ACTIVITIES FOR MARCH 22-23, 1975

In response to your request of April 26, I will provide your investigating committee with a few brief comments about the activities of the Central Emergency Control Center (CECC) on March 22-23. As you know, the CECC Director has the responsibility for evaluating, coordinating, and directing TVA's overall activities involved in coping with a radiological emergency. These responsibilities are set forth in the Radiological Emergency Plan (REP) which describes TVA policies, purposes, delegations, standards, and guidelines in order to protect TVA personnel, plants, and properties, as well as the health and welfare of the general public during a radiological emergency.

J. A. Coffey, Division of Power Production, notified me at 1605 on March 22, 1975, of the fire at Browns Ferry Nuclear Plant and informed me that the REP should be activated and the CECC staffed. I immediately informed E. A. Belvin, Environs Emergency Staff Director, and requested him to alert the State of Alabama. I arrived at the CECC in the Edney Building at about 1625 and learned that, as first alternate, I was to direct the CECC. The following division representatives were notified and asked to stand by on alert: H. G. Moore, Division of Environmental Planning; D. G. Powell, Division of Law; and W. B. Brown, Water Control Planning. Active participants were Dr. R. L. Craig, Division of Medical Services; C. E. Eckl, Information Office; C. F. McBride, Power Information Advisor, Office of Power; and Communication Coordinator, C. J. Hodges. I have drafts of a chronological log by J. A. Oppold, Dr. R. L. Craig, and C. J. Hodges.

The CECC notified the Nuclear Regulatory Commission in Atlanta of the Browns Ferry fire at 1703, and at 1712 a followup contact was made with the Alabama State Department of Public Health. Several telephone conversations were held with each agency throughout the night in an attempt to keep each apprised of the situation at Browns Ferry. This was possible since oral communications and frequent briefings were held with the Division of Power Production Emergency Center. At 2105 the Tennessee State Department of Public Health was notified as a public relations move rather than fulfilling a requirement of the REP.

EXHIBIT D4

PAGE 1 of 3

Harry S. Fox
April 28, 1975

SUMMARY OF CECC ACTIVITIES FOR MARCH 22-23, 1975

Five areas of concern prevailed in the CECC on March 22:

1. Getting the fire out;

2. Maintaining control of the reactors 1 and 2;

3. Detecting, measuring, and controlling any possible radioactive releases into the environment;

4. Recognizing and controlling all potential hazards in the work environment; and

5. Observing workers involved in fighting the fire and treating and caring for anyone who became sick or injured during the fire at Browns Ferry.

Items 1 and 2 were handled by the Division of Power Production's Emergency Center; item 3 by the Division of Environmental Planning's Environs Emergency Staff (see memo, Oppold to Fox, dated April 15, 1975); item 4 by health physicists and industrial hygienists of the Division of Environmental Planning (memos from Oppold to Green, dated April 21, 1975, copy attached, and from Oppold to Thomas, with copy to you, dated April 8, 1975, provide respective findings); item 5 by the Browns Ferry nurses and Division of Medical Services physicians.

There was no evidence of any acute severe injuries or illnesses to any of the employees examined and followup medical evaluations revealed no remedial health effects. No radioactive materials were detected in the environment around Browns Ferry above those normally measured, and inplant worker radiation exposure was no more than normally expected. Several airborne toxic materials were searched for; only carbon monoxide in unit 1 reactor building was measured in significant quantities. Because of the carbon monoxide levels and the unknown makeup of the soot/ash dust, the industrial hygienists were directed to stay inplant until carbon monoxide levels dropped to a safe level and until, in their professional opinion, workers could safely enter areas affected by the fire without self-contained air supplies.

At about 2330 the CECC secured for the night. Two representatives of the Division of Power Production took charge of their Emergency Center. All CECC participants left with these two individuals telephone numbers where each could be reached during the night.

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Harry S. Fox
April 28, 1975

SUMMARY OF CECC ACTIVITIES FOR MARCH 22-23, 1975

At approximately 0800 on March 23 I went to the CECC and remained until about 1615. The log is available on this time period if needed.

TVA's Radiological Emergency Plan and the activation of the CECC were effectively implemented in my opinion. In retrospect, the role of the CECC may have been minimal in this incident, but it certainly provided a necessary function.

James A. Oppold

JAO:DV
Attachments
CC (Attachments):
E. Floyd Thomas, 716 EB, Chattanooga (Attn: J. R. Calhoun)
R. L. Craig, 320 EB, Chattanooga

MC: E. A. Belvin, ROB, MS
Wayne Holley, 150 401B, C
P. A. Krenkel, 268 401B, C
R. S. Rainey, ROB, MS
D. M. Trayer, ROB, MS

EXHIBIT D4
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APPENDIX 7

FINAL REPORT OF TVA PRELIMINARY INVESTIGATION DATED MAY 7, 1975

FIRE AT BROWNS FERRY NUCLEAR PLANT

TENNESSEE VALLEY AUTHORITY

MARCH 22, 1975

FINAL REPORT

OF

PRELIMINARY INVESTIGATING COMMITTEE

MAY 7, 1975

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I. INTRODUCTION

A preliminary investigating committee was established on March 23, 1975, to conduct an early fact-finding investigation of the fire and related events which occurred at the Browns Ferry Nuclear Plant on March 22, 1975. The interdivisional committee consists of the following members:

H. S. Fox, Chairman - Division of Power Production

Charles Bonine, Jr. - Division of Construction

Harry S. Collins, Reporter - Manager of Power's office

David G. Powell - Division of Law

M. N. Sprouse - Division of Engineering Design

Felix A. Szczepanski - Manager of Power's office

The committee's charter is included as appendix A. The committee reported to the plant on March 24, 1975, to initiate its investigation of the fire. A preliminary assessment of the damage was made, numerous interviews were conducted, and a preliminary report of the committee's findings was transmitted to the Manager of Power on April 7, 1975.

II. PURPOSE AND SCOPE

The purpose of this report is to present the committee's findings of facts on conditions and events relative to the fire and to provide a point of reference for other evaluations which may be required.

This report describes events leading up to, during, and after the fire until each unit was placed in the cold shutdown condition.

III. FINDINGS

A. Construction and Operational Status of Plant at the Time of the Fire

Units 1 and 2 were operating at normal full-load capacity, and construction work was proceeding on unit 3.

B. Relevant Design and Construction Features

1. Plant

A positive air pressure is maintained in the control bay, which includes the cable spreading room, with respect to the reactor building. In order to maintain the pressure differential, all penetrations between the control bay and reactor building are designed to provide an air pressure seal. A vertical cross section of the reactor building, control room, and spreading room, which is the area under consideration, is shown as figure 1.

2. Electrical Cable Penetrations

Electrical cable penetrations provide a means of routing cables through barriers such as floors and walls. They can be in the form of conduit or special fabricated steel sleeves.

a. <u>Wall Penetration as Designed</u>

The cable penetration where the fire started is contained in a 48-inch-square opening through the concrete wall separating the units 1 and 2 cable spreading room from the unit 1 reactor building. Division of Engineering Design (DED) drawings require the installation of a 1/2-inch-thick steel plate bulkhead slightly less than 48 inches square in the center of the opening in the concrete wall. Ten openings are cut in the bulkhead plate, and two stacks of five 18-1/2- by 5-1/2-inch steel sleeves are welded into the openings. The steel sleeves are 6 inches long and extend 3 inches on each side of the bulkhead centerline. The vertical clearance between the sleeves is 4 inches, and the horizontal clearance is 5 inches. The steel bulkhead assembly is framed and attached to the wall inside the concrete opening by 1-1/2- by 1-1/2- by 1/4-inch mounting angles. The cable trays abut the wall and are secured to angle iron extending horizontally across the face of the wall. Only the cables extend through the wall penetration. (See figures 2 and 3.)

The design requires that the penetration sleeves, with the cables installed, be filled with polyurethane foam to create an air pressure seal. (See figure 3.) A flameproofing compound, Flamemastic 71A, was specified to be applied 1/8 to 1/4 inch thick over the foam and the cables on both sides of the bulkhead for a distance of 12 inches to form a fire stop.

Field tests were conducted on a typical cable penetration at the site in 1973. Later a test sample was sent to the TVA Singleton laboratory for fire performance testing. A DED staff engineer evaluated the test data and approved the results.

b. Wall Penetration as Originally Constructed

To facilitate sealing of the penetrations and to provide a practical starting point for filling the space around the cables with polyurethane foam, a means of forming a dam is required to prevent the liquid foam from flowing out of the sleeves. A preformed, resilient polyurethane foam was cut to size for insertion into the sleeve opening to form a dam. Other materials, such as styrofoam, were also used in some instances as a back dam. Pourable polyurethane foam was applied over and around the installed cables; after hardening of the pourable polyurethane foam, sprayable Froth Pak Insta-Foam polyurethane was used to finish filling the sleeve. The pourable foam is used since it more easily fills the voids between the cables. The sleeve and 12 inches of cables on both sides of the penetration were then coated with Flamemastic to provide the fire stop. The steel bulkhead as constructed was mounted in the opening with the centerline 3 inches from the surface of the wall on the reactor building side and 23 inches from the surface of the wall on the spreading room side, as indicated by dimensions on figure 4. Materials in addition to polyurethane foam were used to form the pressure seal.

3. <u>Materials Used in Penetrations</u>

Materials used for construction of fire stops, air pressure seals, and resealing after modifications to penetrations are described on table 1.

Diligent efforts are being made to secure from the manufacturers the physical and chemical properties of the materials in table 1, items 1-8, and will be made available if received.

For small leaks in cable penetrations, RTV silicone rubber was typically used as a sealant. For larger leaks, resilient polyurethane foam was typically used as a dam or a plug to contain the RTV silicone rubber or polyurethane foam.

4. <u>Status of Penetration at Time of Fire</u>

The penetration in which the fire started had been originally sealed with polyurethane foam. There is evidence that the penetration had originally been coated with Flamemastic on the spreading room side. An examination after the fire indicates that Flamemastic had been applied to the unit 1 reactor building side of the penetration at some time prior to the fire and modifications which made resealing necessary.

Additional cables had been pulled through the penetration since initial installation. In order to make an opening for additional cables through the penetration, holes were punched with a wooden stick similar to a broom handle. This resulted in breaching any flameproofing that had been applied. This process usually resulted in pieces of polyurethane and Flamemastic in the penetration being knocked onto the cables on both sides of the penetration. This procedure has been generally followed when additional cables are pulled through completed penetrations. Fragments of these materials were observed on the cables in a number of other trays adjacent to the penetrations.

C. Activities Preceding the Fire

The areas within the plant are designed such that the air movement from one plant area to another will always be toward the area of possible higher radiation. This is controlled by supply and exhaust fans. The area of the reactor building and refueling floor (secondary containment) is the area of lowest pressure, and any leakage between secondary containment and other plant areas will be inleakage into the secondary containment.

Under certain conditions, the standby gas-treatment system must exhaust air from the reactor building to maintain a negative pressure. In order not to exceed the capacity of the system, inleakage to the reactor building must be kept at a minimum.

In the completed plant, the refueling zone is common for all three reactor units. During construction an airtight partition is required between operating units and those under construction; and one exists between operating units 1 and 2, and unit 3 which is under construction. Before this partition between units 2 and 3 could be removed, it was necessary to ascertain the degree to which the standby gas-treatment system could handle the added inleakage from the unit 3 reactor building. The Division of Power Production (DPP) was requested to run leakage tests on the units 1 and 2 reactor buildings. The results of those tests indicated that leakage had to be reduced to a minimum if the unit 3 reactor building could be included and inleakage remain within the requirements of the units 1 and 2 technical specifications.

In a program to reduce leakage, the Division of Construction (DEC) wrote workplan 2892. The plan required (1) that all leaks be identified and listed, (2) that leaks be sealed, and (3) that work be verified and signed off by an engineer.

The method for detecting air leaks was largely left to the discretion of the engineer in charge. Several methods have been employed at Browns Ferry. These include smoke devices, soap solutions, and candles. The movement of the flame of a candle was an effective method in locating leaks in dimly lighted areas and generally was the method used.

A list was made of all leaking penetrations. These were identified by elevation and wall location, cable tray identification, and conduit number. The list was given to the electrical craft supervision with a requirement for the foreman to sign off for completed items.

Checking the resealed penetrations was basically the same as inspecting for leaks. However, experience had shown that as the number of leaks was reduced, the differential pressure increased; and other penetrations that originally did not seem to leak began to show airflow. Therefore, the inspectors (engineering aides) were instructed to check all penetrations in their assigned areas. The inspectors were accompanied by electricians who sealed any leaking penetrations as they were discovered. The inspectors often aided the electricians by checking penetrations as they were being sealed.

A successful leakage test and its documented approval were considered as evidence of the pressure seal's integrity.

For production efficiency, application of the Flamemastic did not immediately follow the sealing activities but was applied at intervals when sufficient numbers of seals were made ready.

On March 22, 1975, DEC workers were in the spreading room, sealing and leak-testing cable penetrations between the cable spreading room and the reactor building, when (at approximately 1220 hours--all times are Central Daylight Time) some of the sealant material in the penetration was unintentionally ignited at cable tray VE.

D. Fire

1. Spreading Room Area

a. Sequence of Events

Six men were working in the units 1 and 2 cable spreading room, checking conduit and cable penetrations for air leaks and sealing leaks.

An engineering aide and an electrician were checking cable penetrations through the wall between the spreading room and the unit 1 reactor building, in a window containing 10 cable trays in 2 vertical rows of 5 trays.

The engineering aide was using a candle flame to detect air leaks.

A differential air pressure existed between the spreading room and the reactor building, with the reactor building having a slightly negative pressure and thus causing air to flow from the spreading room through leaks into the reactor building.

The aide detected a strong air leak in the penetration for the second tray from the bottom on the west row.

The leak was caused when additional cables were pulled through the penetration, which resulted in breaching the originally installed air pressure seal and fire stop.

The electrician could not reach the penetration since it was recessed into the wall farther than he could reach.

The aide volunteered to seal the leak for the electrician. The electrician handed the aide two pieces (about 2 inches by 2 inches by 4 inches) of resilient polyurethane foam which the aide inserted into the hole.

After inserting the resilient polyurethane foam into the leak, the aide placed the candle about 1 inch from the resilient polyurethane foam.

The airflow through the leak pulled the candle flame into the resilient polyurethane foam, which sizzled and began to burn.

The aide immediately told the electrician that the candle had started a fire.

The electrician handed the aide a flashlight, which was used to try to beat out the fire with no success.

Another construction worker heard the aide state that there was a fire and gave the aide some rags to use to smother the fire, which was also unsuccessful.

The electrician called for fire extinguishers.

When the rags were pulled away from the penetration, they were smoldering.

Meanwhile, the other worker brought a CO_2 fire extinguisher to the aide.

The fire burned for about 1-1/2 minutes before the first extinguisher arrived.

The entire contents of this CO_2 extinguisher was emptied on the fire. The fire appeared to be out.

About 1/2 to 1 minute later, the fire started up again.

The aide stated that the fire was now on the reactor building side of the wall.

Two construction workers left the spreading room for the reactor building to fight the fire.

The electrician took two fire extinguishers to the aide who remained in the spreading room. Each extinguisher gave only one good puff.

When the aide received the third extinguisher, he heard a fire extinguisher being discharged on the reactor building side of the wall.

As the aide prepared to discharge the fourth extinguisher, the spreading room CO_2 system alarm was sounded; and all workers evacuated the spreading room.

A plant operator, assistant shift engineer (ASE), after ensuring that no workers were in the spreading room, attempted to initiate the spreading room fixed CO_2 system from outside the west door to the room but was unable to do so because it had been deenergized while workmen were in the spreading room.

The ASE then ran to the east door of the spreading room, where he restored the electrical power and initiated the CO_2 system, which then operated properly.

Another ASE later operated the CO_2 system a second time.

After the CO_2 system had been operated the second time, the first ASE checked the spreading room and found that the fire had restarted.

He then directed the fire brigade in fighting the fire in the spreading room.

At 1310 hours, the ASE in charge of the reactor building fire requested the Athens Fire Department to come to the plant.

Employees from the Athens Fire Department assisted in fighting the spreading room fire.

The spreading room CO_2 system was operated one additional time.

An off-duty shift engineer (SE) arrived about 1500 hours and took charge of firefighting in the spreading room and relieved the ASE.

The spreading room fire was extinguished between 1600 hours and 1630 hours, primarily by using dry chemicals.

b. <u>Description of Fire in the Spreading Room</u>

The material ignited by the candle flame was resilient polyurethane foam.

Once the foam was ignited, the flame spread very rapidly.

After the first application of the CO_2, the fire had spread through to the reactor building side of the penetration.

Once ignited, the resilient polyurethane foam splattered as it burned.

After the second extinguisher was applied, there was a roaring sound from the fire and a blowtorch effect due to the airflow through the penetration.

The airflow through the penetration pulled the material from discharging fire extinguishers through the penetration into the reactor building.

Dry chemicals would extinguish flames, but the flame would start back up.

c. <u>Equipment</u>

Portable CO_2 and dry-chemical fire extinguishers were used in the spreading room fire.

The spreading room fixed CO_2 system was activated three times.

Breathing apparatus (air packs) received limited use in the spreading room.

The doors to the spreading room were kept open most of the time to assist in keeping smoke out of the control room.

An inplant fire hose was run from an outlet in the turbine building to the spreading room. This was not used.

The Athens Fire Department made available in the spreading room about 5 gallons of an agent which, when combined with water, forms "light water." This was not used.

Athens Fire Department employees discussed with the SE the possibility of using water on the fire in the spreading room.

No water was used in the spreading room since there was no assurance that the cables were deenergized.

d. <u>Time of Events</u>

(Approximate times shown with ~)

~ 1220	Fire started in penetration
~ 1230	Two construction workers leave spreading room for reactor building
1235	Plant fire alarm sounded. Fire logged in SE's log
~ 1237	First fire extinguisher discharged in reactor building
~ 1240	CO_2 alarm sounded in spreading room; CO_2 system operated
?	Spreading room CO_2 system operated second time

?	ASE assumes direction of fire brigade in fighting fire
?	Spreading room CO_2 system operated third time
∿ 1500	SE assumes charge of spreading room firefighting
∿ 1600-1630	Spreading room fire extinguished

e. Reporting the Fire

Two construction workers left the spreading room at about 1230 hours to go to the reactor building to fight the fire.

One worker stopped at post 8D, a construction portal manned by the Public Safety Service (PSS), and informed the public safety officer on duty that there was a fire in reactor building number 1 and took the fire extinguisher with him to use in fighting the fire.

The officer immediately called the SE and reported a fire in unit 1 reactor building.

The ASE who received the fire report immediately gave the message to the SE and the unit 1 operator and then proceeded to the control room and switched the fire alarm to assure continuous sounding.

The unit operator (UO) immediately began to announce over the PA system that there was a fire in the unit 1 reactor building.

At this time, operators in the control room did not know the exact location of the fire.

An ASE located the fire in the unit 1 reactor building shortly after the construction workers had begun to fight it there. He telephoned the exact location to the operators in the control room.

Shortly thereafter another ASE in the reactor building reported the spreading room fire to the operators in the control room.

2. Reactor Building Area

a. Sequence of Events

When workers in the spreading room saw that the fire had spread into the reactor building, two construction workers left the spreading room and proceeded to the reactor building to fight the fire.

One worker told the public safety officer at post 8D that there was a fire in the reactor building and took a fire extinguisher with him. The other construction worker proceeded to the reactor building where he met a third worker; each of the three workers took a fire extinguisher to the fire.

All three workers arrived at the fire at about the same time. It was burning in the trays which were 20 feet above the second floor of the reactor building. One moved a ladder, already at the scene, next to the fire. Another worker climbed the ladder and discharged a dry-chemical extinguisher on the fire. This application knocked down the flames, but the fire flared up again.

One of the workers alerted other workers on the second level of the unit 1 reactor building of the fire.

The worker who applied the first extinguisher was affected by the smoke and fumes around the cable trays at the top of the ladder.

The unit 1 control room operator was informed by telephone of the precise location of the fire by a plant operator on the scene.

An ASE then arrived and, along with another operator, discharged a CO_2 and a dry-chemical extinguisher simultaneously on the fire. The ASE assumed charge of firefighting activities. Construction workers were instructed to leave the operating units.

Smoke was becoming so dense that breathing apparatus was required; approximately 5 minutes after it was requested, it was available. Until it arrived, CO_2 was applied to the cable trays from the floor.

After the breathing apparatus (air packs) arrived, it was utilized in fighting the fire until visibility became so bad that the workers could not get near the fire. The smoke backed them up to the area of the reactor building closed cooling water system heat exchangers.

The ASE left the fire to assist in unit shutdown. An assistant unit operator (AUO) assumed charge of firefighting activities. The first floor of the reactor building was also evacuated. The AUO went to the control room due to some ill effects of the smoke. Another ASE assumed charge of firefighting activities.

Power to the elevator was lost. The second floor of the reactor building was then evacuated. Some time was utilized to check 5 floors of the reactor building for the elevator to ensure that no one was trapped on the elevator. A head count was made, and from that point on a count was kept of all personnel leaving and entering the reactor building.

About 1330 hours, lighting was lost in the reactor building.

Limited firefighting was resumed in the reactor building for a period between 1430 hours and 1500 hours. A wire was used to rig a guideline. At this time the fire was still confined to the area in the cable trays near the north wall and had not proceeded very far on the south trays.

At this time, the doors between units 1 and 2 were opened, which improved visibility on the second level of unit 1 to about 5 feet.

At about 1630 hours, the SE who had been directing activities in the spreading room took charge of firefighting in the reactor building in order to concentrate activities there. The SE consulted the plant superintendent frequently during fighting of the reactor building fire.

On inspection of the fire at 1630 hours, the major fire was in the cable trays running south from the penetration, with a smaller fire in the cable trays running west from the penetration.

The SE established a routine of sending 2 to 3 people in at a time to fight the fire, using dry chemicals primarily.

Shortly after 1630 hours, temporary d.c. lighting was strung on the second level of unit 1.

A rope was utilized as a guideline, which assisted employees from the Athens Fire Department in approaching the fire to inspect it. The SE went into the vicinity of the fire between 1730 hours and 1800 hours.

On one of his trips into the second level, the SE laid out the fire hose installed there and checked to ensure that water was available. The plant superintendent authorized the use of water as an emergency backup, for example, in case a worker's clothing caught fire. Otherwise, there was a decision not to use water on the fire due to the electrical shock hazard. The Athens fire chief suggested that water would be the best thing to use on the fire if it could be used.

The SE suggested to the plant superintendent that water be used on the fire. The superintendent made the decision to allow the Athens Fire Department employees to use water on the fire.

Water was initially applied to the trays running west; however, from the floor level, the water would effectively reach only the bottom tray. Athens Fire Department employees attempted to utilize one of their nozzles on the hose, but the thread did not match; and the nozzle came off when pressure was applied.

Water was also applied to the fire in the cable trays along the north wall and successfully extinguished it.

Firefighters began using Chemox respirators as the supply of compressed air for the air packs ran low.

The SE and two other operations workers entered the area of the fire to utilize water to fight the fire. The SE took the hose and climbed within four feet of the fire with assistance of the other two men. He sprayed water on the fire in the south cable trays for approximately 10 seconds, which extinguished the fire.

The fire hose was left stuck in a position so that it continued to apply water to the south cable trays.

The second level was entered again and water reapplied. It was then determined that the fire was out. There were subsequently some reports of sparks, but investigation failed to reveal any further fire.

During the course of the fire, it was noticed that a small diameter station control air line under about 90 pounds of pressure, running along the north wall, had parted. The line was later isolated.

Several fire extinguishers were discharged early in the the fire from the third floor through an opening in the floor, but all missed the fire in the cable trays since the opening was not directly over the fire.

b. <u>Description of Fire in Reactor Building</u>

The fire was initially observed in the lower cable trays, extending out from the penetration a distance of 2 to 4 feet. Height of the flames varied from a few inches to a few feet, dying down as extinguishing materials were applied and flaring up between applications. The flames were coming straight up.

Some polyurethane foam was flowing from the penetrations into the trays, and bright yellow flames were coming from the penetrations.

The fire did not advance significantly into the south trays until after 1500 hours.

Scaffold boards had been previously placed below the trays in the unit 1 reactor building, near the cable tray penetration where the fire started. These boards were used to work from in pulling cables through the penetration. These boards were charred by the fire. The charring did not extend to the side away from the fire, indicating little influence as fuel for the fire.

c. Equipment

Portable CO_2 and dry-chemical fire extinguishers were used in the reactor building fire.

MSA air packs were used that had a rating of 30 minutes for moderately heavy activity of the user. A cascade system of large air cylinders was available for charging the packs, but the supply was eventually depleted. There are no air compressor facilities at the plant to fully recharge the air packs. The charges in some air packs did not last 30 minutes. Air packs from Athens Fire Department were also used along with their recharging facilities on their truck and at their station in Athens.

MSA Chemox respirators were used. Several users experienced difficulty when using these for very strenuous activity.

The fire hose and nozzle provided in the second level of the reactor building functioned properly and successfully extinguished the fire.

A nozzle from the Athens fire truck did not fit the threads on the hose on the second floor of the reactor building.

Ladders present on the second level of the reactor building were utilized.

Temporary d.c. lighting was utilized.

A wire and a rope were utilized as guidelines.

A fire hose was laid out on the third floor of the reactor building but was not utilized.

d. Time of Events

~ 1230	Two construction workers leave spreading room for reactor building
~ 1237	First fire extinguisher discharged in reactor building
~ 1240	Unit operator informed of exact location of fire in reactor building
?	Air packs requested and received
~ 1310	ASE requested that Athens Fire Department come to the plant
~ 1330	Lighting lost in reactor building
~ 1645	Temporary d.c. lighting installed
~ 1835	Water applied to fire
~ 1930	Fire determined extinguished

e. Minor Fires on Thrusday, March 20

There were two minor fires on Thussday, March 20, arising from the use of candles for leak-testing in electrical cable penetrations different from the penetration involved in the March 22, 1975, fire. In the first fire, the candle flame ignited some RTV silicone rubber. The construction worker using the candle extinguished the flareup with his fingers.

In the second fire, the candle flame ignited dust and debris in the cable tray. The fire lasted about 30 seconds and was extinguished with a discharge from a CO_2 extinguisher.

The first fire was reported orally to construction supervisory workers; the second fire was entered in the SE's log and reported in writing to construction supervisory workers.

There was no damage from either fire.

E. Effect on Plant Systems and Operations

1. Status of Plant Operations Prior to Fire

At the time of the fire on March 22, 1975, units 1 and 2 were each producing approximately 1,100 MWe gross. Unit 1 was declared in commercial operation on August 1, 1974, and unit 2 on March 1, 1975.

2. Unit 1

The ignition of the fire in the cable penetration has been established as accurately as possible to have occurred at 1220 hours on March 22, 1975. The first indication of its effect on unit 1 operation came 20 minutes later, at 1240 hours. This was 5 minutes after the UO's were notified of the fire and the alarm initiated at 1235 hours.

The first effect on the unit was almost simultaneous annunciation of several events: residual heat removal (RHR) or core spray (CS) automatic blowdown permissive, reactor water level low-automatic blowdown permissive, and core cooling system/diesel initiate.

At this point the UO observed that normal conditions of reactor water level, reactor steam pressure, and drywell atmosphere pressure existed.

Over the next 7 to 8 minutes, a mounting number of events occurred, including the automatic starting of RHR and CS pumps, high-pressure coolant-injection (HPCI) pump, and reactor core isolation coolant (RCIC) pump; control board indicating lights were randomly glowing brightly, dimming, and going out; numerous alarms occurring; and smoke coming from beneath panel 9-3, which is the control panel for emergency core cooling systems (ECCS). The operator shut down equipment that he determined was not needed, such as the RHR and CS pumps, only to have them restart again.

When the reactor power became affected by an unexplained runback of the reactor recirculating pumps, the SE instructed the operator to reduce recirculating pump loading and scram the reactor. While this was being done, the recirculating pumps tripped off. The reactor was scrammed by the operator at 1251 hours.

The turbogenerator was then removed from service; steam from the reactor was bypassed around the turbine to use the condenser as a heat sink; and unneeded condensate, condensate booster, and reactor feed pumps were removed from service. One of each pump was left running to maintain reactor water level. Beginning at approximately 1255 hours and continuing for about 5 minutes, several electrical boards were lost, supplying control voltages and power voltages of 120, 480, and 4,160 volts a.c. and 250 volts d.c. These mainly affected reactor shutdown equipment.

As a result of the loss of these electrical boards and previous effects, many of the systems used in cooling the reactor after it is shut down became inoperative. This included the RHR system, core spray system, HPCI, and RCIC. This is attributed to loss of valve control signals, valve power voltage, motor control signals, motor power voltage, or a combination of these. In addition, many of the instruments and indicating lights were put out of order. Also, the outboard main-steam isolation valves (MSIV's) closed. This isolated the steam generated by reactor decay heat from the condenser heat sink. The valve closure also isolated the steam supply to the turbine-driven reactor feed pumps, and consequently this high-pressure source of water to the reactor was lost. At this time the water input to the reactor was limited to the control rod drive pumps as a high-pressure water source since the steam pressure built to a pressure of 1,080 psi and was being relieved by automatic operation of the relief valves to the suppression pool.

Alternative systems were available and were used effectively to shut down and cool the reactor. This was accomplished by manual opening of the relief valves to reduce reactor pressure below 350 psi where the condensate booster pumps could pump an adequate supply of water to the reactor. The reactor water level decreased during this operation, but it did not drop below a point 48 inches above the top of the active fuel and was returned to normal level by 1345 hours.

Early in the chain of events, the diesel generators started and were allowed to run on standby. During a short period of time the four diesel generators were used to supply their respective shutdown buses. About 1443 hours one of the diesel generators became unavailable.

Soon after the loss of electrical boards, operating workers began attempts to restore the electrical supplies.

Initially, this was generally unsuccessful. Attempts to manually position valves and locally operate the equipment were hampered by darkness and the smoke and fumes from the fire filling the reactor building, requiring the use of air-breathing packs. Some smoke and CO_2 came into the units 1 and 2 control room from firefighting efforts in the spreading room, but it was not necessary to vacate the control room at any time. Two of the operators in the unit 1 control area donned breathing apparatus for a short period of time because of the smoke and fumes. To establish the electrical supply boards, maintenance electricians joined the operators in isolating faulted circuits in order that the boards could be reenergized. This was done over several hours, and needed equipment to provide suppression pool cooling and reactor long-term shutdown cooling was gradually made available.

With adequate electrical power, along with some manual valve alignment, the operators established suppression pool cooling at 0130 hours on March 23, 1975, 12 hours 39 minutes after the unit 1 reactor was scrammed. Normal reactor shutdown cooling was achieved at 0410 hours on March 23, 1975, 15 hours 19 minutes after the unit was scrammed.

3. Unit 2

Nine minutes after unit 1 was scrammed, abnormal events began occurring on unit 2. At 1300 hours the 4-kV shutdown bus 2 deenergized; and the operator observed decreasing reactor power, many scram alarms, and the loss of some indicating lights. The operator put the reactor in shutdown mode, and it scrammed at 1300 hours.

The turbine was immediately tripped, along with the reactor feed pumps. In approximately 4 minutes after scram, the MSIV's closed, isolating the reactor steam from the condenser heat sink and the reactor feed pumps steam supply. RCIC was immediately initiated for reactor water level control and the HPCI to aid as a heat sink for the steam being generated in the reactor by decay heat. These two systems tripped several times over the next hour, and at approximately 1345 hours HPCI became unavailable. RCIC continued to run and supply high-pressure water to the reactor.

When suppression pool temperature began to increase from relief valve steam heating, RHR suppression pool cooling was established at 1320 hours; and the temperature of the water in the torus did not exceed 135° F.

When the MSIV's closed, reactor pressure was relieved by manual operation of the relief valves. Manual operation of the relief valves was lost at 1320 hours and the relief valves lifted intermittently on pressure until 1415 hours, when manual operation was restored; and the reactor was depressurized by use of the relief valves.

At 2010 hours the MSIV's were reopened, making the condenser heat sink available. At 2020 hours on March 22, 1975, equipment was made available to establish operation of the RHR system to be used for reactor long-term shutdown cooling. This was 7 hours 20 minutes after the unit was scrammed.

4. Detailed Operating Events, Operator Action, and Equipment Response and Nonresponse

Tables 6 and 7 provide the sequence of events, operator action, and equipment response which occurred during the fire and until conditions were stabilized (initiation of shutdown cooling) on both units 1 and 2. The events listed on tables 6 and 7 are arranged chronologically, with the best possible establishment of times without the benefit of complete operator logs.

Most of the time, particularly during the early stages of the fire, operators were too busy to log the frequent events and actions. Some of the times and facts were established by charts and printers but for the most part by interviews with operating personnel, both individually and in groups.

5. <u>Status of Major Plant Equipment and Systems and Plant Parameters at the Initiation of Reactor Long-Term Shutdown Cooling</u>

a. <u>Unit 1 at 0410 hours on March 23, 1975</u>

Reactor coolant temperature 360° F

Reactor vessel water level normal

Suppression pool water level +5"

Suppression pool water temperature 153° F

Control rod drive pump and condensate pumps providing makeup water to reactor vessel

Standby liquid control system available

Core neutron monitoring provided by two temporary source range monitors connected outside primary containment with the monitors manned by a licensed reactor operator in communication with a licensed reactor operator in the control room

Primary and secondary containment integrity being maintained

All 4-kV shutdown boards available

Shutdown bus 2 available and supplying offsite power to the shutdown boards

Remote indications (amps, watts, and volts) being read locally at shutdown boards where equipment operation required

Diesel generators A, B, and D available and operable from shutdown boards—diesel generator C unavailable because of control cable problems

RHR loop I pumps and valves available

RHR loop II pump B and valves available

Control for 3 RHR pumps available from control room; control from local stations for most valves

All loop I and loop II core spray pumps and valves available

Four relief valves remotely operable from unit control board

No automatic initiation of diesel generators, core spray system, or RHR system in low-pressure coolant-injection (LPCI) mode available

Suppression pool cooling in service

Suppression pool water level indication and drywell pressure indication operable

Train A of standby gas-treatment system operable

Control rod drive pump in operation--system flow and pressure indication unavailable

Process computer in service (40 analog inputs damaged by fire)

Telephone communication out of service for unit 1 reactor building, offgas vent building, and stack; in service for other areas

Liquid monitor on the effluent from the reactor building closed cooling water system, raw cooling water, and residual heat-removal heat exchangers out of service.

Grab samples of effluent water taken periodically by chemical laboratory personnel.

b. Unit 2 at 2240 hours on March 22, 1975

Reactor coolant temperature 260° F

Reactor vessel water level normal

Control rod drive and condensate pumps providing makeup water to reactor vessel

All RHR pumps operable

HPCI pump inoperable

Core spray loop I pumps A and C and RHR loop I pumps A and C operable only from shutdown boards

Conditions of long-term reactor shutdown cooling were considered normal

F. Damage Assessment (Cable Tray System, Conduit and Grounding System, and all Cables Routed Through These Raceway Systems)

This section summarizes the extent of the physical damage to the cables and the raceway systems involved in the fire at Browns Ferry on March 22, 1975, and indicates the detail to be found in a complete report provided by DED for use in the restoration program. The complete report is numbered BF-DED(BHP-1).

Excluded from the damage assessment are the effects of faults in these cables to mechanical and electrical systems; damage to other equipment resulting from products of combustion and the chemicals and water used in extinguishing the fire; possible structural and concrete damage; and damage outside the zone of influence of the fire. These areas are being evaluated in detail by others within TVA.

A fire consultant has been retained by DED to perform a thorough investigation with the purpose of providing a factually accurate and professional determination or assessment of the mechanisms and their interactions responsible for the initiation, propagation, magnitude, duration, and extent of damage of the fire. The consultant's report has not been received at the time of issuance of this report.

1. <u>Zone of Influence of the Fire</u>

It has been determined that the fire started when an open flame came into contact with material used as the seal around the cables where they penetrate the wall between the units 1 and 2 control bay spreading room and the unit 1 reactor building. Figures 5 and 6 indicate the area being considered in this description. Figure 6 shows the zone of influence of the fire. Figure 7 shows a cross section of trays near the point where the fire started. The cables and raceways in the spreading room were damaged approximately 5 feet north of the wall penetration; and the fire propagated along all trays, as marked on figure 8, in the reactor building on floor elevation 593. Many photographs were taken, and 10 key ones are included in this report as appendix B. Figure 9 shows affected trays and their intersections in single-line representation. Checkpoints used for routing cables on each cable tray are also shown. (See table 2 for loading of cable types onto each tray at each checkpoint.)

Visible damage in the reactor building was observed east along the double stack of 3 trays to the wall between units 1 and 2, south along the 4 trays to a fire stop approximately 28 feet from the wall between the reactor building and the control bay, and west along the double stack of 5 trays, for a distance of approximately 38 feet from the wall between units 1 and 2. Cables were also damaged on 2 of the 4 vertical trays from the top about 10 feet down, and cables in 1 of the other 2 trays were damaged about 4 feet down. Figures 10-12 show the zone of influence of the fire for all damaged or assumed-damaged conduits and grounding systems.

2. Identification of Damaged Conduits, Cable Trays, and Cables Routed Through Raceways

A total of 117 conduits and 26 cable trays was damaged by the fire, and it is assumed that all supports for the raceway system were also damaged. There was a total of 1,611 damaged cables, and these are tabulated on 204 cable tablulation sheets prepared by DED. Table 3 is a sample sheet of the 204 cable tabulation sheets which show the purpose of each cable and other pertinent information needed by DEC to be used in a procedure for identification and removal of damaged cables.

This procedure is being written by DEC to require that the damaged portion of each cable be identified and measured during its removal. This procedure will also require that a section of the undamaged portion of each cable be removed, identified, and stored for future reference. This section will be cut to assure that all manufacturer's data stamped on the outer jacket will be included in the sample.

As of this date there have been 1,169 cables identified as damaged for unit 1, 75 for unit 2, 27 for unit 3, and 340 common to plant. Of the total cables identified and listed in table 3, it was determined that a total of 628 safety-related cables was damaged. These are grouped into categories shown on table 4.

The bare ground cable used for grounding the cable tray system was also damaged by the fire. It was routed along the 480-volt power trays FM, FK, and FO-ESII through the zone of influence.

3. <u>Materials Available as Possible Fuel For the Fire</u>

Of the 1,611 cables, there were 65 different-type cables involved in the fire, as listed on table 5. Figure 7 shows a cross section of the cable trays where the fire started. (See table 2, sheets 8 and 9, for the type cables found there.) These types are representative of each voltage level tray in the area. Types WBB through WNF are power and control cables manufactured in accordance with TVA standard specification and are composed of insulating material footnoted on table 5, sheets, 2, 3, and 4. The remaining types are signal cables which are specified and documented on numerous individual contracts. These are composed of insulating material footnoted in table 5, sheets 2, 3, and 4. In all cases, the ctual types used will be verified in the removal of cables and will be included in the final DED report BF-DED(BHP-1). The filler materials in these cables and cable ties are included in the listing at the conclusion of this section.

Another possible "fuel" was the wall penetration pressure seal materials used between the spreading room and the reactor building. A typical penetration is shown in figures 2 and 3. The sealant material was polyurethane expandable foam, a pressure seal, which is covered with Flamemastic, a flameproofing compound. Another sealant material which is a possible fuel source would be the RTV silicone rubber compound used in sealing conduits through walls and in some cases to seal around new cables added through penetrations.

4. <u>General List of Materials Associated With the Fire</u>

   a. Candle

   b. Polyurethane foam, Froth Pak Insta-Foam

   c. Polyurethane, pourable type

   d. Polyethylene

e. Nylon

f. Cross-linked polyethylene

g. Polyvinyl-chloride

h. Mylar

i. Aluminum foil and rigid aluminum conduit

j. Polyolefins

k. Chlorosulfonated polyethylene

l. Neoprene

m. Fiberglass

n. RTV silicone rubber

o. Galvanizing material on raceways

p. Carbon

q. Thermoplastic nonhygroscopic cable filler material

r. Preformed, resilient polyurethane foam

s. Marinite panels

t. Styrofoam

u. Copper

v. Steel

w. Flamemastic 71A

G. Radiological Assessment

Based on interviews with the plant health physics supervisor and the plant chemical engineer, and information provided by the Plant Results Section and the Division of Environmental Planning, the following has been established.

1. Releases Within the Plant and Personnel Exposures

   a. At the time of the fire, one health physics technician was present at the facility. As requested, off-shift technicians reported to the plant, with the health physics supervisor arriving at approximately 1600 hours. At one time as many as 9 health physics workers were onsite.

   b. Direct radiation surveys conducted within the reactor building indicated there was no increase in direct radiation above normal levels.

   c. Numerous samples to detect airborne radioactivity present within the reactor buildings showed that the only significant particulate or halogen isotope present was the isotope Rubidium 88, a daughter product of the fission gas Krypton 88, with a half-life of 17 minutes. The buildup of Rubidium 88 is attributed to the shutdowns of the reactor building ventilation systems during the fire.

   d. Analyses of the samples showed the maximum concentration of this isotope approximated only 35 percent of the maximum concentration permitted under NRC regulations in 10CFR20 for a 40-hour workweek.

e. Following the fire, a number of individuals, including operations and construction workers, who were considered the most likely to have received internal radiation exposure from being in the unit 1 reactor building, were whole-body counted (on March 24 and 25). All whole-body counts showed no indication of internal deposition of radioactive material.

f. Based on dosimetry information, no plant individual is shown to have exceeded the daily radiation exposure limit; and the film badge readings for the Athens Fire Department employees indicated they received no detectable radiation exposure.

2. <u>Releases From the Plant</u>

a. As a result of the fire, the radiation detectors that monitor the ventilation air exhausted from the unit 1 and the unit 2 reactor buildings were made inoperable. The unit 2 monitor was restored at about 1900 hours on March 22, 1975, and the unit 1 monitor restored at 1600 hours on March 23, 1975.

b. During the course of the fire and the time the monitors were out of service, grab samples were taken from the units 1 and 2 exhausts on the reactor building roof starting at approximately 1645 hours and each hour thereafter and analyzed in the plant radiochemistry laboratory to determine concentrations of radioactivity. Charcoal filter and particulate filter samples were also taken from these airstreams periodically during the event.

c. All other required building ventilation duct monitors and the plant stack release monitors remained operable.

d. Gamma spectrum analysis of the grab samples indicated that the principal isotopes present were Xenon 133, Xenon 133m, Krypton 85m, and the Rubidium 88 detected in the inplant air samples. Analysis of the charcoal samples indicated no detectable amount of iodine.

e. Review of the airborne release rate information shows that the total plant release rate was the highest at 2200 hours on March 22 and corresponds to about 8 percent of the technical specification allowable limit for gross activity release.

f. Liquid radwaste is discharged from the plant periodically and on a batch basis. The last batch released before the fire occurred was on March 19. While as a direct result of the fire the liquid radwaste monitor became inoperable, no release from the plant was being made at the time; and the monitor was returned to operation on March 24 before the next batch was released.

3. Environmental Consequences

a. While not required, the Environs Radiological Emergency Plan was activated for precautionary purposes at approximately 1500 hours on March 22, with the Environs Emergency Staff remaining active until approximately 0500 hours on March 23.

b. A report on the radiological environmental consequences of the fire, made at the committee's request, is summarized below:

(1) Analyses of air particulate and charcoal filter samples collected by monitoring teams in the downwind direction from the plant, based on continual evaluation of data from the plant's meteorological station, show that no radioactivity except that due to naturally occurring radionuclides was detected in the environment.

(5) The report states that "Based on actual measurements and collected data, calculations showthat during the incident at the Browns Ferry NuclearPlant, amounts of radionuclides released to the environment were well below the plant technical specification limits. Conservative calculations show that the radioactivity released to the environment had a very minimal and insignificant environmental impact."

H. Personnel Injuries

Information provided by the TVA medical director states that 7 TVA employees (6 from DPP and 1 from DEC) reported to the Browns Ferry construction project medical office and the health station with complaints associated with smoke inhalation. Under the direction of a TVA physician, each was evaluated and treated by the nurses on duty and released with instructions to report immediately any delayed effects. Shortly after being seen, one of the employees reported the onset of generalized chest discomfort on respiration. He was referred immediately to a local hospital, where he was examined and released by the physician. None of the employees revealed evidence of severe effects from their exposure.

Followup medical evaluations revealed no residual effects from the activities and exposures associated with fighting the fire.

There has been no medical indication for lost time from work. Each employee was medically approved to resume full duties on the next scheduled work shift.

I. <u>Administrative Controls</u>

1. <u>DPP-DEC Interface for Work by Construction Forces in an Operating Unit</u>

a. Under DEC Quality Control Procedure BF-104, Administrative Procedures to Maintain Physical Separation Between Construction and Operating Units and Control of Work in Restricted Access Areas, all modifications and completion work required on a licensed unit by construction employees are done under a workplan. This procedure also specifies (1) that workplans can be written by either DEC or DPP, (2) must be approved by the DEC coordinator, and (3) the DPP coordinator will determine the level of review required within DPP and finalize approval with his signature.

b. BFNP Standard Practice BFA-28, Plant Modifications, describes how modifications to the plant will be requested, performed, and documented, including the approvals necessary, depending on whether the modification is categorized as safety related or nonsafety related.

c. The work being performed at the time the fire started was approved by the DEC coordinator and authorized by the DPP plant modification coordinator under BFNP workplan 2892 which was issued under BF-104 on March 7, 1975.

d. On workplan 2892, the work to be performed is described as follows: "Check electrical and mechanical sealing for secondary containment. (1) make a punch list of sleeves and cable penetrations that require sealing, (2) complete sealing, (3) verify and sign off areas that were found leaking."

A list of identified secondary containment air leaks is attached to the workplan.

e. The space provided for identification of drawings associated with the work has the letters N/A (not applicable) entered.

f. A review of workplan 2892 and applicable administrative procedures indicates the work being performed under this workplan was not processed as a modification under BFA-28 but was processed under BF-104 which does not require that an unreviewed safety question determination be made according to the provisions of 10CFR50.59.

2. Construction Work Control

With regard to the control of the work being performed by construction forces, the committee established the following:

a. There were no written procedures or work instructions covering the sealing and testing of penetrations for the original installation or the modifications except for notations on DED drawings.

b. At the time the fire started, the engineering aide whose assigned responsibility was to inspect the work (i.e., to find the air leaks) was actually doing the work himself (i.e., plugging the leaks) instead of the journeyman electrician.

3. Fire Reporting

a. The existence of a fire was not reported immediately by construction workers discovering the fire. When reported to the PSS officer manning construction portal post 8D, the exact location of the fire was not specified.

b. BFNP Standard Practice BFS3, Fire Protection and Prevention, instructs DPP personnel discovering a fire, whether in a construction area or an area for which DPP is responsible, to report the fire to the construction fire department, telephone 235. BFNP Fire, Explosion, and Natural Disaster

Plan instructs personnel discovering a fire to dial 299 (PAX). The construction extension cannot be dialed from the PAX system, and the plant extension cannot be dialed from the construction phone system.

c. Dialing instructions for reporting fires are located on telephones and are also included on the emergency procedure sheet posted at various locations in the operating areas.

4. Work Hazards Control

While control requirements exist for certain potentially hazardous work, e.g., welding and burning operations, no written procedures or instructions have been issued at Browns Ferry regarding the introduction into and use of potentially hazardous materials or substances in connection with construction work in operating plant areas such as ignition sources and flammables.

J. Other Findings

The possibility of sabotage was investigated, and no reason to suspect sabotage was found.

IV. OTHER GENERAL INFORMATION

A. Central Emergency Control Center (CECC)

1. The CECC was activated on March 22, 1975, during the Browns Ferry fire as a precautionary measure, although no

radiological emergency existed. The CECC was directed from the Edney Building in Chattanooga, beginning at 1525 hours CDT on March 22, 1975, by the Assistant to the Director of Environmental Planning. Other available members of the CECC were notified of the fire.

2. The CECC performed a valuable function—keeping the Nuclear Regulatory Commission (in Atlanta), the Alabama State Department of Public Health, and the Tennessee State Department of Public Health informed rather than fulfilling a requirement of the Radiological Emergency Plan (REP). The CECC was in direct communication with the DPP Emergency Control Center.

3. The CECC office was secured at 2230 hours on March 22, 1975.

B. DPP Emergency Control Center

1. The DPP Emergency Control Center in Chattanooga was established at 1510 hours on March 22, 1975, with the Chief, Nuclear Generation Branch, in charge. By 1630 hours, approximately 20 DPP staff members had assembled at the control center, including the division director and other key management personnel. The branch chief and others were in frequent communication with the superintendent at Browns Ferry. This management team participated in all major decisions associated with the plant operation and firefighting activities.

2. The major group of the staff assembled left at 2200 hours on March 22, 1975. A small group manned the DPP Emergency Control Center until 1500 hours on March 23, 1975.

C. Other Programs for Repair and Return to Service of Equipment

A number of programs have been initiated to evaluate various aspects of the fire and its consequence and return to service of the equipment. A memorandum from E. F. Thomas to R. H. Dunham and H. H. Mull dated March 28, 1975, subject "Repair of Damage Caused by the Cable Fire and Return to Service of Browns Ferry Nuclear Plant Units 1 and 2" has been issued and is being updated to provide directions for these efforts.

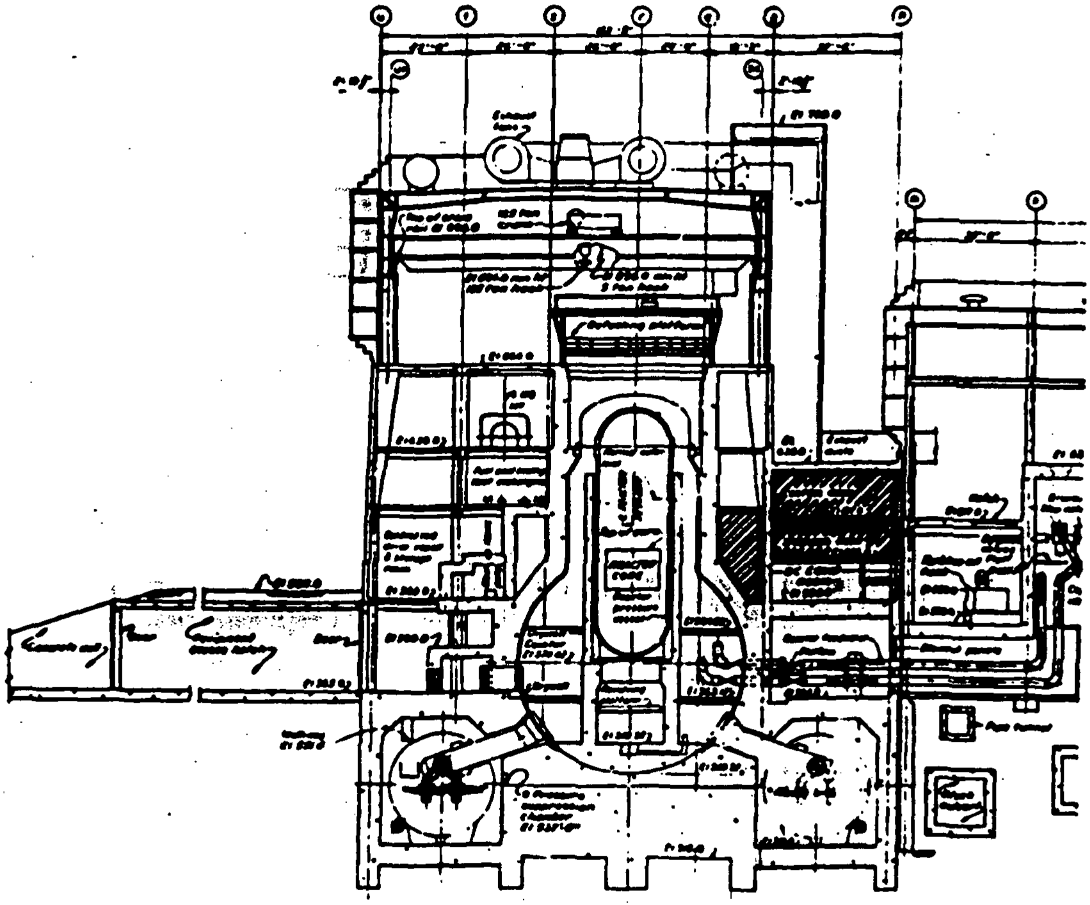
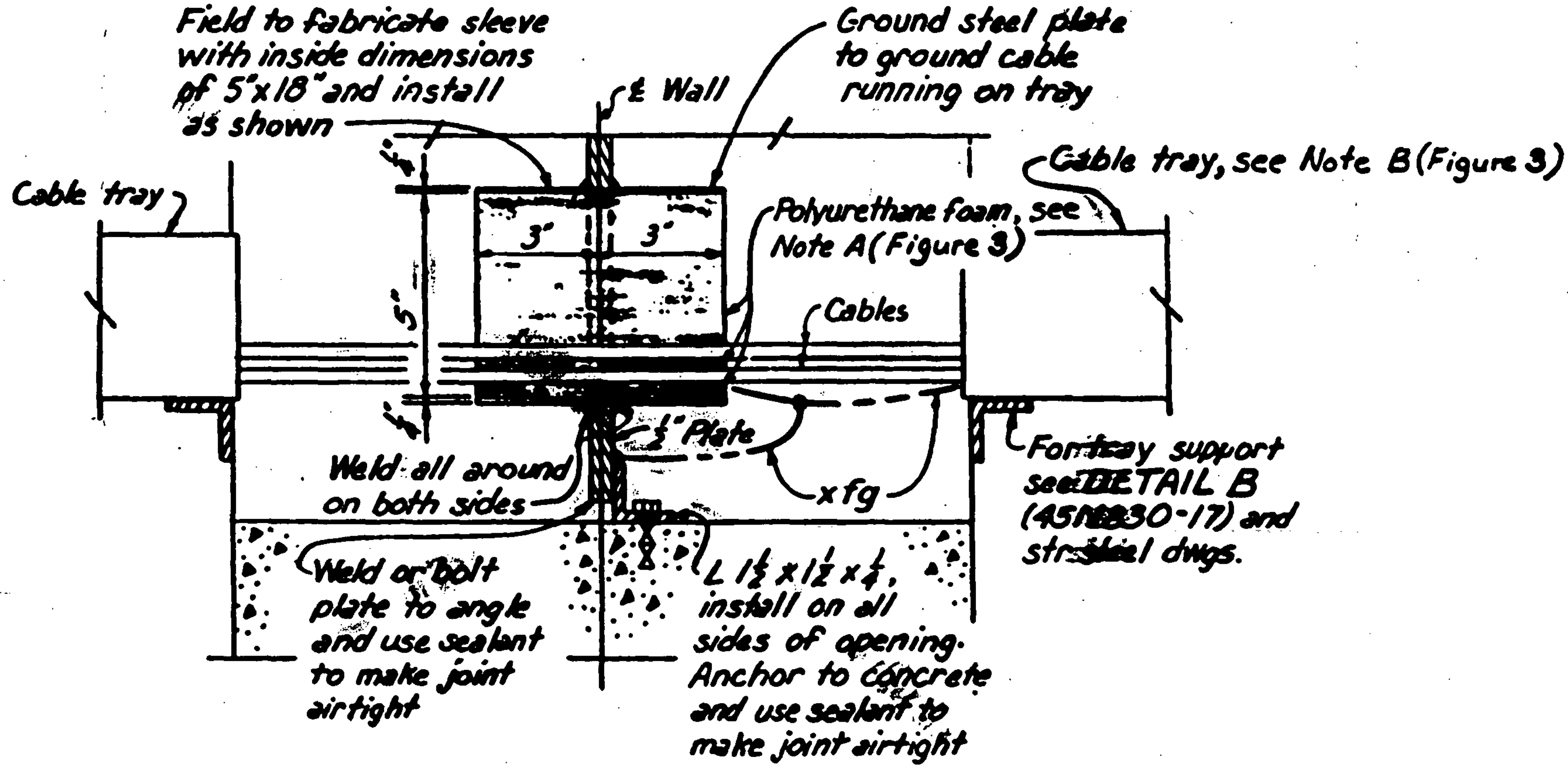


FIGURE 1
Vertical Cross Section

Reactor Building, Control Room,
and Spreading Room

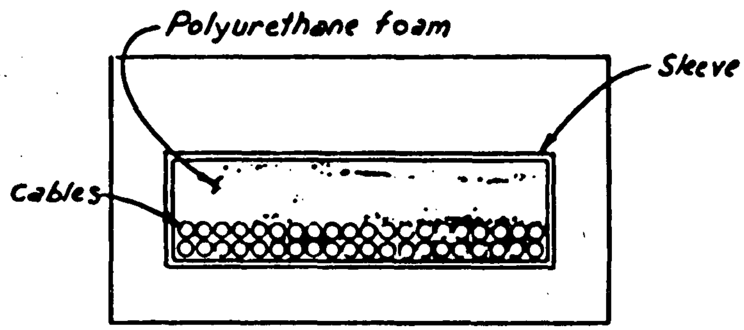
FIG. 1



SIDE VIEW
Scale: 3"=1'-0"

FIGURE 2
TYPICAL WALL PENETRATION

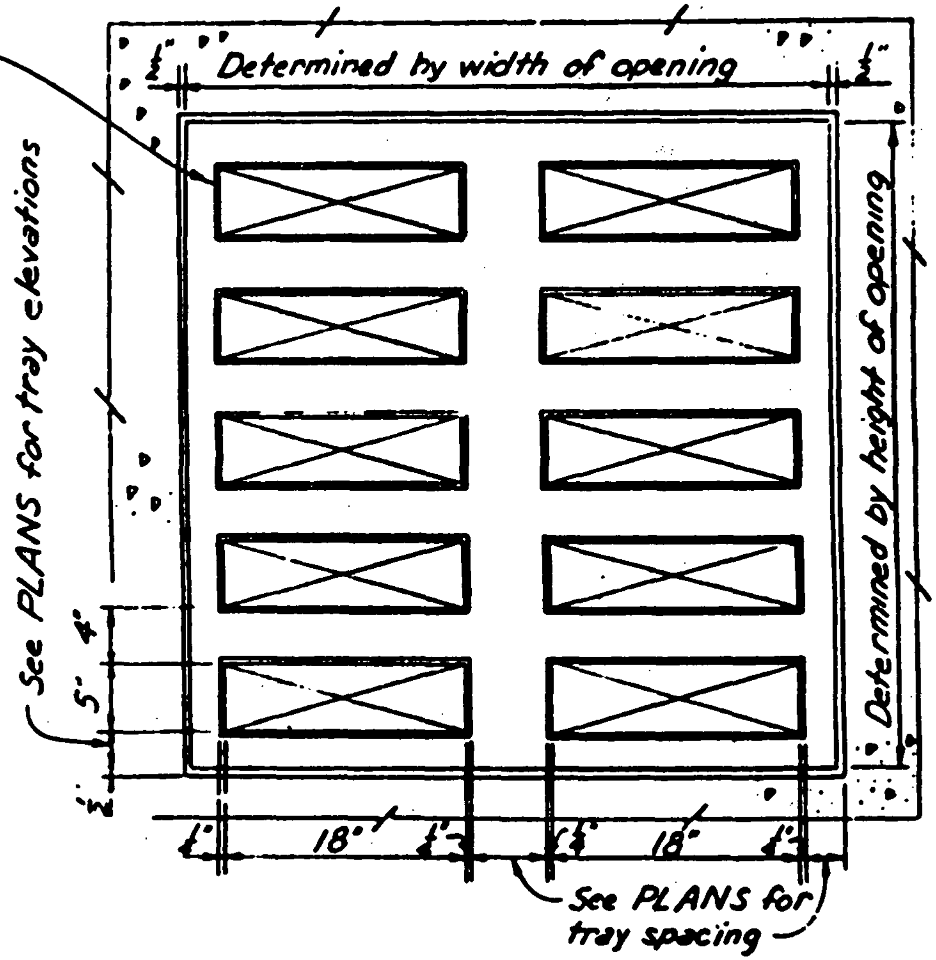
FIG. 2



NOTES:

A. When all cables have been installed through cable sleeve, seal the remaining opening and voids with 6" ± of Urethane foam or equal before applying flame proofing compound, Flamemastic 71A or equal.

B. Apply approximately 1/8" to 1/4" of flame proofing compound on the steel sleeve andon both top and bottom of the tray and cables for 12" on both sides of the barrier.

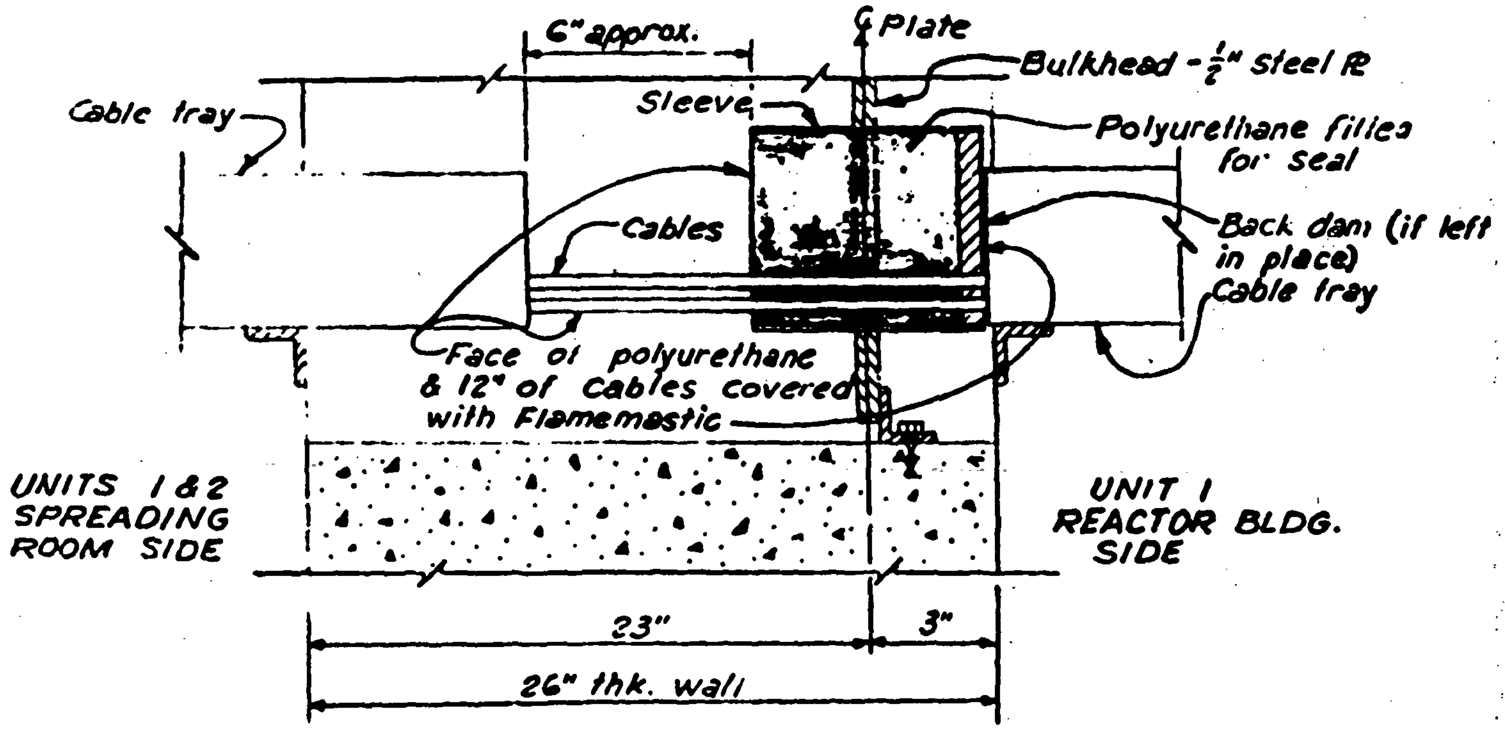


FRONT ELEVATION
Scale: 1"=1'-0"

FIGURE 3
TYPICAL WALL PENETRATION

FIG. 3

NOTE: FIRE STARTED IN SECOND PENETRATION FROM BOTTOM-TRAY VE



SIDE VIEW

FIGURE 4

PARTIAL CROSS SECTION OF PENETRATIONS
(TO SHOW BULKHEAD LOCATION IN WALL AS CONSTRUCTED)

FIG. 4

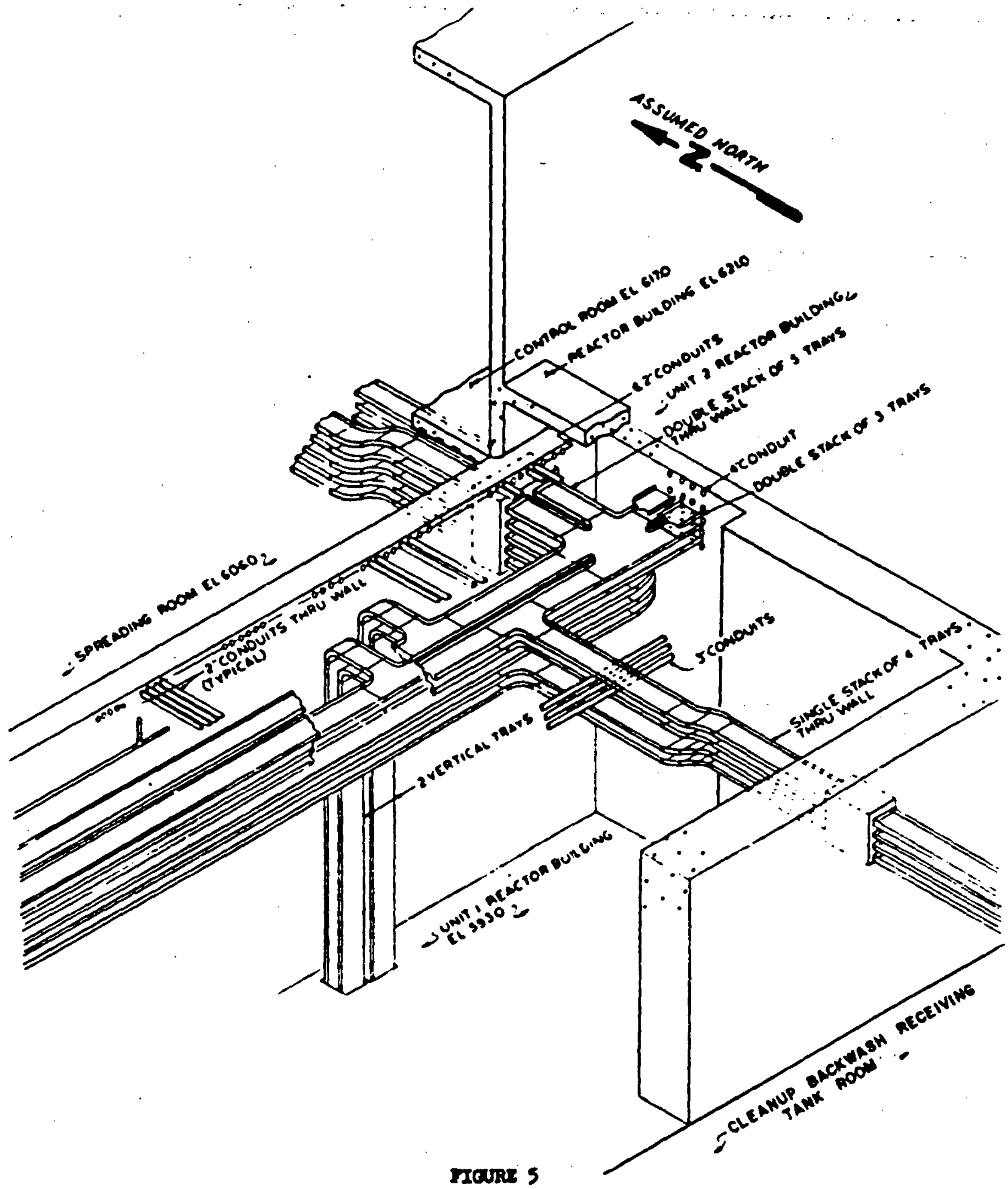


FIGURE 5

AREA OF FIRE

FIG. 5

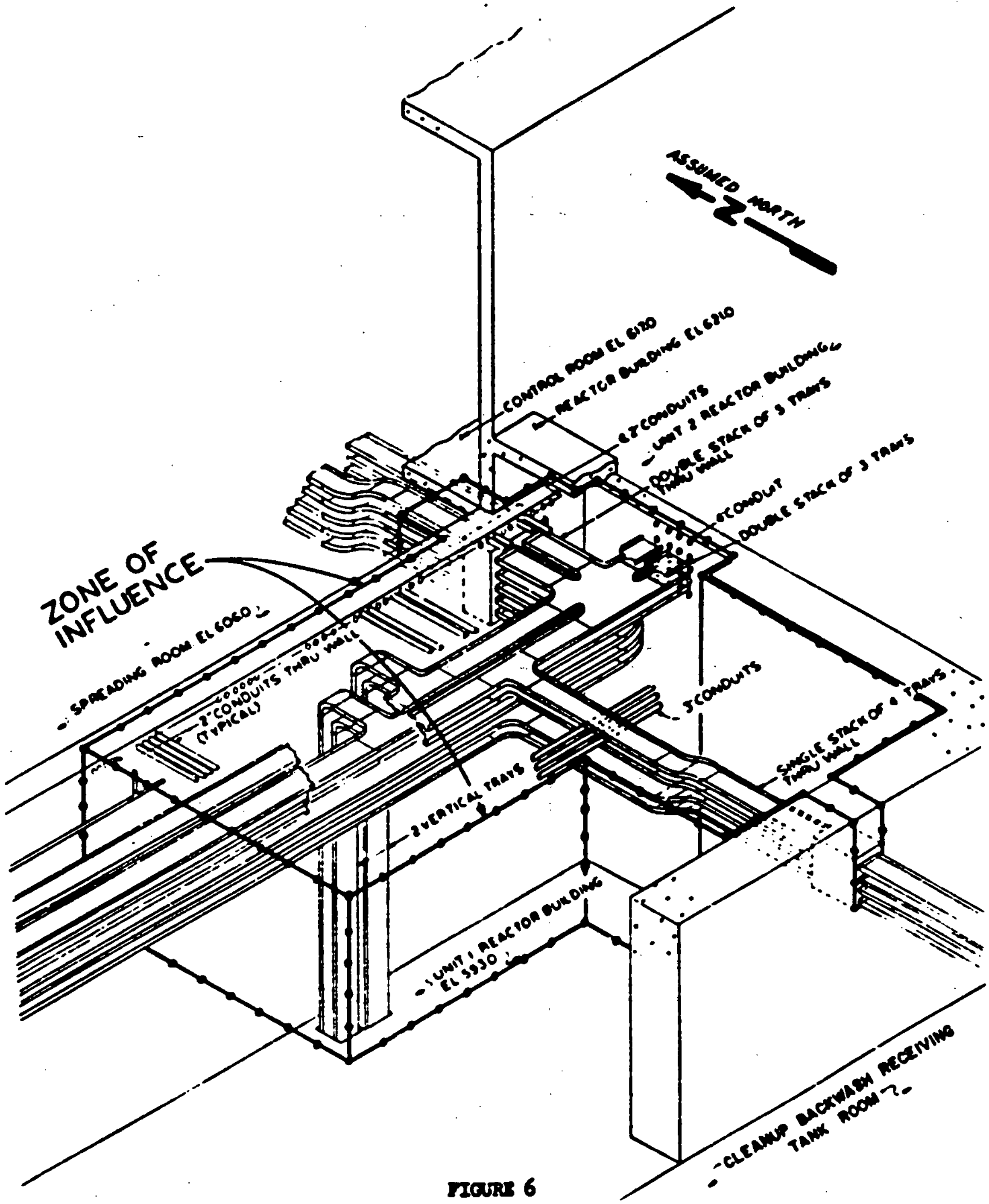
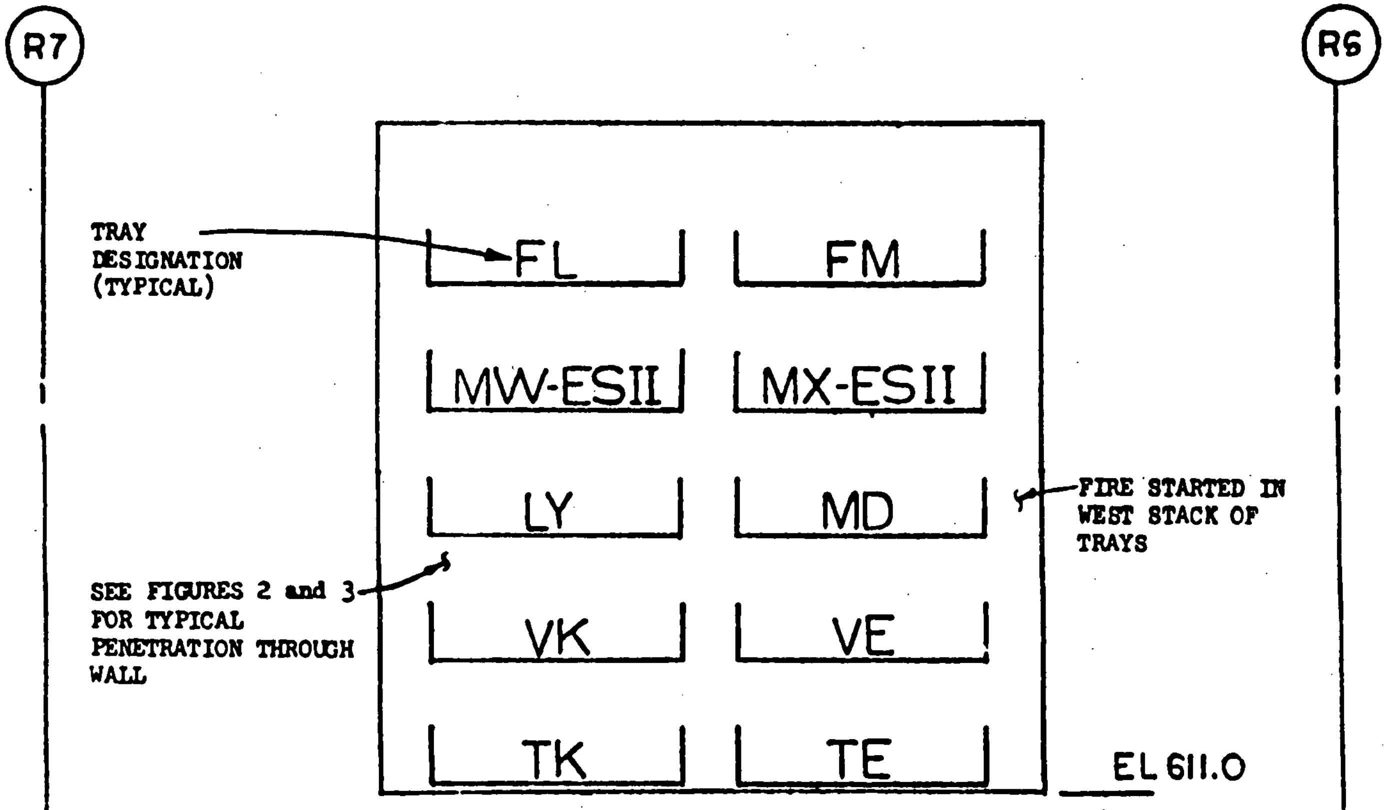


FIGURE 6

AREA OF FIRE

FIG. 6



CABLE TRAYS TO REACTOR BLDG.
(LOOKING SOUTH)

SAME AS CHECKPOINT 131 EXCEPT OPPOSITE HAND

SPREADING ROOM FLOOR EL606.0

FIGURE 7

FIG. 7

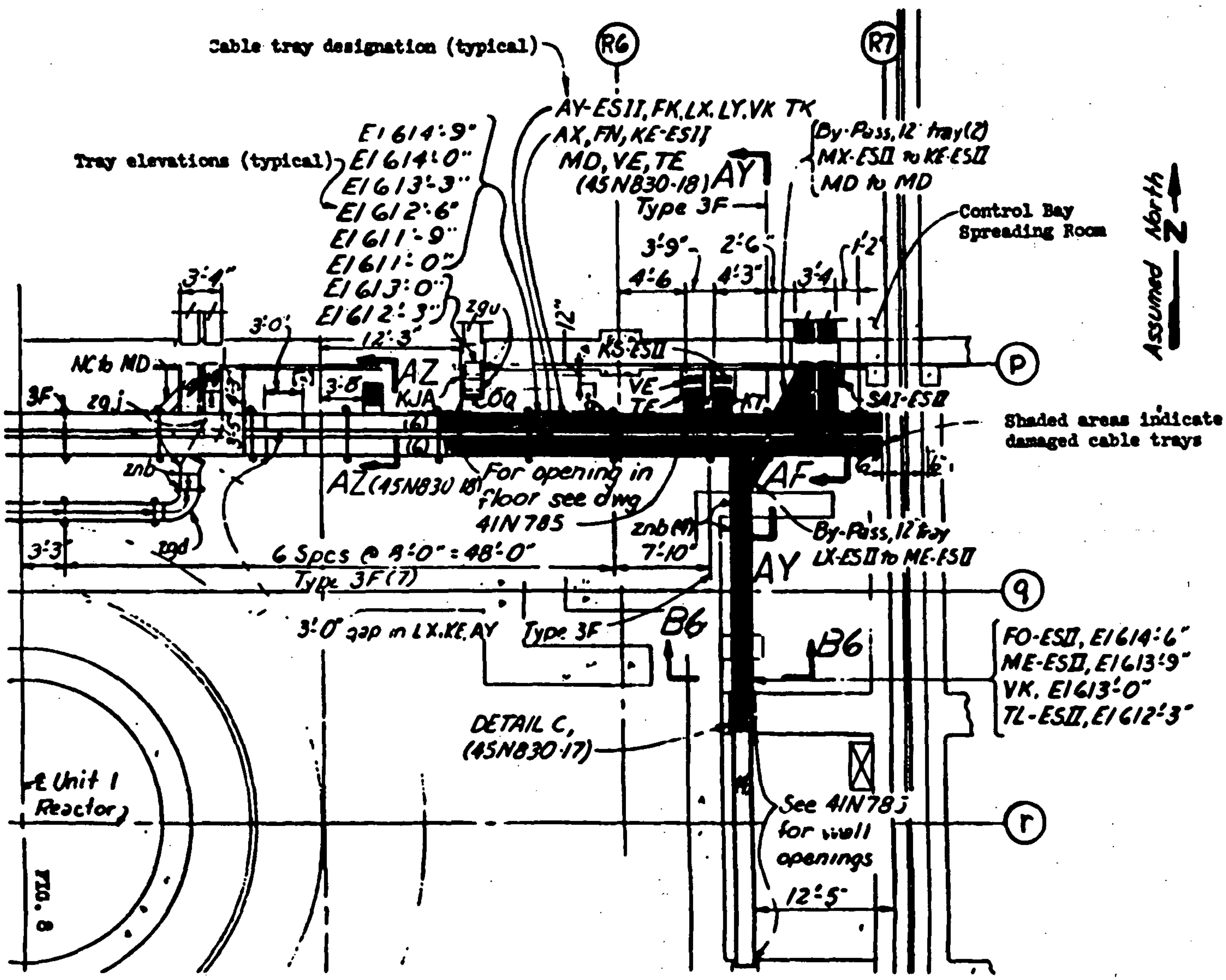


FIGURE 8

PART PLAN VIEW OF CABLE TRAYS

FIGURE 9
CABLE TRAY SINGLE LINE

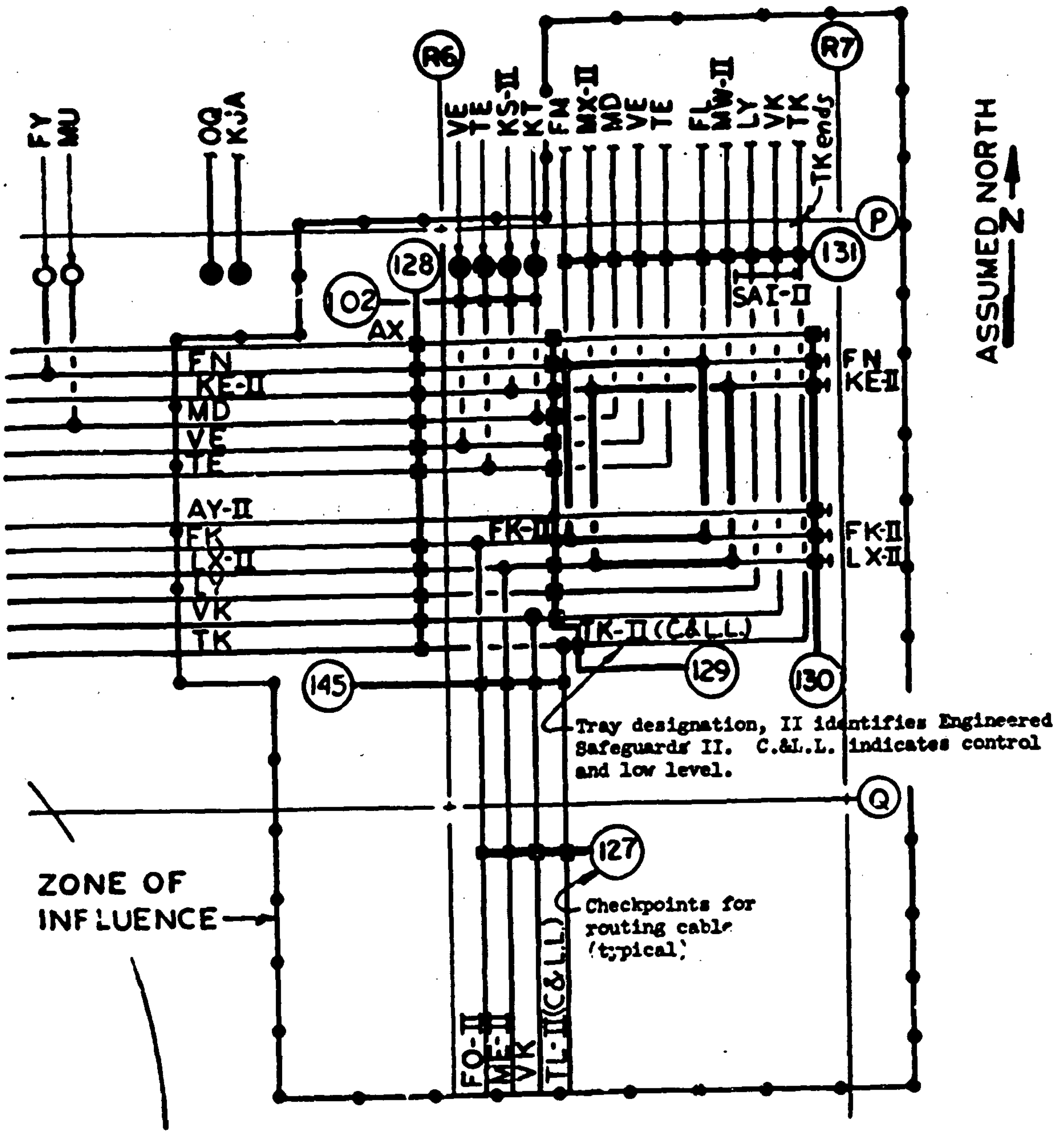


FIG. 9

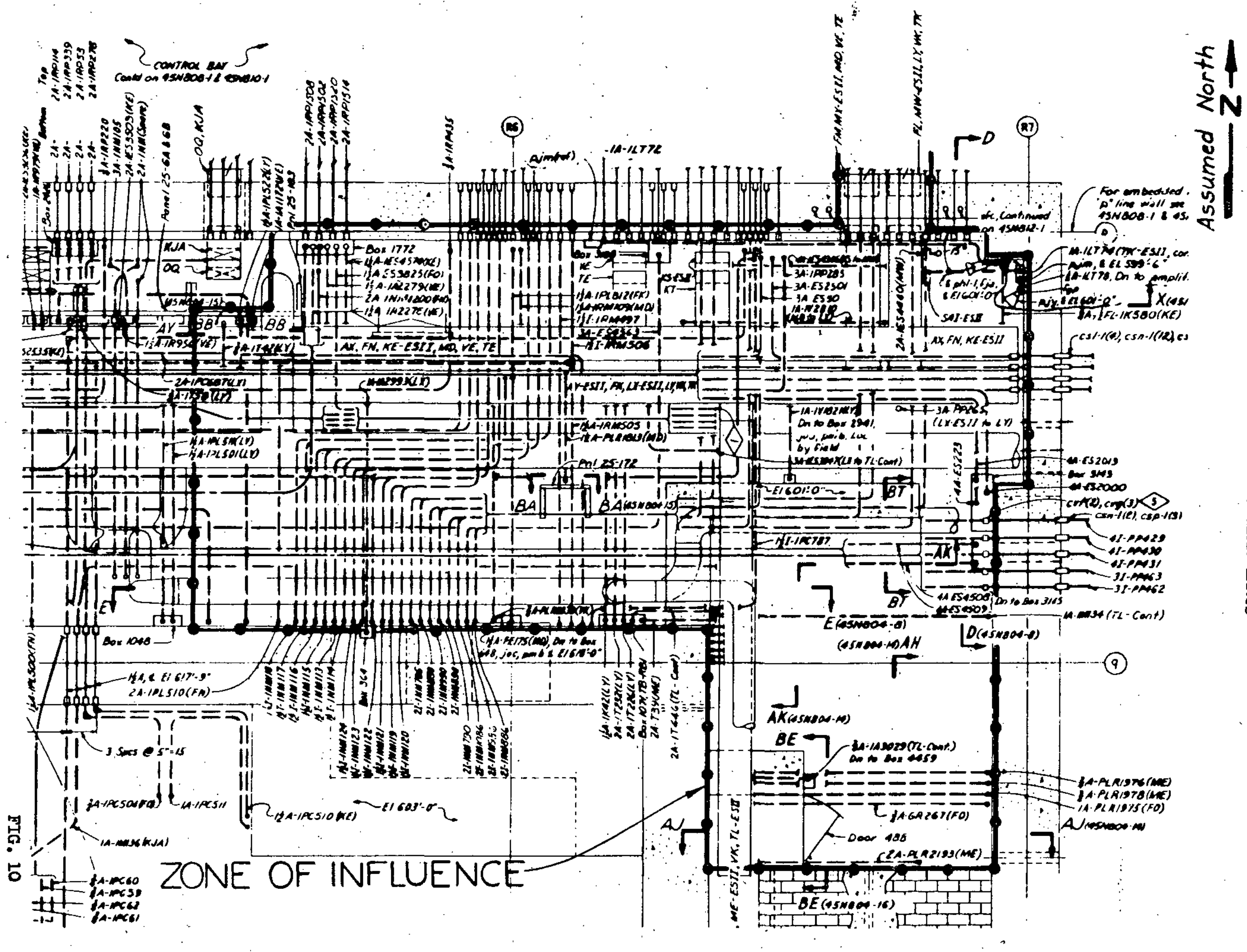


FIGURE 10
PART PLAN VIEW OF CONDUITS NEAR TRAYS

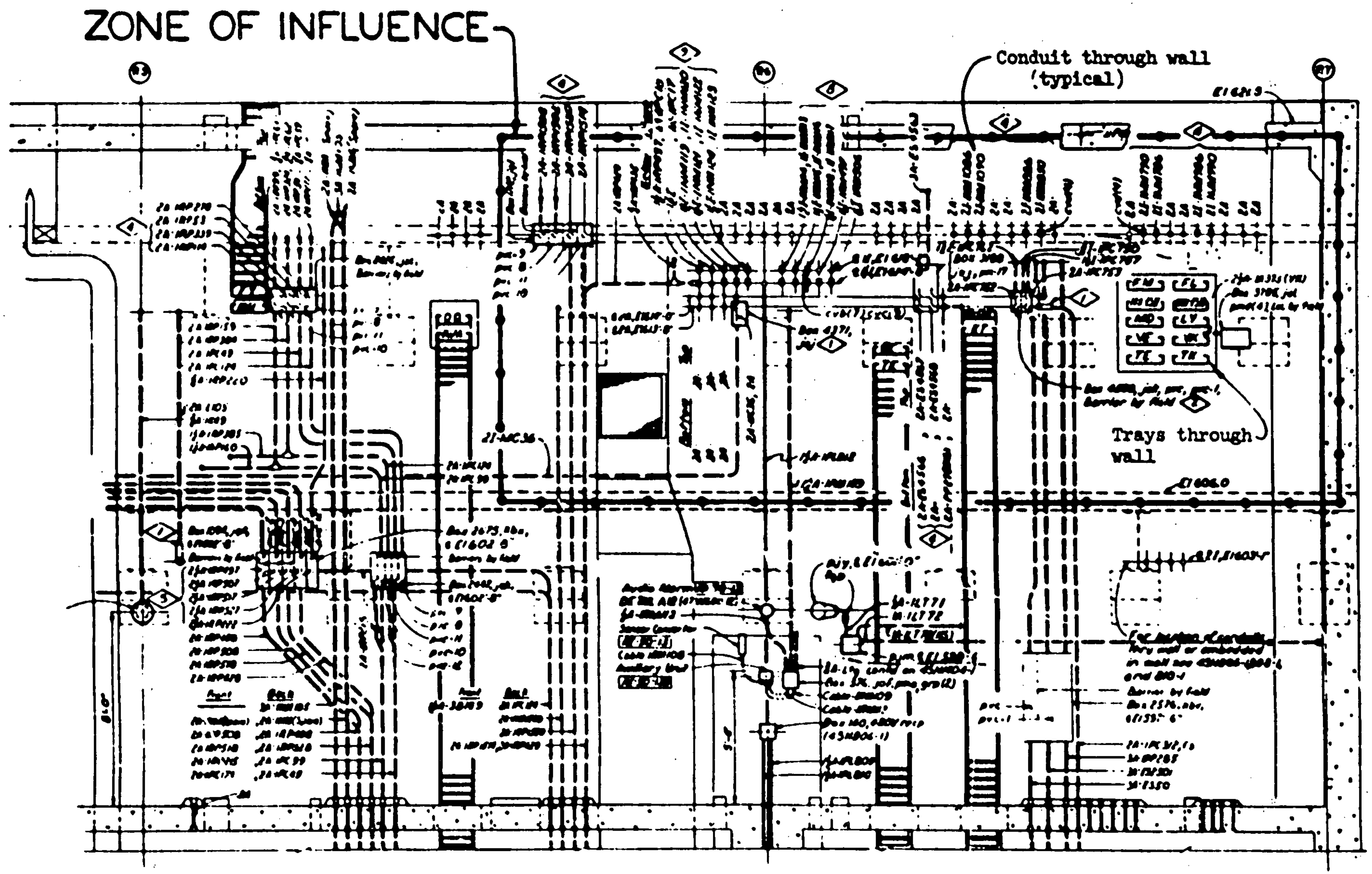


FIGURE 11
ELEVATION VIEW LOOKING NORTH TOWARD CONTROL BAY
FROM REACTOR BLDG UNIT 1 EL 593 SHOWING CONDUITS
AND TRAYS IN ZONE OF INFLUENCE

FIG. 11

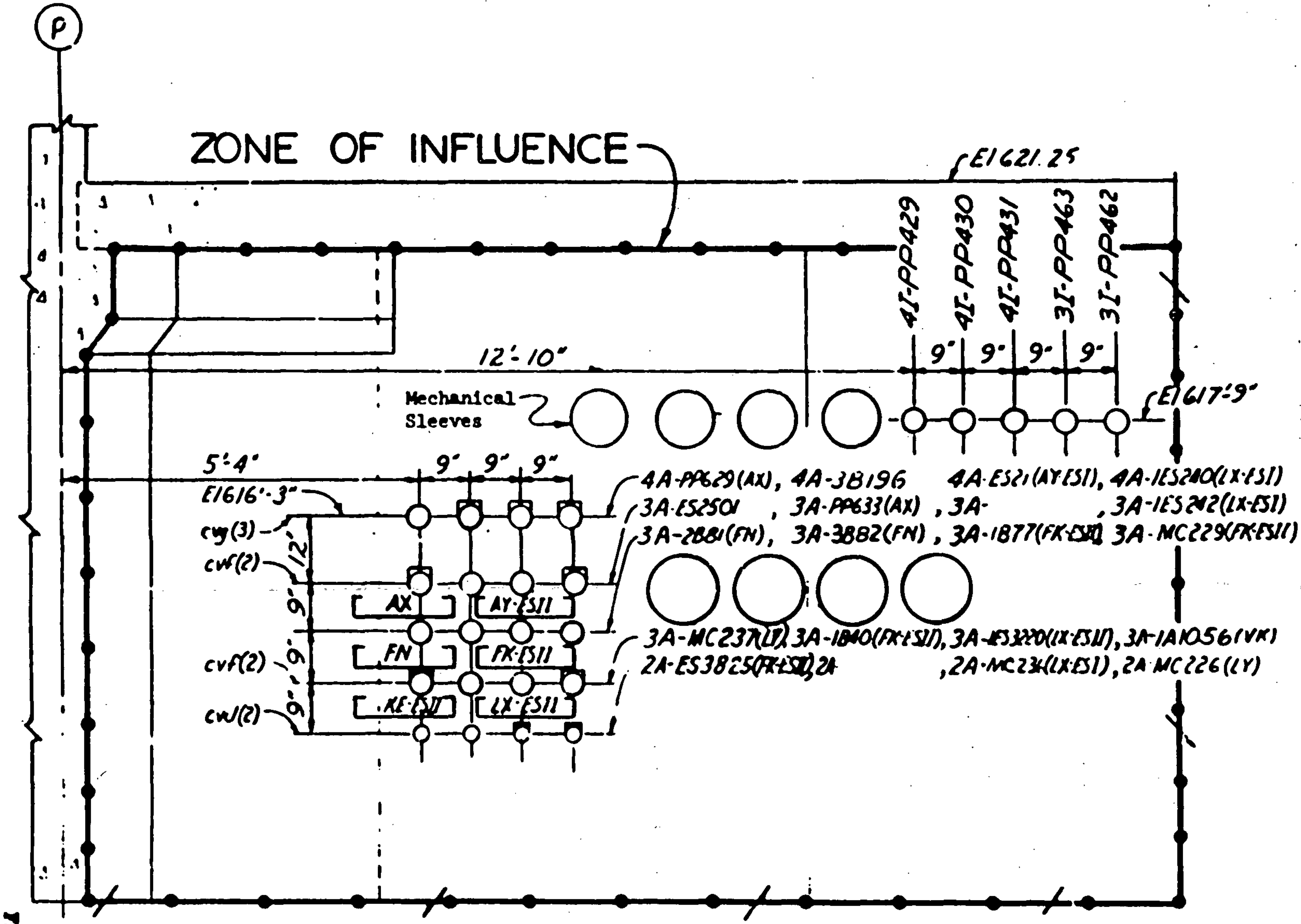


FIG. 12

ELEVATION VIEW LOOKING EAST TOWARD UNIT 2
(cable trays run to wall and stop and cables are fed
into unit 2 through conduits.)

FIGURE 12

DESCRIPTION OF SPECIALTY ITEMS ASSOCIATED WITH PENETRATIONS*

Item	Description	Manufacturer
1. Froth Pak	Insta-Foam	Insta-Foam Products Company Joliet, Illinois

Froth Pak Insta-Foam is the trade name for a kit using an aerosol dispensing unit which contains the chemical components for making rigid polyurethane foam. When the unit is activated, high-quality froth foam is dispensed from two pressurized containers, forming a rigid cellular polyurethane product in less than 1 minute.

2. Polyurethane	Pourable type Part A No. 0293A Pourable type Part B No. 67010	Witco Chemical Company New Castle, Delaware

Polyurethane, pourable type, produces a rigid cellular polyurethane product similar to that produced by the Froth Pak Insta-Foam. The liquids, part A and part B, are mixed equally by pouring back and forth between two containers until mixed and reaction starts. Before it expands, it more readily flows into small crevices to effect a better seal upon expansion.

3. Flamemastic	71A	Dyna-Therm Corporation 598 West Avenue Los Angeles, California

Dyna-Therm Flamemastic coatings are compounded of thermoplastic resinous binders, flame-retardant chemicals, and inorganic incombustible fibers. They have a gray fibrous appearance when dry.

4. Marinite panels	No. 36, type B	Johns-Manville

Marinite panels are composed of incombustible asbestos fibers, diatomaceous silica, and a hydrothermally-produced inorganic binder. They were originally developed to isolate and prevent the spread of shipboard fires. They are hard, dense boards.

5. Resilient polyurethane foam		Hickory Springs Manufacturing Company 2200 Main Avenue, SE. Hickory, North Carolina

Resilient polyurethane foam is a preformed, resilient, cellular polyurethane foam material which was developed primarily to make furniture cushions.

6. Styrofoam	---	Unknown

Styrofoam is a lightweight, preformed thermal-insulating material and packing material. It is commonly used for making ice chests. It is readily found on construction sites since it is also used as protective packing material for fragile equipment.

TABLE 1
SHEET 1 OF 2

	Item	Description	Manufacturer
7.	RTV 102 white	Silicone rubber	General Electric Company Silicone Products Department Waterford, New York

RTV (room temperature vulcanising) silicone rubber is a liquid "rubber" (not a natural rubber) which cures at room temperature to a resilient, tough adhesive. It was originally developed for sealing space vehicles. It is commonly used in the home to seal around bathtubs.

8.	Ty-Rap cable ties	TY-525M	Thomas and Betts Elizabeth, New Jersey

Ty-Rap cable ties are small straps about 1/32 inch thick and 1/8 inch wide, of varying lengths, with a loop in one end for binding cables together. They are generally made of nylon or similar plastic.

9. Other materials may have been used in construction penetration seals.

*These "descriptions" are provided by the committee to assist the laymen in understanding the various materials. The descriptions should not be construed as definitions or precise technical descriptions.

