NRC STAFF REPORT

ON

UNION OF CONCERNED SCIENTISTS' PETITION

FOR EMERGENCY AND REMEDIAL ACTION

December 15, 1977

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I. INTRODUCTION

A. UCS Petition

On November 4, 1977, the Union of Concerned Scientists (UCS) filed a petition with the Commission alleging certain safety deficiencies in connection with the design of nuclear power plants and requesting the Commission to take certain emergency and remedial actions. The alleged deficiencies involved the environmental qualification of electrical connectors and fire protection requirements. The bases stated in the petition are the results of certain recent testing work from the Commission's Qualification Testing Evaluation Program and Fire Protection Research Program, both of which are conducted for NRC by Sandia Laboratories in New Mexico.

The UCS petition alleges that electrical connectors of the type used in nuclear power plants failed in the Sandia tests under environmental conditions similar to those that would occur inside of containments in the event of a loss-of-coolant accident (LOCA). Further, the petition asserts that the tests revealed deficiencies regarding the flammability of electrical cables and the separation of redundant divisions of cables required by current NRC regulations. In particular, the petition alleges that there is a failure to meet (a) the Commission's General Design Criterion 4, the single failure criterion of Appendix A to 10 CFR Part 50 and 10 CFR Part 50.55a(h), with regard to equipment qualification; and (b) General Design Criterion 3, the single failure criterion of Appendix A to 10 CFR Part 50 and Part 50.55a(h), with regard to fire protection.

The UCS petition requested the following actions:

"a. The Commission shall direct the Staff to accelerate a testing program to determine the type of physical separation between electrical cables necessary to maintain the independence and to meet the single failure criterion for redundant safety systems.

"b. The Commission shall direct the Staff to accelerate a testing program for environmental qualification of connectors.

"c. The Commission shall direct the Staff to independently verify the environmental qualifications of all safety-related systems, components and structures.

"d. All Licensing and Appeal Boards should immediately be notified that no further construction permits or operating licenses can be issued until such times as Applicants can demonstrate compliance with the applicable regulations, including specifically General Design Criterion 3 and 4 of Appendix A to 10 CFR Part 50, 10 CFR Section 50.55a(h), and the single failure criterion of Appendix A to 10 CFR Part 50.

"e. All holders of construction permits shall immediately be notified to cease all construction activities involving the connectors identified as defective and all activities relating to electrical cables.

"f. All operating reactors shall immediately be ordered to shut down until such time as the operators can demonstrate compliance with the applicable regulations, including specifically General Design Criteria 3 and 4 of Appendix A to 10 CFR Part 50, 10 CFR Section 50.55(h), and the single failure criterion of Appendix A to 10 CFR Part 50."

A supplemental affidavit was filed by the UCS on November 10, 1977 which contained additional comments regarding electrical connectors and fire protection. A second supplemental affidavit was filed by UCS on November 17, 1977 responding to the Staff's comments made at the Commission's November 11 public briefing. These supplemental affidavits also expressed concern regarding the qualification of other electrical equipment, including electrical penetrations, cables and cable terminations, for a LOCA environment.

B. Commission Actions

As a result of the November 4, 1977 UCS petition the Commission directed the Staff on November 7, 1977 to report on the need for immediate action in response to the petition. The Staff's report on this aspect of the petition was transmitted to the Commission on November 9, 1977.

On November 9, 1977, the Commission issued an Order, published in the Federal Register on November 11, 1977 (42 FR 58803), soliciting views from licensees and members of the public by November 25, 1977 on the UCS petition.* On November 10, 1977 the Commission's Offices of General Counsel and Policy Evaluation developed a number of written questions concerning the Staff's November 9, 1977 report. An open briefing of the Commission by the staff on the emergency aspects of the UCS petition was held on November 11, 1977.

Thereafter the Staff filed additional reports to the Commission on the UCS petition on November 18, 22, and 25, 1977 and December 6, 1977. These reports address the matters raised in the UCS petition and provide updated information on ongoing staff actions to obtain information on the use of electrical connectors and penetrations in operating reactors. The November 22, 1977 report also included staff responses to the questions raised by the NRC Office of the General Counsel and the NRC Office of Policy Evaluation. All of these reports and the UCS petition are on file in the NRC public Document Room.

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*To date 44 comments have been received. None of these responses identified any new safety information that had not already been covered by the UCS petition or the Staff's reports. The comments are addressed elsewhere in this report.

On December 8, 1977 the Commission held a second open meeting on the UCS petition. Both the staff and the UCS made presentations at that meeting.

A chronology of correspondence related to the UCS petition is given in Appendix A.

C. Summary of Safety Aspects of Petition

In this report and in the previous staff reports on the UCS petition, the Staff has addressed the safety aspects of the UCS petition. Specifically, the Staff views are as follows:

1. Fire Protection

   a. The information presented in the UCS petitions, the public comments, and fire test data available from Sandia do not warrant acceleration of the fire protection research program now underway at Sandia;

   b. There is no need to revise the current NRC acceptance criteria for fire protection of nuclear power plants; and

   c. Actions taken and underway by the staff, licensees and applicants prior to and after devlopment of the current fire protection criteria for nuclear power plants provide adequate protection to the public health and safety, pending full implementation of the current criteria.

2. Electrical Components

   a. On the basis of information and actions to date the Staff has concluded there is reasonable assurance that electrical connectors and containment electrical penetrations will perform their required functions in the LOCA environment.

b. The Staff is continuing to review the responses to IE Bulletins 77-05/05A/06 in regard to the performance of electrical connectors and penetrations under postulated accident conditions other than LOCA.

c. The Staff will review the entire topic of electrical equipment qualification as the first topic of the Systematic Evaluation Program for Operating Reactors.

II. STAFF RESPONSE TO UCS PETITION

This section addresses the actions requested by the UCS in its November 4, 1977 petition. On the basis of the information in this report and the previous staff reports filed with the Commission, the Staff concludes that this petition should not be granted for the reasons stated below.

"(a) The Commission shall direct the Staff to accelerate a testing program to determine the type of physical separation between electrical cables necessary to maintain the independence and to meet the single failure criterion for redundant safety systems."

RESPONSE

The Staff does not believe that the Fire Protection Research Program should be accelerated.

The Fire Protection Research Program is intended to provide a data base for use in evaluating design standards and regulatory guides for fire protection and control. At the present time, the major emphasis is directed toward the study of the effects of cable tray spacing on fire propagation; however, the program includes other aspects of fire research, such as the effects of materials, coatings, barriers, detection and suppression.