TABLE 1
SHEET 2 OF 2

Checkpoint 102
(Looking North)

VE

TE

KS-ESII

KT

TRAY DESIG	CABLE TYPE	QTY	CABLE OD	TOTAL AREA
VE	WVA	67	.353	6.566
	WVA-1	38	.333	3.306
	WVB	3	.371	.324
	WVC	14	.401	1.764
	MFR	1	.242	.046
TOTAL		123		12.006
TE	WUB	117	.231	4.914
	WUB-1	25	.339	2.250
TOTAL		142		7.164

TRAY DESIG	CABLE TYPE	QTY	CABLE OD	TOTAL AREA
KS-ESII	WFB	1	.490	.189
	WFE	2	.659	.684
	WGB	40	.425	5.68
	WGC	3	.444	.465
	WGD	2	.484	.368
	WGE	1	.559	.246
	WGG	3	.660	1.026
	WGI	1	.710	.396
	WGK	1	.789	.490
	WHB	6	.384	.696
	WHC	2	.405	.258
	WHE	4	.480	.608
	WHG	1	.519	.212
	WHI	3	.640	.966
	WHJ	1	.710	.396
	WLB	1	.509	.204
TOTAL		72		12.884
KT	WGB	1	.425	.142
	WGG	1	.660	.342
	WHB	2	.384	.232
	WHD	7	.439	1.064
	WHG	1	.519	.212
	WTO	1	.340	.091
	WTR	1	.360	.102
	WVE	1	.461	.167
	WVG	1	.587	.271
	WVI	11	.834	7.205
	WVJ	8	1.012	6.440
	WVR	101	.650	33.532
	WVU-1	2	.439	.304
TOTAL		138		50.104

TABLE 2
SHEET 1 OF 11

Checkpoint 127
(Looking North)

FO-ESII
ME-ESII
VK
TL-ESIIL | TL-ESIIC *

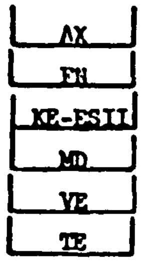
TRAY DESIG	CABLE TYPE	QTY	CABLE OD	TOTAL AREA
FO-ESII	WDD	3	.340	.273
	WDF	4	.429	.580
	WDG	7	.485	1.295
	WDH	3	.619	.903
	WDI	3	.660	1.026
	WLB	1	.509	.204
	WLC	4	.539	.916
	WLN	1	.559	.245
	WLO	4	.627	1.236
TOTAL		30		6.678
ME-ESII	WDE	2	.379	.226
	WDD	4	.340	.364
	WFB	1	.490	.189
	WGB	27	.425	3.834
	WGD	3	.484	.552
	WGE	4	.559	.984
	WGG	1	.660	.342
	WGI	35	.710	13.860
	WGK	5	.789	2.450
	WGM	1	.874	.600
	WHB	6	.384	.696
	WHC	9	.405	1.161
	WHD	4	.439	.608
	WHE	5	.480	.905
	WHG	4	.519	.848
	WHI	4	.640	1.288
	WHJ	7	.710	2.772
	WHL	1	.781	.479
	WLB	1	.509	.204
	WTO	8	.340	.728
	WTR	1	.360	.102
	WVU-1	2	.439	.304
	WWN	2	.0172	.0344
TOTAL		137		33.530

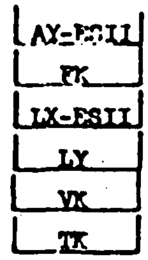
TRAY DESIG	CABLE TYPE	QTY	CABLE OD	TOTAL AREA
VK	WVA	8	.353	.784
	WVA-1	20	.333	1.740
	WVB	9	.371	.972
TOTAL		37		3.496
TL-ESIIC	WFB	8	.490	1.512
	WFD	1	.600	.283
	WGB	21	.425	.284
	WGC	5	.444	.775
	WGD	1	.484	.184
	WGE	2	.559	.492
	WGI	2	.710	.792
	WGK	3	.789	.147
	WHB	6	.384	.696
	WHD	1	.439	.152
	WHE	1	.480	.181
	WHG	4	.519	.848
	WTO	9	.340	.819
	WTR	1	.360	.102
	Belden 8213	1	.405	.129
TOTAL		70		7.396
TL-ESIIL	WUB	40	.231	1.680
	WUB-1	4	.339	.360
	WVA	8	.353	.784
TOTAL		52		2.824

* C - indicates control level portion of TL
L - indicates low level portion of TL

TABLE 2
SHEET 2 OF 11

Checkpoint 128
(Looking East)





TRAY DESIG	CABLE TYPE	QTY	CABLE OD	TOTAL AREA
AX	WHB	6	.915	3.942
FN	WLC	1	.539	.229
	WLS	1	.817	.524
	WDG	12	.485	2.220
	WDN	13	.940	9.022
	WGD	5	.484	.920
TOTAL		32		12.915
KE-ESII	WHE	2	.480	.362
	WHJ	3	.710	1.188
	WDD	2	.340	.182
	WFE	1	.659	.342
	WGB	4	.425	.568
	WGG	1	.660	.342
	WHB	22	.384	2.552
	WWN	1	.148	.0172
TOTAL		36		5.708
MD	WHD	7	.439	1.064
	WHE	1	.480	.181
	WHG	1	.519	.212
	WGB	9	.425	1.278
	WGE	1	.559	.246
	WGG	1	.660	.342
	WHB	2	.384	.232
	WVA-1	2	.333	.174
	WVI	1	.834	.655
	WVJ	8	1.012	5.240
	WVU-1	1	.439	.152
TOTAL		34		9.776

TRAY DESIG	CABLE TYPE	QTY	CABLE OD	TOTAL AREA
AY-ESII	0			
FK	WDF	8	.429	1.160
	WDG	10	.485	1.850
	WDH	5	.619	1.505
	WDI	3	.660	1.026
	WDK	1	.769	.465
	WDN	8	.940	5.552
	WDO	4	1.029	3.332
	WFB	2	.490	.378
TOTAL		41		15.268
LX-ESII	WHC	1	.405	.129
	WHE	1	.480	.181
	WTO	7	.340	.637
	WGC	1	.444	.155
	WGD	1	.484	.184
	WGE	2	.559	.492
	WGG	1	.660	.342
	WGI	1	.710	.396
	WGK	2	.789	.980
TOTAL		17		3.496
LY	WTD	1	.638	.320
	WTO	24	.340	2.184
	WTR	2	.360	.204
	WDD	2	.340	.182
	WFB	1	.490	.189
	WFD	4	.600	1.132
	WGB	10	.425	1.420
	WGC	2	.444	.310
	WGD	1	.484	.184
	WGE	4	.559	.984
	WGG	5	.660	1.710
TOTAL		56		8.819

TABLE 2
SHEET 3 OF 11

Checkpoint 128
(Looking East)
(Continued)

TRAY DESIG	CABLE TYPE	QTY	CABLE OD	TOTAL AREA
VE	WVA	94	.353	9.212
	WVA-1	85	.333	7.395
	WVB	12	.317	1.296
	WVC	4	.401	.504
TOTAL		195		18.407
TE	WUB	80	.231	3.360
	WUB-1	23	.339	2.070
	WVA	3	.353	.294
TOTAL		106		5.724

TRAY DESIG	CABLE TYPE	QTY	CABLE OD	TOTAL AREA
LY	WGK	1	.789	.490
	WHB	6	.384	.696
TOTAL		7		1.186
VK	WVA	29	.353	2.842
	WVA-1	91	.333	7.917
	WVB	9	.371	.972
TOTAL		129		11.731
TK	WUB	16	.231	.672
	WVA	6	.353	.588
TOTAL		22		1.260

Checkpoint 129
(Looking East)

AX	AY-ESII
FN	FK-ESII
KE-ESII	LX-ESII
MD	LY
VE	VK
TE	TK-ESIIL | TK-ESIIC *

TRAY DESIG	CABLE TYPE	QTY	CABLE OD	TOTAL AREA
AX	Same as checkpoint 128			
FN	Same as checkpoint 128			
KE-ESII	WDD	2	.340	.182
	WFB	1	.490	.189
	WFE	1	.659	.342
	WGB	43	.425	6.106
	WGC	3	.444	.465
	WGD	6	.484	1.104
	WGE	1	.559	.246
	WGG	4	.660	1.368
	WGI	1	.710	.396
	WHB	26	.384	3.016
	WHC	2	.405	.258
	WHE	4	.480	.724
	WHG	2	.519	.424
	WHI	2	.640	.644
	WHJ	4	.710	1.584
	WLB	1	.509	.204
	WWN	1	.0172	.0172
TOTAL		104		18.024
MD	Same as checkpoint 131			
VE	Same as checkpoint 131			
TE	Same as checkpoint 131			

TRAY DESIG	CABLE TYPE	QTY	CABLE OD	TOTAL AREA
AY-ESII	0	0	0	0
FK-ESII	WDF	10	.429	1.45
	WDG	7	.485	1.295
	WDH	2	.619	.602
	WDK	4	.769	1.860
	WDN	8	.940	5.552
	WDO	4	1.029	3.332
	WFB	2	.490	.378
	WHB	1	.384	.116
	WLB	1	.509	.204
	WLN	1	.559	.245
	WLO	2	.627	.618
	WVA	2	.353	.196
TOTAL		44		14.544
LX-ESII	WDD	4		.364
	WDE	2		.226
	WFB	1	.490	.189
	WGB	25	.425	3.550
	WGD	3	.484	.552
	WGE	5	.559	2.230
	WGG	2	.660	.684
	WGI	36	.710	14.256
	WGK	5	.789	2.450
	WGM	1	.874	.600
	WHB	6	.384	.696
	WHC	9	.405	1.161
	WHD	4	.439	.608
	WHE	5	.480	.905
	WHG	3	.519	.636
	WHI	4	.640	1.288
	WHJ	7	.710	2.872
	WHL	1	.781	.479
	WLB	1	.509	.204
	WTA	1	1.139	1.020
	WVU-1	2	.439	.304
	WWN	2	.0172	.034
TOTAL		123		33.718

* C - indicates control level portion of TL
L - indicates low level portion of TL

TABLE 2
SHEET 5 OF 11

Checkpoint 129
(Looking East)
(Continued)

TRAY DESIG	CABLE TYPE	QTY	CABLE OD	TOTAL AREA	TRAY DESIG	CABLE TYPE	QTY	CABLE OD	TOTAL AREA
					TK-ESIIC	WFB	8	.490	1.512
						WFD	1	.600	.283
						WGB	16	.425	2.272
						WGC	6	.444	.930
						WGD	2	.484	.368
						WGE	2	.559	.492
						WGI	2	.710	.792
						WGK	4	.789	1.960
						WHB	4	.384	.464
						WHC	1	.405	.129
						WHD	1	.439	.152
						WHE	2	.480	.362
						WHG	4	.519	.848
						WTO	1	.340	.091
						WTR	1	.360	.102
						Belden 8213	1	.405	.129
					TOTAL		56		10.886
					TK-ESIIL	WHB	1	.384	.116
						WUB	34	.231	1.428
						WUB-1	4	.339	.360
						WVA	4	.353	.392
					TOTAL		43		2.296
					LY	Same as checkpoint 128			
					VK	Same as checkpoint 128			

Checkpoint 130
(Looking West)

FN
KE-ESII

AY-ESII
FK-ESII
LX-ESII

TRAY DESIG	CABLE TYPE	QTY	CABLE OD	TOTAL AREA
AX	Same as checkpoint 129			
FN	WDN	4	.940	2.576
	WLS	1	.817	.524
TOTAL		5		3.100
KE-ESII	WLB	1	.509	.204
TOTAL		1		.204

TRAY DESIG	CABLE TYPE	QTY	CABLE OD	TOTAL AREA
AY-ESII		0		
FK-ESII	WDF	8	.429	1.160
	WDG	8	.485	1.480
	WDH	2	.619	.602
	WDK	1	.769	.465
	WDN	8	5.552	.940
	WFB	2	.490	.378
	WLO	1	.627	.309
TOTAL		30		5.334
LX-ESII	WGB	4	.425	.568
	WGD	1	.484	.184
	WGG	1	.660	.342
	WGK	2	.789	.980
	WHB	2	.384	.232
TOTAL		10		2.306

Checkpoint 131
(Looking North)

FM
MX-ESII
MD
VE
TE

FL
MW-ESII
LY
VK
TK

TRAY DESIG	CABLE TYPE	QTY	CABLE OD	TOTAL AREA
FM	WDF	2	.429	.290
	WDG	12	.485	2.220
	WDK	3	.769	1.395
	WDN	3	.940	2.082
	WDO	4	1.029	3.332
	WLC	1	.539	.229
TOTAL		25		9.573
MX-ESII	WDD	2	.340	.182
	WDE	2	.379	.226
	WFB	1	.429	.145
	WFE	2	.659	.684
	WGB	58	.425	8.236
	WGC	1	.444	.155
	WGD	3	.484	.552
	WGE	4	.559	.984
	WGG	4	.660	1.026
	WGI	37	.710	14.652
	WGK	3	.789	1.47
	WGM	1	.874	.600
	WHB	26	.384	3.016
	WHC	10	.405	1.290
	WHD	4	.439	.608
	WHE	7	.480	1.267
	WHG	4	.519	.848
	WHI	4	.640	1.288
	WHJ	4	.710	1.584
	WHL	1	.781	.479
	WVJ-1	2	.439	.304
	WWN	1	.148	.0172
TOTAL		181		38.290
MD	WGB	1	.425	.142
	WGE	1	.559	.246
	WHB	1	.384	.116
	WHD	2	.439	.304
	WHE	1	.480	.181
	WTO	1	.340	.091
	WTR	1	.360	.102
	WVE	1	.461	.167

TRAY DESIG	CABLE TYPE	QTY	CABLE OD	TOTAL AREA
FL	WDN	6	.940	4.164
	WLB	1	.509	.204
	WLN	1	.559	.245
	WLO	1	.627	.309
TOTAL		9		4.922
MW-ESII	WFB	9	.490	1.701
	WFD	1	.600	.283
	WGB	23	.425	3.266
	WGC	6	.444	.930
	WGD	3	.484	.552
	WGE	2	.559	.492
	WGI	3	.710	1.188
	WGK	3	.789	1.47
	WHB	14	.384	1.624
	WHE	5	.480	.905
	WHJ	1	.710	.396
	WLB	1	.509	.204
	WTA	1	1.139	1.020
	Belden 8213	1	.405	.129
TOTAL		73		11.919
LY	WDD	2	.340	.182
	WFB	1	.490	.189
	WFD	3	.600	.849
	WGB	8	.425	1.136
	WGC	2	.444	.310
	WGD	2	.484	.368
	WGE	5	.559	1.230
	WGG	5	.660	1.710
	WGK	1	.789	.490
	WHB	6	.384	.696
	WHC	1	.405	.129
	WHI	3	.640	.966
	WVA-1	2	.333	.174
TOTAL		41		5.783

TABLE 2
SHEET 8 OF 11

Checkpoint 131
(Looking North)
(Continued)

TRAY DESIG	CABLE TYPE	QTY	CABLE OD	TOTAL AREA	TRAY DESIG	CABLE TYPE	QTY	CABLE OD	TOTAL AREA
MD	WVG	1	.587	.271					
(Continued)	WVI	11	.834	7.205	VK	WVA	41	.353	3.332
	WVJ	16	1.012	12.880		WVA-1	46	.333	4.002
	WVR	101	.650	33.592					
	WVU-1	3	.439	.456	TOTAL		87		7.334
TOTAL		141		55.693	TK	WUB	34	.231	1.428
						WUB-1	25	.339	2.250
VE	WVA	52	.353	5.096		WVA	2	.353	.196
	WVA-1	61	.333	5.307					
	WVB	15	.371	1.620	TOTAL		61		3.874
	WVC	10	.401	1.260					
	MPRS	1	.242	.046					
TOTAL		139		13.329					
TE	WUB	37	.231	1.974					
	WUB-1	47	.339	4.230					
	WVA	3	.353	.2940					
TOTAL		87		6.498					

Checkpoint 145

(Looking North)

FO-ESII

ME-ESII

VK

TL-ESIIL TL-ESIIC *

TRAY DESIG	CABLE TYPE	QTY	CABLE OD	TOTAL AREA
FO-ESII	WDD	3	.340	.273
	WDF	2	.429	.290
	WDG	6	.485	1.110
	WDH	1	.619	.301
	WDI	1	.660	.342
	WLB	1	.509	.204
	WLC	1	.539	.229
	WLN	1	.559	.245
	WLO	4	.627	1.236
TOTAL		20		4.230
ME-ESII	WDE	2	.379	.226
	WDD	4	.340	.364
	WFD	1	.600	.283
	WGB	27	.425	3.834
	WGD	3	.484	.552
	WGE	4	.559	.984
	WGG	1	.660	.342
	WGI	35	.710	13.860
	WGK	3	.789	.147
	WGM	1	.874	.600
	WHB	5	.384	.580
	WHC	8	.405	1.032
	WHD	4	.439	.608
	WHE	5	.480	.905
	WHG	4	.519	.848
	WHI	4	.640	1.288
	WHJ	7	.710	2.772
	WHL	1	.781	.479
	WLB	1	.509	.204
	WTA	1	1.139	1.020
	WTO	8	.340	.728
	WTR	1	.360	.102
	WVU	2	.376	.222
	WWN	2	.0172	.034
TOTAL		134		32.014

TRAY DESIG	CABLE TYPE	QTY	CABLE OD	TOTAL AREA
VK	WVA	10	.353	.980
	WVA-1	22	.333	1.914
	WVB	9	.371	.972
TOTAL		41		3.866
TL-ESIIC	WFB	8	.490	1.512
	WFD	1	.600	.283
	WGB	29	.425	4.118
	WGC	5	.444	.775
	WGD	1	.484	.184
	WGE	4	.559	.984
	WGI	2	.710	.792
	WGK	3	.789	.147
	WHB	6	.384	.696
	WHC	1	.405	.129
	WHD	1	.439	.152
	WHE	2	.480	.362
	WHG	4	.519	.848
	WTO	8	.340	.728
	WTR	1	.360	.102
	BELDEN 8213	1	.405	.129
TOTAL		77		11.941
TL-ESIIL	WUB	50	.231	2.100
	WUB-1	4	.339	.360
	WVA	8	.353	.784
TOTAL		62		3.244

* C - indicates control level portion of TL
  L - indicates low level portion of TL

TABLE 2
SHEET 10 OF 11

Note: Tray loading for vertical tray connecting trays MW-II and TK-II south of checkpoint 131

SAI-ESII

TRAY DESIG	CABLE TYPE	QTY	CABLE OD	TOTAL AREA	TRAY DESIG	CABLE TYPE	QTY	CABLE OD	TOTAL AREA
SAI-ESII	WFB	8	.490	1.512					
	WFD	1	.600	.283					
	WGB	19	.425	2.698					
	WGC	4	.444	.620					
	WGD	2	.484	.368					
	WGE	2	.559	.492					
	WGI	2	.710	.792					
	WGK	3	.789	1.470					
	WHB	6	.384	.696					
	WHC	1	.405	.129					
	WHD	1	.439	.152					
	WHE	2	.480	.362					
	WHG	4	.519	.848					
	Belden 8213	1	.405	.129					
TOTAL		56		10.551					

COMPUTED BY CABLE TRAY FQ-ESII CHECKED BY BU-BU DATE 120 & 145

CABLE	PURPOSE	FROM	TO	SCHEM	TYPE	WHERE TO DISCONNECT OR DE-ENERGIZE AND OTHER COMMENTS (40)
B331	Diesel Gen Air Compr Back up Mot Alternate Feed	Batt Bd Pnl 7 Bkr 712 45N701-1	DC throwover sw 45N786-11	45N776	wdg	Open breaker 712, BATT BD 4, panel 7
ES3825-II	Battery & Bd Rm exhaust fan 1B Supply	480 Cont Bay Vent Bd B Pnl 6C 45N788-2	Fan Motor 1B 45N788-4	45N769-7	w/o	Open breaker compartment 6C, 480V Cont. Bay Vent Bd B
ES3900-II	Bd Rm Emer Supply fan 1B Supply	480V Reac MOV Bd 1B pnl 17A 45N1750-6	Fan Motor 45N788-4	45N769-7	wdd	Open breaker compartment 17A, 480V REAC MOV Bd 1B
PL478	Reactor Bldg elevator No. 1 supply	Common Bd Pnl 7B 45N781-2	Mtrs JB 45N781-2	45N764-1	wdg	Open breaker compartment 7B, 480V Common Bd 1
PLR475	Clean-up Backwash Transfer Pump 1A Supply	480V R.B. Vent Bd 1B Pnl 6A 45N1756-2	Pump Motor 45N1756-9	45N779-5	w/o	TRIP BKR 480V REACT BLDG VENT BD 1B, COMPT 6A
K321	SEE SHEET 9					
1B89-1E	Alternate Supply, 250V CONTROLS	Batt Bd 2 Pnl 7 Bkr 708 45N702-1	480V Shutdn Bd 1A Pnl 4 45N1748-2	45N769-1	wdt	TRIP BKR 708, BATT BD 2, PNL 7
1B93-1E	Alternate Supply, 250V CONTROLS	Batt Bd 3 Pnl 7 Bkr 709 45N703-1	480V Shutdn Bd 1B, Pnl 4 45N1748-3	45N769-2	wdt	TRIP BKR 709, BATT BD 3, PNL 7
1ES3000-II	Core Spray Sys II outboard disch valve (FCV-75-53) Supply	480V R MOV Bd 1B Pnl 7E 45N1750-3	FCV-75-53 45N1750-11	45N751-3	w/o	TRIP BKR 480V REACT MOV BD 1B, COMPT 7E
1ES3013-II	Core Spray Sys II outboard disch valve (FCV-75-51) Supply	480V Reac MOV Bd 1B Pnl 8E 45N1750-3	FCV-75-51 45N1750-11	45N751-3	w/o	TRIP BKR 480V REACT MOV BD 1B; COMPT 8E
1K575	Evacuation Alarm System	480V fuse Pnl A 45N785-12	Local Starter in JB 5612 55N2779	55N2779	w/b	PULL FU 2AX & 2AY EVAC ALARM FUSE PNL A EL 606, COL PR11
1K606	Evacuation Alarm System	480V fuse Pnl B 45N786-12	Local Starter in JB 5586 55N2779	do	w/n	PULL FU 1BX & 1BY EVAC ALARM FUSE PNL B EL 606, COL PR12
1PC504-II	Reactor water Clean-up Sys isol valve (FCV-69-12) supply	480V Reactor MOV Bd 1B Pnl 17E 45N1750-6	FCV-69-12 45N1750-7	45N751-3	w/c	TRIP BKR 480V REACT MOV BD 1B, COMPT 17E
1PL474	Unit Preferred MMG Set 1 mor fdr supply	480 Shutdn Bd 1A pnl 7D 45N1748-1	Unit Preferred MMG Set Control Cab 45N708-2	45N799-1	wdh	TRIP BKR 480V SHTDN BD 1A COMPT 7D

TABLE 3
SHEET 1 OF 1

SAMPLE CABLE TABULATION SHEET

Number of each class of safety related cables routed in fire zone.

Plant Usage	Number	Safety Classification	Channel or Division*
Common	20	Engineered Safeguard - ECCS	I
	20	Engineered Safeguard - ECCS	II
	13	Engineered Safeguard - Diesel A	IA
	33	Engineered Safeguard - Diesel C	IIC
	5	Engineered Safeguard - Diesel D	IID
	7	Load Shedding - Diesel A	A1
	9	Load Shedding - Diesel C	B1
	7	Supporting Auxiliaries - Electrical	IE
Subtotal	114		
Unit 1	6	Engineered safeguard - ECCS	I
	182	Engineered Safeguard - ECCS	II
	4	Load Shedding - Diesel A	A1
	5	Load Shedding - Diesel C	B1
	1	Load Shedding - Diesel D	B2
	52	Neutron Monitoring (also activates	IA
	52	Neutron Monitoring RPS)	IB
	52	Neutron Monitoring	IIA
	52	Neutron Monitoring	IIB
	14	Primary Containment Isolation	I
	39	Primary Containment Isolation	II
	2	Reactor Protection (control rod	IA
	2	Reactor Protection scram)	IB
	2	Reactor Protection	IIA
	2	Reactor Protection	IIB
	3	Reactor Protection	IIIB
	12	Supporting Auxiliaries - Electrical	IE
Subtotal	482		
Unit 2	15	Engineered Safeguard - ECCS	I
	3	Engineered Safeguard - ECCS	II
	4	Supporting Auxiliaries - Electrical	IE
Subtotal	22		
Unit 3	4	Engineered Safeguards - ECCS	I
	3	Engineered Safeguards - ECCS	II
	3	Supporting Auxiliaries - Electrical	IE
Subtotal	10		
TOTAL	628		

*See sheets 2 & 3 for channel or division definitions.

TABLE 4
SHEET 1 OF 3

SUMMARY OF CABLE SUFFIX DEFINITIONS

1. Cables in the PP series with an A or B suffix are to be separated from each other. They are not engineering safeguard cables, but a separate routing is desirable. They involve off-site power.

2. The following suffixes apply to all cable series:

   I - Division I engineering safeguard or Primary Containment Isolation cables

   II - Division II engineering safeguard or Primary Containment Isolation cables

   IA - Diesel generator A shutdown logic cables (may be routed in cable tray with Division I cables)

   IB - Diesel generator B shutdown logic (routed in conduit)

   IIC - Diesel generator C shutdown logic (may be routed in cable tray with Division II cables)

   IID - Diesel generator D shutdown logic cables (routed in conduit)

3. The following suffixes apply to LS series:

   A1 - 480V load shedding logic channel A1: (routed with IA-Diesel A)

   A2 - 480V load shedding logic channel A2: (routed with IB-Diesel B)

   B1 - 480V load shedding logic channel B1: (routed with IIC-Diesel C)

   B2 - 480V load shedding logic channel B2: (routed with IID-Diesel D)

4. The following suffixes apply to RP (Reactor Protection) or NM (Neutron Monitoring) series:

   IA - RPS logic channel A1

   IIA - RPS logic channel A2

   IB - RPS logic channel B1

   IIB - RPS logic channel B2

5. The following suffixes apply to RP (Reactor Protection) series:

   IIIA - RPS manual and back-up scram solenoid channel A.

   IIIB - RPS manual and back-up scram solenoid channel B

   A - 120V a-c RPS channels A1, A2, and A3 supply (RPS MG set A)

   B - 120V a-c RPS channels B1, B2, and B3 supply (RPS MG set B)

   G1 - RPS scram solenoid Group 1

TABLE 4
SHEET 2 OF 3

G2 - RPS scram solenoid Group 2

G3 - RPS scram solenoid Group 3

G4 - RPS scram solenoid Group 4

6. Suffix IE - Applies to supporting auxiliaries needed for safe shutdown of plant.

Summary of cable types involved in fire.

CABLE TYPE (MARK)	DESCRIPTION NO. & SIZE OF CONDUCTORS	INSULATED*	NO. CABLES DAMAGED
WBB	1/c # 12	1	11+
WCA	1/c # 14	1	2+
WDD	1/c # 8	2	4+
WDE	1/c # 6	2	1+
WDF	1/c # 4	2	7+
WDG	1/c # 2	2	13+
WDH	1/c #1/0	2	2+
WDI	1/c #2/0	2	2+
WDK	1/c #4/0	2	2+
WDN	1/c #300	2	6+
WDO	1/c #400	2	1+
WFB	2/c # 10	3	16
WFC	3/c # 10	3	1
WFD	4/c # 10	3	6
WFE	5/c # 10	3	1
WGB	2/c # 12	3	157
WGC	3/c # 12	3	18
WGD	4/c # 12	3	37
WGE	5/c # 12	3	13
WGG	7/c # 12	3	14
WGI	9/c # 12	3	45
WGK	12/c # 12	3	18
WGM	16/c # 12	3	2
WGN	19/c # 12	3	2
WHB	2/c # 14	3	62
WHC	3/c # 14	3	15
WHD	4/c # 14	3	18
WHE	5/c # 14	3	13
WHG	7/c # 14	3	10
WHI	9/c # 14	3	8
WHJ	9/c # 14	3	7
WHL	16/c # 14	3	1
WHT	7/c # 16	3	8

* Numbers listed correspond to insulation of cable type as shown below.

\+ Number of individual cable designations. Actual number of conductors appear on checkpoint sheets showing tray fill.

...mary of cable types involved in fire.

CABLE TYPE (MARK)	DESCRIPTION NO. & SIZE OF CONDUCTORS	INSULATED*	NO. CABLES DAMAGED
WLB	2/c # 12	4	8
WLC	3/c # 12	4	5
WLG	7/c # 12	4	1
WLN	2/c # 10	4	1
WLO	3/c # 10	4	5
WLS	7/c # 10	4	1
WNB	1/c #2/0	5	6+
WNC	1/c #4/0	5	2+
WNF	1/c #500	5	3+
WTA	50 pr #19	6	1
WTD	12 pr #19	6	1
WTJ	COAX	7	22
WTK	COAX	8	212
WTK-1	COAX	9	8
WTK-2	COAX	10	4
WTO	2 pr #18	11	42
WTR	2 pr #14	12	4
WUB	Thermocouple	13	167
WUB-1	Thermocouple	14	51
WVA	2/c # 16	15	154
[illegible]-1	2/c # 18	15	206
[illegible]	3/c # 16	15	33
WVC	4/c # 16	15	10
WVE	7/c # 16	15	1
WVG	12/c # 16	15	1
WVI	27/c # 16	15	11
WVJ	37/c # 16	15	16
WVR	29/c # 20	16	101
WVU	5/c # 18	16	4
WVU-1	8/c # 18	16	5
BELDEN 8213	COAX	7	1
MFR'S TV CABLE (BELDEN 8212)	COAX	7	1

* Numbers listed correspond to insulation of cable type as shown below.

+ Number of individual cable designations. Actual number of conductors appear on checkpoint sheets showing tray fill.

1. Single conductor power or control cable with polyethelene insulation and a nylon jacket over the polyethelene. (Termed "PN" per TVA Specification)

2. Single conductor power or control cable with cross-linked polyethelene insulation and a polyvinyl-chloride insulation jacket over the cross-linked polyethelene. (Termed "CPJ" per TVA Specification)

3. Multiple-conductor cable with a core of the specified number of single conductors as in 1 above covered by a polyvinyl-chloride outer jacket. (Termed "PNJ" per TVA Specification)

4. Multiple-conductor cable with a core of the specified number of single conductors as in 2 above covered by a polyvinyl-chloride outer jacket. (Termed "CPJJ" per TVA Specification)

5. Single conductor high-voltage (5000 volts) power cable with extruded stand and cross-linked polyethelene insulation with metallic electrostatic shielding and polyvinyl chloride jacket overall. (Termed "CPSJ" per TVA Specification)

6. Telephone cable with high density polyethelene over each conductor, mylar backed rubber cable tape, aluminum shield, and high density polyethelene jacket overall. Some of these had polyvinyl chloride jacket overall.

7. Coaxial signal cable with both conductor and overall jacket insulated with polyethelene.

8. Coaxial signal cable with conductor insulated with polyethelene and polyvinyl chloride jacket overall.

9. Coaxial signal cable with conductor and overall jacket insulated with irradiated blend of polyolefins and polyethelene and noise free. Some of these types had cross-linked polyethelene over both.

10. Same as 8 but made noise free by a carbon suspension.

11. Same as 6 except without shield.

12. Twisted pair cable with polyethelene over each conductor and polyvinyl chloride jacket overall.

13. Thermocouple cable with high density polyethelene over each conductor, aluminum foil/mylar type laminated shield, and high density polyethelene overall.

14. Thermocouple cable with heat and light stabilized cross-linked polyethelene over each conductor, aluminum foil/mylar tape shield, and chlorosulfonated polyethelene jacket overall.

15. Signal cable with heat and light stabilized cross-linked polyethelene over each conductor, aluminum foil/mylar tape laminated shield, fiberglass reinforced silicone tape assembly wrap, and chlorosulfonated polyethelene jacket overall.

16. Multiple-conductor cable with core of specified number of single conductor cables insulated with cross-linked polyethelene and a neoprene jacket overall.

TABLE 6
SHEET 1 OF 7

BROWNS FERRY UNIT 1

SEQUENCE OF SIGNIFICANT OPERATIONAL EVENTS AT TIME OF FIRE

Time	Event	Action	Response or Nonresponse
3/22/75 Prior to 1235	Initial Condition	Routine Operation	Unit Load 1,100 MWe
1235	Report of fire received by assistant shift engineer from public safety officer.	Assistant shift engineer set off fire alarm and proceeded to fire. Fire alarm sealed in by unit operator who then used paging system to inform plant personnel of fire location.	Operating personnel fire brigade reported to fire and began fire-fighting activities (describe elsewhere in investigation report).
~ 1240	Received the following alarms in unit 1 control room: 1. RHR or core spray pumps running/ auto blowdown permissive 2. Reactor level low/ to blowdown permissive 3. Core cooling system/diesel initiate	Unit operator observed control board and determined normal reactor water level and steam pressure, drywell pressure normal at 0.45 psig, and emergency core cooling system (ECCS) equipment aligned in normal standby status. (Reactor water level instrumentation activates the emergency core cooling systems, this being normal, indicated a lack of need for these systems.) (Normal drywell pressure indicated that piping was intact inside the primary containment.)	All diesel generators (D/G's) started from ECCS logic signal which started the core spray pumps.
~ 1242	Residual heat removal (RHR) and core spray (CS) pump running alarm received. High-pressure coolant injection pump (HPCI), reactor core isolation coolant pump (RCIC) started.	Unit operator observed pumps running and RHR aligned to reactor in low-pressure coolant injection (LPCI) mode. Verified reactor water level normal and stopped pumps. Operator attempted to reset alarm. (All four of these systems are ECCS and with normal level were not required.)	Pumps stopped. Alarm would not reset with reactor pressure and level normal.
~ 1244	RHR and core spray pumps restarted with no apparent reason.	Operator observed reactor level normal and attempted to stop RHR and core spray pumps. Pumps could not be stopped from benchboard.	Operator did stop pumps at ~ 1248 from benchboard.
~ 1248	Reactor recirculation pumps ran back for no apparent reason. Began losing electrical boards. Indicating lights over valve and pump control switches on panel 9-3 were glowing brightly, dimming, and going out. (Panel 9-3 is the control board location for all ECCS equipment.) The lights being lost on control circuits for ECCS pumps and valves precluded reliable operation from that control board.	Operator observed reactor power decreasing and average power range monitors (APRM) responding. Also noted reactor level 2 to 3 inches high. Operators observed smoke from control wiring under panel 9-3.	Unit power decreased from 1,100 MWe to 700 MWe.

Time	Event	Action	Response or Nonresponse
1248 (Contd.)	Lost 1/2 of reactor protection system (RPS).		
	Lost remote manual control of a number of relief valves.		
	Numerous alarms occurred on all control panels and unit in unstable swing.		
1251	Shift engineer instructed operator to zero recirculating pump loading and scram the reactor.	Operator reduced loading signal to recirculating pumps and manually scrammed the reactor and placed reactor mode switch in the shutdown position.	Recirculating pumps tripped by unknown cause at approximately 20 percent loading. Reactor scrammed and all rods inserted.
		Shift engineer reported plant conditions to supervisors by phone.	
1253	Confirmed that all control rods were fully inserted.	Operator tripped B and C reactor feed pumps (RFP), B and C condensate booster pumps, and C condensate pump. Reduced loading on reactor feed pump subpanel to prevent "over shoot" on reactor level return. RCIC started manually as backup.	Pumps responded to trip signal and reactor level was maintained by reactor feed pump A and RCIC.
1254	Unit conditions indicated need for tripping turbogenerator.	Assistant shift engineer (ASE) initiated turbine trip upon observing generator load at 100 MW. Also opened generator field breaker and motor-operated disconnects (MOD's).	Turbine bypass valves opened to compensate for turbine valve closure and maintain pressure normal. The main-steam isolation valves (MSIV's remain open allowing reactor pressure control through the turbine bypass valves to main condenser heat sink.
		Unit operator inserted source range and intermediate range neutron monitors and observed reactor power decrease.	Neutron monitoring responded normally.
	HPCI started.		HPCI automatically aligned in normal injection mode to reactor vessel.
	Reactor water level restored to approximately normal range.	Operators shut down HPCI and RCIC.	HPCI and RCIC shutdown. Problems incurred upon shutdown with valve operation associated with th systems.
1255	Lost 120-V unit preferred power. One of the feeds from this source is the unit control rod position indication on panel 9-5 (reactor control panel).	Operator placed reactor mode switch in "Refuel" mode to verify one rod withdraw permit. (All rods must be fully inserted or the indicating light for one rod withdraw in refuel mode will not illuminate.)	Received white permit light.
	Lost all neutron monitoring.	Operator observed no indication on average power range, intermediate range, or source range monitor.	Capability to monitor core was lost.

TABLE 6 2 OF 7

Time	Event	Action	Response or Nonresponse
~ 1256	By this time the following electrical boards were lost: 1A 250-V D.C. Reactor MOV board* 1B 250-V D.C. Reactor MOV board 1A 480-V A.C. Reactor MOV board 1B 480-V A.C. Reactor MOV board 1C 480-V A.C. Reactor MOV board 1A 480-V A.C. Shutdown (SD) board 1B 480-V A.C. Shutdown board 120-V A.C. unit preferred	Indication from the unit control room as to electrical sources feeding the various equipment and as verified by ASE as he checked the individual boards.	This caused the loss of vital equipment being fed from these electrical boards. Loss of power to MSIV's caused them to go closed (all 4 outboard valves), placing the unit in isolation from the main condenser heat sink and cutting off the steam supply to the reactor feed pump turbines. All emergency core cooling systems were lost with the exception of 4 relief valves which could be operated from the unit control board.
~ 1258	Reactor pressure rapidly increased to 1,100 psig.	ASE was unsuccessful in opening MSIV's from backup control center.	Relief valves opening and closing to maintain pressure between 1,080 and 1,100 psig. Relieving to the suppression pool (torus).
		Operator manually opened main-steam relief valves; then closed as pressure came back to desired range.	Pressure decreased to 850 psig; then rapidly increased to 1,080 psig.
		Attempts to place RCIC in service were unsuccessful from control room or backup control panel.	Valve 71-2 (steam supply to turbine) was apparently the only valve loss on RCIC but rendered it inoperable...This valve was later opened by use of temporary power.
			The HPCI was previously rendered inoperable by loss of valve controls.
~ 1259	Reactor water level decreasing due to almost constant blowing down to the torus.	The only water input left with the capability to overcome a pressure above ~ 350 psig was the control rod drive pump; it was increased to the maximum.	
	Torus cooling became essential.	ASE was unsuccessful in placing emergency power on RHR valves at local MOV board. (Those valves required for torus cooling.)	RHR system was unavailable for torus cooling as a result of electrical board losses.
		Shift engineer and two electricians making attempts to restore 480-V 1A and 1B reactor MOV boards and 250-V D.C. boards.	
1300	4-kV SD board C undervoltage shutdown bus 2 undervoltage. (As noted on electrical printer.)		4-kV voltage continued to be supplied to SD boards A and B by shutdown bus 1. Shutdown boards C and D transferred to D/G's C and D.

*MOV - Motor operated valve

TABLE 3
SHEET 3 OF 7

Time	Event	Action	Response or Nonresponse
1320	SD bus 1 undervoltage.		A and B D/G's tied onto their respective boards. The A and B diesel generators were running and up to speed prior to this but had not received an undervoltage signal to close them onto their boards until this event. This placed all the 4-kV shutdown board equipment upon the diesel generators.
1321	Process computer lost.		No further printout until ~ 1600 hours.
~ 1330	Decision made to depressurize reactor by blowing down to torus.	Manually opened 4 main-steam line relief valves. Checked condensate booster pumps running. Reactor feed pump bypass valve 3-53 checked opened ready to admit water from condensate system.	Reactor pressure decreased; water level decrease Reactor water level dropped from normal 201 inches above top of active fuel to 48 inches abo top of active fuel.
~ 1334	Shutdown bus 2 transferred. (In maintaining a normal configuration on the plant electrical system, the 4-kV shutdown boards will be lined up to feed from the unit or offsite source feed. This allows the diesel generators (D/G's) to be a highly reliable backup, giving two sources of voltage should the need arise.)	Manually initiated by ASE by normal procedure of synchronizing the D/G's with the SD bus; then dropping off D/G feed to the SD board.	Bus 2 energized from unit 2. A,B, and D SD boards transferred to SD bus 2. D/G's remained on running standby.
	Shutdown bus 1 continued to be deenergized.	ASE was unsuccessful in an attempt to manually energize.	
	Shutdown board C remains energized from C D/G.	ASE was unsuccessful in an attempt to manually change C SD board feed from D/G to shutdown bus 2.	Breaker stayed closed for 5 to 10 seconds; then opened. Feed transferred back to D/G.
	D shutdown board deenergized for ~ 5 minutes.	ASE reenergized D board.	D board remained feeding from shutdown bus 2.
~ 1345	Restored unit preferred from unit 2.	ASE manually transferred.	Unit preferred back on both units.
	Reactor steam pressure decreased to 350 psig.	From continued manual operating of relief valves.	Reactor water level increasing as a result of condensate booster pump input.
~ 1355	Water level approaching normal.	Attempted to throttle the feedwater bypass valve 3-53.	No response on feedwater bypass valve 3-53.

TABLE 6
SHEET 4 OF 7

Time	Event	Action	Response or Nonresponse
∼ 1337	Water level going high.	AUO dispatched to bypass valve. Manually closed down on valve.	Restored level to approximately normal.
	Outgoing (PAX) phones and page inoperable for some time.	All operations requiring control room monitoring set up on a periodic in-call basis.	
∼ 1400	Reactor steam pressure at 200 psig.	Operator maintaining reactor steam pressure through four relief valves and level control through RFP bypass and CRD pumps. This had to be controlled via phone communication since the paging system was inoperative.	
	480-V SD boards A and B restored.	ASE manually initiated.	Boards appeared heavily loaded as indicated by loud "humming." Boards remained in service.
		ASE tried to restore reactor 480 MOV boards A and B and reactor 250 MOV boards A and B.	Initially unsuccessful.
	Off-duty maintenance personnel began reporting.	Electricians and operators working to restore these electrical boards by isolating faulted circuits.	Restored approximately two hours later.
∼ 1448	Voltage lost to 4-kV shutdown board C.		There was no control room indication of this condition.
	B D/G found tripped with field breaker open.	ASE closed field breaker on B D/G and brought back to running standby.	
∼ 1500	Attempt made to align one RHR system up for torus cooling and the other for SD cooling.	Four AUO's working in pairs using breathing air packs. Made two entries, but insufficient air supply aborted attempts.	
∼ 1527	Voltage restored to C shutdown board.	ASE found C D/G running at approximately 1/2 speed. Brought D/G to synchronous speed and closed breaker to board.	C shutdown board was deenergized. C shutdown board was also lost from 1545 to 1557. However, at ∼ 1630 C D/G was tied onto the board, its breaker tripped, and prevented C D/G from being used.
∼ 1600	RHR system 1 aligned for torus cooling.	Decision made not to start in this condition since it could not be established that system was charged with water.	This system was subsequently checked for proper alignment and charge and placed in service later.

TABLE 6
SHEET 5 OF 7

Time	Event	Action	Response or Nonresponse
1530	480-V reactor MOV board 1A reenergized.	Result of electrician and operator action.	Energized electrical board allowed main turbine to be placed on turning gear and reactor protection system MG set A to be placed in service which energized trip channel A. Restored power to 1/2 of the process monitoring. Restored power to these ECCS valves feeding from that board, etc.
	Valve restoration to ECCS equipment.	By operator interview, it has been established that the following procedure was used in valve operation where valves were not operable from the control room. Placed select switch in emergency position at the electrical board, ran valve to desired position, reopened breaker, immediately returned to control room and established fact that level was not affected by possible electrical fault misalignment. Tagged valve control switch on unit control board showing valve position. This was a safeguard against draining the vessel down.	Level remained normal.
		ASE observation of conditions as appeared on panel 9-3 in control room.	Core spray loop I A and C pumps appear operable from unit control board. All valves and both pumps had indicating lights. Core spray loop II had a few valves that were inoperable. RHR loop II had a few valves available. RHR loop I--same.
1640	Request to start reactor building exhaust fan to remove smoke and fumes.	Started locally from 480-V reactor building vent board.	Fan responded normally. Dampers controlled manually at the damper.
1700	Request to stop reactor building exhaust fan as airflow appeared to aide fire.	Stopped locally by operator.	Fan stopped.
1800	Relief valves inoperable by remote manual control from benchboard due to loss of instrument and control (I&C) voltage to solenoid in air supply to diaphragm valve in air header to primary containment.	Operator observed lights indicated relief valves open. Other indications suggested that valves were closed. Restarted drywell air compressor. Craftsmen bypassed solenoid valve to provide control air supply to primary containment equipment.	Reactor pressure increasing from 200 psi. The compressor started but discharge isolation prevented airflow to primary containment and relief valve control. Allow relief valve remote manual operation at 2150 hours.

TABLE 6
SHEET 6 OF 7

Time	Event	Action	Response or Nonresponse
1900	Decision made to restart the reactor building exhaust fan.	Manually initiated.	Remained in service.
	PAX telephones restored to unit control room.	ASE cleared problem on phones.	Gave control room the capability to call out.
2008	High torus level from earlier blowdown.	Manually aligned and started RHR drain pump to main condenser hotwell.	Torus level decreased. Hotwell level increased.
2040	Venting drywell via standby gas treatment systems to plant stack. Pressure was about 2.5 psig.	Steamfitters manually opened 2-inch vent to standby gas treatment system.	Drywell pressure decreasing.
2150	Relief valves operable by remote manual control.	Switch left in open position anticipating voltage return. Manually operated relief valves to reduce reactor pressure.	Reactor pressure decreasing from 580 psig maximum.
2200	Secondary containment reestablished.	Operator stationed at reactor building entrance.	Shift engineer approval before entering required. Breathing apparatus required.
2230	ASE made attempt to prove D D/G operable from electrical control board in the control room.		Could not be operated from control room.
		ASE synchronized to D 4-kV shutdown board, locally, picked up load, placed on standby.	Could be used if needed by operating from D shutdown board.
3/23/75 0000	Need for flushing RHR system II prior to placing in shutdown cooling.	Existing procedures could not be used under present circumstance. Two senior reactor operators approved temporary flushing procedure.	System flushed and placed in service at 0410.
0100	Two source range monitors placed in temporary service located on the reactor side of the fire.	Licensed reactor operator stationed at these monitors in the area of unit 1 drywell continuous air monitor unit.	Established capability to monitor core. 10 counts per second reading on monitors.
0130	Torus cooling continues to be a necessity as blowdown continues.	Valves aligned manually by operators and system placed in service.	Decreasing torus temperature.
0212	Torus level instrumentation in service.		Level indicated +1". (Normal level is indicated as 0 with a deviation of + or - 5".)
0245	Restoration of equipment had progressed to the point that A and C core spray pumps could be tested from panel 9-3 in unit control room.	Operator action from unit control room.	Pumps and injection valves operable, thus giving part of the ECCS equipment available if needed.
0410	Shutdown cooling achieved by normal flow path.	Manually aligned system.	Allowed operator control of vessel temperature.

BROWNS FERRY UNIT 2

SEQUENCE OF
SIGNIFICANT OPERATIONAL EVENTS
AT TIME OF FIRE

Time	Event	Action	Response or Nonresponse
3/22/75 Prior to 1300	Initial Condition	Routine Operation	Unit Load 1,100 MWe
∼ 1300	4-kV Shutdown bus 2 deenergized (relay action).		Lost reactor protection system (RPS) motor generator (MG) set 2B; 1/2 scram on RPS giving red lights on panel 9-5; reactor recirculation pump automatically decreasing reactor power. Lost voltage to instrument and control bus B. Lost indicating lights on system I residual heat removal (RHR) and system I core spray - alarms on RHR and core spray "start," "overcurrent," "pump trip."
	Operator observed decreasing reactor power indication and many scram alarms on control panel.	Operator placed reactor mode switch in "shut-down" and inserted nuclear instrumentation (source and intermediate range).	Reactor scrammed inserting all control rods.
∼ 1301	Reactor water level dropped and returned to normal (normal reaction from trip).	Tripped reactor feed pumps A, B, and C - Tripped turbine. Tripped exciter field breaker and opened generator motor-operated disconnects (MOD's).	Equipment response normal.
1308	Main-steam isolation valves (MSIV) closed.	Operator initiated reactor core isolation cooling (RCIC) for level control; initiated high pressure cooling injection (HPCI) for heat sink. Manually initiated relief valves for pressure control.	Equipment response normal. After this start and before ∼ 1415 RCIC and HPCI tripped several times from high reactor water level. Neither of these could be restarted with the controller in "manual." Operator was unable to get any signal from the subpanel control in "manual." Pumps would start with controller in "automatic." At ∼ 1345 HPCI was restarted and brought to ∼ 1/4 speed. It held for about 1 minute. The speed then dropped off with no further response from HPCI; thereafter it was unavailable.

TABLE 1
1 OF 8

Time	Event	Action	Response or Nonresponse
~ 1320	Lost remote manual operability of relief valves depriving operator of ability to reduce reactor pressure below set point.	Unit 2 assistant shift engineer made an attempt to operate relief valves from backup control panel but was unsuccessful.	Relief valves continued to lift on pressure. Maintaining reactor pressure at 1080 psig and below.
	Torus temperature increasing due to relief valve discharge into torus.	Placed the following pumps in service to establish torus cooling: B RHR pump, B2 RHR service water pump, and B1 emergency equipment cooling water pump.	Torus cooling established at ~ 1330. Torus temperature did not exceed 135° F.
~ 1400	Reactor depressurizing apparently from a relief valve that had lifted on pressure and stuck open.	No indication of coolant leak, and pressure decreasing at desired rate.	Reactor pressure decreasing at desired rate: 130 psig at 1500 65 psig at 1900 10 psig at 2040
~ 1415	Pressure starting to decrease.	Placing the condensate system in service.	Anticipating the pressure level that reactor vessel could be supplied from that source.
	Remote manual operability of relief valve restored.	Maintenance and operations personnel working during this period of non-operation checking the instrument and control (I&C) voltage to solenoids, drywell air compressor for proper operation and cutting in the backup control air supply. It is uncertain which of these operations reestablished remote operability of relief valves.	Gave operator discretion on relief valve operation.
~ 1430	Loss of some reactor water level instrumentation.	Determined that level indicator 362 appeared to be reliable and that 2 Yarway level indicators in backup control center corresponded with this indication.	Reactor water level never decreased below 160" above the top of the active fuel. Other level indication began to respond at ~ 1450.
~ 1450	Torus level increasing due to relief valve discharge.	Manually aligned RHR drain pump to transfer torus water to condenser hotwell.	Torus level never increased above + 5".
~ 1957	Voltage restored to C shutdown (SD) board.	Restored power to 480-V SD board 2A by manual operator action. Started B2 RHR service water pump on. Started B RHR pump on. Placed turbine on turning gear (T.G.)	These power supplies allowed turbine to be placed on T.G. and B RHR pump to be tested.
~ 2010	Condenser heat sink available. Allowed use of turbine bypass valves to reduce pressure.	Cleared up electrical trouble with mechanical vacuum pumps and established vacuum in main condenser.	Vacuum above 7" Hg allowing opening of turbine bypass valves for steam admission.
~ 2020	Torus temperature within limits; shut down torus cooling to allow flushing of lines for reactor shutdown cooling.	Aligned valves and flushed system II prior to placing system II in reactor shutdown cooling mode.	
~ 2045	Reactor pressure at 10 psig.	Opened reactor headvents.	
~ 2240	Reactor in shutdown cooling using RHR system II.	Manually aligned system.	Shutdown cooling achieved by normal flow path.

UNITED STATES GOVERNMENT

Memorandum

TENNESSEE VALLEY AUTHORITY

TO : Preliminary Investigation Committee for Fire at Browns Ferry Nuclear Plant

FROM : James E. Watson, Manager of Power, 818 PRB-C

DATE : March 31, 1975

SUBJECT: ESTABLISHMENT OF COMMITTEE TO INVESTIGATE THE MARCH 22, 1975, FIRE AT BROWNS FERRY

This memorandum will confirm and formalize the establishment on March 23, 1975, of a preliminary committee to investigate the fire at Browns Ferry on March 22, 1975. The committee reported to the site on March 24, 1975. It is composed of the following members:

H. S. Fox, Chairman	- Division of Power Production
M. N. Sprouse	- Division of Engineering Design
Charles Bonine	- Division of Construction
David G. Powell	- Division of Law
Felix A. Szczepanski	- Power Manager's Office, Safety Review Board Staff
Harry S. Collins	- Secretary, Safety Review Board

Your participation on this committee is greatly appreciated and while we recognize that it will in all probability cause you personal hardships, we believe it is extremely important to give the committee your full effort for the immediate future.

Attached is an outline of the committee's assignment. Many of the points covered in the outline have already been addressed but this will confirm the assignment to the committee.

Again, I appreciate your serving on this committee and if I can help you in any way, please let me know.

Attachment

CC: R. H. Davidson, 303 PRB-C
J. E. Gilleland, 831 PR1-C
G. H. Kimmons, 607 UB-K
R. H. Marquis, 629 NSB-K
Nuclear Safety Review Board, 210 PRB-C
H. G. Parris, 403 PRB-C
E. F. Thomas, 716 EB-C

PRELIMINARY INVESTIGATION

of

BROWNS FERRY FIRE

Preliminary Incident Evaluation Committee

Scope

Conduct an early fact-finding investigation into events leading up to, during and after the incident until plant conditions were stabilized. Consider actions of people involved, applicable administrative controls, response and nonresponse of plant systems, alternative measures taken, utilization and adequacy of plant firefighting equipment, assessment of extent of damage, status of plant and plant systems; determine if there were any radioactive releases, radioactive exposure and/or injury to plant personnel, and if so, determine the extent thereof; ensure preservation of adequate incident documentation; and provide a point of reference for other evaluations.

Other Considerations

The committee should:

- **Use its discretion in extending its scope and in carrying out its functions to achieve its objectives.**
- **Recommend and seek approval for additional committee members (i.e., consultants or other TVA members); consider qualifications.**
- **Make preliminary report to the Manager of Power within two weeks (by April 7).**

Membership

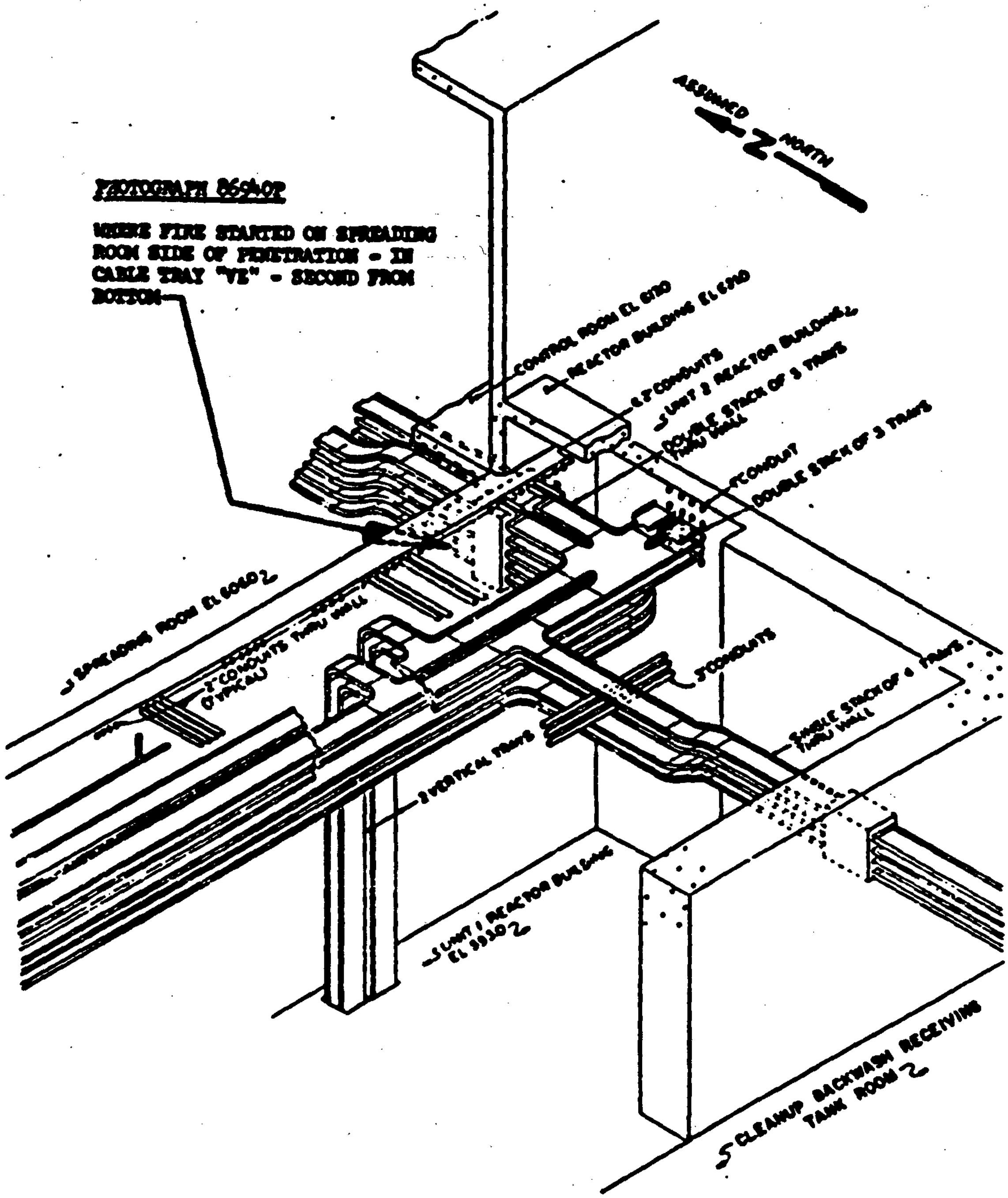
Harry Fox (Chairman)	- DPP
F. A. Szczepanski	- NSRB
C. Bonine	- DEC
M. Sprouse	- DED
D. Powell	- Law
E. S. Collins	- NSRB Staff

APPENDIX A
SHEET 2 OF 2

APPENDIX B

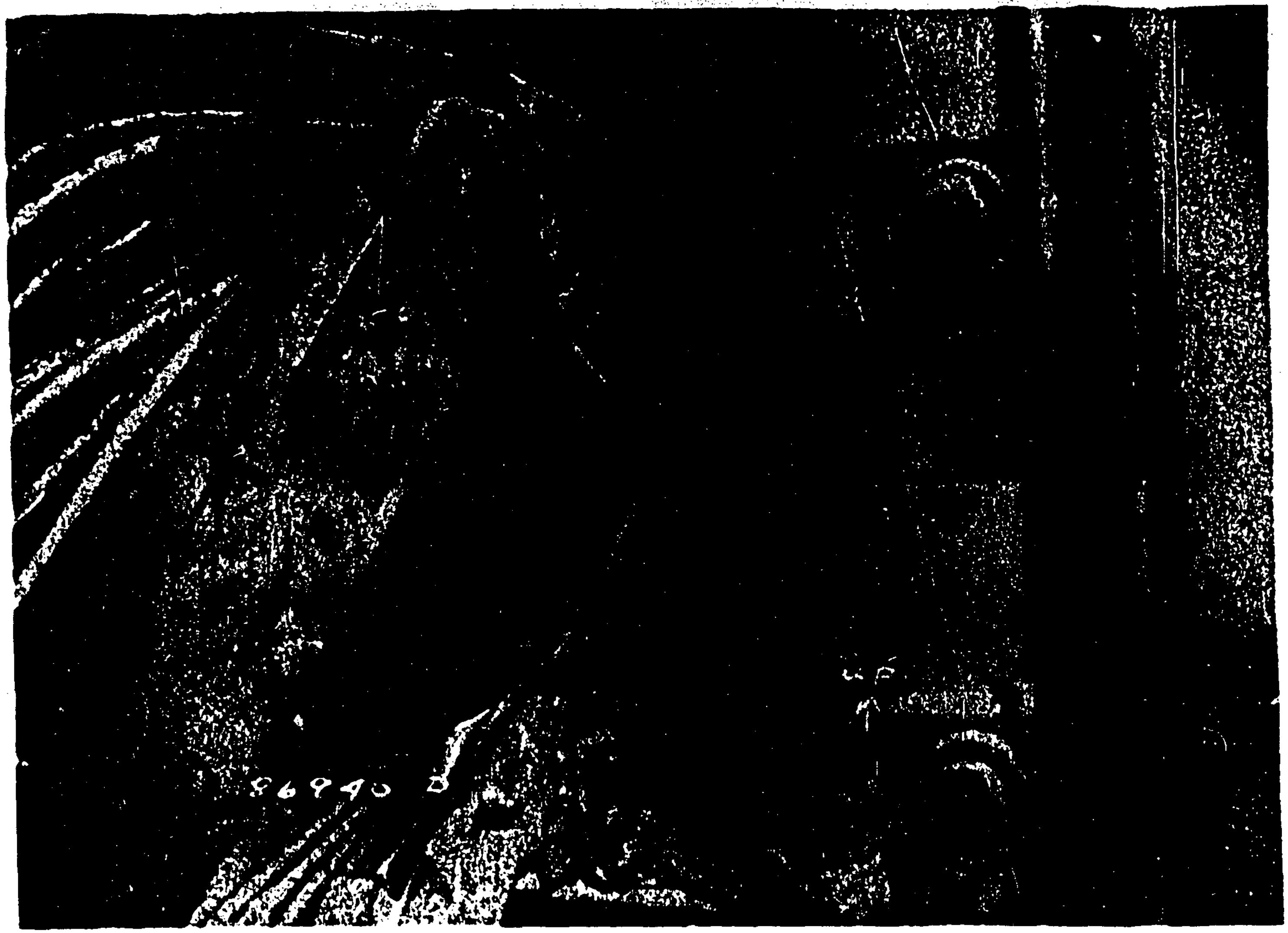
KEY PHOTOGRAPHS OF FIRE AREA

INDEX



AREA OF FIRE

1



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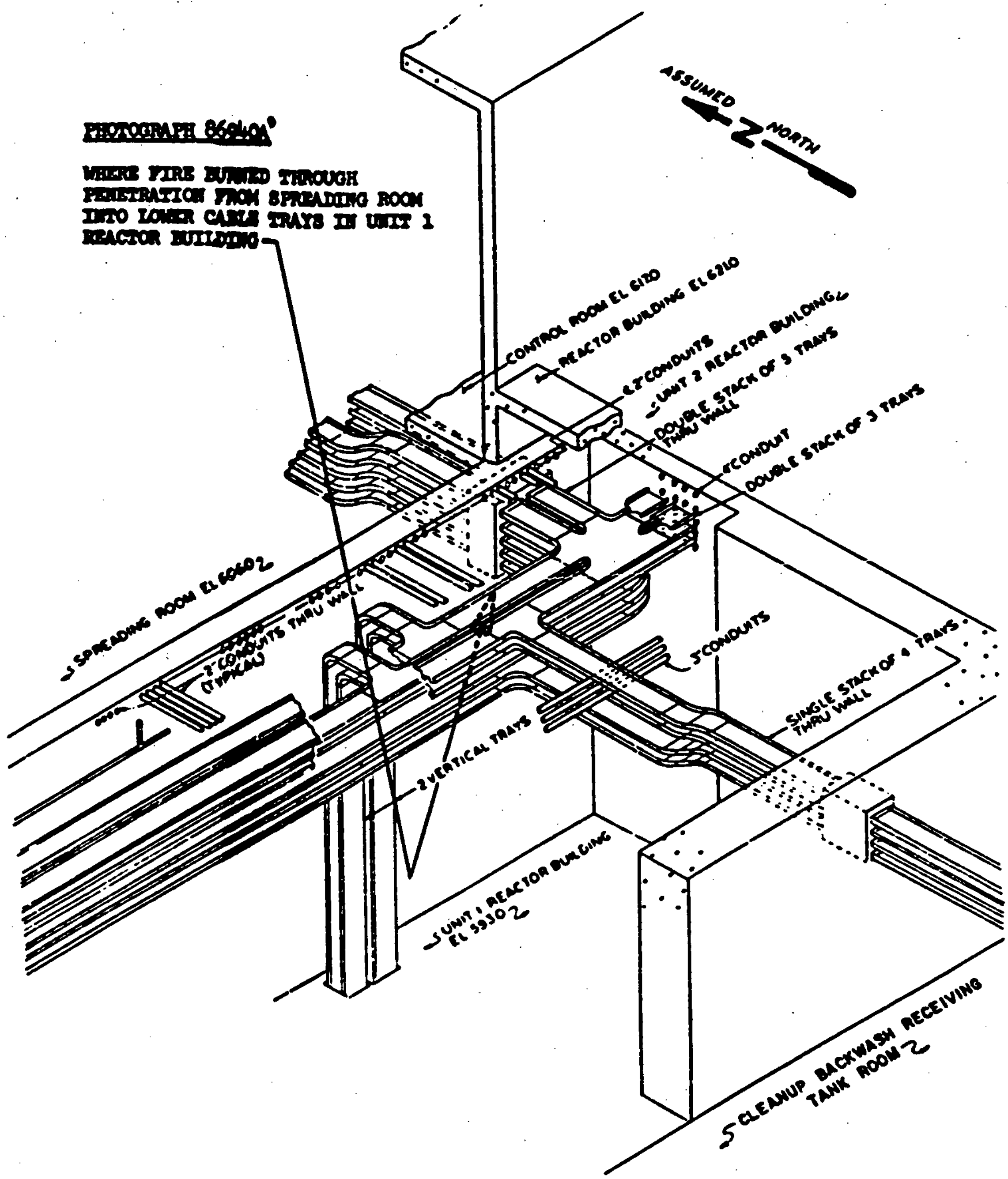
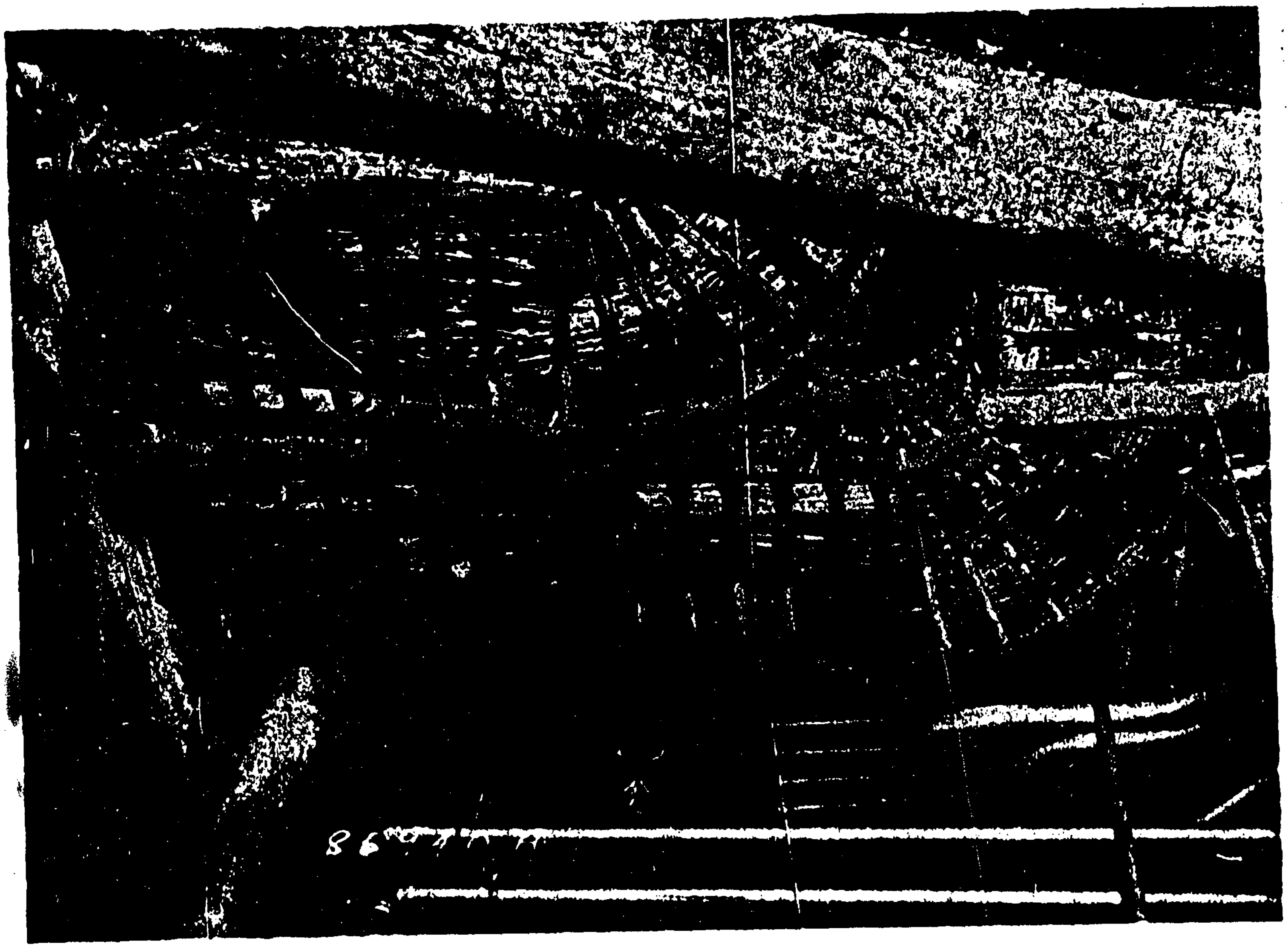

PHOTOGRAPH 869404
WHERE FIRE BURNED THROUGH PENETRATION FROM SPREADING ROOM INTO LOWER CABLE TRAYS IN UNIT 1 REACTOR BUILDING
ASSUMED NORTH
CONTROL ROOM EL 6120
REACTOR BUILDING EL 6210
2-3" CONDUITS
UNIT 2 REACTOR BUILDING
DOUBLE STACK OF 3 TRAYS THRU WALL
3" CONDUIT
DOUBLE STACK OF 3 TRAYS
SPREADING ROOM EL 6060
3" CONDUITS THRU WALL (TYPICAL)
3 CONDUITS
SINGLE STACK OF 4 TRAYS THRU WALL
2 VERTICAL TRAYS
UNIT 1 REACTOR BUILDING EL 5930
CLEANUP BACKWASH RECEIVING TANK ROOM

AREA OF FIRE



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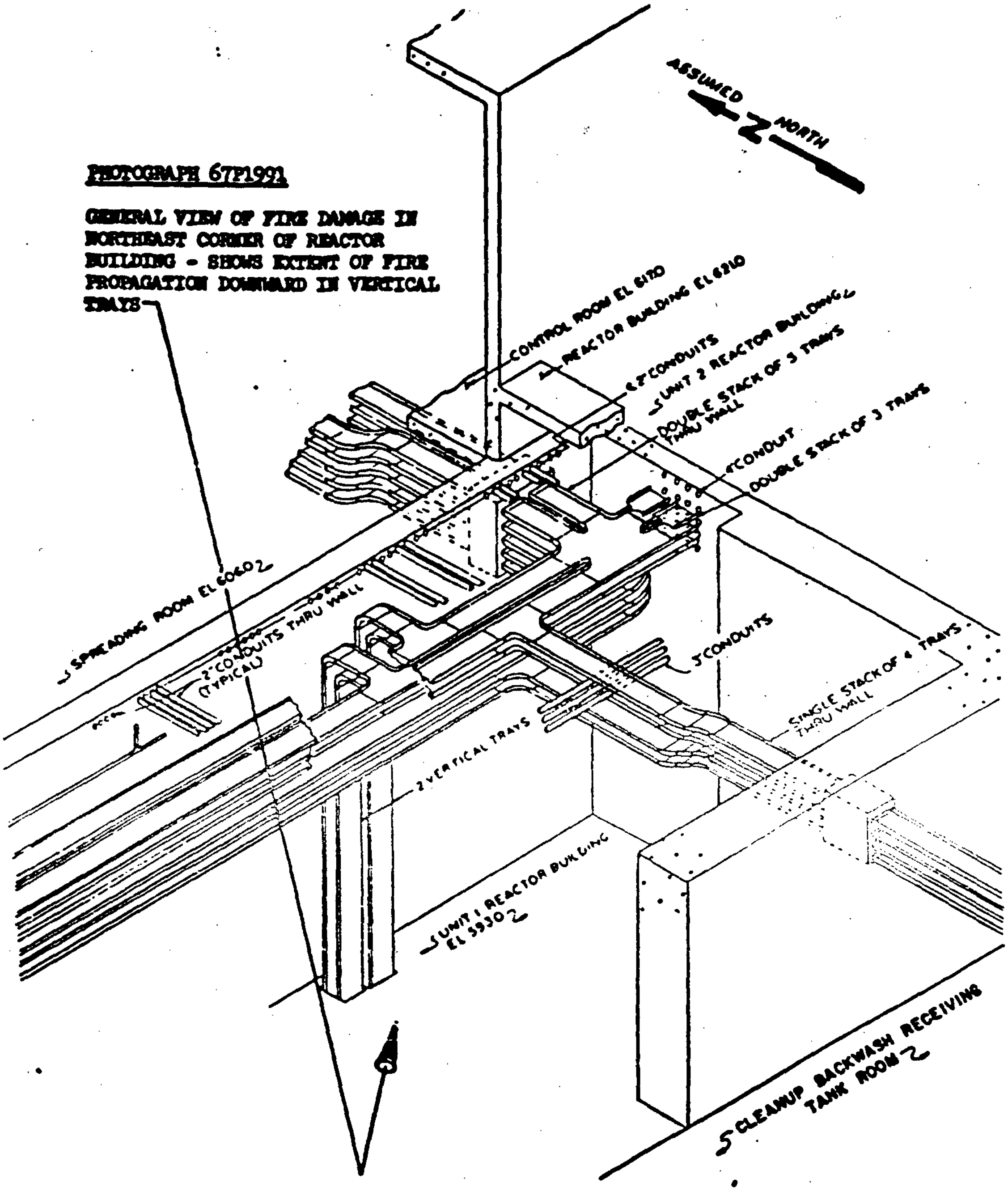
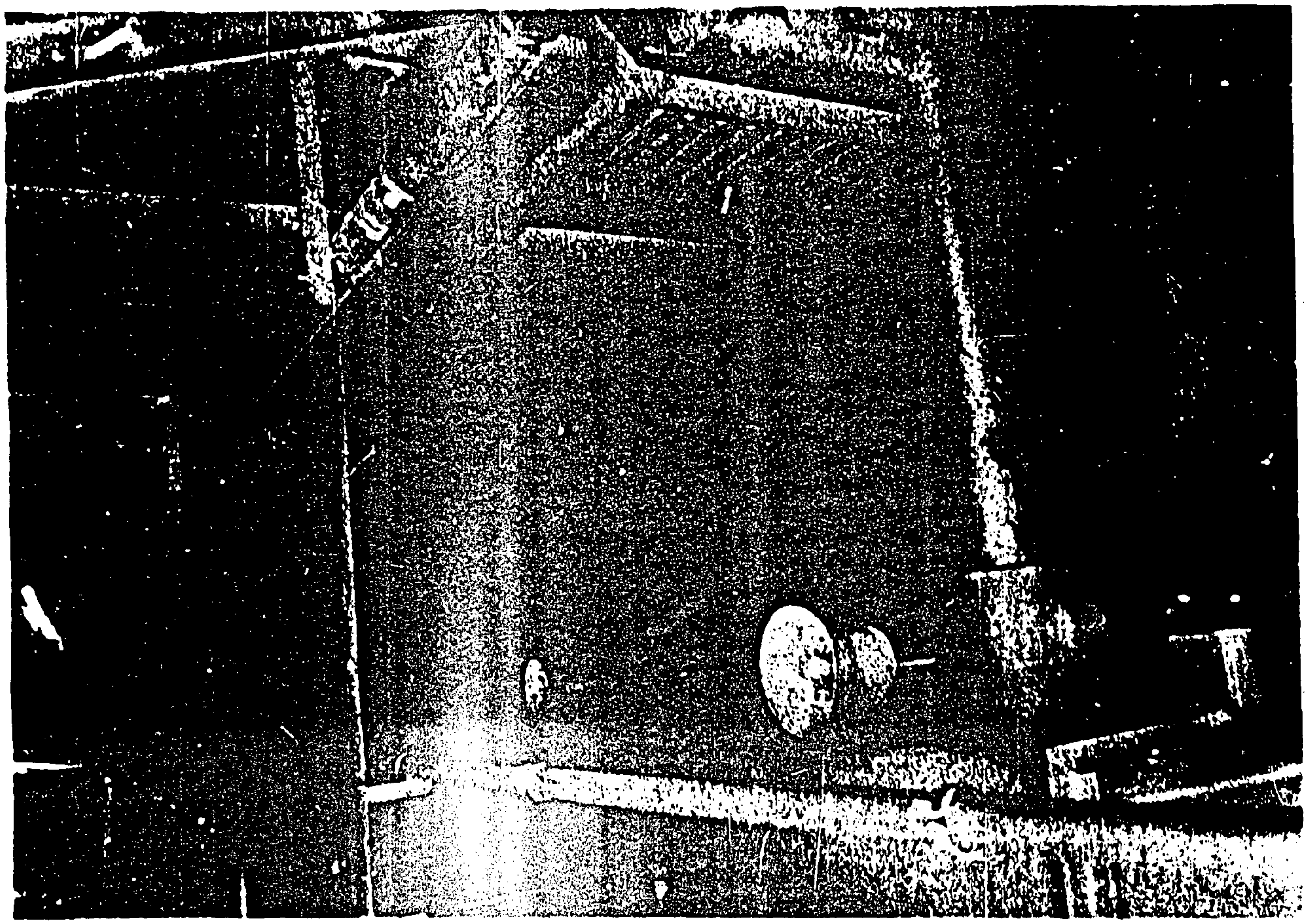


PHOTOGRAPH 67P1991
GENERAL VIEW OF FIRE DAMAGE IN NORTHEAST CORNER OF REACTOR BUILDING - SHOWS EXTENT OF FIRE PROPAGATION DOWNWARD IN VERTICAL TRAYS
ASSUMED NORTH
CONTROL ROOM EL 617.0
REACTOR BUILDING EL 621.0
6 2" CONDUITS
UNIT 2 REACTOR BUILDING
DOUBLE STACK OF 3 TRAYS THRU WALL
4" CONDUIT
DOUBLE STACK OF 3 TRAYS
SPREADING ROOM EL 606.0
2" CONDUITS THRU WALL (TYPICAL)
3 CONDUITS
SINGLE STACK OF 4 TRAYS THRU WALL
2 VERTICAL TRAYS
UNIT 1 REACTOR BUILDING EL 593.0
CLEANUP BACKWASH RECEIVING TANK ROOM

AREA OF FIRE



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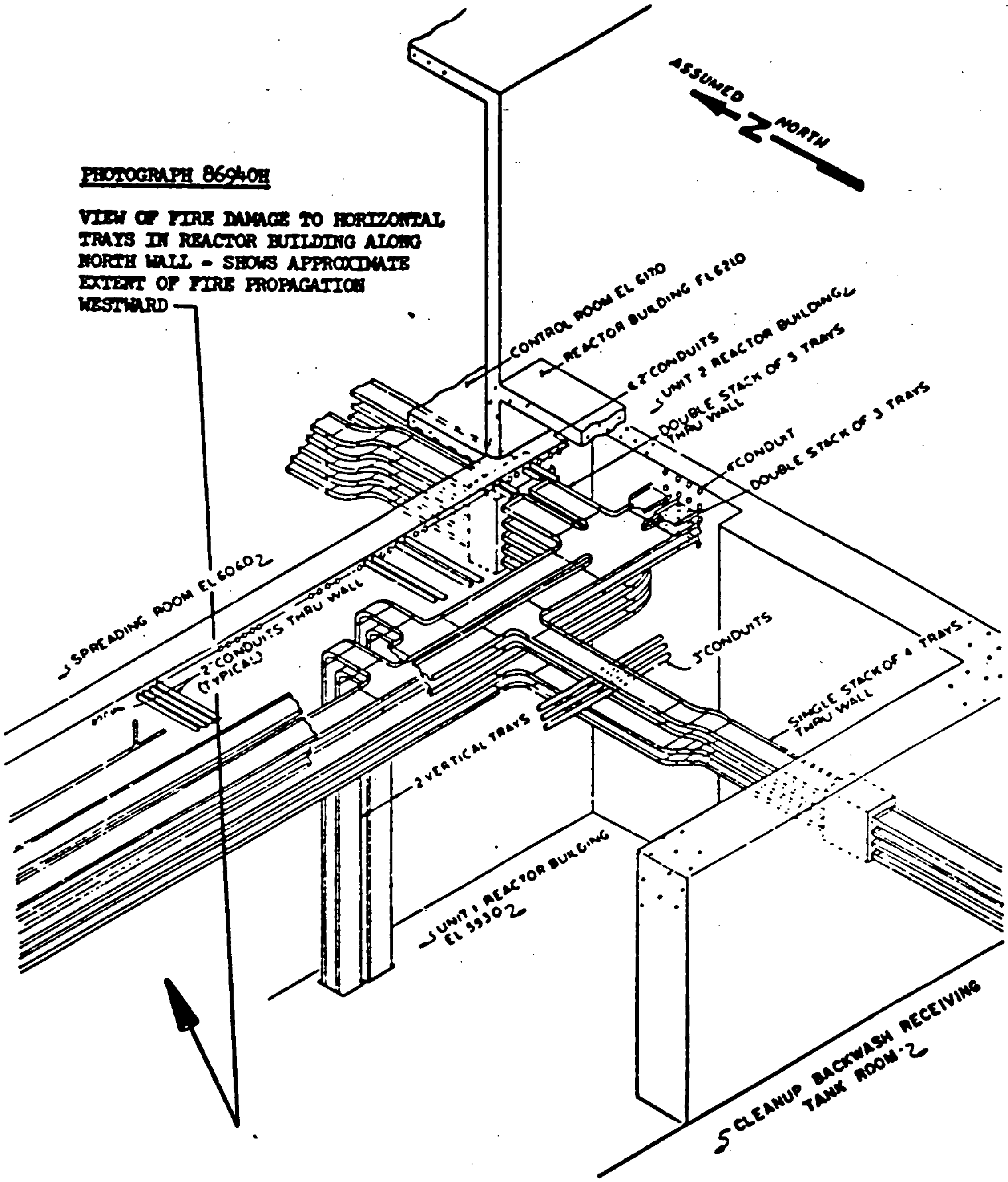
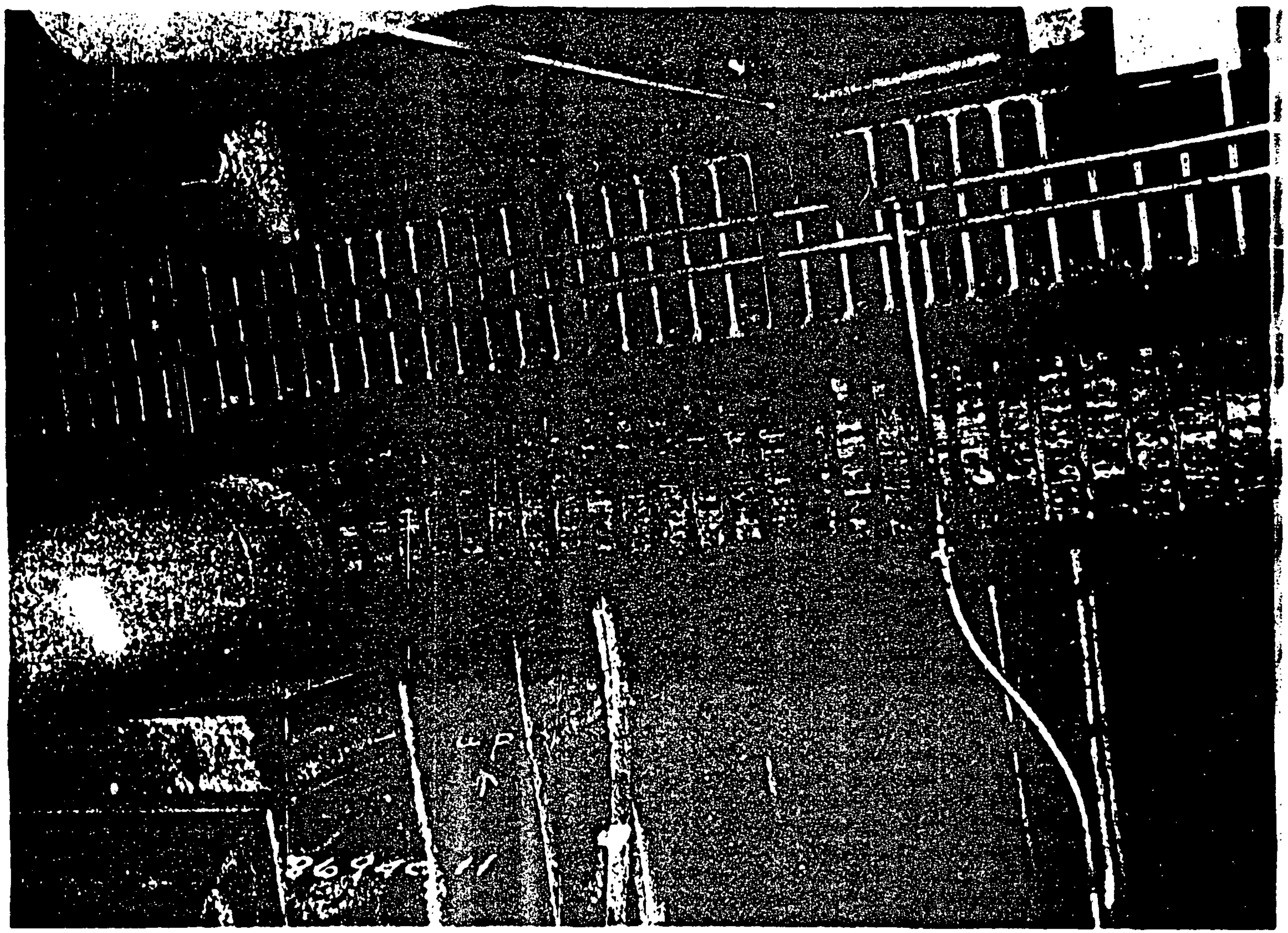

ASSUMED NORTH
PHOTOGRAPH 86940H
VIEW OF FIRE DAMAGE TO HORIZONTAL TRAYS IN REACTOR BUILDING ALONG NORTH WALL - SHOWS APPROXIMATE EXTENT OF FIRE PROPAGATION WESTWARD
CONTROL ROOM EL 6120
REACTOR BUILDING EL 6210
6 2" CONDUITS
UNIT 2 REACTOR BUILDING
DOUBLE STACK OF 3 TRAYS THRU WALL
CONDUIT
DOUBLE STACK OF 3 TRAYS
SPREADING ROOM EL 6060
2" CONDUITS THRU WALL (TYPICAL)
3 CONDUITS
SINGLE STACK OF 4 TRAYS THRU WALL
2 VERTICAL TRAYS
UNIT 1 REACTOR BUILDING EL 5930
CLEANUP BACKWASH RECEIVING TANK ROOM

AREA OF FIRE



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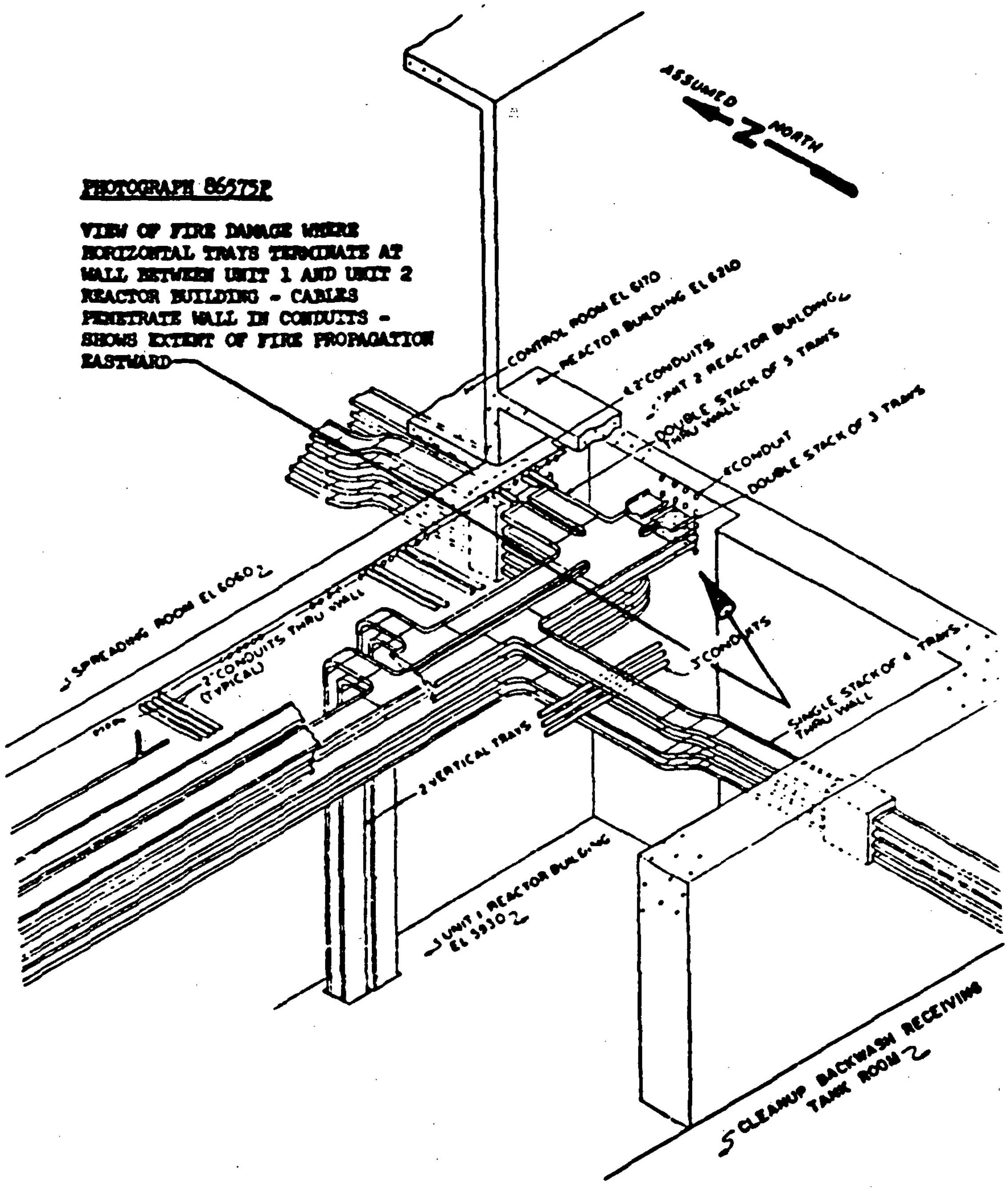
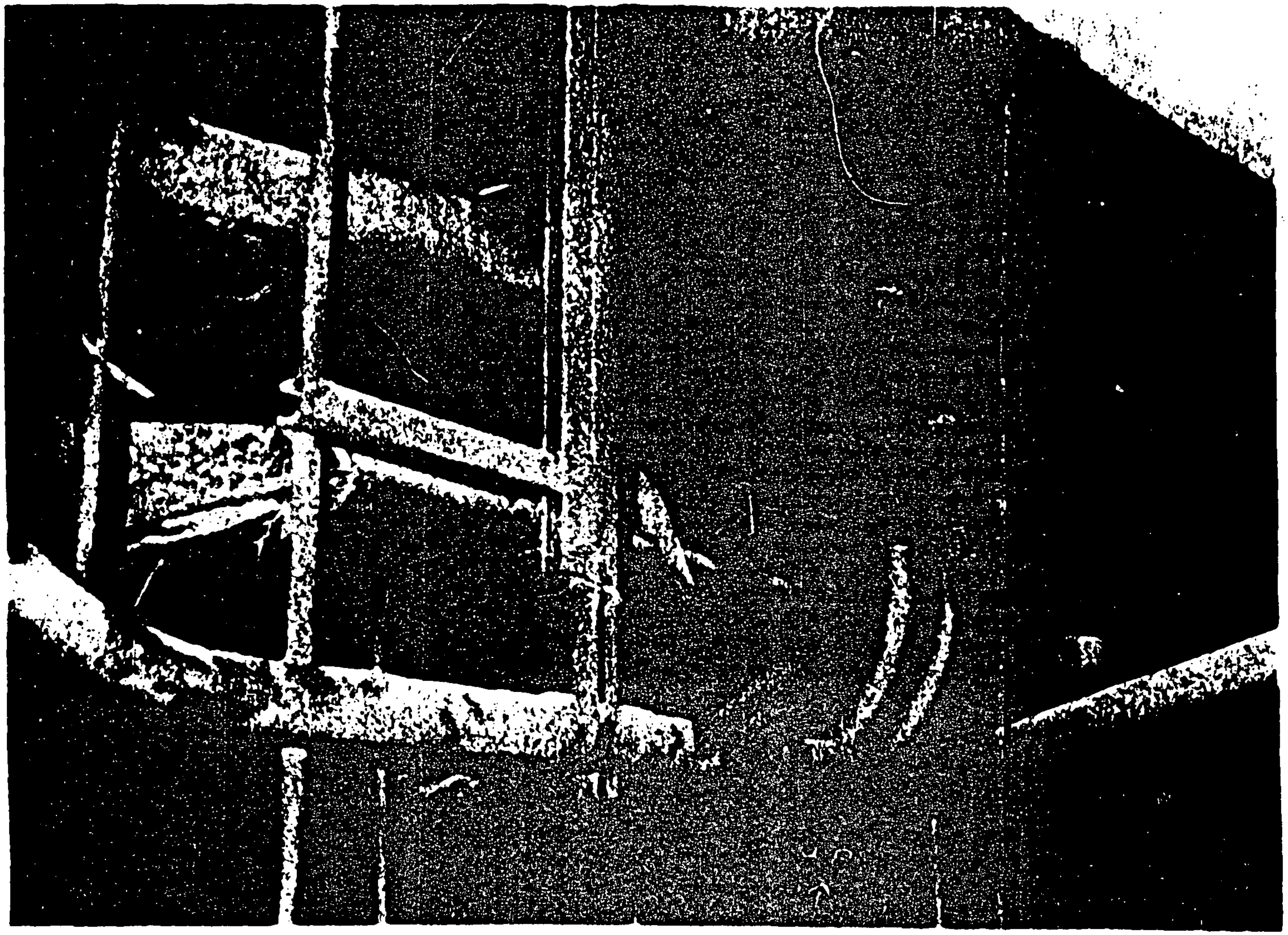
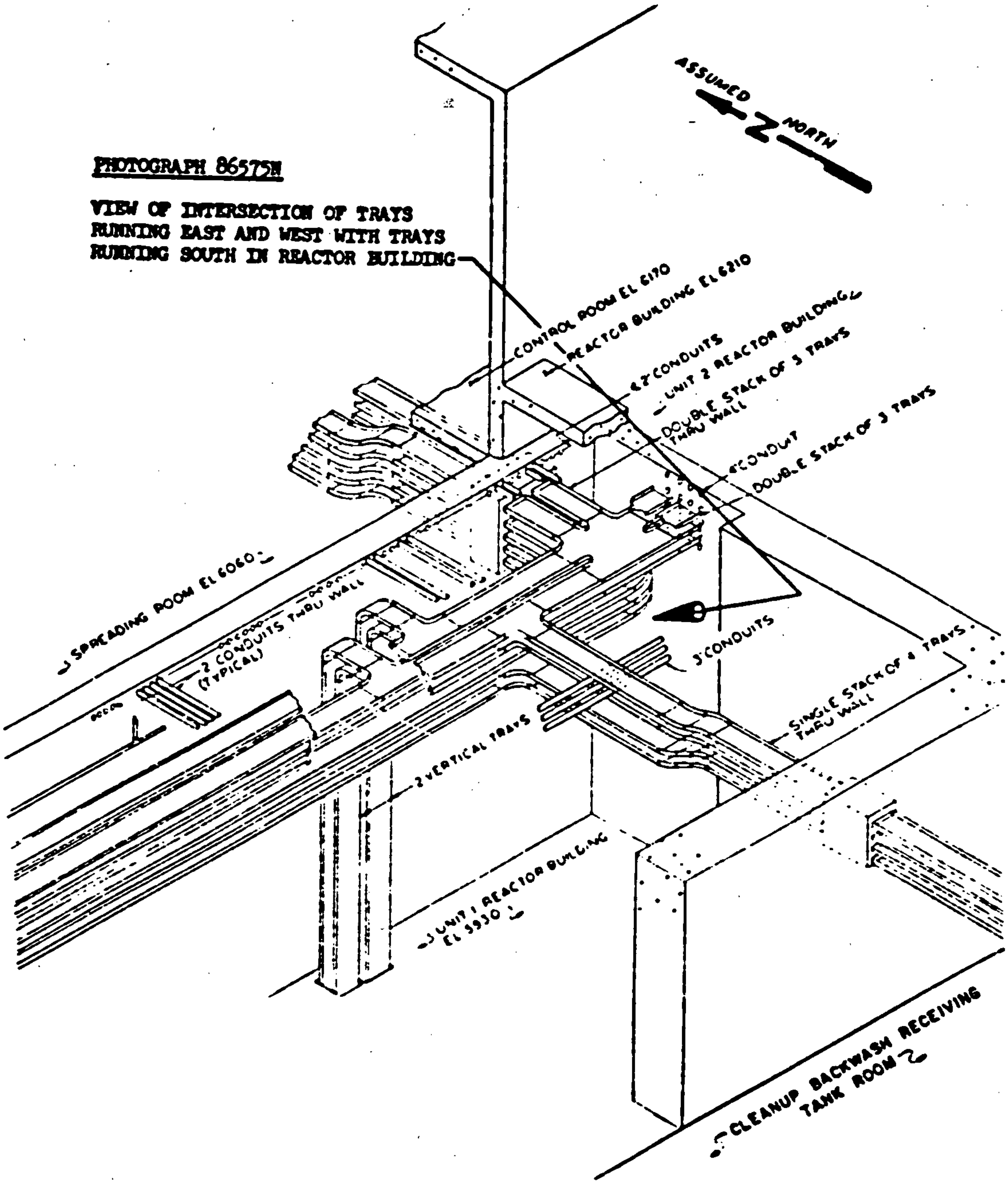


AREA OF FIRE

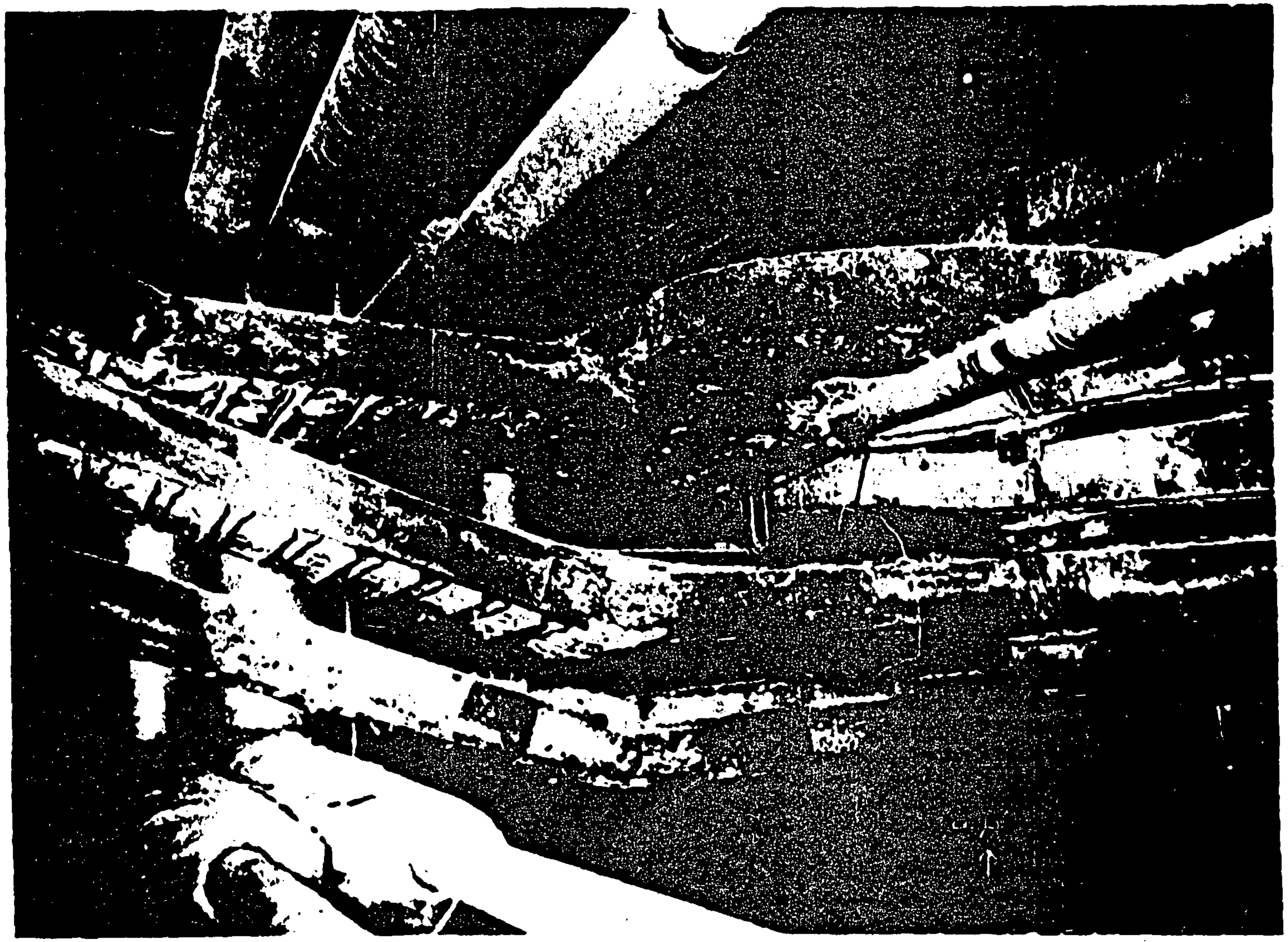
5



86575r


ASSUMED NORTH
N
PHOTOGRAPH 86575N
VIEW OF INTERSECTION OF TRAYS RUNNING EAST AND WEST WITH TRAYS RUNNING SOUTH IN REACTOR BUILDING
CONTROL ROOM EL 6170
REACTOR BUILDING EL 6210
UNIT 2 REACTOR BUILDING
DOUBLE STACK OF 3 TRAYS THRU WALL
DOUBLE STACK OF 3 TRAYS
3" CONDUITS
SINGLE STACK OF 4 TRAYS THRU WALL
SPREADING ROOM EL 6060
2 CONDUITS THRU WALL (TYPICAL)
2 VERTICAL TRAYS
UNIT 1 REACTOR BUILDING EL 5930
CLEANUP BACKWASH RECEIVING TANK ROOM

AREA OF FIRE



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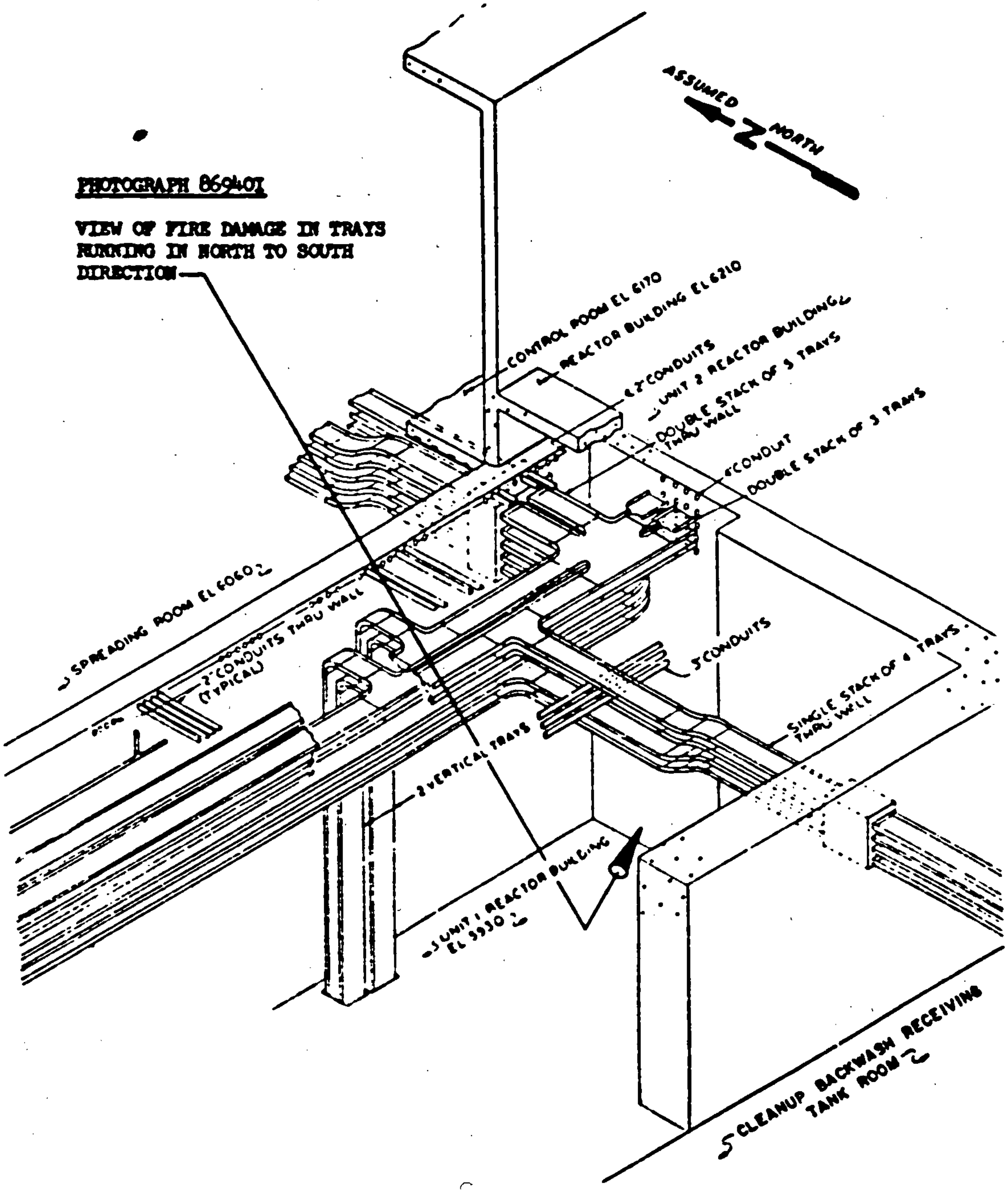
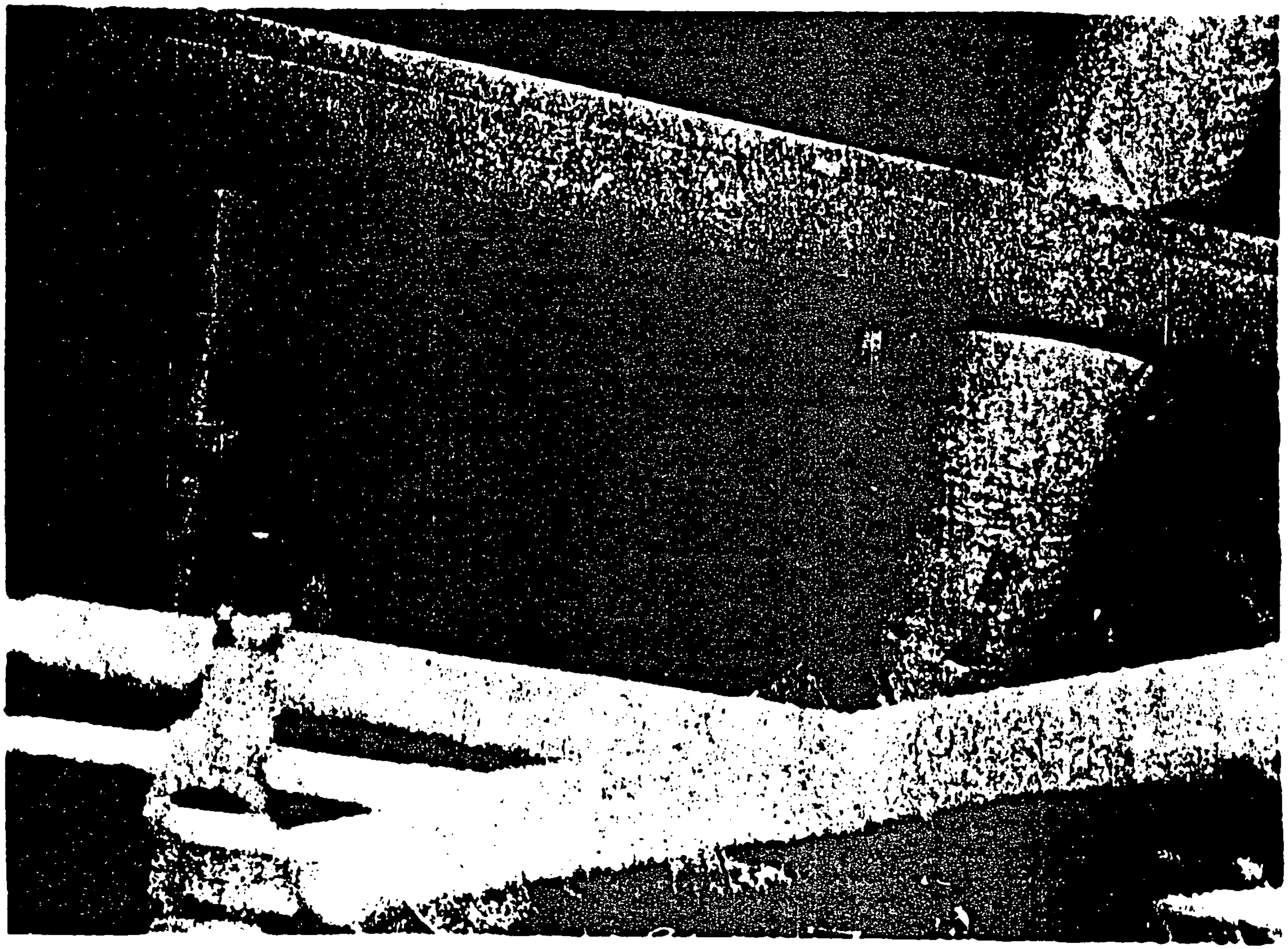

ASSUMED
N
NORTH
PHOTOGRAPH 869401
VIEW OF FIRE DAMAGE IN TRAYS RUNNING IN NORTH TO SOUTH DIRECTION
CONTROL ROOM EL 6170
REACTOR BUILDING EL 6210
4 2" CONDUITS
UNIT 2 REACTOR BUILDING
DOUBLE STACK OF 3 TRAYS THRU WALL
CONDUIT
DOUBLE STACK OF 3 TRAYS
SPREADING ROOM EL 6060
2" CONDUITS THRU WALL
2" CONDUITS (TYPICAL)
3 CONDUITS
SINGLE STACK OF 4 TRAYS THRU WALL
2 VERTICAL TRAYS
UNIT 1 REACTOR BUILDING EL 5930
CLEANUP BACKWASH RECEIVING TANK ROOM

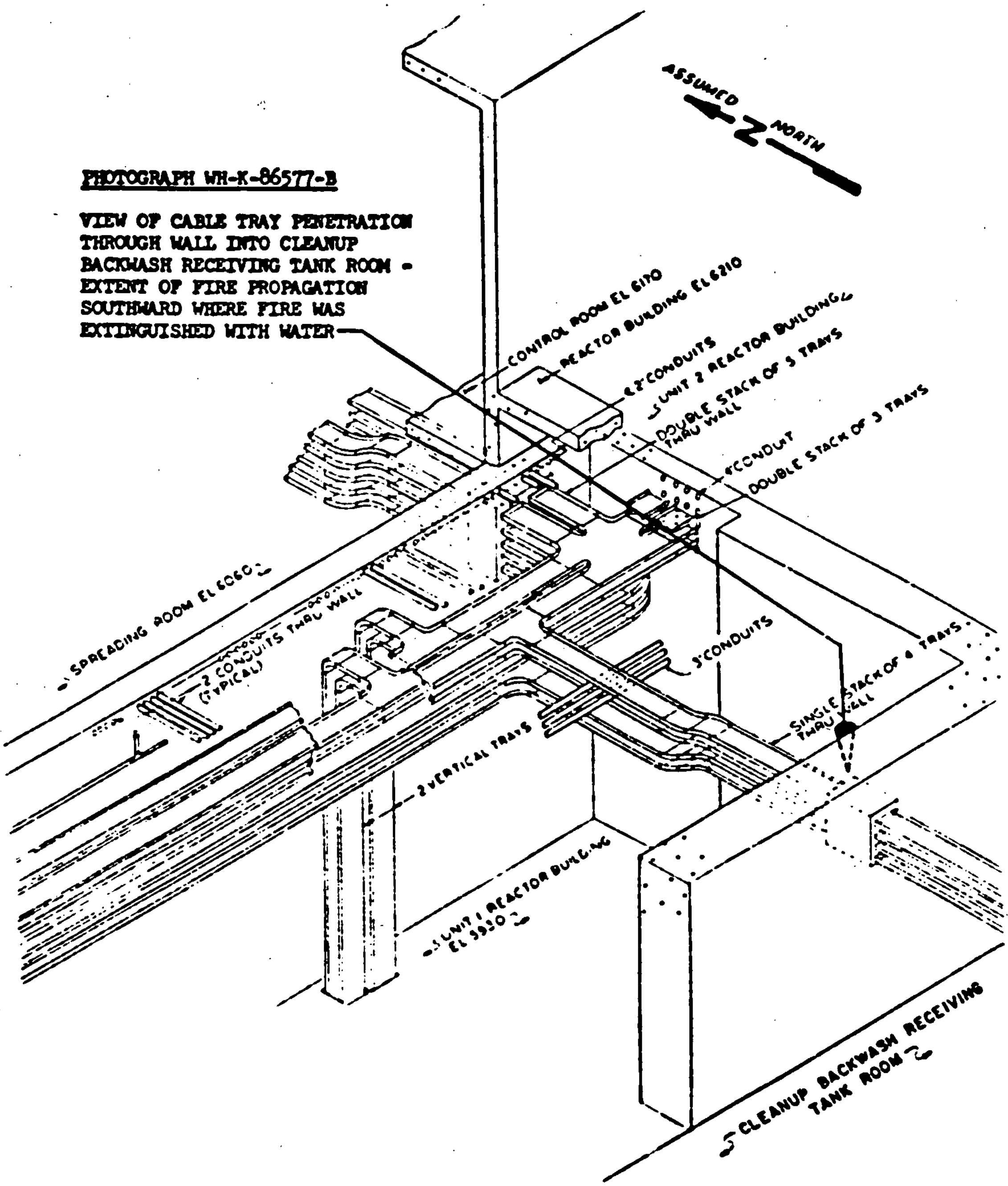
AREA OF FIRE



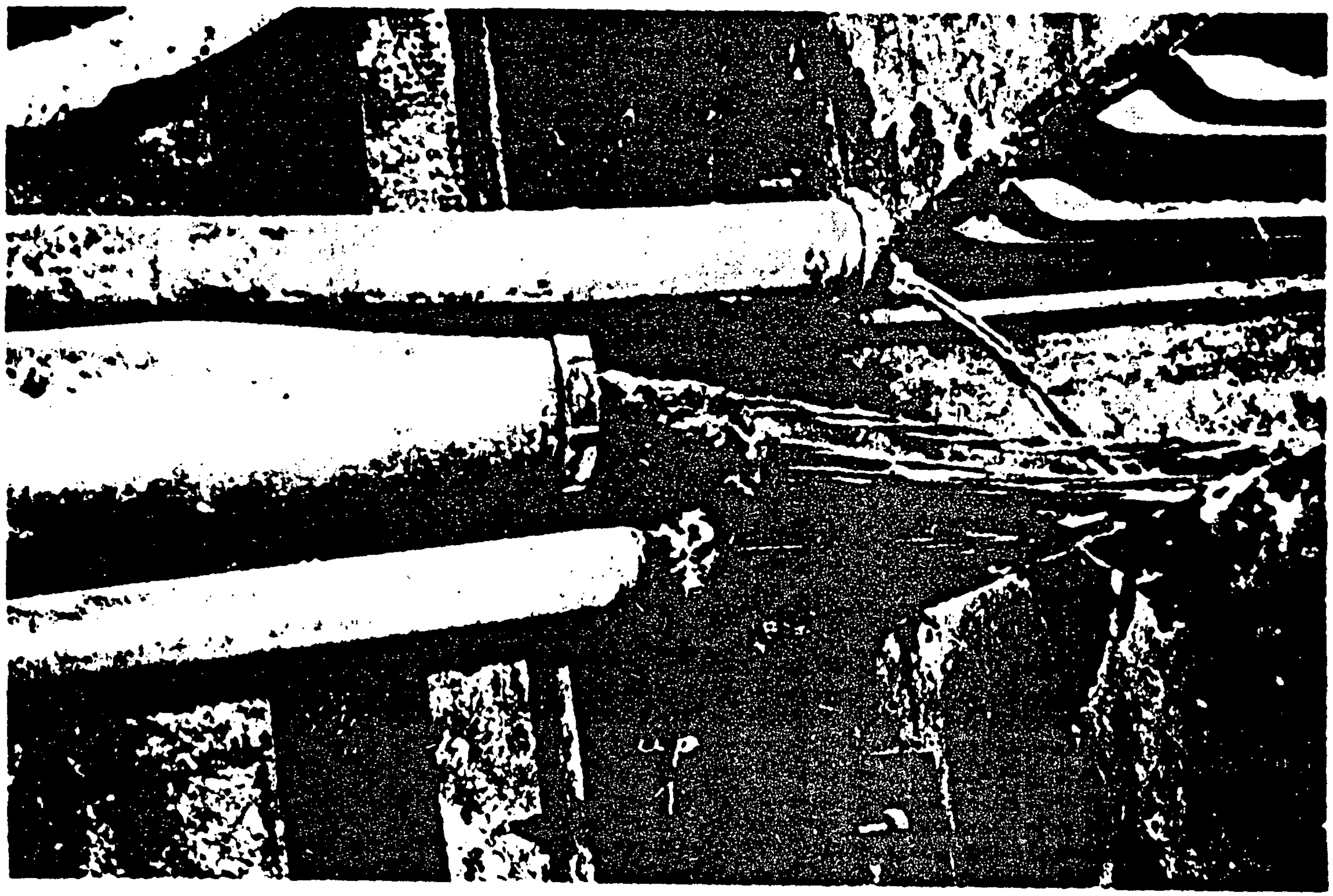
S69401

PHOTOGRAPH WH-K-86577-B

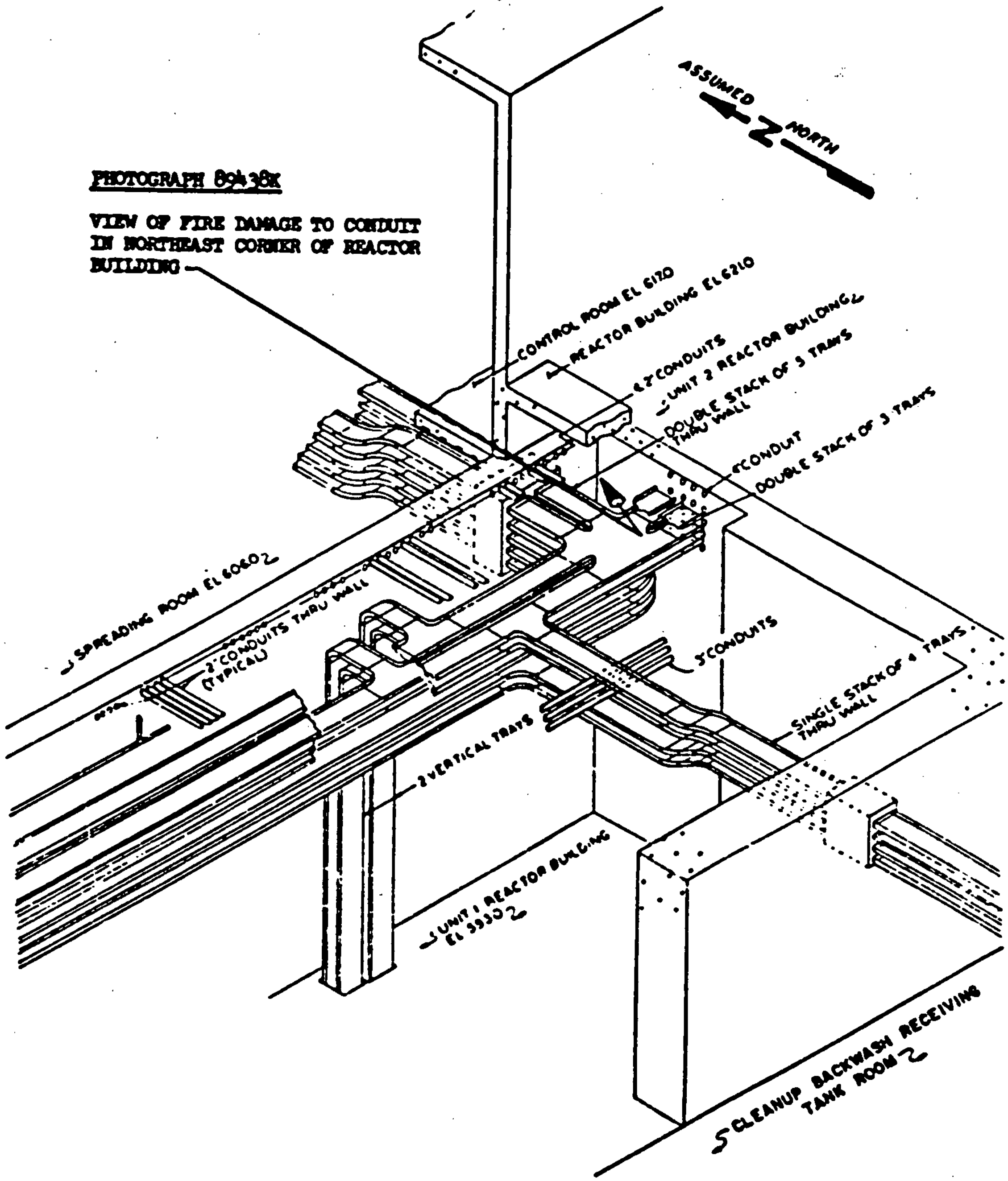
VIEW OF CABLE TRAY PENETRATION
THROUGH WALL INTO CLEANUP
BACKWASH RECEIVING TANK ROOM -
EXTENT OF FIRE PROPAGATION
SOUTHWARD WHERE FIRE WAS
EXTINGUISHED WITH WATER



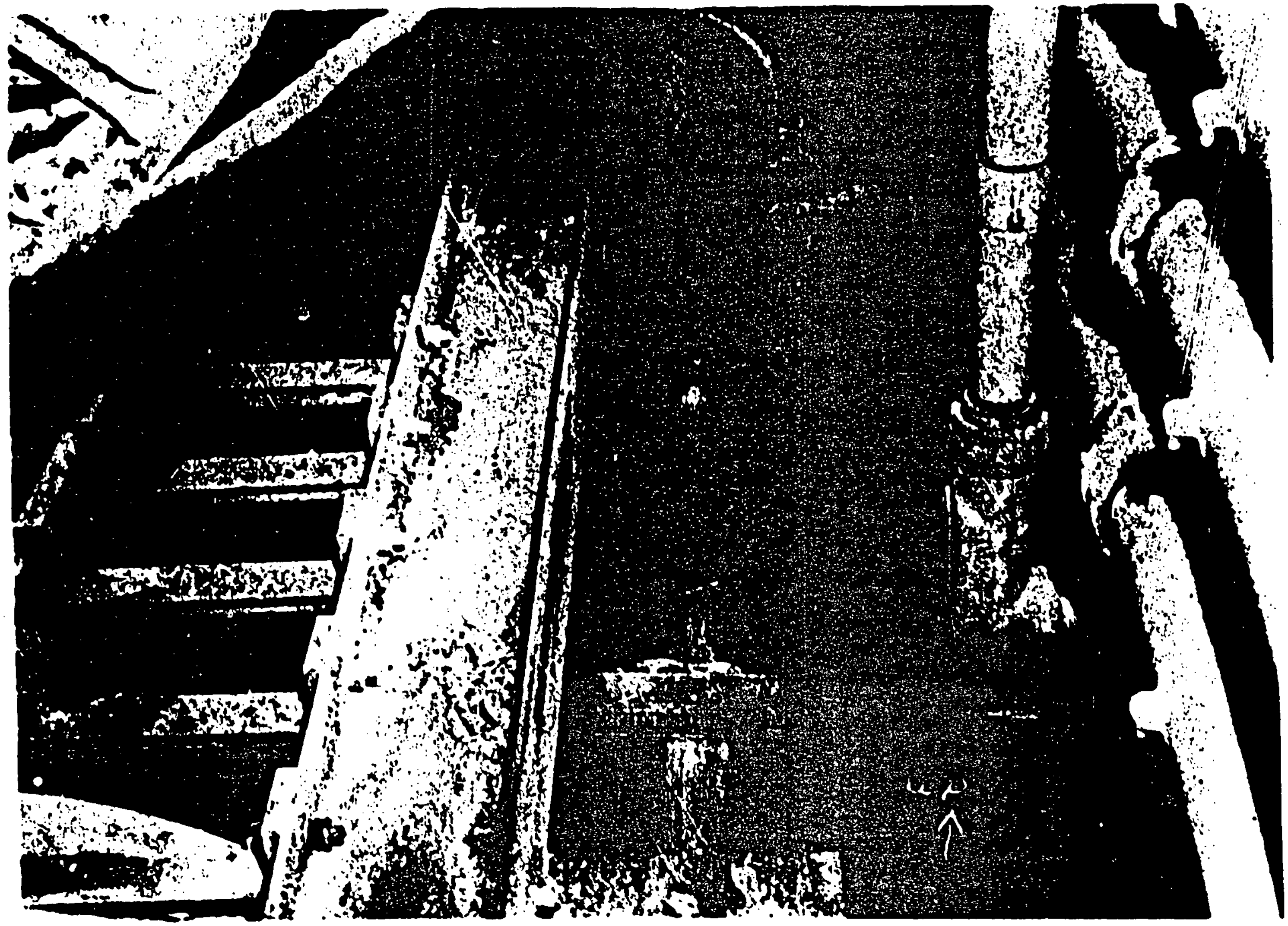
AREA OF FIRE



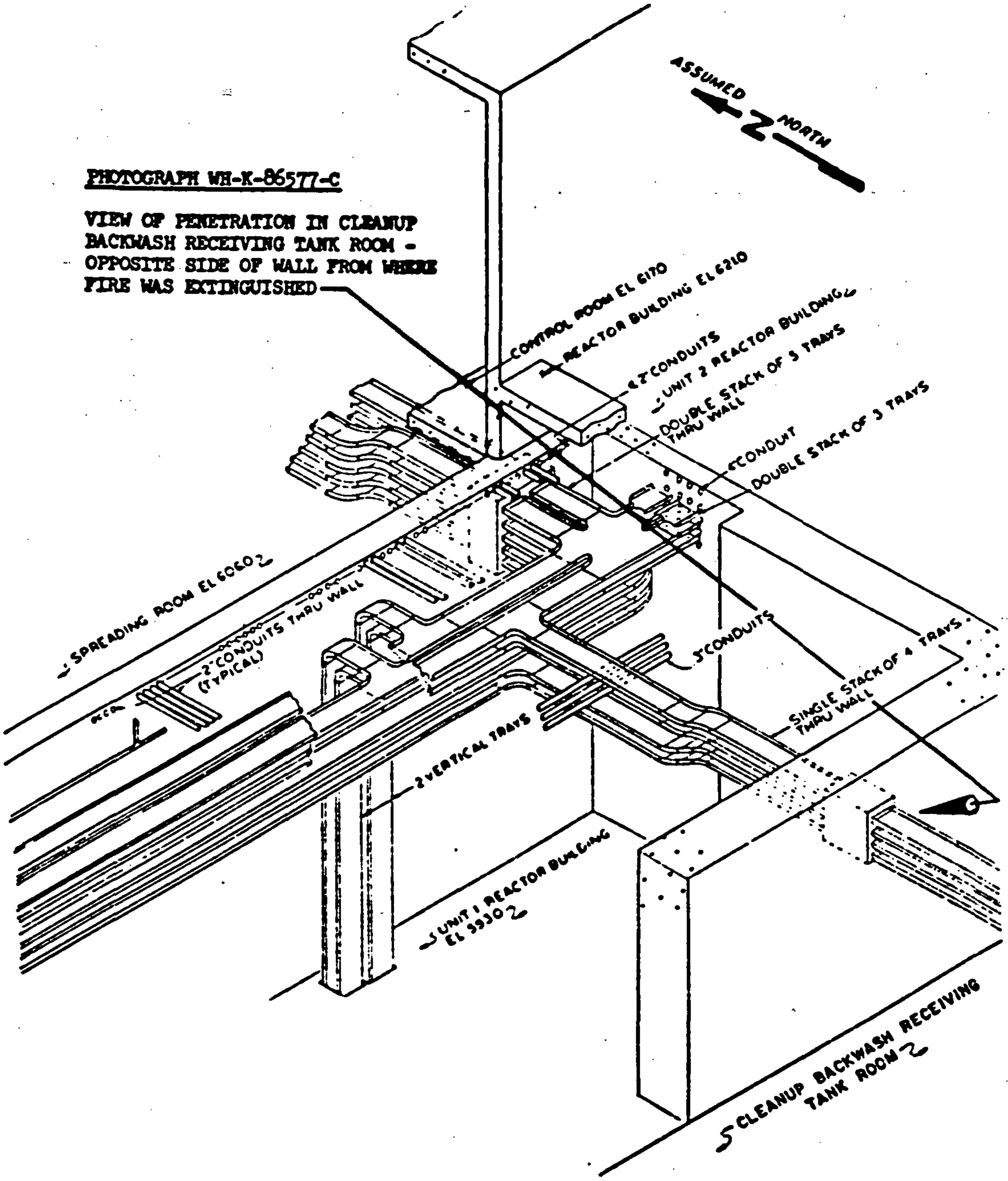
WH-K-86577-B


ASSUMED NORTH
PHOTOGRAPH 89438K
VIEW OF FIRE DAMAGE TO CONDUIT IN NORTHEAST CORNER OF REACTOR BUILDING
CONTROL ROOM EL 6120
REACTOR BUILDING EL 6210
2" CONDUITS
UNIT 2 REACTOR BUILDING
DOUBLE STACK OF 3 TRAYS THRU WALL
CONDUIT
DOUBLE STACK OF 3 TRAYS
SPREADING ROOM EL 6060
2" CONDUITS THRU WALL (TYPICAL)
3" CONDUITS
SINGLE STACK OF 4 TRAYS THRU WALL
2 VERTICAL TRAYS
UNIT 1 REACTOR BUILDING EL 5930
CLEANUP BACKWASH RECEIVING TANK ROOM

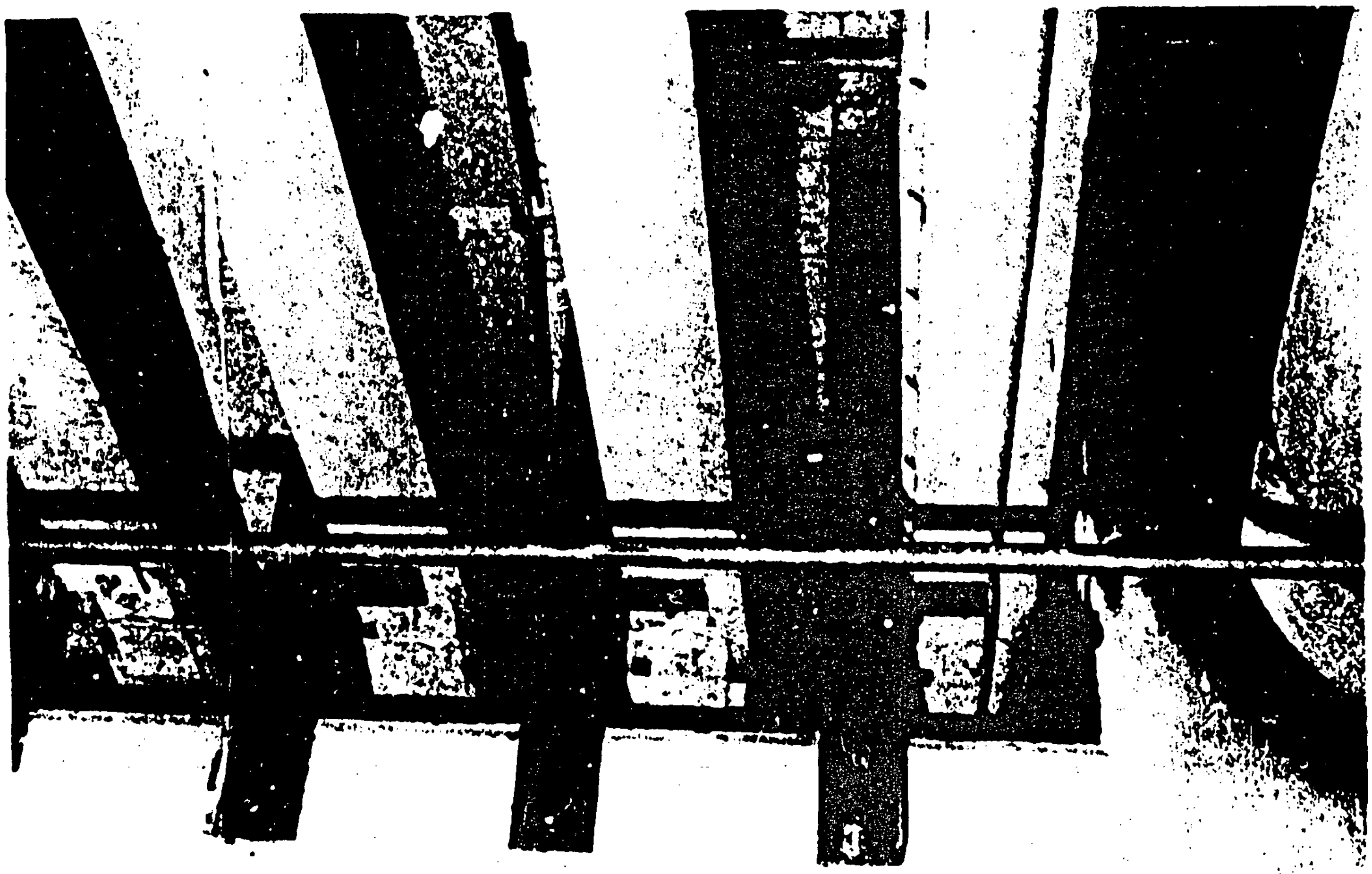
AREA OF FIRE



89438K


PHOTOGRAPH WH-K-86577-C
VIEW OF PENETRATION IN CLEANUP BACKWASH RECEIVING TANK ROOM - OPPOSITE SIDE OF WALL FROM WHERE FIRE WAS EXTINGUISHED
ASSUMED NORTH
N
CONTROL ROOM EL 6170
REACTOR BUILDING EL 6210
2" CONDUITS
UNIT 2 REACTOR BUILDING
DOUBLE STACK OF 3 TRAYS THRU WALL
CONDUIT
DOUBLE STACK OF 3 TRAYS
SPREADING ROOM EL 6060
2" CONDUITS THRU WALL (TYPICAL)
3 CONDUITS
SINGLE STACK OF 4 TRAYS THRU WALL
2 VERTICAL TRAYS
UNIT 1 REACTOR BUILDING EL 5930
CLEANUP BACKWASH RECEIVING TANK ROOM

AREA OF FIRE



WH-X-965??-C

APPENDIX 8

INVESTIGATION REPORT BY THE NUCLEAR ENERGY LIABILITY AND PROPERTY INSURANCE ASSOCIATION (NELPIA) DATED MAY 1975

TVA's Browns Ferry Nuclear Plant
Athens, Alabama
NEL-PIA Policy (NF - 198)

General

During April 7-9, 1975, an investigation into the March 22, 1975 cable fire that occurred at the Browns Ferry Nuclear Station was conducted by J. A. Elsner and R. J. Kasper. During this three day period, an analysis was made of the events that occurred up to, during and after this untimely incident. As a result of this study, a report has been prepared summarizing what happened and why it happened. Applicable recommendations suggesting possible fire protection improvements have been developed for consideration.

Introduction

TVA's Browns Ferry BWR nuclear power plant is located on the Tennessee River south of Athens, Alabama. Completed in 1974, its Unit 1 became operational on August 1, 1974. Seven months later (March 1, 1975) its Unit 2 also became commercially operational. Since each Unit was engineered to produce 1100 megawatts (MW) of electricity, a total of 2200MW were being supplied to TVA's power grid when a fire started in the plant's cable spreading room.

Origin of Fire

The fire that started at 12:35 P.M. on March 22 began in the Cable Spreading Room (CSR) for Units One and Two. This room, at elevation 606 ft., is located directly under the Control Room (CR) for Units One and Two. Constructed of reinforced concrete the CSR is cut-off from (1) the CR by a floor having a fire resistance rating of at least 3-hours and (2) the Reactor Building by a 2 foot, 2-inch thick reinforced concrete wall of at least 8-hours fire resistance rating.

Story of Loss

The night before the loss occurred, workmen were pulling cable from the RB into the CSR through a 2" x 4" hole in the 26-inch thick concrete wall separating the Reactor Building from the Control Building. To understand the procedures followed imagine a 4-ft x 4- ft wall opening containing ten 18-inch wide x 4" high galvanized steel cable trays. Tiered five high, the parallel runs of tray are sealed on the RB side of the wall to prevent flame propagation from one fire area to another. Whenever new cable has to be pulled, an opening is made through the fire stop construction. Upon completion, the opening is resealed first using a pillow-type foamed polyurethane for backing. To both sides of this backing material is next applied a 2-3 inch thick layer of foamed-in-place polyurethane or silicone rubber (currently used) followed by a 1/4-inch layer of Flamemastic 71A to complete the seal.

At the time of the loss workmen were in the CSR checking for air leaks into the RB. Apparently, the Standby Gas Treatment System's capacity (one complete air change/day) was being overtaxed from excessive air leakage into the RB. Using a 10-inch candle the cable tray penetrations were being checked. By holding a candle close to the penetrations, the workmen observed if the candle flame would flicker or not. The CSR is maintained at a positive pressure with respect to the RB to insure that air flow is towards the RB whenever radiation escapes.

When examining BE cable tray penetration, the candle flame was drawn into a hole not sealed airtight by the pillow-type foam material and not yet coated with foamed polyurethane and Flamemastic. It immediately ignited the pillow-type cushioning material and chips of rigid polyurethane foam which in turn ignited plastic insulated control cables. Without portable fire extinguishers immediately available nor a "fire watch" on the other side of the fire wall, the fire began to burn uncontrolled.

Not able to put out the fire by hand using a flashlight and a bundle of rags, the workmen hurried to find some extinguishers. Having located one dry chemical and three CO_2 extinguishers, they attempted to extinguish the fire but found that the burning cable would repeatedly reflash once the extinguishers were emptied.

When hunting-down extinguishers to fight this fire, the public security officer was told of a fire in the CSR. He notified the shift engineer in the Control Room who called in the station alarm by dialing 299 on the telephone. Firefighting personnel responded to this alarm and proceded to extinguish the fire in the CSR and the RB.

The difficulties encountered putting out the fire in the CSR prompted the brigade to activate the Cardox total flooding system. After having experienced numerous problems with this system, the CO_2 system was finally discharged sometime after 1:20 P.M. Dumped three times during the fire, it controlled the fire at its source in the CSR and prevented a catastrophe from occurring.

The fire also spread rapidly into the Reactor Building through the opening in the cable tray seal. Hampered by smoke and the location of the fire, firefighters wearing breathing apparatus used several dry chemical and carbon dioxide fire extinguishers. Discharging them from scaffolding and ladders, they encountered continual reflash. The inaccessibility of the multi-tiered cable trays located 20-25 feet overhead, the occasional dripping of flaming insulation and the failure of the building's ventilation system compounded the firefighting effort.

At two o'clock firefighting efforts in the Reactor Building were abandoned in favor of an orderly station shutdown. For the next two hours, the cable fire was allowed to burn uncontrolled for lack of manpower, lighting and visibility. Once Units One and Two were shutdown, the firefighters resumed fighting the stubborn fire with portable 15-and 20-lb extinguishers.

Up until 6:00 P.M. multiple fires continued to burn in the cable trays. Continually, extinguishers were discharged onto the fires, often the fires would go out and repeatedly the fires would reflash. This phenomenon required a total of 51 fire extinguishers (20-15 lb. CO_2, 31-20 lb. dry powder "purple K"), 2-125 lb. "purple K" carts and 1-100 lb. CO_2 cart!

At 6:00 P.M. a decision was made to use water on the fire. The ineffectiveness of the portable extinguishers and the dust concentrations created by the discharged dry powders finally prompted firefighters to use a 1 1/2" hose stream. Backed by 3-2500 gpm @150 psi electric fire pumps, its use by the Athens Fire Department was found to be extremely effective, and within fifteen minutes or so the fire was essentially out.

Damage

Aside from corrosion damage and clean-up, the cable tray systems received the most damage. Approximately thirty 18-inch wide cable trays or sections thereof were damaged by fire. Assuming there were 100 cables in each of the trays involved in the fire, it is estimated that 60,000-65,000 feet of control and thermocouple cable were destroyed!

The loss of this amount of cable affected or impaired the operation of numerous systems. For example, the manner in which the cable failed simulated a loss of coolant accident. Steam isolation valves closed, motor operated valve control was lost and 250 volt D.C. power controlling the operation of circuit breakers failed. Also affected were numerous control boards, core spray pumping systems, exhaust systems, M-G sets, 4KV bus, and lighting circuits.

The repair, replacement and rearrangement of the cable trays and cable sections will require approximately three months. If Units One and Two are shut down this long, the TVA has estimated it will cost an additional $44M in fuel costs to supply sufficient power to its grid.

Cable Spreading Room, Similar Rooms and Other Areas

The Cable Spreading Room at Browns Ferry was considered to be poorly designed from an operational and fire protection standpoint. First of all it was engineered to house the cable spreading rooms of Units One and Two. Normally one would expect each unit to have its own cable room constructed in such a manner that a fire in one would not spread to the other. Were such an incident to occur that shutdown both units for months at a time, the repercussions would be far-reaching.

The congestion in this CSR was inexcusable. Cable trays were situated in every direction conceivable. The close proximity of horizontal and vertical arrays from floor to ceiling the length of the room (150 ft.) is not possible to relate in terms of the severity of the problem.

Such a massive array of cable trays in the absence of aisles voids any realistic fire-fighting effort without subjecting firemen to possible injury or even loss of life. To go into the heart of this CSR full of smoke and acrid fumes to put out a fire even wearing self-contained breathing apparatus would be considered beyond the realm of heroism. Most firefighters would ultimately resort to a manual defense of the fire outside the room using hose streams and high expansion foam.

Unless automatic fixed fire extinguishment is provided to protect the hazards inherent in these rooms, a loss beyond imagination should be anticipated. To overcome such a catastrophe, a first class, designed and approved, automatic fire protection system is needed.

Of the systems currently available, it is recommended that automatic sprinkler protection be provided. It offers the most effective and efficient way of putting out a burning cable fire. By wetting-down and cooling the red hot conductors, the Class A fire can be readily extinguished without much concern over reflash.

Although total flooding high expansion foam, halon and carbon dioxide are considered acceptable extinguishing agents by some agencies, they do not possess the coolant properties so necessary to ultimately put out the fire. Without their presence, continued reflash of the cables can be expected until the temperature of the conductors is well below the auto-ignition temperature of the cable insulation and jacket

Recommendations

Design

1. A CSR for each unit should be provided. Each spreading room should be cutoff and arranged totally independent of other CSR's by a fire barrier wall of 3-hours fire resistance rating.

2. At least one 3-foot wide 8-foot high aisle should be provided the length and width of the CSR to insure firefighting access.

Class A fire doors should be installed at each entrance (at least two) into the room.

Fire Protection

1. A standard installation of open head, water spray sprinklers controlled by an automatic deluge valve and products-of-combustion acutated detectors should be provided in each CSR. The deluge valve should be located outside the room and connected to the station's annunciator system.

2. One-inch hose connections, equipped with 75 ft. of 1 1/2" woven-jacket lined fire hose and adjustable spray nozzle should be provided in the CSR and located at approximately 100 ft. intervals.

3. The concrete floor should be pitched to drain the sprinkler and hose stream discharge to a suitable drainage facility.

4. Approved smoke and heat venting of the CSR utilizing a powered mechanical exhaust system actuated by products-of-combustion detectors should be provided.

Note: These recommendations are also pertinent to Cable Penetration Rooms and other areas where concentrations of cable exist.

Cable Constructions

The widespread use of combustible nonmetallic cable constructions in nuclear power stations has introduced a fire hazard that is not sufficently appreciated by station operators. Unless those in authority begin to recognize the seriousness of this hazard and its consequences, losses involving millions of dollars can be expected.

The premise that grouped nonmetallic cables will not burn or propagate flame has been a fallacy often promoted by certain interest groups. Up until recently users of electrical cable often believed that their tray cable was "nonburning", "nonpropagating "fire resistant" or "self-extinguishing". Not so! Recent cable fires and cable tray tests have substantiated that grouped cables do burn and are in fact a recognized fire hazard.

Often times manufacturers, designers and users will specify the pass-fail criteria of a cable construction to justify its intended use. When they do, flame resistance properties are often one of the first sacrificed for improved moisture, corrosion, oil, radiation and ozone resistance. The result is a cable construction that will burn in almost every conceivable tray configuration.

The corrosivity and toxicity of burning cable are also often ignored when selecting a cable construction. Whenever polyvinylchloride, neoprene or hypalon cables are burned, hydrogen chloride (HCL) gas is evolved. Combined with the water vapor in air, the hydrochloric acid produced will contaminate building steel, electrical equipment and instrumentation. The result is a major corrosion loss.

At Browns Ferry an assortment of cable constructions were noted. In Table No. 1 most of the types of cables involved in the fire are listed. All of the cables are 600 v. control cable or thermocouple cable. Typically, the cables are two conductor (2C) No. 16 and occasionally one conductor (1C) 300 MCM and three conductor (3C) No. 16.

All of the cables consisted of a polyethylene (PE) or cross-linked polyethylene (XLPE) insulation and a polyvinylchloride (PVC), hypalon or nylon overall jacket.

Considered combustible by manufacturers and some users, these constructions have too often misled people that they do not burn. In fact, they do burn. Certain fire test and numerous fire losses have verified this.

To be more specific, the TVA decided that cable specimens passing Underwriters Laboratories Standard No.83 are acceptable for use in their stations. In the vertical test, a Tirrill gas burner is applied to a 12-inch specimen for 15 seconds then removed for a minimum of 15 seconds. This procedure is repeated five times. Cables that flame or glow longer than 60 seconds after any of the flame applications are judged not acceptable.

Reviewing the flame-retardant requirements of cable at Browns Ferry, all single and multi-conductor cables No.8 and larger were required to pass the misleading vertical flame test. All cable No.9 and smaller were required to pass the even more misleading IPCEA horizontal test. Suprisingly, single and multi-conductor cable having a nylon jacket were not required to pass any flame test!

Because these small-scale tests can be very misleading, especially for grouped multi-conductor control and power cable, users should insist that stringent cable fire tests indicative of "real-life" situations be required. One test currently being used by some utilities is the IEEE No.383 flame test; others, such as NEL-PIA/MAERP, insist that fully loaded, tiered trays be tested, using more severe ignition sources.

In summary, minimal consideration was given to the obvious burning characteristics of the cables used at Browns Ferry. Were there concern, it would have been evidenced by more stringent flame tests, better grade, truly fire retardant cable constructions and fixed fire protection for the cable tray systems.

Recommendations

1. A re-evaluation of current cable testing requirements should be made to establish the pass-fail criteria for flame propagation of "real-life" cable tray systems.

2. As a minimum requirement today only those cable constructions that will pass the current IEEE No.383 flame test should be used. Note: This does not infer that cables passing this test do not require fire protection nor that certification of cables passing more realistic tests are not essential for tommorrow's cables.

3. Whenever practical, cables that do not liberate copious quantities of corrosive gases should be used particularly in strategic, relatively inaccessible and highly susceptible areas.

Table No. 1

Cable Constructions

	Manufacturer	Size	Cable Description
1.	Boston Insulated Wire	2/C No. 16 7 strand tinned copper	30 mils XLPE insulation, .001" mylar tape, 46 mils overall hypalon jacket
2.	Cerro	3/C No. 16 1969 WVR	20 mils PE insulation, 10 mils PVC conductor jacket <u>or</u> 26 mil PE insulation, 4 mil nylon conductor jacket, 80 mil overall PVC jacket
3.	Continental Wire	2/C No. 16 1968 WUB	25 mils 90°C XLPE insulation, 30 mils overall hypalon jacket
4.	General Cable	1967 600 v. CPJ	XLPE insulation, PVC overall jacket
5.	Okonite	2/C, 3/C No. 16 7 strand	20 mils 90°C XLPE, 40 mils overall hypalon jacket
6.	Plastic Wire & Cable	1971 600 v. PJJ 2/C No. 12	20 mils PE, 10 mils conductor jacket, 60 mils overall jacket
7.	Simplex	600 v. 1/C 300MCM PNJ	PE insulation, 4 mils nylon jacket

Symbols:

XLPE - cross-linked polyethylene

PE - polyethylene

PVC - polyvinylchloride

PJJ - PE insulated, PVC jacketed, single conductor

CPJJ - XLPE " , PVC " , multi-conductor

PNJ - PE " , nylon " , single conductor

CPSJ - XLPE " , shielded, PVC jacketed, single conductor

Cable Tray Protection

The cable trays involved in the fire were not protected by automatic or manual fixed fire protection systems. The only fire protection equipment immediately available were a 1 1/2-inch hose connection and a 20 lb. dry chemical extinguisher which was ironically never used.

Having already emphasized the combustibility of the cables involved in the fire, the configuration and inaccessibility aspects of the cable trays should also be studied. Consider 18-inch wide, open ladder trays with rungs every six inches stacked vertically five high with only six inches of vertical separation between trays. Next, imagine another vertical array of trays similiar to the first six inches away. Finally visualize this arrangement 20-25 feet above the floor in a windowless area with access from only one direction.

Such a large and important arrangement of trays filled with plastic jacketed cable is one that should have been protected by a system of automatic sprinklers. The presence of sprinklers would have effected control and extinguishment of the fire within minutes. Once discharged, the water would have sufficiently cooled the copper conductors preventing cable reflash. The amount of smoke generated and corrosive gas liberated would in turn have been considerably less.

Recommendations

1. Cable tray systems should be protected by automatic, zoned, open-head, water spray sprinkler systems arranged to discharge directly onto the cables in the trays.

2. An approved fast acting products-of-combustion type detection system should be provided to actuate the deluge system having sectional control.

3. Adequate floor drains and curbs should be provided to safely remove discharged sprinkler water. Drainage water should be monitored for radioactive materials before being released to the environment. Curbs should be provided around all floor penetrations.

4. Approved noncombustible fire stop constructions should be located in each cable tray and spaced at maximum intervals of 10 feet horizontally and 10 feet vertically. Note: Cable derating should be given consideration when installing fire stops.

5. Wherever practical, isolate, shield, or relocate water damageable equipment.

Consequential Corrosion

Whenever PVC, neoprene and hypalon cable constructions are involved in a fire, frequently the fire damage is insignificant compared to the consequential damage caused by corrosion. The hydrogen chloride gas liberated by these burning materials readily combines with water from the combustion process and water vapor in the air to form hydrochloric acid. When this acid comes into contact with a metal surface, it will react with it to form conventional rust. When it comes into contact with building materials having a calcium element, the hydrochloric acid will react with the calcium constituent causing flaking, tensile strength reduction and steel reinforcement corrosion.

The introduction of such acids into a nuclear station is certainly not advisable. The damage caused by HCL followed by progressive corrosion is often so devastating that months and millions of dollars are required to reactivate a unit shutdown by a fire of relative unimportance.

At Browns Ferry, there was considerable evidence of corrosion damage in the Reactor Building. Throughout elevation 593, stainless steel piping, copper tubing, instrumentation, electrical relays, conduit, valves, pull chains, sleeves, bolts and nuts were found rusted or coated with a brown film.

At elevation 620', there was also evidence of corrosion damage to equipment. Had the open steel-grate stairwells and numerous floor openings been enclosed or sealed air tight, it is doubtful corrosion damage and carbon deposits would have spread to this floor elevation.

Although the TVA does not know how much the cost will be to repair or replace all the equipment damaged by corrosion, it is conceivable that it will take at least three months and cost millions of dollars. Had sufficient consideration been given to this possibility when cables were initially selected, it is doubtful that corrosion would have been a major factor in this loss.

Measures that should be taken to reduce the risk of corrosion damage have been outlined in other sections of this report. Basically, plastic cables having corrosive by-products should be avoided, floor penetrations should be tightly sealed, automatic fire extinguishing systems should be installed, and ventilation systems should be arranged to localize the fire and minimize the concentration of corrosive gases.

Appraisal of Firefighting Concept

Three sections in the "International Guidelines for the Fire Protection of Nuclear Power Plants" should be referenced at this juncture. They are: 1) Appraisal of firefighting concept 2) Mobile firefighting and 3) Equipment of the plant fire brigade. These sections not only emphasize the importance of automatic fixed fire protection installations, they equally stress the importance of a coordinated effort between the local fire department and station fire brigade.

Unless a previous agreement has been reached amongst station personnel and public fire authorities, a situation is apt to present itself that will result in a misguided and costly firefighting approach. For every conceivable loss, a detailed action plan and firefighting approach should be developed in close cooperation with the insurance company and other competent authorities.

Although there is some merit to the firefighting procedures used at Browns Ferry, another approach using water as the extinguishment is often considered more effective. Since plastic electrical cables are Class A materials when they burn, extinguishing agents compatible with Class A fires should be used, e.g. water or light water. Extinguishing agents including carbon dioxide, halon, B:C dry powders and foams that do not have a cooling effect are normally ineffective and should not be used.

For a typical Class C electrical fire, with voltages of 600 volts or less to ground, there is virtually no danger to firefighters directing hose streams on cables. This is especially true of cable tray systems located 20 feet or more above the floor.

Were this concept employed soon after the fire started, it is postulated that the fire would have been extinguished in a matter of minutes. Cable damage would have been minimal, corrosion damage would have been insignificant and an orderly station shutdown would have been possible. In addition, the Reactor Building's ventilation system would not have been impaired and conceivably redundant circuits would not have failed.

Indoor Hose Connections

Although a 1 1/2" hose connection equipped with 100 ft. of woven-jacket fire hose was located 66 feet from the origin of the fire, its use was not as effective as it could have been. First of all, the decision not to use water during the first five hours of the fire proved costly. Were water used from the fire's inception there is little doubt that the fire would have been out without reflash within minutes and that cable and equipment damage would have been minimal.

When the decision was finally made to use water on the fire, plant personnel were reasonably assured that all electrical power was off in the fire area. The Athens Fire Department proceeded to attack a small vertical cable tray fire to see the water's effect. Noting that it worked fine without electrical arcing, they proceeded to attack the fire in the stacked upper trays located 20-25 feet above the floor.

Adjusting the spray nozzle they observed that the water spray wouldn't reach up into the trays. Firemen decided to use a combination spray-straight stream nozzle on the fire truck. Removing the adjustable spray nozzle, they quickly learned that the hose line threads were not compatible with those of the fire department's nozzle. The nozzle flew off!

Ultimately the firemen used ladders and scaffolds to get to the seat of the fire with the hose line. Firemen reported only 2-3 feet visibility, blue flames in the trays and no electrical arcing. Within minutes the fire was out although much localized steaming was noted.

Recommendations

1. Fire protection equipment including hose, nozzles, standpipe valves, and hydrants should have compatible threads with existing equipment and the local fire department.

2. Combination spray-straight stream nozzles should be provided on each hose connection to effectively combat Class A fires normally inaccessible e.g. cable tray fires.

3. Standpipe risers should be sealed on each floor to prevent smoke and corrosive gases from penetrating into areas normally unexposed to the effects of fire.

Smoke and Heat Removal

The effective removal of toxic gases and heavy concentrations of smoke is essential when combating a fire in a nuclear station. The primary reason for heat and smoke venting is that it enables firefighting personnel to gain early access into the fire area to extinguish the fire, to shutdown the station in an orderly manner and to perform emergency rescue work. In addition it will help to reduce smoke and acid vapor contamination.

At Browns Ferry, the smoke and acrid fumes generated by the burning cable were so intense that firefighters were required to wear self-contained breathing apparatus. Without such equipment on hand there was no possible way to manually combat the fire. Fortunately, plant personnel properly saw the need to acquire a supply of breathing apparatus just in case such a situation arose.

Had the ventilation system for the Reactor Building continued to operate unimpaired, the fire would not have burned for seven hours. Instead, within fifteen minutes of the start of the fire, the ventilation system shutdown. Apparently its power supply was located in one of the cable trays involved in the fire.

Running temporary wires to bypass the damaged cables, the ventilation system was turned back on at 4:40 P.M. Allowed to run for twenty minutes, it helped to clear the area of smoke and noxious gases and allow firemen to evaluate their next move. At 5:00 P.M. someone ordered the ventilation system shut down reportedly because it was helping to spread the fire.

Recommendations

1. Approved smoke and heat venting facilities independent of the station's normal ventilation system should be provided throughout areas having a combustible occupancy. Each system should be actuated by products-of-combustion detectors and arranged to contain the release of radioactive materials.

2. The mechanical exhaust system should be powered from electrical feeders run outside the fire area. If inside wiring is necessary, mineral-insulated metal sheathed cable should be used.

3. Additional preventative measures outlined in the "International Guidelines for the Fire Protection of Nuclear Power Plants" should be implemented. Specifically, sections 6.1 (extraction of smoke and heat) and 6.2 (preventing corrosion) are applicable.

Cable Penetrations (Trays, Conduit)

Whenever cable is installed in trays or conduit, wall and floor penetrations are inherently required. To insure the integrity of the walls and floors through which these systems pass, penetration seals are needed. Normally, approved seals that will maintain the fire resistance rating of the wall or floor are required by the authority having jurisdiction.

To Monday - morning quarterback which penetration seal specification should have been used in the Cable Spreading Room is not the intent of this report. What is important is an awareness that Underwriters' Laboratories and Factory Mutual are currently listing and approving devices and constructions for wall openings. Subjected to the ASTM E119 fire test, wall assemblies designed to protect electrical cables and/or conduits against the passage of flame and smoke are being tested.

Assemblies should be tested in conjunction with their intended use. If used in a wall having a 3-hour fire resistance rating, a wall opening protective having a rating of three hours should be installed. Likewise, if mounted in a floor opening, a device listed specifically for floor openings should be used. If a protective seal that is being considered for use in walls and floors is not currently listed or approved, the manufacturer should be requested to have his product tested according to the E119 test procedure by a recognized testing laboratory.

It is also important that wall and floor opening protectives be installed and tested without risk of fire or personnel injury. Installation and final test procedures should not allow the use of open flames nor should they permit temporary unprotected openings overnight and over the weekend. Sufficient precautions should always be taken to insure that all wall and floor openings are sufficiently sealed to prevent the spread of fire from one fire area to another.

Recommendations

1. All wall and floor openings through which electrical cables or conduits penetrate should be protected against the passage of flame and smoke by devices and constructions approved or listed by recognized testing laboratories.

2. Temporary wall and floor openings should be sufficiently sealed with a noncombustible material at the end of each working day to insure the fire integrity of the wall or floor.

3. Open flames should never be used to check the installation, gas tightness and integrity of penetration seals. Whenever protected openings are examined, fire extinguishers should be immediately available to those checking for openings on both sides of the wall.

Self-contained Breathing Apparatus

Discussions with station personnel that fought the fire uncovered a wealth of information about the use and effectiveness of self-contained breathing apparatus. First of all, it was learned their use was imperative to attack the fire, manually shut down Unit One and continue working in the Control Room. To also emphasize how invaluable they were, the fire would have burned uncontrolled for several more hours had they not been used. An orderly station shutdown would have been jeopardized and personnel would have been subject to possible injury including death.

Certain problems concerning their use were noted. Their service life was not adequate by the time they were able to get to the seat of the fire. Continually firefighters had to leave the fire to replenish their air supply having been in the fire area only ten minutes. To shut some valves manually it often required men wearing Scot Air Paks 2-3 trips. Those with long hair often experienced smoke inside their mask from a poor fit.

Overall Chem-Ox respiratory protective equipment was considered a better unit than the Scot Air Paks. They lasted longer and fit better. On the other hand some of the Chem-Ox units were found inoperative whereas the others all worked.

Recommendations

1. Self-contained breathing apparatus approved by the United States Bureau of Mines and described in NFPA No.19B should be provided for all firefighting and control room personnel. Preferably, their service or operating life should be one hour.

2. An on-site reserve air supply should be available and arranged to expediently replenish the supply of air in each unit so that the designed service life is available.

Physical Independence of Redundant Circuits

The loss of some of Unit One's redundant systems illustrates the importance of physically independent redundant circuits. Permitting redundant circuits in trays that adjoin other trays loaded with highly combustible cable is not a sensible nor safe practice. Ultimately a fire in one tray will expose and eventually involve cables in the adjoining trays.

NRC's Regulatory Guide 1.75 "Physical Independence of Electric Systems" contains specific separation criteria for Class IE circuits. Although the specifications in this Guide are an improvement over previous guidelines, they do not require that all redundant circuits be separated by a fire barrier wall. Instead, they permit IE circuits to be installed in locations commesurate with the potential hazard. Such analysis and judgment considerations often dilute the specific criteria intended.

In order to insure a safe and expedient station shutdown, redundant circuits should be separated from each other by a minimum three hour fire barrier wall. By arranging the circuits in this manner, a catastrophe involving one system should not conceivably interfere with the continued operation of the other. Such a requirement should be incorporated into the station's design during the planning stages.

The use of mineral insulated metal sheathed cable or highly flame retardant cable should be considered for one of the two redundant circuits. These cables will insure reliance of the electrical continuity of the primary circuit and additional security that an orderly station shutdown will take place.

Recommendations

1. All redundant Class IE circuits and the equipment served by these circuits should be separated from the primary Class IE circuits by a minimum three hour fire wall. This will require that a redundant cable spreading room be constructed.

2. Mineral insulated metal sheathed cable or equivalent fire resistant cables should be used in one of the two Class IE electrical circuits.

Cardox Total Flooding System

A 17-ton capacity Cardox System was installed to provide protection for the Cable Spreading Room. Arranged to operate manually, the carbon dioxide system was designed to discharge gas into the room for approximately 2 1/2 minutes if actuated electrically. If operated pneumatically the system was arranged to discharge CO_2 continuously until either the valve was closed or the supply was depleted.

At the time of this loss the workmen in the CSR had turned off the electrical power supply to each of the two Push-button Stations. By doing so, the system can not be operated electrically until the toggle switch (located on Unit Two side of the CSR) is turned back on.

When the decision was made to discharge the Cardox system into the CSR, the Assistant Shift Engineer went to the Unit One side of the CSR to operate the Pushbutton Station. It did not work. Noting that the power supply had been turned off (red light out), he attempted to trip the system pneumatically from the Electro Manual Pilot Control

Cabinet. Because a steel plate used in shipment to protect the glass over this cabinet had not been removed when installed, he could not break the glass to open the manual valve!

Next he hurried to the Unit Two side of the CSR, turned back on the power supply at the Relay Cabinet and finally tripped the Pushbutton Station that discharged the total flooding system.

Although flooding the CSR with CO_2 prevented a major fire from occurring inside, it should be noted that the system's capacity was the key to the minimal damage incurred. For some unknown reason, the ventilation system for the CSR had not been interlocked to shutdown when the CO_2 system discharged. When the gas entered the room the exhaust system continued to operate dissipating the CO_2 concentration. This caused reflash! Were the exhaust system interlocked to shutdown it would not have been necessary to discharge the CO_2 system three times.

According to discussions with station personnel, the Kidde ionization detection system in the CSR did not operate during the incipient stages of the fire. However, about the time CO_2 began discharging into the CSR, the detection system did in fact actuate, not from the products-of-combustion but from the CO_2.

Recommendations

1. The ventilation system in the CSR should be arranged to shutdown whenever the Cardox system is discharged.

2. The Cardox system should be rearranged to operate automatically upon actuation of the ionization detection system. Note: A one minute delay should be incorporated into the system to allow workers ample time to leave.

3. A written procedure and permit system should be adopted that would require employees to obtain written permission to impair fire protection equipment.

4. An acceptance test of the fire protection system, including a complete discharge, should be conducted and witnessed by the installer and authority having jurisdiction.

5. An investigation into the compatability of the ionization detectors with the products-of-combustion generated by the burning cable should be made to insure that the detectors will in fact operate during the incipient stages of a fire.

Control Room

Each time it was necessary to discharge the Cardox total flooding system in the Cable Spreading Room, CO_2 gas and smoke from the fire would enter the Control Room through unplugged holes in the floor. Two 4" x 16" and two 6" x 24" unprotected floor openings were found under the "electrical desk" in the Control Room. To compound this problem the gas and products of combustion that escaped through these openings forced the operators in the Control Room to wear Scot Air-Paks when shutting down Unit Two.

Recommendations

1. All floor openings between the CSR and the CR should be sealed air tight with a material that will insure the fire resistance integrity of the floor. Only penetration seals listed by Underwriters' Laboratories or approved by the Factory Mutuals should be considered. Cellular concrete, and inorganic assemblies as described in the "International Guidelines" may also be considered.

2. Self-contained breathing apparatus approved by the United States Bureau of Mines should be located in the CR to insure an orderly station shutdown and to minimize breathing hazards to personnel. The supply should be sufficient for the number of operators and the time it takes to effect a safe shutdown.

Stairwells, Vertical Openings, Mechanical Penetrations

It was disturbing to note the presence of unprotected steel-grated stairways and floor openings around steam pipes, standpipes and mechanical penetrations in the Reactor Building. Although fire did not spread through these openings into other areas, smoke and hydrochloric acid fumes did. An examination of the building structure and equipment on the floors above the fire revealed heavy carbon deposits on walls and ceilings and substantial corrosion damage to stainless steel piping, electrical equipment, and instrumentation.

Were all vertical openings between floors sufficiently sealed to prevent the passage of smoke and corrosive vapors, corrosion damage and general clean-up would have been confined to floor elevation 593. Instead, clean-up and repair operations have to be expanded to include those areas needlessly exposed to carbon deposits and acid fumes

Recommendations

1. All stairwells, elevators, chutes and other vertical openings should be enclosed in approved masonry towers with air-tight, automatic closing Class B fire doors at each opening into the building.

2. All unprotected vertical openings between floors (hoistways, steam pipes, etc.) should be sealed air-tight.

UL-NEL-PIA Cable Testing Program

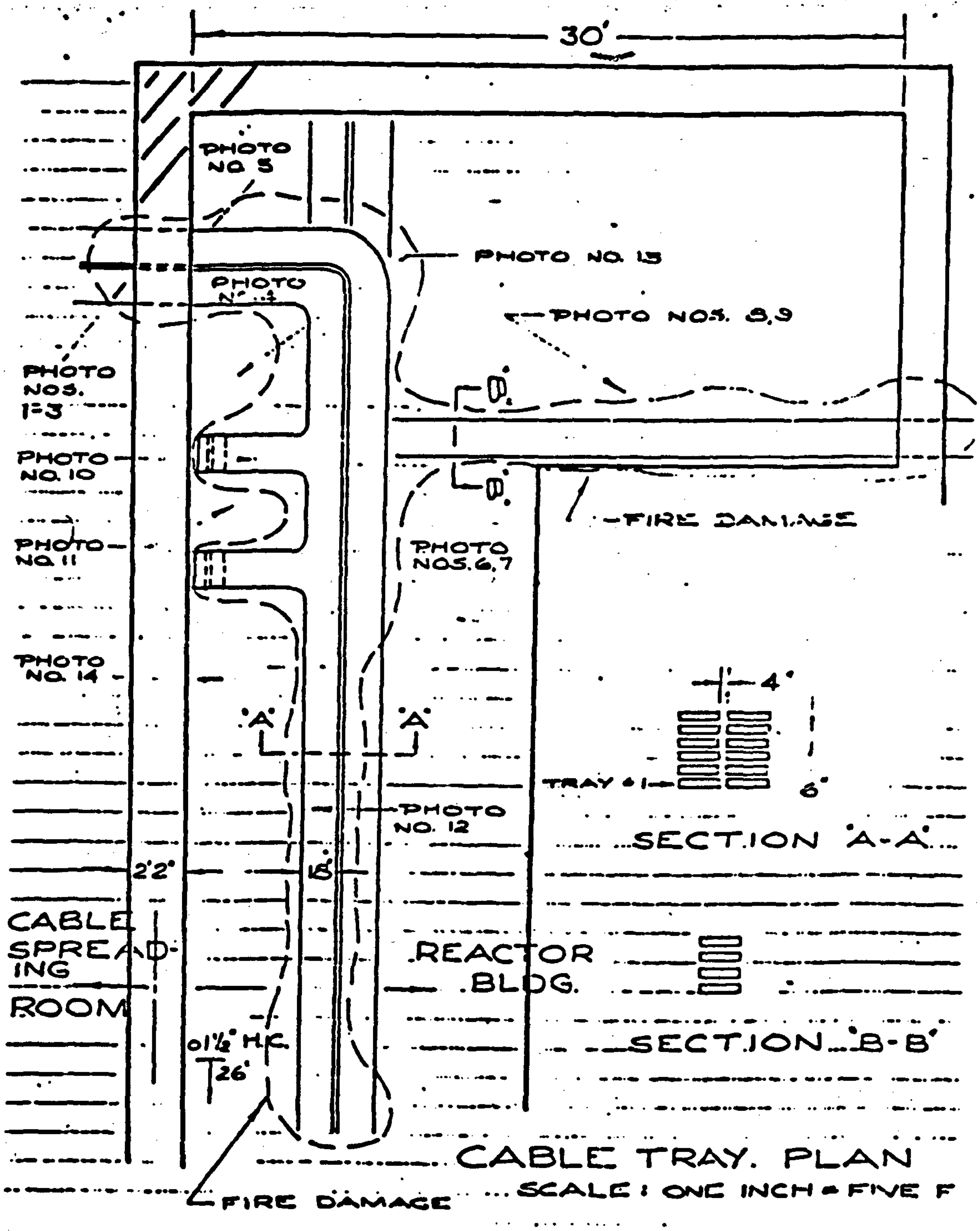
A NEL-PIA sponsored cable testing program is being conducted at Underwriters' Laboratories to determine the relative performance of cables when subjected to the IEEE Standard No.383 vertical flame test using a 70,000, 210,000 and 400,000 Btu/hour gas burner. Various control cable constructions will be tested vertically and horizontally in multi-tiered trays to determine the effects of these ignition sources on flame propagation and circuit integrity.

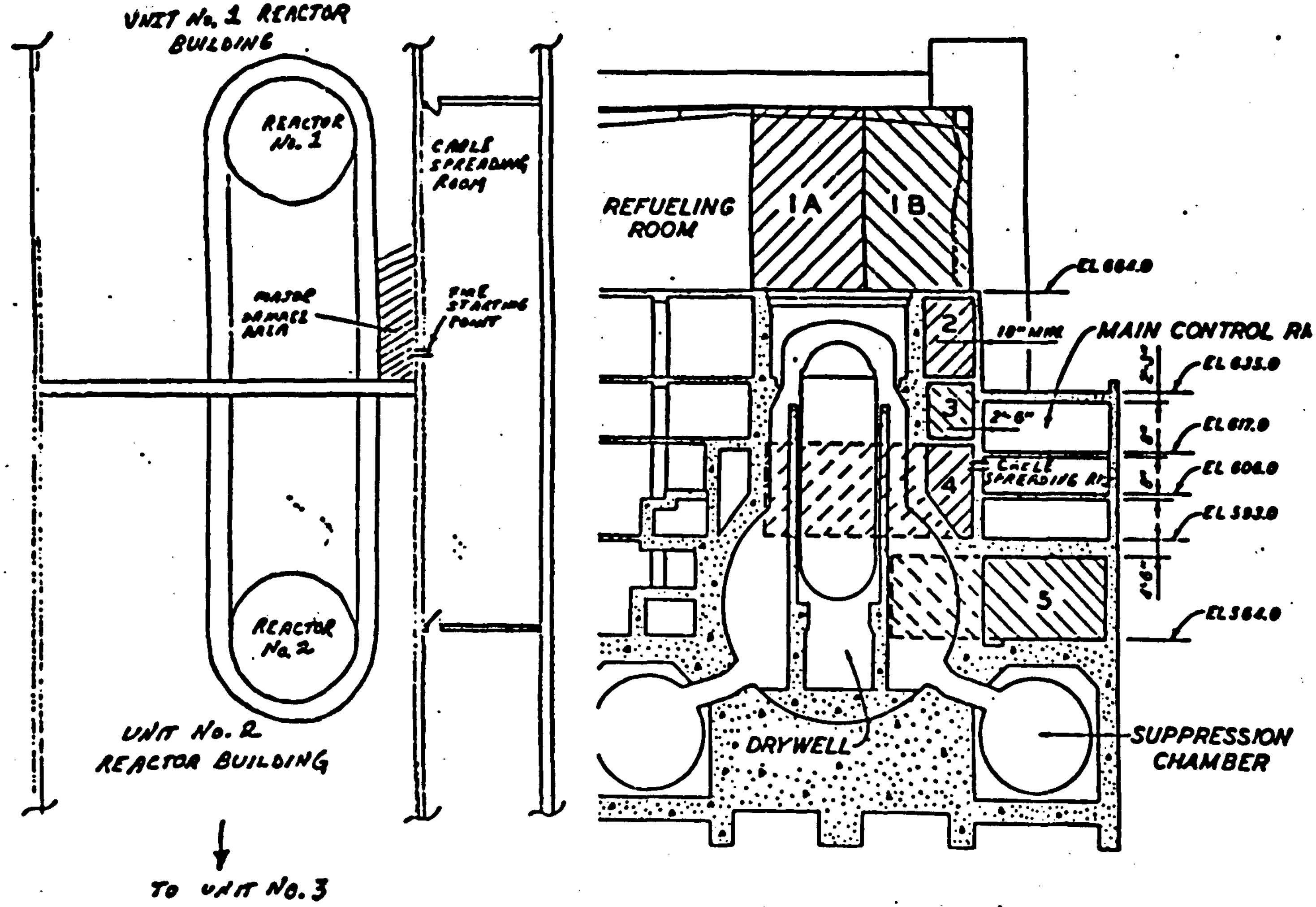
It is strongly recommended that the TVA consider including representative samples of cable involved in the Browns Ferry fire in this full scale testing program. Tested horizontally in multi-tiered trays and vertically, an understanding of the cable's "real-life" (not laboratory) burning characteristics can be demonstrated. In addition, the flame propagation and circuit integrity properties of cables considered highly flame retardant could be compared with those of the cables destroyed in Unit One.

Time Schedule of Events (Approximate)

Saturday, March 22, 1975

12:35 P.M.	Fire started in Cable Spreading Room,
12:40 P.M.	Fire Alarm 299 called in.
12:50 P.M.	Ventilation system in Reactor Building shutdown.
12:51 P.M.	Unit One reactor scrammed.
12:55 P.M.	Public Safety Service fire truck arrived.
1:02 P.M.	Unit Two reactor scrammed.
1:09 P.M.	Athens Fire Department notified.
1:20 - 1:30 P.M.	Cardox total flooding system discharged.
1:25 P.M.	Athens Fire Department arrived with one truck (chief plus six men)
1:30 P.M.	Reactor Building lighting lost.
1:30 - 2:00 P.M.	Self-contained breathing apparatus required in Control Room.
2:30 - 3:00 P.M.	Cardox system discarged the second time.
2:00 - 4:00 P.M.	Cable fire in Reactor Building burning unhampered. Firefighting effort abandoned in order to shutdown Units One and Two.
3:00 - 4:00 P.M.	Cardox system discharged for the third time.
4:40 P.M.	By running temporary wires, the ventilation system in the Reactor Building was reactivated.
5:00 P.M.	The Reactor Building's ventilation system was again shutdown.
6:00 P.M.	Hose stream first used on fire in Reactor Building.
6:45 P.M.	Fire considered out in Reactor Building.

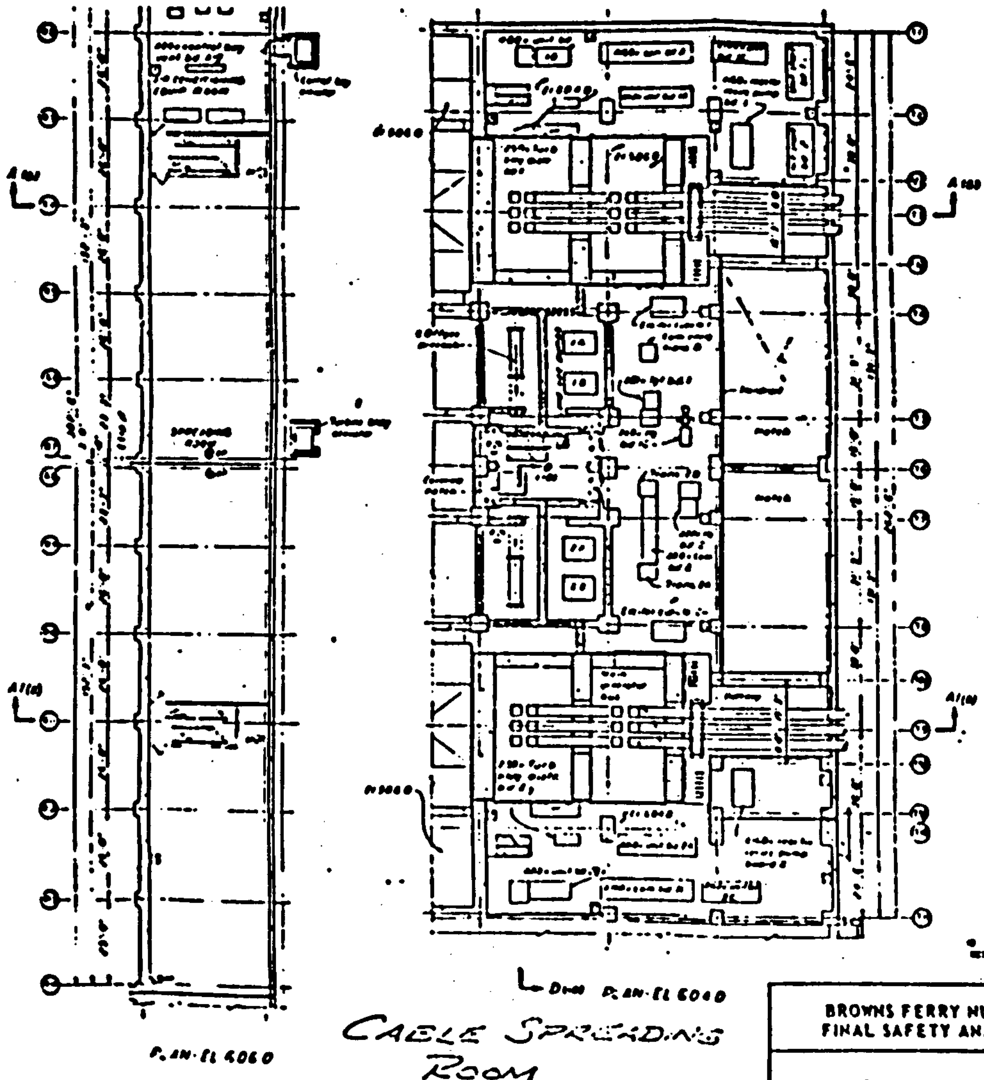

30'
PHOTO NO. 5
PHOTO NO. 13
PHOTO NO. 4
PHOTO NOS. 8,9
PHOTO NOS. 1-3
PHOTO NO. 10
PHOTO NO. 11
FIRE DAMAGE
PHOTO NOS. 6,7
PHOTO NO. 14
"A"
"A"
4"
TRAY # 1
6"
PHOTO NO. 12
SECTION "A-A"
2'2"
"B"
CABLE SPREADING ROOM
REACTOR BLDG.
SECTION "B-B"
o1½" H.C.
26'
CABLE TRAY. PLAN
SCALE: ONE INCH = FIVE F
FIRE DAMAGE



PLAN VIEW

SECTIONAL VIEW

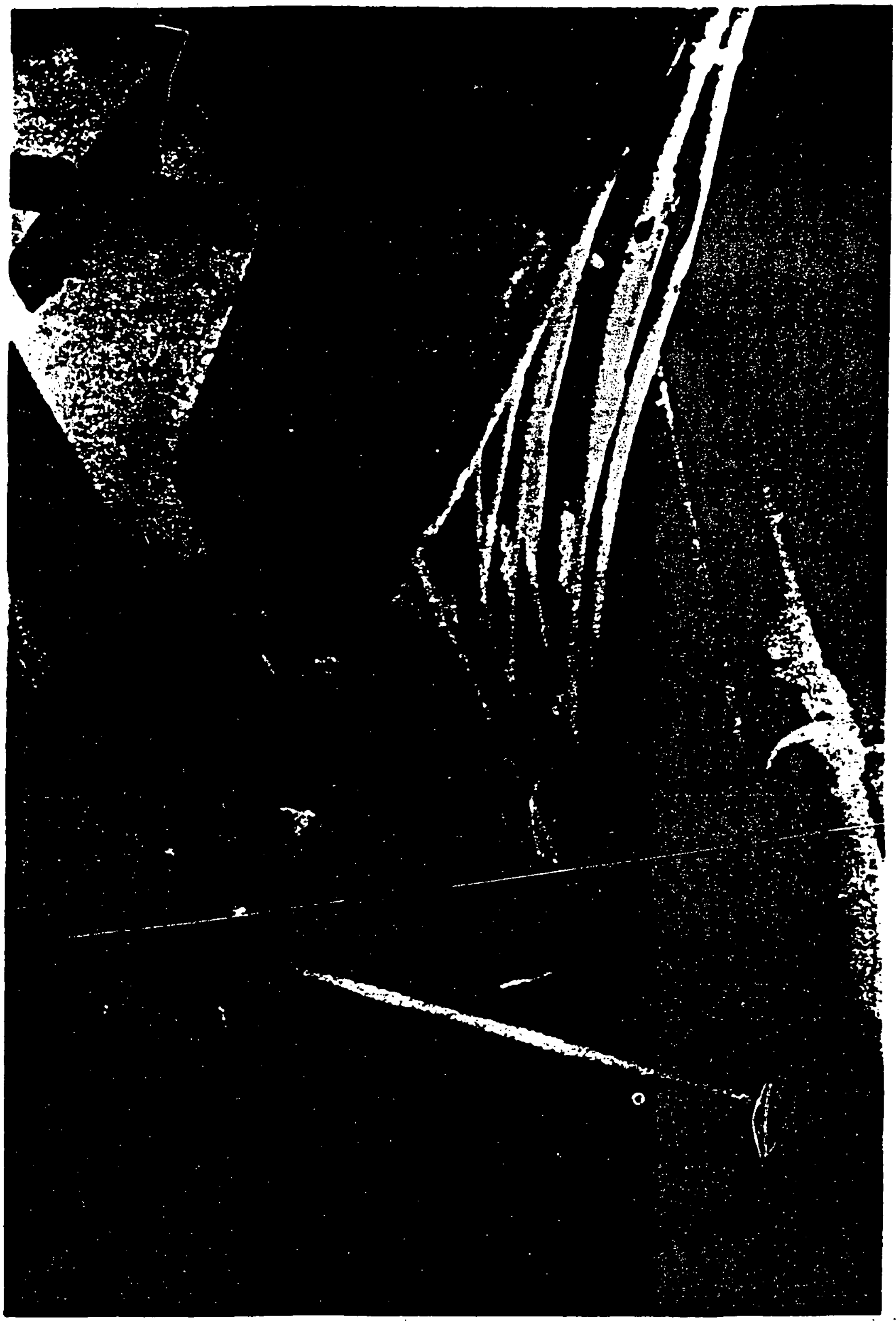
FIGURE No. 1



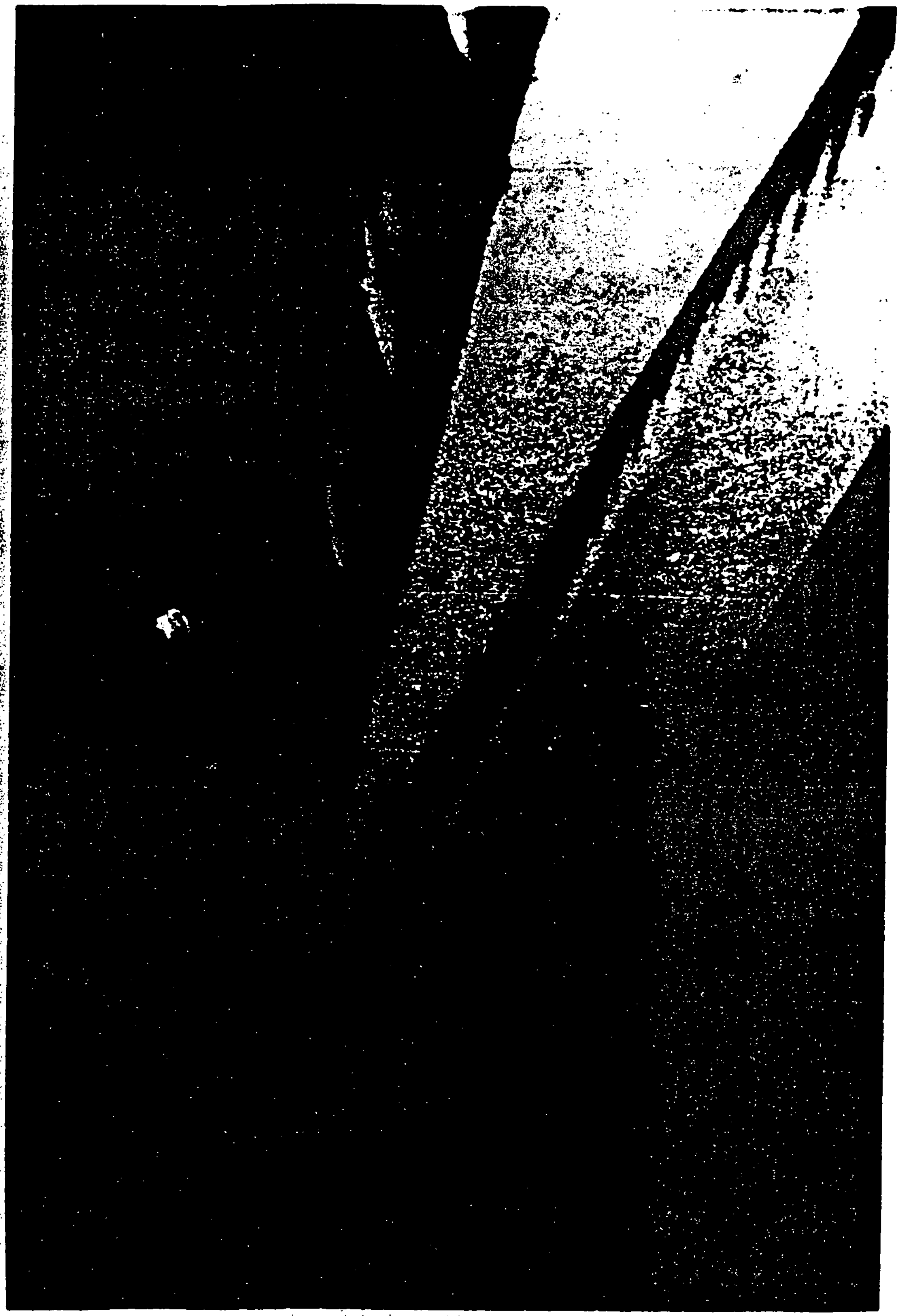
BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

Powerhouse Equipment Plans—

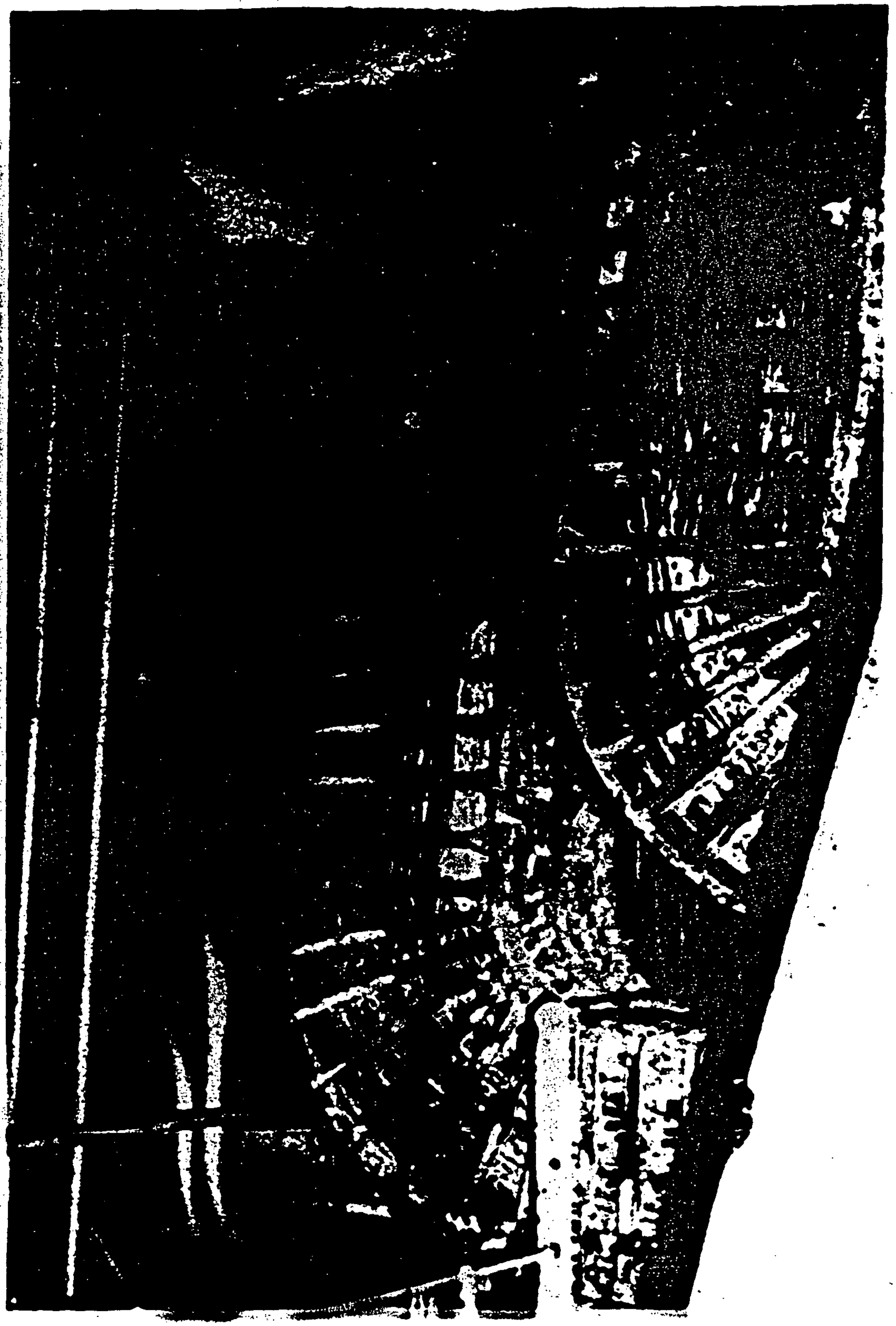
The fire originated in the middle tray. Taken inside the Cable Spreading Room, this photograph shows the cable trays passing through the wall into the Reactor Building.



Also taken inside the Cable Spreading Room, this photo shows the top three trays involved in the fire. Observe the exposed conductors in the top and bottom trays and the close proximity of the trays to each other. The white powder on the middle tray is from a dry powder extinguisher.

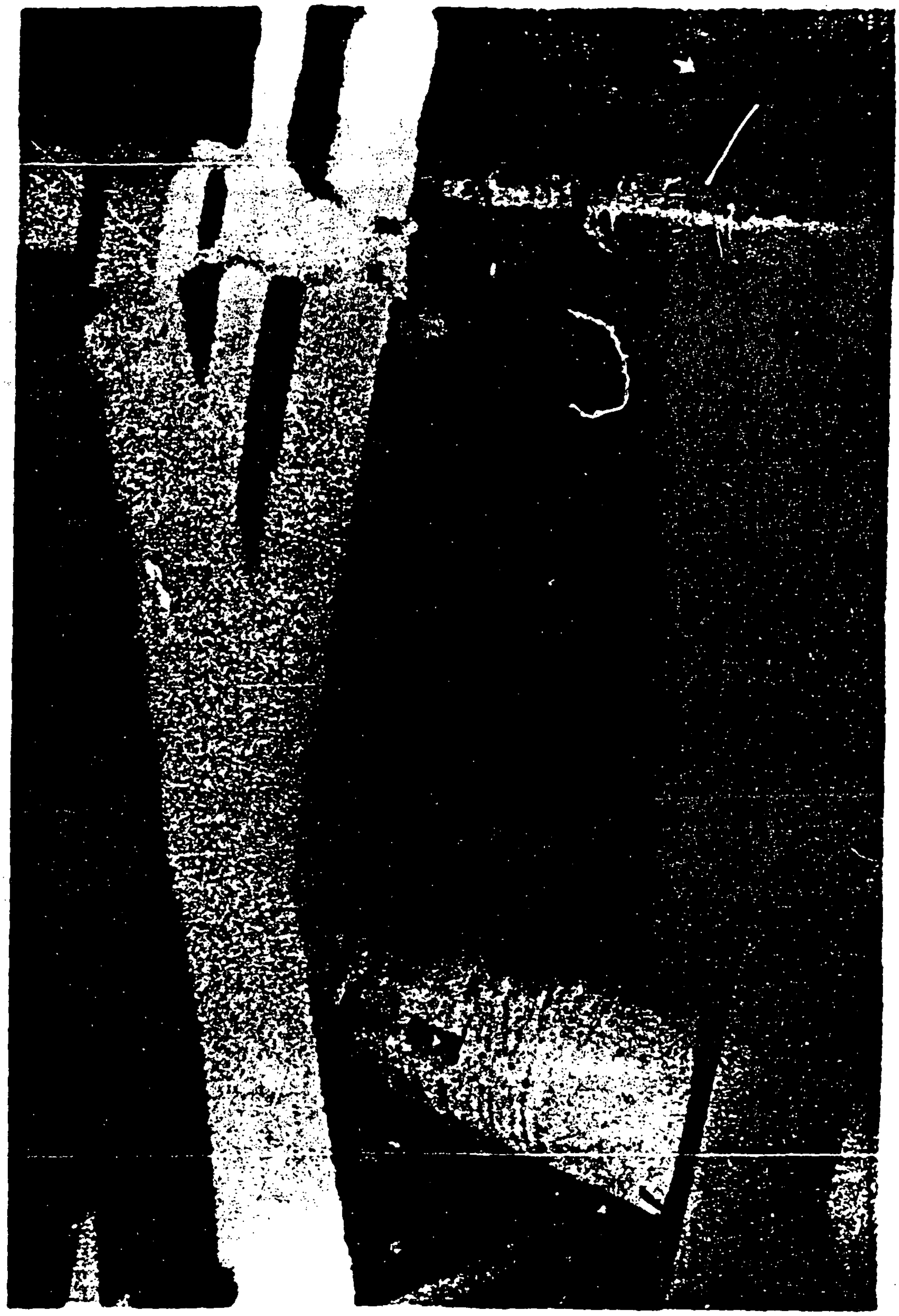


This is a close-up view of the top cable tray. This view clearly shows 1) the thickness of the wall between the Cable Spreading Room and the Reactor Building, 2) a portion of the fire stop and 3) the products of combustion from the burning cables in the top tray.

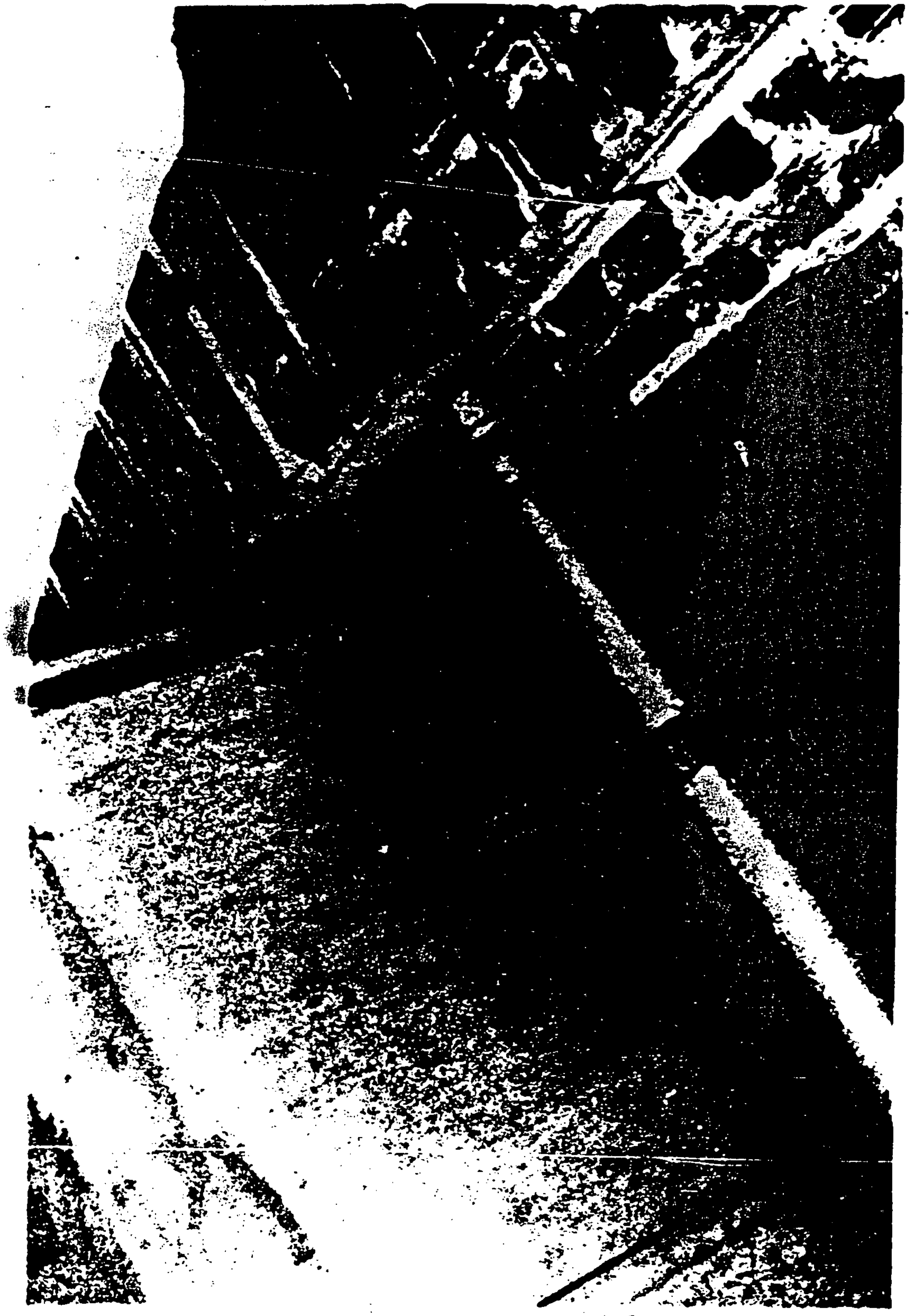


Looking up, this photo was taken in the Reactor Building. It shows the cable trays coming through the fire wall separating the Cable Spreading Room and the Reactor Building. Note the multi-tiered trays, the dry chemical extinguishment (white), the tray corrosion, the exposed conductors and the scaffolding.

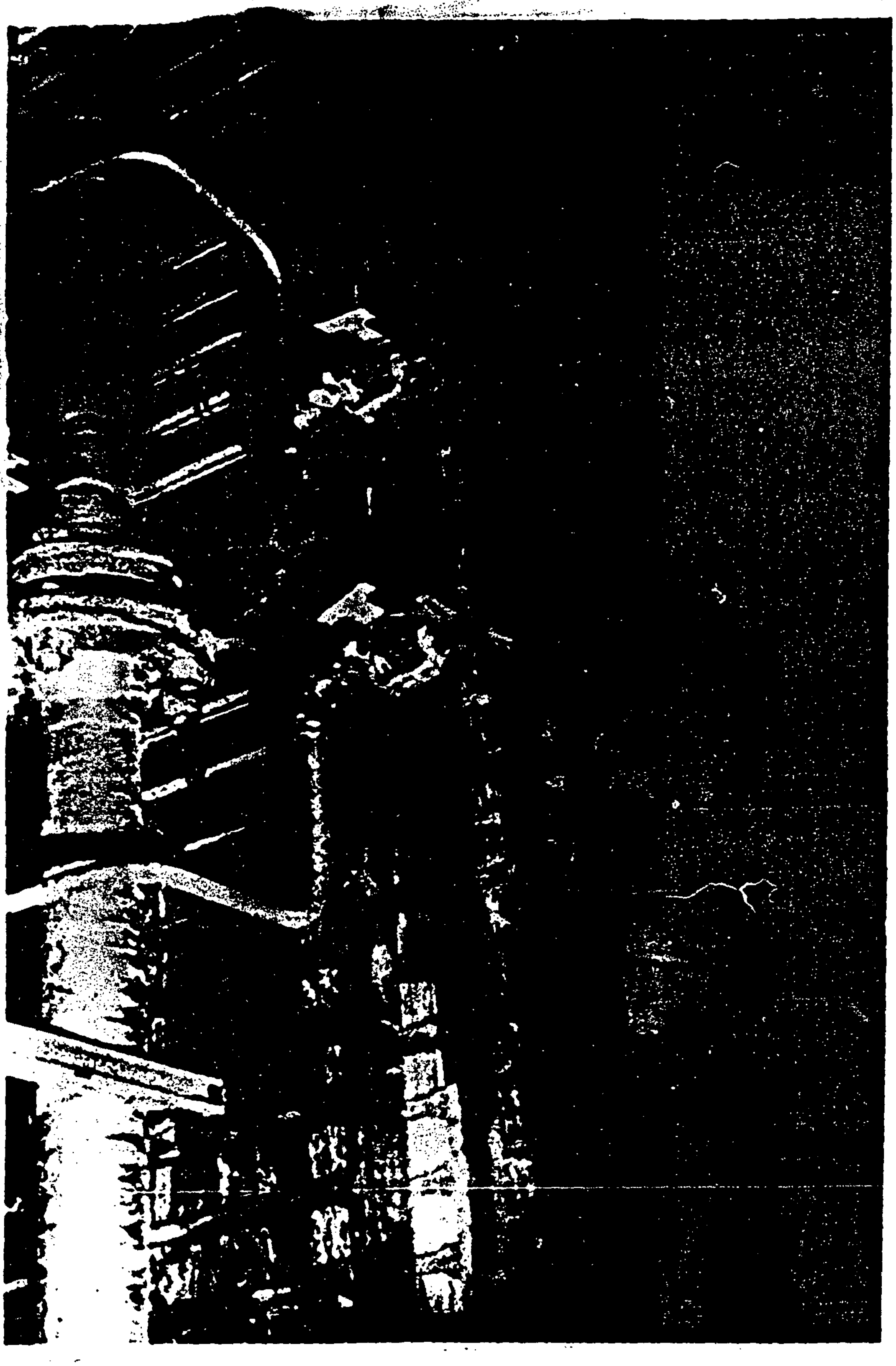
This is another close-up view of the five high, multi-tiered cable trays passing through the Reactor Building wall. Observe the cable damage, soot on the ceiling and acid marking on the conduit.



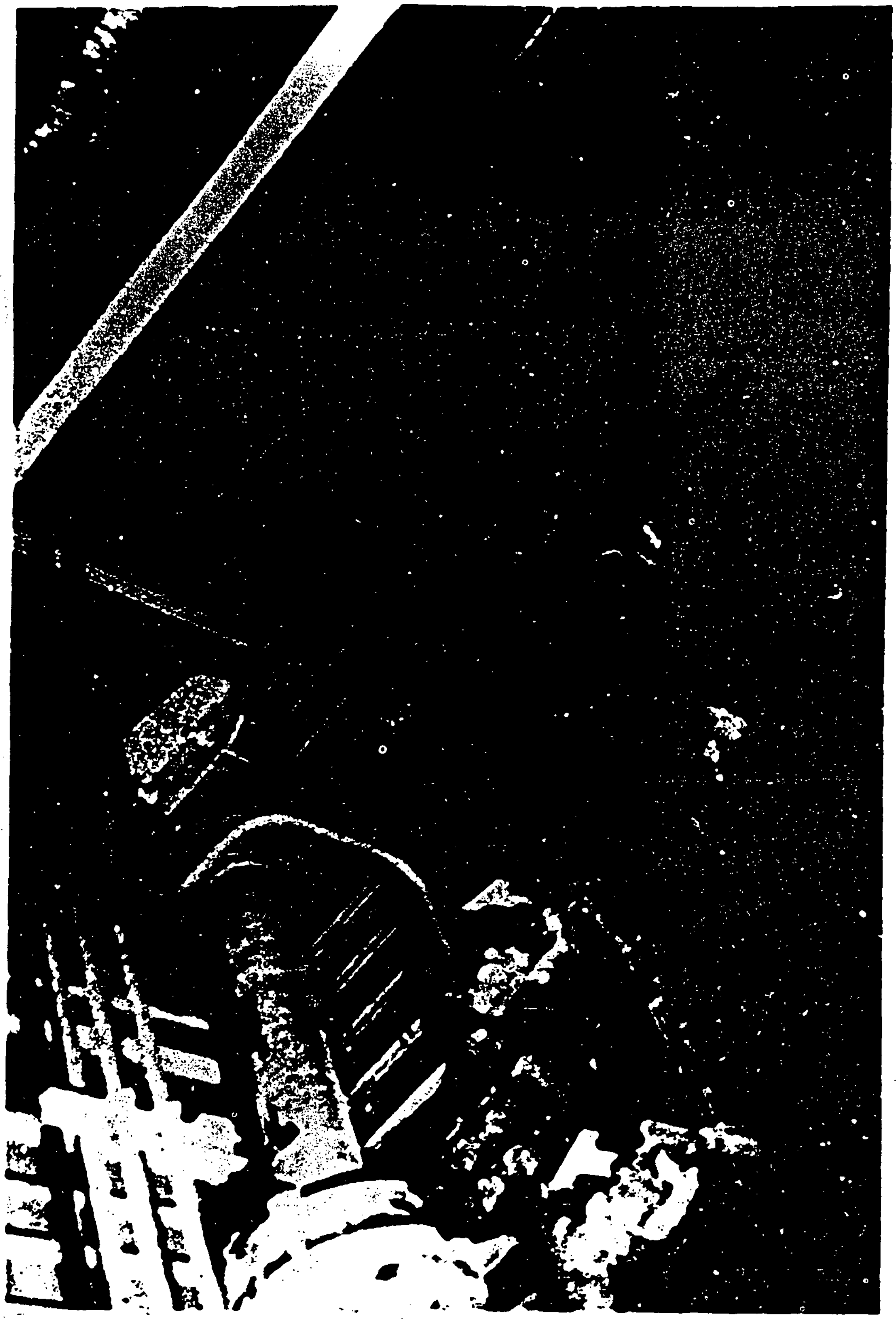
Taken from the floor, this photo shows a four high tray array approximately 15–20 feet from the cable penetration. Imagine the difficulties encountered putting out the fires in these trays some 20–25 feet overhead.



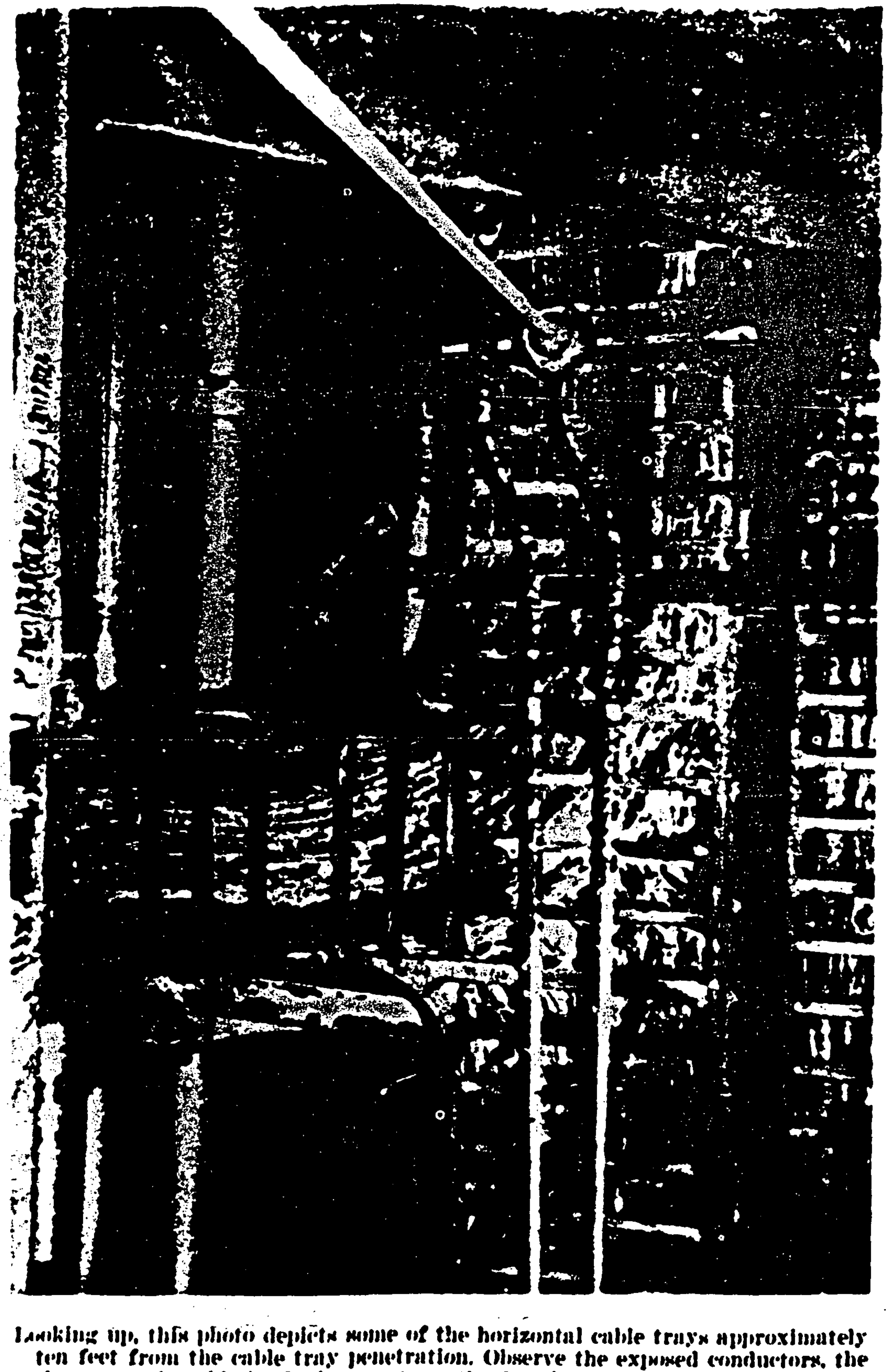
This is a close-up view of Photo No. 6. Observe the tray corrosion (a bluish-green color), the carbon deposits on the walls and ceiling and the exposed conductors in all four trays.