For the reasons outlined in Section IIIA of this report, physical separation is not the only significant consideration relied upon in providing adequate fire protection.

Separation requirements alone are not being emphasized in this program at this time. The Staff sees no immediate need to devote resources to physical separation tests beyond those presently in the test program. The Staff is concentrating its efforts in the area of overall fire protection to ensure that adequate fire protection measures are available for safe plant shutdown in the event of a major fire. Section IIIA of this report discusses the NRC Fire Protection Action Plan for assuring safe operation of nuclear power plants.

"(b) The Commission shall direct the Staff to accelerate a testing program for environmental qualification of connectors."

RESPONSE

The Staff does not believe that the Qualification Testing Evaluation Program should be accelerated.

The present NRC-sponsored Qualification Testing Evaluation Program at Sandia was specifically developed to obtain data to examine current standards and regulatory guides for the environmental qualification testing of safety-related equipment required to operate in a LOCA environment. In view of this objective, the Sandia tests were performed to evaluate the adequacy of a testing methodology and not to verify the qualifications of any particular electrical component to withstand a LOCA event. The recent connector tests resulting in failures were specifically performed to determine if there are synergistic effects with these materials resulting from simultaneous radiation and LOCA steam exposure, as compared with the method in IEEE-323 of sequentially applying radiation followed by the LOCA environment. We believe the program schedule is appropriate for this goal.

The failure of the connectors in the Sandia tests was associated with the inadvertent use of unqualified components that were thought to have been properly qualified for the test environment.

A proper response to concerns regarding the qualification of connectors in nuclear power plants is accelerated identification of instances where connectors are used in safety systems and the determination of the adequacy of qualification of such connectors. This is the course of action taken by the Staff starting on November 8, 1977. The results of this effort and additional staff plans for future action are discussed in Section III.B and Appendix B of this report.

"(c) The Commission shall direct the Staff to independently verify the environmental qualification of all safety-related systems, components and structures."

RESPONSE

To the extent that this request proposes that the NRC undertake the environmental qualification test work we do not agree. That is a responsibility of licensees.

The Staff has been investigating the development and evaluation of qualification test procedures, test methods and test results with regard to environmental qualification of safety-related equipment since before 1968. The Staff's concerns and recommendations have been reflected in both the 1968 and 1971 versions of IEEE Std. 279-1971, "Criteria for Protection Systems for Nuclear Power Generating Stations," and the 1971 and 1974 versions of IEEE Std. 323, "Standard for Qualifying Class IE Equipment for Nuclear Power Plants." The Staff further undertook the Sandia Qualification Test Program as a logical extension of its standards

development efforts to evaluate the adequacy of the qualification testing methodology. In concert with the standards development and the testing methodology verification program at Sandia, the Staff has been actively investigating and evaluating specific equipment qualification tests performed by licensees. A generic task action plan (A-24) has been developed which will include comparisons of the Sandia test program results with tests of safety equipment by the responsible licensees or their equipment suppliers. Moreover, the Staff has undertaken a comprehensive review of the qualification work done in connection with connectors and penetrations. This review is still in progress. We are carefully assessing the environmental qualification method used by the licensees to independently assess the adequacy of these methods in demonstrating qualification. Staff plans for further action are discussed in Section III.B and Appendix B of this report. We see no need for further action by the Commission with respect to this matter.

"(d) All Licensing and Appeal Boards should immediately be notified that no further construction permits or operating licenses can be issued until such time as Applicants can demonstrate compliance with the applicable regulations, including specifically General Design Criteria 3 and 4 of Appendix A to 10 CFR Part 50, 10 CFR 50.55a(h), and the single failure criterion of Appendix A to 10 CFR Part 50."

RESPONSE

For the reasons discussed at length in Section III A of this report, we believe that all such facilities have adequate programs for implementation of fire protection to satisfy the requirements of the GDC-3. Starting in 1978, the staff will assure that fire protection program reviews conducted in accordance with current guidance in the Standard Review Plan are completed before operating licenses are issued.

We believe that present requirements for environmental qualifications satisfy the requirements of GDC-4. In any instance in which items or components are identified for which sufficient basis can not be demonstrated to assure qualification, such information will be brought to the attention of any licensing board that may be considering an application for such facility. This would be rare in construction permit cases, since the details of final system designs and environmental qualification programs are not usually a part of a construction permit application.

"(e) All holders of construction permits shall immediately be notified to cease all construction activities involving the connectors identified as defective and all activities relating to electrical cables."

RESPONSE

All holders of construction permits were directed by I&E bulletins 77-05 and 77-05a to (1) identify all connectors utilized in safety systems which are required to function to mitigate an accident where the accident itself could adversely affect the ability of the system to perform its safety function, amd (2) indicate the status of the qualification of such connectors.

In any instance in which unqualified or inadequately qualified components are identified, the Staff will take appropriate action to assure that qualified components are provided before operating licenses are issued.

Similarly, as indicated in Section III A, starting in 1978 fire protection reviews in accordance with the current guidance of the Standard Review Plan will be completed before operating licenses are issued. All present holders of construction permits have been notified of the requirement to assure that any application for an operating license includes specific consideration of fire protection in accordance with that guidance.

"(f) All operating reactors shall immediately be ordered to shut down until such time as the operators can demonstrate compliance with the applicable regulations, including specifically General Design Criteria 3 and 4 of Appendix A to 10 CFR Part 50, 10 CFR 50.55(h), and the single failure criterion of Appendix A to 10 CFR Part 50."

RESPONSE

As discussed in detail in Section III A, with the additional improvements of fire safety made in operating plants since the Browns Ferry fire and with the current program for the review and implementation of current guidance for operating plants, those plants can continue to operate safely.

The overall fire protection programs in such plants provide adequate implementation of the goals of GDC-3, until full implementation of the new guidance can be completed for these plants.

As discussed in Section III B, the Staff has already undertaken a review of the qualifications of connectors and penetrations to determine the adequacy of environmental qualification of those items. In general, equipment qualification has eventually been demonstrated, despite some early inadequacies in information supporting such qualification. The Staff has, where appropriate, taken prompt action to obtain proper qualification or replacement of unqualified items.

Section IIIB and Appendix B also discuss the staff program for future actions concerning environmental qualification of other

safety-related electrical components. During this program, the Staff will assure that prompt appropriate action is taken in the event that items of doubtful qualification are identified.