This is another view of the cable trays in Photos No. 6 and 7 but taken from the other side.

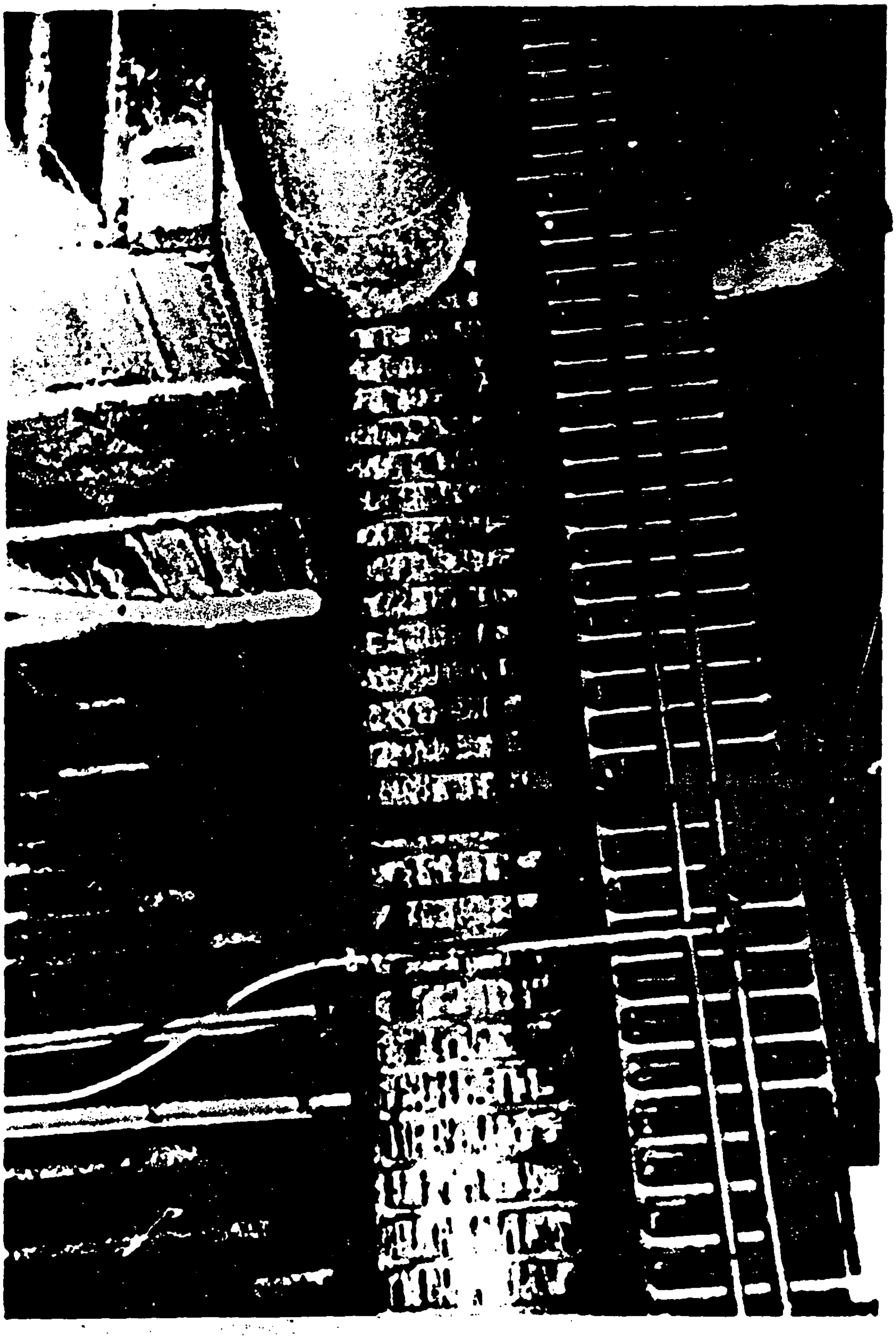


This photo shows the cable trays in Photos No. 6–8 passing through the wall directly opposite the Reactor Building-Cable Spreading Room wall. It was reported the fire penetrated this wall and continued another 3–4 feet.

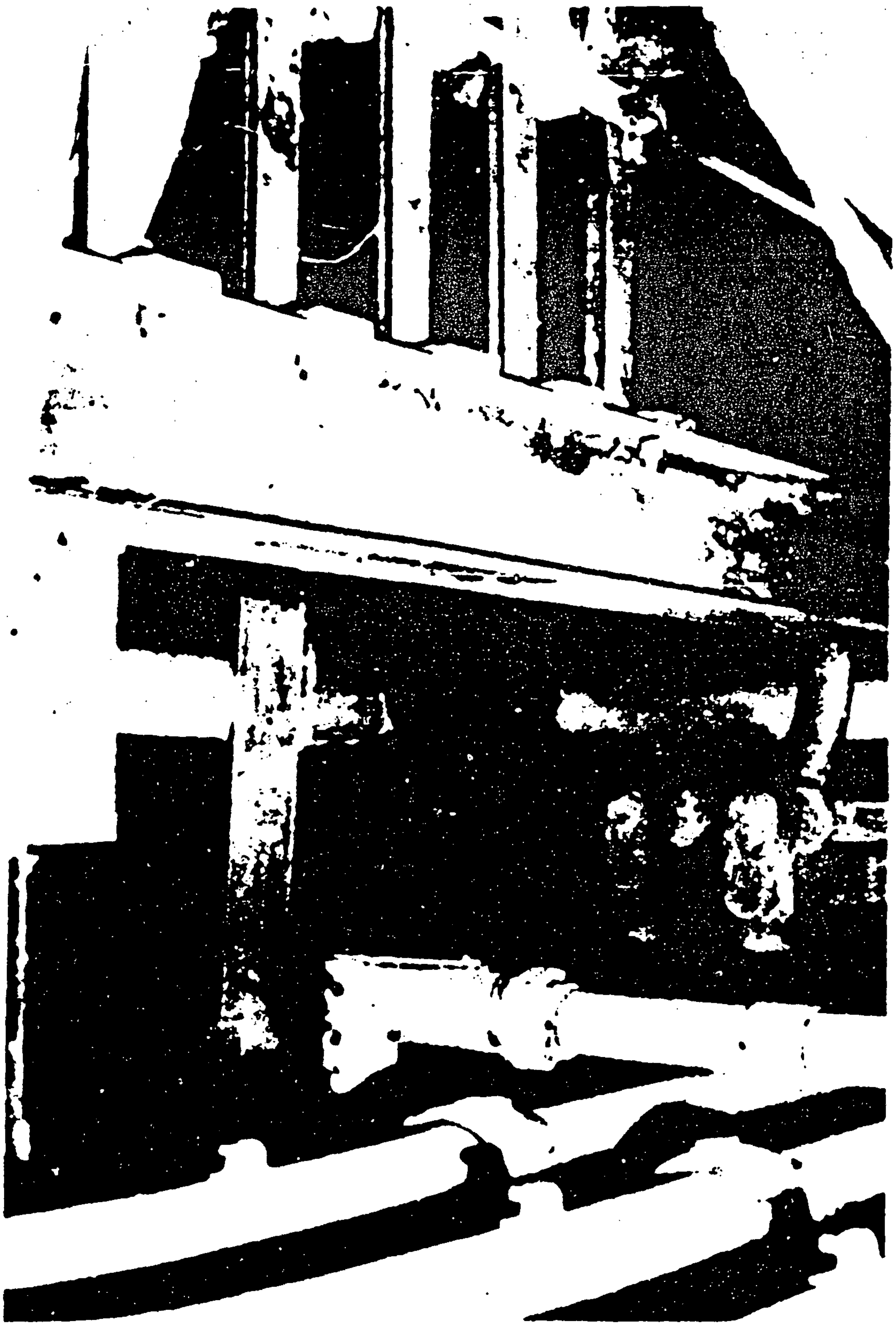


Looking up, this photo depicts some of the horizontal cable trays approximately ten feet from the cable tray penetration. Observe the exposed conductors, the thermocouple cable in the bottom tray, the dry chemical powders, the scaffolding and the close proximity of the parellel runs of vertically tiered trays. Apparently, redundant circuits were located in the upper cable tray.

Once again looking up is a view of the horizontal cable trays approximately 15-20 feet away from the 4' x 4' cable tray penetration. Note where the fire stopped in the vertical trays in the center of this photo whereas in the top right corner tray, the fire continued downward 7-8 feet.



This photo illustrates how far the cable fire spread (35–40') in the direction parallel to the Reactor Building-Cable Spreading Room wall. Note the damage to the cable in the adjacent cable tray (upper), the dry chemical powders in the bottom tray, the acid markings on the conduit and the carbon deposits on the ceiling.



The cable fire also affected a 4160 volt power supply mounted inside an aluminum conduit. Either the heat of the fire or electrical arcing melted this portion of electrical conduit, exposing the conductors.

This is a close-up view of some electrical conduit penetrating the Reactor Building-Cable Spreading Room wall. Note the effects of corrosion on the conduit.

APPENDIX 9

QUESTIONS AND ANSWERS: JCAE, NRC

LETTER DATED SEPTEMBER 18, 1975, MURPHY TO KAMMERER, FORWARDING ADDITIONAL QUESTIONS TO NRC AND NRC RESPONSE.

JOINT COMMITTEE ON ATOMIC ENERGY,
U.S. CONGRESS,
Washington, D.C., September 18, 1975.

Mr. CARLTON KAMMERER,
Director, Office of Congressional Liaison, Nuclear Regulatory Commission, Washington, D.C.

DEAR MR. KAMMERER: Enclosed is a list of additional questions relative to the September 16, 1975 NRC testimony on the Browns Ferry fire which the Committee would like you to provide answers to for the record. We also wish to receive your responses to several other matters raised during the hearing and for which it was agreed that material would be provided for the record. These include, but are not limited to, (a) a listing of the inspections that had been conducted on the Browns Ferry cable room and a copy of the most recent inspection report, (b) an analysis of agreements and disagreements between the NRC investigation report and the report prepared by the Nuclear Energy Liability and Property Insurance Association (NELPIA), (c) material addressing the probability of the Browns Ferry accident from the Rasmussen analysis viewpoint, and (d) an analysis of the paperwork required as a result of the accident, with specific regard to the NRC investigatory process. Please refer to the transcript of the hearings (to be provided to you within the next day or two) to determine the additional responses required.

Since we are endeavoring to publish and distribute the hearing record as rapidly as possible, it would be appreciated if you would provide your answers to the enclosed questions and the other responses no later than October 1, 1975.

Thank you for your cooperation in this matter.

Sincerely yours,

GEORGE F. MURPHY, Jr.,
Executive Director.

Enclosure.

QUESTIONS ON NRC TESTIMONY

Browns Ferry Hearing—September 16, 1975

1. On page 4 of Dr. Knuth's testimony, it is stated that use of an open flame to test for air leakage has been a relatively common practice in the utility industry. Was NRC aware of this practice before the fire, and had it been approved?

2. On page 14 of Dr. Knuth's testimony, it is noted that TVA has taken exception to some of the findings in NRC's investigation report, and that NRC is evaluating the validity of TVA's rebuttal. Does NRC have any conclusions yet on whether or not TVA's positions are valid?

3. On page 15 of Dr. Knuth's testimony, it is indicated that the use of punitive sanctions against TVA is not warranted. Could you describe in further detail the basis on which NRC reached this conclusion, considering the many deficiencies noted in its investigation report and the serious situation that resulted?

4. Apparently, the installation of the cable tray penetration sealant and fire retardant coating had not been completed prior to operations of Units 1 and 2 (Investigation Report—I-2). How did this manage to slip past detection by both TVA and NRC?

5. The procedures for notifying States and local support agencies left something to be desired. How closely had NRC reviewed these procedures?

6. How does NRC reconcile the need for water deluge systems in cable rooms (page 7 of Dr. Hanauer's testimony) with TVA's contention that the use of water might have led to further loss of control of the reactors?

7. The Committee understands that a petition has been filed to change the NRC regulations to require plant operators to distribute emergency plans to the public and conduct yearly drills. What is the status of NRC's consideration of this matter?

8. Over the years, the primary emphasis in nuclear safety has been placed on hypothetical nuclear accidents, such as loss of coolant accidents and reactor excursions. The fire at Browns Ferry and other fires over the years would appear to suggest that the accidents that have occurred at nuclear plants have been more conventional in nature. In this regard:

(a) What consideration does NRC give to fires and other conventional accidents in its safety reviews?

(b) How is NRC organized to assure such hazards are recognized; i.e., does NRC have any special expertise in fire detection and prevention?

(c) Does NRC intend to make any organizational changes to put increased emphasis on preventing fires and other conventional hazards?

9. Mr. David Comey has made two specific recommendations to NRC as a result of the fire:

(a) That operation of nuclear units not be permitted at stations where additional plants are under construction.

(b) That all reactors not fully complying with IEEE standards on separation of electrical cables be shut down.

What is the status of NRC's actions on these recommendations?

10. What is NRC's position regarding the continuing applicability of technical specifications in the event some departure from them is required as the result of an on-site emergency?

11. Please provide background on why the polyurethane material of the type tested by the applicant was not the type used. Also, discuss why these tests were relied upon, when it has been documented that small scale laboratory tests are inadequate to determine the fire properties of a material in large-scale applications. (AEC Serious Accidents Bulletin No. 336, September 19, 1974). Finally, why was polyurethane foam approved for use in such an inaccessible location, when the fire hazards of this practice have long been known? (AEC Health and Safety Bulletin No. 170, dated September 10, 1963 recommends that consideration be given to attaching polyurethane so that it can be removed more easily in case of fire).

[Nuclear Regulatory Commission response, subsequently received, follows:]

U.S. NUCLEAR REGULATORY COMMISSION,
Washington, D.C., October 3, 1975.

Mr. GEORGE F. MURPHY, Jr.,
Executive Director, Joint Committee on Atomic Energy, Congress of the United States.

DEAR MR. MURPHY: Enclosed for inclusion in the record of the September 16, 1975 hearing of the Joint Committee on the Browns Ferry Nuclear Plant fire are the following:

1. Responses to requests for supplemental information made during the hearing.
2. Responses to the eleven questions transmitted to the NRC with your letter of September 18, 1975, to the Director, Office of Congressional Affairs.

Your letter called attention to four matters as having been raised during the hearing on which material should be supplied for the record, namely, (a) list of inspections of the cable room and a copy of the latest inspection report, (b) an analysis of the Nuclear Energy Liability and Property Insurance Association's report and NRC comments thereon, (c) material addressing the probability of the Browns Ferry fire from the Rasmussen analysis viewpoint, and (d) an analysis of the paperwork resulting from the occurrence.

Items (a), (b), and (d) are included in the responses contained in Enclosure 1. With respect to your request for additional material regarding the probability of the Browns Ferry fire—item (c)—the testimony presented at the hearing by Mr. Saul Levine (transcript pages 112-114) reflects all the information presently available on this matter. The analyses that formed the basis for the statement at the hearing are still preliminary, and are being finalized in preparation for the publication of the final report on the Reactor Safety Study. The material requested concerning Browns Ferry is expected to be developed in about a month, and will be furnished to the Joint Committee when completed.

Sincerely,

BEN C. RUSCHE,
(For Lee V. Gossick),
Executive Director for Operations.

Enclosures:
(1) Items Requested for the Record.[1]
(2) Questions on NRC Testimony.

1. On page 4 of Dr. Knuth's testimony, it is stated that use of an open flame to test for air leakage has been a relatively common practice in the utility industry. Was NRC aware of this practice before the fire, and had it been approved?

Response: As stated in Dr. Knuth's testimony, the use of an open flame to test for air leakage in checking condenser vacuum has been a practice in the utility industry. The condensers are generally not located in areas containing vital safety equipment. Responses to IE Bulletins following the Browns Ferry fire indicate that open flames had been used for checking penetration leakage in areas containing vital safety equipment at three other reactor sites.

To the extent that many people on the NRC staff have had experience in the utility industry, the practice of using an open flame to check for condenser leaks was known to exist. The practice was neither explicitly approved nor disapproved by NRC in the condenser application. The more usual practice in industry, as indicated in the responses to our bulletins following this fire, is to administratively control the use of open flames in areas containing vital safety equipment by means of an "open flame" work permit.

2. On page 14 of Dr. Knuth's testimony, it is noted that TVA has taken exception to some of the findings in NRC's investigation report, and that NRC is evaluating the validity of TVA's rebuttal. Does NRC have any conclusions yet on whether or not TVA's positions are valid?

Response: Since the JCAE hearing on the Browns Ferry fire, the I&E Staff completed it evaluation of TVA's response to the NRC Notice of Violation. A copy of its letter to TVA dated September 29, 1975, on this matter is enclosed. The accident investigation report, TVA's response, and the I&E Staff evaluation of the TVA response will all be reviewed by the Special Review Group and by the Commission.

NUCLEAR REGULATORY COMMISSION,
REGION II,
Atlanta, Ga., September 29, 1975.

Attn.: Mr. J. E. Watson
TENNESSEE VALLEY AUTHORITY,
Manager of Power,
Power Building,
Chattanooga, Tenn.

GENTLEMEN: This refers to your September 2, 1975 response to the Notice of Violation sent to you by this office on July 28, 1975.

Of the items of apparent noncompliance in Appendix A of our July 28, 1975 Letter, we have no further questions on Items 1, 2a(1), 2a(2), 3a, 3b, and 3c. Appendix A to this letter notes that item 3d(2) is withdrawn as an item of noncompliance and gives our position on your reply to Items 2a(3), 2a(4), and 3d(1). Appendix B of this letter supplies our position on certain of your replies to the Areas of Concern contained in our letter of July 28, 1975. We will be examining these matters during future inspections and no further reply is requested of you at this time.

Appendix C to this letter provides our comments to certain of your responses on the Conclusions and Summary of Findings of Facts sections of our investigation report.

If you have any questions regarding these matters, please contact me.

Sincerely,

NORMAN C. MOSELEY,
Regional Director.

Attachments:
Appendices A, B, C.

[1] Included at appropriate point in record.
[2] See Appendix 14 for September 2, 1975 letter.

IE POSITIONS RELATIVE TO TVA RESPONSE TO CERTAIN ITEMS OF NONCOMPLIANCE IDENTIFIED IN THE LETTER OF SEPTEMBER 2, 1975, FROM J. E. GILLELAND, TVA, TO NORMAN C. MOSELEY, NRC

APPENDIX A

Item 2.a(3)

The Browns Ferry Procedure for "*Fire—Explosions—Natural Disasters,*" contains the following:

"Shift Engineer on Duty Will:

1. Proceed to scene and appraise situation (the Shift Engineer may choose to remain in the control area and delegate on-scene responsibility to an Assistant Shift Engineer) . . .

5. Remain at the scene and supervise activities."

A normal shift complement at Browns Ferry is one Shift Engineer who is in overall charge of the entire plant and at least one Assistant Shift Engineer for each unit. In the event of a fire the TVA procedure requires delegation of on-scene responsibility when the Shift Engineer is not at the scene to supervise activities. In this case the Shift Engineer departed from the scene without making such delegation. This constitutes noncompliance with TVA procedures, adherence to which is required by TVA technical specifications. Although the Shift Engineer may have thought the Assistant Shift Engineer would assume this responsibility, the TVA procedure is clear in requiring specific delegation.

Our position continues that failure to make a specific delegation of responsibility constitutes noncompliance.

Item 2.a(4)

The first sentence of the quotation from the Browns Ferry Standard Practices Manual BFS3 is "Plant fire protection systems shall be fully operational at all times." (emphasis added). At the time of the fire, the Cable Spreading Room CO_2 installed system was not fully operational because (1) the metal plates were installed under the breakout glass used in the mechanical valve actuation method and (2) the manual electric system was disabled on the Unit 1 side by a switch located on Unit 2 side. From the Unit 1 side manual electric is the only actuation method, therefore, at the time of the fire the CO_2 system could not be promptly activated at all from that side because the switch on the Unit 2 side had de-energized the control on the Unit 1 side. Accordingly, the CO_2 system could not be considered fully operational.

Our position continues that this item represents noncompliance since the system was not fully operational.

Item 3.d(1)

In amendment 25 to the FSAR, beginning on page R7.5–11, the following sentence appears: "The power cables from the 4160 volt shutdown boards to the respective transformers for the 480 volt shutdown boards and the 480 volt diesel auxiliary boards are installed in separate conduits."

Whatever the original design classification of the 4 Kv shutdown board cables was, it is clear that the quoted statement from Amendment 25 requires that they be in separate conduits. Accordingly, the as-installed condition is not in compliance with the FSAR as amended.

Item 3.d(2)

The table on page III–8 of our investigation report is in error in that 1C Core Spray and Residual Heat Removal Normal Valve power supply is 4 Kv SD Board A through NOV Board 1B. Therefore, we concur that Division I and II separation was not violated in the cases discussed in this item. This apparent item of noncompliance is withdrawn.

IE POSITIONS RELATIVE TO TVA RESPONSE TO CERTAIN AREAS OF CONCERN IDENTIFIED IN THE LETTER OF SEPTEMBER 2, 1975 FROM J. E. GILLELAND, TVA, TO NORMAN C. MOSELEY, NRC

APPENDIX B

Item 4

"Plant operations personnel and Public Safety Services personnel at the plant were not familiar with the reactor building fire extinguishing equipment in that they did not know that the nozzle on the hose was specifically designed for fighting electrical fires."

This concern is based in part on actions by Shift Engineer U and Assistant Shift Engineer Y. Both of these individuals participated in the first attempt to use water. The attempt was not successful and led to a decision to try the nozzle from the Athens Fire Truck. Had these individuals been knowledgeable of the characteristics of the nozzle on the hose, it is doubtful that they would have concluded that another nozzle would perform better. These individuals had received TVA fire training and have job responsibilities for fire fighting. These actions led to a conclusion that these key people did not demonstrate knowledge of the equipment available. Your September 2, 1975, letter stated that additional training has been accomplished. This area will be reviewed in future inspections.

Item 5

"Emergency breathing apparatus appears not to have been properly maintained."

This concern was developed from statements made by various personnel who were interviewed by the investigators. Excerpts from specific statements included in the investigation report are as follows:

(1) *ASE-XXXX.*—"The unit that PSO-CC had would not puncture the canister and I don't remember the problem with the other unit. But the Chem-Ox mask units did look to be in sad shape in my opinion."

(2) *IE-JJJJ.*—"I heard that the air supplies were hard to find and did not fit well together."

(3) *AUO-Q.*—"About half the airpacks were unusable because they were marked 'not charged' or 'no mask'." In an explanatory note, AUO-Q stated, "This is very misleading. The airpacks that were brought to the fire site by some of the maintenance people were unuseable due to marking. They simply failed to notice the markings. We had enough airpacks at the site but a lot of the airpacks brought to the fire site were not charged or had the masks in them."

These statements coupled with additional statements concerning lack of available air in cylinders led to our concern that maintenance activities may not have been promptly performed on emergency equipment.

Item 7

"There was apparently no attempt by CECC management to obtain expert advice on methods of fighting the fire."

The plant superintendent, by his discussions with TVA management, gave evidence of his concern regarding his decision not to use water on the fire at an early stage. In view of the fact that local expert advice for earlier use of water—first given by the Athens Fire Chief at about 2:00 p.m. was not heeded—it appears that TVA management should have secured and provided expert advice to the plant superintendent on methods of fighting the fire. The investigation revealed no positive recommendation—nor definitive advice—by TVA management concerning use of water. It appears that such recommendations are an assigned function of the off-site management control in an emergency such as this fire.

We continue to believe that earlier use of water to extinguish the fire would have reduced the impact of the fire on your plants. The increasing inoperability of equipment as the fire progressed, in our view, supports the position that earlier positive extinguishing of the fire by water was called for. The water nozzle at the scene was designed to permit its use with minimum hazard to personnel.

Item 10

"There was apparently no attempt by the CECC Director to direct the operations of the center. The plant superintendent was involved in extensive communications with the center rather than directing the fire fighting and recovery operations. As a consequence the efforts at both locations were not coordinated, management personnel were not provided with accurate information and their actions were relatively ineffective."

The information gained during the IE investigation indicates that the control centers served more as communication centers than as centers to provide an overseeing management control of the situation. Particularly difficult to identify were examples where the control centers evaluated and made recommendations concerning the situation and actions taken to assure that support efforts were adequate.

We could identify no control center review of the measures being used to control the emergency. The control centers appeared to be deeply involved in the hour-by-hour unfolding of the situation rather than in providing the technical and managerial perspective which were needed for a more controlled response to the situation.

Item 11

"Although plant procedures specify that the unit operator has the authority to shutdown the reactor in the event of an emergency, his responsibility to effect a shutdown promptly has not been defined by management."

Your response states that the operator "may operate on his own initiative" (emphasis added) to take actions in emergencies where there is not time to consult with supervision. This position, more positively stated, should be emphasized to your operators as the TVA management position. We believe that the operator has the positive responsibility—shall operate—to take actions on his own initiative when significant, unexplained anomalies occur and he does not have time to consult with supervision.

IE COMMENTS TO TVA'S RESPONSE CONCERNING CONCLUSIONS AND SUMMARY OF FINDINGS OF FACTS IN THE LETTER OF SEPTEMBER 2, 1975, FROM J. E. GILLELAND, TVA, TO NORMAN C. MOSELEY, NRC

Since the Items of Noncompliance (Appendix A) and Areas of Concern (Appendix B) are founded on the Conclusions and Summary of Findings of Facts sections of the investigation report, those comments in Appendix C deal only with matters not already addressed in Appendix A and Appendix B.

APPENDIX C

CONCLUSIONS

B.1.(c)

"Inadequate training of TVA personnel in fire fighting procedures and equipment."

This conclusion is based on the overall response of TVA in fire fighting founded on statements made by individuals during the IE investigation. Some individuals stated they had received no fire fighting training; others said that the training they had received was some years prior to the fire.

B.2.(b)

"Lack of initiative on the part of onsite supervisory personnel to coordinate shutdown activities until after the first manual initiation of depressurization of Unit 1."

Collective finding of the IE investigating team indicates that there was a lack of coordination of shutdown and restoration activities until after the initial manual depressurization of Unit 1. Though TVA does not agree with our conclusion regarding the extent of direct supervisory involvement in the decision to initiate the blowdown of Unit 1, it is based on the investigators' discussions with those involved and rests heavily on the impression that the operator did not believe he had been directed to initiate the blowdown. In any case, the plant superintendent's involvement in coordinating shutdown activities during the time in question could not have been extensive since he arrived onsite between approximately 1:20 p.m. and 1:35 p.m. The initiation of manual blowdown of Unit 1 occurred at approximately 1:40 p.m. The conclusion, therefore, is based primarily on the actions of supervision on the site during the entire period in question.

B.2.(c)

"Failure to establish priorities for restoration of equipment. This resulted in uncoordinated, and in some cases, counter productive or duplicate attempts to restore key equipment to service."

This conclusion is related to activities prior to extinguishing the fire. Our investigation revealed no evidence of a systematic approach to establishing response action alternatives and the associated priorities during that period.

A.5

The location of the fire barrier in the penetration did not conform to the approved design drawings.

We concur with your contention that the location of the fire barrier in the penetration would have no bearing on the function of the fire barrier if the installation had been completed with the flameproof coating. As discussed, however, in the conclusion section of the report (IE Inspection Report, Summary Section B.1.(h)), the inaccessibility of the initial ignition location due to the position of the penetration contributed to the cause of the fire and to its severity. If the penetration had been located ten inches closer to the spreading room wall as specified in the design drawing, the likelihood of the fire would have been decreased and initial efforts at extinguishing the fire would have been aided through improved accessibility.

A.14

"Contrary to good safety practices, the plant procedures do not restrict the use of elevators during fires; both operations and construction personnel used the plant elevators while the fire was in progress."

In the event of the failure of an elevator during a fire, any passengers would be exposed to hazards created by that fire until rescued. Although no one was trapped, at least one elevator used by Browns Ferry personnel failed during the fire.

D.1

"The Athens Fire Department (AFD) responded promptly to the notification of the fire and was prepared to assist within approximately 35 minutes of receiving the alarm. Athens is about ten miles from the site and, upon arrival, the firemen had to be issued personnel radiation monitoring devices prior to entry."

The information concerning issuance of radiation monitoring devices to firement prior to plant entry was included to explain the 35 minute time interval, not to criticize the practice. We do believe that attention should be given to minimizing delays in emergency response while maintaining due attention to safety and security concerns.

D.6

"The Director of the CECC was not aware that the fire had been extinguished until approximately 8:45 p.m. when in fact the fire was extinguished prior to 7:45 p.m."

We do not agree that the Director of the CECC was aware at about 7:45 p.m. CDT that the fire had been extinguished as you maintain. The CECC Director's log and that of individual CECC–AF, a co-worker, indicate that at about 8:05 p.m. CDT the State of Tennessee was advised of the fire. The State of Tennessee records indicate that they were advised of the fire at 8:15 p.m. CDT. Neither the Director nor CECC–AF, therefore, were aware that the fire was out at about 8:05 p.m. CDT.

Subsequent to the call to Tennessee, an untimed late entry was made which stated: "(no time) 8:45 p.m. fire (reported) out." The next entry was made at 10:30 p.m. CDT. We concur with you that it is conceivable that the time in the Director's log could be 8:45 p.m. EDT rather than CDT. This does not establish that the Director was aware at 7:45 p.m. CDT that the fire was out since the entry was made subsequent to 8:05 p.m. CDT. We conclude that the Director became aware that the fire was out between about 8:15 p.m. and 10:30 p.m. CDT, the time of the next log entry. The terminology of the untimed entry indicates that it was at 8:45 p.m. when the event was reported to him.

D.7

"Communications by the CECC with State and local agencies focused on plant operating status rather than offsite radiological releases which is the prime responsibility of these agencies."

The results of the investigation including interviews with personnel from the states and reviews of their records indicate that much of the information provided by TVA directly or in response to questions was in the nature of "raw" operational details rather than specific information which would facilitate evaluation by the states of the probability and nature of possible offsite radiological releases.

We did not intend to imply that *only* offsite radiological release information is to be provided to the states. The states do need to know plant operational status in order to plan their activities in their jurisdiction.

3. On page 15 of Dr. Knuth's testimony, it is indicated that the use of punitive sanctions against TVA is not warranted. Could you describe in further detail the basis on which NRC reached this conclusion, considering the many deficiencies noted in its investigation report and the serious situation that resulted?

Response: The primary emphasis of the NRC's enforcement program has been to bring about corrective action on the part of violators. Punitive action has not been a primary objective of the enforcement program. This philosophy is set forth in a letter dated December 31, 1974 from Dr. Donald F. Knuth to all licensees transmitting the criteria used by the Staff in determining the appropriate enforcement for any given situation. Copies of the December 31, 1974 letter and the enforcement criteria are enclosed.

NRC formal enforcement actions carried out by the Office of Inspection and Enforcement are taken in accordance with IE Manual Chapter identified as MC-0800 which is part of the public record. Under the enforcement criteria, the action decided upon in a given case depends essentially on three factors:

The significance of each time of noncompliance,
The total number of items of noncompliance, and
The enforcement history of the licesnee involved.

These factors require the exercise of judgment in their application to specific situations. To assist in guiding this judgment general criteria have been established. These criteria categorize items of noncompliance into three levels of significance as follows: violations, infractions and deficiencies.

The most significant items of noncompliance are denominated "violations". Violations are defined in terms of items of noncompliance which would lead to or result in relatively severe types of incidents or occurrences such as, for example, a radiation exposure exceeding 50 times the established regulatory limits.

The next most significant items of noncompliance are denominated "infractions." Infractions" are defined in terms of several specific types of incidents or occurrences of a less severe nature than those amounting to "violations." For example, a radiation exposure merely exceeding established regulatory limits would normally be an "infraction."

The least significant item of noncompliance is defined as a "deficiency." Deficiencies are minor matters such as failures to comply with records, posting, or labeling requirements.

In addition to the categorization of items of noncompliance into levels of significance, further guidance in the exercise of the judgment as to the appropriate enforcement action in any given case is set forth in the Office of Inspection and Enforcement Manual. There a point system is established to guide the determination of what enforcement action is appropriate in a given situation.

Basically, three levels of enforcement action are available to the NRC in the discharge of its enforcement responsibilities:

Notice of violation.
Civil penalty.
Order modifying, suspending or revoking a license or to cease and desist from an unauthorized activity.

The selection of the appropriate enforcement action, as indicated above, is made after consideration of both the significance of the individual items of noncompliance as well as the accumulated number of points assigned to the total noncompliance items. Thus, for example, under our criteria, a single serious item of noncompliance could result in a civil penalty or even an order modifying, suspending, or revoking a license. On the other hand, a whole series of minor items of noncompliance might not result in a sufficient number of points to warrant anything more than a notice of violation under our criteria.

Of course, the point system is merely guidance. Similarly, the important determination as to whether an item of noncompliance falls into this category or that category can never be completely automatic. Judgment is always involved. Indeed in the definition of "violation" there is a catch-all category:

"Other similar items of noncompliance having actual or potential consequences of the same maugnitude."

Each of the items of noncompliance in the Browns Ferry case was evaluated under our criteria as to its level of significance. None of the items of noncompliance could be reasonably considered as a "violation" since none of the five QA items nor the one technical specification item can fairly be said to have caused, contributed to or aggravated (or to have had substantial potential for so doing) an incident or occurrence of the type set forth in our criteria defining "violations." In view of the catch-all category "other similar items of noncompliance having actual or potential consequences of the same magnitude," the position could to taken that inadequate Q/A contributed to a serious fire and that a serious fire is a "consequence of the same magnitude," as, for example, radiation levels exceeding 50 times established limits. However, the fire "resulted in a reduction of preventive capability below requirements but redundant controls precluded an item of noncompliance of the violation category"—an explicit definition of an "infraction." Consequently, the individual items of noncompliance all are categorized as infractions-redundancy was reduced but still existed. Evaluating the six "infractions" revealed by the investigation against our guidance did not result in the number of "points" normally associated with a civil penalty.

Based on this evaluation, a civil penalty was not considered to be the appropriate enforcement action because no single item of noncompliance reached the significance of a violation nor did the aggregate of the items of noncompliance reach the significance of a violation.

U.S. ATOMIC ENERGY COMMISSION,
Washington, D.C., December 31, 1974.

To: All AEC Licensees.

CRITERIA FOR DETERMINING ENFORCEMENT ACTION AND CATEGORIES OF NONCOMPLIANCE WITH AEC REGULATORY REQUIREMENTS—MODIFICATIONS

On November 1, 1972, the Commission issued criteria for enforcement actions to be taken for noncompliance with its rules and with license conditions in accordance with Sections 161, 186, and 234 of the Atomic Energy Act and Subpart B of Part 2, 10 CFR. On June 5, 1973, the Commission notified licensees that categories of violation with AEC regulatory requirements had been established because the Commission and the nuclear industry recognized that the significance of violations varies in the potential for affecting the health and safety of the public, the common defense and security, and the environment.

Based on a review of the experience with the criteria for determining enforcement action and the categories of noncompliance, modifications of the use of these criteria and these categories are being made. Comments explaining the modifications are enclosed as Attachments A and B.

The changes in the criteria and categories are primarily administrative in nature and should result in a higher level of understanding of the enforcement program—and the results of the program—on the part of the public and the industry. The basic purpose of the enforcement program—enhancement of the health and safety of the public, the common defense and security, and the environment—remains the same. The long standing practice of requiring corrective action for each identified item of noncompliance (violations) is not changed. The enforcement program continues to emphasize corrective action where necessary to assure that regulated activities meet applicable requirements and are conducted with due regard for public health and safety, common defense and security and protection of the environment.

The modifications clarify the enforcement criteria and categories of noncompliance in the areas of safeguards and environmental matters and provide more explicit definitions to aid in a better understanding of the enforcement program. These definitions make clear the applicability of the program in matters of quality assurance, management control, and systems performance. Also, because the Commission relies to a degree on reports from licensees to assure that timely corrective action is taken and to assure that the industry is notified of important matters of generic interest, a reporting requirement is viewed from the enforcement standpoint to be of the same level of importance as the matter for which the report is required. As a part of the correspondence between a licensee and the AEC subsequent to an inspection, notifications will be made to a licensee of apparent failures on the part of the licensee to meet his commitments contained in his application or in correspondence to the AEC and of deviations from appropriate codes, standards or guides.

The levels of enforcement actions available to the Commission in the exercise of its regulatory responsibilities are the same as those set forth in the letter of November 1, 1972. These include written notices of violation, civil monetary penalties, and orders to "cease and desist" or for modification, suspension, or revocation of a license.

The criteria for issuance of a "Notice of Violation" are essentially unchanged.

The criteria for civil penalties have been modified to elaborate upon those situations for which civil penalties may be imposed. The amount of civil penalty in any given case, within the confines of the amounts established by the Atomic Energy Act, is determined by consideration of several factors including:

1. Potential or actual consequences associated with the item of noncompliance. This includes consideration of the categories of noncompliance.

2. Type of licensee. This includes the purpose for which licensed and the quantity, form and kind of radioactive material authorized.

3. The licensee's recent enforcement history, if applicable. This includes the nature and number of items of noncompliance, the frequency of noncompliance, whether items of noncompliance were repetitive of the same or similar requirements, promptness of corrective action, and the licensee's management of its program for assuring compliance with regulatory requirements.

The criteria clarify that repetitiveness of noncompliance or history of noncompliance is not an essential ingredient for consideration for civil penalty. In some cases of a single instance of noncompliance, a civil penalty may be the appropriate enforcement sanction.

The criteria for orders emphasize the importance of quality assurance and are broadened to include all aspects of the regulatory program. Under these criteria, an order to suspend a license or a portion thereof may be issued for authorized activities of licensees or permit holders which are performed in such a manner as to constitute an immediate or potential threat to employees or the public; or for construction deficiencies which, if not suspended immediately, could eventually result in significant or essentially irreversible construction defects which impact on safety or which increase the potential for or the potential severity of an accident. If, for example, a quality assurance requirement for a specific construction activity is not implemented, this activity may be suspended until full compliance with the requirement is achieved.

Regulatory Operations Bulletins and Immediate Action Letters have been used not only to disseminate information but also as a means of accomplishing voluntary action on the part of licensees to inspect, report and make commitments to correct problems on a timely schedule. These two communications are recognized in these revisions. If these methods are ineffective in achieving the desired action, an order may be promptly issued requiring the action.

The enforcement record of a licensee may be a consideration in selecting the appropriate enforcement sanction in any given case. A licensee's enforcement history is evaluated in terms of distribution of items of noncompliance by importance and by the degree of repetitiveness of noncompliance with the same basic requirement. However, regardless of the history, consideration will be given to the more significant enforcement sanctions as a result of any inspection that reveals items of particular importance to safety and management.

The former system of severity categorization, which was the subject of a letter to licensees dated June 5, 1973, has been revised to place items of noncompliance with regulatory requirements (Violations) more clearly in perspective with regard to their relative significance to the public health, safety and interest and the common defense and security. As shown in Attachment B to this letter, the revised system for categorizing violations (items of noncompliance) has three levels of relative importance which are designated in descending order as (1) "violation," (2) "infraction," and (3) "deficiency," each of which is a legal violation in the statutory sense.

It should be recognized that the enforcement criteria and the categories of noncompliance apply only to situations where there is an apparent failure on the part of a licensee to meet regulatory requirements. The licensee may also be notified of deviations from commitments and appropriate codes, standards, or guides. The significance of these failures generally is judged against the actual or potential consequences resulting from the failures and from the standpoint of licensee awareness and management of his program. From the viewpoint of enforcement, a licensee failure that results in the potential for consequences is equally important with the failure that results in the consequences—both represent instances of failure of the licensee to properly perform. However, from the impact of health and safety, common defense and security, the protection of the environment, actual consequences—when the event did occur—and potential consequences—when the opportunity for occurrences exists but the event did not happen—of a item of noncompliance are quite different. In reporting the more important items of noncompliance, those items that caused or resulted in actual consequences will be differentiated from those that merely provided the potential for the consequences.

The enforcement criteria and the categories of noncompliance apply to situations where there is an apparent failure on the part of a licensee to meet regulatory requirements, commitments, and appropriate codes, standards or guides. There do occur events—such as some equipment malfunctions—at licensee facilities which are not founded in the failure of the licensee to meet requirements, commitments, and appropriate codes, standards, and guides. Such events are not included within the enforcement program.

The enforcement criteria and the categories of noncompliance have been placed in the Public Document Room, 1717 H Street, N.W., Washington, D.C., and a notice has been placed in the Federal Register concerning their availability to all persons upon request.

Sincerely,

DONALD F. KNUTH,
Director of Regulatory Operations.

Enclosures:
A. Criteria for Determining Enforcement Action.
B. Categories of Items of Noncompliance.

CRITERIA FOR DETERMINING ENFORCEMENT ACTION

IN CONNECTION WITH LICENSING AND REGULATORY PROVISIONS OF THE ATOMIC ENERGY ACT OF 1954, AS AMENDED, AND REGULATIONS AND LICENSES ISSUED THEREUNDER

Introduction

The purpose of the AEC enforcement program is the enhancement of the health and safety of the public, the common defense and security, and the environment. The enforcement program emphasizes corrective action, where necessary, to assure that regulated activities meet applicable requirements and are conducted with due regard for public health and safety, common defense and security and protection of the environment. Corrective action is required for each identified item of noncompliance.

Results of AEC inspections and investigations of licensed activities have shown that licensees have not in all cases complied with the regulatory requirements, and it has been necessary to take specific enforcement actions commensurate with the items of noncompliance. This document sets out the criteria for enforcement actions to be taken with respect to future noncompliance with the Atomic Energy Commission's requirements in accordance with Sections 161, 186 and 234 of the Atomic Energy Act and Subpart B of Part 2, 10 CFR.

Levels of enforcement actions available to the Commission

The formal actions available to the Commission in the exercise of its enforcement responsibilities are of three basic types (notices of violation, civil penalties, and orders) which may be applicable to a specific enforcement situation.

1. Written Notices of Violation (10 CFR 2.201).

Notices of Violations are written notices to licensees, citing the apparent instances of failure to comply with regulatory requirements (Violations) which for purposes of categorization have been classified violations, infractions and deficiencies. Such items of noncompliance are generally observed or identified during investigations, inspections, or inquiries.

The same letter enclosing a Notice of Violation may also enclose a notification of apparent deviations from licensee commitments and the provisions of appropriate codes, standards or guides.

2. *Civil Monetary Penalties (10 CFR 2.205).*

The Commission may levy civil monetary penalties against licensees for violations, infractions or deficiencies with respect to requirements in licensing provisions of the Act or any rule, regulation, order, or license issued thereunder. The Commission is required to issue a "notice of violation" to the person charged before instituting proceedings to impose a civil penalty.

3. *Orders to Cease and Desist; and Orders for Suspension, Modification, or Revocation of a License (10 CFR 2.202 and 2.204).*

The AEC has authority to issue orders to "cease and desist," and orders to suspend, modify, or revoke licenses. Such orders are ordinarily preceded by certain procedural requirements, including a written "notice of violation" to the licensee providing him with an opportunity to respond as to the corrective measures being taken. In the event the licensee fails to respond to the notice or to demonstrate that satisfactory corrective action is being taken, an order to show cause may be issued requiring the licensee to show why the particular order (either of revocation, or modification, or suspension) should not be made effective. In some instances where the health, safety, or interest of employees or the public so requires or deliberate noncompliance with the Commission's regulations is involved, the notice provision may be dispensed with and, in addition, the particular order may be made immediately effective pending further order.

In addition to proceeding by way of order, the Commission may also, pursuant to Section 232 of the Act, request the Attorney General to obtain an injunction or other court order to enjoin licensees from violating the Act or any regulation or order issued thereunder.

NOTICE OF VIOLATION—CRITERIA

Section 2.201 of 10 CFR requires that before any formal enforcement action is taken for alleged noncompliance, the AEC will serve on the licensee a written "notice of violation" except when the Director of Regulation finds that the public health, safety, or interest so requires, or that noncompliance is deliberate, the "notice of violation" may be omitted and an order to show cause issued.

Generally, a "notice of violation" may be considered sufficient enforcement action in those cases where:

(*a*) Items of noncompliance are readily correctable, or

(*b*) Items of noncompliance are not repetitive or numerous, and do not constitute an immediate or serious threat to the health and safety of the licensee's employees or the public, to the environment, or to the common defense and security, and

(*c*) There is no indication that appropriate corrective action will not be taken.

CIVIL MONETARY PENALTIES—CRITERIA

The Commission may levy civil monetary penalties on licensees who do not comply with the licensing provisions of the Act or any rule, regulation, order, or license issued. Generally, the type of cases that are appriatepr for civil nplae or license issued. Generally, the type of cases that are appropriate for imposing civil penalties are those involving significant items of noncompliance and which represent a threat (but not necessarily immediate) to the health, safety, or interest of the public, or to the common defense or security, or the environment. As a matter of judgment, civil penalties may be used in lieu of license suspension when there is no immediate threat to the health and safety or the common defense and security and license suspension would deprive the licensee or his employees of their means of livelihood, or the public of essential service.

Civil penalties may be the appropriate enforcement action in cases or situations which meet one or more of the following criteria:

(*a*) Those cases of noncompliance with the same basic requirements that were brought to the attention of the licensee in a "notice of violation" following a previous inspection; or

(*b*) Those cases of noncompliance in which the licensee fails to carry out in a timely manner the corrective action the licensee stated would be taken in response to a previous written notice; or

(*c*) Those cases involving the *deliberate* failure of a person to comply with regulatory requirements;[1] or

(*d*) Those cases involving items of noncompliance in which (1) the licensee's history is one of chronic noncompliance, or (2) due to the nature and number of items of noncompliance, it is apparent that management, having been afforded an opportunity to correct previous items of noncompliance, is not conducting its licensed activities in conformance with regulatory requirements, or

(*e*) Those cases where (1) an order for immediate, but temporary, suspension or to "cease and desist" is issued to remove an immediate threat to the health or safety of the licensee's employees or to the public, to the environment or to the common defense and security, and (2) punitive action is deemed necessary to assure future compliance; or

(*f*) Those cases involving activities under construction permits where there are repeated items of noncompliance with regulatory requirements; or

(*g*) Those cases where an item of noncompliance resulted in or contributed to the cause or to the seriousness of an accident or an incident; or

(*h*) Those cases involving items of noncompliance in the Violation category; or

(*i*) Those cases where the nature and number of items of noncompliance with the regulatory requirements identified during an inspection or an investigation demonstrate that management is not conducting its licensed activities with adequate concern for the health, safety or interest of its employees or the public or the common defense and security; or

(*j*) Those cases where licensees knowingly use materials which are not authorized by the license or utilize authorized materials for uses which are not authorized; or

(*k*) Those cases where significant matters[2] were not reported to the Commission in a timely manner as required by the regulatory requirements.

Civil penalties may be assessed for other cases having comparable types of items of noncompliance and situations for which the Commission deems civil penalties to be appropriate and necessary.

ORDERS—CRITERIA

The AEC has authority to issue orders to "cease and desist" or to suspend, modify, or revoke licenses. The Commission is empowered to enforce these orders and obtain any other appropriate relief by injunction from Federal district courts, if necessary. Cases involving an immediate threat to the public health and safety, or the common defense and security, require immediate steps to remove the threat and are handled by this type of action. Persons who deliberately violate, attempt to violate, or conspire to violate the Commission's regulations and orders, are, upon conviction of the violations, subject to fine up to $5,000 and imprisonment for not more than two years (Section 223 of the Act).

In the event the licensee fails to respond to a "notice of violation" or to demonstrate that satisfactory corrective action is being taken, an order to show cause may be issued requiring the licensee to show why the particular order (either of revocation, or modification, or suspension) should not be made effective. In those instances where the health, safety, or interest of employees or the public, or the common defense and security so requires, or deliberate noncompliance with the Commission's regulations is involved, the notice provision may be dispensed with and, in addition, the particular order may be made immediately effective pending further order.

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[1] NOTE: Section 221(b) of the Atomic Energy Act requires the FBI to investigate all suspected or alleged criminal violations of the Act.

[2] Such significant matters may include, but are not limited to, exposure of personnel to doses in excess of limits, release of radioactive concentrations in effluents in excess of limits, incidents involving an attempt to commit a theft or unlawful diversion of SNM, or to commit an act of sabotage of certain facilities, failure of safety systems, emergency core cooling or other related safety systems to perform their design function, or the MUF of SNM in excess of applicable limits, or similar matters.

(a) Orders to cease and desist

An order to cease and desist is ordinarily issued when a person is conducting unauthorized activities and has been notified of the need for authorization but fails to terminate the activity and other similar circumstances as appropriate.

(b) Orders to suspend a license

An order is ordinarily issued for immediate suspension of a license, or a portion thereof, as necessary to remove an immediate threat to the health, safety or interest of licensee's employees or the public, or to the common defense and security; or for noncompliance with AEC requirements relating to construction of a facility which, if not corrected immediately, could subsequently result in a significant threat to the health, safety or interest of employees or the public, or the common defense and security.

(c) Order to modify a license

An order for the modification of a license, in whole or in part, is ordinarily issued as an enforcement sanction when it is determined that a licensee's operations or activities must be limited or modified to protect the health, safety, or interest of the licensee's employees or the public, or the common defense and security.

(d) Orders to revoke a license

An order is ordinarily issued to revoke a license when:

1. The licensee's performance shows that he is not qualified to perform the activities covered by the license; or
2. Civil penalty proves to be ineffective as an enforcement action; or
3. The licensee refuses to correct items of noncompliance; or
4. A licensee does not respond to a "notice of violation"; or
5. A licensee's response to a "notice of violation" indicates inability or unwillingness to maintain compliance with regulatory requirements; or
6. Any material false statement is made in the application or in any statement of fact required under Section 182 of the Act.

(e) Denial of application for license renewal

Denial of an application for a license renewal is ordinarily used in lieu of an order for revocation where license renewal is pending or the expiration of the license term is imminent.

(f) Orders for other items of noncompliance

Orders to cease and desist, or for suspension, modification or revocation of a license are ordinarily issued for other comparable types of violations, infractions or deficiencies when the Commission deems such sanctions to be appropriate and necessary.

In all cases where orders are issued to impose civil penalties, to require a licensee to "cease and desist," or to suspend, modify, or revoke a license, the person so ordered may demand a hearing under 10 CFR Part 2. The hearing will be granted prior to implementation of the order except in cases where the Commission finds that the violation is deliberate or the public health, safety, or interest requires that the proposed action be temporarily effective pending the outcome of the hearing and/or further order.

REGULATORY OPERATIONS BULLETINS—CRITERIA

A Regulatory Operations Bulletin may be issued to a class of licensees requesting specific actions as a result of safety related equipment design inadequacies, defects, operating inadequacies, malfunctions, or failures of a generic nature that have occurred at a similar facility or operation. The Bulletin will specify that licensees inspect for and/or correct the inadequacies described in the Bulletin, notify Regulatory Operations of the corrective action taken or planned, and the date when action was or will be completed. An order may be issued if the response to a Bulletin is not prompt and effective.

IMMEDIATE ACTION LETTERS—CRITERIA

A Regulatory Operations Immediate Action Letter is ordinarily issued to solicit or confirm a licensee's commitment to certain actions for investigating, reporting, controlling, and correcting situations involvng defects, deviations, failures, or administrative controls, at the licensee's facility. An order may be issued if the response to an Immediate Action Letter is not prompt and effective.

Categories of Items of Noncompliance

The Commission and representatives of the nuclear industry have recognized that the significance of items of noncompliance with AEC requirements varies in the potential for affecting the health and safety of the public, the common defense and security, and the environment. The Commission considers that it is desirable to include in Notices of Violation an indication of the significance of each item of noncompliance cited. As a means of categorizing the items of noncompliance into an order of importance which will express their relative significance, the Commission has established three categories of items of noncompliance as follows:

Violation

A violation is an item of noncompliance of the type listed below, or an item of noncompliance (1) which has caused, contributed to or aggravated an incident of the type listed below, or (2) which has a substantial potential for causing, contributing to or aggravating such an incident or occurrence; e.g., a situation where the preventive capability or controls were removed or otherwise not employed and created a substantial potential for an incident or occurrence with actual or potential consequences of the type listed below:

(*a*) Exposure of an individual in excess of the radiation dose specified in 10 CFR 20.403(b) or exposure of a group of individuals resulting in each individual receiving a radiation dose which exceeds the limits of 10 CFR 20.101 and a total dose for the group exceeding 25 man-rems.

(*b*) Radiation levels in unrestricted areas which exceed 50 times the regulatory limits.

(*c*) Release of radioactive materials in amounts which exceed specified limits, or concentrations of radioactive materials in effluents which exceed 50 times the regulatory limits.

(*d*) Fabrication, or construction, testing, or operation of a Seismic Category I system or structure in such a manner that the safety function or integrity is lost.

(*e*) Failure to function when required to perform the safety function or loss of integrity of a Seismic Category I system, or structure; or other component, system, or structure with a safety or consequences limiting function.

(*f*) Exceeding a safety limit as defined in technical specifications associated with facility licenses.

(*g*) Industrial sabotage of utilization or fuel facilities.

(*h*) Radiation or contamination levels in excess of limits on packages or loss of confinement of radioactive materials in packages offered for shipment on a common carrier.

(*i*) Diversion or theft of plutonium, uranium 233, or uranimum enriched in the isotope U-235.

(*j*) A breakdown in management or procedural controls as evidenced by items of noncompliance in several areas of the QA criteria and license requirements.

(*k*) Other similar items of noncompliance having actual or potential consequences of the same magnitude.

Failure to report the above items as required constitutes a violation of the same importance level.

Infractions

An infraction is an item of noncompliance of the type listed below, or an item of noncompliance (1) which resulted in a reduction of preventive capability below requirements but redundant controls precluded an item of noncompliance of the violation category, or (2) which caused, contributed to or aggravated an incident of the type listed below, or (3) which has a substantial potential for causing, contributing to or aggravating such an incident or occurrence; e.g., the preventive capability or controls were removed or otherwise not employed and there was substantial potential for an accident or occurrence with actual or potential consequences of the type listed below:

(*a*) Exposure of an individual or groups of individuals to radiation in excess of permissible limits but less than the values in 10 CFR 20.403.

(*b*) Release of radioactive materials in concentrations or rates which exceed permissible limits but in amounts less than permissible limits.

(c) Failure to function or loss of integrity of a Seismic Category I system or structure, or other component, system or structure with safety or consequences limiting function during test; or failure to meet surveillance frequencies.

(d) Fabrication, or construction, testing, or operation of a Seismic Category I system or structure in such a manner that the safety function or integrity is impaired.

(e) Exceeding limiting conditions for operation (LCO).

(f) Inadequate management or procedural controls.

(g) Safety system settings less conservative than limiting safety system settings.

(h) A quantity of SNM unaccounted for which exceeds permissible limits.

(i) Exceeding limits or limiting conditions for operation in licenses, technical specifications, guides, codes, or standards which are imposed for the purpose of minimizing adverse environmental impact.

(j) Other similar items of noncompliance having actual or potential consequences of the same magnitude.

Failure to report the above items as required constitutes an item of noncompliance of the same category.

Deficiency

A deficiency is an item of noncompliance in which the threat to the health, safety, or interest of the public or the common defense and security is remote; and no undue expenditure of time or resources to implement corrective action is required; and deficiencies include such items as noncompliance with records, posting, or labeling requirements which are not serious enough to amount to infractions.

Failure to report deficiencies as required constitutes an item of noncompliance of the same category.

4. Apparently, the installation of the cable tray penetration sealant and fire retardant coating had not been completed prior to operations of Units 1 and 2 (Investigation Report—I-2). How did this manage to slip past detection by both TVA and NRC?

Response: Inspections conducted by NRC personnel using the inspection manual in effect prior to the Browns Ferry fire did not include checking of the cable penetrations or fire stops. The inspection manual has since been revised to require such inspections. Further, it is understood that TVA did not include such inspections within its quality assurance program.

5. The procedures for notifying States and local support agencies left something to be desired. How closely had NRC reviewed these procedures?

Response: The TVA Radiological Emergency Plan (REP) states that the responsibility for the coordination of a nuclear emergency with the appropriate public agencies is vested in the Central Emergency Control Center (CECC) Director.[1] During a nuclear emergency at a plant site the CECC staff will function to provide assistance as necessary to the site and division emergency organizations and will provide all information requested by outside agencies. The chain of notification is shown in Figure 1.[2]

When a nuclear emergency is detected at a site, the Site Emergency Director notifies the CECC Director who then notifies the appropriate federal, state, and local authorities.[3] The detailed procedures for this notification are found in Section III of the TVA-REP. These procedures list several conditions of gas release, liquid release, and abnormal radiation as action levels at which protective measures would be initiated.[4] As no action levels associated with these conditions were reached during the fire,[5,6] the CECC was not initially activated. Some increase in airborne activities did occur[7] as a result of the shutdown of the ventilation system but this situation did not constitute a nuclear emergency, as defined in the REP (radiological hazard to onsite personnel or radiation levels in excess of 10 CFR Part 20 values to the public). The REP was activated however, primarily to expedite communications and the decision making process.

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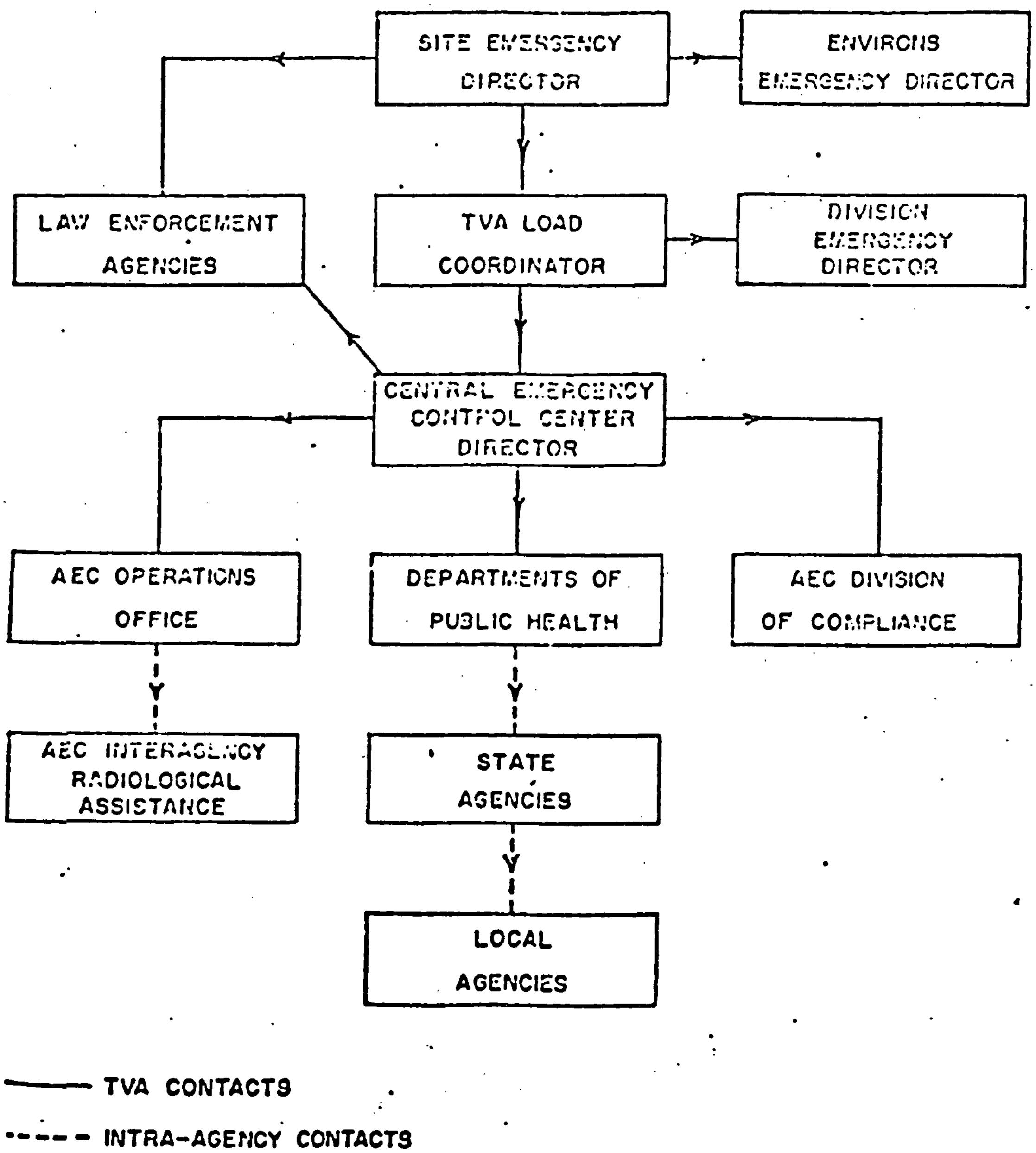
[1] TVA Radiological Emergency Plan for the Browns Ferry Nuclear Plant, revised June 5, 1972, page 37.
[2] Ibid, page 19.
[3] Ibid, page 7.
[4] Ibid, page 11.
[5] IE Report No: 50-259/75-1D (preliminary), page 4.
[6] IE Report No: 50-259/75-1B (preliminary), page 3.
[7] TVA Plan for Evaluation, Repair & Return to Service of Browns Ferry Unit 1 and 2, Section III, page 43.

The NRC staff reviewed these procedures. It was found that:

1. Authorities and responsibilities of key individuals had been delineated;

Figure I

CHAIN OF NOTIFICATION



2. Communication links in the form of call lists had been established for notifying emergency personnel;
3. Action levels had been chosen for notification of offsite agencies in radiological emergency situations; and,
4. Written agreements with offsite emergency organizations had been provided.

We found that the emergency planning program for the Browns Ferry Nuclear Plant met the requirements of 10 CFR 50, Appendix E, and was acceptable. This finding was presented in the Safety Evaluation Report of June 26, 1972. In particular, the TVA-REP met the requirements of Part IV D of Appendix E, in that procedures for the notification of local, State, and Federal agencies were included in the plan and were found satisfactory.

Office of Inspection and Enforcement inspectors do not review procedures prepared by State and local support agencies. They do, however, verify statements made by the licensee in his emergency plan concerning his notification arrangements with State and local authorities by contacting the agencies identified. In the case of Browns Ferry, such contacts by inspectors were made on September 30, October 1–4, October 20–21, and November 1, 1974. These visits are reported in publicly available Inspection Reports.

The fact that the specified action levels contained in the TVA-REP were not reached may have been a factor in the manner in which the notification procedures were implemented. Apparent deficiencies in their implementation were reported to the licensee in the IE Inspection Report 50-259/75-1 dated July 25, 1975. The TVA responded to the alleged infractions and areas of concern by letter dated September 2, 1975.[8] These responses were discussed in IE letter dated September 29, 1975.[9]

6. How does NRC reconcile the need for water deluge systems in cable rooms (page 7 of Dr. Hanauer's testimony) with TVA's contention that the use of water might have led to further loss of control of the reactors?

Response: There is considerable room for judgment and professional disagreement on this issue. Mr. Green of TVA pointed out at the hearing the potential for water on the cables disabling electrical circuits that might otherwise operate successfully in spite of the fire damage. He stated his preference for waiting "until the situation had stabilized" and until equipment available for service had been evaluated before pouring water on these cables.

Past experience in other situations has indicated that it is generally preferable on a risk-benefit basis, to extinguish the fire as quickly as possible with whatever agents are needed. In this way the spread of damage can be more quickly and more surely limited. Delay in the use of water would necessarily incur the risk of additional equipment becoming unavilable due to further spread of the fire.

This is not to say that the indiscriminate use of water is required under all circumstances for all fires. Or that it is totally clear that the early use of water will result in less risks to the public for all fires involving nuclear power plants. Such decisions during an emergency must be left up to qualified personnel on the scene since operating manuals and procedures cannot detail all possible situations and events. However, water has often been needed to put out most electrical cable fires of more than trivial size. This view is also held by the Fire Underwriters and other fire experts we have consulted. They have informed us that current practice is to provide water deluge systems even for such complex systems as large computer installations. They state that this recommendation is based on a balance of the risks and benefits in fighting fires in electrical equipment and has been shown by experience to be preferable.

The course of the Browns Ferry fire (as shown in Figure 4 of Mr. Gilleland's testimony) shows a progressive onset of unavailability of various equipment. We believe that the spread of the fire made some contribution to this. For example, some of the redundant circuits disabled unexpectedly were in metal conduits that were overheated by the fire when it progressed to that location. It therefore appears to us that, in general, fires should be fought promptly and as effectively as possible including the use of water when necessary. Also, in hindsight, it is known that the electric shock considerations were not significant at Browns Ferry. The special hose nozzles available, when used, put out a divided spray that does not endanger the fireman holding the hose.

It is noteworthy that TVA now proposes to install water deluge systems at the Browns Ferry plants in regions of high concentration of electrical cables.

7. The Committee understands that a petition has been filed to change the NRC regulations to require plant operators to distribute emergency plans to the public and conduct yearly drills. What is the status of NRC's consideration of this matter?

Response: By letter dated August 6, 1975, the Public Interest Research Group and thirty other specified citizen groups petitioned the NRC to require licensees and applicants to instruct citizens in public evacuation procedures in case of major nuclear incidents and to test public evacuation plans in realistic drills. The petition has the docket number PRM-50-14 and was published in the Federal

[8] Letter TVA to NRC Region II of 9/2/75, pages 14, 25, 38, 39, and 40.
[9] Letter NRC Region II to TVA of 9/29/75.

Register on September 23, 1975, with a 60-day comment period. The NRC will consider public comments in its subsequent response to the petition.

8. Over the years, the primary emphasis in nuclear safety has been placed on hypothetical nuclear accidents, such as loss of coolant accidents and reactor excursions. The fire at Browns Ferry and other fires over the years would appear to suggest that the accidents that have occurred at nuclear plants have been more conventional in nature. In this regard,

(a) What consideration does NRC give to fires and other conventional accidents in its safety reviews?

(b) How is NRC organized to assure that such hazards are recognized: i.e., does NRC have any special expertise in fire detection and prevention?

(c) Does NRC intend to make any organizational changes to put increased emphasis on preventing fires and other conventional hazards?

Response (a)—pertaining to fires: The NRC review and evaluation of the fire protection of a proposed nuclear power plant includes the fire potential as described in the applicant's Safety Analysis Report (SAR), and a review of the design layout of the fire protection system (FPS) showing the characteristics and locations which define the "fire prevention" and "fire protection" portions of the system. The review also includes the description, identification and types of fire hazards that can exist and fire risk evaluations for each of the postulated hazards. Examples of subjects reviewed include:

1. the selection of fire fighting methods, manual or automatic equipment, and safety devices, including the detection, suppression, control, and extinguishing systems as described in the SAR;

2. the selection of appraisal and trend evaluation systems to be used, and the design of the fire detection and alarm system; and,

3. the general plans for performing inspection checks and the frequency of testing to maintain a reliable detection and alarm system.

The staff reviews the building and facility arrangements and structural design features which control selection of the methods for fire prevention, control and extinguishment, and control of fire hazards. Fire barriers, egress routes, fire walls, and the isolation and containment features are included in the review.

The staff determines from the SAR it appreciable amounts of combustibles are to be located on site, and reviews analyses of the effects of these hazards on safety-related equipment located nearby. The staff varifies that these analyses include a conservative selection of design basis fires, as determined from the types and quantities of stored combustible materials.

The functional performance of extinguishing systems, as described in the SAR, is reviewed to verify the adequacy of the FPS to protect electrical equipment. On multiple unit applications, the additional fire protection and control provisions during construction of the remaining units are reviewed to verify that the integrity of the fire protection system for the safe operation and shutdown of the facility is maintained.

Information that NRC has requested to be incorporated in the applicant's SAR is set forth in Regulatory Guide 1.70.4, "Additional Information Fire Protection Considerations for Nuclear Power Plants", February 1974.

Response (a)—other conventional accidents: The NRC, the vendors, and plant operators have always recognized that nuclear plants are subject to the same types of accidents that can affect any industrial facility. However, nuclear plants are different in that these conventional industrial accidents, if uncontrolled, could in some cases affect the cooling of the core and potentially result in the release of radioactive material.

The staff includes in its safety review an evaluation of these conventional types of accidents. A list of these accidents and the Standard Review Plans which detail the review of each, is attached. These other events include natural phenomena such as tornadoes, wind storms, floods, tsunamis, seiches, and earthquakes. The effect of accidents at nearby locations such as waterways, railways and roads that might produce explosions, missiles, smoke, or toxic or chemical containing gases are considered. At locations where the potential is significant, the consequences of aircraft crashes are considered. Accidents within the plant that might result in these effects are also considered. Radioactive spills are considered and, although not conventional, the causes of such events are usually conventional.

Additionally, a large portion of the safety review consists of evaluating the response of the plant to those transients to which every power plant, conventional or nuclear, is potentially subjected. Some of these transients are well within the normal operating conditions of the plant. However, more attention is directed toward abnormal anticipated transients, since their effects are more severe. Abnormal operational transients are those events which can be reasonably expected to occur during the life of the plant. These events include turbine trips, losses of feedwater, inadvertent actuation of equipment such as the ECCS, and other events which would result from operator errors or equipment malfunctions.

Response (b): Within the NRC staff, especially in the Offices of Nuclear Reactor Regulation and Inspection and Enforcement, personnel are assigned the responsibility for reviewing reactor applications and for inspecting reactor facilities to assure that various potential hazards are recognized, reviewed and evaluated. The purpose of these reviews and inspections is to determine whether plant design criteria and bases are adequate to assure safe reactor shutdown in the event of the occurrence of any of the postulated hazards. Some of these "conventional hazards" considered are fires, both inside and outside the plant, natural phenomena, aircraft crash and its fire effects (where applicable), and/or water surface transportation accidents that could involve fire or explosions and hazards that may be related to conditions surrounding the reactor site.

The NRC does have expertise in the areas of fire detection and prevention. However, if an unusual situation is noted by the staff, consultants are available and used from other federal agencies to assist in the resolution of problems. Agencies that have been or are being used for this service include ERDA, NASA and the Coast Guard.

Response (c): No organizational changes are planned to place increased emphasis on preventing fires and other conventional hazards. However, increased attention is being given to fire hazards in already established organizational components that review fire prevention and protection and other conventional hazards. The Office of Inspection and Enforcement has developed guidance for inspectors in the area of fire protection equipment and procedures. NRC has under consideration the degree of recognition which can be given to inspections performed by the insurance groups as third party inspections.

Task forces to review and evaluate the Browns Ferry fire have been established and have been involved since the fire occurred. The work being done by these groups is being fed back into the existing NRC organization. The staff involved in evaluating the Browns Ferry fire has been augmented by consultants. These consultants are assisting in the evaluation of the fire, in the development of criteria and guides for use in evaluation of fire protection programs for nuclear power plants, and are conducting tests on various aspects of fire conditions that require more detailed knowledge, and providing general advice on fire protection.

Listing of Postulated Accidents

Accident and standard review plan	*Branch with primary responsibility*
Accidents at nearby locations (smoke, chemicals, gases): 2.2.3—Evaluation of potential accidents.	Accident Analysis.
Storms:	
2.3—Meteorology	Site analysis.
3.3—Wind and tornado loadings	Structural engineering.
Earthquakes:	
2.5—Geology and seismology	Site analysis.
3.2—Classification of structures, components and systems.	Reactor systems.
3.7—Seismic design of structures	Structural engineering.
3.8—Design of category I structures	Do.
3.10—Seismic design of category I instrumentation and electrical equipment.	Electrical, instrumentation and control systems.
Floods:	
2.4—Hydraulic engineering	Site analysis.
3.4—Water level (flood) design	Auxiliary and lower conversion systems.

Listing of Postulated Accidents—Continued

Accident and standard review plan	*Branch with primary responsibility*
Missiles:	
3.5.1.1—Internally generated missiles (outside containment).	Auxiliary and lower conversion systems.
3.5.3—Barrier design procedures	Structural engineering.
3.5.1.2—Internally generated missiles (inside containment).	Reactor systems.
3.5.1.4—Missiles generated by natural phenomena.	Accident analysis.
3.5.1.5—Site proximity missiles (except aircraft).	Do.
3.5.1.6—Aircraft hazards	Do.
3.5.2—Structures, systems and components to be protected from externally generated missiles.	Auxiliary and power conversion systems.
Anticipated transients:	
15.1.1–15.1A—Decrease in feedwater temperature, increase in feedwater flow, increase in steam flow, and inadvertent opening of a steam generator relief or safety value.	Reactor systems.
15.1.5—Spectrum of steam system piping failures inside and outside of containment (PWR).	Do.
15.2.1–15.2.5—Steam pressure regulator failure (closed), loss of external load or turbine trip, closure of main steam isolation valve (BWR) and loss of condenser vacuum.	Do.
15.2.6—Loss of nonemergency A-C power to the station auxiliaries.	Do.
15.2.7—Loss of normal feedwater flow.	Do.
15.2.8—Feedwater system pipe breaks inside and outside containment (PWR).	Do.
15.3.1–15.3.2—Loss of forced reactor coolant flow including trip of pump and flow controller malfunctions.	Do.
15.3.3–15.3.4—Reactor coolant pump seizure and reactor coolant pump shaft break.	Do.
15.4.4–15.4.5—Startup of inactive loop or recirculation loop at an incorrect temperature, and flow controller malfunction causing an increase in BWR core flow rate.	Do.
15.4.6—CVCS malfunction that results in a decrease in boron concentration in the reactor coolant (PWR).	Do.
15.6.1—Inadvertent opening of a PWR pressurizer safety/relief valve or a BWR safety/relief valve.	Do.
15.7.3—Postulated radioactive releases due to liquid-containing tank failures.	Effluent treatment systems.
15.5.1–15.5.2—Inadvertent operation of ECCS and CNCS malfunction that increases reactor coolant inventory.	Reactor systems.
15.6.2—Failure of small lines carrying primary coolant outside containment.	Accident analysis.

Listing of Postulated Accidents—Continued

Accident and standard review plan	*Branch with primary responsibility*
Anticipated transients—Continued	
15.6.3—Radiological consequences of steam generator tube failure (PWR).	Do.
15.7.1—Waste gas system failure____	Do.
15.7.5—Spent fuel cask drop accidents.	Do.
15.7.2—Radioactive liquid waste system leak or failure.	Do.
15.4.1—Uncontrolled control rod withdrawal from a subcritical or low power startup condition.	Reactor systems.
15.4.2—Uncontrolled control rod assembly withdrawal at power.	Do.
15.4.3—Control rod misoperation (system malfunction or operator error).	Core performance.
15.4.7—Inadvertent load and operation of a fuel assembly in an improper position.	Do.
15.4.8—Spectrum of rod ejection accidents (PWR).	Do.
15.4.9—Spectrum of rod drop accidents (BWR).	Do.
15.6.4—Radiological consequences of main steam line failure outside containment (BWR).	Accident analysis.
15.6.5—Loss-of-coolant accidents resulting from spectrum of postulated piping breaks within the reactor coolant pressure boundary.	Reactor systems.
15.8—Anticipated transients without scram.	Do.

9. Mr. David Comey has made two specific recommendations to NRC as a result of the fire:

(*a*) That operation of nuclear units not be permitted at stations where additional plants are under construction.

(*b*) That all reactors not fully complying with IEEE standards on separation of electrical cables be shut down.

What is the status of NRC's actions on these recommendations?*

Response (a): Mr. David Comey's recommendation "that licensees be required to shut down operating units of nuclear power units at multi-unit facilities during periods when work on a unit under construction could compromise the integrity of the engineered safety features of an operating unit or units", has been treated as a Petition for Rulemaking since it involves a potential change to NRC Regulations. The Petition was published in the Federal Register on May 9, 1975, with comments requested from the public by July 8, 1975. Three comment letters were received. We expect to respond to this petition, which will include consideration of these comments, by February 1, 1976.

Response (b): On March 24, 1975, Mr. David Comey, citing the March 22, 1975 occurrences at Browns Ferry, asked the NRC to identify those operating plants that do not fully comply with IEEE–279 and to show cause why those plants should not be shut down until they are retrofitted to meet that standard.

On May 5, 1975, NRC provided Mr. Comey with the identity of these eight plants which, since they were issued operating licenses prior to the issuance of IEEE–279, were evaluated against pre-IEEE–279 criteria. In that letter, he was informed of steps that had been and would be taken that in our judgment provide reasonable assurance that all operating plants can continue to operate without undue risk to the health and safety of the public pending completion of the ongoing NRC investigation of the Browns Ferry fire.

In response to this information provided by the NRC, on June 16, 1975, Mr. Comey limited his March 24, 1975 request to a review by the NRC staff of two

*See Appendix 13 for exchange of correspondence between NRC and David Comey.

plants, Indian Point 1 and Dresden 1, pending completion of the report by the NRC Special Review Group on the investigation of the fire.

On August 4, 1975, the NRC staff informed Mr. Comey that on June 11, 1973 the Consolidated Edison Company had been directed pursuant to 10 CFR 50.109 to make certain design changes on Indian Point 1 based on a comparison of the as-built plant protection system with the criteria of IEEE-279. Indian Point 1 has been shut down since October 31, 1974.

Our August 4, 1974 letter also stated that on July 31, 1975 Commonwealth Edison had been requested to compare the as-built design of that plant with the IEEE-279 criteria (1968 Edition); to identify features that do not conform to that standard; and to determine whether, based on its evaluation, there are changes in the facility which could be accomplished that provide substantial additional protection which is required for public health and safety. Commonwealth Edison provided its report on the matter on October 1, 1975. The staff is now evaluating the report to determine what actions, if any, are necessary on that plant to protect the health and safety of the public.

10. What is NRC's position regarding the continuing applicability of Technical Specifications in the event some departure from them is required as the result of an on-site emergency?

Response: Technical Specifications for each facility are required by Commission Regulations (Paragraph 50.36). These specifications set forth limitations and restrictions needed to maintain a facility in a safe condition during plant operation. They further set forth requirements on equipment operability for the purpose of assuring that the requisite number of components are operable in the event of postulated accidents.

Technical Specifications are not expected to cover all conceivable conditions that might arise during an on-site emergency. Should an emergency condition arise which, in the interest of public health and safety, required action inconsistent with a Technical Specification, we would not construe the Specification to be applicable.

11. Please provide background on why the polyurethane material of the type tested by the applicant was not the type used. Also, discuss why these tests were relied upon, when it has been documented that small scale laboratory tests are inadequate to determine the fire properties of a material in large-scale applications. (AEC Serious Accidents Bulletin No. 336, September 19, 1974). Finally, why was polyurethane foam approved for use in such an inaccessible location, when the fire hazards of this practice have long been known? (AEC Health and Safety Bulletin No. 170, dated September 10, 1963 recommends that consideration be given to attaching polyurethane so that it can be removed more easily in case of fire.)

Response: The pieces of sheet polyurethane used to seal the leaks and provide a dam prior to placement of the polyurethane foam represented material not tested by TVA. Approval to use the untested sheet polyurethane in the penetration sealing activity was given by one TVA employee (NRC Investigation Report, p I-10). This individual was aware that the sheet polyurethane had not been tested (NRC Investigation Report, Statement EE-FF). He also was aware of the TVA flammability tests of the sealant material (another form of polyurethane) covered with the flame-resistant material.

In reply to the NRC Notice of Violation concerning the alleged failure of TVA to have detailed written procedures approved by the plant superintendent and reviewed by the TVA Plant Operation Review committee for this penetration sealing activity, TVA replied ". . . it was not recognized that this work on the penetration was of such a nature as to have an effect on the safety of the reactor, on surveillance or testing requirements, or had the potential for release or radioactivity; therefore, leak testing sealing, and inspection of the penetrations were not accomplished by detailed, approved, written procedures as would be required for safety-related items." (p 3, of Enclosed Notice of Violation, to TVA letter of September 2, 1975, from J. E. Gilleland, TVA, to Norman C. Moseley, NRC).

TVA conducted tests of cable penetration fire stops in May 1973 for the purpose of examining the effectiveness of urethane pressure seals. A copy of the TVA evaluation memo concerning these tests was provided as Exhibit A4 in the I&E report of July 25, 1975. The staff does not consider these tests to be representative of the actual installation in that a differential pressure was not utilized to simulate actual in-plant conditions. TVA is now conducting additional tests to more closely simulate the as-built penetrations; application of pressure will be included. The purpose of these tests is to determine whether properly constructed fire stops of the original design are effective and whether those fire stops not required to be replaced due to restoration work can be left in place.

The inaccessibility of the location was, indeed, a handicap in performing any work or testing on the seals at this location. This inaccessibility contributed to the ignition of the fire and to the difficulty in extinguishing it immediately (as had been done for other similar small fires). However, it should be noted that a properly constructed seal was shown to have been an effective fire stop and it should have been possible, and will be accomplished in the future, to devise procedures for working on and testing these seals even in relatively inaccessible locations without any significant fire hazard.

APPENDIX 10

QUESTIONS AND ANSWERS: JCAE/TVA

LETTER DATED SEPTEMBER 17, 1975, MURPHY TO WAGNER, WITH ADDITIONAL QUESTIONS TO TVA

JOINT COMMITTEE ON ATOMIC ENERGY,
U.S. CONGRESS,
Washington, D.C., September 17, 1975.

MR. AUBREY J. WAGNER,
Chairman, Board of Directors,
Tennessee Valley Authority, Knoxville, Tenn.

DEAR MR. WAGNER: Enclosed is the transcript of the September 16, 1975 hearing on the Browns Ferry fire. The TVA witnesses should review and make any necessary corrections of your testimony in accordance with the instructions attached to the transcript. Also enclosed is a list of additional questions on the TVA testimony which the Committee would like you to provide answers to for the record.

Since we are endeavoring to publish and distribute the hearing as rapidly as possible, it would be appreciated if you would return the corrected transcript and your answers to the additional questions by no later than September 29, 1975.

Thank you for your cooperation in this matter.

Sincerely yours,

GEORGE F. MURPHY, JR.,
Executive Director.

QUESTIONS FOR TVA WITNESSES

Browns Ferry Hearing—September 16, 1975

1. How close, in TVA's view, was the plant to having a serious accident—such as fuel damage or core meltdown—which might have presented a serious public health and safety problem? Describe briefly and factually the equipment remaining in operation or which could be made operable within the necessary time frame to prevent core damage. Summarize both the regular and "emergency" function of this equipment. Show whether the availability of and need for this equipment was perceived during the time of the accident, and state the degree of confidence in whether this equipment would actually have been used, if needed.

2. There had been a past history of small fires at the Browns Ferry plant. Why hadn't preventive actions been taken by TVA? (Inspection Report, page I-11)

3. Apparently, there was considerable difficulty with finding sufficient operable air breathing equipment (Inspection Report I-13). Why hadn't TVA fire inspectors detected this situation in their normal periodic inspections of fire equipment?

4. Does TVA have an internal safety review committee to assess plant operating practices? At what level of management was the open flame inspection procedure approved?

5. How often have NRC inspection and enforcement personnel visited the plant and when was the last such visit? What infractions were reported at that time, and were any of these still uncorrected at the time of the fire? If so, did any of these contribute to the accident or its aftermath in any way?

6. Why had the material around the penetration in question not been coated with flamemastic? Why had this penetration not been adequately filled with foam in the first place? Was operation of the plant under these conditions in violation of NRC regulations?

7. Do you consider that TVA was adequately prepared to fight this fire? Are you making any changes to fire control plans and procedures as a result of this fire? Are there any actions that should have been taken by NRC or the State that might have enabled TVA to better respond to the situation?

8. What improvements does TVA believe desirable to make in the procedures used to notify the State of Alabama health officials and other local and county civil defense and police officials in the event of incidents like the Browns Ferry fire?

9. According to the NRC report (II-7), there was considerable concern on the part of the operators that the water level might drop below the top of active fuel. Is that an accurate statement, and what would the consequences have been if the fuel had indeed been uncovered?

10. You note on page 4 that both Units 1 and 2 were shut down shortly after anomalous events began to occur. Could this fire, or any fire, cause an inability to scram? What is the basis for your response?

11. What has TVA's experience been with respect to fires in its fossil fired units? How does this compare with that of its nuclear units?

[Subsequent TVA response follows:]

TENNESSEE VALLEY AUTHORITY
KNOXVILLE, TENNESSEE 37902

September 24, 1975

The Honorable George F. Murphy, Jr.
Executive Director
Joint Committee on Atomic Energy
Washington, D.C. 20510

Dear Mr. Murphy:

In accordance with your request of September 17, 1975, we have reviewed the transcript of the September 16, 1975, Joint Committee hearing on the Browns Ferry fire and prepared answers to your additional questions. Our comments and responses are enclosed.

Very truly yours,

Aubrey J. Wagner
Chairman of the Board

Enclosure

Enclosure

Question 1

How close, in TVA's view, was the plant to having a serious accident--such as fuel damage or core meltdown--which might have presented a serious public health and safety problem? Describe briefly and factually the equipment remaining in operation or which could be made operable within the necessary time frame to prevent core damage. Summarize both the regular and "emergency" function of this equipment. Show whether the availability of and need for this equipment was perceived during the time of the accident, and state the degree of confidence in whether this equipment would actually have been used, if needed.

Answer

Because there was always more than one means available to supply water to the reactor in quantities sufficient to keep the core covered, the likelihood of fuel damage due to the fire event was, for all practical purposes, nonexistent. With the ability to keep the reactor fuel covered with water and with primary and secondary containment intact, there was no serious threat to public health and safety. The following is a summary of equipment remaining in operation or which could be made available during the fire event. These evaluations are restricted to unit 1 since a review of the events makes it clear that the adverse effects of the fire on unit 2 were far less than on unit 1 and at all stages of the incident unit 2 had more capability for maintaining a safe shutdown condition. In addition, due to a longer operating history, the decay heat removal requirements were greater for unit 1. The capability of many of the following components depends on the reactor system pressure. For reference pressures see figure 5 of the statement of Jack E. Gilleland.

Steam turbine-driven feedwater pumps - 3 pumps - capacity 10,800 gpm each

The function of the feedwater pumps is to supply water to the reactor during power operation and during a period following reactor shutdown while steam pressure above 350 pounds still exists. The feedwater pumps have no emergency function. These pumps were available at the time the fire occurred and remained

Question 1 (continued)

available and in use until approximately five minutes after reactor shutdown when they became unavailable due to closure of the main steam isolation valves which cut off their steam supply. The status of the feedwater pumps was known to the operator at all times. TVA systems analyses undertaken subsequent to the fire event have revealed that there were alternate means available by which the main steam isolation valves could have been opened, thus making the steam-driven feedwater pumps available. This was not considered by the plant operator during the fire event.

Condensate Booster Pumps - 3 pumps - capacity 10,800 gpm each

Three condensate booster pumps have the normal function of supplying water to the steam-driven feedwater pumps and supplying water directly to the reactor at pressures less than 435 pounds per square inch. These pumps have no assigned emergency function, but were available throughout the fire event and one of these pumps was in continuous operation and supplied water to the reactor after its pressure was reduced.

Condensate Hot-Well Pumps - 3 pumps - capacity 10,800 gpm each

The normal purpose of these pumps is to supply water to the condensate booster pumps during normal operation. They have the capability of supplying water directly to the reactor at pressures less than 160 pounds per squrre inch. They have no assigned emergency function. These pumps were available and two were in continuous operation throughout the fire event. The status of all three pumps was known to the operator.

Question 1 (continued)

Control Rod Drive Pumps - one per unit plus one shared spare - capacity 225 gpm each

Normally, one control rod drive pump is aligned to each reactor and one spare is available which may be aligned to either reactor. The purpose of these pumps is to supply operating water under pressure to the control rod drive units as well as to supply cooling water for the control rod drives. In normal operation, each of these pumps supplies 100 gpm of water in the reactor while performing its assigned function. It has no emergency function. During the fire event, one control rod drive pump was aligned to each reactor and ran continuously throughout the event. Using controls located in the control room, the operator increased the unit 1 pump flow output to approximately 130 gpm. Had the operator opened an accessible manual valve in the reactor building, the flow could have been increased to 225 gpm. The availability of the common spare pump was known but the pump was not operated. In addition, the unit 2 pump could have been aligned to unit 1, but the operator did not perceive this mode of operating at that time. However, as indicated in our testimony, the unit 1 pump alone would have been adequate.

Standby Liquid Control Pumps - 2 pumps - capacity 56 gpm each

Standby liquid control pumps had the assigned emergency function of injecting a boron solution into a reactor at/up to full pressure should the normal shutdown method fail. Using a test mode, these pumps could have been used to supply in excess of 100 gpm of demineralized water on each reactor. These pumps were not used but the operator was aware of this capability.

Question 1 (continued)

Reactor Core Isolation Coolant - one pump - capacity 600 gpm

The RCIC system is not considered a part of the nuclear safety systems but provides high-pressure makeup water to the reactor vessel if the feedwater system does not operate properly or if the reactor becomes isolated and the coolant inventory is decreasing. The pump is steam driven, normally using steam from the reactor vessel but in a test mode may be operated using steam from an auxiliary boiler. This requires installation of a short piece of pipe available for that purpose which requires approximately one hour to install. The RCIC was initially available and was shut down by the operator since it was not required. Approximately five minutes after reactor shutdown, the RCIC system became unavailable in its normal configuration due to closure of a steam supply valve. It appears that this system could have been made available within an hour had the operator made the decision to install the spool piece to establish a steam flow path from the auxiliary boiler. During the fire event, the operator did place the auxiliary boiler into operation and bring it to full steam pressure.

High Pressure Coolant Injection - one pump - 5,000 gpm capacity

The high pressure coolant injection system operates using reactor steam as a driving force. It has no normal function but has the emergency function of supplying high pressure water to the reactor during a loss of coolant accident. The HPCI was initially available to the operator but was shut down since it was not required. The HPCI became unavailable approximately five minutes after reactor shutdown and was unavailable throughout the remainder of the fire event.

Question 1 (continued)

Residual Heat Removal - 4 pumps - 10,000 gpm capacity each

The RHR is a multipurpose system. In the low pressure coolant injection mode, it is intended to supply water to the reactor at pressures less than 300 pounds during the loss of coolant event. In the torus cooling mode, it has the normal and emergency purpose of circulating suppression chamber water through heat exchangers as a means of controlling suppression chamber temperatures. It has the capability of spraying water into the drywell during a loss of coolant accident thereby limiting containment pressure and temperature. In the shutdown cooling mode, with reactor vessel pressure below 100 psig, it circulates reactor vessel water through a heat exchanger and back to the pressure vessel. The RHR system was initially available in all modes. During the fire, damage to valve and pump wiring disabled two of the RHR pumps. Use of the other two pumps was temporarily lost until power was restored to the valves which were required to line up the system for operation. At approximately 4:30 p.m., two RHR pumps were aligned in the torus cooling mode but were not used because of the possibility that during the valve alignment without position indication the pump discharge piping may have been drained and to start a pump on an empty pipe had the potential of system damage. The status of the system was known to the operator but conditions never were such that use of the available components under these circumstances was justified. The RHR system was placed into service in both the torus cooling and shutdown cooling modes later in the night after a procedure was prepared to manually align and check the systems under existing conditions.

Question 1 (continued)

Standby Coolant Supply - 2 pumps - 4,500 gpm capacity each

Browns Ferry has the capability to supply river water directly to the reactor, at pressures less than 185 pounds using RHR service water pumps. These pumps ran continuously throughout the fire event and this capability was known to the operator. Circumstances did not justify use of this extreme measure since use of river water instead of clean dimeralized water might jeopardize future operability of the reactor. The piping arrangement of this system is such that had the RHRSW pumps become inoperable, any other pumping means such as a fire truck could have been used to pump any available water, including river water, into the reactor through this system. Although this capability was recognized, its use was not seriously considered.

Core Spray System - 4 pumps - 3,100 gpm capacity each

The core spray system is provided to supply coolant to the reactor at pressures less than 300 pounds during the loss of coolant accident event. The core spray system initially aligned and was shut down by the operator because it was not needed. Two of the core spray pumps became unavailable due to the fire damage and the position of the valves in the core spray system was not known to the operator because of loss of position indication. After the fire, it was determined that had the remaining two core spray pumps been operated, water injection into the reactor would have occurred. Core spray pump operation was not an avenue pursued by the operator during the fire event, but the operator did give consideration to trying it in the event other means to supply water to the reactor were lost.

Question 1 (continued)

Condensate Transfer Pumps - 2 pumps - 1,000 gpm each

One of the normal functions of the two condensate transfer pumps is to supply charging water to the discharge piping of the core spray and RHR pumps. The system could have been used to put water into the reactor vessel at low pressure, less than 85 psi, by closing a head tank isolation valve and opening the inboard isolation valve on either core spray or RHR system. These pumps were not used, but the operator was aware of this capability.

Question 2

There had been a past history of small fires at the Browns Ferry Plant. Why hadn't preventive actions been taken by TVA? (Inspection Report, page I-11)

Answer

The plant operators were aware of one small fire which had occurred on Thursday, March 20, 1975. This fire had been discussed in a meeting of plant supervisors on Friday, March 21, and investigative efforts were in progress. Plant management was not aware of any other minor fires which may have occurred in this material because they were not reported. TVA has since made procedural changes controlling the use of open flames, fire reporting, and use of fire attendants when activities with a fire potential are in progress.

Question 3

Apparently, there was considerable difficulty with finding sufficient operable air breathing equipment (Inspection Report I-13). Why hadn't TVA fire inspectors detected this situation in their normal periodic inspections of fire equipment?

Answer

TVA believes that the emergency breathing apparatus was well maintained. Plant procedures for the care of this equipment were in effect and were being used, and the specified quantity of equipment was available.

Question 3 (continued)

Some air cylinders which were in the shop for reconditioning and were specifically marked were inadvertently brought to the scene of the fire, but they were not used and did not endanger personnel safety.

TVA investigations identified some problems with the air masks consisting of service life limitations, recharging deficiencies, and isolated cases of improper operator use. TVA has taken action to resolve these difficulties.

TVA investigations did not identify any deficiencies associated with the plant's inspection and maintenance program for breathing apparatus.

Question 4

Does TVA have an internal safety review committee to assess plant operating practices? At what level of management was the open flame inspection procedure approved?

Answer

TVA does have internal committees to review operating practices. At the time of the fire, procedures required review of construction activities by a licensed senior reactor operator to determine if safety considerations were involved. The senior reactor operator was empowered to authorize activities not having safety significance. Those activities recognized as having safety significance were reviewed by the Plant Operations Review Committee to determine if the necessary prerequisites have been met to authorize their performance. The leak test activities in progress at the time of the fire initiation had been reviewed and approved by the senior reactor operator who did not recognize the safety significance of this activity

Question 4 (continued)

and did not recognize that an open flame would be used. TVA has made procedural changes requiring plant supervisory review of all significant activities and review by the plant quality assurance staff to verify that all significant activities are supported with adequate instructions necessary for their performance. These reviews are a prerequisite to activity authorization by the plant superintendent.

Question 5

How often have NRC inspection and enforcement personnel visited the plant and when was the last such visit? What infractions were reported at that time, and were any of these still uncorrected at the time of the fire? If so, did any of these contribute to the accident or its aftermath in any way?

Answer

The NRC inspection and enforcement personnel have inspected the operation and construction activities at Browns Ferry Nuclear Plant 206 times prior to March 22, 1975. Prior to the fire, the NRC inspectors averaged three inspections per month. The last visits by the NRC inspectors prior to the fire were on February 25-28 and March 21, 1975. The summary of findings of that inspection and TVA's responses to those findings are attached. The only unresolved infraction concerned the format and method of approval of certain procedures relating to the nonradiological monitoring program. None of these items were related in any manner to the fire event.

ATTACHMENT TO QUESTION 5 AND ANSWER

Excerpt from NRC Inspection Report from Visit of February 25-28 and March 20, 1975

SUMMARY OF FINDINGS

I. Enforcement Items

A. Infractions

1. Contrary to Appendix B Technical Specification 2.2.3, the total residual chlorine in the condenser cooling water discharge was not monitored on a weekly basis when the auxiliary raw cooling water systems were being chlorinated.

   This infraction had the potential for causing or contributing to an occurrence related to health and safety.

2. Contrary to Appendix B Technical Specification 5.5.1 written procedures for the in-plant nonradiological monitoring program had not been prepared by the Division of Power Production.

   This infraction had the potential for causing or contributing to an occurrence related to health and safety.

B. Deficiencies

1. Contrary to Appendix B Technical Specification 5.3.2 an audit of the in-plant nonradiological environmental monitoring program had not been conducted.

2. Contrary to Appendix A Technical Specification 6.3.A.1 the reactor was twice started up on a period shorter than the period permitted by the startup operating procedure.

TVA Responses to NRC Inspection Report from Visit of February 25-28 and March 21, 1975

1.A.1 - The technical specifications required that total residual chlorine in the condenser cooling water discharge be determined weekly during periods when the raw cooling water system was being chlorinated. We met this specification by measuring the concentration in the auxiliary raw cooling water flows and computing the concentration in the main condenser cooling water discharge to the river based upon the relative flow volumes of the two systems. Because of the very low residual chlorine concentration in the condenser cooling water discharge, direct measurement with accuracy was impractical using standard instrumentation and techniques. To avoid future misunderstandings with the Inspection and Enforcement Branch, the technical specifications have been revised to specifically authorize the method in use for determining residual chlorine levels. We deny that the alleged infraction had the potential for causing or contributing to an occurrence related to health and safety.

ATTACHMENT TO QUESTION 5 AND ANSWER

I.A.2 - Written procedures were in effect for the in-plant nonradiological monitoring program but had been approved by plant section supervisors rather than the plant superintendent. A system of environmental surveillance instructions is being assembled to ensure uniformity and to provide a means to document requirement compliance. These instructions will be approved by the plant superintendent and will be completed prior to the return of any unit to service.

I.B.1 - TVA was in error in not conducting this audit during calendar year 1974. This was due, in part, to the ever-changing audit requirements. At the time this deficiency was noted, the audit was scheduled to be conducted in the second quarter of calendar year 1975. The audit was conducted on schedule and will be conducted annually hereafter.

I.B.2 - Except for minor details, the facts contained in the NRC report concerning this alleged deficiency are essentially accurate. At the time of the occurrence, the referenced operating instruction read as follows: "Observe the period meter when pulling rods, and govern withdrawal rate to avoid having a period shorter than 60 seconds." The operator was following the procedure in that he was observing the period meter and power level indication when pulling rods and was governing rod withdrawal in the most conservative, prudent manner with a one-notch-at-a-time withdrawal, waiting between notches. The reactivity change resulting from the smallest incremental movement between notches 12 and 14 was sufficient to give a period shorter than 60 seconds. The operator immediately inserted the rod a sufficient amount to stop the power increase. Following confirmation of this high notch worth, the rod program was revised. At no time was rod worth great enough to be a significant safety concern. The operating procedure has been revised in a less conservative direction to permit shorter periods without threat of violation.

Question 6

Why had the material around the penetration in question not been coated with flamemastic? Why had this penetration not been adequately filled with them in the first place? Was operation of the plant under these conditions in violation of NRC regulations?

Answer

Flamemastic was not present on all penetrations because it had been necessary to run additional cable through some penetrations during the accomplishment of plant improvement modifications. It was work prerequisite to reapplication of flamemastic material that led to the fire ignition.

The primary purpose of the polyurethane material was to seal around conductors leading into the secondary containment area. There was no requirement that individual penetrations be leak tight, but there was a requirement that the licensee be able to maintain 1/4 inch of water vacuum in the secondary containment with an in-leakage of less than 9,000 cfm. This requirement was always met.

The design of the penetration seal provided for coating completed penetrations with a flamemastic material. The fire barrier function of the flamemastic coated penetration seals is not mentioned in the Final Safety Analysis Report or other license documents. Because of this, TVA did not consider operation of the plant, under the conditions which existed, to be in violation of NRC regulations.

Question 7

Do you consider that TVA was adequately prepared to fight this fire? Are you making any changes to fire control plans and procedures as a result of this fire? Are there any actions that should have been taken by NRC or the state that might have enabled TVA to better respond to the situation?

Question 7 (continued)

Answer

TVA exhibited adequate preparation for handling a fire emergency situation by effectively implementing emergency plans and actions that ensured reactor safety and control of the fire. Reactor safety was of paramount importance and was assured at all times. The fire in the cable spreading room was controlled and suppressed at all times until extinguishment occurred. The fire in the reactor building remained in an extremely slow propagating, controlled condition until extinguishment occurred. Although preparations were adequate and effectively implemented, the cable tray fire has resulted in a reappraisal of TVA's program of fire prevention and protection for Browns Ferry. Immediate actions taken to improve fire protection at Browns Ferry have included expanding and improving the fire training of fire brigade members and leaders; establishing controls for cutting, welding, and open flames; providing trained fire watches in special areas during restoration work; providing additional portable fire-fighting equipment; and improving the scope and frequency of fire surveys. Design studies have been conducted to identify needed improvements in the fixed fire protection systems, fire water systems and related fire protection such as sealing penetrations, fire walls, etc.

TVA knows of no action that could have been taken by NRC or the State that would have improved our response and handling of the fire emergency at Browns Ferry.

Question 8

What improvements does TVA believe desirable to make in the procedures used to notify the State of Alabama health officials and other local and county civil defense and police officials in the event of incidents like the Browns Ferry fire?

Question 8 (continued)

Answer

Notification of State of Alabama health officials and other local officials of the Browns Ferry fire was carried out in accordance with the provisions in the Radiological Emergency Plan. These notifications were effective, and TVA does not plan to make any changes to the notification procedures. On September 18, 1975, TVA staff met with representatives of all local, county, and state agencies required to respond to an emergency at Browns Ferry. The primary discussion involved prompt notification of appropriate groups. Additionally, periodic reorientation meetings (at least annually) will be held with these organizations.

Question 9

According to the NRC report (II-7), there was considerable concern on the part of the operators that the water level might drop below the top of active fuel. Is that an accurate statement, and what would the consequences have been if the fuel had indeed been uncovered?

Answer

Maintaining an adequate water level in the reactor at all times is a primary concern of the operators. However, as indicated in the TVA testimony, the reactor cores were adequately cooled throughout the event and the water level remained above the reactor fuel. The reactor fuel can be uncovered for short periods of time without exceeding fuel cladding temperature of 2200° F. At this temperature, the core geometry is still amenable to cooling and the cladding would not fail upon the subsequent quenching by the addition of cooling water. Studies also show that keeping the water level above 2/3 of the active fuel is sufficient to limit the peak temperature to below 900° F.