The single failure criterion requirements of Appendix A to 10 CFR Part 50 and 50.55a(h) applicable to fire protection and environmental qualification requirements do not establish a set of design basis events. Rather, they establish standards for design and performance of electrical systems to assure that such systems are capable of performing as required. The staff reviews discussed in detail in Section III show that plants meet the requirements and that the Sandia tests do not bear upon consideration of "single failure" requirements, but rather upon the basic question of conformance with overall design goals.

The staff concludes that no additional Commission action is required with respect to these matters.

III. SAFETY CONSIDERATIONS

A. Fire Protection in Nuclear Power Plants

1. NRC Fire Protection Requirements

The Commission's basic requirements for fire protection for nuclear power plants are set forth in General Design Criterion (GDC) 3. Prior to October, 1972, the Staff had not developed guidelines for implementing GDC 3. Determinations of the adequacy of fire protection were made on a case-by-case basis in the review of individual applications. Subsequently the Staff provided guidance for implementing GDC 3 in Section 9.5.1 of the October, 1972 issue of the "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants" and in the initial issue (September 1975) of the Standard Review Plan, Section 9.5.1. This guidance was general in nature, and did not provide acceptance criteria for fire protection programs and systems for specific safety-related areas in nuclear power plants. Overall fire protection acceptance criteria referred, in a general manner, to industry standards, such as those of the National Fire Protection Association, Underwriters Laboratories, and the National Electrical Manufacturers Association.

Following the fire at the Browns Ferry Nuclear Power Station on March 22, 1975, the Nuclear Regulatory Commission established a Special Review Group to identify the lessons learned from this event and to make recommendations concerning fire protection at nuclear power plants in light of these lessons. The results of this study were published in NUREG-0050, "Recommendations Related to the Browns Ferry Fire," February 1976.

The Special Review Group recommended improvements in four broad categories: (1) guidance to applicants and licensees; (2) evaluation, inspection, and enforcement procedures; and (3) the fire protection programs at licensed facilities; and (4) local governments' emergency procedures. To implement these recommendations, the NRC established an agency wide action plan called the Fire Protection Action Plan which involves the major program offices, i.e., Nuclear Reactor Regulation, Inspection and Enforcement, Standards Development, Nuclear Regulatory Research, Nuclear Materials Safety and Safeguards, and State Programs. Brookhaven National Laboratory and Sandia National Laboratory have been engaged to provide technical assistance to this program. This action plan brings all NRC activities regarding the implementation of the Review Group's recommendations together into a single integrated plan. NRC management periodically reviews the progress of this program via the Status Summary Report for the Fire Protection Action Plan and monthly reports to the Commission regarding MBO VII. In May 1976, as part of this plan, the NRC Staff revised section 9.5.1 of the Standard Review Plan to issue new fire protection guidelines for the implementation of GDC 3.

2. <u>NRC Fire Protection Actions Following Browns Ferry Fire</u>

The following is a summary of actions taken by the NRC as a result of the Browns Ferry fire:

(a) Promptly after the fire, the Office of Inspection & Enforcement sent special bulletins to all licensees of operating power reactors on March 24, 1975 and April 3, 1975 directing certain controls over ignition sources, a review of procedures for controlling plant maintenance and modifications that might affect safety, a review of emergency procedures for alternate shutdown and cooling methods, and a review of flammability of materials used in floor and wall penetration seals. Some of the changes and improvements at operating plants whch have come about as a result of these bulletins are:

(1) modifications of work procedures to assure consideration of the safety significance of electrical cables and piping in the work area;

(2) incorporation of the control of combustible materials into plant procedures;

(3) improved plant procedures for the control of ignition sources;

(4) development of new procedures and guidelines covering the use of water on electrical cable;

(5) study and development of procedures for a variety of means to provide decay heat removal;

(6) addition, upgrading and repair of cable penetration firestops; and

(7) addition of fire suppression equipment.

(b) Promptly after the fire, special inspections by the NRC Office of Inspection and Enforcement were completed at all operating power reactors in April and May 1975 covering the installation of fire stops on electrical cables and penetration seals. Inspection findings which reflected noncompliance with current NRC requirements resulted in requiring corrective action by licensees. Follow-up inspections have confirmed that licensees took required corrective actions and that administrative control procedures were in place.

(c) More detailed procedures for inspection of fire prevention and protection measures have been incorporated in the NRC Operating Reactor Inspection Program. Since September 9, 1975 the Office of Inspection and Enforcement has been conducting detailed annual inspections of licensees' fire protection programs as one of the routine I&E inspection modules. Additionally, a plant tour is conducted quarterly, during which the inspector looks for conditions that might contribute to fires. These inspections include review of fire insurance inspection reports.

(d) Partly in response to the Browns Ferry fire, quality assurance inspection procedures have been improved steadily over those in effect at the time of preoperational QA inspections of Browns Ferry Units 1 and 2. Specifically in response to the fire, inspection procedures for plants under construction and in the preoperational testing phase were revised to include increased focus on verifying conformance with regulations on installation of electrical cables and penetration seals.

(e) When the results of the review and evaluation of plant modifications for the Browns Ferry plant became available, they were used as an example for the development of new guidelines on fire protection for inclusion in the Standard Review Plan, applicable to all nuclear plants.

(f) Factory Mutual Research Corporation--an expert and independent fire protection development and testing organization--was engaged by NRC to provide technical consultation on fire protection and to identify alternate fire protection methods and systems for NRC consideration.

(g) As a result of the Special Review Group's work on the Browns Ferry fire, a number of areas were identified in which improvements were needed in standards. Since a number of organizations have relevant expertise, the Executive Committee of the Nuclear Standards Management Board of the American National Standards Institute (ANSI) has provided a coordinating Steering

Committee to direct and coordinate standards improvements. The NRC Staff has prepared an outline of possible additional or improved standards needed to provide supplemental guidance for implementing the Staff's fire protection guidelines set forth in the Standard Review Plan. The Brookhaven National Laboratory is providing technical support to the NRC staff in this effort.

(h) Sandia Laboratory is conducting NRC-sponsored research to investigate standards for electrical cable separation at nuclear plants, and, in addition, has augmented its fire protection research program to include such areas as the effect of exposure fires, testing of the effects of cable tray separation, and testing of electrical cables in conduits and with fire barriers. High priority has been placed on the testing of fire retardant coating materials for cables. Future programs will include upgrading of flame tests for individual cables; evaluation of cable fire propagation characteristics of aged cables and coating materials; evaluation of early warning fire detection systems; and evaluation of fire extinguishing mechanisms.

3. <u>NRC Fire Protection Guidelines Since Browns Ferry Fire</u>

As indicated above, in May 1976, NRC issued new fire protection guidelines for nuclear power plants. These guidelines are largely based on the lessons learned from the Browns Ferry fire and recommendations from

NUREG-0050 and the NELPIA and the Factory Mutual System. The guidelines for construction permit applications docketed after July 1, 1976 are contained in Standard Review Plan Section 9.5.1 dated May 1, 1976. These guidelines were also issued for public comment as Regulatory Guide 1.120 in June 1976, and a revision to the Guide was reissued for comment in November 1977. Among the more significant features of the guidelines in Standard Review Plan Branch Technical Position 9.5-1 are:

(1) early involvement and assigned responsibility of upper licensee management in the fire protection program throughout plant life, and a management system for implementing fire protection administrative controls, quality assurance measures, fire brigade training and fire protection system testing and maintenance;

(2) early coordination of plant building and systems design and layout with fire protection requirements, more effective separation and isolation--by use of fire barriers--of redundant safety systems, guidance on acceptable fire resistant building and system construction materials; and

(3) specific NRC guidance for each major safety related plant area with respect to acceptable fire detection and suppression, fire barrier separation and fire protection support requirements for ventilation, emergency lighting and communication.

Significant details of the new guidelines for fire protection programs can be summarized functionally as requiring the applicant or licensee to:

(1) provide separation (by fire barrier or distance) between redundant safety-related systems to assure that a safety function will not be lost;

(2) limit the amounts of combustibles, inherent or transient, in safety-related areas;

(3) control the deliberate movement of combustibles and ignition sources;

(4) provide fire detection systems in all safety-related areas;

(6) provide a supplemental hierarchy of fire suppression systems for all safety-related areas that will be adequate to suppress both small and large fires involving the expected combustibles in a manner that does not introduce an adverse affect on safety capability;

(7) provide arrangements to supplement the fire brigade;

(8) provide plant Technical Specifications to assure the availability of the fire detection and suppression systems during plant operation and shutdown; and

(9) provide operating procedures for achieving safe shutdown conditions under a spectrum of fire conditions.

Appendix A issued in August, 1976, to Branch Technical Position 9.5-1 modifies, as appropriate, the guidelines in the Branch Technical Position for those nuclear power plants for which applications for construction permits

were docketed prior to July 1, 1976. Although the guidelines of the Branch Technical Position provide preferred guidance for fire protection programs, alternative fire protection guidelines are identified in the Appendix. The alternatives apply for areas where, depending on the construction or operational status of a given plant, application of the guidelines could have significant impact, e.g., where the building and system designs are already complete and construction is in progress, or where the plant is in operation. These alternative guidelines are intended to provide adequate and acceptable fire protection consistent with safe plant shutdown requirements.

As is true with all Staff regulatory guides (as distinguished from Commission regulations), alternatives may be proposed by applicants and licensees. These alternatives are evaluated by the NRC staff on a case-by-case basis where such departures are justified. Among the alternatives that may be considered is the provision of a "dedicated" system or alternative method for assuring continued safe shutdown of the plant in the event of a fire. This dedicated system must be completely independent of other plant systems, including the power source; however, for fire protection, it may not be necessary for the system to be designed to seismic Category I criteria or meet single failure criteria because it serves only as a system of last resort when other safety systems have failed. Manual fire fighting capability to protect other safety-related systems is still required even when the dedicated system alternate is chosen.

Regulatory Guide 1.120 was issued for public comment in June 1976. It was revised and issued again for public comment on November 7, 1977, clarifying areas addressed in the public comments. The Standard Review Plan Section 9.5.1 is being modified again, to be consistent with Revision 1 to Regulatory Guide 1.120. Among the more significant technical changes in the revised Guide are improved guidance on evaluating postulated fires as part of fire hazards analysis required for each plant; improved guidance on applying the single failure criterion (for both active and passive components) to the fire suppression systems; and improved guidance for fire protection of cable trays outside cable spreading rooms.

4. Plant Fire Protection Staff Evaluation Program

When the new fire protection guidelines were promulgated in May 1976, all licensees were requested to compare their fire protection programs with the new guidelines and to propose modifications for compliance, or to justify non-compliance. As information came in from operating reactors additional guidelines were developed for specific areas of concern and additional information was requested from licensees, including requests to provide fire hazards analyses and Technical Specifications for fire protection systems. These analyses were to be performed by persons expert in nuclear safety and fire protection, to consider the consequences of a postulated fire in each plant area, to consider electrical cable insulation as combustible material, and to consider the effects of the loss of cables due to a fire when redundant cables are routed in close proximity to one another.

The licensees began submitting their comparisons with the new guidelines in July 1976, and the staff reviews of these reports are continuing. As of November 7, 1977, the guideline comparisons, the fire hazards analyses and the proposed Technical Specifications had been received from licensees for all but one operating power plant.* The staff has established multi-disciplinary review teams with expertise in nuclear safety, fire protection, and fire fighting. A team visits each plant to review the complete fire protection program. During this visit, the team considers the qualifications of site personnel responsible for fire protection and suppression, unique situations that may hinder manual fire fighting, and arrangements for supplementing the fire brigade.

The staff evaluates the plant features related to fire protection, as well as the analysis of consequences of a postulated fire in each area of the plant designated as a fire area. We assume that any combustibles may burn and estimate the effects of an unmitigated fire on safety and on the capability for safe shutdown. We evaluate the various alternatives proposed by licensees on a case-by-case basis. Because the primary objective of our action plan is to improve fire protection at nuclear power plants, our objectives and acceptance criteria are designed to reduce the probability that a major fire will occur in any plant and to increase each plant's capability to achieve and maintain a safe shutdown condition should a fire occur.

*Arkansas 1 has indicated that some additional information will be submitted in February1978.

Simply stated, the objectives of the fire protection program in a nuclear power plant are to (a) reduce the likelihood of occurrence of fires; (b) promptly detect and extinguish fires, when they occur; (c) maintain the capability to shut down the plant safely when fires occur; and (d) prevent the release of a significant amount of radioactive material, when fires occur.