Question 10

You note on page 4 that both units 1 and 2 were shut down shortly after anomalous events began to occur. Could this fire, or any fire, cause an inability to scram? What is the basis for your response?

Answer

The March 22 fire did not result in the loss of the ability to scram nor will any other fire result in the loss of the ability to scram. This response is based on the diversity and separation designed into the reactor protection system. The reactor protection system is fail-safe in that the failure modes in any of the logic and control circuits that would be involved in a fire results in a scram. Loss of electric power to the reactor protection system results in a scram.

Question 11

What has TVA's experience been with respect to fires in its fossil-fired units? How does this compare with that of its nuclear units?

Answer

TVA's overall experience with respect to fires in its fossil-fired plants has been quite favorable. In operating 63 fossil-fired units at 12 major steam plants, TVA has experienced only two serious fires, one at the one-unit Bull Run Steam Plant in June of 1968, which resulted in a plant outage of eight weeks, and the other at Unit 7 of the Widows Creek Steam Plant in September of 1960, which resulted in an outage of six weeks. Of course, as in similar major industrial operations, there have been other minor fires. For example, in the four-year period from 1971 to present there were 17 such small fires, 15 of which involved only local damage to equipment and two of which, though more significant, did not result in more than a few days' outage.

These latter included a coal silo fire at Cumberland Steam Plant in September of 1973, and a crusher building fire at Shawnee Steam Plant in September of 1974.

The Bull Run fire was ignited when a fuel oil gasket failed allowing the fuel to spray on hot equipment. The resulting fire burned up through eight levels of the boiler bay, causing damage to several pieces of equipment such as lube oil pumps and ducting, as well as numerous cable trays which were in the bays. The fire was controlled and extinguished in a fairly short period of time using water. The Widows Creek fire involved the unit 7 boiler fuel pump-turbine in that a line flange separated and sprayed fuel oil on hot surfaces, causing ignition. Extensive damage resulted to boiler associated equipment and controls.

Our operating experience at nuclear power plants has not been sufficient to allow us to say that experience with fires at nuclear plants will be similar to that at our fossil-fired plants.

APPENDIX 11

QUESTIONS AND ANSWERS: JCAE/STATE OF ALABAMA

LETTER DATED SEPT. 17, 1975, MURPHY TO GODWIN, WITH ADDITIONAL QUESTIONS FOR STATE OF ALABAMA

RESPONSE BY ALABAMA DEPARTMENT OF HEALTH

JOINT COMMITTEE ON ATOMIC ENERGY,
U.S. CONGRESS,
Washington, D.C., September 17, 1975.

Mr. AUBREY GODWIN,
Division of Radiological Health,
State Health Department, Montgomery, Ala.

DEAR MR. GODWIN: Enclosed is the transcript of the September 16, 1975, hearing on the Browns Ferry fire. You should review and make any necessary corrections of your testimony in accordance with the instructions attached to the transcript. Also enclosed is a list of additional questions on your testimony which the Committee would like you to provide answers to for the record.

Since we are endeavoring to publish and distribute the hearing as rapidly as possible, it would be appreciated if you would return the corrected transcript and your answers to the additional questions by no later than September 30, 1975.

Thank you for your cooperation in this matter.

Sincerely yours,

GEORGE F. MURPHY, Jr.,
Executive Director.

Enclosures:

QUESTIONS FOR AUBREY GODWIN (STATE OF ALABAMA), BROWNS FERRY HEARING, SEPTEMBER 16, 1975

1. What are the responsibilities of the Division of Radiological Health (of the State Department of Health) with respect to the Browns Ferry Nuclear Plant?

2. Do you have any review or approval functions with regard to any aspect of nuclear power plant siting, design, construction or operation in the State of Alabama? Do you believe you should have such functions?

3. In your opinion, did the State receive adequate and timely information from the TVA regarding the fire so that the State could effectively monitor the situation and take any corrective action necessary?

4. The NRC report also indicates that, during the course of the fire and upon being informed of the status of events, the Division of Radiological Health concluded that the core cooling system was degraded and must be watched. To whom were these observations made, and what actions were taken as a result?

5. What actions, if any, does the State plan to take as a result of this accident? Are any changes to emergency planning or radiological monitoring procedures contemplated?

6. According to the NRC report (IV-13) some consideration was to be given to requesting additional appropriations from the State to increase surveillance activities around reactor sites. Has any action been taken with respect to this recommendation?

[Response dated September 25, 1976 subsequently received follows:]

Answers to Questions of the Joint Committee on Atomic Energy

1. What are the responsibilities of the Division of Radiological Health (of the State Department of Health) with respect to Browns Ferry Nuclear Plant?

Response:

Alabama Law assigns the State Board of Health (Alabama Department of Public Health) the following responsibilities:

1. Regulatory control of all sources of ionizing radiation consistent with Federal Regulatory Control of byproduct, source and special nuclear material.

2. During emergencies the Alabama Department of Public Health shall issue such orders as necessary to protect the public health and safety from sources of ionizing radiation.

3. The Alabama Department of Public Health shall inform the Governor of all invasions or threatened invasions of disease.

In summary, control of ionizing radiation sources from the reactors on the Browns Ferry Nuclear Plant site is Federal; control of ionizing radiation sources off the Browns Ferry Nuclear Plant site, including discharges in excess of U.S. Nuclear Regulatory Commission limits is by the Alabama Department of Public Health.

2. Do you have any review or approval functions with regard to any aspect of nuclear power plant siting, design, construction, or operation in the State of Alabama?

Response:

Our legislative mandate requires us to review any source of ionizing radiation which may have a public health or safety impact, regardless of who licenses the use of the source. To carry out this mandate, we regularly participate in all licensing proceedings for nuclear power plants in Alabama. In these proceedings, if we have reason to believe that inappropriate siting criteria, design, construction or operation is proposed, we will oppose the application. Certain portions of the plant design, construction and operation not related to radiation control may be under the regulatory control of other Divisions of the Alabama Department of Public Health or other Agencies of the State of Alabama.

Do you believe you should have such functions?

If you are asking should a program similar to the Agreement State program be established for nuclear power plants, my answer is NO.

The following are my reasons:

A. No state agency presently has an adequate staff to do the necessary multi-discipline review of a nuclear power plant. To acquire the necessary staff, a state would have to spend substantial sums of already short funds.

B. A given state will likely have only 4 sites and may get "stale" in its review of the operations of such a small number of facilities.

If you are asking should a state review these functions and take a position, my answer is YES.

I suggest that the following should be considered by the Joint Committee on Atomic Energy:

> Those matters adjudicated as a part of a state process should not be readjudicated by the U.S. Nuclear Regulatory Commission, but should accept the sworn testimony and conclusions of the State Agency. This should not prevent the U.S. Nuclear Regulatory Commission from coming to a different conclusion from the State Agency on the overall project. However, the U.S. Nuclear Regulatory Commission should not be able to overrule a State Agency denial of a certificate of necessity or polution discharge permit. This should shorten the current hearing process, yet permit the hearing board to make the determinations required by NEPA.

3. In your opinion, did the State receive adequate and timely information from the TVA regarding the fire so that the State could effectively monitor the situation and take any corrective action necessary?

Response:

> I do not feel that the State of Alabama was informed of the fire as promptly as we desired. Subsequently, we have clarified our position on notification by the TVA to include, not only radiation related events, but any event requiring emergency assistance by "off site" agencies. Once notified, the State was properl informed of this situation.

4. The NRC report also indicates that, during the course of the fire and upon being informed of the status of events, the Division of Radiological Health concluded that the core cooling system was degraded and must be watched. To whom were these observations made, and what actions were taken as a result?

Response:

These conclusions were given to:

A. Mr. W.T. Willis, Director
Environmental Health Administration
Alabama Department of Public Health

B. Ira L. Myers, M.D.
State Health Officer

The actions taken included:

A. The activation of the Alabama Radiation Emergency Plan

B. The activation of air sampling stations near Browns Ferry Nuclear Plant

C. The notification of Governor George C. Wallace

D. The close attention given to the current status of the reactors during each contact with the TVA

5. What actions, if any, does the state plan to take as a result of this accident? Are any changes to emergency planning or radiological monitoring procedures contemplated?

Response:

The State has taken the following actions as a result of its review of this accident:

A. Check and update the telephone directory at least every six months. The latest amendment to the Alabama Radiation Emergency Plan containing such changes is dated September 1, 1975.

B. Require each recipient of the Alabama Radiation Emergency Plan to document filing of changes. This procedure started with the September 1, 1975, amendment.

C. Now included in the Alabama Radiation Emergency Plan is a statement defining state and local responsibilities for informing officials of the Plan.

D. Arrangements have been made with the TVA to notify this Agency any time emergency assistance is requested of an "off-site" agency. This is not a part of the Alabama Radiation Emergency Plan since these calls will not be about a radiological incidnet. Potential radiological incidents will remain a portion of the Alabama Radiation Emergency Plan.

6. According to the NRC (IV-13) some consideration was to be given to requesting additional appropriations from the state to increase surveillance activities around reactor sites. Has any action been taken with respect to this recommendation?

Response:

The Division of Radiological Health's budget for the year ending September 30, 1975 consisted of the following sources of funds:

General Fund		$65,510.00
Transferred to the Division from funds appropriated for general public health use--Cigarette Tax		36,4[illegible].00
U.S. Nuclear Regulatory Commission contract for reactor monitoring		6,[illegible].00
	Total	$12[illegible],[illegible].00

The Alabama Department of Public Health proposed in its 1976 budget the following:

General Fund		$159,868.00
To be transferred to the Division from funds appropriated for general public health use--Cigarette Tax		12,000.00
U.S. Nuclear Regulatory Commission contract for reactor monitoring		6,500.00
	Total	$178,368.00

The budget bill HB No. 490 in the current substitute proposes for the Division the following:

General Fund		$35,510.00
To be transferred to the Division from funds appropriated for general public health use--Cigarette Tax		36,490.00
U.S. Nuclear Regulatory Commission contract for reactor monitoring		6,500.00
	Total	$128,500.00

The Department of Public Health requested the introduction of a fee bill which would permit the charging of fees for radioactive material licenses and the the registration of sources of radiation. This is HB No. 967.

The projected revenue from this bill is $158,000 of which $22,000 would not be available to the Division. These funds would replace the Cigarette Tax funds and would allow for equipment expenditures not proposed in the original Departmental Budget.

This bill is currently pending before the House of Representatives of the Legislature.

APPENDIX 12

AEC BULLETINS ON USE OF POLYURETHANE

ISSUE No. 170

ISSUE No. 254

ISSUE No. 336

HEALTH and SAFETY information
United States Atomic Energy Commission

ISSUE NO. 170 September 10, 1963

FOAMED POLYURETHANE FIRE PROVES DIFFICULT TO CONTROL

While the incident described below did not occur at an AEC facility, the lessons learned are of interest at AEC locations utilizing foamed polyurethane.

The incident occurred in a sphere (diameter 41') containing a 12'x 12'x 10' room designed to provide a very high degree of sound absorption through the use of numerous large pieces of foamed polyurethane which were individually shaped in the form of pyramids and bonded to an external metal framework. Eleven hours following the conduction of a test, this room was entered and nothing unusual was reported to have been noted. Approximately five hours later, when the room was next entered, smoke was noted coming from two of the pyramids. These were removed and work on modifying the ceiling of the room continued. From the details of the ensuing events given below, it is apparent that what was initially thought to be a minor fire problem turned out to be a complex firefighting task that could not be solved by application of large quantities of inert gas or thousands of gallons of water.

Work continued with two employees on top of the room walls drilling holes in angle iron support frames with portable electric drills. Smoke became evident in the room and fire in another part of the structure. One employee attempted to remove the burning pyramids while the fire department was being called. Smoke conditions were such that, without breathing equipment, the efforts to remove the burning materials had to be abandoned.

Upon arrival at 12:55 p.m., the fire department found light smoke coming out of the top and bottom of the sphere. The technician in charge requested that water be avoided, if possible, because of the possible damage it would do to electronic test equipment in the setup. An attack on the fire was made with CO_2 extinguishers which had little or no effect. Smoke increased until visibility was less than one foot.

A booster line was taken into the bottom opening of the sphere. Firefighters donned self-contained breathing apparatus and attacked the fire with water, which seemed to be controlling the surface fires. Visibility by this time was zero. A 2½" line was laid from the nearest hydrant (500' distant) and a 750 gpm pumper was placed in operation. This line was wyed into two 1½" lines, which were taken to the top opening of the sphere.

ISSUE NO. 170 September 10, 1963

A blower was placed in the lower opening in an effort to force out some of the smoke and improve visibility. The water applied from the top was able to reach two sides of the room; however, this knocked down burning insulation to provide a heavy bed of glowing coals in the bottom of the sphere.

For approximately two hours this condition existed: a 1½" fog nozzle was used at the top opening discharging 90 gpm: a 1½" fog nozzle at the same location was used to protect the plyboard structure; and a 1½" line was operated through the bottom opening. Firefighters on top were able to operate without breathing equipment, but no entry through the bottom was possible without complete respiratory protection. Water was being drained out through plugs that were removed from the bottom of the sphere and was running out of the access hole where the attack was being made on the fire. It was evident that the overall temperature of the structure was being kept to a safe level but the water was doing nothing to extinguish fires within the pyramids.

The top hose was shut down and an exhaust fan placed over the opening to reduce smoke within the structure. Smoke conditions improved and firefighters were able to pick out hotspots and remove burning panels. A pole was used to break open the pyramids and improve accessibility to the deep-seated fires. A fire watch was posted until 9:00 p.m. when security was turned over to the guard force.

During the fire, approximately 75 tanks of breathing air, at least 24,000 gallons of water, and considerable quantities of carbon dioxide were used.

This incident pointed out the need for:

(a) A better appreciation of polyurethane fire risks by all personnel involved.

(b) Smoke-ejecting equipment and additional portable breathing equipment.

(c) Prompt notification of the fire department in case of <u>any</u> small fire.

(d) Consideration of attaching polyurethane so that it can be removed more easily in case of fire. (Note: No method is currently known for effectively controlling deep-seated fires within large pieces of foamed polyurethane.)

Division of Operational Safety
U. S. Atomic Energy Commission
Washington 25, D. C.

HEALTH and SAFETY
information
United States Atomic Energy Commission

ISSUE NO. 254 July 14, 1967

RECENT FOAMED POLYURETHANE FIRES ON NON-AEC FACILITIES

Incident #1

While cutting off bolts on the outside of an enclosed 20' x 20' x 80' test facility, two hot bolts and some molten metal fell (illustration 1) and lodged between the exterior aluminum wall cladding and the interior sprayed-on polyurethane insulation. Shortly thereafter, a firewatch smelled smoke and, after picking up an extinguisher, ran into the building but was immediately driven off by the intense heat and smoke. Smoke coming from the top of the tower attracted attention of fire department personnel who responded without waiting for receipt of a fire alarm. Intense smoke and heat prevented entrance through any opening into the building interior. Conduct of effective firefighting, even by personnel using self-contained respirators, was impossible. Hose streams applied to the external top of the building were ineffective. During the 20-minute duration of the blaze, periodic explosive sounds were heard, followed by puffs of smoke and fire.

Illustration 2 shows the point, 16' above floor level, at which hot bolts penetrated into the building interior to initiate a "chimney" fire which, as shown in illustrations 3 and 4, extended upward about 4'. Above this point, the blaze fanned out both vertically and horizontally to consume virtually all of the foamed polyurethane fuel in its path (see illustration 5) while leaving lower building wall portions unaffected.

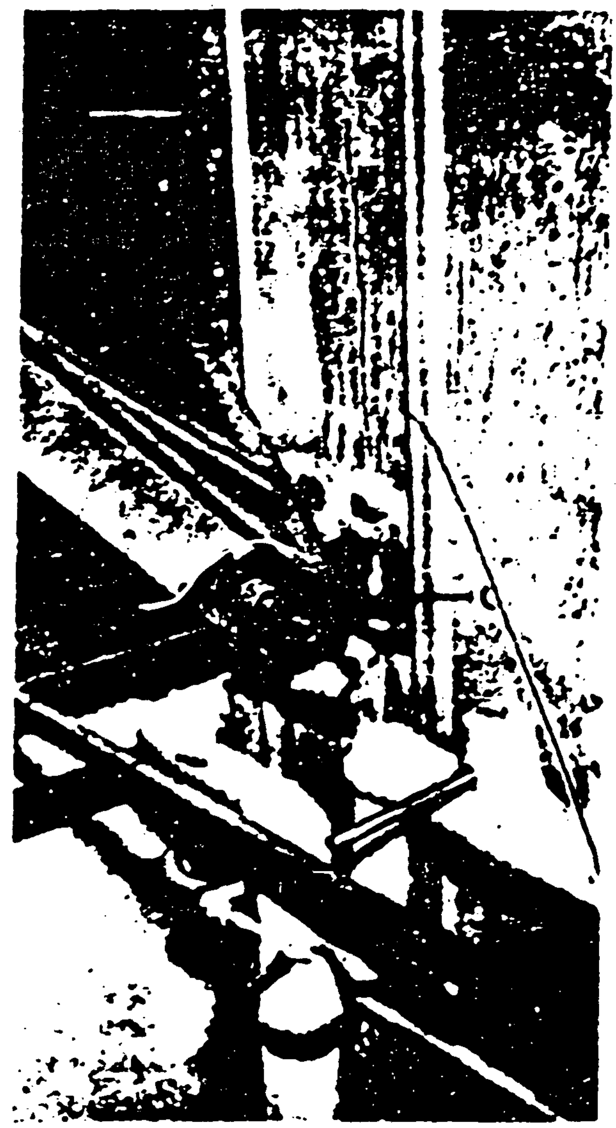
We understand that subsequent foamed polyurethane tests demonstrated ability to ignite test specimens with an acetylene torch but that self-sustaining combustion of the urethane did not persist after removal of the torch. Red hot bolts placed on other test specimens reportedly produced smoke and charring but no flame. These (and other) tests by the company were interpreted as showing that the polyurethane insulation did possess fire-resistant properties, but that under particular conditions of flame or sparks, and with the proper amounts of air, the material was proved capable of supporting and sustaining combustion. It was concluded that polyurethane insulation should not be used where there is a possibility of an open flame and spark and where a limited amount of air is present but that the insulation could be used where

(over)

- Illustration 1 -

Location "A" indicates the position of the two one-inch carbon steel bolts which were burned off the support walkway.

Location "B" indicates the point of penetration into the tower.



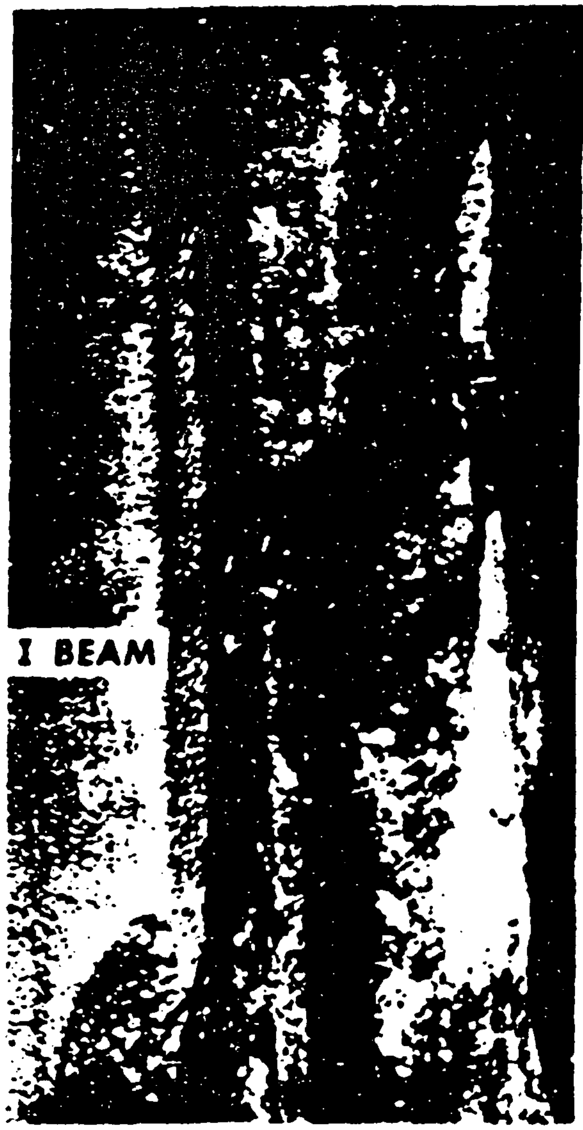
- Illustration 2 -

Location "C" is a closeup shot of the area indicated as Location "B" in illustration 1.

(continued)

- Illustration 3 -

Indicates the "chimney" where the fire started within the tower. The area behind the I-beam is a corner.



- Illustration 4 -

Indicates results of the fire behind the I-beam depicted in illustration 3. Fire burned in the "chimney" approximately four or five feet vertically before horizontal spr--'' occurred.

- Illustration 5 -

Indicates completeness of burning. Note that only a small amount of urethane insulation remains after the fire and is charred.

(continued)

an infinite amount of air, or no air, was present. We understand that this conclusion will be implemented by applying foamed polyurethane to the building exterior but that in doing so extensive care will be taken to avert void areas, frequent noncombustible firebreaks will be provided, and a fire-resistant coating will be applied to the externally exposed portions of the insulation. (Note: While the conclusion concerning accentuated polyurethane combustibility under conditions involving restricted access to air was found surprising, it is of interest to note that all reports of polyurethane fire known to this office have uniformly stressed the profuse generation of dense smoke, a fact suggesting credibility of combustion under conditions limiting fire access to oxygen. It is suspected by many that the heat insulating properties of the foam may also play a major role in influencing urethane combustion.)

Incident #2

This incident occurred within a reactor containment enclosure. Smoke was noted about three hours after foamed polyurethane insulation had been sprayed onto some piping. Identification of the fire source was seriously impeded by the high density of the smoke. Using self-contained respirator equipment, it was found that the recently applied insulation was smoldering and was the source of "considerable smoke and soot." Roughly two hours after its detection, and after applying the contents of a large portable extinguisher, tripping a built-in CO_2 fire extinguishing system, and application of water sprays, the fire was extinguished. Subsequent investigation revealed fire damage was confined to insulation on a few pipelines, but smoke and soot damage was much more extensive.

This fire was attributed to spontaneous ignition resulting from the use of improper techniques in applying polyurethane foamed insulation. One report on the incident stated:

> "The curing process of the sprayed polyurethane insulation is exothermic. If applied in successive layers without sufficient curing time, its temperature will increase, breaking down its structure, and eventually causing ignition."

Following the incident, it is understood that all remaining polyurethane insulation was removed from the affected facility and replaced with materials considered to be less hazardous.

(over).

ISSUE NO. 254 July 14, 1967

Comments

The problem of controlling polyurethane fire was previously emphasized in HEALTH AND SAFETY INFORMATION Issue #170. Information concerning polyurethane fire risks is currently being gathered from a variety of sources. Analysis of this information is complicated by clear disagreement as to acceptable yard-sticks for testing and evaluating these risks. Such conflicts are not surprising when it is recalled that many of the above-referenced tests, made following incident #1, could individually have been construed as providing evidence attesting that incident #1 could not occur. It is anticipated that a summary of findings resulting from study of urethane fire risks will be published during the next few months.

Division of Operational Safety
U. S. Atomic Energy Commission
Washington, D. C. 20545

ISSUE NO. 33
September 19, 1974

SERIOUS Accidents

THE UNITED STATES ATOMIC ENERGY COMMISSION

TWO FIRES INVOLVING URETHANE FOAM

The AEC recently experienced two fire losses in which polyurethane foam plastics played a major role. Following is a brief description of these fires, some background information on foams, and a summary of the lessons learned:

Occurrence Number 1

At a large laboratory, a number of accelerator beam tunnels of corrugated steel construction are connected to a 150- by 230-foot-experimental building. Near the far end of one 430-foot-long tunnel, workers were cutting a bolt hole with a welding torch from the outside of the structure. The tunnel, which is approximately 12-feet high, had been lined with a 1-1/2-inch thick, sprayed-on, polyurethane foam and during the cutting operation, this foam was ignited. Although the resulting fire revealed a number of fire protection deficiencies, only the problem of the foam is discussed here.

It was intended that the foam for this tunnel meet a flame spread rating of 25 and a smoke developed rating of 50 or less in accordance with UL 723 "tunnel test." The material that was actually installed was described as "self-extinguishing" by the ASTM 1692-67T test, but it did not have a "tunnel test" rating. In addition, it had been planned to coat the foam with an intumescent paint, install a 10-foot, glass fiber insulation firebreak at the tunnel end adjoining the experimental building and install a sprinkler water curtain at this end. At the time of the fire, the sprinklers were installed but valved off due to incomplete construction, the 10-foot firebreak was installed, and 150 feet of the foam closest to the experimental building were coated with a fire-retardant (but non-intumescent) paint.

Despite the above considerations, the fire swept the entire 430-foot length of the tunnel in less than 8 minutes, filled the laboratory with smoke, and burned plastic-reinforced fiberglass panels and plastic windows in the adjoining wall of the laboratory. The fire progressed rapidly with intense smoke development, similar to a test in a mine tunnel with
dimensions which

Research Establishment in the March 1972 issue of the English magazine Fire. At the peak of the firefighting effort, over 2,000 gallons of water per minute were being applied to the fire by several fire departments.

Occurrence Number 2

A second incident occurred in a foundry-type building where urethane foam panels had been installed in a 5 by 17 by 13-foot-high breakout station to reduce noise levels. Lead-loaded vinyl curtains and plexiglass windows constituted the remaining combustibles. Occasionally, a leaking mold allows water to react with the hot metal, producing some pyrophoric residue and flammable gas which sometimes flashes when the mold is inverted in the breakout operation, resulting in a small fireball at the mouth of the pot. In this instance, the vinyl material and urethane ignited and forced evacuation of the building before the operators could take effective firefighting action. Dense smoke built up so fast that an ionization-type smoke detector some 260 feet south, in a separate room, alarmed within 5 minutes of the initial "flash."

Although both the curtain material and the foam were supposedly "self-extinguishing," tests showed the curtain material burned readily, producing large amounts of smoke, and the foam burned as long as it was exposed to flame. A test on a piece of older foam material showed that it ignited and burned freely, indicating an adverse aging effect.

Background on Foams

Small scale laboratory tests have long been recognized as inadequate to determine the fire properties of a material in large-scale applications. ASTM D635 and D1692 tests are used to compare relative flammabilities of plastic materials on a laboratory scale. Unfortunately, the terms "self-extinguishing" or "slow burning" as defined by these laboratory scale tests do not necessarily indicate a low combustibility in large scale wall and roof covering applications. The unsuitability of this interpretation was best demonstrated in an article in the NFPA quarterly for October 1962, wherein J. W. Wilson, Chief Construction Engineer for the Factory Mutual Engineering Division, utilized the tests to compare samples of four woods. Only balsa wood was classified as "burning" by the D635 test; all other woods were classified as "self-extinguishing." The further unsuitability of the D635 test for extrapolating into large scale applications is apparent from a review of 1971-72 Modern Plastics Encyclopedia where 146 of 149 plastics tested in accordance with D635 were listed as "self-extinguishing," "slow burning," etc.

(continued)

The application of the fire retardant paint to the surface of the foam in Occurrence 1 illustrates another difficulty in accepting test results without a full understanding of the application. While the material used was UL listed and FM approved, the applications applied to wood, masonry, plaster, and drywall surfaces. The application instructions did not cover the coating of foams. In fact, there is no fire retardant paint approved as a protective coating for foams.

The AEC Health and Safety Information Bulletin number 254 of July 14, 1967, reports two urethane foam fires which demonstrate another problem with these laboratory scale tests. While fire tests are performed by exposing the source of ignition to the exposed surface of the foam, fires have resulted from the ignition source applied to the outside, or protected face, or by a source, such as hot bolt, dropping between the metal wall and the insulating foam. This may be a more severe exposure than the normal test procedure since the heat may be concentrated by the insulating quality of the foam rather than being dissipated by the open air atmosphere which surrounds the normal test setup.

A summary of tests reported in the April 1973 issue of Fire concludes that flame retarded foams may be as easy to ignite as untreated foams when realistic ignition sources, such as a sheet of newspaper, are used. In fact, since the treated foams burned hotter, produced much denser smoke and at least as much carbon monoxide and hydrogen cyanide, it appears that a "fire retardant" type of foam could be a greater hazard in life safety situations than the untreated type.

The May 1974 issue of Fire Technology notes that the Urethane Safety Group and the Plastics in Construction Council, both of the Society of the Plastics Industry, have recently urged that the nation's building codes require that rigid urethane foam insulation be covered with a "suitable barrier against ignition" and not be left exposed after installation. The Urethane Safety Group is now preparing a new set of safety guidelines on applications which is expected to be available from the Society of the Plastics Industry, Department U-100, 250 Park Avenue, New York, New York 10017, in the near future. Factory Mutual Loss Prevention Data Sheets 1-57 and 8-29 are useful guides regarding methods of protecting foams. In the past, tests have been made on specific applications at AEC contractor sites and various guidelines were developed for specific situations or applications. In view of the current nationwide interest, the many tests now being made, and the new test methods under development, AEC criteria can be expected to change to reflect any new information.

Lessons Learned

While there are few significant new lessons to be learned from these two occurrences, a number of old lessons are evidently worth relearning. Specifically:

(over)

1. All new installations and modifications need a safety analysis--even modifications required for safety reasons. This is not the first instance in which a modification that was made to solve one safety problem created another.

2. The suitability of materials for any application must be based on the actual installation and its surrounding environment. Acceptance of fire tests in particular is not valid unless the tests apply to the intended installation. For example, a fire may be an acceptable risk while the smoke or toxic fumes which are generated may not be acceptable.

3. Merely citing that a material meets a test is meaningless without specifying the limits that are acceptable to the user. Claims that a product "meets a test" may be misleading in that tests merely standardize <u>how</u> a test is performed, not what results are acceptable. A statement that a plastic "meets ASTM E-84" (the tunnel test) is no more a guarantee of fire safety than a statement that a concrete "meets ASTM C-143" (the slump test) is a guarantee of quality. In both cases the user must specify the limit that is acceptable for his application.

4. Control of construction programs should consider sequencing of hazardous operations. Cutting and welding operations should preferably be completed before any necessary combustibles are introduced to the building. If this is not practical, it is preferable to delay the hazardous work until the protection systems are in service.

5. Fire protection systems should be in service before the building or equipment they protect is in service.

Division of Operational Safety
U. S. Atomic Energy Commission
Washington, D. C. 20545

APPENDIX 13

CORRESPONDENCE: NRC/COMEY

EXCHANGE OF CORRESPONDENCE BETWEEN NRC AND DAVID COMEY MARCH 24, 1975 THROUGH AUGUST 4, 1975

BUSINESS AND PROFESSIONAL PEOPLE FOR
THE PUBLIC INTEREST (PBI),
Chicago, Ill., March 24, 1975.

[Mailgram]

WILLIAM ANDERS,
Chairman, U.S. Nuclear Regulatory Commission,
Washington, D.C.:

Copy of mailgram sent this date to Bernard C. Rusche, Director, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555.

DEAR DR. RUSCHE: The March 22 occurrences in Dockets 50-259, 50-260, and 50-296 represent a serious failure to meet the intent of criterion 22, Appendix A, 10CFR 50. Therefore, pursuant to 10CFR 2.206, 2.202, and 50.100, BPI hereby formally requests you take the following specific actions.

(1) Review and identify those operating nuclear power plants that do not fully comply with IEEE-279 criteria for nuclear power plant protection systems and show cause why these plants should not be required to shut down until they are retrofitted to meet said criteria.

(2) Adopt regulation requiring shutdown of operating units at multi-unit facilities during periods when work on a unit under construction could compromise the integrity of the engineered safety features on an operating unit or units.

DAVID DINSMORE COMEY,
Director of Environmental Research, BPI.

U.S. NUCLEAR REGULATORY COMMISSION,
Washington, D.C., April 8, 1975.

Mr. DAVID DINSMORE COMEY,
Director of Environmental Research, Business and Professional People for the Public Interest, 109 North Dearborn Street, Suite 1001, Chicago, Ill.

DEAR MR. COMEY: This is to acknowledge receipt of your March 24, 1975 letter which refers to the recent fire at the Browns Ferry Nuclear Station and makes two requests for Commission action. These requests, filed under 10 CFR 2.206, are under active consideration and you may be assured that within a reasonable time, we will take the action appropriate in the circumstances.

The first request is being handled as a Petition for Order to Show Cause under 10 CFR 2.202, to operators of nuclear power plants "that do not fully comply with IEEE-279 criteria for nuclear power plant protection systems". A copy of a related notice that is being filed with the Office of the Federal Register for publication is enclosed.

The second request is being handled separately as a petition for rulemaking. We plan to file a notice on this matter with the Office of the Federal Register for publication. A copy of our notice on your rulemaking petition will be sent to you by separate letter.

Sincerely,

EDSON G. CASE,
Acting Director,
Office of Nuclear Reactor Regulation.

Enclosure.

U.S. NUCLEAR REGULATORY COMMISSION,
Washington, D.C., April 8, 1975.

Docketing and Service Section, Office of the Secretary of the Commission.

PETITION FOR SHOW CAUSE ORDER

Two signed originals of an Order identified as follows are enclosed for your transmittal to the Office of the Federal Register for filing and publication: Nuclear Regulatory Commission, David D. Comey, Petition for Show Cause Order.

Twelve additional conformed copies of the Order are enclosed for your use.

A. SCHWENCER,
Light Water Reactors Branch 2-3,
Division of Reactor Licensing.

Enclosure.

NUCLEAR REGULATORY COMMISSION—DAVID D. COMEY

PETITION FOR SHOW CAUSE ORDER

Notice is hereby given that on March 24, 1975, David Dinsmore Comey made a request pursuant to 10 CFR § 2.206 of the Commission's Rules of Practice for the issuance of an Order to Show Cause why operation of nuclear powerplants should not be suspended pending full compliance with IEEE-279 criteria for nuclear powerplant protection systems. In accordance with the procedures specified in 10 CFR § 2.206, appropriate action will be taken on this request within a reasonable time.

A copy of the Petition for the Order to Show Cause is available for inspection in the Commission's Public Document Room, 1717 H Street, N.W., Washington, D.C. 20555.

Dated at Bethesda, Maryland this 8th day of April 1975.

FOR THE NUCLEAR REGULATORY COMMISSION
EDSON G. CASE,
Acting Director,
Office of Nuclear Reactor Regulation.

U.S. NUCLEAR REGULATORY COMMISSION,
Washington, D.C., August 4, 1975.

Mr. DAVID DINSMORE COMEY,
Director of Environmental Research, Business and Professional People for the Public Interest, 109 North Dearborn Street, Suite 1001, Chicago, Ill.

DEAR MR. COMEY: This is in response to letter of June 16, 1975, which limits the request in paragraph (1) of your March 24 letter to a review by the NRC Staff of two plants (Indian Point 1 and Dresden 1) pending receipt of Dr. Hanauer's report on the investigation and review of the Browns Ferry fire. We are able to satisfy a portion of your request at this time.

For Indian Point 1, the Staff required the licensee in 1973 to provide information which compared the as-built plant protection system to IEEE-279, as published in 1968. Based on our review of that information, we determined that certain changes in the plant would provide substantial additional protection which is required for public health and safety. Accordingly, on June 11, 1973, the licensee was ordered, pursuant to 10 CFR 50.109, to complete these modifications within one and one-half years. Although these modifications have not yet been completed, the Indian Point 1 facility has been shut down since October 31, 1974. A copy of our June 11, 1973, order to the Consolidated Edison Company is enclosed for your information.

With regard to Dresden 1, we have recently requested Commonwealth Edison to compare the as-built design of the plant with the 1968 edition of IEEE-279, to identify features of the plant that do not comply with this standard, and to determine whether, based on this review, there are changes in the facility that should be accomplished which provide substantial additional protection which is required for public health and safety. A copy of our letter to Commonwealth Edison dated July 31, 1975 is enclosed. Based on our evaluation of the licensee's report, the staff will then determine what actions, if any, with regard to Dresden 1 are necessary to protect the health and safety of the public.

We expect that the licensee and Staff review will require several months to complete, and will provide you with the results of these reviews.

Sincerely,

EDSON G. CASE,
Deputy Director,
Office of Nuclear Reactor Regulation.

Enclosures:

ENCLOSURE 1

U.S. ATOMIC ENERGY COMMISSION,
Washington, D.C., June 11, 1973.

Attn: MR. WILLIAM E. CALDWELL, *Vice President,*
Consolidated Edison Co. of New York, Inc.,
4 Irving Place, New York, N.Y.

GENTLEMEN: As you know, we have under review your application for a full-term operating license for Unit No. 1 at the Indian Point Nuclear Generating Station. Our review includes consideration of the need for backfitting Unit No. 1 to provide substantial, additional protection which is required for the public health and safety. Our review of portions of the plant protection system has been completed, and we find that modifications to the system are necessary.

In the event of an accident jeopardizing the integrity of the core, Section 6 (Accident Analysis) of the Safety Analysis Report for Core B indicates that the plant protection system would function to terminate the accident before fuel and fuel clad melting could occur in the hottest channel and result in the release of fission products to the reactor coolant system. Although containment is provided to retain fission products should they be released from the reactor coolant system, protection systems for modern nuclear plants are designed to meet the single failure criterion of IEEE Std. 279–1968, "Criteria for Nuclear Power Plant Protection Systems", and the General Design Criteria of Appendix A to 10 CFR Part 50. By letters dated March 31, 1972, and April 28, 1972, from your attorneys, you submitted in support of your application additional information which compares the plant protection system to IEEE 279 as published August 30, 1968, and to General Design Criterion 20, "Protection Systems Redundancy and Independence", and General Design Criterion 21, "Single Failure Definition", as published in the *Federal Register* on July 11, 1967. Our review is based on this information and other information in the public record.

We find that improvements in the plant protection system, including circuits for isolating containment and for protection against steam line breaks, will provide substantial, additional protection for the health and safety of the public, and that such improvements are required for the health and safety of the public. Therefore, pursuant to Section 50.109, "Backfitting", of 10 CFR Part 50, you are hereby directed to submit, within six months from the date of this letter, the following:

1. Details of proposed modifications that are necessary to make the design of protection circuits essential to safety conform with the requirements of Section 4.2, "Single Failure Criterion", of IEEE Std. 279–1968.

2. Details of proposed modifications and analyses of such modifications that are required to assure that dc electric power systems essential to safety meet the single failure criterion.

3. Your detailed analyses of all portions of the protection system including the proposed modifications which show that the modified system meets the single failure criterion.

4. Results of analyses and/or tests which demonstrate that all instrumentation and electric equipment essential to safety can function in its environment during and following an accident. If satisfactory results are not obtained, describe the modifications necessary to assure that all instrumentation and electric equipment essential to safety can function in its accident environment.

Further, you are hereby directed to complete within one and one-half years from the date of this letter, all such modifications and tests necessary to conform to the requirements of the single failure criterion.

In developing the proposed modifications necessary to satisfy the requirements specified in 1. and 3. above, the following concepts should be applied in your analyses of the modifications.

1. The types of single failures to be considered are:

a. failure to any single component or module or a single circuit fault. (Circuit faults include short circuits, open circuits, grounds, and the application of any available ac or dc potential.)

b. multiple component failures and/or circuit faults that can be predicted to occur as a result of a single event within or external to the system.

2. Each single failure is considered to occur coincident with any and all combinations of nondetectable failures. (Nondetectable failures are failures that cannot be detected by periodic tests, anomalous indications, or alarms.)

3. If an event that results in the need for protective action can cause failures in the protection system or if failures in circuits common to control and protective functions can result in the need for protective action, it must be shown that no single failure, in addition to the event-caused failures or the failures in the common portion of the protection system, will prevent the protection system from performing the required protective action(s).

Based on our review of portions of the protection system, the following suggested modifications are examples of the kind of modifications that might be made to the protection system in order to satisfy the single failure criterion:

(a) install an additional scram bus similar to the existing scram bus;

(b) connect the two scram buses to the scram solenoid valves in a manner which ensures that either bus is capable of deenergizing the solenoid valves regardless of whether the other bus is operable or failed;

(c) disconnect one of the nuclear instrumentation logic networks from the existing scram bus and connect it to the additional similar scram bus;

(d) transfer either the high pressurizer pressure or the high primary pressure trip relay contacts from the existing scram bus to the additional similar scram bus;

(e) install additional independent high primary temperature, low primary pressure, and low boiler feedwater pressure instrumentation for use with the additional similar scram bus;

(f) transfer one flux/flow computer from the existing scram bus to the additional similar scram bus and install additional flux averaging amplifiers, one for use with the additional similar scram bus and one for use with the automatic reactor control system;

(g) modify the existing scram bus and design the additional similar scram bus (or design their connections to the scram solenoid valves) in a manner that permits testing individually the two scram buses during reactor power operation; and

(h) provide redundant initiating devices in the criteria for protecting against the effects of a steam line break and for isolating containment.

If we can provide additional guidance with regard to your preparation of these required submittals, please do not hesitate to call (Mr. Roger Woodruff is the Regulatory Project Manager).

You should appreciate that, in addition, there are other technical issues related to your pending application for a full term operating license for Indian Point Unit No. 1 now under review by the AEC staff. In this connection, we are concerned with the frequent delays that have been experienced in your responses to our requests for additional information on these issues. We believe that this matter warrants prompt management attention and request that you take the necessary steps to remedy this situation.

Sincerely,

A. Giambusso,
Deputy Director for Reactor Projects,
Directorate of Licensing.

SAFETY EVALUATION BY THE DIRECTORATE OF LICENSING—CONSOLIDATED EDISON CO. OF NEW YORK, INC.

INDIAN POINT 1; DOCKET NO. 50-3

Background

As part of our review of Consolidated Edison Company's application for a full-term operating license for Unit No. 1 at the Indian Point Nuclear Generating Station, we have completed a review to determine the need, if any, for backfit of the plant protection system. Our position and the reasons for it are discussed in the following sections.

Description of the Plant Protection System

Principal features of the Unit No. 1 plant protection system are a scram bus, two logic networks, and circuits for isolating containment and for isolating steam lines.

When values of important plant parameters exceed their limiting safety system settings, relays in the scram bus automatically cause insertion of the control rods by deenergizing a solenoid valve in the hydraulic system for each control rod. Principal relays are controlled by instrumentation which monitors reactor coolant temperature and pressure, boiler feedwater pressure, and the ratio of neutron flux to reactor coolant flow.

Neutron flux sensors monitor reactor power and provide input signals to the logic networks. These networks supply electrical power to the scram bus and are intended to interrupt electrical power to the scram bus if reactor power exceeds the limiting safety system setting.

Multiple input signals to the logic networks are provided by each of 18 neutron flux sensors. The signals are connected in sets of three to each of nine "two-out-of-three" modules in each network. If any two signals to a "two-out-of-three" module exceeds the limiting safety system setting, the module initiates a scram signal. The outputs of the nine "two-out-of-three" modules are connected to a "one-out-of-nine" module. On receipt of a scram signal, a solid state switch in this module "opens" interrupting power to two trip actuator amplifiers. A solid state switch in each of the parallel trip actuator amplifiers then "opens", disconnecting power from the scram bus. Because the two logic networks are connected to the scram bus in parallel, both logic networks must function to deenergize the scram bus.

Three of the 18 neutron flux sensors also provide signals to a flux averaging amplifier. The output from this amplifier and signals from reactor coolant flow sensors are connected to two computers which calculate the ratio of flux to flow. If the value of the ratio is excessive, each computer demands that each of two pairs of parallel relay contacts opens in the scram bus. Other pairs of parallel relay contacts in the scram bus are opened by sensors monitoring boiler feedwater pressure, reactor cooling system pressure, and reactor cooling system temperature.

To prevent overcooling of the reactor core in the event of a steam-line-break accident and to limit the reactivity and power transients which would result, circuitry is provided to close the steam stop valve in the affected loop if a reactor scram occurs in coincidence with low steam pressure. A pressure drop across the steam stop valve in coincidence with scram causes closure of the stop valves in the associated reactor cooling loop. These functions are initiated by a single switch or relay.

Containment is partially or completely isolated automatically in the event of high containment pressure, high radiation, and very low pressurizer level. For each case, isolation is initiated by a single relay.

Safety Evaluation

IEEE 279 establishes several criteria for the design of plant protection systems. However, the fundamental aspects of IEEE 279 is the single failure criterion which is intended to assure that the protection system will function in the event that any single failure or fault exists somewhere in the protection system. Our safety evaluation is based on this fundamental aspect of IEEE 279.

With regard to the scram bus, application of stray power to it can disable all relays and solid state switches from the point of application of stray power back to, and including, the trip actuator amplifiers, thus, defeating the scram functions associated with the relays and solid state switches. Other kinds of single failures, e.g., failed components, short circuits, open circuits, and grounds, can also disable relays in the scram bus and cause loss of specific scram functions.

In the logic networks, random failure of any one of six solid state switches in the "closed position" will disable the high neutron flux scrams. In the circuits for isolating containment and for isolating steam lines, single failures of initiating mechanisms would cause loss of protective action.

Although we have not reviewed all portions of the plant protection system, it is clear that single failures can be expected to inhibit the plant protection system and that backfitting the plant protection system will provide substantial, additional protection which is required for the health and safety of the public.

Backfit and Information Requirements

Pursuant to 10 CFR Part 50, Section 50.109, we are requiring that Consolidated Edison submit, within six months from the date of this evaluation, proposed modifications to the plant protection system which will provide significantly greater assurance that the plant protection system will perform its intended function should any of the following kinds of single failures occur: failed components, short circuit, open circuits, grounds, and application of any available ac or dc potential. The proposed modifications must include capability at power for conducting surveillance on the plant protection system to assure that it is capable of performing its intended function. Further, we are requiring that Consolidated Edison complete approved modifications of the plant protection system within one and one-half years from the date of this evaluation.

R. W. WOODRUFF,
Operating Reactors Branch #1,
Directorate of Licensing.
ROBERT J. SCHEMEL,
Operating Reactors Branch #1,
Directorate of Licensing.

U.S. NUCLEAR REGULATORY COMMISSION,
Washington, D.C., July 31, 1975.

Attention Mr. J. S. Abel.
COMMONWEALTH EDISON CO.,
Nuclear Licensing Administrator, Boiling Water Reactors,
P.O. Box 767, Chicago, Ill.

DEAR MR. ABEL: As you may know, on March 24, 1975, Mr. David Comey, representing Businessmen for the Public Interest, requested, among other things, that the NRC identify those operating nuclear power plants that do not fully comply with IEEE-279 criteria for nuclear power plant protection systems, and show cause why such plants should not be required to shut down until they are retrofitted to meet these criteria. A copy of Mr. Comey's request is enclosed (Enclosure 1). Based on a review of information in our files, those plants which did not comply with IEEE-279 were identified and provided to Mr. Comey in a letter dated May 5, 1975 (Enclosure 2). In this letter, we suggested to Mr. Comey that BPI might wish to modify its show cause request pending completion of the NRC's ongoing investigation and review of the Browns Ferry fire. In his reply of June 16, 1975, Mr. Comey agreed to such a modification, except as his initial request related to Indian Point 1 and Dresden 1. A copy of Mr. Comey's June 16 letter is also enclosed (Enclosure 3).

With regard to the Indian Point 1 facility, the NRC staff has previously conducted a review of the design of this facility with regard to compliance with IEEE-279 and ordered the licensee pursuant to 50.109 to make certain modifications to the plant. A copy of our letter to Consolidated Edison dated June 11, 1973, concerning this matter is enclosed for your information (Enclosure 4). With regard to Dresden 1, the information available to the staff is not sufficient to make a comparable evaluation. Accordingly, pursuant to 50.109(c) of the Commission's regulations, you are requested to compare the as-built design of the Dresden 1 plant with the 1968 edition of IEEE-279, to identify features of the plant that do not comply with this standard, and to determine whether, based on your evaluation, there are changes in the facility that should be accomplished which provide substantial additional protection which are required for public health and safety. The results of your evaluation should be provided to the staff within 60 days of the date of this letter.

If you have any questions concerning this matter, please let me know.

Sincerely,

EDSON G. CASE, *Deputy Director,*
Office of Nuclear Reactor Regulation.

Enclosures:

ENCLOSURE 1

BENARD C. RUSCHE,
Director, Office of Nuclear Reactor Regulations, U.S. Nuclear Regulatory Commission, Washington, D.C.

DEAR DR. RUSCHE: The March 22 occurrences in dockets 50-259, 50-260, and 50-296 represent a serious failure to meet the intent of criterion 22, Appendix A, 10 CFR 50, therefore, pursuant to 10CFR 2.206, 2.202, 50.100, BPI hereby formally requests you take the following specific actions.

(1) Review and identify those operating nuclear power plants that not fully comply with IEEE-279 criteria for nuclear power plant protection systems and show cause why these plants should not be required to shut down until they are retrofitted to meet said criteria.

(2) Adopt regulation requiring shutdown of operating units at multi-unit facilities during periods when work on a unit under construction could com-

promise the integrity of the engineered safety features on an operating unit or units.

DAVID DINSMORE COMEY,
Director of Environmental Research, BPI.

ENCLOSURE 2

MAY 5, 1975.

Mr. DAVID DINSMORE COMEY,
Director of Environmental Research, Business and Professional People for the Public Interest, 109 North Dearborn Street, Suite 1001, Chicago, Ill.

DEAR MR. COMEY: Your March 24 letter asked, among other things, that those operating nuclear power plants that do not fully comply with IEEE-279 criteria for nuclear power plant protection systems be identified. As you know, the IEEE-279 criteria were initially published in August 1968 (as "Proposed Criteria"), and have been revised once since that time (in 1971).

Enclosure 1 is a list of those operating plants that have been evaluated by the NPC staff for compliance with the IEEE-279 criteria and found to be acceptable, and the date of issuance of the applicable IEEE criteria. Enclosure 2 lists the remaining eight operating plants that were evaluated for compliance against earlier staff criteria available at the time those plants were reviewed and approved for operation. These eight plants were all licensed prior to August 1968. Although some of the earlier criteria used by the staff in its evaluations of the reactor protection systems for the plants listed in Enclosure 2 were subsequently included in what is now IEEE-279, the degree of compliance with the 1968 or 1971 IEEE-279 criteria varies for these plants.

Your March 24 letter also requested, in view of the Browns Ferry fire, that the NRR review those plants that do not fully comply with the IEEE-279 criteria for nuclear power plant protective systems, and show cause why these plants should not be required to shut down until they are retrofitted to meet these criteria. As I discussed with you by telephone, such a review by NRC would initially involve a report by the involved licensees dealing with the issue that you have raised, and a subsequent evaluation and judgment by the staff. I would expect that these steps would take at least six months to complete and would involve considerable staff effort.

Pending completion of the ongoing NRC investigation and review of the cause and implications of the Browns Ferry fire, the NRC has taken actions through its Office of Inspection and Enforcement that we believe provide reasonable assurance that all operating plants, including those listed in Enclosure 2, can continue to operate without undue risk to the health and safety of the public. These actions are set forth in Enclosures 3 and 4 to this letter, and were taken in late March and early April of this year.

The NRC actions taken to date as a result of the Browns Ferry fire deal primarily with measures to reduce the likelihood of occurrence of serious fires at operating nuclear power plants. A related important issue is whether the criteria (such as those in IEEE-279) relevant to limiting the damage from such a fire, if one should occur, should be revised in light of the Browns Ferry occurrence. For this reason, one of the specific objectives of the ongoing NRC investigation and review is to determine whether or not such criteria should be modified or supplemented and, if so, to which plants these revised or supplemental criteria should be applied. The results of the Browns Ferry investigation by NRC's Office of Inspection and Enforcement are presently expected to be publicly available in July 1975. The schedule for completion of the technical review by the group under the direction of the Technical Advisor to the Executive Director for Operations has not yet been established.

In view of the foregoing, it appears possible that you may wish to modify the request in paragraph (1) of your letter of March 24. For example, you might wish to consider deferral of any further action by NRC on this portion of your request until the NRC's investigation and review of the Browns Ferry fire is completed, limiting this portion of your request to only some of the plants listed in Enclosure 2, or other alternative courses of action.

I would be pleased to discuss this matter with you at your convenience before we continue action on your request. If you need any further information, please let me know.

Sincerely yours,

EDSON G. CASE,
Deputy Director, Office of Nuclear Reactor Regulation.

Enclosures:

OPERATING NUCLEAR POWER REACTORS THAT WERE EVALUATED AGAINST THE CRITERIA CONTAINED IN IEEE STD 279 AND FOUND ACCEPTABLE

Docket No.	Plant	Applicable issue of IEEE-279	Operating license issuance date
50-219	Oyster Creek	1968	Apr. 9, 1969
50-220	Nine Mile Point 1	1968	Dec. 26, 1974
50-237	Dresden 2	1968	Dec. 22, 1969
50-244	Ginna	1968	Sept. 19, 1969
50-245	Milestone Point 1	1968	Oct. 7, 1970
50-247	Indian Point 2	1968	Oct. 19, 1971
50-249	Dresden 3	1968	Jan. 12, 1971
50-250	Turkey Point 3	1968	July 19, 1972
50-251	Turkey Point 4	1968	Apr. 10, 1973
50-254	Quad Cities 1	1968	Oct. 1, 1971
50-255	Palisades	1968	Mar. 24, 1971
50-259	Browns Ferry 1	1968	June 26, 1973
50-260	Browns Ferry 2	1968	June 24, 1974
50-261	H. B. Robinson 2	1968	July 31, 1970
50-263	Monticello	1968	Sept. 8, 1970
50-265	Quad Cities 2	1968	Mar. 21, 1972
50-266	Point Beach 1	1968	Oct. 5, 1970
50-267	Fort St. Vrain	1968	Dec. 21, 1973
50-269	Oconee 1	1968	Feb. 6, 1973
50-270	Oconee 2	1968	Oct. 6, 1973
50-271	Vermont Yankee	1968	Mar. 21, 1973
50-277	Peach Bottom 2	1968	Aug. 8, 1973
50-278	Peach Bottom 3	1968	July 2, 1974
50-280	Surry 1	1968	May 25, 1972
50-281	Surry 2	1968	Jan. 29, 1973
50-282	Prairie Island 1	1968	Aug. 9, 1973
50-285	Fort Calhoun	1968	May 24, 1973
50-287	Oconee 3	1968	July 19, 1973
50-289	Three Mile Island 1	1968	Apr. 19, 1974
50-293	Pilgrim 1	1968	June 8, 1972
50-295	Zion 1	1968	Oct. 19, 1973
50-298	Cooper Station	1968	Jan. 18, 1974
50-301	Point Beach 2	1968	Nov. 16, 1971
50-304	Zion 2	1968	Nov. 14, 1973
50-305	Kewaunee	1968	Dec. 21, 1973
50-306	Prairie Island 2	1968	Oct. 29, 1974
50-309	Maine Yankee	1968	Sept. 15, 1972
50-312	Rancho Seco 1	1968	Aug. 16, 1974
50-313	Arkansas One, Unit 1	[1] 1971	May 21, 1974
50-315	D.C. Cook 1	1968	Oct. 25, 1974
50-317	Calvert Cliffs 1	1968	July 31, 1974
50-321	Hatch 1	1968	Aug. 6, 1974
50-324	Brunswick 2	[1] 1971	Dec. 27, 1974
50-331	Duane Arnold	1968	Feb. 22, 1974
50-333	Fitzpatrick	[1] 1971	Oct. 17, 1974

OPERATING NUCLEAR POWER REACTORS THAT WERE EVALUATED AGAINST PRE IEEE 279 CRITERIA

Docket No.	Plant	Operating license issuance date
50-003	Indian Point 1	Mar. 26, 1962
50-010	Dresden 1	Sept. 28, 1959
50-029	Yankee Rowe	Dec. 24, 1963
50-133	Humboldt Bay	Aug. 28, 1962
50-155	Big Rock Point	Aug. 30, 1962
50-206	San Onoire 1	Mar. 27, 1967
50-213	Connecticut Yankee	June 30, 1967
50-409	LaCrosse	July 3, 1967

[1] The differences between the 1968 and 1971 version of IEEE-279 do not appear.

CABLE FIRE AT BROWNS FERRY NUCLEAR POWER STATION—IE BULLETIN NO. 75-04

DESCRIPTION OF CIRCUMSTANCES

Preliminary information from Tennessee Valley Authority regarding a fire which occurred on March 22, 1975, at their Browns Ferry site near Athens, Alabama, indicates that the fire was started as a result of construction activities. The fire resulted in the shutdown of two operating nuclear plants and made several safety systems inoperative, including systems normally used for decay heat removal during shutdown. The workmen were engaged in construction activities on a third unit not yet licensed for operation by NRC.

Initial information indicates that during the installation and testing of cable through-wall penetrations an open flame ignited a flammable material used in the penetration seals.

ACTION TO BE TAKEN BY LICENSEES:

The following actions are requested of selected Licensees with operating power reactor facilities and major construction activities at a common site:

1. Review your overall procedures and system for controlling construction activities that interface with reactor operating activities, with particular attention to the installation and testing of seals for electrical cables between compartments of the reactor building, e.g., control room to cable spreading room.

2. Review the design of floor and wall penetration seals, with particular attention to the flammability of materials.

3. Evaluate your procedures for the control of ignition sources which may be used for leak testing or other purposes in areas containing flammable materials.

4. Report to this office, in writing within 20 days of the date of this Bulletin, the results of your reviews or evaluations regarding items 1 through 3 above.

ACTIONS TO BE TAKEN BY LICENSEES MAY BE REVISED:

The actions requested by Licensees above may be revised as additional details of the Browns Ferry occurrence are available and evaluated by the NRC.

CABLE FIRE AT BROWNS FERRY NUCLEAR PLANT—IE BULLETIN No. 75-04A

The following material supplements and modifies IE Bulletin 75-04.

DESCRIPTION OF CIRCUMSTANCES

Additional, though still preliminary, information has become available related to the fire which occurred at the Browns Ferry Site on March 22, 1975. The fire started in the cable spreading room at a cable penetration through the wall between the cable spreading room and the reactor building for Unit 1. A slight differential pressure is maintained (by design) across this wall, with the higher pressure being on the cable spreading room side. The penetration seal originally present had been breached to install additional cables required by a design modification. Site personnel were resealing the penetration after cable installation and were checking the airflow through a temporary seal with a candle flame prior to installing the permanent sealing material. The temporary sealing material was highly combustible, and caught fire. Efforts were made by the workers to extinguish the fire at its origin, but they apparently did not recognize that the fire, under the influence of the draft through the penetration, was spreading on the reactor building side of the wall. The extent of the fire in the cable spreading room was limited to a few-feet from the penetration; however, the pressure of the fire on the other side of the wall from the point of ignition was not recognized until significant damage to cables related to the control of Units 1 and 2 had occurred.

Although control circuits for many of the systems which could be used for Unit 1 were ultimately disabled by the fire, the station operating personnel were able to institute alternative measures by which the primary system could be depressurized and adequate cooling water supplied to the reactor vessel. Unit 1 was shut down manually and cooled using remote manual relief valve operation and condensate booster pump, and control rod drive system pumps. Unit 2 was shut down and cooled for the first hour by the RCIC. After depressurization, Unit 2 was placed in the RHR shutdown cooling mode with makeup water available from the condensate booster pumps and control rod drive system pump.

ADDITIONAL ACTIONS TO BE TAKEN BY LICENSEES

1. Because the occurrence appears to have resulted from modifications being made to an operating unit, all power reactors with operating licenses should address the actions requested in Bulletin 75-04 as well as the actions described below.

2. Review your policies and procedures relating to construction or maintenance and modification work to assure that activities which might affect the safety of a unit in operation, including the ability to shut down and cool the unit are properly controlled. Your review should consider particularly your policy on deferring construction, maintenance or modification work on a unit

until a shutdown period except for emergency maintenance vital to continued safe operating or safe shutdown of the unit.

3. Review your policies and procedures to assure that for construction or modification and maintenance activities during plant operation, particular attention is given to the following areas:

(a) The degree of safety significance of affected and nearby cabling and piping.

(b) The use and control of combustible materials.

(c) The use and control of equipment that may be an ignition source.

(d) The assignment of personnel, knowledgeable of plant arrangement and plant operations, whose sole temporary responsibility is monitoring the safe performance of construction or maintenance and modification work, including attention to otherwise unattended areas adjacent to the work areas.

(e) Provision of installed or portable equipment to provide the monitoring personnel with prompt communication with the operating staff in the control room.

(f) Provision of adequate fire prevention and fire suppression equipment, installed or portable, for the following locations:

(1) Areas where work is being performed.

(2) Areas where occurrence of a fire has high safety significance, even though the probability of occurrence is relatively small.

(g) Recognition that fire, even one involving electrical equipment, may, if of sufficient intensity require water as the ultimate suppression medium.

4. Review your emergency procedures to assure that consideration for alternate methods for accomplishing an orderly plant shutdown and cooldown are provided in case of loss of normal and preferred alternative shutdown and cooldown systems for any reason (e.g. a fire). In this connection, assure that the minimum information necessary to assist the operators in such shutdown actions, the minium protection system actions required (e.g. scram) and the spectrum of alternative paths available to the operators to supply cooling water and remove decay heat dependent on plant conditions are included in your emergency procedures.

5. Report to this office, in writing, within 20 days of the date of this Bulletin, your schedule for review in each of the above areas.

6. Upon completion of your reviews provide the results of the review and the schedule for accomplishment of any revisions to your policies and procedures, and any proposed changes to the facility, and the date by which the changes are scheduled to be completed. If this latter date is more than 30 days after the date of the initial report, provide a monthly summary report detailing your progress in the review and/or proposed procedure or facility modifications. Reports requested by Bulletin 75-04 may be incorporated with the initial response to this Bulletin.

BUSINESS AND PROFESSIONAL PEOPLE FOR THE PUBLIC INTEREST,
Chicago, Illinois, June 16, 1975.

EDSON G. CASE,
Deputy Director, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, D.C.

DEAR MR. CASE: Thank you for your letter of May 5, 1975, responding to my March 24, 1975, letter.

We have reviewed Enclosure 2 to your letter, listing those operating nuclear power reactors that were evaluated against pre-IEEE-279 criteria.

We have no desire to tie up NRC staff with a review of all the plants listed in Enclosure 2, and if the actions taken by NRC pending review of the cause and implications of the Browns Ferry fire will assure adequate protection of the public health and safety, and if the results of the Hananer review are going to be forthcoming soon, we are willing to modify the request in paragraph (1) of my letter of March 24, 1975.

There are, however, two plants listed in Enclosure 2 which we believe ought to be treated separately. These are the two oldest plants, Indian Point 1 and Dresden 1, the operating license issue date for which are March 26, 1962, and September 28, 1959, respectively. Because of their early operating license review dates, it appears likely that these two plants may have the least compliance with IEEE-279. Moreover, of the eight reactors listed in your Enclosure 2, these two reactors are the closest to large metropolitan population centers (New York and Chicago, respectively).

In addition, both Indian Point 1 and Dresden 1 have rather poor operating records, both with respect to numbers of abnormal occurrences and also non-compliance with NRC regulations, and, therefore, we believe that it appears that there is a higher probability of a serious accident at these facilities than at the others listed in Enclosure 2.

Accordingly, as you suggest in the next to last paragraph of your letter of May 5, 1975, we will limit our request in paragraph (1) of my March 24 letter to a review by NRC, including a report by the involved licensees and an evaluation and judgment by the staff, of only some of the plants in Enclosure 2, namely, Indian Point 1 and Dresden 1. As you further suggest, we are willing to consider deferral of any further action by NRC with respect to other plants on that portion of our request until we have received the Hanauer investigation and review of the Browns Ferry fire.

Very truly yours,

DAVID DINSMORE COMEY,
Director, Environmental Research.

ENCLOSURE 4

U.S. ATOMIC ENERGY COMMISSION,
Washington, D.C., June 11, 1973.

Attn: Mr. WILLIAM E. CALDWELL, *Vice President,*
Consolidated Edison Co. of New York, Inc.,
4 Irving Place, New York, N.Y.

GENTLEMEN: As you know, we have under review your application for a full-term operating license for Unit No. 1 at the Indian Point Nuclear Generating Station. Our review includes consideration of the need for backfitting Unit No. 1 to provide substantial, additional protection which is required for the public health and safety. Our review of portions of the plant protection system has been completed, and we find that modifications to the system are necessary.

In the event of an accident jeopardizing the integrity of the core, Section 6 (Accident Analysis) of the Safety Analysis Report for Core—indicates that the plant protection system would function to terminate the accident before fuel and fuel clad melting could occur in the hot-test channel and result in the release of fission products to the reactor coolant system. Although containment is provided to retain fission products should they be released from the reactor coolant system, protection systems for modern nuclear plants are designed to meet the single failure criterion of IEEE Std. 279-1968, "Criteria for Nuclear Power Plant Protection Systems", and the General Design Criteria of Appendix A to 10 CFR Part 50. By letters dated March 31, 1972, and April 28, 1972, from your attorneys, you submitted in support of your application additional information which compares the plant protection system to IEEE 279 as published August 30, 1968, and to General Design Criterion 20, "Protection Systems Redundancy and Independence", and General Design Criterion 21, "Single Failure Definition", as published in the *Federal Register* on July 11, 1967. Our review is based on this information and other information in the public record.

We find that improvements in the plant protection system, including circuits for isolating containment and for protection against steam line breaks, will provide substantial, additional protection for the health and safety of the public, and that such improvements are required for the health and safety of the public. Therefore, pursuant to Section 50.109, "Backfitting", of 10 CFR Part 50, you are hereby directed to submit, within six months from the date of this letter, the following:

1. Details of proposed modifications that are necessary to make the design of protection circuits essential to safety conform with the requirements of Section 4.2, "Single Failure Criterion," of IEEE Std. 279-1968.

2. Details of proposed modifications and analyses of such modifications that are required to assure that dc electric power systems essential to safety meet the single failure criterion.

3. Your detailed analyses of all portions of the protection system including the proposed modifications which show that the modified system meets the single failure criterion.

4. Results of analyses and/or tests which demonstrate that all instrumentation and electric equipment essential to safety can function in its environment during and following an accident. If satisfactory results are not obtained, describe the modifications necessary to assure that all instrumentation and electric equipment essential to safety can function in its accident environment.

Further, you are hereby directed to complete within one and one-half years from the date of this letter, all such modifications and tests necessary to conform to the requirements of the single failure criterion.

In developing the proposed modifications necessary to satisfy the requirements specified in 1. and 3. above, the following concepts should be applied in your analyses of the modifications.

1. The types of single failures to be considered are:

a. failure to any single component or module or a single circuit fault. (Circuit faults include short circuits, open circuits, grounds, and the application of any available ac or dc potential.)

b. multiple component failures and/or circuit faults that can be predicited to occur as a result of a single event within or external to the system.

2. Each single failure is considered to occur coincident with any and all combinations of nondetectable failures. (Nondetectable failures are failures that cannot be detected by periodic tests, anomalous indications, or alarms.)

3. If an event that results in the need for protective action can cause failures in the protection system or if failures in circuits common to control and protective functions can result in the need for protective action, it must be shown that no single failure, in addition to the event-caused failures or the failures in the common portion of the protection system, will prevent the protection system from performing the required protective action(s).

Based on our review of portions of the protection system, the following suggested modifications are examples of the kind of modifications that might be made to the protection system in order to satisfy the single failure criterion:

(a) install an additional scram bus similar to the existing scram bus;

(b) connect the two scram buses to the scram solenoid valves in a manner which ensures that either bus is capable of deenergizing the solenoid valves regardless of whether the other bus is operable or failed;

(c) disconnect one of the nuclear instrumentation logic networks from the existing scram bus and connect it to the additional similar scram bus;

(d) transfer either the high pressurizer pressure or the high primary pressure trip relay contacts from the existing scram bus to the additional similar scram bus;

(e) install additional independent high primary temperature, low primary pressure, and low boiler feedwater pressure instrumentation for use with the additional similar scram bus;

(f) transfer one flux/flow computer from the existing scram bus to the additional similar scram bus and install additional flux averaging amplifiers, one for use with the additional similar scram bus and one for use with the automatic reactor control system;

(g) modify the existing scram bus and design the additional similar scram bus (or design their connections to the scram solenoid valves) in a manner that permits testing individually the two scram buses during reactor power operation; and

(h) provide redundant initiating devices in the criteria for protecting against the effects of a steam line break and for isolating containment.

If we can provide additional guidance with regard to your preparation of these required submittals, please do not hesitate to call (Mr. Roger Woodruff is the Regulatory Project Manager).

You should appreciate that, in addition, there are other technical issues related to your pending application for a full term operating license for Indian Point Unit No. 1 now under review by the AEC staff. In this connection, we are concerned with the frequent delays that have been experienced in your responses to our requests for additional information on these issues. We believe that this matter warrants prompt management attention and request that you take the necessary steps to remedy this situation.

Sincerely,

A. GIAMBUSSO,
Deputy Director for Reactor Projects,
Directorate of Licensing.

Enclosure:

Safety Evaluation by the Directorate of Licensing Consolidated Edison Company of New York, Inc.—Indian Point 1: Docket No. 50-3

Background

As part of our review of Consolidated Edison Company's application for a full-time operating license for Unit No. 1 at the Indian Point Nuclear Generating Station, we have completed a review to determine the need, if any, for backfit of the plant protection system. Car position and the reasons for it are discussed in the following sections.

Description of the Plant Protection System

Principal features of the Unit No. 1 plant protection system are a scram bus, two logic networks, and circuits for isolating containment and for isolating steam lines.

When values of important plant parameters exceed their limiting safety system settings, relays in the scram bus automatically cause insertion of the control reds by deenergizing a solenoid valve in the hydraulic system for each control red. Principal relays are controlled by instrumentation which monitors reactor coolant temperature and pressure, boiler feedwater pressure, and the ratio of neutron flux to reactor coolant flow.

Neutron flux sensors monitor reactor power and provide input signals to the logic networks. These networks supply electrical power to the scram bas and are intended to interrupt electrical power to the scram bus if reactor power exceeds the limiting safety system setting.

Multiple input signals to the logic networks are provided by each of 18 neutron flux sensors. The signals are connected in sets of three to each of nine "two-out-of-three" modules in each network. If any two signals to a "two-out-of-three" module exceeds the limiting safety system setting, the module initiates a scram signal. The outputs of the nine "two-out-of-three" modules are connected to a "one-out-of-nine" module. On receipt of a scram signal, a solid state switch in this module "opens" interrupting power to two trip actuator amplifiers. A solid state switch in each of the parallel trip actuator amplifiers then "opens", disconnecting power from the scram bus. Because the two logic networks are connected to the scram bus in parallel, both logic networks must function to deenergize the scram bus.

Three of the 18 neutron flux sensors also provide signals to a flux averaging amplifier. The output from this amplifier and signals from reactor coolant flow sensors are connected to two computers which calculate the ratio of flux to flow. If the value of the ratio is excessive, each computor demands that each of two pairs of parallel relay contacts opens in the scram bus. Other pairs of parallel relay contacts in the scram bus are opened by sensors monitoring boiler feedwater pressure, reactor cooling system pressure, and reactor cooling system temperature.

To prevent overcooling of the reactor core in the event of a steam-line-break accident and to limit the reactivity and power transients which would result, circuitry is provided to close the steam stop valve in the affected loop if a reactor scram occurs in coincidence with low steam pressure. A pressure drop across the steam stop valve in coincidence with scram causes closure of the stop valves in the associated reactor cooling loop. These functions are initiated by a single switch or relay.

Containment is partially or completely isolated automatically in the event of high containment pressure, high radiation, and very low pressurizer level. For each case, isolation is initiated by a single relay.

Safety Evaluation

IEEE 279 establishes several criteria for the design of plant protection systems. However, the fundamental aspects of IEEE 279 is the single failure criterion which is intended to assure that the protection system will function in the event that any single failure or fault exists somewhere in the protection system. Our safety evaluation is based on this fundamental aspect of IEEE 279.

With regard to the scram bus, application of stray power to it can disable all relays and solid state switches from the point of application of stray power back to, and including, the trip actuator amplifiers, thus, defeating the scram functions associated with the relays and solid state switches. Other kinds of single failures, e.g., failed components, short circuits, open circuits, and grounds, can also disable relays in the scram bus and cause less of specific scram functions.

In the logic networks, random failure of any one of six solid state switches in the "closed position" will disable the high neutron flux scrams. In the circuits for isolating containment and for isolating steam lines, single failures of initiating mechanisms would cause loss of protective action.

Although we have not reviewed all portions of the plant protection system, it is clear that single failures can be expected to inhibit the plant protection system and that backfitting the plant protection system will provide substantial, additional protection which is required for the health and safety of the public.

Backfit and Information Requirements

Pursuant to 10 CFR Part 50, Section 50.109, we are requiring that Consolidated Edison submit, within six months from the date of this evaluation, proposed modifications to the plant protection system which will provide significantly greater assurance that the plant protection system will perform its intended function should any of the following kinds of single failures occur: failed components, short circuits, open circuits, grounds, and application of any available ac or dc potential. The proposed modifications must include capability at power for conducting surveillance on the plant protection system to assure that it is capable of performing its intended function. Further, we are requiring that Consolidated Edison complete approved modifications of the plant protection system within one and one-half years from the date of this evaluation.

R. W. Woodruff,
Operating Reactors Branch No. 1,
Directorate of Licensing.
Robert J. Schemel,
Chief, Operating Reactors Branch No. 1,
Directorate of Licensing.

APPENDIX 14

CORRESPONDENCE: TVA/NRC

Letter Dated September 2, 1975, TVA to NRC, Responding to Violation Letter of July 28, 1975

TENNESSEE VALLEY AUTHORITY,
Chattanooga, Tenn., September 2, 1975.

Mr. NORMAN C. MOSELEY,
Director, Regional Office,
U.S. Nuclear Regulatory Commission, Atlanta, Ga.

DEAR MR. MOSELEY: We have reviewed the material provided with your letter to J. E. Watson dated July 28, 1975, (50-259/75-1, 50-260/75-1), related to the Browns Ferry fire. In accordance with your request, we have prepared responses to the alleged infractions in the Notice of Violation and items mentioned in Areas of Concern, Appendix A and Appendix B respectively, to your July 28 letter. In addition, we have prepared responses related to the Conclusions and Summary of Findings of Facts included in your Regulatory Investigation Report. The TVA responses are enclosed. With reference to Section 2.201, Items (1), (2), and (3), in the fourth paragraph of your letter, you will note that except in the cases of those allegations TVA has denied, the additional administrative procedures have already been placed into effect.

In addition to revisions in administrative controls, TVA has, as you know, also committed to numerous revisions to the Browns Ferry plant design to minimize the likelihood of a similar fire. These are discussed in great detail in TVA's "Plan for Evaluation, Repair, and Return to Service of Browns Ferry Units 1 and 2 (March 22, 1975, Fire)" dated April 13, 1975. You have been provided copies of all such documents and correspondence and we assume I & E is familiar with the status of the plant.

Much of the controversy among investigators of the incident hinges on the timing of the use of water as an extinguishing agent on the burning electrical cables. The plant operators were concerned that additional components would be rendered inoperable if their cables were shorted out by spraying water. TVA has no general policy prohibiting the use of water on electrical fires, and we believe that our proposed design changes to the plant will enable water to be used in the future to extinguish cable tray fires without compromising plant safety.

The Browns Ferry Nuclear Plant is currently shut down with both reactor cores unloaded. TVA is pursuing as diligently as possible all actions necessary to put the plant back into operation as soon as possible, and this activity is requiring major effort by TVA management and staff. Because of these conditions, and because we believe the seriousness of the alleged infractions warranted more time for a thorough and measured response, we asked for additional time beyond September 2 (which you did not grant) to allow us to respond more fully. We hope that any future correspondence on this matter will consider the status of the plant and the overall recovery plan.

We believe our overall corrective action program has reduced to an extremely low level the probability of the occurrence of another fire at Browns Ferry, and has reduced to even lower levels the probability that a fire would result in a significant consequence.

Very truly yours,

J. E. GILLELAND,
Assistant Manager of Power.

Enclosure.

ENCLOSURE: NRC JULY 28, 1975, DOCKET NOS. 50–259, 50–260

NOTICE OF VIOLATION

1. FAILURE TO COMPLY WITH 10 CFR 50.59

Items appearing to be in noncompliance with 10 CFR 50.59, "Changes, Tests and Experiments," as indicated below:

a. 10 CFR 50.59, requires, in part, that records be maintained of changes to the facility to the extent that such changes constitute changes to the facility as described in the Safety Analysis Report. It further requires that these records shall include a written safety evaluation which provides the bases for the determination that the change does not involve an unreviewed safety question. The Browns Ferry FSAR Section 5.3.3.5 specifies, in part, that all electrical penetrations are sealed with sealant around conductors.

Contrary to this requirement, a safety evaluation was not made of the "change to the facility as described in the Safety Analysis Report" which was constituted by operation of the reactor with containment penetrations unsealed while concurrently sealing and testing the penetrations.

This infraction had the potential for causing or contributing to an occurrence related to health and safety.

TVA Response

The Browns Ferry Nuclear Plant Final Safety Analysis Report (FSAR) was written to describe a completed 3 unit plant. To varying degrees, the FSAR describes most all plant features regardless of their nuclear safety significance.

Before issuance of the operating license for each unit, the NRC (then Regulatory Staff of the AEC) made findings that, among other things, the facility construction was substantially complete in conformity with the construction permit and the rules and regulations of the Commission. The Browns Ferry units were licensed and permitted to operate one unit at a time while construction continued on the remaining units. This was not considered to be a violation of the operating license or a violation of 10 CFR 50.59 since it was the basis upon which the licenses were issued.

TVA has made written safety evaluations in accordance with 10 CFR 50.59 before making changes to completed portions of the facility but has not been in the practice of making safety evaluations for work on components where construction work had never been completed except where a safety-related interface was recognized to exist. The section of the FSAR referenced in the allegation is a description of the secondary containment system describing not only the seals around electrical cables but all other building seals which contribute to the integrity of the secondary containment. Unsealing penetrations to add additional cable and resealing them upon completion was not considered as a change to the facility design nor was it recognized to have safety significance as long as the limiting conditions for operation of the secondary containment described in the technical specifications were maintained.

To reduce the potential for this type occurrence in the future, TVA has made procedural changes to require additional review before activities are authorized. The changes in procedures which have been presented to NRC are discussed in "Plan for Evaluation, Repair, and Return to Service of Browns Ferry Units 1 and 2 (March 22, 1975, Fire)," a copy of which has been provided you.

2. FAILURE TO COMPLY WITH TECHNICAL SPECIFICATIONS

Items appearing to be in noncompliance with the facility Technical Specifications, as indicated below:

a. The Technical Specifications, Sections 6.3.A and 6.3.B state, in part:

"A. Detailed written procedures, including applicable check-off lists covering items listed below shall be prepared, approved and adhered to * * *.

"4. Emergency conditions involving potential or actual release of radioactivity * * *.

"5. Preventive or corrective maintenance operations which could have an effect on the safety of the reactor.

"6. Surveillance and testing requirements * * *.

"B. Written procedures pertaining to those items listed above shall be reviewed by PORC and approved by the plant superintendent prior to implementation * * * Such changes shall be documented and subsequently reviewed by PORC and approved by the plant superintendent."

(1) Contrary to these requirements, the leak testing, sealing and inspection of the penetrations were being accomplished; but detailed written procedures approved by the plant superintendent and reviewed by PORC had not been developed for the control of this work.

TVA Response

It was not recognized that this work on the penetrations was of such a nature as to have an effect on the safety of the reactor, on surveillance or testing requirements, or had the potential for release of radioactivity; therefore, leak testing, sealing, and inspection of the penetrations were not accomplished by detailed, approved, written procedures as would be required for safety-related items. Procedural changes have been made which require greater in-depth review of all significant activities prior to their authorization. In addition, procedural changes have been made which require review of plant activities by the plant Quality Assurance Supervisor to verify that the activities are supported by approved instructions necessary for their performance.

(2) Contrary to these requirements, persons discovering the fires on March 20 and March 22, 1975, did not adhere to the provisions of the Emergency Procedure in that they did not initiate the fire alarm.

TVA Response

The construction workers first attempted to extinguish the fire, whereas the Browns Ferry Nuclear Plant Emergency Procedure specifies that the fire alarm be sounded first. The officer reporting the fire telephoned the shift engineer's office rather than calling either of the numbers listed in the procedure. To correct this condition, greater emphasis on emergency procedures has been and is being stressed as part of the Browns Ferry Nuclear Plant employee orientation program and fire drills accompanied by instruction over the public address system are being utilized as a means of training all employees in the proper action to take in case of fire.

(3) Contrary to these requirements, the Browns Ferry Emergency Procedure was not adhered to in that the Shift Engineer did not delegate onscene responsibility for fire fighting to an assistant shift engineer when he departed the fire area.

TVA Response

TVA denies this allegation and believes that the Browns Ferry Emergency Procedure was followed. The shift engineer and assistant shift engineer both reported to the scene of the fire. The shift engineer was recalled to the control room by the reactor operator leaving the assistant shift engineer in charge of fire-fighting activities. The assistance shift engineer was automatically in charge when the shift engineer was called away, the assistant shift engineer knew he was in charge, and his actions support that he assumed this responsibility. The NRC investigation report substantiates that the assistant shift engineer coordinated fire-fighting efforts, evacuated people from the area, made reports to the shift engineer, and placed the call for offsite assistance.

(4) Contrary to these requirements and the requirements of Browns Ferry Standard Practices Manual which specify, in part, in Standard Practice BFS3 that:

"Plant fire protection systems shall be fully operational at all times. Removal of a plant fire protection system from service for any reason other than as required in a test procedure requires the approval of the plant superintendent. Removal of a system from service for more than seven days requires a review of PORC."

The fire protection system for the cable spreading room was not fully operational in that metal plates had been installed under the glass in the manual stations during the construction of the plant and had not been removed. The approval of the installation of the plates had not been documented prior to or subsequent to the issuance of the operating license and the installation had not been reviewed by PORC. Additionally, the CO2 manual-automatic initiation system had been electrically disabled by the construction workers without documented approval of the Plant Superintendent.

This infraction had the potential for causing or contributing to an occurrence related to safety.

TVA Response

TVA denies this allegation; the CO_2 system was not removed from service. The presence of the metal plates under the breakout glass used in the mechanical valve method did not prevent manual electric actuation nor slow application of the CO_2. The CO_2 system in the spreading room can be actuated manual electrically or manual mechanically. The system does not have automatic actuation. Of the two means available to actuate manually the system, the operator used the manual electrical method, which is the normal method and the faster method. The method used (in accordance with established instructions) was significantly faster than the mechancal valve method. As a safety precaution, the metal plates had been installed to prevent inadvertent operation of the CO_2 system in the spreading room while workers were present. The condition of the CO_2 system was given consideration and determined not be contrary to the standard practice requirements in that the CO_2 flooding system was operational and not removed from service. In addition, written plant instructions were in effect describing the condition of the system and specifying its mode of operability.

3. FAILURE TO COMPLY WITH APPENDIX B TO 10 CFR 50

Items appearing to be in noncompliance with Appendix B to 10 CFR 50, "Quality Assurance Requirements for Nuclear Power Plants and Fuel Reprocessing Plants," as indicated below:

a. Criterion XVI of Appendix B to 10 CFR 50 and the related commitments in the FSAR, Appendix D.4, "Operational Quality Assurance Program Plan," Section D.4.2.4.7 specifies, in part, that measures be established to assure that conditions adverse to quality are promptly identified and corrected; that measures assure that causes of conditions be determined and action taken to preclude repetition; and that the corrective actions are documented and reported to appropriate levels of management.

Contrary to these requirements, during the penetration sealing operations, the conditions adverse to quality were not promptly identified and corrected; the causes of conditions were not determined and actions taken to preclude repetition; and the required documentation was not supplied in the two instances that fires were reported to management.

This infraction had the potential for causing or contributing to an occurrence related to safety.

TVA Response

The minor fire that occurred in the sealing material on March 20, 1975, had been discussed in the plant management staff meeting on March 21 and was in the process of being followed up by supervisory personnel when the major fire of March 22 occurred. Plant administrative instructions for reporting conditions adverse to quality have since been revised and are also included as a part of the general employee training program.

b. Criterion X of Appendix B to 10 CFR 50, requires, in part, that a program for inspection of activities affecting quality be established to verify conformance with documented instructions, procedures, and drawings; and that persons assigned the responsibilities for such inspections shall be independent of individuals directly responsible for work performance. Related commitments are spelled out in the FSAR, Appendix D.4., Sections D.4.2.3.1 and D.4.2.1.1., respectively.

Contrary to these requirements, inspections of the sealing of cable penetrations were not conducted so as to assure conformance with drawings; and inspectors were involved in the work activities for which they had inspection responsibilities.

This infraction had the potential for causing or contributing to an occurrence related to safety.

TVA Response

The response to item 2.a(1) is also applicable to this item. Since the fire, procedures have been issued applicable to all penetrations under repair or construction. These procedures provide for instruction of craft personnel engaged in penetration sealing and flameproofing and provide for independent inspection by engineers. The inspectors are prohibited from engaging in the work.

c. Criterion XVIII of Appendix B to 10 CFR 50, and the related commitments set forth in the FSAR, Appendix D.4, "Operational Quality Assurance Program Plan," specifies, in part, that a comprehensive system of planned audits be carried out to assure compliance with all aspects of the quality assurance program.

Contrary to these requirements, a review of the records of the audits conducted at Browns Ferry and discussions with responsible individuals indicated that no audits had been conducted of the penetration installation.

This infraction had the potential for causing or contributing to an occurrence related to safety.

TVA Response

Audits are not required for non-safety-related items, therefore, audits were not conducted of the records pertinent to penetration installation because their safety significance had not been recognized. Since the fire, a detailed audit has been made of all penetrations and work of this nature has been added to the quality assurance audit program.

d. Criterion III of Appendix B to 10 CFR 50, and the related commitments set forth in the FSAR, Appendix D.2, "TVA Quality Assurance Plan for Design and Construction," Section D.2.4.3.4., specifies, in part, that certain basic design drawings, such as single line diagrams, are reviewed to determine that they meet the design bases, design criteria and other design input requirements.

The FSAR, Amendment 25, "Response to AEC Question 7.5," states, in part, that cables for the Engineered Safeguards Systems are separated into two redundant divisions (Division I or Division II) such that no single credible event could damage the cables of redundant counterparts. This section further states that power cables from the 4160-Volt Shutdown Boards are installed in separate conduits. It further states that the electric circuits of one of the two loops of the Core Spray System including the pump motors and electrically operated valve, are in Division I; and the circuits of the other loop are in Division II. Additionally, it states that the electric circuits associated with pumps A and C, and their valves, of the LPCI sytsem are in Division I; and the electrical circuits of pumps B and D, and their valves are in Division II.

(1) Contrary to this requirement, the power cable supplying 480 Volt Shutdown Board 1B from 4KV Shutdown Board C (Division II) is routed in the same tray as the power cable supplying 480 Volt Shutdown Board 2A from 4KV Shutdown Board B (Division I).

TVA Response

The routing indicated in the finding was not a violation of the separations criteria and the electrical design was reviewed and approved by the NRC. This finding is an oversimplification of a highly technical point. The original design did not assign divisional classification to 4-kV cables between the 4-kV shutdown boards and the 4-kV/480-volt transformers. These cables were assigned an IE designation. Separation requirements for IE cables are that the normal and alternate feeders for the same board will not be run in the same tray. The design was analyzed to and met the single failure criterion.

(2) Contrary to this requirements, RHR Pump 1C and Core Spray Pump 1C are supplied from 4KV Shutdown Board B, and their associated valves are supplied from 4KV Shutdown Board C. Shutdown Board B is in Division I and Shutdown Board C is in Division II.

This infraction had the potential for causing or contributing to an occurrence related to safety.

TVA Response

The IE report contains a mistake and therefore erroneously concludes that separations criteria were violated. Core Spray Pump 1C is supplied from 4-kV

Shutdown Board B, as stated, but power for the valves associated with this core spray pump is supplied from 4-kV Shutdown Board A, not Board C. Power for the pump and its associated valves is taken from Division I.

The RHR pumps have three modes of operation, namely torus cooling, shutdown cooling, and low-pressure coolant injection (LPCI). RHR Pump 1C is supplied from 4-kV Shutdown Board B in Division I as stated. In the first two modes of operation, power for the associated valves is supplied from 4-kV Shutdown Board A, which also is in Division I. In the LPCI mode, the injection valve (FCV 74-53) associated with this pump receives power from 480-volt Reactor MOV Board 1C which receives its normal power from 4-kV Shutdown Board C through 480-volt Shutdown Board 1B. This 480-volt shutdown board supplies Division II loads. Reactor MOV Board 1C has an automatic transfer capability upon loss of normal power, and would transfer from 480-volt Shutdown Board 1B to 480-volt Shutdown Board 1A which is supplied from 4-kV Board A. This 480-volt shutdown board supplies Division I loads. The automatic transfer feature of 480-volt Reactor MOV Board 1C permits the power supplies to be arranged as described without violation of separations criteria. This design feature was received and approved by NRC.

Appendix B—Areas of Concern

1. Construction personnel were involved in work using open flame but had not been given fire fighting training nor did they have personnel with this training assigned to their area.

TVA Response

Fire fighting in an operating unit is provided by trained plant operations personnel. Construction personnel are not required to fight fires in an operating unit and therefore do not receive fire fighting training. Procedures have been changed such that when working in an operating unit area, they do receive orientation in emergency procedures, including reporting of fires. Since the fire, TVA has issued written instructions prohibiting the use of any flame at Browns Ferry to check for airleaks. In addition, a procedure has been written establishing control over the use of cutting, welding, and open flames in operating plants. This procedure requires a physical survey of the areas where the work is to be performed and the establishment of safeguards including additional fire fighting equipment and a fire watch prior to any work involving open flame or welding.

2. Construction personnel working in the plant were not familiar with the plant emergency procedures, and some were not familiar with the plant layout.

TVA Response

The response to alleged violation 2.a.(2) is also applicable to this concern.

3. Operations personnel on shift were not fully aware of the ongoing construction activities.

TVA Response

The shift engineer o. duty was aware of the ongoing construction activities through the information contained in the workplans. A copy of each workplan is held by the shift engineer, and all work on transferred equipment is scheduled with him. Because the area where the fire was initiated is normally kept locked, the shift engineer's approval was required prerequisite to entry. Since the fire, procedural changes have been made requiring workplans to contain a more detailed description of the work to be performed. In addition, all workplans now receive supervisory and quality assurance review, are listed on a daily activity sheet, and their performance requires authorization of the plant superintendent.

4. Plant operations personnel and Public Safety Services personnel at the plant were not familiar with the reactor building fire extinguishing equipment in that they did not know that the nozzle on the hose was specifically designed for fighting electrical fires.

TVA Response

Plant operations personnel who have responsibility for fire fighting are instructed in the use of fire extinguishing equipment. Fog-type nozzles are utilized in our fire training classes, and an individual would only have to look at the nozzle to see the type it was. The only statement we can find to support the conclusion that personnel were not familiar with the fire hose nozzle is in the I&E

report where an operator in his statement said it worked more like a shower head than a fog nozzle.

5. Emergency breathing apparatus appears not to have been properly maintained.

TVA Response

TVA contends the emergency breathing apparatus was well maintained. Plant procedures for the care of this equipment were in effect and were being used. Some air cylinders which were in the shop for reconditioning and were specifically marked were inadvertently brought to the scene of the fire, but they were not used and did not endanger personnel safety.

6. Certain of the operations personnel as well as the construction personnel were not familiar with use of the breathing apparatus.

TVA Response

Plant operating personnel are responsible for fire fighting in an operating unit. Construction personnel are not normally trained in the use of air-breathing equipment. This is a part of the training provided operations personnel and the effectiveness of this training can be adjudged by the total reliance on air-breathing equipment by employees engaged in fire fighting and other activities associated with safe reactor shutdown without fatality or significant personnel injury.

7. There was apparently no attempt by CECC management to obtain expert advise on methods for fighting the fire.

TVA Response

The CECC is not designated the responsibility for advising on fire fighting in the Radiological Emergency Plan. The DECC had as part of their staff considerable expertise for judging the probability of short circuiting vital equipment by using water. The use of water was discussed several times throughout the fire with the Division Emergency Control Center (DECC). The main objectives of TVA's efforts on March 22 were to ensure the safety of the reactors and to minimize personnel injury. The report fails to recognize these points in the repeated implications that water should have been used on the fire at an earlier time. The plant superintendent made the conscious decision not to use water (fog, foam, or straight) because of the possibility of shorting circuits and further degradation of the plant to a condition that would have been more difficult to control. Reactor safety concerns under the circumstances took precedence over extinguishing a localized fire. Even in retrospect and in the light of nearly five months of engineering evaluation, this decision still stands as a correct decision. Of course, water was used to extinguish the fire after a more stable condition had been reached.

8. A design review subsequent to the fire revealed two cable trays with cables installed in excess of the design criteria. The practice of the construction forces of abandoning cable in-place cou'd result in additional cable trays being overfilled.

TVA Response

The cases indicated do not constitute violations of the cable tray fill criteria. Question 7.5 of May 22, 1971, requested that TVA provide the criteria and their bases which establish the minimum requirements for preserving the independence of redundant reactor protection systems, engineered safety feature systems, and Class 1E Electrical Systems through physical arrangement and separation, and assure minimum availability during any design basis event. In answering this question, we included a statement that cable trays carrying control and signal cables are loaded to a maximum of 60 percent of the cross-sectional area of the tray. This answer applied to the safety related cables referenced in the question.

The 60 percent guideline figure, which is the ratio of the cross-sectional area of the cable to the cross-sectional area of the tray, is obtained from past experience wherein it had been determined that a cable tray would be approximately level full at this point. Physical space for cables was the primary consideration because the weight limitation and electrical current considerations contained ample margin for filling to the worst case. If the cable tray became full before the 60 percent guideline was reached because of cable configuration in the tray no additional cables were routed in that tray. The same general guideline for non-safety-related cables was also followed although it is not specifically addressed in the FSAR.

Because of certain factors such as number of various sizes of cable or special installations in trays, it was possible that some trays would not be full when the 60 percent value was reached. In some cases, the 60 percent guideline was exceeded up to the point of the tray being level full by installing cables in these trays with special care. Although two cable trays are referenced, one is actually a continuation of the other.

9. The Browns Ferry Emergency Procedures and Standard Practice Manual contain errors and inconsistencies relating to emergency response. The Standard Practice Manual, BFA-34, permits the DPP Coordinator to review work plans for safety significance and does not require further review of his evaluations. This could result in a Technical Specification violation.

TVA Response

This item appears to contain a clerical error in report preparation in that the statements do not seem to be related. However, we believe one concern intended is that plant procedures contained two different phone numbers for use in reporting a fire. One phone number initiates the fire alarm and is a standard number used in all TVA operating plant phone systems. The other number on the construction phone system rings the fire marshall responsible for fire fighting on units under construction and before the operating license is received. Dialing either on the appropriate telephone system would result in prompt reporting of a fire. However, instructions existing within the plant operating area have been revised to delete the phone number of the construction fire marshall. The concern related to work plan review is addressed in the TVA response to alleged infraction 2.a.(1).

10. There was apparently no attempt by the CECC director to direct the operations of the center. The plant superintendent was involved in extensive communications with the center rather than in directing the plant fire fighting and recovery operations. As a consequence, the efforts at both locations were not coordinated, management personnel were not provided with accurate information and their actions were relatively ineffective.

TVA Response

TVA denies this allegation.

The TVA Radiological Emergency Plan (REP) sets forth the policies; purposes; delegations; standards; guidelines; and, where feasible, specific instructions necessary for TVA to discharge its responsibilities during a radiological emergency in order to comply with pertinent directives applicable to the protection of TVA personnel, plants, and properties and the health and welfare of the general public.

The general delegation of responsibility among the various emergency staffs as outlined in the Browns Ferry Nuclear Plant Radiological Emergency Plan are as follows:

A. *Central Emergency Control Center (CECC)*

Ensure that the various staff members man their positions.

Notify and inform other Government agencies.

Establish communications with Division Emergency Director and Environs Emergency Director.

Protect the public.

Gather offsite information.

Review with Division Emergency Director measures taken to control emergency.

B. Division of Power Production Emergency Center (DECC)

Receive reports from Site Emergency Director.

Evaluate and make recommendations.

Ensure that suport efforts are adequate.

C. Site Emergency Staff

Assume responsibility for emergency.

During the Browns Ferry fire, these responsibilities were carried out as designated. There was, according to the logs of various staffs, information and direction given between the individuals involved.

Although the alternate CECC director did not realize immediately on his arrival that he was acting as director at the CECC, his logs reveal that he, upon notification (1604 EDT), contacted the Environ Director and directed him to notify the Alabama Department of Public Health. This delay in recognition did not slow down the Radiological Emergency Plan from being implemented.

Upon his arrival at the CECC (1625 EDT), the alternate CECC director was briefed by the director, Division Emergency Center, on plant status. The public information officer was briefed. NRC was called at 1703 EDT. Personal contact with the Alabama Department of Public Health was made at 1712 EDT. In brief, coordination was evident and all divisions of responsibility were being carried out. This is evidence of coordination of positive leadership.

The focus of communications with other agencies centered on plant and environmental conditions. We do not believe this can be considered a misdirection since one directly affects the other. Since data provided indicated no radioactive release, the various agencies requested information on plant status and used this information to assess the potential for environmental release.

The status of the fire at Browns Ferry was discussed numerous times according to the Division Emergency Center log and the plant log. The information was not always timely because the fire was reported out and then would flare up again. This fact may have caused some confusion, but communications on reporting the conditions of the fire were successful even though other plant problems took precedence at times.

As pointed out in preface statements, fire fighting efforts were the responsibility of the Site Emergency Director with advice from the division emergency center. At no time was control of the fire beyond the scope of the resources already under the direction of the plant superintendent (site emergency director). There is no basis for stating that CECC guidance was inadequate.

The plant superintendent could not have personally directed all fire fighting activities nor does good management technique suggest that he should. Had he attempted to do so he would not have been able to carry out other specified responsibilities which have a much broader impact on reactor safety and the health and safety of the public. The superintendent was not engaged in *extensive* communication with the center, and this communication did not impair the superintendent's performance of his duties and responsibilities.

11. Although plant procedures specify that the Unit operator has the authority to shut down the reactor in the event of an emergency, his responsibility to effect a shutdown promptly has not been defined by management.

TVA Response

TVA denies this allegation.

Plant instructions specify that the reactor operator is responsible for the operation of his unit and give him authority to shut it down when he deems its safety is in jeopardy. Further, his job description specifies that, in emergencies where there is not time to get advice from his supervisor, he may operate on his own initiative to correct existing conditions or to save life or property. It is impractical and probably impossible to define in detail all conditions for which prompt shutdown is desirable.

CONCLUSIONS

A. General

1. The radiological impact on the public, plant personnel or the environment resulting from this fire was no more significant than from the routine shutdown of the reactors.

TVA Response

This finding is correct. Priority efforts at the plant were directed towards the primary objectives of ensuring safety and minimizing personnel injury and these objectives were achieved.

2. Both Unit 1 and Unit 2 cores remained adequately cooled throughout this occurrence. Although some systems were rendered inoperable, plant operators were successful in bringing into operation alternate means to cool the reactors.

TVA Response

This finding is correct. Priority efforts at the plant were directed towards ensuring safety and minimizing personnel injury.

3. Some minor injuries occurred as a result of the fire.

TVA Response

A few Browns Ferry employees were treated for smoke inhalation. No member of the public was injured in any way.

B. Direct Cause of the Fire, its severity or its consequences

1. The following factors contributed directly to the cause or severity of the fire:

(a) Failure to evaluate the hazards involved in the sealing operation and to prepare and implement controlling procedures.

TVA Response

Prior to operation of Browns Ferry unit 1, TVA engineers participated in and reviewed reports of fire tests made on penetrations and cables similar to those which were to be used at Browns Ferry. The test results indicated that fire propagation through a completed penetration was an unlikely event. On the basis of these tests, a penetration and cable fire was not considered to be a high degree risk; therefore, special design control and installation procedures were not considered necessary.

Since the March 22, 1975, fire, TVA management is aware that cable and penetration fires are a potentially severe hazard. Administrative control and inspection have been upgraded in consideration of the hazard.

(b) Failure of workers to report numerous small fires experienced previously during the sealing operations, and the failure of supervisory personnel to recognize the significance of those fires which were reported, and to take appropriate corrective actions.

TVA Response

In the future the procedures to be followed in fire reporting will be stressed to all employees at the plant as a part of the overall training program. A minor fire that occurred in the sealing material on March 20, 1975, has been discussed in a plant management staff meeting on March 21 and was in the process of being followed up by supervisory personnel when the major fire of March 22 occurred.

(c) Use of an open flame without fire precautions specific to this activity.

TVA Response

Since the fire, TVA has issued written instructions prohibiting the use of any flame at Browns Ferry to check for air leaks. In addition, a procedure has been written establishing the control of the use of cutting, welding, and open flames in operating plants.

(d) Ineffectual leadership by TVA in fire fighting activities.

TVA Response

Since the fire additional plant supervisory personnel have attended the fire brigade leader training course so that now all Browns Ferry Shift Engineers and Assistant Shift Engineers are certified as having participated in this training.

(e) Inadequate training of TVA personnel in fire fighting procedures and equipment.

TVA Response

TVA believes there was adequate training in fire fighting procedures and equipment. Apparently training records at Browns Ferry were not reviewed by the NRC even though the records were made available to the investigators. Extensive training in fire fighting techniques had been conducted before the fire and the records so show. In addition, a special briefing on fire fighting has been conducted for over fifty Browns Ferry operations personnel since the fire.

(f) Delay in application of water in fighting the fire.

TVA Response

The main objectives of TVA's efforts on March 22 were to ensure the safety of the reactors thereby insuring public health and safety and to minimize personnel injury. The NRC Office of Inspection and Enforcement report fails to recognize these points in the repeated implications that water should have been used on the fire at an earlier time. The plant superintendent made the conscious decision not to use water (fog, foam, or straight) because of the possibility of short circuits and further degradation of the plant to a condition that would have been more difficult to control. Reactor safety concerns under the circumstances took precedence over extinguishing a localized fire. Even in retrospect and in the light of nearly five months of engineering evaluation, this decision still stands as the correct decision. Of course, water was used to extinguish the fire after a more stable condition had been reached at the plant.

(g) Difficulties encountered in use of self-contained breathing apparatus, caused by inadequate training of personnel in its use, inadequate maintenance and inability to recharge air bottles fully.

TVA Response

The types of self-contained breathing apparatus provided at Browns Ferry are well maintained, are fully adequate, and are in common use in fire fighting. A special training course has been conducted for plant employees in the use of the equipment. Some air cylinders brought to the scene of the fire had been specifically marked as being in the shop for reconditioning. Additional recharging equipment will be installed at the plant to provide additional capability onsite to recharge a large number of tanks fully and more rapidly.