The results of our evaluations will be published in safety evaluation reports which accompany amendments to the Plant Technical Specifications related to the Fire Protection Program. Our review of the fire protection programs for operating facilities has been in progress since May 1976. Except for Arkansas Unit 1, all of the operating facilities have submitted their fire protection programs for our review. Draft Safety Evaluation Reports, which reflect the results of our review, are in preparation for fifteen plants. We expect to issue these fifteen Safety Evaluation Reports by February, 1978, which is one month behind the schedule in the NRC Fire Protection Action Plan. We expect to complete all of our reviews of operating plants by December, 1978 as currently scheduled in the action plan.

We have also completed interim reviews of the current fire protection capability of each operating plant and have issued safety evaluations and proposed interim Technical Specifications regarding fire protection to govern the period until we complete the full evaluation of plans to achieve conformance with the Appendix A guidance.

In addition to the fire protection review of individual plants, teams also conduct generic studies of fire protection problems, and guideli~~s. These generic studies have considered several areas of concern in the conduct of fire hazards analyses. For example, we have compared data obtained from several fire testing methods with fire damage experienced in actual fires to estimate how such test data can be used in predicting fire damage. We have made our own estimates of design basis fire conditions in postulated configurations that are representative of certain areas of licensed facilities. We have studied the data available on oil fires in power plants to estimate the magnitude of such hazards. We have collected data appropriate for estimating the size of fires that may be associated with selected combustible configurations found in licensed facilities.

It is against this background of staff effort to assure adequate fire protection in nuclear power plants that the Sandia tests can be placed in proper perspective.

5. <u>Sandia Fire Protection Research Program</u>

a. <u>Cable Tray Fires - Test Descriptions</u>

The cable tray fire tests referenced in the UCS petition were a series of tests conducted as part of the NRC Fire Protection Research Program. The purpose of this program is to provide a data base to be used in evaluating design standards and regulatory guides for fire protection and control. The particular tests discussed in the UCS petition were performed to study the effect of electrical cable tray spacing on fire propagation.

The tests were conducted in a building 60 ft long, 19 ft high, and 24 ft wide, simulating an open plant configuration where the effects of reflected heat from walls and ceiling are minimized. Fully loaded (40% volume fill), horizontally oriented, steel ladder cable trays eighteen inches wide were used. Cables meeting the flame retardant standards of IEEE 383-74 were used in all of the tests.

Fire propagation tests were first conducted b, initiating the fire with simulated electrical overloads and short circuits in a single tray. Cable tray spacing was varied from 5 ft vertical and 3 ft horizontal separation to 10.5 in. vertical and 8 in. horizontal separation with no propagation or cable damage in any tray, except the ignition (donor) tray.

The next phase of propagation tests investigated the resistance to fire propagation in fourteen closely packed horizontal cable trays (2 rows - 7 trays high) identical to the cable trays described above. The fourteen cable trays were arranged in two stacks with each tray spaced 10.5 in. vertically with 8 in. horizontal separation between stacks. These cable trays simulated a single fully stacked safety division. Three additional identical and fully loaded cable trays, representing a redundant safety division, were arranged with one located 5 ft above the 14-tray stacked safety division, the second located 3 ft to the side, and the third located 5 ft above the second. The vertical and horizontal separation distances between safety divisions correspond to those listed in Regulatory Guide 1.75 for open plant areas. Propagation tests were conducted using the electrical initiation method employed in the previous tests and there was no propagation or cable damage in any tray except the ignition (donor) tray.

The next series of tests conducted on July 6, 1977 utilized an exposure fire to study the current regulatory position that calls for consideration of exposure fires in design of fire protection systems. The tests did not include the use of any fire suppression system, nor were they designed to determine the effectiveness of such a system. Since there is no single exposure fire that could be considered typical, the objective of the test was to develop the severest fire possible in an ignition (donor) tray without the ignition source itself influencing any of the adjacent cable trays. For the type of cable utilized, experiments showed that a fully developed cable tray fire (in terms of near maximum cable bundle and flame temperatures) could be obtained if the cable tray fire involved an area of cable about 18 in. by 36 in. and that two 70,000 Btu/hr capacity burners placed beneath the cable tray would usually produce such a fully developed cable tray fire. It was also determined that at least five minutes were required before the near maximum steady state cable bundle and flame temperatures were reached. Therefore, an insulation barrier was placed above the ignition (donor) tray that kept the temperature rise in the adjacent trays negligible while the fire in the ignition tray was brought to the fully developed state.

After the predetermined time of five minutes, the burners were shut off and the insulating barrier removed. In this manner, cable tray propagation could be studied for a fully developed cable tray fire started by an exposure fire without the exposure fire itself burning or contributing to the propagation from tray to tray. Two tests were conducted. In the first test a fully developed cable tray fire in the ignition tray was not obtained. After determining that the temperature rise in the adjacent cable trays was still negligible, the insulating fire barrier was replaced above the donor tray and the cable in the ignition tray was repositioned to allow for a more optimum air fuel mixture to support combustion. The testing to date indicates that the most important factor in determining if a cable tray fire will develop is the spacing between cables. In both exposure fire tests, optimum spacing with a random fill pattern was sought. Usually in power plant cable trays the cables are in a more uniform pattern, making it more difficult to start a fire. After repositioning the cable for a second test, the propane burners were reignited with the insulating fire barrier in place. After five minutes, the burners were turned off and the insulating fire barrier removed. This second test resulted in a fully developed fire in the ignition tray with subsequent propagation to the tray above it in about ten minutes. Propagation proceeded up to the two seven tray stacks and caused ignition of the simulated redundant safety division above this stacked safety division in about twenty-five minutes. Five

minutes later, the supportstructure collapsed and the test was terminated. Ignition of the simulated redundant safety division five feet above the safety division was verified prior to termination of the test. Before terminating the test, no attempt was made to extinguish the fire. After terminating the test, complete suppression of the fire required 75 gallons of water and was accomplished in 15 minutes. It should be pointed out that the ignition tray fire burned out shortly after the fire propagated to the tray above.

Even with allowances for the fuel consumed in the ignition tray for the first test, it would appear that a fire barrier with a relatively low rating (about 15 minutes) would have prevented the fire propagation. Again, it should be emphasized that no suppression systems such as barriers, coatings, or sprinklers were in use.

b. Role of Cable Flame Retardancy and Safety Division Physical Separation Requirements in Plant Fire Protection

The UCS Petition of November 4, 1977 (page 5) alleges that the Sandia cable fire tests demonstrate that the cable flame retardancy provisions of IEEE-383 and the physical separation

provisions of Regulatory Guide 1.75 and IEEE-384 are not effective in minimizing the effects of fires. In his Supplemental Affidavit submitted by the UCS letter of November 10, 1977 Mr. Pollard states his belief (page 3) that the cable flame retardancy and separation standards are not adequate, even when used together with other fire detection and suppression standards (such as those of Regulatory Guide 1.120) as part of the defense-in-depth principle.*

Simply stated, the Staff disagrees with these allegations and Mr. Pollard's second supplemental affidavit, dated November 17, 1977. As we indicated in our report of November 9, 1977 (pp. 6-7), the Sandia tests confirmed the Staff position taken since the Browns Ferry fire; namely, that the Regulatory Guide 1.75 and IEEE-383 requirements, by themselves, are not sufficient to protect against exposure fires. Thus, additional fire protection measures are required by NRC. These additional measures include fire barriers between redundant safety division cable trays; fire retardant coatings on cabling; automatic fire detection systems; automatic fire extinguishing systems, such as sprinklers in plant areas of high cable density; backup fire suppression capability (fire hoses and portable extinguishers); administrative procedures and controls to minimize fire hazards due to poor housekeeping or to plant maintenance activities; and fire brigade training and drills to assure adequate response to fire emergencies.

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* It is unclear whether this remains the position of UCS in light of Mr. Pollard's oral presentation concerning the adequacy of current NRC fire protection guidelines to the Commission on December 8, 1977.

As we indicated in enclosure 1 of the Staff report of November 22, 1977 (pp. 2-7), additional fire protection measures contained in Branch Technical Position 9.5-1 and its Appendix A, and in Regulatory Guide 1.120 do not suggest that no protection is afforded by use of the IEEE-383 and Regulatory Guide 1.75 guidance. Rather, they serve to supplement and diversify protection against exposure fires, and to provide ample margin for assuring a high degree of fire protection safety at nuclear plants.

6. Basis for Continuation of Plant Operation and Licensing

The staff has previously indicated its basis for the continued operation of licensed plants pending completion of the full implementation of our current fire protection guidance in the November 9, 1977 report (pp. 8-9) and the November 22, 1977 report (pp. 12-18). This basis includes, 1) the actions taken as a result of the I&E inspections and subsequent follow-up actions by licensees; 2) the conclusion of the Browns Ferry Fire Special Review Group Report (NUREG-0050) that the probability of fires of a large and disruptive nature of the magnitude of the Browns Ferry fire is small and that "there is no need to restrict operation of nuclear power plants for public safety", and 3) improvements made since that time by licensees in fire prevention measures and fire brigade capability and training that have been noted in the plants visited to date and are expected to exist in the remaining plants, which further reduce the probability and consequences of fires.

The petitioners' technical arguments for the shutdown of operating reactors can be characterized as: If in an operating plant a single failure can render redundant safety systems inoperable, the operation of the plant presents an undue risk to public health and safety. A public comment on the UCS petition from Mr. Myron J. Miller, of Factory Mutual Research, dated November 22, 1977, raised a related question, "Does there presently exist in any operating nuclear plants a situation in which a fire could disable redundant safety systems?"

The staff has recognized since the Brown's Ferry fire that there are certain locations in some operating plants in which an unmitigated fire could affect redundant systems. Several sections of the report of the Special Review Group NUREG-0050 address this point and place the safety significance of such events in perspective. Briefly, the Special Review Group recognized that:

(1) The Browns Ferry fire induced common mode failures or redundant core cooling systems.

(2) Manual actions can restore the operability of cooling systems.

(3) Isolation of redundant safety equipment and associated cables is not fully achievable in real life.

(4) An area intended to be non-hazardous with regard to fires will not necessarily remain non-hazardous for the life of the plant.

(5) Operating plants and those under construction are in many respects similar in design to Browns Ferry and a re-evaluation is needed.

(6) For each plant, a suitable combination of measures such as electrical isolation, physical distance, barriers, resistance to combustion, and sprinkler systems should be applied.

(7) The independence of normal cooling systems from emergency cooling systems should be considered.

(8) Each of the Review Group's recommendations that is relevant to existing plants must be considered as a recommendation for possible backfitting. Whether to implement such recommendations must be decided on a plant-by-plant basis.

(9) At Browns Ferry, a fire disabled a substantial amount of core cooling equipment. However, in the absence of a loss-of-coolant accident, this equipment was not required to function. The reactors were safely shutdown and cooled.

(10) There was no radioactivity release greater than normal occurrence, and the health and safety of the public were not afrected.

As part of the reviews which are being conducted under BTP 9.5-1 and Appendix A the safety consequences of those few instances where a fire can disable redundant systems are specifically being evaluated plant-by-plant. The licensee's fire hazards analysis evaluates the consequences of a postulated non-mechanistic fire in each fire area of the plant. Both mitigated and unmitigated

fires are postulated and the effect of postulated fire damage on the capability to achieve and maintain safe shutdown is assessed. The goal of this evaluation is to determine whether there is reasonable assurance that at least one method of achieving and maintaining safe shutdown is independent of the influence of the postulated fires. Where such assurance can not be shown, the staff requires modifications to achieve that goal.

In areas of plants containing safety-related equipment, the procedures for strict control of combustible and potential ignition sources and the ability to detect and mitigate fires by suppression are being evaluated. The evaluations to date show that in most fire areas only one safety system may be affected by an unmitigated fire. They also show that in a few fire areas, such fires may affect cables of either division of redundant systems, cables of both divisions of some redundant safety systems, or cables of all systems used to achieve safe shutdown conditions. Mitigated fires in the same fire areas may affect, at most, a single division of safety systems. In those few instances where a fire could still affect the cables of all systems used to achieve safe shutdown, an alternative shutdown method may be required, as explained above.

While our evaluations are ongoing, the continued operation of plants is acceptable since the probability of a fire which would threaten public health and safety is low as concluded in the Browns Ferry Special Review Group Report (November 9, 1977 Report, p. 8). In addition, as shown in the instance of the fire at

Browns Ferry, time is available for manual actions and emergency modifications to restore the cooling system.

Moreover, there has been substantial progress since the Browns Ferry fire in reducing the potential for severe damaging fires including:

a. strict administrative controls over the handling and storage of combustibles and ignition sources in areas that contain safety-related systems;

b. modifications which have been and continue to be made to provide fire retarding, fire detection, and fire fighting capability;

c. operating procedures that have been developed by licensees to assure safe shutdown in the event of a fire; and

d. the additional modifications now being made to the operating plants to decrease the severity of a fire and increase the plant's capability to cope with an unmitigated fire.