(h) Inaccessibility of the initial ignition location due to the position of the penetration and testing of seals prior to flameproofing.

TVA Response

The location of the fire barrier in the penetration has no bearing on the function of the fire barrier. While technically it would be a deviation from the design drawing, it would not be a reportable design deficiency.

(i) Lack of familiarity with the performance characteristics of the fire nozzle provided for such fires and ultimately used successfully to fight the fire.

TVA Response

Plant Operations personnel who have responsibility for fire fighting are instructed in the use of fire extinguishing equipment. The only evidence that might support this conclusion is that one operator in his statement said it worked more like a shower head than a fire nozzle. We do not understand the basis for this conclusion. Fog-type nozzles are utilized in our fire training classes and an individual would only have to look at the nozzle to see the type it was.

2. The following factors contributed to difficulties experienced during the post-shutdown cooling of the reactors.

(a) For Unit 1, fire rendered inoperable for a significant period of time portions of the control and power circuits for components used in the normal cooldown of the reactor and in emergency core cooling systems.

TVA Response

The Emergency Core Cooling System (ECCS) is designed to provide adequate core cooling in the unlikely event of a loss of normal core cooling water including a postulated break in the largest reactor cooling water pipe during full power operation. This accident, of course, did not take place. While the availability on unit 1 of emergency core cooling water throughout the event would have greatly facilitated unit 1 core cooling, the reactor was safely shut down using other cooling water sources in conjunction with the operation of the reactor depressurization features of the ECCS. Adequate ECCS for unit 2 was available.

(b) Lack of initiative on the part of onsite supervisory personnel to coordinate shutdown activities until after the first manual initiation of depressurization of Unit 1.

TVA Response

An objective review of the detailed statements included in the NRC I&E report will substantiate that supervisory management personnel were onsite and did exercise the initiative to coordinate the activities pertaining to reactor shutdown and restoration; that the plant superintendent was in the control room and states that the decision to blowdown (reduce reactor pressure) was his; and that once the fire was extinguished an orderly approach to verifying actual conditions was undertaken and temporary power was reestablished to vital equipment.

(c) Lack of knowledge of location and severity of the fire on the part of control room personnel and delay in rapid shutdown of the reactor when there were anomalous instrumentation indications.

TVA Response

As noted in Conclusions A1 and A2 both units 1 and 2 were safely shut down and cooled down. See TVA response to Area of Concern No. 11.

(d) Reduction of the options for cooling available during the period of repressurization of the Unit 1 reactor cooling system.

TVA Response

Overpressure protection was available at all times for both reactors. On Unit 1 the capability to blowdown the reactor to low pressure was lost for a period of time. The NRC is reviewing a TVA-proposed design change to eliminate this problem in the future.

(e) Failure to establish priorities for restoration of equipment. This resulted in uncoordinated and, in some cases, counter-productive or duplicative attempts to restore key equipment to service.

TVA Response

An objective review of the detailed statements included in the I&E report will substantiate that supervisory management personnel were onsite and did coordinate the activities pertaining to reactor restoration. Once the fire was extinguished, an orderly approach to verifying actual conditions was undertaken and temporary power was reestablished to vital equipment.

As wires were disconnected or contacts jumpered, the steps were written down and later put in a log as required by standard plant practices. In addition, the Plant Operations Review Committee met at 8:45 the night of the fire to discuss plant conditions and restoration activities to be taken during the night. The few examples cited of counterproductive or duplicative effort in the I&E transcripts are overwhelmed by statements to the contrary.

3. The following factors contributed to the adverse interaction between safety-related systems of Unit 1 and Unit 2:

(a) Two identified cases of failure to follow the TVA cable separation criteria described in the FSAR.

TVA Response

See TVA response to alleged infraction 3.e (1).

(b) Some Unit 1 and Unit 2 ECCS equipment is supplied from common 4 KV boards in accordance with the FSAR. The failure of any of these boards causes a loss of power to the connected equipment in both units.

TVA Response

It is important to note that there are two divisions of redundant emergency core cooling equipment for each reactor. Either division can provide adequate core cooling in case of a design basis accident. Failure of a single 4 KV board would affect, at most, equipment in one division on each unit.

This is a well known design feature of the Browns Ferry plant. It was thoroughly reviewed by the Atomic Energy Commission and was approved by the AEC as being completely acceptable. See TVA response to alleged infraction 3.d(2).

C. Other Important Factors Not Directly Contributing to the Cause of the Fire, its Severity or its Consequences

1. Rapid manual operation of the installed carbon dioxide system in the cable spreading room was precluded because metal plates installed behind breakout glass during construction had not been removed.

TVA Response

See TVA response to 2.a(4).

2. Operation of TVA's Central Emergency Control Center (CECC) was not well coordinated; information sometimes was not exchanged internally. CECC communications with personnel at the Browns Ferry site were not effective in keeping the CECC currently informed concerning the status of the plant or of recovery activities. Consequently, communications with other agencies led to misunderstanding of plant status by those agencies.

TVA Response

TVA believes that the CECC, Division of Power Production Emergency Center, the Site Emergency Staff, and the Environs Emergency Center fulfilled their responsibilities and that good coordination existed throughout the fire and recovery period. The information available to CECC was expeditiously communicated to outside agencies with the full knowledge that, in retrospect, some of the information could have been incomplete. There was no deliberate attempt by TVA to mislead outside agencies.

SUMMARY OF FINDINGS OF FACTS

A. Events Leading to the Fire and Fire Fighting Efforts

During the investigation of events leading to the fire and the fire fighting efforts the following facts were disclosed:

1. The immediate cause of the fire was the ignition of polyurethane which is used at BFNP as a cable penetration sealing material. Construction workers

were checking for leaks in a penetration connecting the cable spreading room with the reactor building. Personnel performing the checks utilized a candle flame to detect air flow from the cable spreading room to the reactor building. The candle flame ignited the polyurethane. (Details I, page 3)

TVA Response

Since the fire, TVA has issued written instructions prohibiting the use of any flame at Browns Ferry to check for air leaks. In addition, a procedure has been written establishing the control of the use of cutting, welding, and open flames in operating plants.

2. There was no approved written procedure provided to control the inspecting, sealing, and testing of cable penetrations. (Details I, page 11)

TVA Response

Since the fire, procedures have been issued to require surveillance by engineering personnel of all penetrations under repair or construction. These procedures also provide for instruction of craft personnel engaged in penetration sealing and flameproofing and provide for inspection by engineers.

3. Personnel responsible for inspecting the penetration seals were also involved in making the seal repairs. (Details I, page 11)

TVA Response

Procedures for installation and inspection of penetration seals have been issued that provide for installation by craft personnel and independent inspection by engineers. The inspectors are prohibited from engaging in the work.

4. Some penetration seals were not properly completed prior to operation of Units 1 and 2, nor were they properly maintained subsequent to operation. (Details I, page 9)

TVA Response

See TVA response to item 1.a.

5. The location of the fire barrier in the penetration did not conform to the approved design drawings. (Details I, page 9)

TVA Response

The location of the fire barrier in the penetration has no bearing on the function of the fire barrier. While technically it would be a deviation from the design drawing, it would not be a reportable design deficiency.

6. The electrical design specified the installation of cable in trays in excess of that permitted by the TVA design criteria in at least two cases. Other trays could contain excess cable in that cable was abandoned in place when designs were changed. (Details I, page 12)

TVA Response

A design guideline for cable tray fill based on 60% of the cross sectional area of a tray carrying control and signal cables is indicated in the FSAR. See TVA response to Area of Concern 8.

7. Although not required by license conditions, laboratory tests had been performed by TVA to verify the ability of the penetration seals to perform as a fire barrier. These tests, however, did not simulate the condition existing at the time of this fire. For example, the tests were not conducted with a pressure differential across the seal, the polyurethane was not exposed to the flame nor was a leakage path established through the seal. Additionally, the test did not include the sheet type polyurethane in use in the cable penetrations. The use of this sheet type foam had not been approved by the TVA design department. (Details I, page 10)

TVA Response

Flame tests were conducted before the operation of Browns Ferry to confirm the completed penetration seal design concept. The results of these tests indicated that a completed fire stop would be effective. Since the fire, TVA has constructed a facility to conduct testing under a differential pressure on a mockup of the penetration in which the fire occurred.

8. Construction personnel were utilizing an open flame in an area which is vulnerable to fire, but specific precautions and trained fire fighters were not present at the time. (Details I, page 12)

TVA Response

Written procedures have been established since the fire to control the use of open flames. In addition, instructions are being issued to require additional surveillance by fire fighting personnel at open penetrations in operating areas.

9. Previous fires in the polyurethane foam materials had not always been reported to the appropriate levels of management, and, on the occasions when reported, no action was taken to prevent recurrence. (Details I, page 11)

TVA Response

In the future the procedures to be followed in fire reporting will be stressed to all employees at the plant as a part of the general training program. A minor fire that occurred in the sealing material on March 20, 1975, had been discussed in a plant management staff meeting on March 21 and the minor fire was in the process of being followed up by supervisory personnel when the major fire of March 22 occurred.

10. The various procedures for responding to fires contained conflicting information. For example, the BFNP Emergency Procedure lists two different telephone numbers to be used in reporting a fire, one in a table of emergency numbers and the second in the test of the procedure. The appropriate number is the one in the test; dialing this number automatically sounds the fire alarm and rings the Unit 1 operator's telephone. (Details I, page 4)

TVA Response

At Browns Ferry two different phone numbers can be used to report a fire. One phone number initiates the fire alarm and is a standard number (299) used in all TVA operating plant phone systems. The other number on the construction phone system (235) rings the fire marshall responsible for fire fighting during construction. Dialing either on the appropriate telephone system would result in prompt reporting of a fire.

11. The BFNP Emergency Procedure was not followed by those involved when reporting the fire. The construction workers first attempted to extinguish the fire, whereas the procedure specifies that the fire alarm be sounded first. The guard reporting the fire telephoned the Shift Engineer's office rather than calling either of the numbers listed in the procedure. (Details I, page 4)

TVA Response

The procedures to be followed at the plant will be stressed as part of the general employee training program.

12. During construction, metal plates were installed under the breakout glass in the CO^2 system manual crank stations located at the entrances to the cable spreading room. These metal plates were not removed prior to plant operation, thus preventing manual initiation of the CO^2 system had it been necessary. The CO^2 system was successfully initiated automatically. (Details I, page 3)

TVA Response

See TVA response to alleged infraction 2.a(4).

13. TVA does not have outside agencies inspect their fire protection equipment and systems. Internal inspections of fire protection equipment had not revealed the presence of the metal plates. (Details I, page 6)

TVA Response

TVA had fire protection instructions in effect explaining fire protection system status including presence of the plates to the operators. The presence of the plates was known by TVA plant operations personnel. See response to Summary of Findings of Facts item A12.

14. Contrary to good safety practices, the plant procedures do not restrict the use of elevators during fires; both operations and construction personnel used the plant elevators while the fire was in progress. (Exhibit 1, Elect.-D, AUO-Q et al.)

TVA Response

We question the relevance to nuclear safety of this comment. Use of the elevators during the early stages of the fire enhanced rapid movement of personnel to perform operations related to safely shutting down the reactors and assisted in fighting the fire.

15. The practice at TVA's facilities has been that manual application of water on electrical fires be avoided for personal safety reasons. CO_2 and dry chemical equipment is provided for use by personnel fighting such fires. TVA's Fire Safety Manual permits the use of water on electrical fires and a nozzle designed for manual application of water on electrical fires had been provided at the hose station in the vicinity of the fire area. (Details I, page 13)

TVA Response

TVA has established no general policy prohibiting the manual use of water on electrical fires for personnel safety reasons. In this fire incident, the use of water was considered, but delayed due to reactor safety considerations. Written procedures have been revised and special training sessions held to emphasize that water is an acceptable extinguishing agent for electrical fires under appropriate circumstances.

16. The BFNP fire hoses and nozzles were not interchangeable with the Athens Fire Department equipment because different types of threads were used. (Details I, pages 8 and 13)

TVA Response

The fire hose threads at Browns Ferry were threads cut to a national standard. A particular nozzle from the Athens Fire Department would not mate with a Browns Ferry hose apparently because it was cut to another national standard. The TVA hose and nozzle were used to extinguish the fire. TVA will insure that fire hose threads at Browns Ferry are compatible with those of the Athens Fire Department.

17. Some TVA personnel experienced difficulty in using the self-contained breathing apparatus. (Details I, pages 6, 9 and 13)

TVA Response

See TVA response to Area of Concern 5.

18. There were no organized efforts to extinguish the fire in the reactor building for about three and one half hours. Sporadic individual efforts may have been made during this period. (Details I, page 7)

TVA Response

During this period, which was about two hours, the plant staff devoted its major efforts to the more important priority tasks of safely shutting down the reactors and fighting the fire in the cable spreading room after which fire fighting in the reactor building was resumed.

19. Although the responsibility for handling this occurrence rested with the Plant Superintendent, he was hesitant in authorizing the use of water on the fire; during discussions with his supervisor he requested permission to use water on the fire. (Details I, page 8 and Details IV, page 1)

TVA Response

The main objectives of TVA's efforts on March 22 were to ensure the safety of the reactors and to minimize personnel injury. The report fails to recognize these points in the repeated implications that water should have been used on the fire at an earlier time. The plant superintendent made the conscious decision not to use water (fog, foam or straight) because of the possibility of short circuits and further degradation of the plant to a condition that would have been more difficult to control. Under the circumstances reactor safety concerns were a more important consideration than extinguishing a localized fire. Even in retrospect and in the light of nearly five months of engineering evaluation, this decision still stands as the correct decision. Of course, water was used to extinguish the fire after a more stable condition had been reached.

20. Only minor injuries to personnel were sustained during the fire. (Details I, page 18)

TVA Response

This finding is correct. See response to Conclusion A3.

21. The fire fighting and the shutdown of Units 1 and 2 were accomplished without overexposure of personnel to radiation and without the release of radioactive effluents from the plant in excess of license limits. (Details I, pages 13-18)

TVA Response

This finding is correct. See response to Conclusion A1.

B. Operation Events and Problems Experienced During Fire and Until Shutdown Cooling Established

During investigation of operational events and problems experienced during the fire and until shutdown cooling was established, the following facts were disclosed:

1. The nuclear cores for both units were adequately cooled during and subsequent to the fire. The minimum water level (Unit 1) was about forty-eight inches above the top of the active fuel. Maintaining reactor water level was never a problem in Unit 2. (Details II, page 7)

TVA Response

This finding is correct.

2. The control room was manned at all times although some smoke and fumes entered the room for a brief period. (Details II, page 5)

TVA Response

This finding is correct.

3. The ability to use the emergency core cooling system (ECCS) as backup to normal cooling was lost for Unit 1. The capability to operate the Unit 1 Standby Liquid Control Systems was lost for at least three hours. (Details II, pages 4 and 8)

TVA Response

The Emergency Core Cooling System (ECCS) is designed to provide adequate core cooling in the unlikely event of a loss of normal core cooling water including a postulated break in the largest reactor cooling water pipe during full power operation. This accident, of course, did not take place. While the availability on unit 1 of emergency core cooling water throughout the event would have greatly facilitated unit 1 core cooling, the reactor was safely shut down using other cooling water sources in conjunction with the operation of the reactor depressurization features of the ECCS. Adequate ECCS for unit 2 was available.

The Standby Liquid Control System (SLCS) is designed to inject boron into the core to shutdown the nuclear reaction if control rods fail to scram (be inserted into the core). The SLCS was not needed at any time during the event as all unit 1 and 2 control rods were observed to have scrammed.

4. Loss of the air supply system for relief valve control resulted from the loss of electrical power to a solenoid valve in the air supply to a diaphragm type isolation valve. This led to the repressurization of the Unit 1 reactor from about 6:40 p.m. to 10:20 p.m. (Details II, page 9)

TVA Response

Overpressure protection was available at all times for both reactors. On unit 1 the capability to blowdown the reactor to low pressure was lost for a period of time for the reason indicated. The valve in question was subsequently bypassed and reactor pressure reduced. The NRC is reviewing a TVA-proposed design change to eliminate this problem in the future.

5. The drywell vent valves for Unit 1 were wired open to prevent drywell pressure buildup. (Details II, page 9)

TVA Response

This was intentionally done to vent the unit 1 primary containment through the charcoal filters of the Standby Gas Treatment System as required by procedure. Because of the fire these valves failed and isolated the unit 1 drywell. However, they were manually opened and wired open to prevent their subsequent inadvertent closure.

6. The spare control rod drive (CRD) pump was inoperable from approximately 1:30 p.m. on. The Unit 2 CRD pump could have been valved to supply Unit 1 if required. (Details II, page 4)

TVA Response

After the fire during a review of engineering drawings, TVA determined that it would have been possible to valve in the unit 2 CRD (control rod drive pump) to the unit 1 reactor. However, the unit 1 pump did not fail and no backup was needed. This item had no impact on the operations on March 22.

V. Effects on Each Unit and Interactions Between Units

During the investigation of the effects on each unit and interactions between units, the following facts were disclosed:

1. The failure of a single 4 KV Shutdown Board results in the loss of power to the ECCS equipment supplied from the board for both Units 1 and 2. This feature is inherent in the design of the electrical systems for Units 1 and 2 and conforms to the FSAR commitments. Unit 3 electrical systems are not shared with Units 1 and 2. (Details III, page 4)

TVA Response

It is important to note that there are two divisions of redundant emergency core cooling equipment for each reactor. Either division can provide adequate core cooling in case of a design basis accident. Failure of a single 4 KV board would affect, at most, equipment in one division on each unit.

This is a well known design feature of the Browns Ferry plant. It was thoroughly reviewed by the Atomic Energy Commission and was approved by the AEC as being completely acceptable.

2. The coordination of the power supplies for the ECCS is such that motors for the systems pumps are not in every case supplied from the same power source as their associated valves. In six cases the pumps and valves are supplied from power sources in the same division, but in two cases the valves are supplied from one division and the pumps from another, in violation of the separations criteria defined in the FSAR. (Details III, page 8)

TVA Response

It is not necessary that pumps and associated valves in the ECCS be supplied from the same power source. The Browns Ferry design is divisionalized such that pumps and associated valves in the ECCS are powered from sources assigned to either division I or II. The loss of pumps and/or associated valves in one division is an acceptable event since either division can adequately cool the reactor in the unlikely event of a postulated design basis accident.

There is an error in the I&E report which apparently has lead to the erroneous conclusion that the criteria were violated.

3. The power cable supplying 480 Volt Shutdown Board 1B from 4 KV Shutdown Board C and the power cable supplying 480 Volt Shutdown Board 2A from 4 KV Shutdown Board B were both routed in the same tray. In that 4 KV Shutdown Board B is in Division I and 4 KV Shutdown Board C is in Division II, this routing constitutes a violation of the TVA separations criteria as defined in the FSAR. (Details III, page 9)

TVA Response

This finding is an oversimplification of a highly technical point. The original design did not assign divisional classification to 4 KV cables between the 4 KV shutdown boards and the 4 KV/480 Volt transformers. The design was analyzed to and met the single failure criterion. Therefore, the routing indicated in the finding was not a violation of the criteri...

4. Power was lost to the common inlet valve for the core standby coolant (raw water) supply for Units 1 and 2. This valve could have been operated manually. (Details III, page 5)

TVA Response

This statement is irrelevant since it ignores the fact that the raw water supply was never required. As an addi[illegible]onal backup safety feature, the Browns Ferry design includes provisions for injection of river water into the reactor should it become ultimately necessary in the event all other cooling water supplies fail. As indicated in the statement the inlet valve could have been operated if it had been necessary. In this incident, however, other more preferable water sources were available and were used.

5. Power from the Unit 2 preferred power bus was lost for a period of time because an operator connected it to the Unit 1 preferred bus which sustained faults. A second operator subsequently opened the tie breaker and restored power to the Unit 2 bus. (Details III, page 5)

TVA Response

The unit preferred power buses are not required for safe reactor shutdown. However, they do power annunciation and indicating circuits. The NRC report indicates that the above item apparently occurred and unit 2 preferred power was affected. However, this was noticed by plant operators and the power supply was restored.

D. Response of TVA Groups and Various Governmental Bodies

During the investigation the following facts were established concerning the response of TVA groups and various state and local governmental bodies:

1. The Athens Fire Department (AFD) responded promptly to the notification of the fire and was prepared to assist within approximately 35 minutes of receiving the alarm. Athens is about ten miles from the site and, upon arrival, the firemen had to be issued personnel radiation monitoring devices prior to entry. (Details IV, page 1)

TVA Response

In accordance with TVA and NRC requirements, personnel must be issued radiation monitoring devices upon entry to the plant.

2. The AFD fire chief initially made the recommendation to use water on the fire at about 2:00 p.m.; however, permission to use water was not given by the Plant Superintendent to the fire fighters until approximately 6:40 p.m. and it was about 7:00 p.m. to 7:20 p.m. when water was actually used. (Details IV, page 1)

TVA Response

The expert fire fighting advice of the Athens Fire Chief was duly considered. However, the plant operators' detailed knowledge of the Browns Ferry design and potential reactor safety implications of the application of water to burning safety-related electrical cables caused them to delay water application to the fire until plant conditions were made more stable. TVA believes in retrospect, after five months of detailed evaluation, that Browns Ferry plant management made the correct decision.

3. The individual who would normally function as TVA Director, Central Emergency Control Center (CECC), was not immediately located. The first alternate was not aware of this and had been at the CECC approximately thirty minutes before realizing that he should have been functioning as Director. (Details IV, page 2)

TVA Response

Although the alternate CECC director did not realize that he was acting as director immediately upon arrival at the CECC, his logs reveal that he, upon notification, contacted the Environs Director and directed him to notify the Alabama Department of Public Health. This delay in recognition did not slow down the Radiological Emergency Plan from being implemented, and the alternate CECC director did carry out the responsibilities of the CECC director for the 30-minute period.

4. Logs kept by individuals at the CECC did not always indicate the times of the events.

TVA Response

TVA believes that all significant or pertinent events were logged at the CECC. Logging of events is a matter of judgment.

5. Information provided by the CECC to others was not always accurate or current. For example, at about 5:00 p.m. an individual at CECC advised NRC that the fire was confined to the spreading room. (Details IV, pages 2 and 3)

TVA Response

The information available to CECC was expeditiously communicated to outside agencies with the full knowledge that, in retrospect, some of the information could have been incomplete. There was no deliberate attempt by TVA to mislead outside agencies.

6. The Director of the CECC was not aware that the fire had been extinguished until approximately 8:15 p.m. when in fact the fire was extinguished prior to 7:45 p.m. (Details IV, page 4)

TVA Response

The fire was extinguished at about 7:45 p.m. *CDT* (Browns Ferry time) and the CECC director in Chattanooga became aware of it at that time. This, of course, is the same as 8:45 p.m. *EDT* (Chattanooga time), which was the time logged in the CECC log provided to NRC.

7. Communications by the CECC with state and local agencies focused on plant operating status rather than offsite radiological releases which is the prime responsibility of these agencies. (Details IV, page 8)

TVA Response

TVA's review of the overall record concludes that CECC communications with outside agencies did focus primarily on the potential for radiological releases. TVA did respond to questions from outside agencies related to plant status with a view toward allowing them to better assess the overall situation.

8. The Alabama equipment for the downwind air sampling was not available for service. Sampling was initiated at Decatur station at approximately 9:00 p.m. on March 22 using a sampler obtained from the Alabama Pollution Control Commission. (Details IV, page 10)

TVA Response

TVA has no comment on responsibilities of the State of Alabama.

9. The State of Alabama emergency plan was out of date, not available to certain responsible officials and some officials were not cognizant of their individual responsibilities. (Details IV, pages 15, 16 and 19)

TVA Response

TVA has no comment on responsibilities of the State of Alabama.

10. Attempts to contact some local officials by state organizations was minimal. (Details IV, page 15 and 16)

TVA Response

TVA has no comment on responsibilities of state organizations.

APPENDIX 15

ATTACHMENTS INCLUDED WITH STATEMENT OF MR. BEN C. RUSCHE, DIRECTOR, OFFICE OF NUCLEAR REACTOR REGULATION, NRC[1]

ATTACHMENT I

NRR EVALUATION OF PLANT SAFETY DURING THE BROWNS FERRY FIRE

Purpose

The purpose of this evaluation is to determine whether alternate methods could have been used to maintain the plant in a safe shutdown condition during the March 22, 1975, fire at the Browns Ferry plant. Such alternate methods are based on the as-built plant design capability. The plant was, in fact, brought to a safe shutdown condition by the operators in a satisfactory manner. At no time was the core uncovered. Our evaluation of the availability of alternate methods of maintaining safe shutdown has been performed to assess the margin of safety that existed throughout the course of the incident.

Approach

1. The evaluations were restricted to Unit 1 since a review of the events made it clear that the effects of the fire on Unit 2 were less, by far, than on Unit 1. At all stages of the incident, Unit 2 had more capability for maintaining a safe shutdown condition. In addition, due to a longer operating history, the decay heat removal requirements were greater for Unit 1.

2. The incident was divided into three specific periods as follows:

   a. Initiation of fire up to reactor scram

   b. Reactor scram, and

   c. Water addition and heat removal.

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[1] See p. 77, text.

3. The calculations, which are basic to assessing capabilities of various systems and components in maintaining the core covered and in removing decay heat, are dependent on initial plant conditions and important parameters such as time and water level. Starting points as realistic as possible which were determined by cross checking of facts were used in these calculations.

I. Fire Initiation to Reactor Scram

Shortly after the fire started, spurious events occurred. These events were due to open circuits, short circuits and/or grounds caused by the burning of electric cables. In most cases, the events from the false signals proceeded in the direction of safety. For example, approximately ten minutes following the announcement of the fire, the Emergency Safeguards Systems on Unit 1 received a false signal of a potential loss-of-coolant accident situation. This signal started all emergency core cooling equipment and all four emergency diesel generators. The entire system functioned as designed. No water was admitted to the reactor vessel at this time from low pressure sources since in the absence of an actual coolant loss the reactor vessel remained pressurized. Some water was injected by operation of the High Pressure Coolant Injection (HPCI) and Reactor Core Isolation Cooling (RCIC) systems. The operator was presented with sufficient information to assure himself that no loss-of-coolant accident condition existed. Since an adequate supply of water to the reactor vessel was available using normal equipment, these emergency systems were shut down. There was some difficulty shutting down these systems initially since the false accident signal was still present. The operator was able to shut down these systems on subsequent attempts.

Other spurious events included various alarms in the control room and conflicting instrumentation readouts. Later in the day, as attempts to bring the Residual Heat Removal (RHR) System into operation were made, a normally-open pump suction valve was found to be closed. It would appear that this valve had been closed by a spurious closure signal.

We have concluded from our investigation that spurious events which occurred in this initial period did not play a particularly important role in shaping the course of subsequent events. However, they were indicative of progressive loss of equipment operability.

Reactor Scram

The Reactor Protection System (RPS) causes automatic reactor shutdown (scram) when preset values of particular variables are exceeded. There are also means for scramming the reactor manually. During the fire, both units were scrammed manually at the judgement of the operator. Automatic scram did not occur prior to this because none of the limits for scram were exceeded. Nevertheless, one may ask whether there was any danger of the scram system being disabled so that the reactor could not be shut down.

Loss of power to the reactor protection system, such as might occur during a fire, is a fail-safe condition that would cause scram. Therefore, the principal question is whether or not the protection system wiring could have been maintained in an energized condition by coming into contact with an extraneous power source when fire damage occurred to cable insulation. All of the reactor protection system cables are run in rigid conduit. No cables containing power from sources other than the RPS motor generator sets are run in these conduits. Thus, the RPS cables are fully protected by the grounded conduit against the possibility of negating scram by becoming powered by external sources. Heating and loss of insulation on protection system cables within the conduit could lead to short circuits. The results would be blown fuses causing associated relays to de-energize. This is a fail-safe condition leading to reactor shutdown.

We have concluded through detailed examination of the RPS design that the system would have shut the reactor down automatically had it been required because of any transient condition. We have concluded also that fire damage would have inevitably caused automatic scram had the reactor not been shutdown manually.

III. Water Fill and Heat Removal

Starting from an operating condition the normal means of reactor shutdown and cooldown following control rod insertion would be as follows:

a. Continue steam flow to the main condenser through the turbine bypass lines until steam pressure is reduced to a point (less than 100 psig) where condenser vacuum could not be maintained; and,

b. With reactor vessel pressure below 100 pisg, initiate the reactor shutdown cooling mode of the RHR system. This system circulates reactor vessel water through a heat exchanger and back to the pressure vessel via an RHR pump.

It should be noted that in the normal method described above, all heat is rejected to outside sources of cooling water, either in the main condenser or the RHR heat exchanger. No makeup water is required since reactor water is circulated to the heat exchanger and returned to the vessel.

When the Main Steam Isolation Valves (MSIV) closed during the fire, the routing of steam to the main condenser was no longer possible. Due to the loss of operability of certain components by fire damage, the normal means for cooling and making up water for the isolated condition could not be accomplished. The normal means of water fill and heat removal under these conditions would have been as follows:
Decay heat increases reactor vessel pressure to the setpoints of the relief valves which open and provide a flow path to the suppression pool. The decay heat is thus transferred for a short time immediately after shutdown from the fuel to the suppression pool via boiling of the water surrounding the core. This transfer involves a loss of water from the vessel which is automatically made up by operation of either the RCIC or HPCI steam driven turbines which pump water from the condensate storage tanks or from the suppression pool to the vessel. Transfer of heat from the suppression pool to an outside cooling source is accomplished by operating the RHR system in the torus cooling mode during which an RHR pump circulates suppression pool water through a heat exchanger cooled by river water. Shortly after shutdown, decay heat levels are low enough to match the heat removed by steam flow to the RCIC turbine. The relief valves will then close.

We will now discuss the methods utilized by the operator and alternate methods that were available. For reference purposes, Table 3 provides a listing of all plant pumps normally capable of providing makeup water to the pressure vessel, including both high pressure and low pressure capability. The availability of these pumps during the incident is covered under the remarks section of the Table.

The discussion can be separated in terms of water makeup and heat removal. Heat removal from the fuel was accomplished automatically through relief valve operation and steam release to the suppression pool. The large volume of the suppression pool gives it a very large heat absorption capacity, and 10 to 12 hours of decay heat can be handled prior to the need for operator action. Heat removal to an ultimate sink is not as critical a consideration as water makeup and will, therefore, be discussed later.

At high vessel pressure, the automatic makeup is normally provided by the feedwater system backed up with either the steam driven HPCI or RCIC systems. Neither the HPCI or RCIC were automatically available following spurious operation at the start of the fire.

With the only source of high-pressure makeup water in operation being a single control rod drive (CRD) pump, the plant operators decided to reduce the pressure in the vessel to a point where they would be able to provide makeup water . low pressure equipment known to be operable. However, other equipment could have been used to maintain the core covered with water.

Besides the CRD pump on Unit 1, other installed sources of high pressure makeup were the CRD pump on Unit 2, a shared spare CRD pump and standby liquid control (SLC) pumps. The CRD pumps, while performing their normal functions associated with the control rod drive system, also provide water to the vessel while at any pressure. One CRD pump is normally in operation per unit and the pump for Unit 1 operated continuously throughout the course of the incident. In addition the SLC pumps are each capable of providing approximately 56 gpm at pressures up to reactor coolant system design pressure. The SLC pumps were not required as a backup reactivity shutdown system since the control rods functioned normally. However, there was a time period of up to three hours following the initiation of the fire during which they were not available due to loss of power. Means of utilizing installed sources of make up water and an analysis of their effectiveness follows. The effectiveness of a given component can be illustrated by utilizing Figures 1 & 2. These figures show the water makeup rate required to keep the core covered as a function of time after shutdown. Their bases are detailed with the figures.

Although our calculations are based on the core covered with water it should be pointed out that this is not necessary to prevent serious core damage or melting. If the core uncovers, steam generated in the portion of the core that is covered will still provide some cooling to the upper and uncovered portion of the core.

Fuel rod cladding temperatures could be allowed to reach at least 1800°F without serious consequences. Therefore, the steam leaving the core could be at nearly 1800°F rather than at saturation temperature. The increase in temperature would increase the enthalpy or heat content for the same mass flow, and thus a lower flow would be adequate. However, since the core remained covered no heatup beyond saturation temperature was experienced during or following the fire.

The CRD pump in operation was part of a system for Units 1 and 2 which consisted of three CRD pumps. One pump normally operates for each unit and the third pump can be used on either unit. Subsequent examination of the actual piping configuration confirmed that it is also possible to align the Unit 2 pump to provide water to Unit 1. Means also exist to increase the output of a CRD pump by valving in a pump test bypass line which provides an additional flow path. It is estimated that by opening this single valve it would have been possible to have provided sufficient water, approximately 225 gpm, to maintain the core covered throughout the course of the incident. No other systems would have been required to provide water to maintain an adequate inventory of water in the reactor vessel and depressurization would not have been necessary. This flow (225 gpm) could have been increased to in excess of 300 gpm with an additional CRD pump. Figure 1 shows that a flow rate of approximately 218 gpm would have been adequate to maintain core coverage throughout the incident.

An additional source of high pressure water mentioned previously as being unavailable due to fire damage was the RCIC system. It would have been capable of providing sufficient flow (600 gpm) for makeup water requirements throughout the entire course of the incident if the operators' decision had been to make it available. ears that this system could have been made available within an hour king this decision. The source of steam for the RCIC system would have been the auxiliary boiler which was used for testing the RCIC prior to plant operation. Two procedures are necessary to provide the steam path. First, the auxiliary boiler must be put into operation. Full steam pressure from this source can be obtained in less than one hour. (The operators actually put the auxiliary boiler into operation later in the incident). The second procedure is the installation of a spool piece to make up the flow path from the auxiliary boiler to the RCIC turbine. This could have been accomplished in less than one hour. The operation of the RCIC would then have been possible from the backup control room where there is control and indication of system operation both in terms of water flow to the vessel and turbine speed.

Beyond the RCIC system, there remains the standby liquid control (SLC) pump mentioned previously. If the decision had been made to make the pump capacity available as high pressure makeup, the operators have indicated that these pumps could have also been made available within an hour. The capacity of these pumps alone would not have been sufficient to provide for complete water requirements. The capacity of these pumps along with the flow rate being obtained from the operating CRD pump (total of approximately 212 gpm) would have provided a sufficient quantity of water to keep the core adequately cooled. Obviously, the use of one or both of the SLC pumps in conjunction with other makeup sources already noted would have provided more than an adequate supply of high pressure water to assure that the core would have remained covered.

We have discussed above various alternate means of obtaining high pressure sources of makeup water which were not utilized. Instead, as we noted previously, a decision was made by the operators to depressurize the reactor. This was accomplished by using the four relief valves available for remote manual operation from the control room. Only one would have been required to perform the function. The remaining seven relief valves with remote manual capability were not operable because of fire damage to the power supplies to the solenoids controlling air flow to the valves. However, it has been established that should the operators have needed any of the other valves it would have been practical to supply solenoid opening power at the appropriate electrical connecting pins at the containment penetration.

Once the reactor vessel pressure was reduced to 350 psig one condensate booster pump and one condensate pump provided an adequate source of makeup water. In addition the operators had available two additional condensate booster pumps and two additional condensate pumps. Any one of the condensate pumps was capable of supplying more than enough water to maintain the core covered at reactor pressure below 160 psig. An additional source of low pressure water for injection into the reactor vessel was river water. If needed, either of two available pumps could have provided sufficient river water to keep the core covered as long as the reactor pressure was down to 185 psig or less.

During the stable low pressure operation in which the condensate booster pump and condensate pump were providing makeup water, operability of the relief valves being used for maintaining low pressure in the vessel was lost as a result of the loss of control air. At this time the relief valves closed and the continued production of decay-heat resulted in a pressure increase in the vessel. The pressure increased beyond the capability of the available low pressure injection systems, and the capability of providing water by these systems was lost. The operators established the cause of the loss of control air and the decision was made to bypass the closed solenoid valve in the common relief valve air supply line. The air supply in the accumulators designed to provide for relief valve operation under isolation conditions had been depleted.

The bypass action was accomplished in approximately 3-1/2 hours following which time the reactor vessel pressure was once again reduced within the capability of the low pressure condensate booster pump and the condensate pump.

There were other courses of action which might have been taken by the operator at the time that remote-manual operability of the relief valves was lost. No immediate problem existed since the pressure would have increased up to the setpoints of the relief valves in their over-pressure protection mode with subsequent steam relief to the suppression pool. The CRD pump was providing a source of makeup water. With the much reduced decay heat, considerable time was available for other operator action. The alternatives discussed under high pressure makeup were still available if control air to the relief valves could not be reestablished.

Calculations, however, indicate that no operator action was necessary to maintain the plant in a safe condition. This is due to the availability of a depressurization and heat removal path via the main steam line drain valves to the condenser. Both of these valves were inoperable by electrical means as a result of fire damage. The operators however, had decided previously to return draining capability to the main steam line and this had been achieved approximately one hour after control air to the relief valves was lost. It is calculated that the quantity of steam being removed from the pressure vessel through the main steam drainline was great enough that the reactor pressure would have leveled off at a point prior to reaching the relief valve set-point. An equilibrium condition would then have been maintained with the reduced reactor pressure reducing the head on the operating TRC pump such that the pump would provide sufficient makeup flow to maintain the core covered throughout the remainder of the incident.

The calculations providing the basis for these statements are as follows:

1. Maximum steam flow rate through the 4 inch drain line during pressurization of the reactor (between 7:30 and 9:30 PM) based upon an average pressure of 530 psig is 122,000 lb/hr.
   a. Adiabatic compressible flow with friction Fanno line with choking,
   b. from observation of the piping: $4(\frac{fl}{D}) = 20$,
   c. inlet mach number = .185, and
   d. inlet velocity = 378 ft/s.

However, the loss of 122,000 lbs of water would cause the reactor vessel liquid level to decrease 114 inches in one hour,

$$\frac{122,000\ \text{lb/hr}}{50\ \text{lb/ft}\ \times\ 20.9\ \text{ft /inch of reactor level}} = 114\ \text{inches/hr}$$

Since the reactor liquid level remained essentially constant through the prepressurization period it is obvious that the actual steam flow was considerably less than the calculated as a maximum. However, the flow would necessarily have been equal to or greater than the makeup flow provided during that time period by the CRD pump. This is based on the level not showing a net increase during the time period even though the CRD pump was continually in operation. We can therefore conservatively state that the drain capacity at the average 530 psig pressure was 152 gpm corresponding to 76,000 lb/hr. The 152 gpm is estimated from the CRD pump head and system resistance curves.

2. From the reactor pressure recorder stripchart it was determined that had the relief valve not opened, the pressure rise, extrapolated to about 2:00 AM would have reached 700 psig; however, at 700 psig the steam discharge rate would have increased, i.e.,

$$\frac{\text{Steam discharge rate @ 700 psig}}{\text{Steam discharge rate @ 530 psig}} = \frac{@\,700}{530}\sqrt{\frac{T_{\text{Abs 700}}}{T_{\text{Abs 530}}}}$$

$$= \frac{.86}{.64} \times \sqrt{\frac{965}{934}} \approx 1.34$$

at 700 psig, steam discharge rate = 102,000 lb/hr X 1200 BTU/lb. This corresponds to a heat load of 122 X 10^6 BTU/hr, i.e., 102,000 lb/hr X 1200 BTU/lb. However, at 2:00 AM (about 47,000 seconds after scram) the reactor deay heat is 0.715% of full power = 23.6 MW which corresponds to 80 X 10^6 BTU/hr. Since for the assumed pressure of 700 psig, more energy leaves the reactor than that which is produced, the reactor pressure could never get as high as 700 psig.

Furthermore, the decay heat rate of 23.6 MW corresponds to a makeup rate of 142 gpm, which is achievable with one CRD pump operating against a reactor pressure of 660 psig (at 700 psig 1 CRD pump will deliver ~ 139 gpm).

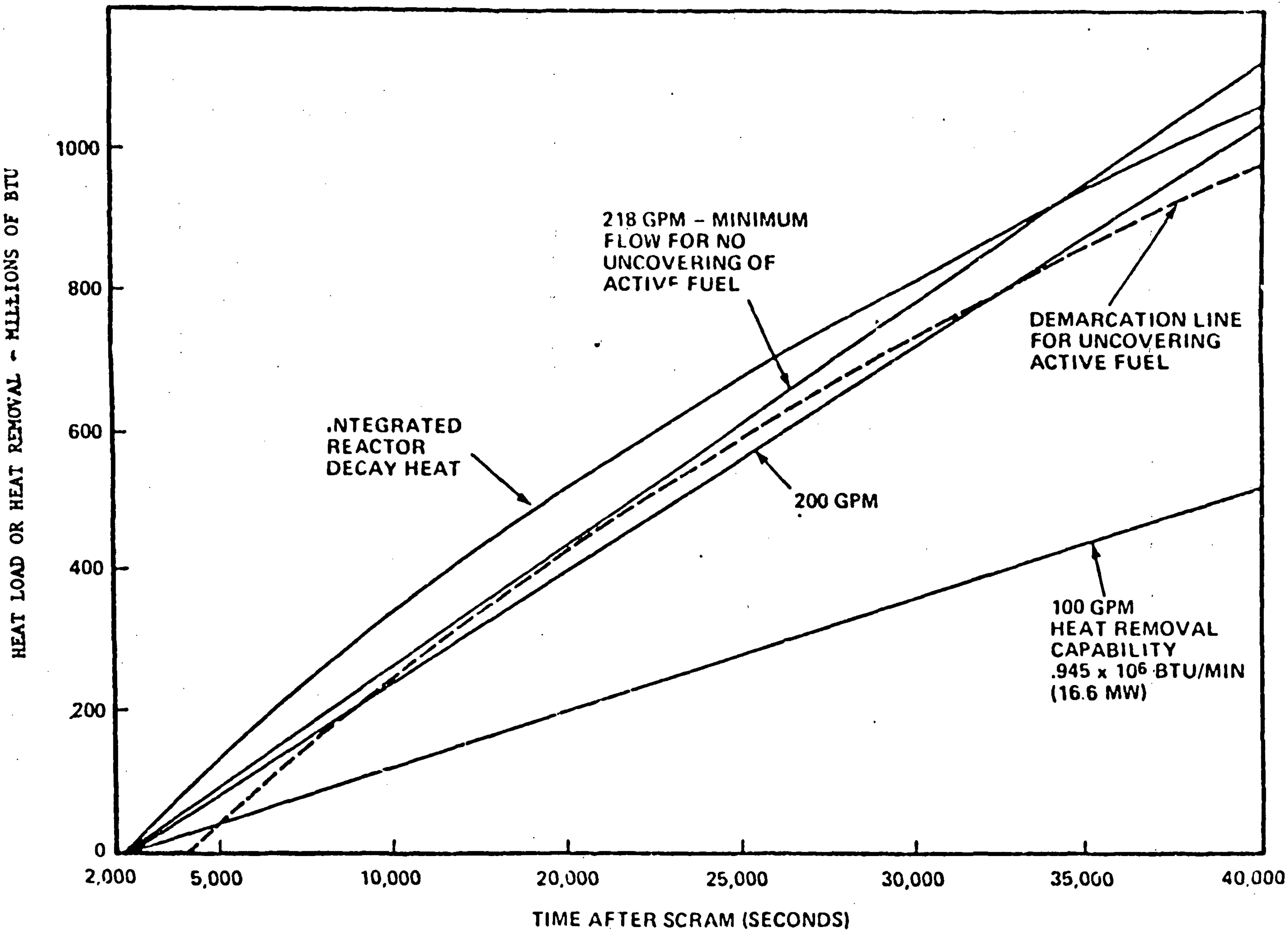
The operators however did regain use of control air and proceeded to depressurize the reactor vessel which permitted use of low pressure pumps. This stable condition continued until normal means became available (which occurred only following clearing of the smoke sufficiently for the operators to establish the proper valve lineup) such that torus cooling could be attained. Thereafter a normal reactor shutdown cooling mode was used.

The means of heat removal available through the course of the incident was not critical due to the availability of relief valves to transfer the decay heat to the suppression pool which is of sufficient volume to preclude the need for any operator action for a time period of 10 to 12 hours following the start of the incident. The suppression pool could then be permitted to boil, if necessary, with the steam vented through the standby gas treatment system to the atmosphere.

There were also other means of heat removal from the reactor. Once the main steam line drain valves were open, a flow path existed to the condenser whereby the decay heat was being transferred to the river water. This greatly reduced the heat load on the suppression pool within the torus and eventually would have permitted removal of all of the generated decay heat. Means of removing the decay heat from the torus prior to availability of normal torus cooling existed. There are several flow paths by which water could be drained from the torus thereby removing the heated water and by which cold makeup water would be introduced to the torus. The reactor heat removal drain pumps were operable and provided flow paths by which water could be transferred from the torus to the main condenser and also from the condensate storage tanks to the torus. Flow rates far in excess of those required were available. The staff concludes that alternate means of heat removal were available throughout the course of the incident. Normal suppression pool cooling was established 12 hours after scram and normal shutdown cooling of the reactor was established 15 hours after scram.

The core remained covered throughout the incident. The operators always had a course of action which they were pursuing and in addition there was always at least one or more redundant means available to cool the core.

953



1000
800
600
400
200
0
2,000
5,000
10,000
20,000
25,000
30,000
35,000
40,000
TIME AFTER SCRAM (SECONDS)
HEAT LOAD OR HEAT REMOVAL - MILLIONS OF BTU
218 GPM - MINIMUM FLOW FOR NO UNCOVERING OF ACTIVE FUEL
DEMARCATION LINE FOR UNCOVERING ACTIVE FUEL
INTEGRATED REACTOR DECAY HEAT
200 GPM
100 GPM HEAT REMOVAL CAPABILITY .945 x 10^6 BTU/MIN (16.6 MW)

Figure 1.

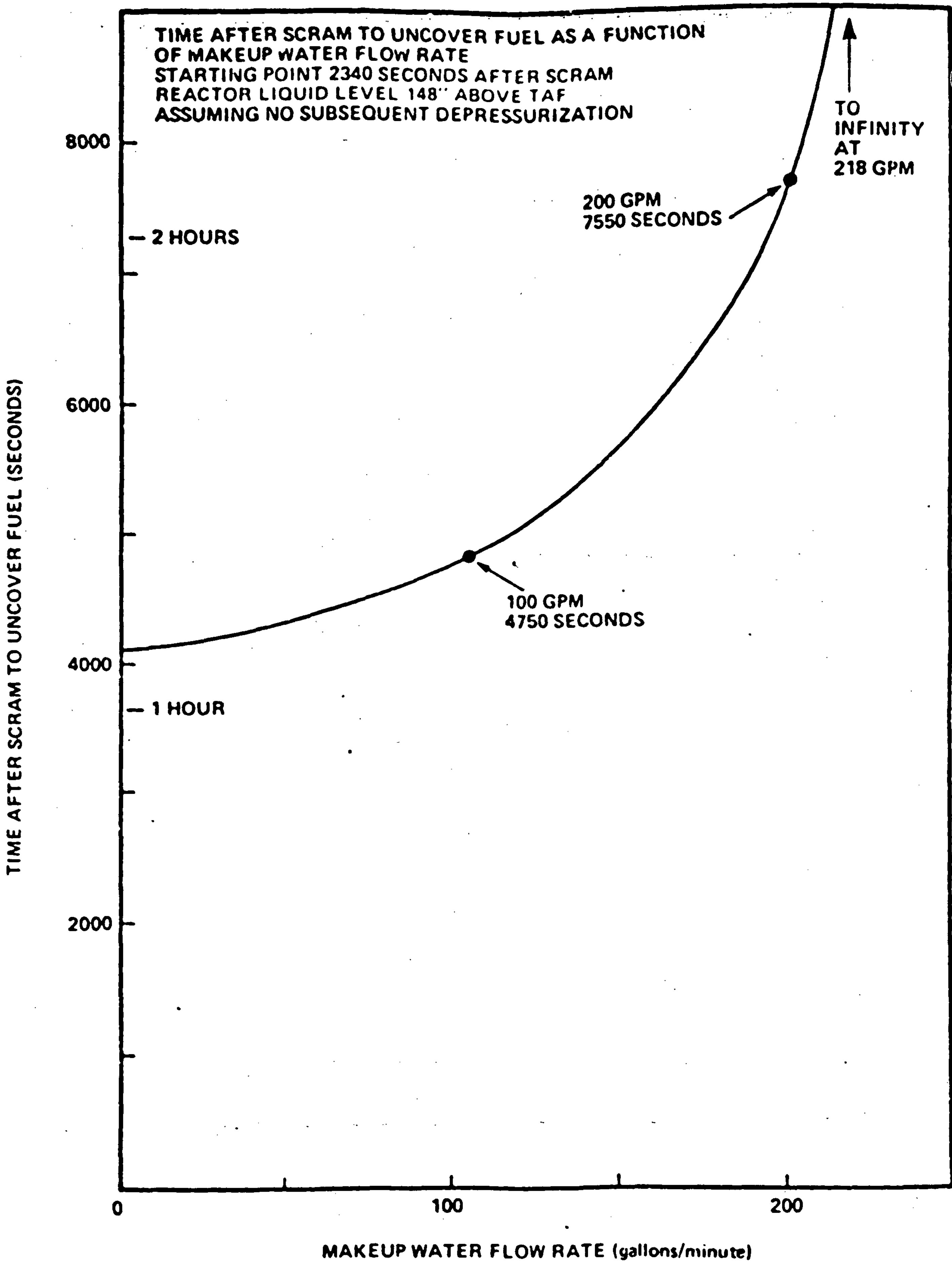

TIME AFTER SCRAM TO UNCOVER FUEL AS A FUNCTION
OF MAKEUP WATER FLOW RATE
STARTING POINT 2340 SECONDS AFTER SCRAM
REACTOR LIQUID LEVEL 148'' ABOVE TAF
ASSUMING NO SUBSEQUENT DEPRESSURIZATION
TO INFINITY AT 218 GPM
200 GPM
7550 SECONDS
100 GPM
4750 SECONDS
8000
2 HOURS
6000
4000
1 HOUR
2000
0
100
200
TIME AFTER SCRAM TO UNCOVER FUEL (SECONDS)
MAKEUP WATER FLOW RATE (gallons/minute)

Figure 2.

BASES FOR DEVELOPMENT OF FIGURE 1

Figure I presents the results of calculations which were made to determine the amounts of coolant which had depressurization not been accomplished about 40 minutes after scram, would have been necessary to keep the fuel covered.

The line labeled reactor decay heat represents the amount of decay heat released by the core starting approximately 2340 seconds after reactor scram. This curve was obtained by numerical integration of the ANS decay heat curves (Shure + actinides + uncertainty), corrected for finite operation (equivalent of 280 full power days). The values used for the numerical integration appear in Tables 1 and 2.

The lines in figure 1 which are labeled 100, 200, 218 gpm show the amount of heat which could be removed by introduction of coolant (at 100, 200, and 218 gpm), starting at 2340 seconds. For these calculations it is assumed that water is introduced at 90°F, and is exhausted as saturated vapor at 1080 psig. Thus a continuous flow of 100 gpm can remove 16.6 mw of heat (945,000 BTU/min)--note that the 100 gpm line is constructed by connecting a straight line through the points: x = 2340 seconds, y = 0 BTU, and x = 8340 seconds, y = 94.5×10^6 BTU. The lines of 200 and 218 gpm are constructed similarly.

At 2340 seconds, the water level observed in the reactor was 148 inches above the top of the active full (TAF); this corresponds to 3095 ft^3 of water. Since the initial condition of the vessel was one of saturation at 1080 psig, the density of the water in the reactor vessel was ~ 45.5 lb/ft^3 and the latent heat of vaporization for the water at that pressure was 631 BTU/lb. The amount of decay heat necessary necessary to boil off the initial volume of water was:

$$148 \text{ in. above TAF} \times \frac{20.9 \text{ ft}^3}{\text{in. of reactor water level above TAF}} \times 45.5 \text{ lb/ft}^3 \times 631 \text{ BTU/lb} = 89 \times 10^6 \text{ BTU}$$

Thus, if there were no makeup water provided, the time at which the core would become uncovered is ≈ 4050 seconds after scram (point on the decay heat curve where 89 x 10^6 BTU are liberated from the core).

If, however, makeup water at the rate of 100 gpm were initiated at time 2340 seconds after scram, at time 4050 seconds after scram the makeup water would have been able to account for 27.5 x 10^6 BTU of decay heat and the core would not have been uncovered. The difference between the integrated decay heat, and the makeup water cooling capacity represents the net amount of heat used to boil off the initial reactor vessel inventory.

Proceeding to time ≈ 4750 seconds after scram, the integrated decay heat is ≈ 127.5 x 10^6 BTU, and the heat removal capability of the makeup water introduced at 100 gpm is ≈ 38.5 x 10^6 BTU. The difference between these values is 89 x 10^6 BTU, enough to uncover the active fuel. Thus, the dashed line - labeled "Demarcation line for uncovering the active fuel" was plotted 89 x 10^6 BTU below the integrated reactor decay heat curve. Therefore, introducing 100 gpm of coolant will assure that the active fuel is covered until 4750 seconds after scram.

Similarly, introducing coolant at a rate of 200 gpm will assure that the active fuel is covered until ≈7520 seconds after scram (note at 7520 seconds after scram the 200 gpm line is 89 x 10^6 BTU below the integrated decay heat curve (7520 seconds, 162.5 x 10^6 BTU, is the point of intersection of the 200 gpm line and the dashed "Demarcation" line).

Similarly, examination of figure 1 reveals that the 218 gpm line will never cross the demarcation line; consequently, introduction of coolant at 218 gpm will prevent the core from ever being uncovered.

Summarizing these findings -- starting at a reactor pressure of 1080 psig with a liquid level of 148 in. above TAF, at 2340 seconds after scram, introduction of cooling water at 90°F will result in the following:

Coolant Rate, Gallons/Minute	Time After Scram To Uncover Fuel, Seconds
0	4050
100	4750
200	7550
218	∞

Figure 1 is a convenient tool for determining the makeup water's fuel covering capability for the aforementioned event.

Figure 2 is a cross plot of Figure 1 to directly show the time from scram to core uncovering as a function of makeup coolant flow rate.

DECAY HEAT CALCULATION

From TVA's plan for Evaluation Repair and Return to Service Part VI Section E page 59 (5/28/75) the reactor had 280 full power days prior to the fire. This equates to,

$$280 \text{ days} \times 86{,}500 \ \frac{\text{sec}}{\text{day}} = 24 \times 10^{6} \text{ seconds}$$

which according to the Shure decay heat curve represents a 0.085% full power correction term (to account for the fact that the reactor had not been in operation for infinite time).

The heat removal capability of cooling water:

assume, $T_{in} = 90°F$, $h_f = 58$ BTU/lb

$P_{out} = 1080$ psig, hg = 1189 BTU/lb

$$1 \ \frac{\text{gallon}}{\text{minute}} \times .1337 \ \frac{\text{ft}^3}{\text{gallon}} \times \frac{\text{lb}}{.016 \ \text{ft}^3} \times (1189 - 58) \ \frac{\text{BTU}}{\text{lb}}, \text{ or}$$

$$1 \text{ gpm} = 9450 \ \frac{\text{BTU}}{\text{min}}, \text{ which is equivalent to, } 0.166 \text{ MW.}$$

TABLE 1

REACTOR DECAY HEAT

Time After Scram (seconds)	Decay Heat % of Full Power	Interval Avg. % of Full Power	Total Interval Heat ($x10^6$ BTU)
40	5.015		
100	4.215	4.615	8.66
160	3.78	3.997	7.52
220	3.53	3.655	6.86
300	3.265	3.398	8.51
400	3.06	3.16	9.9
600	2.765	2.913	18.2
660	2.665	2.715	5.1
1000	2.5	2.583	27.4
1300	2.1	2.3	21.6
1900	1.99	2.045	38.4
2500	1.76	1.875	35.3
4300	1.59	1.67	94.5
6100	1.44	1.515	85.5
7900	1.3	1.37	77.3
9100	1.2	1.25	47.0
15,000	1.02	1.11	209.0
21,100	.915	.965	181.0
27,100	.815	.865	163.0
39,100	.715	.765	288.0
51,100	.7	.707	265.0

1% fp - sec = $3.13x10^4$ BTU

Table 2

Decay Heat Subsequent to Time of

First Reactor Depressurization 2340

Seconds After Scram (Water Level 148" above TAF)

Time Sec	Decay Heat per Interval (BTU x 10^6)	Decay Heat Integral (BTU x 10^6)
2340		
2500	8.95	9
4300	94.5	103.5
6100	85.5	189
7900	77.3	266
9100	47	313
15,100	209	522
21,100	181	703
27,100	163	866
39,100	288	1154
51,000	265	1419

TABLE 3

PLANT SOURCES OF MAKEUP WATER

	Number Provided	Flow rate for each Pump	Comments
Control rod drive pumps	1 per unit plus 1 shared spare All 3 could have been aligned for 1 Unit	Normal operation: 100 gpm at reactor pressure of 1080 psig 182 gpm at reactor pressure of 0 psig With pump test: bypass line open 225 gpm at reactor pressure of 1080 psig	Available throughout event (used in normal operating mode)
Standby liquid control pumps	Two	56 gpm each at reactor pressure of 1080 psig (positive displacement pumps)	Available for majority of post scram conditions (powerless for several hours due to fire damage. Could have been energized in less than 1 hr. if needed.
Reactor Core Isolation Cooling Pump (Steam turbine driven)	One	600 gpm at reactor pressure of 1080 psig	Unavailable due to damage to connecting wiring-however could have been made available in less than 1 hour if needed

Table 3 (continued)

Plant Sources of Makeup Water

	Number Provided	Flow Rate for Each Pump	Comments
High pressure Coolant Injection Pump (Steam turbine driven)	One	5000 gpm at reactor pressure of 1080 psig	Unavailable due to fire damage
Feedwater Pumps (Steam turbine driven)	Three	11200 gpm	Unavailable due to closure of main steam isolation valves, inadequate steam supply
Condensate and Condensate Booster Pumps	Three	10800 gpm (low pressure < 350 psig)	Available throughout event - could only be used in low pressure mode
Condensate Pumps	Three	10800 gpm (low pressure)	Available throughout event - could only be used in low pressure mode
Low Pressure Coolant Injection Pumps	Four	10000 gpm (low pressure)	Two unavailable due to fire damage to cables
Core Spray Pumps	Four	3125 gpm (low pressure)	Two unavailable due to fire damage to cables
Residual Heat Removal Service Water Pumps	Two	4500 gpm (low pressure)	Both available

ATTACHMENT II

LETTER FROM THE NRC TO THE TVA DATED MAY 9, 1975, WITH ENCLOSURES

AMENDMENT NO. 9 TO LICENSE NO. DPR-33

AMENDMENT NO. 6 TO LICENSE NO. DPR-52

SAFETY EVALUATION

FEDERAL REGISTER NOTICE

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

May 9, 1975

Docket Nos. 50-259
and 50-260

Tennessee Valley Authority
ATTN: Mr. James E. Watson
Manager of Power
818 Power Building
Chattanooga, Tennessee 37201

Gentlemen:

The Commission has issued the enclosed Amendment No. 9 to Facility License No. DPR-33 and Amendment No. 6 to Facility License No. DPR-52 for the Browns Ferry Nuclear Plant, Units 1 and 2. These amendments include Change No. 10 to the respective Technical Specifications and are in response to your request dated May 8, 1975.

The amendments revise the Technical Specifications, taking into account the present condition of plant systems, so as to ensure that the two units will remain in a safe and stable posture during the shutdown time interval between now and our subsequent authorization for the start of repair and restoration of the facility.

Copies of the related Safety Evaluation and the Federal Register Notice are also enclosed.

Sincerely,

Robert A. Purple, Chief
Operating Reactors Branch #1
Division of Reactor Licensing

Enclosures:
1. Amendment No. 9 to License No. DPR-33
2. Amendment No. 6 to License No. DPR-52
3. Safety Evaluation
4. Federal Register Notice

cc w/enclosures:
See next page

cc w/enclosures:
Robert H. Marquis
General Counsel
629 New Sprankle Building
Knoxville, Tennessee 37919

Athens Public Library
South and Forrest
Athens, Alabama 35611

Mr. Thomas Lee Hammons
Chairman, Limestone County Board
of Revenue
Athens, Alabama 35611

Anthony Z. Roisman, Esquire
Berlin, Roisman & Kessler
1712 N Street, NW
Washington, D.C. 20036

cc w/ enclosures & incoming
Ira L. Myers, M.D.
State Health Officer
State Department of Public Health
State Office Building
Montgomery, Alabama 36104

Mr. Dave Hopkins
Environmental Protection Agency
1421 Peachtree Street, NE
Atlanta, Georgia 30309

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

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-259

BROWNS FERRY NUCLEAR PLANT UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 9
License No. DPR-33

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Tennessee Valley Authority (the licensee) dated May 8, 1975, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations; and

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

2. Accordingly, the license is amended by a change to the Technical Specifications as indicated in the attachment to this license amendment and Paragraph 2.C.(2) of Facility License No. DPR-33 is hereby amended to read as follows:

"(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications, as revised by issued changes thereto through Change No. 10."

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert A. Purple, Chief
Operating Reactors Branch #1
Division of Reactor Licensing

Attachment:
Change No. 10 to Technical
Specifications

Date of Issuance: May 9, 1975

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

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-260

BROWNS FERRY NUCLEAR PLANT UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 6
License No. DPR-52

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Tennessee Valley Authority (the licensee) dated May 8, 1975, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations; and

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

2. Accordingly, the license is amended by a change to the Technical Specifications as indicated in the attachment to this license amendment and Paragraph 2.C.(2) of Facility License No. DPR-52 is hereby amended to read as follows:

"(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications, as revised by issued changes thereto through Change No. 10."

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert A. Purple, Chief
Operating Reactors Branch #1
Division of Reactor Licensing

Attachment:
Change No. 10 to Technical Specifications

Date of Issuance: May 9, 1975

<u>ATTACHMENT TO LICENSE AMENDMENT NO. 9 TO LICENSE NO. DPR-33</u>

<u>AND LICENSE AMENDMENT NO. 6 TO LICENSE NO. DPR-52</u>

<u>(CHANGE NO. 10 TO THE TECHNICAL SPECIFICATIONS)</u>

<u>DOCKET NOS. 50-259 AND 50-260</u>

Revise Appendix A as indicated.

FACILITY OPERATING LICENSES DPR-33 AND DPR-52

TEMPORARY TECHNICAL SPECIFICATIONS AND BASES

FOR

BROWNS FERRY NUCLEAR PLANT

UNIT 1 AND UNIT 2

INTRODUCTION

As a result of equipment damage resulting from the fire which occurred at the facility on March 22, 1975, various technical specifications set forth in the technical specifications governing operation,[1] are no longer appropriate. For example, certain items are not "operable" as defined in the technical specifications governing facility operation, but will nontheless perform the functions necessary to assure plant safety. These Temporary Technical Specifications and their respective bases will provide assurance that the plant equipment and systems in their present status and configurations are adequate to provide for plant safety for the work activities being carried out at the facility during the period in which these specifications are effective.

Certain work is planned for the period during which these specifications are to be effective. To assure that certain components, which will be required to function during restoration and repair of the facility (which is not covered by their specifications) are protected during this period, some components will be rerouted and reconnected to obtain a configuration which is protected during repair. Also, certain instruments or components will be changed from automatic operation to manual operation or will be locked in a safe configuration. These modifications will be performed during the period in which these temporary specifications are effective.

However, during the period in which these specifications are effective, it is intended that no intentional action will be taken which would entirely remove a safety system from service except for essential maintenance. To assure this, no intentional action will be taken to cause components required to achieve the objectives of these technical specifications to lose their functional capability except for maintenance

[1] The term "technical specifications governing facility operation" means Facility Operations Licenses DPR-33 and DPR-52, Technical Specification and Bases for Browns Ferry Nuclear Plant Unit 1 and Unit 2, Limestone County, Alabama, Tennessee Valley Authority, Docket Nos. 50-259, 20-260, effective prior to May 5, 1975.

required to prevent a substantial increase in the risk of failures to the component or any other system or component presently functional in the facility. However, automatic features in components may be altered when such actions enable such components to be operated manually or locked in a safe configuration to achieve the status specified for such component in the "TVA Safety Analysis of the BFNP Units 1 and 2 During Operations Related to Removal of Damaged Cables, Cable Trays, and Conduit," dated May 1, 1975, and included as Part 6, Section E, of the Plan for Evaluation, Repair, and Return to Service of Browns Ferry Units 1 and 2 (March 22, 1975, Fire).

These temporary technical specifications will be effective immediately upon issuance and will remain effective until superseded. During the period in which these technical specifications are effective, each reactor will be maintained in cold shutdown. In this condition, the mode switch shall be locked in shutdown, the reactor coolant temperature will be equal to or less than 212° F, and the reactor will be maintained in the vented position.

These temporary specifications do not authorize removal of the fire-affected features or restoration of the facility. Before such activities are authorized, these temporary specifications will be superseded in the whole or in part.

These temporary specifications may otherwise be supplemented, amended, or modified from time to time.

The temporary specifications set forth in this addendum, are in the form of specified sections, subsections, or page changes as identified on the particular change. These changes are temporary only.

1.0 DEFINITIONS

This technical specification (pages 2-7) remains unchanged.

SAFETY LIMIT

1.1/2.1 FUEL CLADDING INTEGRITY

Delete present section (pages 8-25) in its entirety and replace with the following:

SAFETY LIMIT

1.1 FUEL CLADDING INTEGRITY

Applicability

Applies to the interrelated variables associated with fuel thermal behavior.

Objective

To establish limits which ensure the integrity of the fuel cladding.

Specification

A. Whenever the reactor is in the cold shutdown condition with irradiated fuel in the reactor vessel, the water level shall not be less than 17.7 inches above the top of the normal active fuel zone.

LIMITING SAFETY SYSTEM SETTING

2.1 FUEL CLADDING INTEGRITY

Applicability

Applies to the devices which are provided to prevent the reactor system safety limits from being exceeded.

Objective

To define the level of the process variables at which protective action is initiated to prevent the fuel cladding integrity safety limit from being exceeded.

Specification

The limiting safety system settings shall be as specified below:

A. Manual operation under procedural controls shall be established to assure the reactor water level remains $\geq$378 inches above vessel zero.

1.1/2.1 BASES: LIMITING SAFETY SYSTEM SETTINGS RELATED TO FUEL CLADDING INTEGRITY

The objective of this specification is to assure that irradiated fuel in the reactor vessel remains covered with water at all times. When plant equipment as specified in facility technical specification is available, this objective is accomplished automatically. This is no longer possible. Accordingly, procedural control of the water level is required as set forth in the specification 3.2.A.1.

With the reactor in cold shutdown, rapid makeup of coolant inventory is not required. The provisions for manual operation under procedural controls provided by the "TVA Safety Analysis of the BFNP Units 1 and 2 During Operations Related to Removal of Damaged Cables, Cable Trays, and Conduit," dated May 1, 1975, and included as Part 6, Section E, of the Plan for Evaluation, Repair, and Return to Service of Browns Ferry Units 1 and 2 (March 22, 1975, Fire), will assure timely isolation of leaks by requiring redundant level alarms and indicators; double isolation features, one of which shall be operative from the control room, will prevent draining of the vessel as a result of any single active component failure. Also, in the unlikely event that coolant inventory is required, a core spray pump, capable of delivering rated flow, will be available to provide the required vessel makeup capability.

The limiting safety systems scram setpoints are not capable of performing their intended function since the automatic actions they would initiate are unavailable. However, the facility has been placed in a condition that prevents the need for these actions.

1.2/2.2 REACTOR COOLANT SYSTEM INTEGRITY

Delete present specification in its entirety (pages 27-30) and replace with the following:

SAFETY LIMIT

1.2 REACTOR COOLANT SYSTEM INTEGRITY

Applicability

Applies to limits on reactor coolant system pressure.

Objective

To establish a limit below which the integrity of the reactor coolant system is not threatened due to an overpressure condition.

Specification

A. The pressure at the lowest point of the reactor vessel shall not exceed 1,375 psig whenever irradiated fuel is in the reactor vessel.

LIMITING SAFETY SYSTEM SETTING

2.2 REACTOR COOLANT SYSTEM INTEGRITY

Applicability

Applies to the devices which are provided to prevent the reactor system safety limits from being exceeded.

Objective

To define the protective action to prevent the pressure safety limit from being exceeded.

Specification

A. The reactor vessel head vent valves shall be kept in the open position.

B. Four safety relief valves which have accumulators shall be functional on each unit with two valves powered from each of two 250V DC MOV boards.

1.?/2.2 BASES: REACTOR COOLANT SYSTEM INTEGRITY

With the reactor in the cold shutdown condition and with redundant RHR shutdown cooling capability, pressurization of the reactor vessel is prevented. In the unlikely event that shutdown cooling capability from the RHR system is lost, manual operation of any one of the four available relief valves prevents pressurization of the reactor pressure vessel above 50 psig preserving the safety limit. This keeps the pressure low enough to permit addition of water by a core spray pump.

Checks of the reactor head vent and relief valve position indicating lights each shift confirm control power available for actuation.

3.1/4.1 REACTOR PROTECTION SYSTEM

Delete present specification (pages 31-49) in its entirety and replace with the following:

LIMITING CONDITIONS FOR OPERATION

3.1 REACTOR PROTECTION SYSTEM

Applicability

Applies to the instrumentation and associated devices which initiate a reactor scram.

Objective

To assure the operability of the reactor protection system.

Specification

A. The setpoints and minimum number of channels required to be functional are listed in Table 3.1.A/4.1.A.

SURVEILLANCE REQUIREMENTS

4.1 REACTOR PROTECTION SYSTEM

Applicability

Applies to the surveillance of the instrumentation and associated devices which initiate reactor scram.

Objective

To specify the type and frequency of surveillance to be applied to the protection instrumentation.

Specification

A. Instrument channels shall be functionally tested and calibrated as indicated in Table 3.1.A/4.1.A.

TABLE 3.1.A/4.1.A

REACTOR PROTECTION SYSTEM INSTRUMENTATION

Minimum Number of Functional Channels per Trip System	Trip Function	Trip Level Setting	Functional Test Frequency	Alarm Level Setting	Calibration Frequency
1	Manual scram	N/A	once/month	N/A	N/A
2 (1)	SRM high-high count	$\leq 5 \times 10^5$ cps	once/week	$\leq 5 \times 10^5$ cps (upscale high high) $\leq 10^5$ cps (upscale high) ≥ 3 cps (downscale)	once/3 months

Notes for Table 3.1.A/4.1.A

(1) The SRM's are presently connected in the non-coincident mode in the RPS, where any 1 SRM upscale high high will initiate a scram. Of the 4 SRM channels, only 2 are required to be functional.

3.1/4.1 BASES: REACTOR PROTECTION SYSTEM

The reactor protection system will be available for rod insertion in the unlikely event that a rod becomes withdrawn. This is highly improbable because the reactor mode switch is locked in the shutdown position and the control rod directional solenoids are electrically disarmed. The reactor protection system is functional as described in Section 7.2 of the FSAR with the manual scram and the SRM's required functional. The SRM's have been placed in the non-coincidence scram mode to provide automatic scram from high-high count rate.

3.2/4.2 PROTECTIVE INSTRUMENTATION

Delete present specifications (pages 50-107) in its entirety and replace with the following:

LIMITING CONDITIONS FOR OPERATION

3.2 PROTECTIVE INSTRUMENTATION

Applicability

Applies to the plant instrumentation which initates and controls a protective function.

Objective

To assure the operability of protective instrumentation.

Specification

A. Surveillance Instrumentation

The limiting conditions for the instrumentation that provides surveillance information required in the period during which these specifications are effective are given in Table 3.2.A.

1. Reactor water level shall be maintained at $\geq$ 555" above vessel zero.

SURVEILLANCE REQUIREMENTS

4.2 PROTECTIVE INSTRUMENTATION

Applicability

Applies to the surveillance requirement of the instrumentation that initiates and controls protective function.

Objective

To specify the type and frequency of surveillance to be applied to protective instrumentation.

Specification

A. Surveillance Instrumentation

The surveillance requirements associated with the equipment listed in Table 3.2.A shall be that indicated in Table 4.2.A.

1. The reactor water level shall be verified once each shift.

TABLE 3.2.A
SURVEILLANCE INSTRUMENTATION
(POST ACCIDENT INSTRUMENTATION)

Minimum No. of Operable Instrument Channels	Instrument No.	Instrument	Type Indication and Range	Alarm Setting	Notes
2	LI-3-206 or LR-3-53 or LI-3-55	Reactor Water Level	Indicator 0" to 60" Recorder 0" to 60" Indicator 0" to 60" Indicator 0" to 400"	Low $\geq$ 27", High $\leq$ 39"	(1)
2	PI-3-54 PR-3-53	Reactor Pressure	Indicator 0-1200 psig Recorder 0-1200 psig	High $\leq$ 1040 psig	(1)
2	PR-64-50 and PI-64-67	Drywell Pressure	Recorder 0-80 psig Indicator 0-80 psig		(1)
2	TI-64-52A and TR-64-52	Drywell Temperature	Indicator 0-400°F. Recorder 0-400°F.	High $\leq$ 145°F	(1)
2	TI-64-55A and TIS-64-55	Suppression Chamber Water Temperature	Indicators 0-400°F.	High $\leq$ 90°F.	(1)
1	LI-64-54A	Suppression Chamber Water Level	Indicator -25" to +25"		(1)
1	NA	Control Rod Position	Continuity		(2)
2	SRM A, B, C, D	Neutron Monitoring	Indicator and Recorder 0.1 to 10^{+6} cps -100 to +10 sec (period)	Downscale $\geq$ 3 cps Retract permit $\geq$ 100 cps Upscale High $\leq$ 10^5 cps Upscale High-High $\leq 5 \times 10^5$ cps Period $\geq$ 30 sec.	(1) (3)

NOTES FOR TABLE 3.2.A

(1) If one of the instrument channels monitoring a parameter should become incapable of performing its intended function, all operations which could affect the associated system will be suspended until the item is returned to service.

(2) The control rod position indicator full-in switches will be operable for every control rod and will provide indication in the control room, or the control rod position will be verified to be full-in by a continuity check of the full-in limit switches at the local panel.

(3) The following Source Range Monitoring Channels will be operable and will provide count rate indication and alarms in the control room:

SRM Channel A or C

SRM Channel B or D

The alarms in the control room will be as follows:

SRM TRIPS

Trip Function	Trip Action
SRM upscale or inoperative	Annunciator, amber light
SRM Downscale	Annunciator, white light

In addition, the SRM's have been placed in the non-coincidence scram mode to provide for protection against high neutron flux.

TABLE 4.2.A

MINIMUM TEST AND CALIBRATION FREQUENCY FOR SURVEILLANCE INSTRUMENTATION

(POST ACCIDENT INSTRUMENTATION)

	Instrument Channel	Calibration Frequency (3)	Instrument Check
1)	Reactor Water Level	Once/6 months	Each Shift
2)	Reactor Pressure	Once/6 months	Each Shift
3)	Drywell Pressure	Once/6 months	Each Shift
4)	Drywell Temperature	Once/6 months	Each Shift
5)	Suppression Chamber Water Temperature	Once/6 months	Each Shift
6)	Suppression Chamber Water Level	Once/6 months	Once/Week (4)
7)	Control Rod Position	NA	Each Shift (1)
8)	Neutron Monitoring (SRM)	Once/3 months (2)	Each Shift
9)	Drywell Pressure (PS-64-67)	Once/6 months	NA

NOTES FOR TABLE 4.2.A

(1) Control rod position indication to verify the full-in position will be maintained on unit 2. Unit 1 control rods which do not have position indication in the control room will be monitored at the local panel by use of the full-in limit switches at intervals not to exceed once/3 days.

(2) The SRM will be functionally tested on frequency of once per week.

(3) Instrument calibration will be verified upon restoration, then at the frequency listed.

(4) Instrument check will be to observe instrument response as transmitter is removed and returned to service. A redundant instrument is not available for comparison.

3.2/4.2 BASES: PROTECTIVE INSTRUMENTATION

For each parameter monitored, as listed in Table 3.2.A, there are equivalent or redundant channels of instrumentation as noted. By comparing readings between two channels, a near continuous surveillance of instrument performance is available. Any deviation in readings will necessitate an early recalibration, thereby maintaining the quality of the instrument readings.

The redundant monitoring of reactor vessel level will be sufficient to provide the operator with information with which to determine the adequacy of the coolant inventory. In addition, overfilling of the reactor vessel will be detected by verifying that reactor vessel water level is not rising to an abnormally high level. The approach of abnormal conditions will be brought to the operator's attention by audible alarms.

Instrumentation has been selected to provide control room indication of adequate information regarding status of the drywell pressure and temperature and torus water level and temperature. Monitoring of information concerning key primary containment parameters will ensure that sufficient control of these parameters can be manually initiated in a timely manner.

A minimum of two channels of the SRM's, each on separate power supplies, will be operable. Each channel will provide indication of count rate and provide alarms as listed in Table 7.5-1 of the FSAR.

An automatic scram from high-high count rate of any SRM's channel as well as a manual scram will be available. One control rod drive pump each for units 1 and 2 will be kept in operation and all the accumulators will be kept charged with nitrogen and water. The scram pilot valves and pneumatically controlled valves to scram each control unit will be operable.

3.2/4.2 BASES: PROTECTIVE INSTRUMENTATION (CONTINUED)

Control rod position indication to verify the full-in position will be maintained on unit 2. The continuity of all the full-in limit switches on unit 1 will be monitored periodically. Under the conditions to be maintained in Temporary Technical Specification 3.3, the probability of a control rod withdrawal is significantly lower than that following a scram from normal conditions.

The reactor water level will be nominally maintained at a level greater than 27" but less than 39" as indicated on LR-3-53. Above or below this level will give an alarm of "Reactor Water Level Abnormal." 27" indication on LR-3-53 corresponds to 555" above vessel zero.

3.3/4.3 REACTIVITY CONTROL

Delete this specification in its entirety (pages 108-121) and replace with the following

LIMITING CONDITIONS FOR OPERATION

3.3 REACTIVITY CONTROL

Applicability

Applies to the operational status of the control rod system.

Objective

To assure the ability of the control rod system to control reactivity.

Specification

A. All control rods shall be inserted in the full-in position.

B. The directional control valves shall be disarmed electrically for all control rods.

C. The manual valves in the drive water supply shall be in the shut position to prohibit rod movement.

D. The control rod accumulators shall be charged.

E. Two SRM channels shall be functional.

F. The control rod drive pump shall be in service.

SURVEILLANCE REQUIREMENTS

4.3 REACTIVITY CONTROL

Applicability

Applies to the surveillance requirements of the control rod system.

Objective

To verify the ability of the control rod system to control reactivity.

Specification

A. Control rod position shall be verified in accordance with Table 4.2.A.

B. Each directional control valve shall be verified to be electrically disarmed at intervals not to exceed once every 3 days.

C. Each control rod insert drive valve shall be verified closed at intervals not to exceed once every 3 days.

D. The accumulator pressure shall be checked once a day.

E. The count rate shall be recorded once each shift.

F. The control rod drive pump discharge pressure shall be checked once each shift.

3.3/4.3 BASES: REACTIVITY CONTROL

To prevent an inadvertent or spurious withdrawal of a control rod the directional control valves of each control rod have been electrically disarmed. As a further precaution, the valve in the drive water supply to each rod will be closed. In the unlikely event that a control rod does become withdrawn, two channels of the SRM's are required to be available for visual indication of neutron level. Although the SRM's may not immediately respond to a single rod movement, they would be adequate to monitor the approach to criticality. Also the SRM's are connected in the non-coincidence scram mode to provide rapid rod insertion from a high-high count rate on either SRM. Additionally, a manual scram will also be available. To assure that the control rods can be scrammed, the control rod accumulators are required to be charged with nitrogen and water pressure and a control rod drive pump is required to be in service. To provide additional indication of a control rod withdrawal, the control rod position indicator full-in switches will be functional for every control rod and will provide indication in the control room or the control rod position will be verified to be full-in by a continuity check of the full-in limit switch at the local panel.

3.4/4.4 STANDBY LIQUID CONTROL SYSTEM

Delete present section in its entirety (pages 122-129) and replace with the following:

LIMITING CONDITIONS FOR OPERATION

3.4 STANDBY LIQUID CONTROL SYSTEM

Applicability

Applies to the operating status of the Standby Liquid Control System.

Objective

To assure the availability of a system with the capability to shut down the reactor and maintain the shutdown condition without the use of control rods.

Specification

A. Normal System Availability

1. The standby liquid control system shall be functional at all times.

2. The injection line manual isolation valve 63-524 will be closed except when the system is required for emergency injection.

SURVEILLANCE REQUIREMENTS

4.4 STANDBY LIQUID CONTROL SYSTEM

Applicability

Applies to the surveillance requirements of the Standby Liquid Control System.

Objective

To verify the operability of the Standby Liquid Control System.

Specification

A. Normal System Availability

The operability of the Standby Liquid Control System shall be verified by the performance of the following tests:

1. At least once per month each pump loop shall be functionally tested.

2. At least once during each operating cycle:

a. Check that the setting of the system relief valves is 1425 ± 75 psig.

b. Manually initiate the system, except explosive valves. Pump boron solution through the recirculation path and back to the Standby Liquid Control Solution Tank. Minimum pump flow rate of 39 gpm

LIMITING CONDITIONS FOR OPERATION

3.4 STANDBY LIQUID CONTROL SYSTEM

B. Operation with Inoperable Components:

1. From and after the date that a redundant component is made or found to be inoperable, Specification 3.4.A.1 shall be considered fulfilled. The inoperable component shall be returned to service within seven days.

SURVEILLANCE REQUIREMENTS

4.4 STANDBY LIQUID CONTROL SYSTEM

against a system head of 1275 psig shall be verified. After pumping boron solution, the system shall be flushed with demineralized water.

c. Manually initiate one of the Standby Liquid Control System loops and pump demineralized water into the reactor vessel.

This test checks explosion of the charge associated with the tested loop, proper operation of the valves and pump operability. Replacement charges shall be selected such that the age of charge in service shall not exceed five years from the manufacturers assembly date.

d. Both systems, including both explosive valves, shall be tested in the course of two operating cycles.

B. Surveillance with Inoperable Components:

1. When a component is found to be inoperable, its redundant component shall be demonstrated to be operable immediately and daily thereafter until the inoperable component is repaired.

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LIMITING CONDITIONS FOR OPERATION

3.4 STANDBY LIQUID CONTROL SYSTEM

C. Sodium Pentaborate Solution

At all times when the Standby Liquid Control System is required to be operable the following conditions shall be met:

1. The net volume - concentration of the Liquid Control Solution in the liquid control tank shall be maintained as required in Figure 3.4.1.

2. The temperature of the liquid control solution shall be maintained above the curve shown in Figure 3.4.2. This includes the piping between the standby liquid control tank and the suction inlet to the pumps.

SURVEILLANCE REQUIREMENTS

4.4 STANDBY LIQUID CONTROL SYSTEM

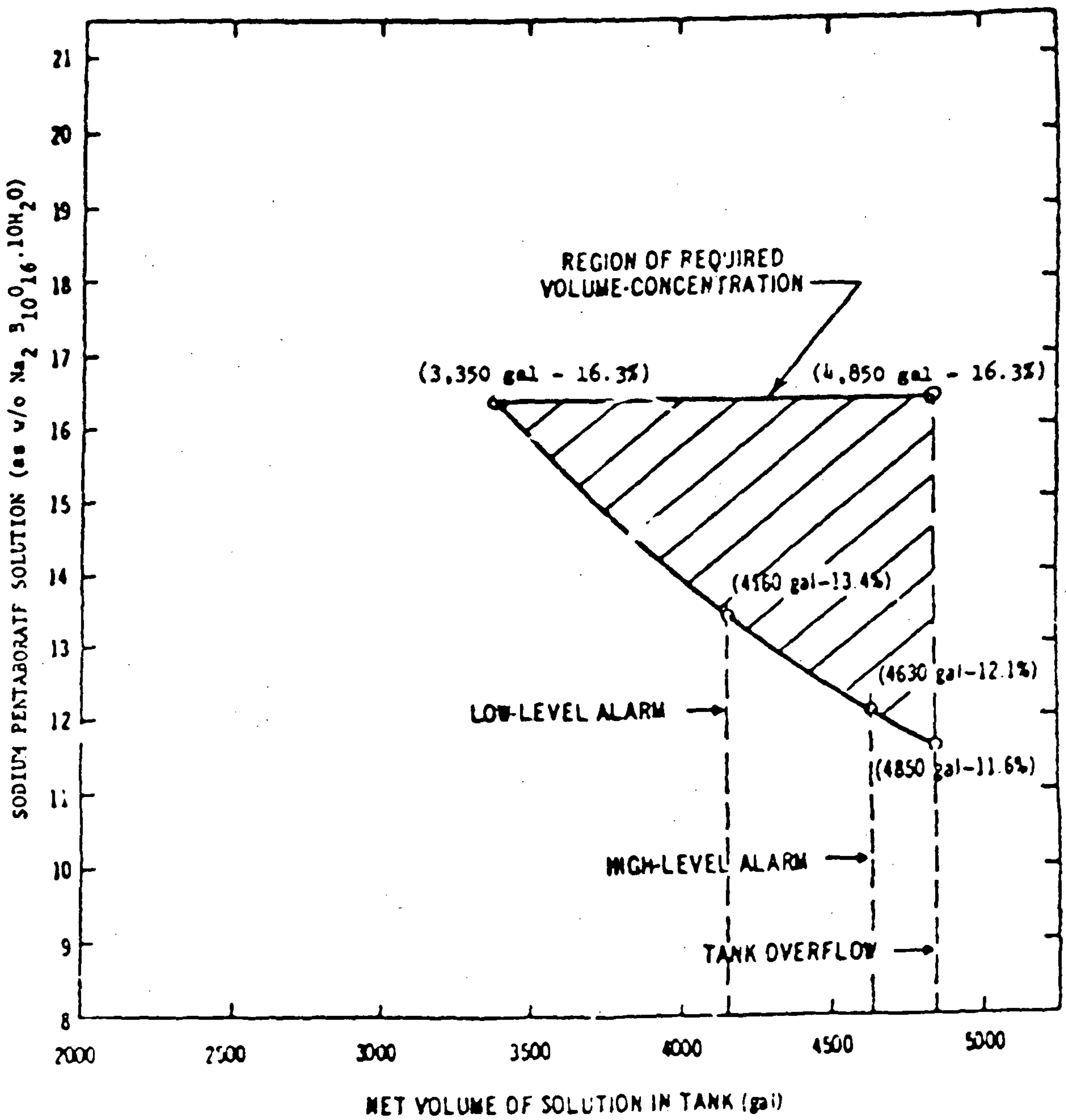
C. Sodium Pentaborate Solution

The following tests shall be performed to verify the availability of the Liquid Control Solution:

1. Volume: Check at least once per day.

2. Temperature: Check at least once per day.

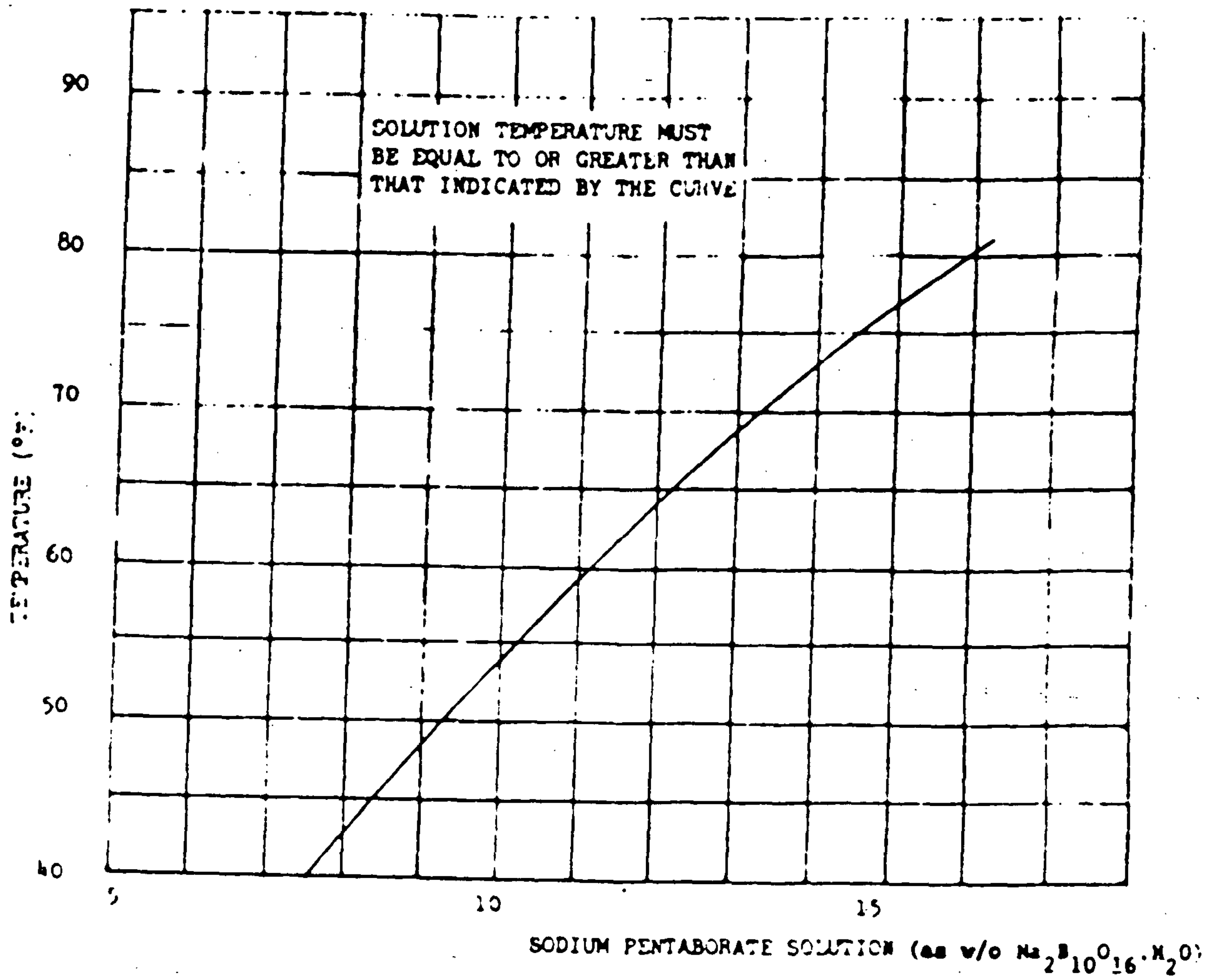
3. Concentration: Check at least once per month. Also check concentration any time water or boron is added to the solution or solution temperature is below the temperature required in Figure 3.4.2.



BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

SODIUM PENTABORATE SOLUTION
VOLUME-CONCENTRATED REQUIREMENTS

FIGURE 3.4-1



BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

SODIUM PENTABORATE SOLUTION
TEMPERATURE REQUIREMENTS

FIGURE 3.4-2

BASES: STANDBY LIQUID CONTROL SYSTEM (Continued)

C. (Continued)

The volume concentration requirement of the solution are such that should evaporation occur from any point within the curve, a low level alarm will annunciate before the temperature-concentration requirements are exceeded.

The quantity of stored boron includes an additional margin (25 percent) beyond the amount needed to shut down the reactor to allow for possible imperfect mixing of the chemical solution in the reactor water.

A minimum quantity of 4,160 gallons of solution having a 13.4 percent sodium pentaborate concentration or the equivalent is required to meet this shutdown requirement as defined in Figure 3.4.1.

Experience with pump operability indicates that the monthly test, in combination with the tests during each operating cycle, is sufficient to maintain pump performance. Various components of the system are individually tested periodically, thus making unnecessary more frequent testing of the entire system.

The solution temperature and volume are checked at a frequency to assure a high reliability of operation of the system should it ever be required.

BASES: STANDBY LIQUID CONTROL SYSTEM

A. The purpose of the liquid control system is to provide the capability of maintaining the reactor in a cold shutdown condition assuming that none of the withdrawn control rods can be inserted. To meet this objective, the liquid control system is designed to inject a quantity of boron that produces a concentration greater than 600 ppm of boron in the reactor core in less than 125 minutes.

The minimum limitation on the relief valve setting is intended to prevent the loss of liquid control solution via the lifting of a relief valve at too low a pressure. The upper limit on the relief valve settings provides system protection from overpressure.

B. Only one of the two standby liquid control pumping loops is needed for operating the system. One inoperable pumping circuit does not immediately threaten shutdown capability. Assurance that the remaining system will perform its intended function and that the long-term average availability of the system is not reduced is obtained from a one-out-of-two system.

C. Level indication and alarm indicate whether the solution volume has changed, which might indicate a possible solution concentration change. The test interval has been established in consideration of these factors. Temperature and liquid level alarms for the system are annunciated in the control room.

The solution is kept at least 10° F above the saturation temperature to guard against boron precipitation. The margin is included in Figure 3.4.2.

3.5/4.5 CORE AND CONTAINMENT COOLING SYSTEMS

Delete the present specification in its entirety (pages 130-150b) and insert the following:

LIMITING CONDITIONS FOR OPERATION

3.5 CORE AND CONTAINMENT COOLING SYSTEMS

Applicability

Applies to the operational status of the core and containment cooling systems.

Objective

To assure the operability of the core and containment cooling systems under all conditions for which this cooling capability is an essential response to plant abnormalities.

Specification

A. Core Spray System (CSS)

1. When the reactor vessel is atmospheric and irradiated fuel is in the reactor vessel, at least one core spray loop with one pump and available diesel generator shall be capable of delivering flow.

B. Residual Heat Removal System (RHRS) Containment and Shutdown Cooling

1. When the reactor vessel

SURVEILLANCE REQUIREMENTS

4.5 CORE AND CONTAINMENT COOLING SYSTEMS

Applicability

Applies to the surveillance requirements of the core and suppression pool cooling systems when the corresponding limiting conditions for operation is in effect.

Objective

To verify the operability of the core and suppression pool cooling systems under all conditions for which this cooling capability is an essential response to plant abnormalities.

Specification

A. Core Spray System (CSS)

1. Core Spray System Testing

	Item	Frequency
a.	Pump operability	upon restoration and monthly thereafter
b.	Motor operated valve operability	upon restoration and monthly there after

2. When it is determined that one core spray loop with one pump and available diesel generator is incapable of delivering

LIMITING CONDITIONS FOR OPERATION

3.5 CORE AND CONTAINMENT COOLING SYSTEMS

is atmospheric and irradiated fuel is in the reactor vessel, at least one RHR loop with two pumps or two loops with one pump per loop shall be capable of manual operation. Diesel generators must also be available to power the pumps. A service water supply must be available.

C. RHR Service Water System (RHRSWS) Emergency Equipment Cooling Water System (EECWS)

1. When the reactor vessel is atmospheric and irradiated fuel is in the reactor vessel, 4 (or 3) RHRSW pumps must be functional and aligned to RHR header service corresponding to the 4 selected RHR pumps (2 of which may have corresponding heat exchangers on one RHRSW header thus requiring only 3 RHRSW pumps).

2. At all times, 2 RHRSW pumps which are capable of automatic restarting shall be

SURVEILLANCE REQUIREMENTS

4.5 CORE AND CONTAINMENT COOLING SYSTEMS

flow, another core spray pump with an available diesel generator shall be selected, and all active components in the flow paths shall be immediately demonstrated to be capable of delivering flow.

B. Residual Heat Removal System (RHRS) (Containment and Shutdown Cooling)

1. Residual Heat Removal System Testing

	Item	Frequency
a.	Pump operability	Upon restoration and monthly thereafter
b.	Motor operated valve operability	Upon restoration and monthly thereafter

2. When it is determined that one RHR pump (containment and suppression pool cooling) or associated heat exchanger is incapable of delivering flow and removing heat at a time when flow capability and heat removal is required, the remaining RHR pump and associated heat exchanger and available diesel generator, and all active components in the flow paths shall be demonstrated

LIMITING CONDITIONS FOR OPERATION

3.5 CORE AND CONTAINMENT COOLING SYSTEMS

assigned to EECW header service with one pump assigned to each header and running continuously. Each pump shall be assigned to a separate diesel power supply.

D. Maintenance of Filled Discharge Pipe

Whenever the core spray system or RHR systems are required to be functional, the discharge piping from the pump discharge of these systems to the last block valve shall be filled. The condensate head tank shall be aligned to serve the discharge piping of the RHR and CS pumps. The pressure indicators on the discharge piping of the RHR and CS pumps shall indicate not less than listed below.

PI-75-20	70 psig
PI-75-48	70 psig
PI-74-51	70 psig
PI-74-65	70 psig

SURVEILLANCE REQUIREMENTS

4.5 CORE AND CONTAINMENT COOLING SYSTEMS

to be capable of delivering flow and heat removal immediately and weekly thereafter until the inoperable RHR pump and associated heat exchanger is returned to service or an alternate pump and heat exchanger with an available diesel generator selected and verified.

C. RHR Service Water System (RHRSWS) And Emergency Equipment Cooling Water System (EECWS)

1. RHR Service Water System

   Each of the 4 (or 3) RHRSW pumps and associated essential control valves on the RHR heat exchanger headers shall be demonstrated to be functional upon restoration and once every three months thereafter if not in continuous service.

2. EECW System

   Each of the 2 RHRSW pumps assigned to EECW service will be in continuous service. Associated essential control valves on the EECW headers shall be demonstrated to function once every three months.

LIMITING CONDITIONS FOR OPERATION

3.5 CORE AND CONTAINMENT COOLING SYSTEMS

SURVEILLANCE REQUIREMENTS

4.5 CORE AND CONTAINMENT COOLING SYSTEMS

3. When it is determined that one RHRSW pump and associated control valves on an RHR heat exchanger header is incapable of delivering flow at a time when flow delivery capability is required, the remaining RHRSW pumps and associated heat exchangers and available diesel generators, and all active components in the flow paths shall be demonstrated to be capable of delivering flow immediately and weekly thereafter until the inoperable RHRSW pump is returned to service or an alternate pump is selected and verified.

4. When it is determined that one RHRSW pump and associated control valves on an EECW header is incapable of delivering flow at a time when flow delivery capability is required, an alternate RHRSW pump on a corresponding diesel generator shall be selected and assigned to the same EECW header.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS

3.5 CORE AND CONTAINMENT COOLING SYSTEMS

4.5 CORE AND CONTAINMENT COOLING SYSTEMS

D. Maintenance of Filled Discharge Pipe

The following surveillance requirements shall be adhered to to assure that the discharge piping of the core spray system and RHR system are filled:

1. Prior to the operation of the RHRS and core spray systems, the discharge piping of these systems shall be vented from the high point and water flow determined.

2. Following any period where the RHR or core spray systems have not been required to be operable, the discharge piping of the nonfunctional system shall be vented from the high point prior to the return of the system to service.

3. When the RHRS and CSS are required to be functional the pressure indicators which monitor the discharge lines shall be monitored daily and the pressure logged.

3.5 BASES: CORE AND CONTAINMENT COOLING SYSTEMS

3.5.A Core Spray System (CSS)

The core spray loop provides one of several means of supplying water to the reactor vessel. The capacity of one core spray pump is in excess of 10 times the amount of water required to cool the reactor core in its present shutdown condition. Power operation of the reactor will not be permitted, therefore, the margin available for this purpose will increase with time. Technical Specification 3.5.A and 3.5.B require that at least three pumps will be available to assure that adequate cooling of the fuel is accomplished. However, considerably more equipment is available, including; additional core spray, RHR, control rod drive, condensate booster, hotwell and raw water pumps. As noted in the introduction availability of these components will be maximized.

Analyses have been performed to demonstrate that in excess of 15 hours is available to restore core cooling if for some unforeseen reason cooling is interrupted. Studies were performed to show that a core spray loop could be manually placed in operation in less than one hour. Hence considerable margin is available for manual actuation.

3.5 BASES: CORE CONTAINMENT COOLING SYSTEMS

3.5.B Residual Heat Removal System (RHRS) (Containment and Shutdown Cooling

The decay heat removal requirements for one unit in the cold shutdown condition can be conservatively met by the operation of one RHR pump and its associated RHR heat exchanger in the shutdown cooling mode. The total heat load for the heat exchanger is estimated to be less than one-fourth of the heat exchanger capability under the required flow and temperature conditions. At least three RHR pumps and associated heat exchangers on each unit are capable of delivering flow and removing the decay heat. The low decay heat and absence of pressure which could foster an unacceptable loss of coolant allows ample time for manual operation in accordance with established operating instructions.

3.5 BASES: CORE AND CONTAINMENT COOLING SYSTEMS

3.5.C RHR Service Water System (RHRSW) Emergency Equipment Cooling Water System (EECWS)

The decay heat removal cooling water requirements for two units in the cold shutdown condition can be conservatively met by the operation of one RHRSW pump on one heat exchanger on each unit. One RHRSW pump is required foreach unit if the units are using heat exchangers which are not on the same service water header. Four RHRSW pumps are presently available and capable of delivering flow to meet this requirement. Less than one-half the flow delivery capability of each pump is needed to remove the present decay heat for each unit. The low decay heat level and ample flow delivery capability allows ample time for manual operation in accordance with established operating instructions.

The standby emergency equipment cooling water (EECW) requirements for two units in the cold shutdown condition can be adequately met by the operation of one RHRSW pump on an EECW header. The EECW system is not required for normal plant shutdown operation because the required cooling water is supplied by the plant raw cooling water system. The principal immediate need for EECW flow is in the event that a diesel engine should be started. In this case, EECW flow must be established at once. To meet this requirement, two RHRSW pumps assigned to EECW service are running continuously with each assigned to a seperate supply header. Either header will supply all EECW requirements. In addition, two other RHRSW pumps are available and capable of delivering flow by manual operations in accordance with established operating instructions. The two RHRSW pumps which are running continuously are assigned to 4.16 kV shutdown boards which have associated operable diesel generators and the pumps are arranged with automatic restart provisions in the event that board undervoltage should occur.

3.5 BASES: CORE AND CONTAINMENT COOLING SYSTEMS

3.5.D Maintenance of Filled Discharge Pipe

If the discharge piping of the core spray and RHR system are not filled, a water hammer can develop in this piping when the pump and/or pumps are started. To minimize damage to the discharge piping and to ensure added margin in the operation of these systems, this Tehcnical Specification requires the discharge lines to be filled whenever the system is in a functional condition. If a discharge pipe is not filled, the pumps that supply that line must be assumed to be nonfunctional for Technical Specification purposes.

The core spray and RHR system discharge piping high point vent is visually checked for water flow prior to any pump operation to ensure that the lines are filled. The visual checking will avoid starting the core spray or RHR system with a discharge line not filled. In addition to the visual observation and to ensure a filled discharge line other than prior to testing, a head tank located approximately 100 feet above the discharge line high point supplies makeup water for these systems. System discharge pressure indicators are used to determine the water level above the discharge line high point. The indicators will reflect approximately 30 psig for a water level at the high point and approximately 70 psig for a water level at the head tank and are monitored daily to ensure that the discharge lines are filled.