In addition, Interim Fire Protection Technical Specifications have already been proposed for all operating plants. We expect them to be in effect shortly. These Interim Technical Specifications will cover the availability of existing fire protection systems and administrative controls, including fire brigade strength and training, and control of combustibles and ignition sources.

Consequently, for continued operation of operating plants the risk to public health and safety is acceptably low, and suspension of operation is not required.

For plants currently under licensing review, and for those plants now under construction, the staff fire protection reviews based on the guidelines of BTP 9.5-1 and Appendix A thereto, will be complete before operating licenses for such plants are issued, starting in 1978, and implementation schedules will be specified for any fire protection improvements that may be required.

7. Public Comments on UCS Petition Relative to Fire Protection

On November 11, 1977, the Commission published a notice in the Federal Register (42 FR 58803) soliciting the views of licensees and the public with respect to the UCS petition, and requesting comments by November 28, 1977. To date, forty-four responses have been received. A list of those responding is provided as Appendix C. None of the responses identified any new safety information that had not already been covered by the UCS petition and the subsequent supplemental affidavits of Mr. Pollard or not previously considered by the staff. Our review of the comments presented in these submittals indicates: a) twenty-four responses recommend denial of the petition, of these one is from an architect-

engineering firm, one is from a state agency, and the remainder are from utilities with nuclear power plants or law firms representing such utilities; b) fourteen responses recommend that the petition be granted; of these, nine are from environmental or public interest organizations and six are from private citizens; and c) five responses contain general statements or a request for additional time to prepare further comments without an accompanying recommendation on the petition. The bases presented in these public comments to support the recommendations they contain range in scope from a single statement and conclusion to a detailed discussion of the Sandia tests, with resulting positions as to their applicability to nuclear power plants.

The staff has reviewed the public comments. Specific comments pertaining to fire protection and the Sandia fire tests generally fell into four categories.

1) Discussion of what the tests demonstrated regarding cable separation and flammability criteria and the adequacy of such criteria.

2) Comparison of the test conditions with actual nuclear power plant conditions, cable tray arrangements, and credible fire sources.

3) Defense-in-depth in fire protection; degree of protection at each level, other fire protection capability in addition to cable separation and flammability limits.

4) Adequacy of NRC fire protection program; evaluation of operating plants under construction, current NRC fire protection guidance.

The comments in each of these categories are discussed below.

1) Cable Separation and Flammability Criteria

Fourteen utilities commented on the NRC cable separation and flammability criteria as related to the Sandia Test results. Basically, it was stated that: exposure fires were not considered to be a credible event; the Sandia test fire because of its severity, was not representative of any exposure fire expected to occur in a nuclear plant cable area; and the IEEE-384 separation criteria used in the Sandia test were valid with respect to internal (electrical) fires only.

Previous staff comments on these points are contained on page 15 to the Staff report of November 9, 1977 and on pages 3, 4 and 7 of enclosure 1 and pages 22-23 of enclosure 2 to the Staff report of November 22, 1977. Briefly stated, exposure fires are considered by the staff to be a low probability, yet credible, event and they are presumed in fire protection reviews. The exposure fire assumption is specifically stated in the fire hazard analysis section of Reg Guide 1.120. In any event, as stressed on page 3 of

Enclosure 1 of the staff report of Novemoer 22, 1977, compliance with Regulatory Guide 1.75 (IEEE-384) and IEEE-383 alone are not sufficient to protect against fires. Additional fire protection measures are required as identified and discussed in this report.

The severity of the Sandia fire test is addressed in this report and in pages 22-23 of the staff report of November 22, 1977, in which the staff concluded that Sandia test configuration is considered to be a conservative but not "worst case" representation of high cable concentrations at plants applying Regulatory Guide 1.75.

Several utilities stated that IEEE-383 was a good standard for performing a relative "screening" of cables for fire hazard classification and served to eliminate undesirable cables. The NRC staff concurs with this comment on IEEE-383. Footnote 2 on page 6 of Staff report of November 9, 1977 points out that Regulatory Guides 1.120 and 1.131 state that cables passing the IEEE-383 test provisions are not exempt from additional fire protection measures or considered to be qualified for any installed cable system configuration.

2) Actual Plant Conditions vs Test Conditions

Several utilities identified actual conditions in their plants that tend to prevent fires or mitigate the consequences of cable fires compared to the Sandia test conditions. These conditions include the following:

a. Administrative controls on the handling and storage of combustibles and ignition sources are such that the conditions required to produce an exposure fire in an area of high cable concentration are unlikely.

b. The circuit protection and capacity of electrical cables are such that occurrence of conditions required to electrically initiate a fire in cable concentrations is unlikely.

c. The test fire was more severe than the exposure fire that might result from a failure of the administrative controls on combustibles and ignition sources.

d. If a fire occurs in an area of high cable concentration, existing fire detection and suppression methods would mitigate the consequences of such a fire.

e. If a fire occurs in areas of high cable concentrations, the propagation of such a fire would be significantly reduced because of a variety of plant specific features, including: (1) cable configurations within trays significantly different from the test configurations; (2) cable trays covered by steel covers or bottoms, flame retardant coatings, or mineral wool blankets; (3) less cable tray fill; (4) less air space between cables; (5) significantly different cable types; (6) steel cable tray supports, rather than aluminum, preventing change in geometry due to collapse of supports; and (7) fire barriers separating redundant divisions.

f. The results of the tests are consistent with the assumptions utilized by NRC in formulating currently applicable fire protection requirements.

The staff agrees with the above comments except for c., e.3 and e.4. For these comments, the staff notes that, while proper plant administrative controls reduce the probability of exposure fires and any such fire that might occur would likely be less severe than the Sandia exposure test fire, it is appropriately conservative to consider exposure fires in the plant fire hazards analysis because they cannot be discounted. The cable tray volume percent fill and air space between cables varies significantly within a plant and from plant to plant; therefore, the Sandia test is reasonably representative of some plant cable installations.

3) Defense-in-Depth

Thirteen respondents to the UCS petition have commented on the NRC concept of defense-in-depth. The comments fall into three categories.

a. The respondents disagree with the UCS interpretation that each level of defense-in-depth must stand alone and be "complete." The respondents agree that each level must be strong, but they do not agree that each level need (or even can) be perfect. For instance, to absolutely prevent fires, all combustible materials and possible ignition sources would have to be eliminated - a condition that is not possible in the real world.

b. The respondents state that use of physical separation and flame retardant cables are not the only methods available or used for controlling fire damage and assuring capability for safe shutdown of the plant.

c. The respondents state that installed automatic and manual fire detection and suppression capability is adequate to control and extinguish cable fires that may reasonably be expected in a nuclear power plant.