3.6/4.6 PRIMARY SYSTEM BOUNDARY

Delete the present specification in its entirety (pages 151-175) and substitute the following:

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENT

3.6 PRIMARY SYSTEM BOUNDARY

Applicability

Applies to the status of the reactor coolant system.

Objective

To assure the integrity of the reactor coolant system.

Specification

A. Thermal and Pressurization Limitations

The reactor vessel head bolting studs shall not be under tension unless the temperature of the vessel head flange and the head is greater than 100°F.

B. Coolant Chemistry

Reactor water shall be maintained within the following limits:

Conductivity,
10 umhos/cm@25°C
Chloride, 0.5 ppm

C. Safety and Relief Valves

Four safety relief valves which have accumulators shall be functional on each unit with two valves powered from each of two 250v DC MOV boards.

D. Reactor Vessel Head Vent

The reactor vessel head vents shall be in the open position.

4.6 PRIMARY SYSTEM BOUNDARY

Applicability

Applies to the periodic examination and testing requirements for the reactor coolant system.

Objective

To determine the condition of the reactor coolant system and the operation of the safety devices related to it.

Specification

A. Thermal and Pressurization Limitations

When the reactor vessel head bolting studs are tensioned and the reactor is in a cold condition, the reactor vessel shell temperature immediately below the head flange shall be permanently recorded.

B. Coolant Chemistry

1. A sample of reactor coolant shall be analyzed:
   a. at least every 96 hours for conductivity and chloride ion content.
   b. at least every 8 hours for conductivity and chloride ion content when the continuous conductivity monitor is inoperable.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENT
3.6 PRIMARY SYSTEM BOUNDARY	4.6 PRIMARY SYSTEM BOUNDARY
	C. Safety and Relief Valves
	The position indicating lights for the four safety relief valves specifie' in 3.6.C.1 shall be checked each operating shift to confirm control power available for actuation.
	D. Reactor Vessel Head Vent
	The position indicating lights on the reactor vessel head vents shall be checked each operating shift to confirm the valves are indicated to be in the open position.

3.6.A/4.6.A BASES: THERMAL AND PRESSURIZATION LIMITATIONS

Tightening the studs on the reactor vessel head flexes it slightly to bring together the entire contact surfaces adjacent to the "O" rings of the head and vessel flange. The reactor vessel head flange and head are constructed such that their initial maximum NDTT is 10° F, as cited in paragraph 4.2.7 of the safety analysis report. Therefore, the initial minimum temperature at which the studs can be placed in tension is established at 40° F + 60° F or 100°F.

3.6.B/4.6.B BASES: COOLANT CHEMISTRY

Materials in the primary system are primarily 304 stainless steel and the Zircaloy cladding. The reactor water chemistry limits are established to prevent damage to these materials. Limits are placed on conductivity and chloride concentrations. Conductivity is limited becasue it is continuously measured and gives an indication of abnormal conditions and the presence of unusual materials in the coolant. The chloride limit is specified to prevent stress corrosion cracking of stainless steel and is based on established relationships between stress corrosion, chloride concentrations and dissolved oxygen. Zircaloy does not exhibit similar stress corrosion failures.

When conductivity is in its normal range, pH and chloride and other impurities affecting conductivity must also be within their normal range. When conductivity becomes abnormal, then chloride measurements are made to determine whether or not they are also out of their normal operating values becaues conductivity could be high due to the presence of a neutral salt which would not have an effect on pH or chloride. In some types of water-cooled reactors, conductivities are in fact high due to purposeful addition of additives. In the case of BWR's however, where no additives [illegible] BASES: COOLANT CHEMISTRY (Continued)

are used and where near neutral pH is maintained, conductivity provides a very good measure of the quality of the reactor water. Significant changes therein provide the operator with a warning mechanism so he can investigate and remedy the condition causing the change before limiting conditions, with respect to variables affecting the boundaries of the reactor coolant, are exceeded.

Methods available for correcting the off-standard conditions, include using the cleanup demineralizer, reducing the input rate of impurities, or changing the supply of cooling water. The conductivity of the reactor coolant is continuously monitored. The samples of the coolant which are taken every 96 hours will serve as a reference for calibration of these monitors and is considered adequate to assure accurate readings of the monitors. If conductivity is within its normal range, chlorides and other impurities will also be within their normal ranges. The reactor coolant samples will also be used to determine the chlorides. Therefore, the sampling frequency is considered adequate to detect long-term changes in the chloride ion content. Reactor coolant sampling is increased to once per shift when the continuous conductivity monitor is unavailable.

[illegible].6.C BASES: SAFETY AND RELIEF VALVES

With the reactor in the cold shutdown condition and with low decay heat, four safety relief valves are more than adequate to assure the integrity of the reactor coolant systems. Control power from different 250V MCV boards assures availability of two valves even in the unlikely event of loss of a 250V MCV board. Surveillance checks each shift assures control power is available for valve actuation.

[illegible].6.D BASES: REACTOR VESSEL HEAD VENT

Maintaining the reactor vessel head vents in the open position with the RHR shutdown cooling system in service keeps the reactor vessel at atmospheric pressure when in the cold shutdown condition. Surveillance checks of the position indicating lights are made each shift.

3.7/4.7 CONTAINMENT SYSTEMS

Delete the present specification in its entirety (pages 175 through 216)

LIMITING CONDITIONS FOR OPERATION

3.7 CONTAINMENT SYSTEMS

Applicability

Applies to the status of the primary containment system.

Objective

To monitor pressure and temperature of the primary containment system.

Specification

A. The suppression chamber water level and temperature shall be monitored (see Table 3.2.A).

B. The drywell pressure and temperature shall be monitored (see Table 3.2.A).

SURVEILLANCE REQUIREMENTS

4.7 CONTAINMENT SYSTEMS

Applicability

Applies to the primary containment.

Objective

To verify the pressure and temperature of the primary containment.

Specification

A. The suppression chamber water level and temperature shall be checked once per day.

B. The drywell pressure and temperature shall be checked once per day.

3.7 BASES: CONTAINMENT SYSTEMS

This specification ensures indication of adequate information regarding status of the drywell pressure and temperature and suppression chamber water level and temperature. Monitoring of information concerning key primary containment parameters will ensure that sufficient control of these parameters can be manually initiated in a timely manner.

3.8/4.8 RADIOACTIVE MATERIALS

Pages 217-220 are unchanged. Page 221 - replace with the attached page which revises specification 3.8.B.8. Page 222 is unchanged.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
8. Radioactive gases released from each units turbine and reactor building roof vents, the radwaste building roof vents, and the main stack shall be continuously monitored. To accomplish this, at least one reactor building and one turbine building vent monitoring system per unit shall be operating whenever that unit's building ventilation system is in service. Also one radwaste building system vent monitoring channel shall be operating whenever the radwaste building ventilation system is in service. At least one main stack monitoring channel shall be operating whenever any units air ejector, mechanical vacuum pump, or a standby gas treatment system train is in service. If normal monitoring systems are not available, temporary monitors or other systems shall be used to monitor effluent. A monitoring channel may be out of service for 4-hours for functional testing and calibration without providing a temporary monitor.	4.8.B Airborne Effluents
	C. Radiological Environmental Monitoring Program An environmental monitoring program shall be conducted as described below and outlined in Table 4.8.F. 1. Atmospheric Monitoring a. The atmospheric monitoring network is divided into three subgroups consisting of 12 monitoring stations of which 9 stations shall be operable at all times. The monitoring stations are shown on Figure 4.8-1. These monitoring locations are subject to change dependent upon continued evaluation of the environmental monitoring program. The station at Muscle Shoals will be used as background reference. Each monitor shall be capable of continuously sampling air at regulated flow of approximately three cubic feet per
9. The primary containment shall be purged through the standby gas treatment system.	

3.8/4.8 RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

There is no change to thi specification (pages 223 - 239).

3.9/4.9 Delete the present specification in its entirety (pages 240-248) and substitute the following:

LIMITING CONDITIONS FOR OPERATION

3.9 AUXILIARY ELECTRICAL SYSTEM

Applicability

Applies to the auxiliary electrical power system.

Objective

To assure an adequate supply of electrical power for those systems required for safety.

Specification

A. Auxiliary Electrical Equipment

For the condition of both units 1 and 2 in cold shutdown, the following conditions apply except as specified in section 3.9.B.

1. Two off-site sources of power to the shutdown system available from either the 500 KV system (backfeeding) or the 161 KV transmission system.
2. 3 diesel generators available and their associated 4 kV shutdown boards. All 4 kV shutdown boards energized unless offsite power is lost.
3. 4 kV shutdown bus 1 or 2 energized.
4. 480V shutdown boards associated with each unit are energized.
5. Under voltage relays on the three 4 kV shutdown boards associated with the functional diesel generators shall be available.

SURVEILLANCE REQUIREMENTS

4.9 AUXILIARY ELECTRICAL SYSTEM

Applicability

Applies to the periodic testing requirements of the auxiliary electrical systems.

Objective

Verify the capability of the auxiliary electrical system.

Specification

A. Auxiliary Electrical Equipment

1. Diesel Generators

a. Each diesel generator shall be manually started and loaded once each month to demonstrate operational readiness. The test shall continue for at least a one-hour period at 75% of rated load or greater.

During the monthly generator test the diesel generator starting air compressor shall be checked for operation and its ability to recharge air receivers. The operation of the diesel fuel oil transfer pumps shall be demonstrated, and the diesel starting time to reach rated voltage and speed shall be logged.

3.9 LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.9 AUXILIARY ELECTRICAL SYSTEM	4.9 AUXILIARY ELECTRICAL SYSTEM

Applicability

Applies to the auxiliary electrical power system.

Objective

To assure an adequate supply of electrical power for those systems required for safety.

Specification

A. Auxiliary Electrical Equipment

For the condition of both units 1 and 2 in cold shutdown, the following conditions apply except as specified in section 3.9.B.

1. Two off-site sources of power to the shutdown system available from either the 500 kV system (backfeeding) or the 161 kV transmission system.
2. 3 diesel generators available and their associated 4 kV shutdown boards. [illegible] kV shutdown boards energized [illegible] less offsite power is lost.
3. 4 kV shutdown bus 1 or 2 energized.
4. 480V shutdown bo[illegible] associated with each unit [illegible] energized.
5. [illegible] on the three 4 kV [illegible] associated with [illegible] diesel generators [illegible] [illegible]

Applicability

Applies to the periodic testing requirements of the auxiliary electrical systems.

Objective

Verify the capability of the auxiliary electrical system.

Specification

A. Auxiliary Electrical Equipment

1. Diesel Generators
   a. Each diesel generator shall be manually started and loaded once each month to demonstrate operational readiness. The test shall continue for at least a one-hour period at 75% of rated load [illegible]

   During the monthly generator test the diesel generator starting air compressor shall be checked for operation and its ability to recharge air receivers. The operation of the diesel fuel oil transfer pumps shall [illegible] and the diesel starting time to reach rated voltage and speed shall be logged.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
	4.9 AUXILIARY ELECTRICAL SYSTEM
	b. Once a month the quantity of diesel fuel available shall be logged.
	c. Each diesel generator shall be given an annual inspection in accordance with instructions based on the manufacturer's recommendations.

LIMITING CONDITIONS FOR OPERATION

3.9 AUXILIARY ELECTRICAL SYSTEM

6. 250V DC shutdown board batteries and chargers for the three functional boards must be available.

7. The 250V unit board batteries and a battery charger for each battery and associated battery boards must be functional.

8. Batteries, chargers, busses, and feeds to the four SRM channels specified in Table 3.2.A shall be operable in each unit.

9. There shall be a minimum of 103,300 gallons of diesel fuel in the standby diesel generator fuel tanks.

B. Cold Shutdown on Units 1 And 2 With Inoperable Equipment

1. From and after the date that off-site power is reduced to one source, another source must be restored within seven days.

2. When one of the three diesel generators become inoperable and a shutdown bus and 3-4 kV shutdown boards are available, 2 days are allowed

SURVEILLANCE REQUIREMENTS

4.9 AUXILIARY ELECTRICAL SYSTEM

d. Once a month a sample of diesel fuel shall be checked for quality. The quality shall be within the acceptable limits specified in Table 1 of ASTM D975-68 and logged.

2. D.C. Power System - Unit Batteries (250-Volt) Diesel Generator Batteries (125-Volt) And Shutdown Board Batteries (250-Volt).

a. Every week the specific gravity and the voltage of the pilot cell, and temperature of an adjacent cell and overall battery voltage shall be measured and logged.

b. Every three months the measurements shall be made of voltage of each cell to nearest 0.1 volt, specific gravity of each cell, and temperature of every fifth cell. These measurements shall be logged.

3. Undervoltage Relays

a. Once every 6 months, the

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS

3.9 AUXILIARY ELECTRICAL SYSTEM

to comply with 3.9.A.2

3. When one 4-kV shutdown board is inoperable, both off-site 161-kV transmission lines and both common station transformers shall be available and the remaining 4-kV shutdown boards and associated diesel generators, and all 480V emergency shutdown boards shall be functional.

4. From and after the date that one of the three 250-Volt unit batteries and/or its associated battery board is found to be inoperable for any reason, the NRC shall be notified within 24 hours of the situation, the precautions to be taken during this period and the plans to return the failed component to an operable state.

5. From and after the date that one of the three 250-Volt shutdown board batteries and/or its associated battery board is found to be inoperable for any reason, the NRC shall be notified within 24 hours of the situation, the precautions to be taken during this period and the

4.9 AUXILIARY ELECTRICAL SYSTEM

conditions under which the undervoltage relays are required shall be simulated with an undervoltage on each shutdown board to demonstrate that the associated diesel generator will start.

b. The undervoltage relays which start the diesel generators from the 4-kV shutdown boards, shall be calibrated annually for trip and reset and the measurements logged.

B. Operation with Inoperable Equipment

1. When one common station transformer, or one 161 kV line is found to be inoperable, all diesel generators and associated boards must be demonstrated to be functional immediately and daily thereafter.

2. When one diesel generator is found to be inoperable, the remaining diesel generators and associated boards shall be demonstrated to be functional immediately and daily thereafter.

LIMITING CONDITIONS FOR OPERATION

3.9 AUXILIARY ELECTRICAL SYSTEM

plans to return the failed component to an operable state.

SURVEILLANCE REQUIREMENTS

4.9 AUXILIARY ELECTRICAL SYSTEM

3. When one 4-kV shutdown board is found to be inoperable, all remaining 4-kV shutdown boards and two associated diesel generators, shall be demonstrated to be functional, immediately and daily thereafter.

3.9 BASES: AUXILIARY POWER SYSTEM

The auxiliary power system consists of 3 functionally operable diesel generators (A, B, & D) capable of automatic start on loss of voltage to their respective 4-kV shutdown boards. The C diesel feed is inoperable but its associated 4-kV shutdown board is available. The 480V shutdown boards 1A, 1B, 2A, and 2B are operable with at least one source of power. All the 480V reactor MOV boards 1A, 1B, 1C, 2A, 2B, & 2C are operable with at least one source of power.

The requirements of this specification are based on the minimum complement of equipment to redundantly maintain the reactor in a safe condition. The capacity requirements of this mode of operation can be adequately addressed with two diesels. The third diesel is to accomodate loss of off-site power and single failures in the diesel generator plant.

All 4-kV and 480V boards are available and the shutdown busses are available. With a single failure in any of the 4-kV and 480V board or feeders, diesel power can be directed to an adequate complement of equipment to accomodate the shutdown mode. This is accomplished by either feeding the required boards and their associated loads directly or by feeding two boards from one diesel through the backfeed capability of the shutdown bus.

There are eight 250-volt d-c battery systems each of which consists of a battery, battery charger, and distribution equipment. Three of these systems provide power for unit control functions, operative power for unit motor loads, and alternative drive power for a 115-volt a-c unit preferred motor-generator set. One 250-volt d-c system provides power for common plant and transmission system control functions, drive power for a 115-volt a-c plant preferred motor-generator set, and emergency drive power for certain unit large motor loads. The four remaining systems deliver control power to the 4160-volt shutdown boards.

3.9 BASES: AUXILIARY POWER SYSTEM (Continued)

Each 250-volt d-c shutdown board control power supply can receive power from its own battery, battery charger, or from a spare charger. The chargers are powered from normal plant auxiliary power or from the standby diesel-driven generator system. Zero resistance short circuits between the control power supply and the shutdown board are cleared by fuses located in the respective control power supply. Each power supply is located in the reactor building near the shutdown board it supplies. Each battery is located in its own independently ventilated battery room.

The 250-volt d-c system is so arranged, and the batteries sized such, that the loss of any one unit battery will not prevent the safe cooldown of both units in the event of the loss of offsite power and single failure in any one unit. The loss of one 250-volt shutdown board battery affects normal control power only for the 4160-volt shutdown board which it supplies. The station battery supplies loads that are not essential for safe shutdown and cooldown of the nuclear system. This battery was not considered in the accident load calculations.

4.9 BASES

The monthly tests of the diesel generators are primarily to check for failures and deterioration in the system since last use. The diesels will be loaded to at least 75 percent of rated power while engine and generator temperatures are stabilized (about one hour). The minimum 75 percent load will prevent soot formation in the cylinders and injection nozzles. Operation up to an equilibrium temperature ensures that there is no overheat problem. The tests also provide an engine and generator operating history to be compared with subsequent engine-generator test data to identify and to correct any mechanical or electrical deficiency before it can result in a system failure.

Battery maintenance with regard to the floating charge, equilising charge, and electrolyte level will be based on the manufacturer's instruction and sound maintenance practices. In addition, written records will be maintained of the battery performance. The plant batteries will deteriorate with time but precipitous failure is unlikely. The type of surveillance called for in this specification is that which has been demonstrated through experience to provide an indication of a cell becoming irregular or unserviceable long before it becomes a failure.

The equalizing charge, as recommended by the manufacturer, is vital to maintaining the Ampere-hour capacity of the battery, and will be applied as recommended.

3.10/4.10 CORE ALTERATIONS

The present technical specification (pages 249 - 260) remains unchanged.

3.11 UNIT 3/UNIT 1 AND 2 INTERACTIONS

Add a new specification as indicated on the attached sheet, p. 260a.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENT
3.11 UNIT 3/UNIT 1 AND 2 INTERACTIONS Applicability Applies to construction activities of unit 3. Objective To assure that unit 3 construction activities do not adversely affect the safety of either unit 1 or 2. Specification All operations, tests, and other activities associated with the construction of unit 3 that can affect the safety of either unit 1 or unit 2 are prohibited.	

3.11 BASES: UNIT 3/UNIT 1 AND 2 INTERACTIONS

Interface documentation that presents the results of analyses to demonstrate that adverse affects associated with interaction between unit 3 and either unit 1 or unit 2 are not available. Since the arrangement of safety systems is such that adverse affects are possible, further degradation in either unit 1 or unit 2 should not be allowed until the interaction analysis is completed.

5.0 MAJOR DESIGN FEATURES

The present technical specification remains unchanged.

6.0 ADMINISTRATIVE CONTROLS

The present specification remains unchanged (pages 263 - 288) except delete page 285 and substitute the following pages which contain a new table 6.8.A and its bases.

TABLE 6.8.A

Minimum Shift Crew Requirements

Shift Position		Type of License
Shift Engineers (SE)	1	SRO
Assistant Shift Engineers (ASE)	1	SRO
Licensed Reactor Operator[1]	1	RO
Unit Operators (UO)	2	RO
Assistant Unit Operators (AUO)	6	None
Health Physics Technician	1	None
Other Personnel	2	
Minimum Shift Crew	14	

Notes: SRO - Senior Reactor Operator
RO - Reactor Operator

Note for Table 6.8.A

[1]This position is normally filled by an assistant shift engineer, but as a minimum it may be filled by a licensed reactor operator. When the incumbent is not a senior reactor operator, he shall not be assigned duties requiring him to direct licensed activities of reactor operators.

BASES

As a result of the damage created by the fire to various units 1 and 2 cables, a number of safety systems cannot be operated from the control room. Equipment monitoring and operation is now required from various positions located throughout the facility. Analyses have shown that in excess of 15 hours are available to perform these various tasks. In addition, studies have been performed to assess the amount of time

BASES (Continued)

required to perform these manual operations including consideration of additional equipment failures. Results of these studies have shown that operation of equipment for which the most degradation is experienced could be achieved in a time interval of several hours. These studies were performed with the assumption of a shift crew of 10 men. The additional 4 men required provide margin as well as to allow for continued monitoring during performance of these manual actions. It is expected that two of these four men would be used for that purpose. When the equipment is aligned as described in Appendix A to "Safety Analysis of the Browns Ferry Nuclear Plants Units 1 and 2 During Operations Related to Removal of Damaged Cables, Cable Trays, and Conduits," dated May 1, 1975, the minimum crew size could be reduced.

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

SAFETY EVALUATION BY THE DIVISION OF REACTOR LICENSING

SUPPORTING AMENDMENT NO. 9 TO FACILITY LICENSE NO. DPR-33

AND AMENDMENT NO. 6 TO FACILITY LICENSE NO. DPR-52

(CHANGE NO. 10 TO TECHNICAL SPECIFICATIONS)

TENNESSEE VALLEY AUTHORITY

BROWNS FERRY NUCLEAR POWER PLANT, UNITS 1 AND 2

DOCKET NOS. 50-259 AND 50-260

INTRODUCTION

On March 22, 1975, a fire at the Browns Ferry facility resulted in substantial damage to power, control, and instrumentation wiring. The facility is not operational and is in a cold shutdown condition with fuel in the core and the vessel vented to the containment atmosphere. Although many systems were damaged, the facility has been maintained in a safe and stable condition utilizing a substantial number of systems and components. In fact, the functioning systems exceed the minimum number usually required for a facility in cold shutdown condition.

In order to prepare the facility for repair of the fire damage, vital equipment to protect the core from damage is being reconnected, or rerouted or verified to be free from the fire affected wiring or systems. Many of these vital systems are being modified to manual control rather than automatic control to isolate them from spurious operation resulting from interconnection with fire damaged wiring or systems. Some systems or components are being disconnected or locked in a safe configuration. The primary system mode switch is, by administrative control, locked in the "shutdown" mode.

Before any repair and removal of cables and systems from the fire affected zone can be authorized, all of the necessary vital systems must be protected, isolated, or verified as free from interconnection with any of the systems or cabling which is to be removed. These activities of rerouting, reconnecting, verification, and isolating systems from potential interaction with systems or wiring to be removed are presently in progress.

The available systems and components to provide plant safety under present conditions, while the facility is in cold shutdown and the plant is being configured into a condition protected against possible adverse interaction resulting from repairs and restoration, have been reviewed and appropriate technical specifications governing this period have been established through conferences between the staff and the licensee and submitted for approval by the licensee. This safety evaluation describes the protection required to provide adequate safety during present conditions. It does not address or authorize cable removal, cutting, or repair of the fire affected zone. Any such authorization will be addressed in a subsequent safety evaluation.

EVALUATION

1. Nuclear Safety

The control rods are each fully inserted in the core. The reactor mode switch is locked in the shutdown position, the control rod directional control valves, which provide the drive fluid to withdraw the rods, have been electrically disconnected, and the manual valves in the drive water supply lines have been closed. In this condition the core is subcritical and cannot be made critical. Under the present low pressure, low temperature conditions, there are no equipment malfunctions or single operator errors which could cause any control rod to withdraw. Nonetheless, rod position is to be verified in the fully-in position on a regular basis. The control rod position indicator full-in switches will be operable or the control rod position will be verified to be full-in by a continuity check of the full-in limit switches at the local panel.

In addition, the electrically disconnected condition of the control rod directional control valves will be verified by visual observation, along with visual observation of the closed position of the manual valve in the drive water supply lines.

In addition to the manual scram, an automatic alarm and scram function has been added for the operable source range monitors, even through this is not ordinarily a function of this system. This system will detect any approach to criticality and will both alarm and automatically cause scram action to drive the control rods in if for any unforeseen reason they might have moved outward. The control rod accumulators will be maintained in a fully charged condition. A standby supply of borated water is available for injection into the reactor to further ensure a subcritical core. These systems provide a high degree of assurance that the facility is protected against the occurrence of criticality under present conditions.

2. Core Cooling and Water Inventory

Under present conditions, the fuel remains in the core under cold shutdown conditions (less than 212 F and atmospheric pressure). Core cooling is presently being provided by the operation of one residual heat removal (RHR) pump and associated heat exchanger. Decay heat which must be removed to prevent fuel damage is very small, amounting to less than one-fourth of the heat exchanger capacity of one RHR system. Nonetheless, two RHR pumps and heat exchangers are required to be operational by the technical specifications and at least three RHR pumps and heat exchangers are presently available on each unit. The low decay heat level provides ample time for manual operation of backup equipment. The RHR pumps take suction from a recirculation loop and pump water through the RHR heat exchangers in which reactor decay heat is transferred to a separate RHR service water system. In the event that the RHR pump providing flow should fail, the second RHR system for the unit is functioning and can provide ample cooling. In the event of the failure of the single valve in the suction line, heat removal can be accomplished by the generation of steam in the core which is vented to the containment through the open vent valves or backup relief valves. The condensed steam returns to the torus which can be cooled by either of the RHR systems operating in a torus cooling mode which does not depend on the failed suction valve. Makeup coolant to the core in this mode of operation could be provided by either of the RHR pumps or the core spray pump taking suction from the torus.

Analyses have been performed to demonstrate that in excess of 15 hours is available to restore core cooling if for some unforeseen reason cooling is interrupted. Studies were performed to show that backup systems could be manually placed in operation in less than 1 hour. Considerable margin is, therefore, available for manual actuation.

The decay heat removal cooling water requirements for two units in the cold shutdown condition can be conservatively met by the operation of one RHRSW pump or one heat exchanger on each unit. One RHRSW pump is required for each unit if the units are using heat exchangers which are not on the same service water header. Four RHRSW pumps are presently available and capable of delivering flow to meet this requirement. Less than one-half the flow delivery capability of each pump is needed to remove the present decay heat for each unit. The low decay heat level and ample flow delivery capability allows ample time for manual operation in accordance with established operating instructions.

The standby emergency equipment cooling water (EECW) requirements for two units in the cold shutdown condition can be adequately met by the operation of one RHRSW pump on an EECW header. The EECW system is not required for normal plant shutdown operation because the required cooling water is supplied by the plant raw cooling water system. The principal immediate need for EECW flow is in the event that a diesel engine should be started. In this case, EECW flow must be established at once. To meet this requirement, two RHRSW pumps assigned to EECW service are running continuously with each assigned to a separate supply header. Either header will supply all EECW requirements. In addition, two other RHRSW pumps are available and capable of delivering flow by manual operations in accordance with established operating instructions. The two RHRSW pumps which are running continuously are assigned to 4.16 kV shutdown boards which have associated operable diesel generators and the pumps are arranged with automatic restart provisions in the event that board undervoltage should occur.

In the present cold shutdown vented condition, with no pressure and low temperature in the primary system, there is no source to cause piping damage. Loss of coolant through inadvertent draining is minimized by locking valve positions so that water cannot be drained from the vessel. However, even in the event of loss of coolant inventory either of the RHR pumps or the core spray pump are capable of taking suction on the torus and providing water to the vessel. Even beyond these pumps there are additional RHR pumps, core spray pumps, and even hotwell and condensate booster pumps capable of supplying coolant to the vessel. Although the technical specifications do not require that these pumps be kept functioning, the technical specifications do provide that the licensee will not deliberately remove any of the pumps or any other component capable of achieving the objectives of the technical specifications from service except for maintenance required to prevent a substantial increase in the risk of failures to the component or other functioning system.

3. Electrical Power

The electrical energy to operate the systems required to maintain adequate cooling of the core is provided by redundant offsite and onsite power supplies. The offsite power is available through two 161 kV circuits connected from the power grid to redundant shutdown buses which in turn supply power to all four 4.16 kV emergency buses. To provide power in the event of loss of the offsite power supply, three of the four plant diesels are functioning along with their associated emergency buses. Only two of the three are required to supply a sufficient amount of electrical energy to power vital equipment. The emergency diesels are cooled through the emergency coolant water system which is presently continuously operating and has been configured to automatically transfer to diesel load in the event of loss of offsite power to assure that these pumps automatically provide sufficient coolant to the diesels to protect against overheating. Redundant 250 V batteries and their associated distribution systems are available to provide DC control power as necessary to the required system.

4. Minimum Shift Crew Considerations

As a result of the damage created by the fire to various Unit 1 and 2 cables, a number of safety systems cannot be operated from the control room. Equipment monitoring and operation is now required from various positions located through the facility. Analyses have shown that in excess of 15 hours are available to perform various tasks involved in transferring to the various functioning backup systems. Manual operation of equipment for which the most degradation is experienced could be achieved in a time interval of several hours, and the licensee has indicated that these operations could easily be accomplished by a shift crew of 10 men. We have required a shift crew of 14 men. The additional four men required provide margin, as well as to allow for continued monitoring during performance of these manual actions. It is expected that two of these four men would be used for that purpose.

5. Quality Assurance

In connection with the work of configuring the plant so various systems and components are free of potential interaction during fire damage removal and facility restoration, the QA procedures have recently been clarified to focus on this work to assure that they conform to the approved QA program. The previously approved QA program itself has not been modified.

6. Unit 3 Construction Activities

As a result of interfaces between Unit 3 and Unit 1 and 2, the technical specifications do not permit operations, tests, and other activities associated with the construction of Unit 3 that could affect the safety of either Unit 1 or 2.

7. Summary

In its present cold shutdown condition, the facility has been protected from damage to the core by disarming control rods so that no mechanism is available which could result in core criticality. The available functioning equipment is adequate to provide ample coolant flow to keep the core cooled and exceeds the minimum required for cold shutdown by the technical specifications governing facility operation prior to the fire.

The technical specification changes associated with this amendment reflect the changes that are necessary to account for the present condition of the plant. Certain additional controls and equipment requirements not required in the pre-fire technical specifications have been added to provide additional assurance that the plant will be maintained in a safe and stable shutdown condition during the present activities of preparing for the restoration program. The technical specifications associated with this amendment include these added controls and equipment requirements.

Except as necessitated by the physical realities that exist due to fire damage, no safety limit, limiting condition for operation, or surveillance requirement in the pre-fire technical specifications that is pertinent to the present cold shutdown condition of the plant has been modified, relaxed, or deleted by this amendment. The resulting technical specifications continue to provide adequate protection to the health and safety of the public and, by accounting for the actual existing condition of the plant systems and components, they minimize potentially unsafe conditions or actions. With the plant locked into a shutdown condition and considering the very low amount of decay heat generation from the shutdown core, the probability of any accident involving the release of radioactivity from the site is very low and the potential consequences of any accident are correspondingly low. In view of the foregoing, we conclude that the issuance of these amendments does not involve a significant hazards consideration.

CONCLUSION

The changes in technical specifications authorized in connection with this evaluation result in enhancement of safety under present conditions, as discussed above. Based on these considerations, we have concluded that: (1) because the change does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the change does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of these amendments will not be inimical to the common defense and security or to the health and safety of the public.

Date: May 9, 1975

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NOS. 50-259 AND 50-260

TENNESSEE VALLEY AUTHORITY

NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES

Notice is hereby given that the U.S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 9 to Facility Operating License No. DPR-33 and Amendment No. 6 to Facility Operating License No. DPR-52 issued to Tennessee Valley Authority which reivsed Technical Specifications for operation of the Browns Ferry Nuclear Plant, Units 1 and 2, located in Limestone County, Alabama. The amendments are effective as of their date of issuance.

The amendments revise the Technical Specifications, taking into account the present condition of plant systems, so as to assure that the two units will remain in a safe and stable posture during the period of plant shutdown resulting from damage due to a fire which occurred on March 22, 1975. The amendments do not authorize removal of fire damaged components or systems or restoration of the facility to operable conditions. This will be subsequently considered.

The application for these amendments complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter 1, which are set forth in the license amendments, Prior public notice of these amendments is not required since the amendments do not involve a significant hazards consideration.

For further details with respect to this action, see (1) the application for amendment dated May 8, 1975, (2) Amendment No. 9 to License No. DPR-33 and Amendment No. 6 to License No. DPR-52 with Change No. 10, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, NW., Washington, D.C. and at the Athens Public Library, South and Forrest, Athens, Alabama 35611.

A copy of items (2) and (3) may be obtained upon request addressed to the U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Reactor Licensing.

Dated at Bethesda, Maryland, this 9th day of May, 1975.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert A. Purple, Chief
Operating Reactors Branch #1
Division of Reactor Licensing

ATTACHMENT III

LETTER FROM THE NRC TO THE TVA DATED JUNE 13, 1975, WITH ENCLOSURES

AMENDMENT NO. 10 TO LICENSE NO. DPR-33

AMENDMENT NO. 7 TO LICENSE NO. DPR-52

SAFETY EVALUATION

FEDERAL REGISTER NOTICE

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

JUN 1 3 1975

Docket Nos. 50-259
and 50-260

Tennessee Valley Authority
ATTN: Mr. James E. Watson
Manager of Power
818 Power Building
Chattanooga, Tennessee 37201

Gentlemen:

The Commission has issued the enclosed Amendment No. 10 to Facility License No. DPR-33 and Amendment No. 7 to Facility License No. DPR-52 for the Browns Ferry Nuclear Plant, Units 1 and 2. These amendments include Change No. 11 to the respective Technical Specifications and are in response to your request dated June 2, 1975. Copies of the related Safety Evaluation and the Federal Register Notice are also enclosed.

The amendments revise the Technical Specifications, taking into account the present condition of plant systems, so as to ensure that the two units will remain in a safe and stable posture during the period of defueling and fuel storage in the fuel pools. The plants will be maintained in this condition until completion of the repairs of damage resulting from the fire which occurred on March 22, 1975. We have completed our review of your proposed plans for removal of fire-affected features and find them acceptable. Accordingly, pursuant to Mr. Rusche's letter to you dated April 17, 1975, these amendments authorize you to commence removal of fire-affected features upon fulfillment of the requirements of Technical Specification 3.10.H.

Our review of your proposed plans related to restoration of Units 1 and 2 is continuing and will be the subject of a subsequent approval action.

Sincerely,

Robert A. Purple

Robert A. Purple, Chief
Operating Reactors Branch #1
Division of Reactor Licensing

Enclosures:
See next page

Enclosures:
1. Amendment No. 10 to License No. DPR-33
2. Amendment No. 7 to License No. DPR-52
3. Safety Evaluation
4. Federal Register Notice

cc w/enclosures:
Robert H. Marquis
General Counsel
629 New Sprankle Building
Knoxville, Tennessee 37919

Athens Public Library
South and Forrest
Athens, Alabama 35611

Mr. Thomas Lee Hammons
Chairman, Limestone County Board
of Revenue
Athens, Alabama 35611

Anthony Z. Roisman, Esquire
Berlin, Roisman & Kessler
1712 N Street, NW
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Ira L. Myers, M.D.
State Health Officer
State Department of Public Health
State Office Building
Montgomery, Alabama 36104

Mr. Dave Hopkins
Environmental Protection Agency
1421 Peachtree Street, NE
Atlanta, Georgia 30309

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

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-259

BROWNS FERRY NUCLEAR PLANT UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 10
License No. DPR-33

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Tennessee Valley Authority (the licensee) dated June 2, 1975, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations; and

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

2. Accordingly, the license is amended by a change to the Technical Specifications as indicated in the attachment to this license amendment and Paragraph 2.C.(2) of Facility License No. DPR-33 is hereby amended to read as follows:

"(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications, as revised by issued changes thereto through Change No. 11."

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert A. Purple

Robert A. Purple, Chief
Operating Reactors Branch #1
Division of Reactor Licensing

Attachment:
Change No. 11 to Technical
Specifications

Date of Issuance: JUN 13 1975

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

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-260

BROWNS FERRY NUCLEAR PLANT UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 7
License No. DPR-52

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Tennessee Valley Authority (the licensee) dated June 2, 1975, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations; and

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

2. Accordingly, the license is amended by a change to the Technical Specifications as indicated in the attachment to this license amendment and Paragraph 2.C.(2) of Facility License No. DPR-52 is hereby amended to read as follows:

"(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications, as revised by issued changes thereto through Change No. 11."

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert A. Purple, Chief
Operating Reactors Branch #1
Division of Reactor Licensing

Attachment:
Change No. 11 to Technical
Specifications

Date of Issuance: JUN 1 3 1975

ATTACHMENT TO LICENSE AMENDMENT NO. 10 TO LICENSE NO. DPR-33

AND LICENSE AMENDMENT NO. 7 TO LICENSE NO. DPR-52

(CHANGE NO. 11 TO THE TECHNICAL SPECIFICATIONS)

DOCKET NOS. 50-259 AND 50-260

These page changes refer to pages of the "Technical Specifications and Bases for Browns Ferry Nuclear Plant, Units 1 and 2, effective prior to May 9, 1975

Revise Appendix A as follows:

Remove Pages	Insert New Pages
8 through 26	8, 15
27 through 30	27, 28
31 through 49	31, 33, 42
50 through 106	50, 54, 59, 81, 99, 100
107 through 121	108, 109, 115
122 through 129	122, 123, 124, 125, 126, 127, 128
130 through 150b	130, 131, 131a, 132, 133, 134, 135, 136, 143, 144, 145, 146, 147, 148, 149
151 through 174	151, 152, 166, 167, 168
175 through 216	175, 176, 176a, 206
221	221
240 through 260	240, 241, 242, 243, 244, 245, 246, 247, 248, 249, 250, 251, 252, 253, 254, 254a, 255, 256, 256a, 257, 258, 259, 260, 261, 262, 263, 264
--	270A, 270B
285	285, 285a

FACILITY OPERATING LICENSES DPR-33 AND DPR-52

INTERIM TECHNICAL SPECIFICATIONS AND BASES

FOR

BROWNS FERRY NUCLEAR PLANT

UNIT 1 AND UNIT 2

SAFETY LIMIT

1.1 FUEL CLADDING INTEGRITY

Applicability

Applies to the interrelated variables associated with fuel thermal behavior.

Objective

To establish limits which ensure the integrity of the fuel cladding.

Specification

A. Whenever the reactor is in the cold shutdown condition with irradiated fuel in the reactor vessel, the water level shall not be less than 17.7 inches above the top of the normal active fuel zone.

LIMITING SAFETY SYSTEM SETTING

2.1 FUEL CLADDING INTEGRITY

Applicability

Applies to the devices which are provided to prevent the reactor system safety limits from being exceeded.

Objective

To define the level of the process variables at which protective action is initiated to prevent the fuel cladding integrity safety limit from being exceeded.

Specification

The limiting safety system settings shall be as specified below:

A. Manual operation under procedural controls shall be established to assure the reactor water level remains ≥378 inches above vessel zero.

1.1/2.1 BASES: LIMITING SAFETY SYSTEM SETTINGS RELATED TO FUEL CLADDING INTEGRITY

The objective of this specification is to assure that irradiated fuel in the reactor vessel remains covered with water at all times. When plant equipment as specified in facility technical specification is available, this objective is accomplished automatically. This is no longer possible. Accordingly, procedural control of the water level is required as set forth in the specification 3.2.A.1.

With the reactor in cold shutdown, rapid makeup of coolant inventory is not required. The provisions for manual operation under procedural controls provided by the TVA Safety Analysis of the BFNP units 1 and 2 included as Part VI, Section E, of the "Plan for Evaluation, Repair, and Return to Service of Browns Ferry Units 1 and 2 (March 22, 1975, Fire)" will assure timely isolation of leaks by requiring redundant level alarms and indicators. Double isolation features, one of which shall be operative from the control room, will prevent draining of the vessel as a result of any single active component failure. Also, in the unlikely event that coolant inventory is required, a core spray pump, capable of delivering flow, will be available to provide the required vessel makeup capability.

The limiting safety systems scram setpoints are not capable of performing their intended function since the automatic actions they would initiate are unavailable. However, the facility has been placed in a condition that prevents the need for these actions.

SAFETY LIMIT

1.2 REACTOR COOLANT SYSTEM INTEGRITY

Applicability

Applies to limits on reactor coolant system pressure.

Objective

To establish a limit below which the integrity of the reactor coolant system is not threatened due to an overpressure condition.

Specification

A. The pressure at the lowest point of the reactor vessel shall not exceed 1,375 psig whenever irradiated fuel is in the reactor vessel.

LIMITING SAFETY SYSTEM SETTING

2.2 REACTOR COOLANT SYSTEM INTEGRITY

Applicability

Applies to the devices which are provided to prevent the reactor system safety limits from being exceeded.

Objective

To define the protective action to prevent the pressure safety limit from being exceeded.

Specification

A. The reactor vessel head vent valves shall be kept in the open position.

B. Four safety relief valves which have accumulators shall be functional on each unit with two valves powered from each of two 250V DC MOV boards, whenever the reactor vessel head is bolted in place.

1.2/2.2 BASES: REACTOR COOLANT SYSTEM INTEGRITY

With the reactor in the cold shutdown condition and with redundant RHR shutdown cooling capability, pressurization of the reactor vessel is prevented. In the unlikely event that shutdown cooling capability from the RHR system is lost, manual operation of any one of the four available relief valves prevents pressurization of the reactor pressure vessel above 50 psig preserving the safety limit whenever the reactor vessel head is bolted in place. This keeps the pressure low enough to permit addition of water by a core spray pump.

Checks of the reactor head vent and relief valve position indicating lights each shift confirm control power available for actuation.

LIMITING CONDITIONS FOR OPERATION

3.1 REACTOR PROTECTION SYSTEM

Applicability

Applies to the instrumentation and associated devices which initiate a reactor scram.

Objective

To assure the operability of the reactor protection system whenever irradiated fuel is in the reactor vessel.

Specification

A. Whenever irradiated fuel is in the reactor vessel the setpoints and minimum number of channels required to be functional are listed in Table 3.1.A/4.1.A.

SURVEILLANCE REQUIREMENTS

4.1 REACTOR PROTECTION SYSTEM

Applicability

Applies to the surveillance of the instrumentation and associated devices which initiate reactor scram.

Objective

To specify the type and frequency of surveillance to be applied to the protection instrumentation.

Specification

A. Instrument channels shall be functionally tested and calibrated as indicated in Table 3.1.A/4.1.A.

TABLE 3.1.A/4.1.A

REACTOR PROTECTION SYSTEM INSTRUMENTATION

Minimum Number of Functional Channels per Trip System	Trip Function	Trip Level Setting	Functional Test Frequency	Alarm Level Setting	Calibration Frequency
1	Manual scram	N/A	once/month	N/A	N/A
2 [(1)] [(2)]	SRM high-high count	$\leq 5 \times 10^5$ cps	once/week	$\leq 5 \times 10^5$ cps (upscale HI HI) $\leq 10^5$ cps (upscale HI) ≥ 3 cps (downscale)	once/3 mos
2 [(3)]	High Water Level in Scram Discharge Tank	N/A	once/3 months	$\leq$ 50 Gallons	once/3 month

Notes for Table 3.1.A/4.1.A

(1) The SRM's are presently connected in the non-coincident mode in the RPS, where any 1 SRM upscale HI HI will initiate a scram. Of the 4 SRM channels, only 2 are required to be functional.

(2) It is recognized that SRM count rate will diminish and eventually disappear during the process of removing fuel from a reactor at which time there is no required proof of functionability.

(3) Channels A or C and B or D required.

3.1/4.1 BASES: REACTOR PROTECTION SYSTEM

The reactor protection system will be available for rod insertion in the unlikely event that a rod becomes withdrawn. This is highly improbable because the reactor mode switch is locked in the shutdown position and the control rod directional solenoids are electrically disarmed. The reactor protection system is functional as described in Section 7.2 of the FSAR with the manual scram and the SRM's required functional. The SRM's have been placed in the non-coincidence scram mode to provide automatic scram from high-high count rate.

The two level switches and their associated alarms provide assurance that an adequate volume exists in the scram discharge volume tank to accommodate a scram.

LIMITING CONDITIONS FOR OPERATION

3.2 PROTECTIVE INSTRUMENTATION

Applicability

Applies to the plant instrumentation which initates and controls a protective function.

Objective

To assure the operability of protective instrumentation.

Specification

A. Surveillance Instrumentation

The limiting conditions for the instrumentation that provides surveillance information required in the period during which these specifications are effective are given in Table 3.2.A.

1. Reactor water level shall be maintained at $\geq$ 555" above vessel zero whenever irradiated fuel is in the reactor vessel.

SURVEILLANCE REQUIREMENTS

4.2 PROTECTIVE INSTRUMENTATION

Applicability

Applies to the surveillance requirement of the instrumentation that initiates and controls protective function.

Objective

To specify the type and frequency of surveillance to be applied to protective instrumentation.

Specification

A. Surveillance Instrumentation

The surveillance requirements associated with the equipment listed in Table 3.2.A shall be that indicated in Table 4.2.A.

1. The reactor water level shall be verified once each shift.

Table 3.2.A

SURVEILLANCE INSTRUMENTATION

Minimum # of Operable Instrument Channels	Instrument #	Instrument	Type Indication and Range	Alarm Setting	Notes
2	LI-3-206 or LR-3-53 or LI-3-53 or LI-3-55 and LI-3-46A or 46B	Reactor Water Level	Indicator 0" to 60" Recorder 0" to 60" Indicator 0" to 60" Indicator 0" to 400" Indicator +60" to -155"	Low ≥ 27", high ≤39"	(1)(4)
2	PI-3-54 PR-3-53	Reactor Pressure	Indicator 0-1200 psig Recorder 0-1200 psig	High ≤ 1040 psig	(1)(5)
2	PR-64-50 and PI-64-67	Drywell Pressure	Recorder 0-80 psig Indicator 0-80 psig		(1)(5)
2	TI-64-52A and TR-64-52	Drywell Temperature	Indicator 0-400° F. Recorder 0-400° F.	High ≤ 145° F.	(1)(5)
2	TI-64-55A and TIS-64-55	Suppression Chamber Water Temperature	Indicators 0-400° F.	High ≤ 90° F	(1)(4)
1	LI-64-54A or LI-64-66	Suppression Chamber Water Level	Indicator -25" to +25"		(1)(4)
1	NA	Control Rod Position	Continuity		(2)(4)
2	SRM A, B, C, D	Neutron Monitoring	Indicator and Recorder 0.1 to 10^{+6} cps -100 to +10 sec. (period)	Downscale ≥ 3 cps Retract permit ≥ 100 cps Upscale HI ≤ 10^5 cps Upscale HI-HI ≤ $5x10^5$ cps Period ≥ 30 sec.	(1)(3)(4)
1	LS-78-2A	Fuel Storage Pool level high	NA	El. 663' 1 1/2"	(6)
1	LS-78-2B	Fuel Storage Pool level low	NA	El. 662' 7 1/2"	(6)
1	TR-74-80 PT 17	Fuel pool temperature	Recorder 0-600°F	≤ 125°F	(6) (7)

NOTES FOR TABLE 3.2.A

(1) If one of the instrument channels monitoring a parameter should become incapable of performing its intended function, all operations which could affect the associated system will be suspended until the item is returned to service.

(2) The control rod position indicator full-in switches will be operable for every control rod and will provide indication in the control room, or the control rod position will be verified to be full-in by a continuity check of the full-in limit switches at the local panel. Prior to the unbolting of the reactor vessel head, control rod position indication to verify full-in position will be established in the control room and maintained thereafter when fuel is in the vessel.

(3) The following Source Range Monitoring Channels will be operable and will provide count rate indication and alarms in the control room:

SRM Channel A or C

SRM Channel B or D

The alarms in the control room will be as follows:

SRM TRIPS

Trip Function	Trip Action
SRM upscale or inoperative	Annunciator, amber light
SRM Downscale	Annunciator, white light

In addition, the SRM's have been placed in the non-coincidence scram mode to provide for protection against high neutron flux. It is recognized that SRM count rate will diminish and eventually disappear during the process of removing fuel from the reactor at which time there is no required proof of functionability.

(4) Only required whenever irradiated fuel is in the reactor vessel.

(5) Only required whenever the reactor pressure vessel head is bolted in place.

(6) Only required when irradiated fuel is in the fuel pool.

(7) If this instrument channel is out of service, alternate manual means will be used until it is returned to service.

MINIMUM TEST AND CALIBRATION FREQUENCY FOR SURVEILLANCE INSTRUMENTATION
(POST ACCIDENT INSTRUMENTATION)

	Instrument Channel	Calibration Frequency (3)	Instrument Check
1)	Reactor Water Level	Once/6 months	Each Shift
2)	Reactor Pressure	Once/6 months	Each Shift
3)	Drywell Pressure	Once/6 months	Each Shift
4)	Drywell Temperature	Once/6 months	Each Shift
5)	Suppression Chamber Water Temperature	Once/6 months	Each Shift
6)	Suppression Chamber Water Level	Once/6 months	Once/Week (4)
7)	Control Rod Position	NA	Each Shift (1)
8)	Neutron Monitoring (SRM)	Once/3 months (2)	Each Shift
9)	Drywell Pressure (PS-64-67)	Once/6 months	NA
10)	Fuel Pool Level (LS-78-2A, 2B)	Once/6 months	Once/Day
11)	Fuel Pool Temperature (TR-74-80)	Once/6 months	Once/Day

NOTES FOR TABLE 4.2.A

(1) Control rod position indication to verify the full-in position will be maintained in the control room on Unit 2. While the reactor vessel head is bolted in place unit 1 control rods which do not have position indication in the control room will be monitored at the local panel by use of the full-in limit switches at intervals not to exceed once/3 days. Prior to and subsequent to the unbolting of the reactor vessel head, control rod position indication to verify the full-in position will be maintained in the control room. This monitoring circuitry will be tested once a week.

(2) The SRM will be functionally tested on frequency of once per week.

(3) Instrument calibration will be verified upon restoration, then at the frequency listed.

(4) Instrument check will be to observe instrument response as transmitter is removed and returned to service.

3.2/4.2 BASES: PROTECTIVE INSTRUMENTATION

For each parameter monitored, as listed in Table 3.2.A, there are equivalent or redundant channels of instrumentation as noted. By comparing readings between two channels, a near continuous surveillance of instrument performance is available. Any deviation in readings will necessitate an early recalibration, thereby maintaining the quality of the instrument readings.

The redundant monitoring of reactor vessel level will be sufficient to provide the operator with information with which to determine the adequacy of the coolant inventory. The approach of abnormal conditions will be brought to the operator's attention by audible alarms.

Instrumentation has been selected to provide control room indication of adequate information regarding status of the drywell pressure and temperature and torus water level and temperature. Monitoring of information concerning key primary containment parameters will ensure that sufficient control of these parameters can be manually initiated in a timely manner.

A minimum of two channels of the SRM's, each on separate power supplies, will be operable. Each channel will provide indication of count rate and provide alarms as listed in Table 7.5-1 of the FSAR.

An automatic scram from high-high count rate of any SRM's channel as well as a manual scram will be available. One control rod drive pump each for units 1 and 2 will be kept in operation and all the accumulators will be kept charged with nitrogen and water. The scram pilot valves and pneumatically controlled valves to scram each control unit will be operable.

3.2/4.2 BASES: PROTECTION INSTRUMENTATION (CONTINUED)

Control rod position indication to verify the full-in position will be maintained on unit 2. The continuity of all the full-in limit switches on unit 1 will be monitored periodically. Under the conditions to be maintained in Temporary Technical Specification 3.3, the probability of a control rod withdrawal is significantly lower than that following a scram from normal conditions. Prior to removing the reactor vessel head bolts, a monitoring system will be provided in the main control room to alert the operator if any rod drift should occur.

The reactor water level will be nominally maintained at a level greater than 27" but less than 39" as indicated on LR-3-53. Above or below this level will give an alarm of "Reactor Water Level Abnormal." 27" indication on LR-3-53 corresponds to 555" above vessel zero.

In the unlikely event that loss of fuel pool level occurs, the low level alarm is set high enough above the fuel such that operator action can be taken to establish makeup before the level reaches 8-1/2 feet above the fuel. Refer to the specification 3.10 for fuel pool water conditions.

LIMITING CONDITIONS FOR OPERATION

3.3 REACTIVITY CONTROL

Applicability

Applies to the operational status of the control rod system.

Objective

To assure the ability of the control rod system to control reactivity while fuel is in the reactor vessel.

Specification

While fuel is in the reactor vessel the requirements of 3.3.A through 3.3.G shall be met.

A. All control rods shall be inserted in the full-in position.

B. The directional control valves shall be disarmed electrically for all control rods.

C. The manual valves in the drive water supply shall be in the shut position to prohibit rod movement.

D. The control rod accumulators shall be charged.

E. Two SRM channels shall be functional.

F. One control rod drive pump shall be in service.

SURVEILLANCE REQUIREMENTS

4.3 REACTIVITY CONTROL

Applicability

Applies to the surveillance requirements of the control rod system.

Objective

To verify the ability of the control rod system to control reactivity.

Specification

A. Control rod position shall be verified in accordance with Table 4.2.A.

B. Each directional control valve shall be verified to be electrically disarmed at intervals not to exceed once every 3 days.

C. The drive water supply valve (85-593) to each hydraulic control unit shall be verified closed and the water supply valves (85-612, 85-613) to each shall be verified open at intervals not to exceed once every 3 days.

D. The accumulator pressure shall be checked once a day.

E. The count rate shall be recorded once each shift.

F. The control rod drive pump discharge pressure shall be checked once per shift.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.3 REACTIVITY CONTROL	4.3 REACTIVITY CONTROL
Specification	Specification
G. Prior to unbolting the reactor vessel head, and thereafter while fuel is in the reactor when normal control rod position indication is not available, the continuity of the control rod full-in limit switches shall be monitored continuously with visual indication and audible alarm in the control room to indicate that any control rod is not fully inserted or that the monitoring circuit is interrupted or deenergized.	G. The temporary circuit provided to monitor control rods without position indication shall be functionally tested prior to unbolting the reactor vessel head and at intervals not to exceed once per week.

To prevent an inadvertent or spurious withdrawal of a control rod the directional control valves of each control rod have been electrically disarmed. As a further precaution, the valve in the drive water supply to each hydraulic control unit will be closed. In the unlikely event that a control rod does become withdrawn, two channels of the SRM's are required to be available for visual indication of neutron level. Although the SRM's may not immediately respond to a single rod movement, they would be adequate to monitor the approach to criticality. Also the SRM's are connected in the non-coincidence scram mode to provide rapid rod insertion from a high-high count rate on either SRM. Additionally, a manual scram will also be available. To assure that the control rods can be scrammed, the control rod accumulators are required to be charged with nitrogen and water pressure and a control rod drive pump is required to be in service. To provide additional indication of a control rod withdrawal, the control rod position indicator full-in switches will be functional for every control rod and will provide indication in the control room except, when the reactor vessel head is bolted in place, the control rod position may be verified to be full-in by a continuity check of the full-in limit switch at the local panel. Prior to unbolting the reactor vessel head, circuitry will be provided to permit continuous monitoring in the control room of continuity through the full-in limit switches when normal indication is not available. The circuitry will be testable and will provide a light and audible alarm in the control room to assure operator attention should any one rod be moved from the fully inserted position. Once the fuel is removed from the reactor vessel, the control of reactivity is assured by the design of the spent fuel storage pool as described in FSAR, Chapter 10, Subsection 3.

LIMITING CONDITIONS FOR OPERATION

.4 STANDBY LIQUID CONTROL SYSTEM

Applicability

Applies to the operating status of the Standby Liquid Control System.

Objective

To assure the availability of a system with the capability to shut down the reactor and maintain the shutdown condition without the use of control rods.

Specification

A. Normal System Availability

1. The standby liquid control system shall be functional whenever fuel is in the reactor vessel.

2. The injection line manual isolation valve 63-524 will will be closed except when the system is required for emergency injection.

SURVEILLANCE REQUIREMENTS

4.4 STANDBY LIQUID CONTROL SYSTEM

Applicability

Applies to the surveillance requirements of the Standby Liquid Control System.

Objective

To verify the operability of the Standby Liquid Control System.

Specification

A. Normal System Availability

The operability of the Standby Liquid Control System shall be verified by the performance of the following tests:

1. At least once per month each pump loop shall be functionally tested.

2. At least once during each operating cycle:

a. Check that the setting of the system relief valves is 1425 ± 75 psig.

b. Manually initiate the system, except explosive valves. Pump boron solution through the recirculation path and back to the Standby Liquid Control Solution Tank. Minimum pump flow rate of 39 gpm

LIMITING CONDITIONS FOR OPERATION | SURVEILLANCE REQUIREMENTS

3.4 STANDBY LIQUID CONTROL SYSTEM

4.4 STANDBY LIQUID CONTROL SYSTEM

against a system head of 1275 psig shall be verified. After pumping boron solution, the system shall be flushed with demineralized water.

c. Manually initiate one of the Standby Liquid Control System loops and pump demineralized water into the reactor vessel.

This test checks explosion of the charge associated with the tested loop, proper operation of the valves, and pump operability. Replacement charges shall be selected such that the age of charges in service shall not exceed five years from the manufacturers assembly date.

d. Both systems, including both explosive valves, shall be tested in the course of two operating cycles.

B. Operation with Inoperable Components:

1. From and after the date that a redundant component is made or found to be inoperable, Specification 3.4.A.1 shall be considered fulfilled if the inoperable component is returned to service within seven days.

B. Surveillance with Inoperable Components:

1. When a component is found to be inoperable, its redundant component shall be demonstrated to be operable immediately and daily thereafter until the inoperable component is repaired.

LIMITING CONDITIONS FOR OPERATION

3.4 STANDBY LIQUID CONTROL SYSTEM

C. Sodium Pentaborate Solution

At all times when the Standby Liquid Control System is required to be operable the following conditions shall be met:

1. The net volume - concentration of the Liquid Control Solution in the liquid control tank shall be maintained as required in Figure 3.4.1.

2. The temperature of the liquid control solution shall be maintained above the curve shown in Figure 3.4.2. This includes the piping between the standby liquid control tank and the suction inlet to the pumps.

SURVEILLANCE REQUIREMENTS

4.4 STANDBY LIQUID CONTROL SYSTEM

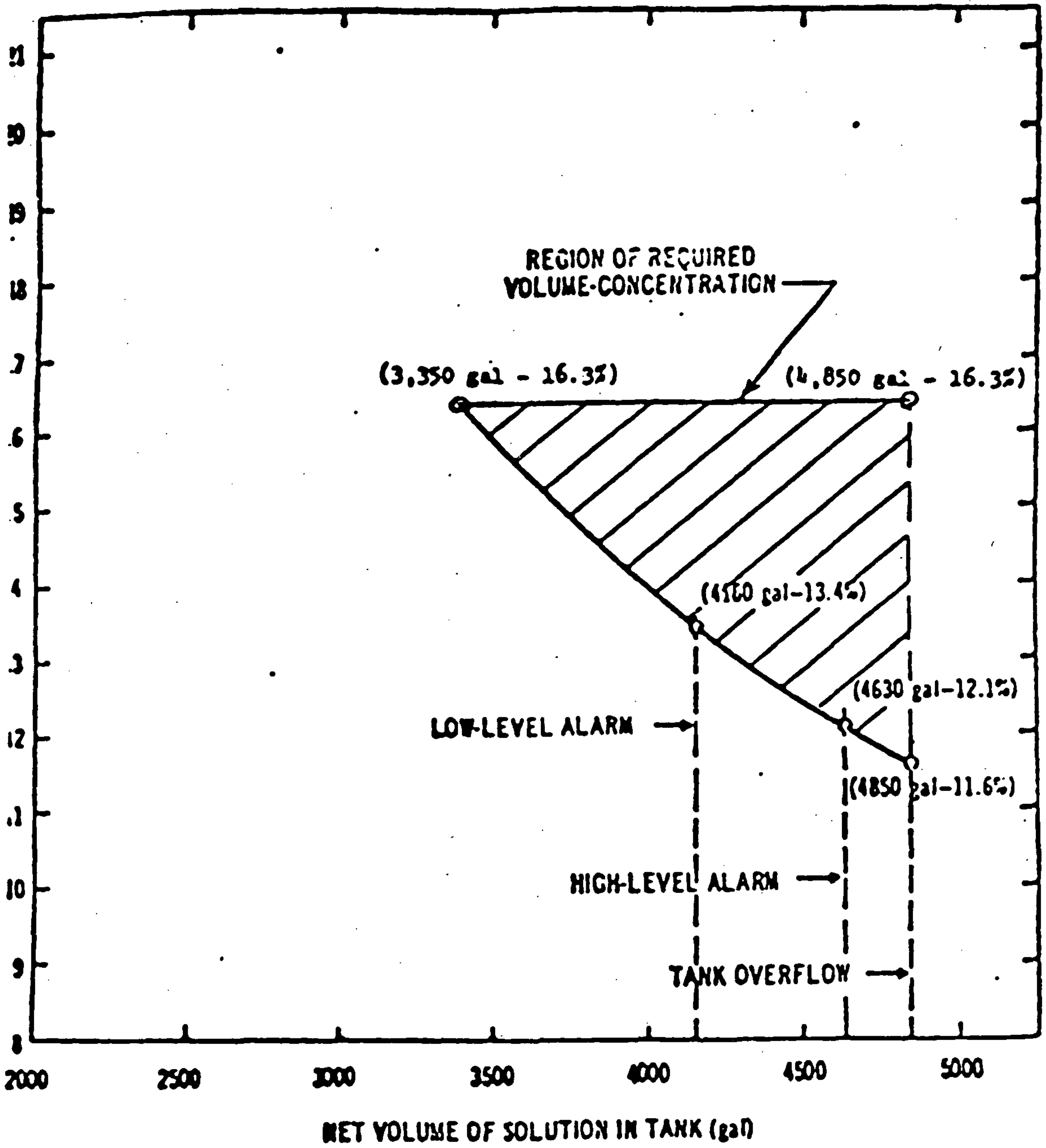
C. Sodium Pentaborate Solution

The following tests shall be performed to verify the availability of the Liquid Control Solution:

1. Volume: Check at least once per day.

2. Temperature: Check at least once per day.

3. Concentration: Check at least once per month. Also check concentration any time water or boron is added to the solution or solution temperature is below the temperature required in Figure 3.4.2.

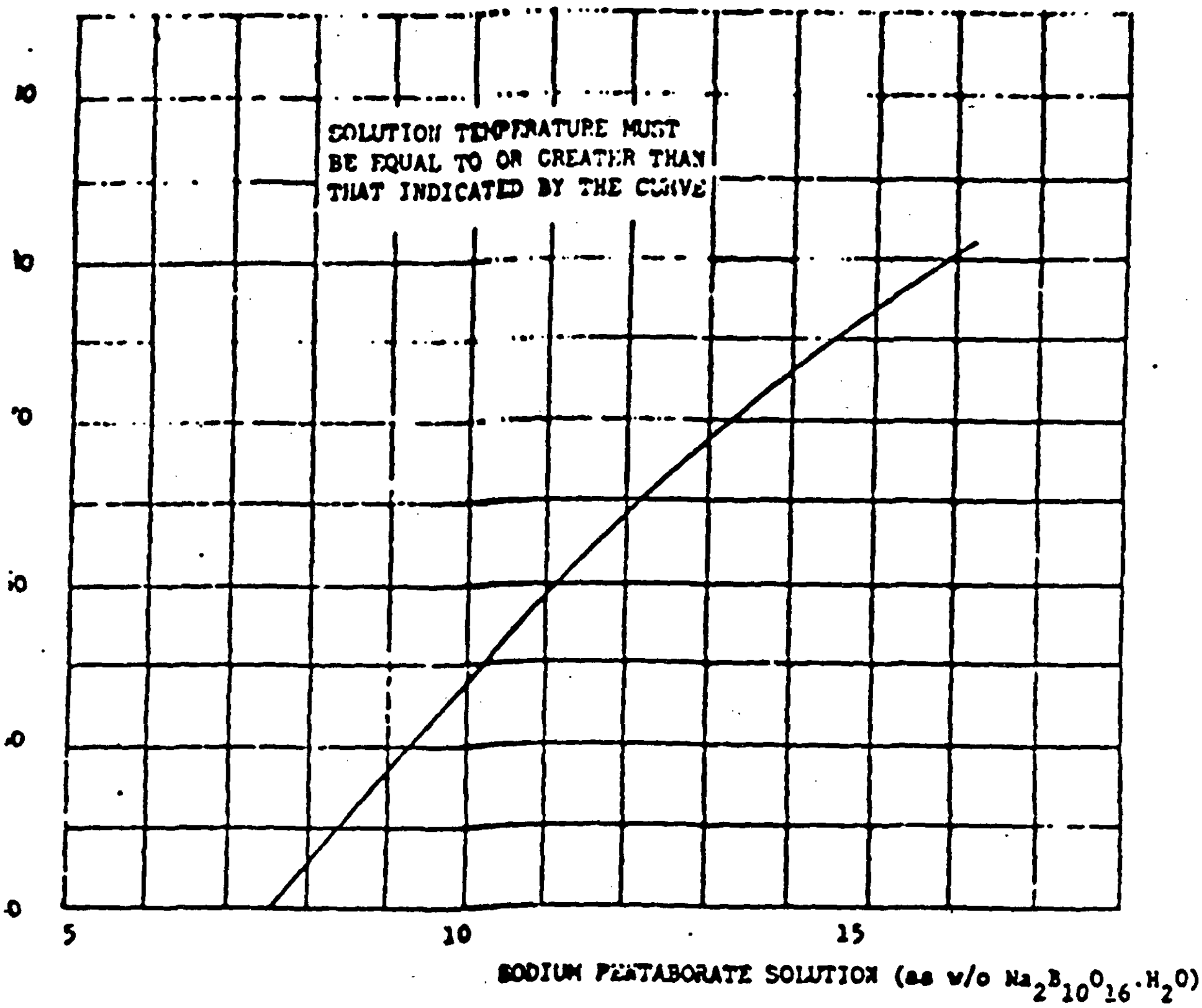


BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

SODIUM PENTABORATE SOLUTION
VOLUME-CONCENTRATED REQUIREMENTS

FIGURE 3.4-1

JUN 1 3 1975



BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

SODIUM PENTABORATE SOLUTION
TEMPERATURE REQUIREMENTS
FIGURE 3.4-2

BASES: STANDBY LIQUID CONTROL SYSTEM

A. The purpose of the liquid control system is to provide the capability of maintaining the reactor in a cold shutdown condition assuming that none of the withdrawn control rods can be inserted. To meet this objective, the liquid control system is designed to inject a quantity of boron that produces a concentration greater than 600 ppm of boron in the reactor core in less than 125 minutes.

The minimum limitation on the relief valve setting is intended to prevent the loss of liquid control solution via the lifting of a relief valve at too low a pressure. The upper limit on the relief valve settings provides system protection from overpressure.

B. Only one of the two standby liquid control pumping loops is needed for operating the system. Assurance that the remaining system will perform its intended function and that the long-term average availability of the system is not reduced is obtained from a one-out-of-two system.

C. Level indication and alarm indicate whether the solution volume has changed, which might indicate a possible solution concentration change. The test interval has been established in consideration of these factors. Temperature and liquid level alarms for the system are annunciated in the control room.

The solution is kept at least 10° F above the saturation temperature to guard against boron precipitation. The margin is included in Figure 3.4.2.

BASES: STANDBY LIQUID CONTROL SYSTEM (Continued)

C. (Continued)

The volume concentration requirements of the solution are such that, should evaporation occur from any point within the curve, a low level alarm will annunciate before the temperature-concentration requirements are exceeded.

The quantity of stored boron includes an additional margin (25 percent) beyond the amount needed to shut down the reactor to allow for possible imperfect mixing of the chemical solution in the reactor water.

A minimum quantity of 4,160 gallons of solution having a 13.4 percent sodium pentaborate concentration or the equivalent is required to meet this shutdown requirement as defined in Figure 3.4.1.

Experience with pump operability indicates that the monthly test, in combination with the tests during each operating cycle, is sufficient to maintain pump performance. Various components of the system are individually tested periodically, thus making unnecessary more frequent testing of the entire system.

The solution temperature and volume are checked at a frequency to assure a high reliability of operation of the system should it ever be required.

LIMITING CONDITIONS FOR OPERATION

3.5 CORE, CONTAINMENT, AND FUEL POOL COOLING SYSTEMS

Applicability

Applies to the operational status of the core, suppression pool, and fuel pool cooling systems.

Objective

To assure the operability of the core and containment cooling systems under all conditions for which this cooling capability is an essential response to plant abnormalities.

Specification

A. Core Spray System (CSS)

1. When irradiated fuel is in the reactor vessel, at least one core spray loop with one pump and available diesel generator shall be capable of delivering flow.

B. Residual Heat Removal System (RHRS) Containment and Shutdown Cooling

1. When irradiated fuel is in the reactor vessel at least two RHR loops with one pump per loop

SURVEILLANCE REQUIREMENTS

4.5 CORE, CONTAINMENT, AND FUEL POOL COOLING SYSTEMS

Applicability

Applies to the surveillance requirements of the core, suppression pool, and fuel pool cooling systems when the corresponding limiting conditions for operation are in effect.

Objective

To verify the operability of the core and suppression pool cooling systems under all conditions for which this cooling capability is an essential response to plant abnormalities.

Specification

A. Core Spray System (CSS)

1. Core Spray System Testing

	Item	Frequency
a.	Pump operability	upon restoration and monthly thereafter
b.	Motor operated valve operability	upon restoration and monthly thereafter

2. When it is determined that one core spray loop with one pump and available diesel generator is incapable of deliverin

LIMITING CONDITIONS FOR OPERATION

3.5 CORE CONTAINMENT, AND FUEL POOL COOLING SYSTEMS

shall be capable of manual operation. Diesel generators must also be available to power the pumps. A service water supply must be available.

C. Spent Fuel Pool Cooling System

1. Whenever irradiated fuel is stored in the spent fuel pool, the cooling system for the spent fuel pool shall be capable of maintaining the temperature of the fuel pool coolant $\leq 125^{\circ}F$.
2. When irradiated fuel is stored in the spent fuel pool, except as specified in 3.5.C.3 two fuel pool cooling pumps and heat exchangers shall be capable of maintaining fuel pool temperatures as specified in 3.5.C.1. Diesel Generators must also be available to power the pumps. The associated RBCCWS loop and service water system must also be operating.

SURVEILLANCE REQUIREMENTS

4.5 CORE CONTAINMENT, AND FUEL POOL COOLING SYSTEMS

flow, another core spray pump with an available diesel generator shall be selected, and all active components in the flow paths shall be immediately demonstrated to be capable of delivering flow.

B. Residual Heat Removal System (RHRS) (Containment and Shutdown Cooling)

1. Residual Heat Removal System Testing

	Item	Frequency
a.	Pump operability	Upon restoration and monthly thereafter
b.	Motor operated valve operability	Upon restoration and monthly thereafter

2. When it is determined that one RHR pump (containment and suppression pool cooling) or associated heat exchanger is incapable of delivering flow and removing heat at a time when flow capability and heat removal are required, the remaining RHR pump and associated heat exchanger and available diesel generator, and all active components in the flow paths [illegible]

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3. Whenever the temperature of the water in the spent fuel pool exceeds 125°F or one of the fuel pool cooling systems is made or becomes unavailable, the RHR supplemental fuel pool cooling mode shall be operable or operating if necessary to maintain the fuel pool temperature less than 125°F.	
4. Whenever irradiated fuel is stored in the fuel pool, the gates on the fuel transfer canal between Unit 1 and Unit 2 shall be left in place and the transfer canal drain valves 1-78-561 and 1-78-562 shall be shut; except if a Unit fuel pool cleanup system becomes unavailable, the transfer canal between Unit 1 and Unit 2 may be opened allowing the other Unit fuel pool cleanup system to be used for both units.	

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS

3.5 CORE, CONTAINMENT, AND FUEL POOL COOLING SYSTEMS

D. RHR Spent Fuel Pool Supplemental Cooling

1. Whenever irradiated fuel is stored in the spent fuel pool at least one RHR loop with one RHR pump capable of being supplied with diesel power shall be capable of manual operation in the fuel pool supplemental cooling mode. The associated RHRSW pump, heat exchanger must also be functional.

E. RHR Service Water System (RHRSWS) Emergency Equipment Cooling Water System (EECWS)

1. When irradiated fuel is in the reactor vessel, 4 RHRSW pumps must be functional and aligned to RHR header service corresponding to the 4 selected RHR pumps.

2. At all times, at least 2 RHRSW pumps shall be assigned to

4.5 CORE, CONTAINMENT, AND FUEL POOL COOLING SYSTEMS

capable of delivering flow and heat removal immediately and weekly thereafter until the inoperable RHR pump and associated heat exchanger is returned to service or an alternate pump and heat exchanger with an available diesel generator selected and verified.

C. Spent Fuel Pool Cooling System

1. The spent fuel pool water temperature shall be checked and recorded at least every 24 hours.

2. Prior to opening the transfer canal between the Unit 1 and Unit 2 fuel pools, and daily thereafter, verify that drain valves 1-78-561 and 1-78-562 are shut.

D. RHR Spent Fuel Pool Supplemental Cooling

1. Routine surveillance for the RHR pump is specified 4.5.B.1.

2. When it is determined that the selected RHR pump for spent fuel pool supplemental cooling is incapable of heat removal, another RHR pump capable of

LIMITING CONDITIONS FOR OPERATION

3.5 CORE, CONTAINMENT AND FUEL POOL COOLING SYSTEMS

EECW header service with one pump assigned to each header. Each pump must run continuously with its loss of voltage trip deactivated, or it shall be capable of automatic start in its normal D/G load sequencing mode of operation. Each pump shall be assigned to a separate diesel power supply.

3. Whenever irradiated fuel is stored in the reactor vessel one RHRSW pump (D1 or D2) capable of being supplied with diesel power normally or alternately assigned to the RHR heat exchanger header supplying the standby coolant supply connection must be available with a flow path to the reactor vessel.

4. Whenever irradiated fuel is stored in the spent fuel pool, two independent flow paths for water make up to the spent fuel pool will be available from two RHRSW pumps, capable of being supplied by diesel power.

SURVEILLANCE REQUIREMENTS

4.5 CORE, CONTAINMENT, AND FUEL POOL COOLING SYSTEMS

being supplied with diesel power shall be selected, and all active components required for heat removal capability, shall be demonstrated to be capable of delivering flow.

3. The RHR pump and valves used for fuel pool supplemental cooling shall be tested upon restoration and monthly thereafter.

4. If requirements 4.5.D.1 and 4.5.D.2 cannot be met, the requirements of specification 4.5.E.5 shall be met.

E. RHR Service Water System (RHRSWS) And Emergency Equipment Cooling Water System (EECWS)

1. RHR Service Water System

Each of the four RHRSW pumps and associated essential control valves on the RHR heat exchanger headers shall be demonstrated to be functional upon restoration and once every three months thereafter if not in continuous service.

LIMITING CONDITIONS FOR OPERATION

3.5 CORE, CONTAINMENT, AND FUEL POOL COOLING SYSTEMS

F. Maintenance of Filled Discharge Pipe

Whenever the core spray system or RHR systems are required to be functional, the discharge piping from the pump discharge of these systems to the last block valve shall be filled. The condensate head tank shall be aligned to serve the discharge piping of the RHR and CS pumps. The pressure indicators on the discharge piping of the RHR and CS pumps shall indicate not less than listed below.

PI-75-20	70 psig
PI-75-48	70 psig
PI-74-51	70 psig
PI-74-65	70 psig

SURVEILLANCE REQUIREMENTS

4.5 CORE, CONTAINMENT, AND FUEL POOL COOLING SYSTEMS

2. EECW System

Each of the 2 RHRSW pumps assigned to EECW service will be in continuous service. Associated essential control valves on the EECW headers shall be demonstrated to function once every three months.

3. When it is determined that one RHRSW pump and associated control valves on an RHR heat exchanger header is incapable of delivering flow at a time when flow delivery capability is required, the remaining RHRSW pumps and associated heat exchangers and available diesel generators, and all active components in the flow paths shall be demonstrated to be capable of delivering flow immediately and weekly thereafter until the inoperable RHRSW pump is returned to service or an alternate pump is selected and verified.

4. When it is determined that one RHRSW pump and associated

LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

4.5 CORE, CONTAINMENT, AND FUEL POOL COOLING SYSTEMS

control valves on an EECW header is incapable of delivering flow at a time when flow delivery capability is required, an alternate RHRSW pump on a corresponding diesel generator shall be selected and assigned to the same EECW header.

5. At intervals not to exceed 7 days each independent fuel pool makeup flow path from its respective EECW header to the fire hose outlet connection on the refueling floor will be tested to verify makeup water supply availability.

F. Maintenance of Filled Discharge Pipe

The following surveillance requirements shall be adhered to to assure that the discharge piping of the core spray system and RHR system are filled:

LIMITING CONDITIONS FOR OPERATION

3.10 CORE ALTERATIONS

H. Restoration Work

1. Removal of fire damaged equipment shall not be initiated until all fuel from both cores is stored in their fuel storage pools and the fuel pools and the fuel pool gates between the fuel pools and the reactor cavities are installed with the canal blocks in place.

2. After all fuel from both cores is stored in its respective fuel storage pool and the fuel pool gates are installed with canal blocks in place, removal of fire damaged equipment may be carried out in accordance with the "Plan for Evaluation, Repair, and Return to Service of Browns Ferry Units 1 and 2" and revisions thereto up to and including Revision 10 as supplemented by licensee's letter dated June 11, 1975 (Gilleland to Rusche, NRC)

3.10/4.10 BASES: CORE ALTERATIONS

A. Refueling Reactivity Controls

The electrical circuits to the insert and withdraw directional control valves of each control rod drive hydraulic control unit will be disconnected at the valves to prevent inadvertent or spurious withdrawal of a control rod during fuel removal. As a further precaution, the manual valve in the drive water supply to each hydraulic control unit will be closed.

Each control rod is mechanically latched in the full-in position and withdrawal would require a prescribed set of operations to (1) relieve pressure on the latch by a slight insertion, (2) release the latch, and (3) withdraw the rod. The likelihood of the required operations occurring asthe result of any spurious, inadvertent, or accident condition is so small as to be incredible. The control rod drives are seismically qualified. Breaking or cutting any hydraulic line will not cause a control rod withdrawal. The full-in positions of all the control rods in both units will be indicated, or monitored, continuously.

Under the conditions to be maintained, the probability of a control rod withdrawal is significantly lower than that following a scram from normal conditions.

Redundant SRM alarms in the control room would alert the operator of significant reactivity increase, and automatically initiate a scram. Manual scram capability is maintained while fuel is in the reactor.

The refueling interlocks are designed to back up procedural core reactivity controls during refueling operations. By sensing

3.10/4.10 BASES: CORE ALTERATIONS

A. Refueling Reactivity Controls (Continued)

the condition of the refueling equipment and the control rods, the interlocks prevent the refueling platform from being moved over the core if a control rod is withdrawn and fuel is on a hoist. Likewise, if the refueling platform is over the core with fuel on a hoist, control rod motion is blocked by the interlock. These interlocks are activated by placing the mode switch in the refueling position. When the mode switch is in the refuel position, one control rod can be withdrawn.

By requiring that the mode switch remain in the shutdown position while removing fuel from the reactor vessel, control rod withdrawal is prohibited. In addition, the insert and withdrawal directional control valves of each control rod drive hydraulic control unit will be disconnected at the valves to prevent inadvertent or spurious withdrawal of a control rod, and the valve in the drive water supply to each hydraulic control unit will be closed. Since the ability to withdraw a control rod is not present, the need for the refueling interlocks no longer exists.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
	4.5 CORE, CONTAINMENT, AND FUEL POOL COOLING SYSTEM 1. Prior to the operation of the RHRS and core spray systems, the discharge piping of these systems shall be vented from the high point and water flow determined. 2. Following any period where the RHR or core spray systems have not been required to be operable, the discharge piping of the nonfunctional system shall be vented from the high point prior to the return of the system to service. 3. When the RHRS and CSS are required to be functional the pressure indicators which monitor the discharge lines shall be monitored daily and the pressure logged.

3.5.A Core Spray System (CSS)

The core spray loop provides one of several means of supplying water to the reactor vessel. The capacity of one core spray pump is in excess of 10 times the amount of water required to cool the reactor core in its present shutdown condition. Power operation of the reactor will not be permitted, therefore, the margin available for this purpose will increase with time. Technical Specifications 3.5.A and 3.5.B require that at least three pumps will be available to assure that adequate cooling of the fuel is accomplished. However, considerably more equipment is available, including additional core spray, RHR, control rod drive, condensate booster, hotwell and raw water pumps.

Analyses have been performed to demonstrate that in excess of 35 hours is available to restore core cooling if for some unforeseen reason cooling is interrupted. Studies were performed to show that a core spray loop could be manually placed in operation in less than one hour. Hence, considerable margin is available for manual actuation.

3.5 BASES: CORE, CONTAINMENT, AND FUEL POOL COOLING SYSTEMS

3.5.B Residual Heat Removal System (RHRS) (Containment and Shutdown Cooling)

The decay heat removal requirements for one unit in the cold shutdown condition can be conservatively met by the operation of one RHR pump and its associated RHR heat exchanger in the shutdown cooling mode. The total heat load for the heat exchanger is estimated to be less than one-fourth of the heat exchanger capability under the required flow and temperature conditions. At least three RHR pumps and associated heat exchangers on each unit are capable of delivering flow and removing the decay heat. The low decay heat and absence of pressure which could foster an unacceptable loss of coolant allows ample time for manual operation in accordance with established operating instructions.

3.5 BASES: CORE, CONTAINMENT, AND FUEL POOL COOLING SYSTEMS

3.5.C Spent Fuel Pool Cooling System

The spent fuel pool cooling system consists of two 600 gpm pumps and heat exchangers. Figure D-2 of the TVA Safety Analysis of the BFNP Units 1 and 2 included as Part VI, Section I, of the "Plan for Evaluation, Repair, and Return to Service of Browns Ferry Units 1 and 2 (March 22, 1975, Fire)" shows that the system is capable of maintaining the spent fuel coolant temperature below 125°F based on the ANS standard 3-year operation decay heat curve. One pump and heat exchanger will provide sufficient cooling capability to prevent pool boiling.

In response to NRC Question 10.1, dated March 25, 1971, TVA committed to modify and upgrade the spent fuel pool cooling system to qualify as a Seismic Class I system. The system was subsequently analyzed and designed in accordance with the requirements of ANSI B 31.1.0, 1967. The loading combinations and allowable stresses used in the analysis were in accordance with ASME Section III, Subsection NC, 1971 requirements. The analysis meets the intent and requirements of ASME Section III, 1974. All piping, valves, and equipment as shown in Figure Q10.1-1 of the response, except that identified as nonseismic, were analyzed.

The gates on the fuel transfer canal between Unit 1 and Unit 2 will be left in place and the transfer canal drain line will be valved out. This provides redundant seismically qualified barriers for the prevention of pool leakage through the transfer canal. The transfer canal connecting the two pools may be opened allowing each fuel pool clean-up system to provide redundancy for the other in the event of a Unit Fuel Pool Clean-up System failure provided the canal drains are verified closed.

3.5 BASES: CORE, CONTAINMENT, AND FUEL POOL COOLING SYSTEMS

3.5.D RHR Spent Fuel Pool Supplemental Cooling

The decay heat removal requirements for a full core stored in the fuel pool can be conservatively met by the operations of one RHR pump and its associated RHR heat exchanger in the fuel pool cooling mode. The total heat load for this mode is estimated to be less than 60 per cent of the heat exchanger capability under the required flow and temperature conditions.

During the original preoperational test program, each RHR pump was operated in the fuel pool cooling mode of operation. The test included operation of the RHR system (1000-2000 gpm) in parallel with the fuel pool cooling system. Two fuel pool cooling pumps were operated with 600 gpm per pump. This test verified the ability to operate the RHR system in parallel with the fuel pool cooling system at design flow rates.

One RHR pump and associated RHRSW pump capable of being supplied with diesel power will be functional any time spent fuel is stored in the spent fuel pool.

3.5.2 RHR Service Water System (RHRSWS) Emergency Equipment Cooling Water System (EECWS)

The decay heat removal cooling water requirements for two units in the cold shutdown condition can be conservatively met by the operation of one RHRSW pump on one heat exchanger on each unit. One RHRSW pump is required for each unit if the units are using heat exchangers which are not on the same service water header. Four RHRSW pumps are presently available and capable of delivering flow to meet this requirement. Less than one-half the flow delivery capability of each pump is needed to remove the present decay heat for each unit. The low decay heat level and ample flow delivery capability allow ample time for manual operation in accordance with established operating instructions.

The standby emergency equipment cooling water (EECW) requirements for two units in the cold shutdown condition can be adequately met by the operation of one RHRSW pump on an EECW header. The EECW system is not required for normal plant shutdown operation because the required cooling water is supplied by the plant raw cooling water system. The principal immediate need for EECW flow is in the event that a diesel engine should be started. In this case, EECW flow must be established at once. To meet this requirement, two RHRSW pumps are assigned to EECW service are running continuously with each aligned to a separate supply header. Each of these pumps are operating continuously (with loss of voltage trips deactivated) or they must be capable of automatic start in their normal diesel generator load sequencing mode of operation. Either header will supply all EECW requirements. In addition, two other RHRSW pumps are available and capable of delivering flow by manual operations in accordance with established operating instructions. The two RHRSW pumps which are running continuously or have automatic restarting capability are assigned to 4.16 kV shutdown boards which have associated operable diesel generators and the pumps are

3.5 BASES: CORE, CONTAINMENT, AND FUEL POOL COOLING SYSTEMS

3.5.E RHR Service Water System (RHRSWS) Emergency Equipment Cooling Water System (EECWS) (Continued)

arranged with automatic restart provisions in the event that board undervoltage should occur.

In the unlikely event that all make up capability is lost, water can be supplied to the reactor or fuel pool by certain RHRSW pumps directly from the river. The D1 or D2 RHRSW pumps can pump through the RHRSW header and standby coolant supply line in to RHR loop II on Unit 1, or RHR loop I on Unit 2. From the RHR loop the water can be routed to the reactor through the LPCI injection valves or the fuel pool through the fuel pool system connections on each RHR loop. An alternate path is available which is independent of the RHR and fuel pool cooling systems. The alternate path will only require any one of four RHRSW pumps and manual valve operation to provide make up coolant directly to the fuel pool through a hose connected to either the north or south EECW header hydrant. The RHRSW pump has an on site power source. The make up capability provided by the RHRSW pump far exceeds the amount needed to replace the water lost at the maximum evaporation rate possible with the present decay heat.

3.5 BASES: CORE, CONTAINMENT, AND FUEL POOL COOLING SYSTEMS

3.5.F Maintenance of Filled Discharge Pipe

If the discharge piping of the core spray and RHR system are not filled, a water hammer can develop in this piping when the pump and/or pumps are started. To minimize damage to the discharge piping and to ensure added margin in the operation of these systems, this Tehcnical Specification requires the discharge lines to be filled whenever the system is in a functional condition. If a discharge pipe is not filled, the pumps that supply that line must be assumed to be nonfunctional for Technical Specification purposes.

The core spray and RHR system discharge piping high point vent is visually checked for water flow prior to any pump operation to ensure that the lines are filled. The visual checking will avoid starting the core spray or RHR system with a discharge line not filled. In addition to the visual observation and to ensure a filled discharge line other than prior to testing, a head tank located approximately 100 feet above the discharge line high point supplies makeup water for these systems. System discharge pressure indicators are used to determine the water level above the discharge line high point. The indicators will reflect approximately 30 psig for a water level at the high point and approximately 70 psig for a water level at the head tank and are monitored daily to ensure that the discharge lines are filled.

3.6 PRIMARY SYSTEM BOUNDARY

Applicability

Applies to the status of the reactor coolant system.

Objective

To assure the integrity of the reactor coolant system.

Specification

A. Thermal and Pressurization Limitations

The reactor vessel head bolting studs shall not be under tension unless the temperature of the vessel head flange and the head is greater than 100° F.

B. Coolant Chemistry

Reactor water shall be maintained within the following limits: Conductivity $\leq$10 umhos/cm @ 25° C. and Chloride $\leq$0.5ppm.

C. Safety and Relief Valves

Four safety relief valves which have accumulators shall be functional on each unit with two valves powered from each of two 250V DC MOV boards when the reactor vessel head is bolted in place.

4.6 PRIMARY SYSTEM BOUNDARY

Applicability

Applies to the periodic examination and testing requirements for the reactor coolant system.

Objective

To determine the condition of the reactor coolant system and the operation of the safety devices related to it.

Specification

A. Thermal and Pressurization Limitations

When the reactor vessel head bolting studs are tensioned and the reactor is in a cold condition, the reactor vessel shell temperature immediately below the head flange shall be permanently recorded.

B. Coolant Chemistry

1. A sample of reactor coolant shall be analyzed:

   a. at least every 96 hours for conductivity and chloride ion content.

   b. at least every 8 hours for conductivity and chloride ion content when the continuous conductivity monitor is inoperable.

LIMITING CONDITIONS FOR OPERATION

3.6 PRIMARY SYSTEM BOUNDARY

D. Reactor Vessel Head Vent

The reactor vessel head vents shall be in the open position when the reactor vessel head is bolted in place and the head vent piping is assembled.

SURVEILLANCE REQUIREMENTS

4.6 PRIMARY SYSTEM BOUNDARY

C. Safety and Relief Valves

The position indicating lights for the four safety relief valves specified in 3.6.C.1 shall be checked each operating shift to confirm control power available for actuation when the reactor vessel head is bolted in place.

D. Reactor Vessel Head Vent

The position indicating lights on the reactor vessel head vents shall be checked each operating shift to confirm the valves are indicated to be in the open position when the reactor vessel head is bolted in place and vent piping is assembled.

3.6.A/4.6.A BASES: THERMAL AND PRESSURIZATION LIMITATIONS

Tightening the studs on the reactor vessel head flexes it slightly to bring together the entire contact surfaces adjacent to the "O" rings of the head and vessel flange. The reactor vessel head flange and head are constructed such that their initial maximum NDTT is 10° F, as cited in paragraph 4.2.7 of the safety analysis report. Therefore, the initial minimum temperature at which the studs can be placed in tension is established at 40° F + 60° F or 100° F.

3.6.B/4.6.B BASES: COOLANT CHEMISTRY

Materials in the primary system are primarily 304 stainless steel and the Zircaloy cladding. The reactor water chemistry limits are established to prevent damage to these materials. Limits are placed on conductivity and chloride concentrations. Conductivity is limited because it is continuously measured and gives an indication of abnormal conditions and the presence of unusual materials in the coolant. The chloride limit is specified to prevent stress corrosion cracking of stainless steel and is based on established relationships between stress corrosion, chloride concentrations and dissolved oxygen. Zircaloy does not exhibit similar stress corrosion failures.

When conductivity is in its normal range, pH and chloride and other impurities affecting conductivity must also be within their normal range. Chloride measurements are made to determine whether or not they are also out of their normal operating values because conductivity could be high due to the presence of a neutral salt which would not have an effect on pH or chloride. In some types of water-cooled reactors, conductivities are in fact high due to purposeful addition of additives. In the case of BWR's, however, where no additives

3.6.B/4.6.B BASES: COOLANT CHEMISTRY (Continued)

are used and where near neutral pH is maintained, conductivity provides a very good measure of the quality of the reactor water. Significant changes therein provide the operator with a warning mechanism so he can investigate and remedy the condition causing the change before limiting conditions, with respect to variables affecting the boundaries of the reactor coolant, are exceeded.

Methods available for correcting the off-standard conditions include using the cleanup demineralizer, reducing the input rate of impurities, or changing the supply of cooling water. The conductivity of the reactor coolant is continuously monitored. The samples of the coolant which are taken every 96 hours will serve as a reference for calibration of these monitors and are considered adequate to assure accurate readings of the monitors. If conductivity is within its normal range, chlorides and other impurities will also be within their normal ranges. The reactor coolant samples will also be used to determine the chlorides. Therefore the sampling frequency is considered adequate to detect long-term changes in the chloride ion content. Reactor coolant sampling is increased to once per shift when the continuous conductivity monitor is unavailable.

3.6.C/4.6.C BASES: SAFETY AND RELIEF VALVES

With the reactor in the cold shutdown condition and with low decay heat, four safety relief valves are more than adequate to assure the integrity of the reactor coolant systems. Control power from different 250V MOV boards assures availability of two valves even in the unlikely event of loss of a 250V MOV board. Surveillance checks each shift assure control power is available for valve actuation. When the reactor vessel head is not bolted in place, the relief valves are not required.

[illegible].6.D/4.6.D BASES: REACTOR VESSEL HEAD VENT

Maintaining the reactor vessel head vents in the open position with the RHR shutdown cooling system in service keeps the reactor vessel at atmospheric pressure when in the cold shutdown condition. Surveillance checks of the position indicating lights are made each shift.

When the reactor vessel head is not bolted in place or the vessel head is removed or the head vent piping is removed from the head, adequate venting capability exists.

LIMITING CONDITIONS FOR OPERATION

3.7 CONTAINMENT SYSTEMS

Applicability

Applies to the status of the primary containment systems and refuel zone ventilation.

Objective

To monitor pressure and temperature of the primary containment system and provide refuel [illegible] ventilation.

Spec[illegible]

A. [illegible]

1. [illegible]

2. [illegible] well [illegible] ture shall [illegible]. (See Table [illegible])

SURVEILLANCE [illegible]

4.7 CONTAINMENT SYSTEMS

Applicability

Applies to the primary containment and refuel zone ventilation.

Objective

To verify the pressure and temperature of the primary containment and assure air exhaust from the refuel zone.

Specification

A. Primary [illegible]

1. The suppression chamber water level and temperature shall be checked once per 24-hours when fuel [illegible] in the reactor.

2. The drywell pressure and temperature shall be checked once per 24-hours when the reactor vessel head is bolted in place.

LIMITING CONDITIONS FOR OPERATIONS	SURVEILLANCE REQUIREMENTS
3.7 CONTAINMENT SYSTEMS	4.7 CONTAINMENT SYSTEMS
	Specification
B. REFUEL ZONE VENTILATION	B. REFUEL ZONE VENTILATION
When the reactor vessel head is unbolted, and when the fuel is stored in the fuel pool, there shall be one refueling floor exhaust fan and one supply fan in service per unit.	When the reactor vessel head is unbolted operation of one refueling floor supply fan and one exhaust fan shall be verified and documented once per shift.
C. STANDBY GAS TREATMENT	C. STANDBY GAS TREATMENT
When the reactor vessel head is unbolted and when the fuel is stored in the fuel pool, one train of the Standby Gas Treatment System shall be functional and aligned to the Reactor Building Zones of Units 1 and 2.	When the reactor vessel head is unbolted the capability of one train of the Standby Gas Treatment System to exhaust air shall be demonstrated weekly.
D. REACTOR BUILDING VENTILATION	D. REACTOR BUILDING VENTILATION
When the reactor vessel head is unbolted and when fuel is stored in the fuel pool the reactor building zones for Units 1 and 2 shall be ventilated such that a negative pressure is maintained with respect to the refueling zone.	When the reactor vessel head is unbolted and when fuel is stored in the fuel pool the reactor building zones for Units 1 and 2 shall be monitored daily to assure a negative pressure is being maintained with respect to the refueling zone.

LIMITING CONDITIONS FOR OPERATIONS

3.7 CONTAINMENT SYSTEMS

E. INOPERABLE COMPONENTS

Whenever the requirements of specifications 3.7.B and 3.7.C cannot be met, all fuel handling activities or any activity over irradiated fuel in the vessel or fuel pool shall not be permitted.

3.7 BASES: CONTAINMENT SYSTEMS

A. PRIMARY CONTAINMENT

This specification ensures indication of adequate information regarding status of the drywell pressure and temperature and suppression chamber water level and temperature when fuel is in the reactor. When fuel is removed this requirement is no longer necessary. Monitoring of information concerning these primary containment parameters will ensure that sufficient control of these parameters can be manually initiated in a timely manner.

B. REFUEL ZONE VENTILATION

This specification ensures that an air flow is established in the refuel zone. Refuel zone ventilation may be accomplished by continuous operation of one refuel floor exhaust fan and one supply fan per unit. Fan dampers will be secured in the open position to ensure flow continuity.

Refuel zone air flow provides a means of exhausting moisture vapor in the unlikely event that spent fuel pool coolant steaming occurs.

C. STANDBY GAS TREATMENT

In the event that normal refuel zone ventilation is inoperable one standby gas treatment train is capable of providing a means of exhausting moisture vapor and air from the Reactor Building Zones of Units 1 and 2 to maintain a slight negative pressure between the reactor building zones and the refueling zone.

D. REACTOR BUILDING ZONES - Δp AND CHLORIDE CONTROL

The normal ventilation systems for the reactor building zones and the refuel zone are separate systems and will be aligned to maintain control of airborne chloride contamination from the reactor building zones to the refuel zone. Using this alignment, the reactor building zones will be maintained at a negative pressure with respect to the refuel zone and thereby preclude airborne chloride contamination of the refuel zone.

In the event of loss of off-site power the standby gas treatment system will perform the function of the reactor zone ventilation system.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
	4.8.B Airborne Effluents

. Radioactive gases released from each units turbine and reactor building roof vents, the radwaste building roof vents, and the main stack shall be continuously monitored. To accomplish this, at least one reactor building and one turbine building vent monitoring system per unit shall be operating whenever that unit's building ventilation system is in service. Also one radwaste building system vent monitoring channel shall be operating whenever the radwaste building ventilation system is in service. At least one main stack monitoring channel shall be operating whenever any units air ejector, mechanical vacuum pump, or a standby gas treatment system train is in service. If normal monitoring systems are not available, temporary monitors or other systems shall be used to monitor effluent. A monitoring channel may be out of service [illegible] 4 hours for functional testing and calibration without providing a temporary monitor.

The primary containment shall be purged through the standby gas treatment system.

C. Radiological Environmental Monitoring Program

An environmental monitoring program shall be conducted as described below and outlined in Table 4.8.F.

1. Atmospheric Monitoring

a. The atmospheric monitoring network is divided into three subgroups consisting of 12 monitoring stations of which 9 stations shall be operable at all times. The monitoring stations are shown on Figure 4.8-1. These monitoring locations are subject to change depending upon continued evaluation of the environmental monitoring program. The station at Muscle Shoals will be used as background reference.

Each monitor shall be capable of continuously sampling air at [illegible] flow of approximately three cubic feet per

CONDITIONS OF OPERATION

3.9 AUXILIARY ELECTRICAL SYSTEM

Applicability

Applies to the auxiliary electrical power system.

Objective

To assure an adequate supply of electrical power for those systems required for safety.

Specification

A. Auxiliary Electrical Equipment

For the condition of both units 1 and 2 in cold shutdown with fuel in the reactor, the following conditions apply except as specified in section 3.9.B.

1. Two off-site sources of power to the shutdown system available from either the 500 kV system (backfeeding) or the 161 kV transmission system.
2. Three diesel generators available and their associated 4 kV shutdown boards. All 4 kV shutdown boards energized unless offsite power is lost.

SURVEILLANCE REQUIREMENTS

4.9 AUXILIARY ELECTRICAL SYSTEM

Applicability

Applies to the periodic testing requirements of the auxiliary electrical systems.

Objective

Verify the capability of the auxiliary electrical system.

Specification

A. Auxiliary Electrical Equipment

1. Diesel Generators

   a. Each diesel generator shall be manually started and loaded once each month to demonstrate operational readiness. The test shall continue for at least a one-hour period at 75% of rated load or greater. During the monthly generato test the diesel generator starting air compressor shall be checked for operation and its ability to recharge air receivers.

CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.9 AUXILIARY ELECTRICAL SYSTEM	**4.9 AUXILIARY ELECTRICAL SYSTEM**

3.9 AUXILIARY ELECTRICAL SYSTEM

3. 4 kV shutdown bus 1 or 2 energized.
4. 480V shutdown boards associated with each unit are energized.
5. Under voltage relays on the three 4 kV shutdown boards associated with the functional diesel generators shall be available.
6. 250V DC shutdown board batteries and chargers for the four functional boards must be available.
7. The 250V unit board batteries and a battery charger for each battery and associated battery boards must be functional.
8. Batteries, chargers, busses, and feeds to the four SRM channels specified in Table 3.2.A shall be operable in each unit.
9. There shall be a minimum of 103,300 gallons of diesel fuel in the standby diesel generator fuel tanks.

4.9 AUXILIARY ELECTRICAL SYSTEM

The operation of the diesel fuel oil transfer pumps shall be demonstrated, and the diesel starting time to reach rated voltage and speed shall be logged.

b. Once a month the quantity of diesel fuel available shall be logged.

c. Each diesel generator shall be given an annual inspection in accordance with instructions based on the manufacturer's recommendations.

d. Once a month a sample of diesel fuel shall be checked for quality. The quality shall be within the acceptable limits specified in Table 1 of ASTM D975-68 and logged.

2. D.C. Power System - Unit Batteries(250-Volt) Diesel Generator Batteries (125-Volt) and Shutdown Board Batteries (250-Volt).

CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.9 AUXILIARY ELECTRICAL SYSTEM	4.9 AUXILIARY ELECTRICAL SYSTEM

3.9 AUXILIARY ELECTRICAL SYSTEM

B. Cold Shutdown on Units 1 and 2 With Inoperable Equipment

1. From and after the date that off-site power is reduced to one source, another source must be restored within seven days.

2. When one of the three diesel generators becomes inoperable and a shutdown bus and three 4 kV shutdown boards are available, 2 days are allowed to comply with 3.9.A.2.

3. When one 4-kV shutdown board is inoperable, both off-site 161-kV transmission lines and both common station transformers shall be available and the remaining 4-kV shutdown boards and associated diesel generators, and all 480V emergency shutdown boards shall be functional.

4. From and after the date that one of the three 250-Volt unit batteries and/or its associated battery board is found to be

4.9 AUXILIARY ELECTRICAL SYSTEM

a. Every week the specific gravity and the voltage of the pilot cell, and temperature of an adjacent cell and overall battery voltage shall be measured and logged.

b. Every three months the measurements shall be made of voltage of each cell to nearest 0.1 volt, specific gravity of each cell, and temperature of every fifth cell. These measurements shall be logged.

3. Undervoltage Relays

a. Once every 6 months, the conditions under which the undervoltage relays are required shall be simulated with an undervoltage on each shutdown board to demonstrate that the associated diesel generator w[ill] start.

b. The undervoltage relays which start the diesel

3.9 AUXILIARY ELECTRICAL SYSTEM

inoperable for any reason, the NRC shall be notified within 24 hours of the situation, the precautions to be taken during this period and the plans to return the failed component to an operable state.

5. From and after the date that one of the four 250-Volt shutdown board batteries and/or its associated battery board is found to be inoperable for any reason, the NRC shall be notified within 24 hours of the situation, the precautions to be taken during this period and the plans to return the failed component to an operable state.

C. Auxiliary Electrical Equipment

For the condition of no fuel in Units 1 and 2 reactor vessels and irradiated fuel in the fuel pools for both Units 1 and 2 the following conditions apply except as specified in 3.9.D.

4.9 AUXILIARY ELECTRICAL SYSTEM

generators from the 4-kV shutdown boards, shall be calibrated annually for trip and reset and the measurements logged.

B. Operation with Inoperable Equipment

1. When one common station transformer, or one 161-kV line is found to be inoperable, all diesel generators and associated boards must be demonstrated functional immediately and daily thereafter.
2. When one of the three diesel generators is found to be inoperable, the remaining diesel generators and associated boards shall be demonstrated functional immediately and daily thereafter.
3. When one 4-kV shutdown board is found to be inoperable, all remaining 4-kV shutdown boards and two associated diesel generators, shall be demonstrated functional, immediately and daily thereafter.

CONDITIONS OF OPERATION	SURVEILLANCE REQUIREMENTS

3.9 AUXILIARY ELECTRICAL SYSTEM

1. Two off-site sources of power to the shutdown system available from either the 500 kV system (backfeeding) or the 161 kV transmission system.
2. Two diesel generators available and their associated 4 kV shutdown boards available for use.
3. 4 kV shutdown bus 1 or 2 energized.
4. The 480-Volt shutdown boards associated with the divisions of fuel pool cooling in service are energized.
5. Under voltage relays on the two 4 kV shutdown boards associated with the functional diesel generators shall be functionally operable.
6. 250-Volt DC shutdown board batteries and chargers for the two functional boards must be available.
7. The 250-Volt unit board batteries and a battery charger

4.9 AUXILIARY ELECTRICAL SYSTEM

C. Auxiliary Electrical Equipment

1. Diesel Generators

a. Each diesel generator shall be manually started and loaded once each month to demonstrate operational readiness. The test shall continue for at least a one-hour period at 75% of rated load or greater. During the monthly generator test the diesel generator starting air compressor shall be checked for operation and its ability to recharge air receivers. The operation of the diesel fuel oil transfer pumps shall be demonstrated, and the diesel starting time to reach rated voltage and speed shall be logged.

b. Once a month the quantity of diesel fuel available shall be logged.

c. Each diesel generator shall

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3.9 AUXILIARY ELECTRICAL SYSTEM

and associated battery boards are functional.

8. There shall be a minimum of 103,300 gallons of diesel fuel in the standby diesel generator fuel tanks.

D. Fuel Removed from Units 1 and 2 Reactors and Stored in Units 1 and 2 Fuel Pools with Inoperable Equipment

1. From and after the date that off-site power is reduced to one source, another source must be restored within seven days.

2. When one of the two diesel generators becomes inoperable and a shutdown bus and two 4 kV shutdown boards are available, 2 days are allowed to comply with 3.9.C.2.

3. From and after the date that one of the three 250-Volt unit batteries and/or its associated battery board is found to be inoperable for any reason, the NRC shall be notified within 24 hours of the situation, the precautions to be taken during

4.9 AUXILIARY ELECTRICAL SYSTEM

be given an annual inspection in accordance with instructions based on the manufacturer's recommendations.

b. Once a month a sample of diesel fuel shall be checked for quality. The quality shall be within the acceptable limits specified in Table 1 of ASTM D975-68 and logged.

2. D.C. Power System - Unit Batteries (250-Volt) Diesel Generator Batteries (125-Volt) and Shutdown Board Batteries (250-Volt).

a. Every week the specific gravity and the voltage of the pilot cell, and temperature of an adjacent cell and overall battery voltage shall be measured and logged.

b. Every three months the measurements shall be made of voltage of each cell to nearest 0.1 volt, specific gravity of each cell, and temperature of every fifth cell. These measurements shall be logged.

CONDITIONS OF OPERATION

3.9 AUXILIARY ELECTRICAL SYSTEM

this period and the plans to return the failed component to an operable state.

4. From and after the date that one of the two 250-Volt shutdown board batteries and/or its associated battery board is found to be inoperable for any reason, the NRC shall be notified within 24 hours of the situation, the precautions to be taken during this period and the plans to return the failed component to an operable state.

SURVEILLANCE REQUIREMENTS

4.9 AUXILIARY ELECTRICAL SYSTEM

3. Undervoltage Relays

a. Once every 6 months, the conditions under which the undervoltage relays are required shall be simulated with an undervoltage on each shutdown board to demonstrate that the associated diesel generator will start.

b. The undervoltage relays which start the diesel generators from the 4-kV shutdown boards shall be calibrated annually for trip and reset and the measurements logged.

D. Operation with Inoperable Equipment

1. When one common station transformer, or one 161 kV line is found to be inoperable, the diesel generators and associated boards must be demonstrated functional immediately and daily thereafter.

2. When one of the two diesel generators is found to be inoperable, the remaining die

CONDITIONS OF OPERATION	SURVEILLANCE REQUIREMENTS
3.9 AUXILIARY ELECTRICAL SYSTEM	4.9 AUXILIARY ELECTRICAL SYSTEM
	generator and associated board shall be demonstrated functional immediately and daily thereafter.

3.9 BASES: AUXILIARY POWER SYSTEM

The auxiliary power system consists of 3 functionally operable diesel generators (A, B, and D) capable of automatic start on loss of voltage to their respective 4-kV shutdown boards. The C diesel feed is inoperable but its associated 4-kV shutdown board is available. The 480V shutdown boards 1A, 1B, 2A, and 2B are operable with at least one source of power. All the 480V reactor MOV boards 1A, 1B, 1C, 2A, 2B, and 2C are operable with at least one source of power.

A and B. Cold Shutdown on Units 1 and 2

The requirements of these specifications are based on the minimum complement of equipment to redundantly maintain the facility in a safe condition. The capacity requirements of this mode of operation can be adequately addressed with two diesels. The third diesel is to accommodate loss of off-site power and single failures in the diesel generator plant.

All 4-kV and 480V boards are available and the shutdown busses are available. With a single failure in any of the 4-kV and 480V board or feeders, diesel power can be directed to an adequate complement of equipment to accommodate the shutdown mode. This is accomplished by either feeding the required boards and their associated loads directly or by feeding two boards from one diesel through the backfeed capability of the shutdown bus.

There are eight 250-volt d-c battery systems each of which consists of a battery, battery charger, and distribution equipment. Three of these systems provide power for unit control functions, operative power for unit motor loads, and alternative drive power for a 115-volt a-c unit preferred motor-generator set. One 250-volt d-c system provides power for common plant and transmission system control functions, drive power for a 115-volt a-c plant preferred motor-generator set, and emergency drive power for certain unit large motor loads. The four remaining systems deliver control power to the 4160-volt shutdown boards.

Each 250-volt d-c shutdown board control power supply can receive power from its own battery, battery charger, or from a spare charger. The chargers are powered from normal plant auxiliary power or from the standby diesel-driven generator system. Zero resistance short circuits between the control power supply and the shutdown board are cleared by fuses located in the respective control power supply. Each power supply is located in the reactor building near the shutdown board it supplies. Each battery is located in its own independently ventilated battery room.

The 250-volt d-c system is so arranged, and the batteries sized such, that the loss of any one unit battery will not prevent the safe cooldown of both units in the event of the loss of offsite power and single failure in any one unit. The loss of one 250-volt shutdown board battery affects normal control power only for the 4160-volt shutdown board which it supplies. The station battery supplies loads that are not essential for safe shutdown and cooldown of the nuclear system. This battery was not considered in the accident load calculations.

<u>C and D. Fuel Removed from Units 1 and 2 Reactors and Stored in Units 1 And 2 Fuel Pools</u>

The requirements of these specifications are based on the minimum complement of equipment to redundantly maintain the fuel in the fuel pool in a safe condition. The capacity requirements of this mode of operation can be adequately addressed with one diesel. The second diesel is to accommodate loss of off-site power and single failures in the diesel generator plant.

Two 4-kV and two 480-volt boards will be available from off-site power supplied by one shutdown bus or from two diesel generators. With a single failure in any of the 4-kV and 480-volt boards or feeders, diesel power can be directed to an adequate complement of equipment to accommodate the fuel pool cooling mode.

The loss of any one unit battery will not prevent the safe cooling of both fuel pools in the event of the loss of offsite power and single failure in any one unit. The loss of one of the two 250-volt shutdown board batteries affects normal control power only for the 4160-volt shutdown board which it supplies.

4.9 BASES

The monthly tests of the diesel generators are primarily to check for failures and deterioration in the system since last use. The diesels will be loaded to at least 75 percent of rated power while engine and generator temperatures are stabilized (about one hour). The minimum 75 percent load will prevent soot formation in the cylinders and injection nozzles. Operation up to an equilibrium temperature ensures that there is no overheat problem. The tests also provide an engine and generator operating history to be compared with subsequent engine-generator test data to identify and to correct any mechanical or electrical deficiency before it can result in a system failure.

Battery maintenance with regard to the floating charge, equalizing charge, and electrolyte level will be based on the manufacturer's instruction and sound maintenance practices. In addition, written records will be maintained of the battery performance. The plant batteries will deteriorate with time but precipitous failure is unlikely. The type of surveillance called for in this specification is that which has been demonstrated through experience to provide an indication of a cell becoming irregular or unserviceable long before it becomes a failure.

The equalizing charge, as recommended by the manufacturer, is vital to maintaining the Ampere-hour capacity of the battery, and will be applied as recommended.

LIMITING CONDITIONS FOR OPERATION

3.10 CORE ALTERATIONS

Applicability

Applies to the removal of fuel from the reactor pressure vessel.

Objective

To prevent criticality during the removal of fuel from the reactor pressure vessel.

Specification

A. Reactivity Control

1. The reactor shall be kept in cold shutdown and all control rods fully inserted during fuel removal.
2. The reactor mode switch shall be locked in the "SHUTDOWN" position during fuel removal.

B. Core Monitoring

During fuel removal two channels of the SRM's, each on separate power supplies, shall be operable. For a SRM to be considered operable, the following conditions must be satisfied:

1. The SRM shall be fully inserted into the core.

SURVEILLANCE REQUIREMENTS

4.10 CORE ALTERATIONS

Applicability

Applies to the periodic verification of rod position and testing of instrumentation used during fuel removal.

Objective

To verify full insertion of control rods and operability of instrumentation during fuel removal.

Specification

A. Reactivity Control

1. Surveillance to verify full insertion of all control rods is specified in 4.3.
2. Prior to removing fuel, and daily thereafter, verify that the reactor mode switch is locked in the "SHUTDOWN" position.

B. Core Monitoring

Prior to the removal of fuel from the core the SRM's shall be functionally tested and checked for neutron response. Thereafter, while required to be operable, the SRM's will be checked daily for

LIMITING CONDITIONS FOR OPERATION

3.10 CORE ALTERATIONS

2. Prior to fuel removal the SRM shall have an initial minimum count rate of 3 cps with all rods fully inserted in the fully loaded core. The count rate will diminish during fuel removal.

C. Spent Fuel Pool Water Conditions

1. Whenever irradiated fuel is stored in the spent fuel pool, the pool water level shall be maintained at or above 8 1/2 feet above the top of the stored fuel.

2. Whenever irradiated fuel is in the spent fuel pool, the pool water temperature shall be ≤ 125°F.

3. Fuel pool water shall be maintained within the following limits:

Conductivity, 10 umho/cm @ 25°C
& Chloride, 0.5 ppm

D. Reactor Building Crane

1. The reactor building crane shall be operable:

a. When a spent fuel cask is handled.

SURVEILLANCE REQUIREMENTS

4.10 CORE ALTERATIONS

response until the count rate diminishes to ≤ 0.3 cps.

C. Spent Fuel Pool Water Conditions

1. Whenever irradiated fuel is stored in the spent fuel pool, the water level shall be recorded daily.

2. Whenever irradiated fuel is stored in the spent fuel pool, the water temperature shall be recorded daily.

3. A sample of fuel pool water shall be analyzed in accordance with the following specifications:

a. At least daily for conductivity and chloride ion content.

b. At least once per 8 hour shift for conductivity and chloride ion content when the fuel pool cleanup system is inoperable.

D. Reactor Building Crane

1. The following operational checks and inspections shall be performed on the reactor building crane prior to

LIMITING CONDITIONS FOR OPERATION | SURVEILLANCE REQUIREMENTS

3.10 CORE ALTERATIONS

b. Whenever new or spent fuel is handled with the 5-ton hoist.

2. Whenever there is irradiated fuel in the spent fuel pool or in the reactor vessel with the reactor vessel head off, crane loads shall not be transported over the spent fuel pool or reactor vessel opening except those operations in the spent fuel pool and reactor vessel which normally require the services of the overhead crane.

3. When the crane is used in the unit 3 area and fuel is in the spent fuel pools or in the reactor vessel with the reactor vessel head off of unit 1 or unit 2, bridge movement to the unit 1 or 2 areas shall be restricted by redundant key lock electrical limit switches. Bypassing the limit switches for required activities in the units 1 and 2 areas shall be under

4.10 CORE ALTERATIONS

handling of a spent fuel cask and new or spent fuel (these need not be performed more frequently than quarterly):

a. The cab and pendant controls shall be demonstrated to be operable on both the 125-ton hoist and the 5-ton hoist.

b. A visual inspection shall be made to ensure structural integrity of the 125-ton hoist, the 5-ton hoist and cask yoke safety wire ropes.

c. The overtravel limit switch interlocks, movement speed control and braking operations for the bridge, trolley, and hoists, the pendant interlocks, the main-auxiliary hoist operation interlock, and the remote emergency stop shall be functionally tested.

d. The bridge drive limit switches shall be functionally tested each time they are placed in service before the crane is loaded.

3.10 CORE ALTERATIONS

strict administrative controls requiring the signature of a senior reactor operator.

4. When the crane is used to handle materials between units 1, 2 and 3 areas and irradiated fuel is in the fuel pools or in the reactor vessel with the reactor vessel head off of units 1 or 2, rail stops shall be installed to prevent trolley movement over the fuel pools or open reactor vessels. Removal of the rail stops to permit required activities specified in 3.10.D.2 shall be governed by strict administrative controls requiring the signature of a senior reactor operator.

5. When the crane is unattended for a period greater than one hour, the bridge and trolley shall be tied down to prevent movement in case of a natural phenomena.

4.10 CORE ALTERATIONS

LIMITING CONDITIONS FOR OPERATION

3.10 CORE ALTERATIONS

E. Spent Fuel Cask

1. Upon receipt, an empty fuel cask shall not be lifted until a visual inspection is made of the cask-lifting trunnions and fastening con- nection

F. Spent Fuel Cask Handling - Refueling Floor

1. Administrative control shall be exercised to limit the height the spent fuel cask is

SURVEILLANCE REQUIREMENTS

4.10 CORE ALTERATIONS

2. At any time when the reactor building crane is in service there shall be a person assigned to the refueling floor who has authority to control crane activities and will observe crane motions and conditions. He shall document at least once per shift that the require- ments of Limiting Conditions for operation 3.10.D.2, 3.10.D.3, 3.10.D.4 and 3.10.D.5 have been co plied with.

E. Spent Fuel Cask

1. Prior to attachment and lifting of an empty spent fuel cask from the shipping trailer, a visual inspection shall be conducted on the lifting trunnions and the fasteners used to connect the trunnion to the cask.

2. A visual inspection shall be made of the assembled trunnion on the empty cask to ensure proper assembly.

3.10 CORE ALTERATIONS

F. Spent Fuel Cask Handling - Refueling Floor

1. (Continued)

raised above the refueling floor by the reactor building crane to 6 inches, except for entry into the cask decontamination chamber where height above the floor will be approximately 3 feet.

2. The spent fuel cask yoke safety cables shall be properly positioned at all times except when the cask is in the decontamination chamber.

G. Refueling Interlocks

1. The fuel grapple hoist load switch shall be set at $\leq$1,000 lbs.

2. If the frame-mounted auxiliary hoist, the monorail-mounted auxiliary hoist, or the service platform hoist is to be used for handling fuel with the head off the reactor vessel, the load sensing limit switch on the hoist to be used shall be set at <400 lbs.

3.10/4.10 BASES: CORE ALTERATIONS

B. Core Monitoring

The SRM's are provided to monitor the core during shutdown and to monitor core reactivity and guide the operator while fuel is being removed from the reactor pressure vessel. All fuel moves during this operation will reduce reactivity. It is expected that the SRM's will drop below 3 cps before all of the fuel is unloaded. Since there will be no reactivity additions during this period, the low number of counts will not present a hazard. When all of the fuel has been removed to the spent fuel storage pool, SRM's will no longer be required.

Requiring the SRM's to be functionally tested prior to fuel removal assures that the SRM's will be operable at the start of fuel removal. The daily response check of the SRM's ensures their continued operability until the count rate diminishes due to fuel removal.

3.10/4.10 BASES: CORE ALTERATIONS

C. Spent Fuel Pool Water Conditions

The design of the spent fuel storage pool provides a storage location for approximately 140 percent of the full core load of fuel assemblies in the reactor building which ensures adequate shielding, cooling, and reactivity control of irradiated fuel. An analysis has been performed which shows that a water level at or in excess of eight and one-half feet over the top of the stored assemblies will provide shielding such that the maximum calculated radiological doses do not exceed the limits of 10 CFR 20. The normal water level provides 14-1/2 feet of additional water shielding. The capacity of the skimmer surge tanks is available to maintain the water level at its normal height for three days in the absence of additional water input from the condensate storage tanks. All penetrations of the fuel pool have been installed at such a height that their presence does not provide a possible drainage route that could lower the normal water level more than one-half foot.

The fuel pool cooling and clean up system can maintain the fuel pool water temperature below 125°F when removing maximum normal heat load from the pool with the RBCCW temperature at its maximum. If the pool temperature approaches 125°F the standby RHR loop through its fuel pool cooling connection will provide for continued heat removal.

The fuel pool water chemistry limits are established to prevent damage to the 304 stainless steel used to fabricate the fuel pool. Limits are placed on conductivity and chloride concentrations. Conductivity is limited because it is easily measured and gives

3.10/4.10 BASES: CORE ALTERATIONS

C. Spent Fuel Pool Water Conditions (Continued)

an indication of abnormal conditions and the presence of unusual materials in the water. The chloride limit is specified to prevent stress corrosion cracking of stainless steel and is based on established relationships between stress corrosion, chloride concentrations, and dissolved oxygen.

When conductivity is in its normal range, pH and chloride and other impurities affecting conductivity must also be within their normal range. Chloride measurements are made to determine whether or not they are also out of their normal operating values because conductivity could be high due to the presence of a neutral salt which would not have an effect on pH or chloride.

No additives are used in the fuel pool water and near neutral pH is maintained, therefore conductivity provides a very good measure of the quality of the water. Significant changes therein provide the operator with a warning mechanism so he can investigate and remedy the condition causing the change before limiting conditions, with respect to variables affecting the boundaries of the fuel pool, are exceeded.

Methods available for correcting off-standard conditions include using the fuel pool filter demineralizer, halting fuel or equipment movements, or using the condensate demineralizers. The samples of the fuel pool water which are taken daily are considered adequate to detect long-term changes in the chloride ion content. Fuel pool sampling frequency is increased to once per eight hour shift when the fuel pool clean up system is inoperable.

3.10/4.10 BASES: CORE ALTERATIONS

D. Reactor Building Crane

The reactor building crane and 125-ton hoist are required to be operable for handling of the spent fuel in the reactor building. The controls for the 125-ton hoist are located in the crane cab. The 5-ton has both cab and pendant controls.

A visual inspection of the load-bearing hoist wire rope assures detection of signs of distress or wear so that corrections can be promptly made if needed.

The testing of the various limits and interlocks assures their proper operation when the crane is used.

Assignment of a person to monitor crane activities and who has authority to direct crane operations assures close, safety monitoring and compliance with crane handling requirements. The placement of the bridge limit switches in their respective drive control circuits and the installation of the rail stops on the trolley rails assures that crane movements cannot be inadvertently made over spent fuel pools with fuel in them. Crane bridge and trolley tie down when the crane is unattended prevents their motion in the case of a natural event, such as an earthquake.

3.10/4.10 BASES: CORE ALTERATIONS

E. Spent Fuel Cask

The spent fuel cask design incorporates removable lifting trunnions. The visual inspection of the trunnions and fasteners prior to attachment to the cask assures that no visual damage has occurred during prior handling. The trunnions must be properly attached to the cask for lifting of the cask and the visual inspection assures correct installation.

F. Restoration Work

Requiring completion of fuel storage and installation of gates and blocks places the fuel in a maximum security condition prior to the restoration activities.

3.10/4.10 BASES: CORE ALTERATIONS

F. Spent Fuel Cask Handling - Refueling Floor

Although single failure protection has been provided in the design of the 125-ton hoist drum shaft, wire ropes, hook and lower block assembly on the reactor building crane, the limiting of lift height of a spent fuel cask controls the amount of energy available in a dropped cask accident when the cask is over the refueling floor.

An analysis has been made which shows that the floor and support members in the area of cask entry into the decontamination facility can satisfactorily sustain a dropped cask from a height of three feet.

The yoke safety cables provide single failure protection for the hook and lower block assembly and limit cask rotation. Cask rotation is necessary for decontamination and the safety ropes are removed during decontamination.

LIMITING CONDITIONS FOR OPERATION | SURVEILLANCE REQUIREMENTS

3.11 UNIT 3/UNIT 1 AND 2 INTERACTIONS

Applicability

Applies to construction activities of unit 3.

Objective

To assure that unit 3 construction activities do not advercely affect the safety of either unit 1 or 2.

Specification

All operations, tests, and other activities associated with the construction of unit 3 that can affect any system or component required to be functional by these Interim Technical Specifications are prohibited until (1) an interation analysis is performed which shows that an activity associated with the construction and testing of unit 3 will not adversely affect the safety of units 1 and 2, and (2) the interaction analysis is approved by NRC.

3.11 BASES: UNIT 3/UNIT 1 AND 2 INTERACTIONS

Since there are shared systems among Units 1, 2, and 3 an interaction analysis is performed which shows that an activity associated with construction of Unit 3 will not adversely affect the safety of unit 1 or unit 2.

TABLE 6.8.A

MINIMUM SHIFT CREW REQUIREMENTS

Shift Position	Present	During One Unit Fuel Unloading	During Two Unit Fuel Unloading	With Fuel in Storage Pool	Type of License
Shift Engineers	1	1	1	1	SRO
Assistant Shift Engineers	1	2	2	1	SRO
Licensed Reactor Operator [1]	1	1	1	1	RO
Unit Operators	2	2	2	2	RO
Assistant Unit Operators	6	6	7	4	NONE
Health Physics Technician	1	1	1	1	NONE
Other Personnel	2	2	2	0	--
Minimum Shift Crew	14	15	16	10	

NOTES: SRO - Senior Reactor Operator

RO - Reactor Operator

NOTE FOR TABLE 6.8.A

[1]This position is normally filled by an assistant shift engineer, but as a minimum it may be filled by a licensed reactor operator. When the incumbent is not a senior reactor operator, he shall not be assigned duties requiring him to direct licensed activities of reactor operators.

BASES

As a result of the damage created by the fire to various units 1 and 2 cables, a number of safety systems cannot be operated from the control room. Equipment monitoring and operation is now required from various positions located throughout the facility. Analyses have shown that in excess of 15 hours are available to perform these various tasks. In addition, studies have been performed to assess the amount of time

TABLE 6.8.A

BASES (Continued)

required to perform these manual operations including consideration of additional equipment failures. Results of these studies have shown that operation of equipment for which the most degradation is experienced could be achieved in a time interval of several hours. These studies were performed with the assumption of a shift crew of 10 men. We have required a shift crew of 14 men for the present cold, shutdown condition, and one additional man per unit during defueling operations, i.e., 15 men if one unit is being defueled and 16 men if both units are being defueled. The additional men will perform defueling operations and, since defueling will be suspended in the event a manual transfer to back-up systems is necessary, will provide margin for performance of the required manual operations and monitoring.

After defueling is completed with all the fuel in the storage pool, the time available to perform manual transfer to back-up systems is extended even further. Therefore, a shift complement of 10 men is then adequate to provide the same level of safety.

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 10 TO THE FACILITY LICENSE DPR-33

AND AMENDMENT NO. 7 TO FACILITY LICENSE NO. DPR-52

(CHANGE NO. 11 TO TECHNICAL SPECIFICATIONS)

TENNESSEE VALLEY AUTHORITY

BROWNS FERRY NUCLEAR POWER PLANT, UNITS 1 & 2

DOCKET NOS. 50-259 AND 50-260

1.0 INTRODUCTION

On March 22, 1975, a fire at the Browns Ferry Nuclear Plant (BFNP) required the shutdown of Units 1 and 2. Since shortly after the start of the fire, both nuclear units have been in the cold shutdown condition with fuel in the cores and the respective reactor vessels vented to the containment atmosphere. Shutdown cooling systems are in operation in both units.

The fire affected various power, instrumentation and control cables which caused a number of emergency systems to become inoperable. Although many systems were damaged, the facility was placed and is being maintained in a safe and stable condition utilizing a substantial number of systems, and with functional backup systems being available. In fact, the total number of functional systems presently available for this purpose exceeds the number normally required for a facility in the cold shutdown condition.

The action now under consideration involves first steps toward the restoration of the facility. This involves the repair and removal of affected cables; however, before any repair and removal of cables and systems from the fire affected zone could be authorized, it was deemed prudent to require that all of the necessary vital systems be protected, isolated or verified as free from interconnection with any of the systems or cables which were damaged by the fire. Therefore, on May 9, 1975, the Commission issued Amendment 9 to Facility License No. DPR-33 (Unit 1) and Amendment 6 to Facility License No. DPR-52 (Unit 2), Change No. 10 to the respective Technical Specifications (designated the Temporary Technical Specifications), and a supporting Safety Evaluation. This safety evaluation assessed the safety of the facility in its shutdown fire damaged condition, and the Temporary Technical Specifications set forth the requirements to assure that the units would remain in a safe and stable condition until fire damage removal and restoration could be subsequently authorized.

As authorized in the May 9, 1975, Amendments, the various vital systems and components required to assure the safety of the facility have been re-established as functional by (1) reconnecting them with electrical power and control cables routed outside of the fire damage zone, (2) verifying that the original cables were outside of the fire damage zone and therefore are unaffected by the fire, (3) converting some systems and components to manual control and (4) placing some systems and components in the desired safe configuration and disconnecting power leads thus preventing spurious operation. Additionally, the primary system mode switch is, by administrative control, locked in the "Shutdown" mode.

The original "Plan for Evaluation, Repair and Return to Service of Browns Ferry Units 1 and 2," hereafter referred to as the "Plan", submitted by the licensee was based on leaving the cores in their respective vessels and isolating the required safety systems, to the extent possible, from potential interaction with fire affected systems during damage removal and restoration operations.

After discussions with the NRC staff, the licensee has determined that significantly greater safety would be provided by removing the fuel from both reactor vessels and placing it in the respective fuel storage pools prior to start of fire-damage removal and restoration operations. This has the effect of (1) virtually eliminating any potential for inadvertent criticality, (2) providing a substantially greater coolant inventory, and (3) reducing the number of systems required to maintain the facility in a safe condition. Thus, in the highly unlikely case where all fuel pool cooling is lost (heat removal would then be through boiling) and assuming no makeup, conservative analyses performed by the licensee have shown that a minimum of 160 hours are available before the pool level reaches the Technical Specification limit. The staff has verified by independent analyses that these estimates are very conservative. Thus more than adequate time is available for the restoration of pool cooling and makeup by manual operation of the available alternative systems. By comparison, approximately 28 hours would be available for this purpose before the core is uncovered if loss of cooling should occur with no makeup if the fuel were in the reactor vessel.

Accordingly, the licensee has requested authorization for removal of fuel from the cores and for its storage in the respective fuel storage pools, and for removal of fire damage and restoration of the facility. A revised Safety Analysis (Section E of Part VI dated 5/28/75 and 6/5/75 of the Plan) has been submitted in support of this request. Therefore, this safety evaluation addresses the safety considerations involved in the removal of the fuel from the cores and its transfer to the respective fuel storage pools, and the subsequent storage of the fuel in the pools while fire damage removal is in progress. Change 11 to the Technical Specifications (designated the Interim Technical Specifications) are being issued concurrently with this safety evaluation as part of Amendments 10 and 7 to Facility Licenses Nos. DPR-33 and DPR-52 respectively. These Interim Technical Specifications were prepared and submitted for approval by the licensee after extensive discussions with the NRC staff. The Interim Technical Specifications (Change 11) include all the provisions for assuring the safety of the facility for the condition with fuel in the reactor vessels that were included in the Temporary Technical Specifications (Change 10). In addition, Change 11 adds the provisions for assuring safety during defueling, fuel storage, and fire damage removal and restoration.

The safety considerations discussed in this safety evaluation and the requirements of the Interim Technical Specifications will assure the safe storage of fuel while facility restoration work is in progress, as well as during fire damage removal work. However, we have not yet been provided with a complete description of the design changes, criteria, and procedures for the repair and restoration of the facility. Therefore, this safety evaluation does not address and these amendments do not authorize the permanent repair and restoration of fire damage. Specific authorization in this regard will be issued after the necessary information has been submitted and reviewed and approved by the staff.

Likewise, this safety evaluation does not address and these amendments do not authorize the refueling and return to power operation of Browns Ferry Nuclear Plant Units 1 and 2. Any such authorization will be addressed in a subsequent safety evaluation and change to the Technical Specifications.

2. EVALUATION

2.1 General

This safety evaluatuion is based on a staff review of (1) the Safety Analysis (dated 5/28/75 and 6/5/75 included as Part VI, Section E, of the Plan; (2) Appendices A through G of the Safety Analysis; (3) other parts of the Plan as referenced in the Safety Analysis; (4) the BFNP FSAR; and (5) the information acquired through site inspections and extensive discussions with the licensee.

This safety evaluation addresses the safety considerations involved in the removal of fuel from the cores and its placement in the respective fuel storage pools (defueling), and the subsequent storage of the fuel in the pools while fire damage removal and restoration operations are in progress. This includes the defueling of Units 1 and 2, or the possible defueling of both units simultaneously. However, no cable cutting or fire damage removal operations will be permitted by the Interim Technical Specifications until the defueling of both units is completed, i.e., until all the fuel is in the respective fuel storage pool.

Safety is assured throughout the defueling-restoration operations by providing the systems and procedures required to accomplish the following:

a. Keep the fuel subcritical and monitor reactivity during fuel removal.
b. Remove decay heat from irradiated fuel in the reactor vessel and in the fuel storage pool and to maintain coolant over the fuel.
c. Assure the physical integrity of the fuel during fuel handling and storage.

d. Monitor and control the release of radioactive effluents to the environment.

e. Monitor and control the level of chloride contamination in the fuel pool.

Chloride contamination is not an immediate safety problem but it could contribute to a long term safety concern after power operation is resumed because of possible chloride stress corrosion effects on stainless steel components. However, careful control and limits on chloride contamination of fuel pool reactor coolant will help eliminate this long term concern.

An essential requirement for maintaining facility safety is the availability of adequate electric power. Therefore, the requirements for electric power is that the emergency power system shall retain delivery capability for the essential protection systems during all phases of the defueling-restoration operation following loss of all offsite power and failure of one diesel generator. The staff has reviewed the systems and inspected the facility to assure that this capability is available and will be provided by the Interim Technical Specifications.

The basic principle for assuring safety is that the minimum complement of systems and components be provided throughout the defueling-restoration operation as required to maintain BFNP Units 1 and 2 in a safe condition.
The Interim Technical Specifications require that this minimum redundancy be available at all times. However, it is expected that during most of the defueling-restoration operation the available safety systems will in fact exceed the minimum redundancy requirement, thus providing additional assurance for the health and safety of the public.

2.2 Reactivity Control

While fuel remains in the core it is essential to assure that control rods cannot be inadvertently withdrawn. To prevent this the reactor mode switch will be locked in the shutdown position, the directional control valves necessary for control rod withdrawal will be electrically disconnected, and the manual valve in the drive water supply to each hydraulic control unit will be closed. The Interim Technical Specifications include provisions to assure that this configuration is maintained thus making it virtually impossible to withdraw control rods. Also, before the reactor head is unbolted, full-in rod position indication and alarm must be operable in the control room. In this configuration the reactor core is subcritical and cannot be made critical by single failure or operator error.

In the highly unlikely event that a control rod drifts or becomes withdrawn, two channels of source range monitoring will be available to indicate an increase in count rate. The scram pilot valves and pneumatically controlled valves to scram each control rod will be operable. The control rod accumulators will be kept in a fully charged condition and a control rod drive pump will be in service. Manual scram is available to the operator and in addition, the source range monitors provide a non-coincidence input to the scram system. This system will detect approach to criticality and will alarm and cause automatic scram action to drive in any control rod that is not in the full-in position.

The standby liquid control system, which can inject borated water to shutdown and maintain the core in a subcritical state, will be available while fuel is in the reactor vessel.

Once the fuel has been removed from the core and placed in the storage racks in the fuel pool, the physical configuration imposed by the racks prevents the attainment of criticality. In this condition, protection against inadvertent criticality is totally independent of any electrically operated system or component.

We have reviewed the measures and systems for reactivity control and monitoring and conclude that they provide a high degree of assurance that the facility will be protected against the occurrence of criticality throughout the defueling-restoration operation.

2.3 Residual Heat Removal Shutdown Cooling (Fuel in the Core)

Adequate cooling of any fuel in the core, either prior to or during the defueling operation, is provided by operation of one Residual Heat Removal (RHR) pump and associated heat exchanger. The decay heat (based on the entire core) which must be removed to prevent fuel damage is very small amounting to less than one-fourth of the heat exchanger capacity of one RHR system. Nonetheless, both the Temporary Technical Specifications (issued May 9, 1975) and the Interim Technical Specifications being issued with this safety evaluation require that two RHR loops with one pump per loop be operational, and three RHR pumps and heat exchangers are presently available for each unit.

The RHR pumps take suction from a recirculation loop and pump water through the RHR heat exchangers in which reactor decay heat is transferred to a separate RHR service water system. In the event that the RHR pump providing flow should fail, the second RHR system for the unit is operational and can provide ample cooling.

In the event of the failure of the single valve in the suction line to the recirculation loop through which the RHR systems take suction, heat removal can be accomplished by the generation of steam in the core which is vented to the reactor building from the open reactor vessel (the head is off). Makeup coolant to the core in this mode of operation could be provided by either of the RHR pumps operating in the LPCI mode or the core spray pump taking suction from the torus. Operating in these modes the RHR systems do not rely on the single valve in the suction line to the recirculation loop.

In the cold shutdown head-off condition, with no pressure and low temperature in the primary system, there is no source to cause piping damage. Loss of coolant through inadvertent draining is minimized by locking valve positions so that water cannot be drained from the vessel and by providing double isolation valves in the operating cooling loop. However, even in the event of loss of coolant inventory, either of the RHR pumps or the core spray pump are capable of taking suction from the torus and providing water to the vessel.

With the reactor well flooded and the transfer canal gate open to the fuel storage pool the low decay heat level and the very large coolant inventory provide very ample time for manual switchover to backup equipment in the event of any operating cooling system failure. The time for such switchover was estimated in our May 9, 1975 Safety Evaluation Report to entail about 1 hour.

Analyses performed indicate that even if the refueling cavity were to accidentally drain, the reactor vessel volume is sufficient so that in excess of 28 hours would be available to restore coolant to the core before decay heat could cause sufficient boil off to reduce water level so as to uncover the top of fuel elements in the core.

We have concluded that sufficient decay heat removal and coolant makeup capability will be provided for the fuel in the core throughout the defueling operation, and that the fuel will thus be adequately protected from damage due to overheating.

2.4 Fuel Pool Cooling and Water Purification

Fuel cooling in the fuel storage pools for Units 1 and 2 is normally accomplished by the Fuel Pool Cooling and Purification System (FPPCPS). Each pool system consists of two fuel pool cooling pumps and heat exchangers arranged in parallel, two pool skimmer tanks, and the associated piping and valving. Each cooling pump and heat exchanger combination has sufficient capacity to remove the decay heat from the entire core.

The purification portion of the system is shared between Units 1 and 2 and consists of three precoat demineralizer trains arranged in parallel such that Unit 1 and Unit 2 use separate demineralizers with the third train maintained on standby to permit the backwashing and precoating of the exhausted demineralizer for either plant unit. Cooling is supplied by the two pumps and heat exchangers of the fuel pool cooling system.

When all fuel has been transferred to the fuel storage pool, one RHR system is available to function as a backup system in the event that all fuel pool cooling systems were lost, even though such loss could not be occasioned by any single event or failure. The Interim Technical Specifications require one RHR system to be operable as a backup for the fuel pool cooling systems, both of which are also required to be operable. One RHR pump and associated heat exchanger can provide ample cooling of the fuel pool, maintaining temperature below 125°F.

The licensee has upgraded the design of the spent fuel pool cooling system to Seismic Category I requirements. It was common for similar facilities licensed before Browns Ferry that the fuel pool cooling systems were not seismic Category I, although they were designed to prevent pool drainage in the event of failure (See Safety Guide 13). Nevertheless, at the time of the operating license review for Browns Ferry Unit 2, the licensee agreed to upgrade the fuel pool cooling system to seismic Category I. This involved installation of seismic Category I isolation valves and connections between the cooling systems and the purification system. The system is now seismic Category I and is designed to sustain normal loads, anticipated transients, the Operating Basis Earthquake, and the Safe Shutdown Earthquake within design limits which are consistent with those outlined in AEC Regulatory Guide 1.48, "Design Limits and Loading Conditions," for ASME Class 2 and 3 components. The specified design basis combinations of loading provide reasonable assurance that in the event (a) an earthquake should occur at the site, or (b) other upset, emergency or faulted plant transients should occur during normal plant operation, the resulting combined stresses imposed on the spent fuel pool cooling system components are not predicted to exceed the allowable design stress and strain limits for the materials of construction.

Limiting the stresses under such loading combinations provides a conservative basis for the design of the system components to withstand the most adverse combinations of loading events without loss of structural integrity. We conclude that the sytem design is acceptable and will perform as a seismic Category I system.

The licensee has analyzed various failure modes of the cooling system assuming a single initiating event followed by a concurrent single active failure. The failure modes include loss of offsite power, pump failures, valves failing in the unsafe condition and pipe cracks in the systems. For these failure modes, pool cooling will be available from at least one system. In addition, for highly degraded conditions where all pool cooling was assumed lost, pool boiling would occur and makeup to the pool would be available from three seismic Category I makeup systems; the standby coolant system connected to the pool through the RHR piping, and the two separated Emergency Equipment Cooling Water System (EECWS) headers each provided with a hose hydrant and hose to reach Unit 1 or Unit 2 pools.

We have reviewed the results of these analyses and agree with the licensee that for all postulated failure modes the stored fuel will be maintained at acceptable temperature levels and would not result in fuel damage.

Maintenance of the reactor coolant and the fuel storage pool water quality will be accomplished by the Reactor Water Cleanup System (when the fuel is in the reactor vessel) and by the fuel pool purification system when the reactor well is flooded and after the entire core is in storage in the fuel pool. The licensee has analyzed the use of the purification system to maintain the Technical Specification limits and the capability of the radwaste system to process and store the effluents from the purification system. The results indicate that on the basis of one backwash and precoating operation a day per unit the radwaste system has excess capacity.

We have reviewed the radwaste capability using a conservative assumption of two backwash and precoating operations per unit per day. Under these conditions, the liquid waste water can be stored, processed and released to the environment at levels below those permitted for normal power operation of the plant. The resin sludge storage capacity is sufficient to store 3900 cubic feet of sludge. In addition, storage liners are on hand that can store 6200 cubic feet of dewatered sludge. Assuming no offsite shipment of resin sludge, adequate capacity exists for operation of the purification systems for approximately 6 months. This finding is based on no additional large discharges of radwaste to the system. Based on the above, we conclude that the fuel pool purification system, in combination with the radwaste system, can maintain the cooling water quality and not cause an increase in releases of radioactivity offsite above that already permitted for normal reactor operation.

2.5 Cooling Water Systems

The cooling water systems provided for Units 1 and 2 consist of the Residual Heat Removal Service Water System (RHRSWS), the Emergency Equipment Cooling Water System (EECWS), Raw Cooling Water System (RCWS) and the Reactor Building Closed Cooling Water System (RBCCWS). Each of these systems serves the function of removing heat from the reactor plant essential systems and transferring it to the river (ultimate heat sink).

Since the fire, these systems have been made operational and have been valved into an arrangement that will assure cooling water to the RHR system, the spent fuel pool cooling system, the diesel generator cooling system, the standby coolant system and the manual fuel pool makeup water systems. Additional essential heat loads can be manually valved into the cooling water systems as required depending on the failure mode that occurs. For example, the EECWS can be manually valved to provide cooling water to the control room air conditioning coolers and emergency cooling units should the need arise.

These systems are redundant and separate and supplied with onsite power to assure a continued supply of cooling water for the essential systems. The Interim Technical Specifications require a minimum number of pumps that must be connected to the separate headers. The flow capacity of any one pump on any header is more than adequate to provide the cooling for the assigned essential systems.

The licensee has performed analyses for all potential failure modes and has demonstrated that a minimum of 8 hours are available to perform any manual operations to cool the fuel in the reactor vessel prior to boiling. When the fuel is in the storage pool and the gates are installed, a minimum of 30 hours are available for manual operations prior to pool boiling. The analysis also demonstrates that with one fuel pool pump and heat exchanger operating, the pool temperature can be maintained well below the boiling temperature. Two fuel pool cooling pumps or one RHR loop is required by the Interim Technical Specifications.

For the highly degraded case where all pool cooling is lost and the gates are in place, analyses demonstrate that, assuming no makeup, a minimum of 160 hours is available before the minimum pool level reaches the Technical Specification limit. This provides for more than adequate time for manual operations to provide makeup water to the pool from the EECW system or to restore pool cooling.

For the case where the refueling bellows seal were to fail during the fuel unloading from the reactor vessel, the analysis demonstrates that a minimum of 10 hours are available prior to boiling to provide makeup water to the fuel pool and assure cooling of the fuel in the reactor vessel.

We have evaluated the licensees analyses and the capability for cooling water systems to be manually aligned to the appropriate header to supply cooling water to the essential systems and to provide makeup water to the fuel pool or the reactor vessel in the event of the above failures, and have concluded that these systems are fully capable of supplying the vital cooling and makeup water needs of the facility throughout the period of defueling - restoration operations.

2.6 Electrical Power

The electrical energy to operate the systems required to maintain the units in a safe condition is provided by redundant offsite and onsite power supplies. The offsite power is supplied by four 161 kV circuits connected from the power grid to redundant shutdown buses which in turn supply power to all four 4.16 kV emergency buses. Three of the four plant diesels are capable of supplying emergency power on loss of offsite power. During fuel removal operations, the three diesels assure that emergency power is available for sufficient cooling equipment to maintain a safe condition upon loss of offsite power plus the application of an additional single failure. During fuel storage and fire damage removal operations, only two diesels are required to assure sufficient cooling with the loss of offsite power and an additional single failure.

The diesels are cooled through the emergency equipment cooling water system whose pumps automatically start on diesel start to ensure sufficient coolant to the diesels to protect against overheating.

Redundant 250V batteries and their associated distribution systems are available to provide d-c control power as necessary to the required systems.

We have concluded that adequate power supplies are available to the systems and equipment required to maintain the facility in a safe condition, and that the provisions of the Interim Technical Specifications assure that this capability will be maintained throughout the defueling, fire damage removal and restoration operations.

2.7 Cable Cutting andDamage Removal

The safety of the facility during cable cutting and damage removal operations will be maintained by adherence to the following criteria and procedures.

The minimum systems required to maintain the facility in a safe condition have been identified. Prior to the start of any cable cutting or damage removal operations, the electrical cables serving these systems will be verified to be independent of the fire damage zone by careful check of the appropriate electrical,schematics, installation drawings, and cable schedules. In some cases, this required replacement of fire damaged cables. Other measures being taken to preclude spurious operation of critical components include physically placing of some components in the required safe position, and the removal of power by opening breakers, pulling fuses and disconnecting wires. All disconnected wires will be taped to cover exposed conductors.

Also, prior to start of any cable cutting or fire damage removal, all fire affected cables will have been identified by careful check of the appropriate electrical schematics, and installation drawings and cable schedules. These cables will be disconnected from their terminations on both sides of the fire affected zone. Before a cable is cut, its disconnected ends will be checked for potential; if potential exists, no cutting will proceed until that cable is rechecked and cleared. Likewise, if potential is found at a cut end, the cable must be identified and cleared before further cutting.

Furthermore, if any of the critical structures, systems or components which are required to maintain the facility in a safe condition becomes inoperable, during cable cutting or damage removal, these operations will be suspended until the condition is rectified.

We have concluded that the measures and procedures governing cable cutting and fire damage removal operations described in Part VI of the Plan and the provisions of the Interim Technical Specifications provide adequate assurance that these operations can be performed without undue risk to the health and safety of the public.

2.8 Fuel Handling Equipment

The fuel handling equipment to be used for preparing for fuel removal and for transfer of the fuel from the reactor vessel to the fuel pool consists of the Reactor Building Crane (125 ton hoist and an auxiliary 5 ton hoist), the fuel handling bridge, and the normal complement of fuel handling tools, accessories and special tools that are required for normal refueling operations.

The reactor building crane will be used to remove the shield plugs, drywell head, reactor vessel head and the upper core internal to prepare for unloading the fuel from the reactor vessel. After the fuel has been transferred to the fuel pool, the crane will be used to secure the water tight integrity of the fuel pool, to replace the shield plugs in the dryer separator storage area and perform any required functions over the reactor well and empty reactor vessel.

The reactor building crane 125 ton hoist has been completely pre-operationally tested since the fire, including a 125% load test for the maximum expected load to be lifted. The 5 ton auxiliary hoist will also receive a similar test before use.

When the crane is not being used for the above described purposes, it will be placed in the area over the Unit 3 refueling zone and prevented from moving into the Unit 1 and 2 area by two functionally redundant and physically separated limit switches. The trolley movement will be similarly restricted from moving over the fuel pool (by rail stops installed on the trolley rails) except for the times when the 5 ton auxiliary hoist is required to be used in Unit 1 and 2 for handling small loads in the fuel pool to accomplish required fuel handling operations. This exception also applies to performance of in-vessel inspection or work after the fuel is in storage. These exceptions will be under strict administrative control with signed approval required.

With these restrictions, limit devices and administrative procedures, we find there is reasonable assurance that heavy loads will not be carried over the fuel pool and thereby preclude damage to the fuel structural integrity or the stored fuel assemblies.

The fuel handling bridge, which removes the fuel from the reactor vessel and places it in the storage racks in the fuel pool, will be demonstrated to be operable prior to use. Testing to be performed will include the bridge drives, brakes, positioning drives, the mast and the fuel grapple. In addition, the lifting devices, including the 1/2 ton hoist, on the bridge will be load tested at 125% of their maximum expected load.

The only exception to the normal refueling safety features is the absence of the refueling interlocks that are associated with control rod drives and reactivity control. These interlocks will not be required for the current operation since the control rods are prevented from being moved by the deactivation of the drive system and by the operating mode switch being in the shutdown position which prohibits control rod withdrawal.

Based on our review, we conclude that the fuel handling and transfer equipment is capable of carrying out its function safely. Any malfunction of the equipment that could result in the dropping of a fuel assembly and cause a release of radioactivity has been analyzed in Section 2.10 and shown not to endanger the health and safety of the public.

2.9 Reactor Building and Control Bay Ventilation

Due to the effects of the fire, the reactor building ventilation system and the standby gas treatment system cannot be operated in their normal modes as required for normal power operation or accident conditions. As a result, the reactor building ventilation system is being manually operated to ventilate the refueling zone and other reactor building zones. One standby gas treatment system train will be verified to be free of the fire area and will be capable of being operated by onsite power. The system will be actuated manually at the standby gas treatment building.

With this arrangement, the reactor buidling will be ventilated such that the refueling zone will be at a positive pressure with respect to the fire affected area to control any chloride contamination of the refueling zone during the cable removal operations.

Since the reactor building ventilation system is not powered by onsite power it will not be available during a loss of offsite power condition. For this condition, the refueling zone will be maintained at essentially atmospheric pressure through dampers that have been secured in the open position; and the standby gas treatment train will be operated to provide a slightly negative pressure in the reactor building zones of Units 1 and 2.

The cooling of the RBCCW pumps and the spent fuel pool cooling pumps will normally be performed by the reactor building ventilation system. For the loss of offsite power condition, these pumps have been analyzed to be capable of continued operation for at least 24 hours assuming no heat loss from the building and no standby gas treatment system flow. In the event of an extended offsite power loss, these pumps would approach their design operating temperature limit and could be lost. In this case, the standby RHR fuel pool cooling system would maintain the pool cooling function since the RHR pumps have individual room coolers that are operable from onsite power and cooled by the cooling water systems previously evaluated in Section 2.5 of this report.

For the highly degraded condition where fuel pool boiling would occur, the vapor from the pool surface would be exhausted to the atmosphere through the dampers that have been secured in the open position, and through the ventilation tower doors to the roof of the reactor building. Any condensation on the inside of the reactor building siding will be collected by the refueling zone floor drain system.

We have reviewed the licensee's method for ventilation of the reactor building during normal and abnormal conditions. Based on our conclusions set forth in Section 2.10 of this report that secondary containment and standby gas treatment are not required to protect the health and safety of the public, we conclude that the proposed method of operation is acceptable.

The control bay ventilation and air conditioning systems are operable from onsite power and will be available to maintain the habitability of the control bay and control the temperature of the essential areas, including the shutdown board rooms. The cooling function will be normally provided by the RCWS. However, as discussed in Section 2.5 Cooling Water Systems, the EECW system can be manually aligned to provide the cooling function if the RCWS is not available.

Based on our review, we conclude that adequate control bay ventilation and cooling can be provided for all postulated failure modes and that the proposed systems configurations are acceptable.

2.10 Accident Analysis

The principal accident event of concern in connection with the defueling and fuel storage operation is the potential for a fuel handling accident.

With low pressure and temperature and seismically qualified equipment, there is no potential for loss of coolant accidents. The analyses discussed in Sections 2.3, 2.4 and 2.5 above, demonstrate that the fuel will be amply protected from cladding damage due to lack of cooling.

The potential release of radioactivity resulting from pool boiling in the event of temporary loss of cooling has been considered. Even if all gap activity in the stored core were somehow released under this condition (an extremely unlikely event) the resultant offsite doses would be small fractions of the 10 CFR Part 100 exposure guidelines.

For the fuel handling accident, the licensee has analyzed the radiological effect of a postulated fuel handling accident (Appendix E of Part VI of the Plan). We have evaluated the licensee's analysis and find it acceptable. However, we also performed an indepencent analysis and conclude that fuel handling at the Browns Ferry plant may be conducted without secondary containment and the Standby Gas Treatment System filters in operation without danger to the public health and safety. Although the licensee performed his analysis assuming only 46 days decay, the actual decay time before planned fuel transfer would be nearer 90 days. The analysis performed at 46 days results in a thyroid dose estimate which would be reduced by a factor of 16 at 88 days decay (32 days later) to a two-hour thyroic dose at the exclusion area boundary of less than 0.1 rem. These values are not only far below the siting guidelines of 10 CFR Part 100; they are, in fact, below the standards for radiation protection from a year's exposure to normal operational releases set forth in 10 CFR Part 20, for the general public. The whole body exposure is less than 0.1 rem at 46 days decay and not significantly less at 88 days. Table 2.10 gives the assumptions and results of the NRC staff analysis that support the conclusion that secondary containment and the standby gas treatment system are not required during fuel handling operations.

2.11 Process and Area Radiation Monitoring

Process and radiation monitoring equipment will be in service to monitor the following release points: (1) reactor turbine ventilation exhaust, (2) refueling zone exhaust, (3) radwaste ventilation exhaust, (4) turbine building roof ventilation, (5) liquid radwaste effluent, (6) main stack and (7) reactor zone exhaust.

In addition, area radiation monitors will be operable in both Units 1

TABLE 2.10 - ASSUMPTIONS AND CONSEQUENCES FOR A FUEL HANDLING ACCIDENT AT BROWNS FERRY

A. Assumptions

Power Level at Shutdown	3300 MWt
Operating Time	1000 days
Decay time from 3/22/75 to 6/17/75)	88 days
X/Q	3.4×10^{-4} sec/m^3
Breathing rate	3.47×10^{-4} m^3/sec
Number of rods in core	37,346
Number of rods breached	111
Iodine DF in water	0.01
Gap Activity	30% Kr-85 10% Other Noble gases and iodines
Power Peaking Factor	1.5
Puff release of all gap activity in breached rods assumed.	
Isotopes considered.	I-131, Kr-85, Xe-133

B. Consequences

Exclusion Area Boundary

Thyroid dose, rem	0.04
Whole Body dose, rem	0.04

and 2 reactor building and refueling floor areas for plant personnel safety.

This monitoring equipment provides assurance that plant releases are properly monitored and plant personnel are promptly warned of hazardous situations.

2.12 Reactor Coolant and Fuel Pool Cooling Water Quality

The licensee has described the precautions that will be employed to preclude the introduction of chloride contamination into the fuel pool cooling water and the reactor coolant. If chloride should be inadvertently introduced into the coolant, the licensee has shown that there are adequate means to detect and reduce the chloride to acceptable levels. The Interim Technical Specification limits on conductivity and chloride concentration and surveillance requirements provide reasonable assurance against conditions that could lead to significant stress corrosion of materials and loss of structural integrity of components.

2.13 Minimum Shift Crew Considerations

As a result of the damage created by the fire to various Unit 1 and Unit 2 cables, a number of safety systems cannot be operated from the control room. Equipment monitoring and operation is now required from various positions located through the facility. Analyses have shown that in excess of 28 hours are available before the fuel is uncovered to perform various tasks involved in transferring to the various functioning backup systems. Manual operation of equipment for which the most degradation is experienced could be achieved in a time interval of several hours, and the licensee has indicated that these operations could easily be accomplished by a shift crew of 10 men.

We have required a shift crew of 14 men for the present cold shutdown condition, and one additional man per unit during defueling operations, i.e., 15 men if one unit is being defueled and 16 men if both units are being defueled. The additional men will perform defueling operations, and since defueling will be suspended in the event a manual transfer to backup safety systems is necessary, will provide margin for performance of the required manual operations and monitoring.

After defueling is completed with all the fuel in the storage pool, the time available to perform manual transfer to backup systems is extended even further. Therefore, a shift complement of 10 men is then adequate to provide the same level of safety.

These personnel will not be involved in any activities other than those required to maintain the facility in a safe condition. We have concluded that these staffing requirements provide a satisfactory level of safety with sufficient margin for anticipated contingincies and are acceptable.

2.14 Quality Assurance

As a result of our review and evaluation of the licensee's QA plan (Part XIII, Section A, dated May 28, 1975) for the restoration of BFNP Units 1 and 2, and as amended by letter dated June 11, 1975 from J. E. Gilleland to B. C. Rusche, we find it acceptable for the control of activities associated with the defueling, fuel storage, and cleanup and removal of fire-affected equipment of BFNP Units 1 and 2. The QA plan, as amended, commits to the QA program requirements as described in TVA's Sequoyah Nuclear Plant FSAR (through Amendment 34), Docket No. 50-327, meets the requirements of Appendix B to 10 CFR Part 50, and meets, with some minor and acceptable exceptions, the requirements of WASH documents 1283, 1309 and 1284.

2.15 Unit 3 Construction Activities

To assure that Unit 3 activities do not adversely affect Unit 1 and 2, the technical specifications do not permit operations, tests, and other activities associated with the construction of Unit 3 that could affect the functioning of any safety system or component required by the Interim Technical Specifications until interface analyses are performed.

3.0 SUMMARY

"The Temporary Technical Specifications and Bases for Browns Ferry Nuclear Plant Unit 1 and 2, Change No. 10 to the Technical Specifications," issued May 9, 1975, included the appropriate specifications to provide assurance that the plant equipment and systems in their present status and configurations are adequate to provide the plant safety. The present status is that the reactors are in cold shutdown with the fuel in the reactor vessels.

In order to isolate the fuel to the maximum extent from possible adverse effects caused by activities associated with removal of fire damaged equipment or restoration work, the Unit 1 and Unit 2 fuel will be removed from the reactor vessels and stored in their respective fuel storage pools.

We have carefully reviewed the safety of the transfer of fuel to the storage pool and its continued storage there until completion of restoration work. The equipment and controls needed to perform the transfer of fuel (defueling) to the storage pool under the present conditions have not been affected by the fire (Ref. Sec. 2.8). The safety of the fuel is ensured by maintaining adequate cooling, with back-up cooling systems required to be operable, for fuel in the reactor vessel and in the storage pool during defueling-restoration operations (Ref. Sec. 2.3 and 2.4).

Maintaining the fuel in a shutdown condition is ensured in the reactor vessel and in the storage pool (Ref. Sec. 2.2). Redundant and diverse sources of make-up water are available in the event of loss of cooling or loss of coolant both in the reactor vessel and in the fuel storage pool (Ref. Sec. 2.3, 2.4, and 2.5).

Storing the fuel in the fuel storage pools provides a very large margin of safety. The present levels of decay heat and fission product inventory in the cores after approximately 90 days of decay are very low. The large coolant inventory provided by the fuel storage pools increases the time available to respond to the most extreme adverse events including the worst case of rupture of the refueling bellows seal that drains the coolant to within five inches of the top of the fuel and results in loss of the fuel pool cooling system (Ref. Sec. 2.5). Even in this case, at least 10 hours are available to take corrective action prior to the fuel storage pool water reaching the boiling temperature. This worst case is precluded after completion of defueling by the installation of the gate between the fuel pool and the reactor well. This is required by the Interim Technical Specifications.

The two pool cooling systems and the independent RHR back-up are both capable of providing ample cooling. These are also backed up by three independent make-up water systems to supply water to the pool to prevent the fuel from uncovering. Even if all cooling were lost it would take over seven days for the water level to drop to the Technical Specification limit, 8 1/2 feet above the fuel, if no make-up water is assumed.

The activities and controls for removal of fire-damaged equipment have been reviewed and found to minimize any adverse effects on safety of the facility (Ref. Sec. 2.7). This is ensured by strict procedures, Quality Assurance requirements, and the isolation of the fuel in the storage pools from the restoration activities.

Therefore, Change No. 11 to the Technical Specifications includes the requirements for equipment and administrative controls needed in addition to those contained in the Temporary Technical Specifications, Change No 10, in order to perform the defueling and maintain the fuel in the storage pools with the protective systems and administrative controls described in this evaluation. The resulting Interim Technical Specifications continue to provide adequate protection to the health and safety of the public and, by accounting for the actual existing condition of the plant systems and components, they minimize any potentially unsafe conditions or actions.

The probability and consequences of any accident involving release of radioactivity from the site are very low, based on the following considerations: (1) The plant is locked into a shutdown condition prior to and during fuel removal from the reactor vessel; (2) The fuel will be stored in a non-critical geometry in the spent fuel pools that provide a large, inviolable   of coolant with multiple sources of make-up; (3) The length of decay time since operation results in a very low amount of decay heat generation from the irradiated fuel assemblies, as well as a low inventory of fission product activity for possible release; and (4) Redundant cooling systems and multiple makeup sources are available and a very long time period is available to restore cooling in the unlikely event of total loss of cooling systems. In view of the foregoing, we conclude that the issuance of these amendments does not involve a significant hazards consideration.

4.0 CONCLUSION

The changes in technical specifications authorized in connection with this evaluation result in enhancement of safety under present conditions, as discussed above. Based on these considerations, we have concluded that: (1) because the change does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the change does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of these amendments will not be inimical to the common defense and security or to the health and safety of the public.

Date: June 13, 1975

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NOS. 50-259 AND 50-260

TENNESSEE VALLEY AUTHORITY

NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES

Notice is hereby given that the U.S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 10 to Facility Operating License No. DPR-33 and Amendment No. 7 to Facility Operating License No. DPR-52 issued to Tennessee Valley Authority which revised Technical Specifications for operation of the Browns Ferry Nuclear Plant, Units 1 and 2, located in Limestone County, Alabama. The amendments are effective as of their date of issuance.

The amendments revise the Technical Specifications, taking into account the present condition of plant systems, so as to assure that the two units will remain in a safe and stable posture during the period of defueling of both units and storage of the fuel in their respective fuel pools. The plant will be maintained in this condition until completion of the repairs of damage resulting from the fire which occurred on March 22, 1975. These amendments also authorize the removal of fire-affected equipment. Approval of restoration of the facility will be the subject of a separate action.

The application for these amendments complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendments. Prior public notice of these amendments is not required since the amendments do not involve a significant hazards consideration.

For further details with respect to this action, see (1) the application for amendment dated June 2, 1975, (2) Amendment No. 10 to License No. DPR-33 and Amendment No. 7 to License No. DPR-52 with Change No. 11, (3) Part VI Section E of "Plan for Evaluation, Repair, and Return to Service of Browns Ferry Units 1 and 2", Revision 7 dated May 28, 1975, and Revision 10 dated June 5, 1975, and (4) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, NW., Washington, D.C. and at the Athens Public Library, South and Forrest, Athens, Alabama 35611.

A copy of items (2), (3), and (4) may be obtained upon request addressed to the U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Reactor Licensing.

Dated at Bethesda, Maryland, this 13th day of June 1975.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert A. Purple, Chief
Operating Reactors Branch #1
Division of Reactor Licensing

ATTACHMENT IV

LETTER FROM THE NRC TO THE TVA DATED SEPTEMBER 2, 1975, WITH ENCLOSURES

AMENDMENT NO. 14 TO DPR-33

AMENDMENT NO. 11 TO DPR-52

SAFETY EVALUATION

FEDERAL REGISTER NOTICE

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

September 2, 1975

Docket Nos. 50-259
and 50-260

Tennessee Valley Authority
ATTN: Mr. James E. Watson
Manager of Power
818 Power Building
Chattanooga, Tennessee 37201

Gentlemen:

Your letter dated August 29, 1975, requested approval of certain restoration activities for Browns Ferry Units 1 and 2 and revised the Final Safety Analysis Report for Units 1, 2, and 3. The restoration activities for Browns Ferry Units 1 and 2 and the details of the fire protection design changes for Unit 3 are described in "Plan for Evaluation, Repair, and Return to Service of Browns Ferry Units 1 and 2 (March 22, 1975 Fire)" dated April 13, 1975, and revisions thereto up to and including Revision 20 (the Plan). In addition, your letter dated August 29, 1975, listed TVA's commitments for additional work not covered by the Plan.

We have reviewed the restoration work and design modifications proposed in the Plan and have considered the commitments made by TVA. We have concluded that TVA may proceed with the restoration and design modifications as proposed in the Plan. We find that implementation of these items are necessary and will not preclude any further modifications resulting from our continuing review, including resolution of the design details for incorporating the commitments made by TVA. Appropriate Amendments and Safety Evaluation are enclosed.

Sincerely,

Robert A. Purple, Chief
Operating Reactors Branch #1
Division of Reactor Licensing

Enclosures:
1. Amendment No. 14 to DPR-33
2. Amendment No. 11 to DPR-52
3. Safety Evaluation
4. Federal Register Notice

ccs: See next page

cc w/enclosures:
Robert H. Marquis
General Counsel
629 New Sprankle Building
Knoxville, Tennessee 37919

Athens Public Library
South and Forrest
Athens, Alabama 35611

Mr. Thomas Lee Hammons
Chairman, Limestone County Board
of Revenue
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-259

BROWNS FERRY NUCLEAR PLANT UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 14
License No. DPR-33

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Tennessee Valley Authority (the licensee) dated August 29, 1975, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations; and

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

2. Accordingly, add Paragraph 2.C(4) to Facility License No. DPR-33 to read as follows:

   "(4) The facility may be modified as described in Section X of "Plan for Evaluation, Repair, and Return to Service of Browns Ferry Units 1 and 2 (March 22, 1975 Fire)" dated April 13, 1975, and revisions thereto up to and including Revision 20."

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert A. Purple

Robert A. Purple, Chief'
Operating Reactors Branch #1
Division of Reactor Licensing

Date of Issuance: September 2, 1975

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

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-260

BROWNS FERRY NUCLEAR PLANT UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 11
License No. DPR-52

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Tennessee Valley Authority (the licensee) dated August 29, 1975, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations; and

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

2. Accordingly, add Paragraph 2.C(5) to Facility License No. DPR-52 to read as follows:

   "(5) The facility may be modified as described in Section X of "Plan for Evaluation, Repair, and Return to Service of Browns Ferry Units 1 and 2 (March 22, 1975 Fire)" dated April 13, 1975, and revisions thereto up to and including Revision 20."

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert A. Purple, Chief
Operating Reactors Branch #1
Division of Reactor Licensing

Date of Issuance: September 2, 1975

SAFETY EVALUATION BY THE DIVISION OF REACTOR LICENSING SUPPORTING VARIOUS RESTORATION ACTIVITIES FOR THE BROWNS FERRY NUCLEAR POWER PLANT UNITS 1, 2 AND SUPPLEMENT 7 TO THE SAFETY EVALUATION BY THE DIVISION OF REACTOR LICENSING U.S. NUCLEAR REGULATORY COMMISSION IN THE MATTER OF TENNESSEE VALLEY AUTHORITY, BROWNS FERRY NUCLEAR PLANT UNITS 1, 2, AND 3 WITH REGARD TO DESIGN CHANGES

1.0 Introduction

On March 22, 1975, a fire at the Browns Ferry Nuclear Plant required a shut down of Units 1 and 2. The facility subsequent to the shutdown was found to have incurred substantial damage to power, control, and instrumentation wiring. Unit 3 of the Browns Ferry facility did not suffer any damage as a direct result of the fire. All three units are presently in the shutdown condition with the fuel removed from the vessels for Units 1 and 2; the Unit 3 reactor is still under construction with operation for that unit scheduled for early 1976. An overall program has been developed by the licensee delineating the necessary activities required to restore damaged portions of the facility to a level so that operation of Units 1 and 2 can be resumed.[1]

As a result of the fire an NRC review plan was developed consisting of three major and parallel elements. The first element was the investigation conducted by the Office of Inspection and Enforcement of events leading to the fire, fire fighting efforts, sequence of operational events and problems experienced with the nuclear steam supply system, interaction between units, and the response of TVA, State and Local authorities. This phase of the NRC plan has been completed.[2]

The second element of the plan, being performed by the Office of Nuclear Reactor Regulation has as its objectives (1) to assure that a safe plant configuration was attained and is being maintained subsequent to the fire, (2) to assure safety during removal of fuel from Units 1 and 2, (3) to assure plant safety during fire damage removal and restoration, and (4) to determine that the design changes that are required for restoration of these plants to operational status are acceptable. Thus far there have been two licensing actions taken to assure that the plant has been placed in a safe configuration following the fire. First, the plant Technical Specifications were changed to Temporary Technical Specifications designed to assure that a safe configuration

[1] "Plan for Evaluation, Repair and Return to Service of Browns Ferry Units 1 and 2 (March 22, 1975, Fire)" and revisions thereafter.

[2] OI&E Investigation Report of the March 22, 1975 Fire at the Browns Nuclear Station, dated July 25, 1975.

was maintained following the fire, while further corrective action plans were under development.[3] Subsequently, the Technical Specifications were again changed to provide for the removal of fuel from both Unit 1 and 2 reactor vessels and placement into the respective storage pools.[4] These changes also assure that the necessary minimum protective equipment to assure protection of the fuel in the storage pools from damage would be available with adequate redundancy and would be protected against adverse interaction with the fire affected wiring and cables, while such fire damage was removed and when the damaged wiring and cabling was replaced.

Thus far the restoration work performed at the facility to date has been limited to various clean-up operations and removal of damaged electrical wiring and cable trays. Thus, the fourth objective of the NRR element of the NRC program remains to be accomplished. This safety evaluation considers a portion of that objective, namely, the approval of certain proposed design features needed to prepare the facility for return to power operation. Consideration of return to operation including a determination of whether additional design features are required and an assessment of the operating procedures to assure safe operation will be discussed in a subsequent safety evaluation, prior to authorizing return to power operation.

The third element of the NRC program consists of the review being performed by the Special Browns Ferry Review Group established on March 26, 1975.[5] Efforts of this review group will be to prepare for NRC recommendations to change, as required, NRC policies, procedures, and technical requirements.

By letters dated August 25 and 29, 1975 TVA requested the approval of various restoration activities required to restore the plant to operational status as a result of the fire. The approval requested by TVA in the August 25

3
See Safety Evaluation issued in connection with Amendment 9 to license DPR-33 (Unit 1) and Amendment 6 to license DPR 1-52 (Unit 2) issued May 9, 1975.

4
See Safety Evaluation issued in connection with <u>Amendment 10 to license DPR-33 (Unit 1) and Amendment 7 to license DPR-52 (Unit 2)</u> issued.

5
Memorandum L Gossick to all NRC employees, "Appointment of Special Review Group," dated March 26, 1975.

letter consists of the following major items:

"1. Approval to proceed with structural work which covers restoration of concrete, embedments, pipe supports, cable tray supports and the replacement of pipe and ducting in the fire-affected area. In general, the plans related to this portion of the restoration effort are described in Part VIII of the Recovery Plan and in the response to Question 1.0 under the "General" heading.

2. Approval of electrical design changes including circuit changes and permission to proceed with the restoration and installation of cable trays, conduit, and cables, including approval for splicing. The work and design changes are described in Parts VI and X of the Restoration Plan.

3. Approval for installation of the fire detection system and for the installation of distribution piping of the fixed spray system in all areas of reactor building except in the fire-affected area. The installation in Unit 2 and Unit 3, and in the area outside the fire zone is scheduled to proceed simultaneously with structural work in the fire zone of Unit 1. These systems are described in subsection 5.0 of Part X of the Recovery Plan."

The replacement activities described in Item 1 above were authorized by letter dated August 28, 1975. The TVA letter dated August 29, 1975 modified Item 3 to some extent discussed below. Our evaluation of the overall restoration of the facility and improvement of facility design from the standpoint of fire protection is still under review. In its August 29, 1975 letter, the license agreed to provide further improvements in fire protection, over and above the improvements reflected in the changes authorized herein. These improvements are listed as commitments in the August 29, 1975 letter. The staff is still considering whether such further improvements are necessary prior to authorization of return to operation or whether such further improvements may be incorporated at a later shutdown period in the future. Although further improvements may be required in the future, including additional changes that may be required before return to operation is permitted, the design modifications authorized herein provide substantial enhancement of the capability of the facility to withstand the effects of a fire.

2.0 Description of Damage Resulting from the Fire

Substantial detail concerning the cause of the fire and the extent of damage to structures, systems and components is provided in the previously noted OI&E report. The following brief summary is provided as background for the discussion of the repair and replacement activities described in this Safety Evaluation.

After ignition, the fire propagated through the penetration in the wall between the cable spreading room and the Unit 1 reactor building. As a result of the pressure differential which is maintained beween the reactor building and the cable spreading room, and because an installed carbon dioxide (CO_2) fire extinguishing system was used in the cable spreading room, only a small amount of burning occurred in the cable spreading room. Damage to the cables in this area was limited to about 2 feet immediately adjacent to the penetration where the fire started. The major damage occurred in the reactor building outside the cable spreading room, in an area roughly 40 feet by 20 feet, where a very high concentration of electrical cables exists. There was very little other equipment in the fire affected area, with the only direct damage other than cables being the melting of a soldered joint on an air line.

Electrical System Damage

The electrical cables, after insulation had been burned off, shorted together, and grounded to their supporting trays or to the conduits, resulting in the loss of control power for much of the installed equipment such as valves, pumps, blowers and the like.

The most significant aspect of the fire damage was that it resulted in the loss of redundant safety equipment.

Soot and Chloride Damage

In addition to the cable damage, the burning insulation created a dense soot which was deposited throughout the Unit 1 reactor building and in some small areas in the Unit 2 reactor building. An estimated 4,000 lbs of polyvinyl chloride insulated cable which burned released an estimated 1400 lbs of chloride to the reactor building. Chlorides are very corrosive to most materials. Components subject to stress corrosion damage from chloride contamination are limited to those constructed of stainless steel and high strength alloy steel such as equipment hanger springs. The licensee has provided a program of cleaning, cleanliness verification, evaluation, and surveillance to mitigate any harmful consequences of the exposure to chlorides.

TVA's cleaning program consisted of dry cleaning and wet detergent cleaning using demineralized water and a chloride-free industrial detergent solution. Cleaning procedures were repeated until measured chloride levels were below 0.38 mg per dm^2. This level of chloride contamination is recognized to be an acceptable level for chloride in contact with stainless steel and is incorporated as the acceptance level in RDT Standard E5-1T. After cleanliness was verified, precautions were taken to prevent recontamination. All sources of contamination were removed and further surveys were performed to verify acceptable cleanliness levels.

In addition to cleaning efforts, the licensee will undertake an extensive program, consisting of liquid penetrant examination, metallurgical sampling and surface replica examination. During the penetrant work, the stainless steel piping and piping components will be tested by liquid penetrant in all of the principle piping areas outside of primary containment in which corrosion damage could have occurred.

Metallurgical samples will be taken from stainless steel piping, stainless components and high strength alloy steel components. The samples will be removed in the fire zone and other areas of the Unit 1 reactor building where heavy residue from the fire was deposited. In addition, surface replicas will be taken from Unit 1 piping. Surface replication is a very sensitive technique with respect to the detection of cracking due to chloride stress corrosion and was an addition to the original metallurgical evaluation plan.

The surveillance program will be worked out in detail between TVA and NRC and will be carried out before return to operation is authorized and thereafter during plant operation, and will be addressed in our subsequent safety evaluation before return to operation is authorized. However, the initial proposed program consists of quarterly visual examinations of all stainless steel components and piping for signs of leakage, cracking or distress.

We conclude that based on our evaluation the cleaning procedures were effective in reducing chloride levels to or below the recognized acceptance standards. Testing to be carried out during the restoration period along with the subsequent surveillance program which will be developed before the plant returns to operation will assure that any deleterious corrosion will be detected and corrected in a timely manner.

3.0 Overall Approach Proposed by TVA

The Browns Ferry Nuclear Facility consists of three boiling water nuclear reactors each of which are designed to produce 1067 Mw of electric power. Units 1 and 2 of that facility were authorized for operation in June 1973 and June 1974 respectively and were in operation up until the March 22, 1975, fire. Unit 3 is still under construction and is expected to go into service early in 1976.

Units 1 and 2 share a common control room with a shared cable spreading room located beneath the control room. Wiring carrying signals between the control room and various pieces of equipment in the plant are routed into the cable trays in the cable spreading room. The fire caused extensive damage to wiring located immediately outside of the cable spreading room.

Some sharing of equipment exists between all three units; the sharing of electrical systems is most extensive in Units 1 and 2. The electrical design concept employed in the design of the Units 1 and 2 is based on using a two-division concept. The purpose of dividing the electrical system into two divisions was to assure that the facility could be maintained in a safe configuration even with the postulated loss of one entire division.

Consideration of a fire as a design basis event, requires that the capability be maintained during and following the fire to safely shutdown the reactor and remove the decay heat. Previous analyses (PSAR and FSAR) have shown that only one division of electrical equipment is required to assure that the facility is maintained in a safe condition. The control rod system in a BWR is designed to be fail safe. That is, a loss of electrical power results in initiating a scram of the control rods and shutdown of the reactor.* Therefore, the major objective for protection of the reactor from the effects of fire is to assure that necessary equipment to remove decay heat remains operable in the event of a fire.

Thus, it is important to limit fire induced failures to one division of the safety related equipment for shutdown heat removal from each reactor which provides, for each reactor at least one core spray system with two associated pumps and valves, and one RHR system with associated pumps, heat exchangers, and valves, together with enough relief valves for reactor system blowdown to low pressure. (It should be noted that non-safety grade equipment is normally used to perform this function). In order to assure that a fire would not cause a loss of both divisions, some changes in the facility design and equipment layout are necessary.

Facility modification that would be required to allow complete physical separation of the divisions would be extremely difficult to achieve in an already designed and existing facility. There are however, combinations of methods which can be used to accomplish the same purpose, e.g., 1) fabrication of the electrical distribution system with fireproof wiring, 2) fire detection and extinguishing systems that protect the critical areas of the plant, 3) elimination of substances which can be the source of fire would also reduce the potential for fire damage to both electrical divisions; and 4) relocation of the divisions as well as installation of physical barriers to limit the susceptibility of damage occurring in both divisions.

In recognition of the difficulty of achieving complete physical independence of the two divisions, the licensee proposed an overall program that con-

* The control rod system in a BWR is a mechanical system whose operating power is derived from a hydraulic system. An interruption of the electrical power to the scram circuitry causes a scram which initiates the hydraulic system and becomes independent of the electrical system and eliminates the need for continuous power supply.

sidered a number of elements. These include design changes, improved fire fighting methods, and improved administrative procedures which would assure that postulated fires within the facility would not result in a loss of both electrical divisions. This overall approach proposed by the licensee is presented in Part X of the recovery plan. The essential ingredients of that proposal are to provide changes to the electrical circuits via either sufficient separation or by the installation of various fire barriers to assure that a fire induced failure in one division could not damage the second division. A one-hour fire protection interval was selected to assure that a fire in any location in one division would not damage any portion of its redundant division even if the fire should last for one full hour. The licensee analyzed its systems taking into account the materials pertaining to the particular arrangement of the division of cabling under consideration to determine the need for additional protection. Using these analytical techniques, fire barriers are proposed to be installed such that the stated objective of a one hour fire protection interval would be available through the facility. On the basis of a one hour fire protection interval, procedures were derived to assure that a fire which could threaten the safe shutdown of the facility, would be extinguished well within one hour. Reliance is placed on using augmented administrative procedures in conjunction with hand held fire protection equipment already installed in the facility and additional hand held extinguishers which will be added throughout the facility. In those areas for which easy access by personnel with hand held fire equipment would be difficult, licensee will install fixed manually operated water sprays. The areas which have been specifically identified as requiring fixed spray systems are as follows:

a. Along cable tray runs parallel to the north wall of Reactor Building at elevation 593 feet of units 1, 2, and 3.

b. All penetrations from the cable spreading room into the turbine building and from the cable spreading room into the Reactor building containing congested cable trays where several trays make each inaccessible.

c. In other Reactor building areas where concentration of cables, as determined by visual surveys, make it difficult for immediate, easy access for fire fighting application of extinguishing agents.

To assure that any fires that may start are promptly detected, so that they may be extinguished promptly, the licensee has proposed installing a substantially extended fire detection system which will detect the start of fires and will promptly alarm in the control room. Operating procedures under development will result in prompt dispatch of fire fighting personnel to extinguish the fire.

Two types of detector will be employed to provide the fire detection function in those areas determined from the plant reanalysis to be critical. Products of combustion detectors will be installed in general on a 30 ft. grid basis. These areas will be zoned and no less than two detectors will be installed in a given zone. All detectors within a given zone will be wired in "OR" logic. Sensitivity or trip point will be 6 milligrams of products of combustion per cubic foot of air. Heat cable detectors will be installed on cable trays in the critical areas. This will provide both redundancy and diversity of detectors. The heat cable detectors will respond when the temperature at any point along the cable reaches 250 F. All sensors will be tested on a periodic basis. Products of combustion detectors within a given zone will provide one alarm for detection and another alarm for circuit trouble. Heat detectors within a given zone will provide one alarm for detection of heat or loss of power. And all detectors will receive power from the plant preferred bus.

Cable Replacement and Repair

The major effort in restoring the facility to a condition in which it will be able to operate, is the replacement and repair of damaged wiring and cables. This amounts to some 9500 conductors to be replaced or spliced. All cabling of the reactor protection system, primary containment isolation system, engineered safeguards (divisional) cables will be replaced from terminal to terminal (without splicing). Only non-divisional cables (that is those not required for providing either power or control to safety systems) will be spliced and repaired. Of these, the specific repair requirements provide for protection against high temperature or interruption of vital circuits that could result from a poor splice. The procedures proposed call for splicing at undamaged locations determined by measurements made on the insulation material, for splices in accordance with applicable codes; for accessibility to facilitate inspection; and for measures that assure no mechanical loading exists on the splices. Appropriate fire stops are provided along the cable trays away from the splice.

Replacement or repair of cabling will be of equivalent quality or better than the original cabling which was damaged by the fire.

Circuitry Changes

In addition to replacement and repair of damaged wiring and cables, TVA proposes design changes to enhance fire protection and eliminate the source of the loss of redundant safety equipment which occurred as a result of the March 22, 1975 fire.

The most significant loss of redundant equipment was associated with failures of their power sources which were caused by short circuits to lamp circuits leading from the control circuits of Reactor Motor Operated Valve (MOV) boards. These cables were considered to be non-divisional because a dropping resistor was provided in the lamp circuit for the purpose of isolating this circuit from the control circuit. Several of these cables from both divisions were run together in common trays and were, subsequently, all damaged by the fire. TVA has proposed to remove the cables leading from Reactor MOV board control circuits to breaker indication lamps to eliminate the major problem which was the loss of ability to operate the boards in both divisions. We agree with this approach and conclude that these circuit changes will eliminate the loss of redundant division MOV board controls.

The second most significant cause for loss of redundant equipment was due to the proximity of conduits containing division I and divison II cables to cable trays which were the primary source of combustible material. Although the use of conduit may provide adequate protection against some affects of fire, the fire which occurred proved that conduit protection alone was insufficient for a fire of that magnitude without additional protection in the form of additional separation or fire barriers. To remedy this TVA will modify the cabling and wiring systems to assure that cross divisional affects during fires due to the proximity of conduits are minimized. To accomplish this the cabling and wiring is to be modified, wherever divisional cables are routed in conduit near open trays carrying cables of the opposite division. Separation of divisional conduits will be provided by structures, distance, barriers, interposing ducts, pipes, etc., or combinations thereof to provide at least a one hour fire protection interval. The proposed criteria to be used to effect these changes are contained in Part X of the plan.

An analysis of the plant cable tray and conduit network has been made using the proposed criteria and approximately 80 locations have been identified for which conformance to the above proposed requirements had to be examined. The supplemental separations requirement involves the use of barriers and fire stops to inhibit cross divisional effects during a fire and to minimize fire propagation within divisions. In addition, conduits are no longer treated as adequate barriers to fire and therefore separation or other barriers are required to protect the conduit circuits. Together these additional requirements will provide a minimum of one hour fire protection interval against cross divisional effects.

A third cause for loss of redundant equipment was damage to their cables sharing the same cable tray contrary to the stated criteria. This third significant cause for loss of equipment will be remedied by separating the involved cables. Selective changes will also be made in the electric power system to improve isolation. These changes will provide individual normal feeders to the 4KV/480-V transformers. One change will eliminate sharing between units of the 4 KV feeder that was the normal supply to 480 V Shutdown Board 2A and 2B through TS3E. Other changes will provide individual 4 KV feeds to transformers TDA and TDB which supply 480-V Diesel Auxiliary Boards A and B, respectively. The only shared 4KV feeder will be the alternate supply to transformers TSIE and TDE.

The requirements for replacement and for splicing and repair of cables damaged by the fire will provide for replacement in which the cabling integrity and performance will be essentially equivalent to that provided in the original installation. The design changes in circuitry will provide substantially enhanced protection for the redundant safety system circuitry by providing increased separation so as to provide at least a one hour fire protection barrier to assure that fires in one division will not adversely affect another division even if the fire should last for a full hour.

With the increased and improved fire detection system proposed and the installation of fixed sprays in areas in which access is restricted to augment improved fire fighting procedures throughout the plant, the design changes which TVA proposes will substantially improve the ability of the facility to withstand fire.

4.0 NRR Evaluation of the TVA Approach

Although the fire that occurred caused damage to the electrical system greater than that which had been considered in our original evaluation of that facility, our evaluation has shown that even with this extensive damage considerable flexibility remained with respect to methods available to remove the decay heat from the reactor and assure the reactor was maintained in a safe condition.

In spite of the capability already inherent in the facility, the fire demonstrates that additional emphasis on the protection of the facility from a fire is needed. In this connection we have considered those features, actions and design approaches that should be a part of fire protection capability. The three basic considerations in minimizing the effects of the fire are: administrative actions that can prevent a fire from occurring; use of separation as a mechanism by which to prevent a fire from damaging redundant safety equipment; and, incorporation of a means to detect and extinguish a fire quickly.

The principal difference in our approach from that used by TVA, is that we would emphasize that each of these elements should be considered by themselves to the extent practical. As previously noted, no one of these three elements can be relied upon to completely eliminate problems associated with fires. TVA's approach instead was based on selection of a one hour fire protection separation and providing detection and fire protection equipment and procedures to extinguish fires within one hour. This is to be combined with strict administative control to prevent fires from occurring.

As indicated above the one hour fire protection separation which the changes in circuitry will provide, will substantially enhance the ability of the facility to withstand fire. However, we believe and have indicated to TVA our position that such separation should be considered as providing a base protection standard and that if practicable application of additional thermal barriers could significantly extend the one hour interval, then such additional barriers should be installed.
TVA will substantially upgrade its operating procedures and training for fire protection to assure that fires can be extinguished within one hour using the hand held fire equipment available in the facility. TVA proposes to install a fixed manually operated spray system, to supplement the hand held equipment, only in areas in which access prevents effective use of hand held fire fighting equipment.

We believe that an automatically actuated fixed spray system should be extended throughout all areas in which the potential for inter-divisional effects exist. The advice we have received from expert consultants supports this approach. The experience gained as a result of the Browns Ferry Fire indicates that if a water spray had been used earlier in the fire, considerably less damage would have resulted and the fire would have been extinguished sooner than it was. After discussions between the staff and TVA, the licensee in a letter dated August 29, 1975, has committed to converting those fixed spray systems he has presently designed to an automatic system within approximately one year. He has also committed in that letter to evaluate the installation of additional fixed spray systems to extend the fire protection coverage throughout all those areas in which such protection is required. Such studies will be completed and any areas for which this extended coverage should be provided will be identified and discussed in our subsequent evaluation prior to authorization of power operation of any of these units.

The operation of the installed CO_2 (Carbon Dioxide) fire extinguishing system in the cable spreading room was effective in preventing substantial damage to the cables located in that room. The licensee did not propose to make any further changes to the fire extinguishing equipment provided in the cable spreading room. Reliance on the manual operation of CO_2 system as is now provided is not adequate. We have informed TVA of our conclusion and TVA has agreed to make necessary changes to provide automatic actuation of the CO_2 system. We have also informed TVA that we believe that there should be a liberal application of a flamemastic material to coat the cable trays located within the cable spreading room. The CO_2 system when exhausted may not have sufficient cooling capability to fully extinguish a deep-seated fire. The licensee has indicated that in such an event the system could be augmented to extinguish any fire by using hand held equipment. Although this may be an appropriate method for augmenting the CO_2 system, we have indicated our position to the licensee that he should study further additional measures, including consideration of installing fixed water spray systems, that could be used within the cable spreading room. The licensee has agreed to provide such a study. The study will be completed and any changes will be identifed and discussed in our subsequent safety evaluation prior to authorization of return to power.

We have informed the licensee that the design of the penetration which requires building a fire stop be modified to incorporate a new material which will be demonstrated by tests to be of a fire resistant nature. This would include all penetrations that were damaged as a result of the fire as well as all new penetrations. Those in the Unit 3 cable spreading room are to be constructed using the new design. We have also informed the licensee to reexamine the penetration that now exists and assure that they are placed into the originally designed condition. We have further asked that whenever it is necessary as a result of the restoration program to breach a fire stop that the repair of that fire stop to the extent practical include removal of the original materials with replacement made of materials used for the design of new fire stops. The licensee has agreed to these provisions.

During our detailed evaluation we and our consultants have looked into the manner in which the ventilation systems throughout the plant should perform. We have concluded that in the event a fire is detected in the cable spreading room, its ventilation system should be stopped. We have not been able to complete our determinations as to whether the ventilation system used for other rooms should also be stopped in the event of a fire. There is a need for ventilation to clear and remove the combustion products thereby providing accessibility into the fire area. On the other hand, the increased ventilation rate may be undesirable with respect to the capability for causing the fire to burn with an increased force. We have informed the licensee that this matter must be studied further prior to authorizing any power operation. The licensee has committed to perform this further evaluation.

Our evaluation of the facility will continue and our subsequent evaluation prior to facility operation will discuss further facility improvements still under study as described above. However, authorization of the changes and the work activities proposed by TVA will improve facility safety as described in Section 5.0. Moreover, such work is of such a nature that further changes to effectuate potential improvements discussed above would not be precluded or hampered by the restoration activities approved at this time.

5.0 Requested Approval

There are three major action items for which the licensee has requested approval, namely, structural work, electricl design changes including installation of cables, and installation of the fire detection and spray systems. Following are the bases for the approval for each of these action items.

1. Approval to proceed with structural work which covers restoration of concrete, embedments, pipe supports, cable tray supports and the replacement of pipe and ducting in the fire-affected area. In general, the plans related to this portion of the restoration effort are described in Part VIII of the Recovery Plan and in the responses to Question 1.0 under the "General" heading.

Basis for NRC Approval

The programs and procedures for restoration of process piping, HVAC ducts and supports, hangers, restraints, cable trays and supports of affected mechanical equipment have been reviewed. The restoration program requires the evaluation of all the above mentioned components for the effects of the cable fire. Thermal sensitivity based on minimum acceptable material properties and safety function will be determined for these components and items and then compared to the fire temperature zones determined by inspection of damage and analyses. Where thermal sensitivity was lower than the actual temperature experienced, the component or item may be eliminated from further consideration and replaced to the original specification or an evaluation will be made to justify continued use of the component.

Evaluation of all process piping is required regardless of the level of exposure to temperature effects. In lieu of further evaluation some piping may be replaced to the original specifications. If piping is to remain in service the restoration program requires removal of insulation, complete cleaning of the pipe surfaces and dye penetrant inspection of the pipe and pipe fitting surfaces.

The scope of the dye penetrant inspection is set in accord with the piping material strengths and sensitivity to the chemical environment resulting from the fire.

Carbon steel piping found to have been in a 500 F temperature zone will receive an initial inspection of twenty percent of the pipe surface and one hundred percent of the fittings. The discovery of fire related cracks will require inspection of one hundred percent of the pipe surface in the same temperature zone. The boundary of the temperature zone will be extended by ten feet if the crack is found within ten feet of the initially determined boundary.

Stainless steel pipe and fittings will receive a one hundred percent inspection.

Aluminum piping will receive an inspection similar to that specified for carbon steel with the exception that the temperature zone is reduced to below 340 F. Any aluminum pipe found exposed to temperatures above 340 F will be replaced without further evaluation. In addition, if any fire related cracks are found within twenty feet of the 340 F zone a one hundred percent inspection will be performed in that twenty foot length and inspection will continue for another twenty feet of length.

We find that these procedures for restoration of process piping are properly set in accord with the material properties of the piping material and in recognition of the sensitivities of the various materials to the chemical environment resulting from the fire and that these procedures provide adequate assurance that the restored process piping will be fully capable of performing all design basis safety functions.

Safety related heating ventilating and air conditioning ducts will be evaluated on the basis of exposure to temperature of 500 F or greater as gaged by the established temperature zones and visible distortion of the ducts. All ducts showing visual distortions will be replaced or reworked to original specifications. All duct work influenced by the fire will be cleaned regardless of temperature exposure.

Evaluation of cable trays, cable tray supports and fixed members of pipe supports will be based on exposure temperatures and stress analysis considering material strengths reduced by any possible annealing affects of higher temperature exposure. Any structural steel found to have been immersed in a temperature of 1000 F or higher will be replaced without further evaluation.

In general all cable trays in direct and near contact with cables actually consumed by the fire will be replaced without further evaluation. The larger cable tray supports will be re-evaluated based on the actual load to be sustained. The acceptable stress levels for these analyses will be set at either seventy five percent of the original design allowable stress or based on the measurement of actual physical properties from a sample of the material exposed to the actual fire conditions.

The requirements specified above for acceptance of cable tray supports will also be employed for acceptance of the fixed members of pipe supports. Additional requirements have been established for acceptance of the variable elements of pipe supports. In the absence of variations in setting from the effects of the fire and the absence of actual fire damage the support setting will be corrected to give a properly aligned piping system. Setting changes as a result of the fire with no visible evidence of heat damage will require evaluation of spring characteristics to assure the necessary range of support as required by the system design. Visible heat damage will require replacement of the spring mechanism and if necessary the entire support.
We find that the procedures set forth for the evaluation and restoration of the safety related HVAC ducts cable trays, cable tray supports, and the fixed and variable members of pipe supports provide adequate assurance that these items will be capable of performing as designed throughout the range of loads and functions specified in the design bases for Browns Ferry Unit 1.

With respect to the structural steel components and the reinforced concrete structures which either support the cables or were in the vicinity of the burning cables TVA has initiated a restoration program which consists of:

1. Identification of the affected structures;
2. Establishment of the most probable temperature to which each structure was subjected during the fire;
3. Evaluation of the extent of damage of each structure;
4. Establishment of criteria for the requirement of replacement or repair of the damaged structure; and,
5. Development of procedures for detailed repair.

The structural steel components affected consist of miscellaneous structural steel supports, steel embedment, and portions of the building superstructure steel. The reinforced concrete structures affected consist of walls, columns, and floor slabs, which form the permanent building superstructure.

The temperature zones are established on the basis of the color of the structures of different materials of construction, melting of metals and burning of other materials. The extent of damage of each structure is evaluated on the basis of the temperature to which the structure was subjected and the physical condition of the structure.

The criteria for the requirement of replacement or repair of damaged structure are as follows:

1. Steel Structures

Structural steel components will be replaced or will continue to be used depending upon whether their thermal sensitivity is lower or higher than the temperature to which the structural component was subjected during the cable fire. In addition any steel structure found to have been immersed in a temperature zone of 1000 F or more shall be replaced.

2. Concrete Structure

For reinforced concrete structures, the procedures for the evaluation of the concrete will consist of detailed visual inspection, comparison of temperature zones and testing of samples from core borings. For reinforcing steel, the procedures of evaluation will consist of the following:

(a) Analyze the adequacy of the structure by neglecting the steel bars exposed to high temperatures, and if found adequate no further evaluation will be done; and,

(b) Conduct sampling and testing of steel bars to determine their material characteristics. The measured material characteristics will be evaluated by rechecking the design of the affected structure. If the reduced characteristics are still adequate to resist the loads imposed on the structure, no modifications are necessary.

On the basis of the above criteria, the floor slab at elevation 621.25 in the reactor building which appeared to be most seriously damaged was evaluated by TVA. Reinforcing steel bar and concrete core samples were taken and tested. The results of the tests show that the material characteristics of the reinforcing steel and concrete were not adversely affected by the fire. On the basis of these findings, TVA concluded that the fire damage to reinforced concrete is structurally minor. Consequently, the repairs to be undertaken are considered non structural.

We have concluded that the criteria, methods and procedures that are used in the evaluation of damage of structures and in the replacement of structural steel components and repair of reinforced concrete structures will assure that all structural components affected by the fire will be returned to their originally acceptable condition. Accordingly, since no facility modification is involved in the repair work, the work was approved by our letter dated August 28, 1975.

2. Approval of electrical design changes including circuit changes and permission to proceed with the restoration and installation of cable trays, conduit, and cables, including approval for splicing. The work and design changes are described in parts VI and X of the Restoration.

Basis for NRC Approval

The basis for the approval of this work is provided in Sections 3.0 and 4.0.

3. Approval for installation of the fire detection system and for the installation of distribution piping of the fixed spray system. These systems are described in sub-section of Part X of the Recovery Plan.

Basis of NRC Approval

The basis for the approval of this work is provided in Sections 3.0 and 4.0. A number of additional modifications were requested regarding the installation of the spray system as proposed above. The modifications include features required to make the fixed spray system capable of automatic actuation. Additional automatic fixed spray systems may also be required. The approval and performance of the work identified above does not preclude these additional requirements. The approval of the final disposition of certain of the commitments made by TVA in their letter dated August 29, 1975 will be the subject of a later safety evaluation prior to the return to operation. Some of the commitments require that TVA investigate and make proposals regarding certain areas of system improvement; (i.e., additional fixed water spray coverage, fixed water spray or alternative in the the cable spreading room, modification of cable spreading room ventilation system, additional penetration seal material). The later safety evaluation, prior to authorization for return to operation will address the final resolutuion of these items. There also may arise during the course of the restoration work additional items that will require resolution and these will also be addressed at that time.

6.0 Conclusions

As discussed in our Safety Evaluation issued in connection with Amendment 10 to DPR-33 and Amendment 7 to DPR-52, June 13, 1975, the requirements of the Technical Specifications issued on that date provide for protection of the fuel, which has been placed in the storage pools. These requirements assure ample safety systems to protect the fuel from damage. Such systems were made independent of systems involved in the restoration work and eliminate the potential for adverse interactions resulting from removal of fire damaged systems and restoration of the facility.

The replacement and repair work authorized in connection with this safety evaluation, will provide for replacement of cabling equivalent to that provided in the original installation. The design changes and improvements in circuit separation and resulting improved fire protection, along with the design improvements in fire detection systems and fire extinguishing systems proposed by TVA, substantially enhance the capability of the facility to withstand fires and will provide reasonable assurance that the public health and safety will not be endangered. These changes do not involve a significant increase in the probability or consequences of accidents previously considered and do not involve a significant decrease in a safety margin.

Based on the foregoing considerations, we have concluded that there is reasonable assurance that the public health and safety will not be endangered by the proposed changes and the restoration activities authorized herein and that such changes do not involve significant hazard considerations.

Our review of the safety of the facility to return to operation is still in progress and additional changes may be required as a result of our continuing review, or as a result of the additional studies which we have indicated are needed and which TVA has proposed as discussed above. This safety evaluation does not address resumption of operation at Browns Ferry Units 1 and 2. Any authorization to resume operation will be considered in a subsequent safety evaluation which will also consider necessary changes to plant operating technical specifications required in connection with such authorizations.

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NOS. 50-259 AND 50-260

TENNESSEE VALLEY AUTHORITY

NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES

Notice is hereby given that the U.S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 14 to Facility Operating License No. DPR-33 and Amendment No. 11 to Facility Operating License No. DPR-52 issued to Tennessee Valley Authority for operation of the Browns Ferry Nuclear Plant, Units 1 and 2, located in Limestone County, Alabama. The amendments are effective as of their date of issuance.

The amendments modify the licenses to authorize modifications to Units 1 and 2 in conformance with "Plan for Evaluation, Repair, and Return to Service of Browns Ferry Units 1 and 2 (March 22, 1975 Fire)" in accordance with the licensee's request dated August 29, 1975. These amendments do not authorize return to operation of Units 1 and 2. That authorization will be the subject of another action upon completion of our review of the total restoration work required.

The application for these amendments complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendments. Prior public notice of these amendments is not required since the amendments do not involve a significant hazards consideration.

For further details with respect to this action, see (1) the application for amendments dated August 29, 1975, (2) Amendment No. 14 to License No. DPR-33 and Amendment No. 11 to License No. DPR-52, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, NW., Washington, D.C. and at the Athens Public Library, South and Forrest, Athens, Alabama 35611.

A copy of items (2) and (3) may be obtained upon request addressed to the U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Reactor Licensing.

Dated at Bethesda, Maryland, this 2nd day of September 1975.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert A. Purple, Chief
Operating Reactors Branch #1
Division of Reactor Licensing

ATTACHMENT V

FIRES AT U.S. AND FOREIGN NUCLEAR PLANTS

FIRES AT U.S. NUCLEAR PLANTS

Peach Bottom Unit 1

During construction activities on February 3, 1965, a cable fire was started as a result of sparks from a welding operation. One of the design changes instituted as a result of the fire was to replace cable using a polyethylene inner sheath with butyl-rubber sheathed cable.

A second fire took place on April 13, 1967. The fire resulted from the ignition of polyurethane thermal insulation on piping associated with the low temperature delay beds and regenerative heat exchangers, components of the helium purification system.

San Onofre Unit 1

There were two fires in 1968; both involved cable trays. The fire on February 7, 1968, resulted from mechanical and electrical overloading of power cables for the pressurizer heaters. The fire on March 12, 1968, resulted from cable tray loadings in excess of design specifications.

Nine Mile Point Unit 1

During the startup testing in 1969 and 1970 a fire occurred which resulted in the destruction of motor control centers. Loose terminations on line conductors at the contactors and breakers caused the fire.

4. Indian Point Unit 2 — There was a construction fire involving wood scaffolding which burned the cable trays and was reported as a personnel error. The fire occurred on November 4, 1971.

5. Quad Cities Unit 2 — There was a fire in two electrical trays in the reactor building. The fire was confined to approximately 5 feet at the end of two cable tray sections. This event occurred July 16, 1972.

6. Beaver Valley Unit 1 — A fire occurred at a motor control center on October 4, 1972, apparently resulting from a short circuit in one of the splices in the temporary wiring for the heat-light bulbs in the base of the motor control center.

7. Oconee Unit 1 — A fire took place August 20, 1970, during construction activity near the reactor coolant pump suction piping. Oil from machining operations ignited and burned, killing one man and injuring three others.

A second fire took place in the control rod drive transfer cabinet on March 9, 1972. The fire was caused by a loose electrical connection. Damage was limited to a few components within the cabinet.

A third and fourth took place on December 30, 1972. These fires were caused by the ignition of oil spilled from the reactor coolant-pump oil system when the oil came in contact with reactor coolant systems components.

8. Oconee Unit 2

A fire took place in the turbine building on June 27, 1975. Oil leaking from the turbine oil purifier system ignited when it came in contact with the purifier heaters. Cables above the fire were charred; however, there was no loss of operational circuits. A report is being prepared by the utility.

9. Salem Unit 1

A construction fire occurred in April, 1974, in a cable tray located in the 4 KV switch-gear room of the auxiliary building. The fire occurred during a welding operation involving a piece of weld slag which fell onto a cable tray.

10. Salem Unit 2

A fire took place on October 18, 1974 during construction activities. Wood construction forms were ignited and burned cables in the area of the heavy equipment hatch of the containment building.

11. Browns Ferry Unit 1 & 2 — A fire occurred on March 22, 1975 in the cable spreading room during a penetration leak test; a candle ignited polyurethane foam material used to seal penetration cable openings.

FIRES AT FOREIGN NUCLEAR PLANTS

1. Muhleberg, Switzerland — During power ascension testing a turbine-generator oil fire occurred. This was a result of failure of a screwed pipe connection of the pressurized oil pipe leading to a control valve. The oil ignited and the fire spread to cables in the immediate area. This fire spread through unprotected cable tray openings into adjacent cable rooms. The fire occurred near the end of 1971.

2. Hinkley, England — Due to failure of a hydraulic relay valve of a turbine, oil sprayed out and was ignited by an adjacent hot steam valve. The fire at the turbine was extinguished quickly by means of a manually controlled water spray system as well as a hose line from an inside hydrant and portable

extinguishers. Burning oil flowing along cable trays caused serious damage to the cables. This fire took place in July, 1966.

3. Tokai Mura/Unit 1, Japan

Due to failure of a cock in a strainer assembly in the reactor building a major quantity of oil sprayed out when a strainer was being changed, and was ignited by a hot pipe handling 360° C steam. The sprinkler system put out the fire in four minutes. The reactor was not damaged but was shut down while repairs were made to the strainer assembly. The fire took place in 1967.

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MICROCOPY RESOLUTION TEST CHART

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