The staff addressed item a above on pages 6-8 of Enclosure 1 of the Staff report of November 22, 1977. The Staff position remains unchanged. To reiterate and emphasize that position, we quote the second paragraph of the discussion of defense-in-depth from Regulatory Guide 1.120 - Fire Protection Guidelines for Nuclear Power Plants namely, "No one of these echelons can be perfect or complete by itself. Strengthening any one can compensate in some measure for weaknesses, known, or unknown, in the others." This clearly indicates that defense-in-depth is achieved by an adequate balance between three defense levels.

The staff has already responded to item b in the staff response regarding cable separation and flammability criteria above. Regulatory Guide 1.120 clearly states that compliance with Regulatory Guide 1.75 (IEEE 384) and Regulatory Guide 1.131 (IEEE 383) alone are not sufficient to protect against fire. Additional fire protection measures are required.

With respect to item c, the staff agreees that the fire detection and suppression capability presently available in operating nuclear power plants is adequate to assure safe plant shutdown in the event of fires that may be reasonably expected in nuclear power plants. However, further improvements of the fire protection program may be necessary as determined by the continuing fire protection evaluation program for operating plants described in Section III.A.4 above.

4) NRC Fire Protection Program

a. Several comments were received from organizations recommending that operating reactors be shut down until full compliance with the regulations had been demonstrated. A number of licensees commented that they had upgraded the fire protection measures in their plants since the Browns Ferry fire and were taking additional actions to comply with the NRC Staff's review of operating plants. Other organizations commented that the NRC Staff has already initiated a systematic review of fire protection at operating plants and that this review is proceeding in a sufficient and timely manner.

b. Regarding plants under construction, or under review for construction permits, comments were received recommending that construction and licensing activities be suspended until new guidelines incorporating the results of the Sandia tests were developed and applicants had demonstrated compliance with those guidelines. A number of commentators said that the current NRC staff guidance was adequate and the evaluation of plants in the licensing process was proceeding in a timely manner. Several applicants commented that they had submitted their fire protection evaluations to the staff and would incorporate additional fire protection measures, as necessary, prior to operation of their plants.

c. Several organizations commented that current staff guidance (Branch Technical Position 9.5-1 and Regulatory Guide 1.120) required fire barrier separation of redundant cable divisions. Other comments stated that the Branch Technical Position and the Regulatory Guide contained adequate guidance for separation and defense-in-depth.

The staff position on the fire protection for plants in operation and under construction (response to items a and b above) is treated in Section III.A.6 of this report.

For item c, the staff agrees. Branch Technical Position 9.5-1 and Regulatory Guide 1.120 present guidelines for a fire protection program acceptable to the NRC Staff for new plants. Appendix A to Branch Technical Position 9.5-1 presents fire protection program guidance for operating reactors and plants under construction. The content and adequacy of this current staff guidance are summarized and discussed in detail in Section III.A.3 of this report.

B. Qualification of Connectors, Penetrations, and Other Electrical Equipment Required for Safety

1. Introduction

The UCS petition and Supplemental Affidavits* allege that the results of tests of electrical connectors at Sandia and the electrical penetration failures that occurred at Millstone Unit 2, indicate (1) a generic applicability to other similar electrical equipment and (2) doubt as to whether the quality assurance documentation audit conducted by the staff is effective for these components and other similar equipment.

The staff discussion that follows provides a summary of staff actions regarding electrical connectors and electrical penetrations, and discusses the qualification of other similar electrical equipment. Public comments on the environmental qualification aspects of the UCS petition are discussed at the end of Section III.

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*The staff in its recent actions has gone beyond the concerns initially expressed in the UCS petition with connectors and penetration assemblies required to function in the LOCA environment This expansion of the scope of interest had several facets: IE Bulletin 77-05A required licensees to look at all connectors in safety-related systems required to function in the environment of any accident they are designed to mitigate; IE Bulletin 77-06 required licensees to examine the qualifications of electrical penetrations in addition to considering the implications of the Millstone failures; and the NRR staff has addressed the state of knowledge of environmental qualifications of all safety-related electrical equipment.

2. Connectors

The bases for the UCS petition are certain test results described in a trip report written by Mr. Ron Feit of RES, dated August 5, 1977 concerning the Commission's Qualification Testing Evaluation Program and Fire Protection Research Programs, both of which are conducted for NRC by Sandia Laboratories located in New Mexico.*

The purpose of the Qualification Testing Evaluation Program is to obtain data to examine the suitability of current standards and regulatory guides for the environmental qualification testing of safety-related equipment required to operate in a loss-of-coolant accident (LOCA) environment. The tests in this program were conducted to evaluate the adequacy of testing methodology, and not to verify the qualification

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* Three interim Quick Look Reports describing the results of the ongoing qualification testing program were provided by Sandia to NRC in January, March, and July 1977, and routinely sent to the NRC Public Document Room at that time. The preliminary results of the tests were also provided to the NRC staff in the August 5, 1977 trip report, and in a memorandum dated August 26, 1977 from S. Levine (RES) to E. G. Case (NRR), et al.

of specific safety-related equipment. One specific objective is to look for synergistic effects, i.e., effects of simultaneous application of all environmental parameters.

The Sandia program on component qualification testing methodology involves testing in simulated LOCA environment, including combinations of steam, radiation, water, high pressure and temperature, and chemical additives. The equipment used in the tests was intended to be qualified according to the IEEE-323 Standard* which describes the basic principles, procedures, and methods of qualification of Class 1E (i.e., safety grade) equipment.

The electrical connectors of interest in the petition were selected for testing early in the program because their physical size was such that they could be accommodated in the available test facility and not because they were known to be used in nuclear power plant safety systems.

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* IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations, endorsed by NRC Regulatory Guide 1.89.

Three suppliers of electrical connector assemblies were contacted by Sandia and requested to provide LOCA-qualified test specimens that could be accommodated in the test facility. The electrical connector assemblies included qualified cables provided by Sandia that were attached to the connectors by the suppliers.

Sequential and simultaneous tests involving radiation, aging and LOCA thermodynamic conditions were conducted on a total of twelve connector assemblies. The test procedures called for a five-day aging cycle at 130°C to simulate a 40-year plant life followed by a 200 megarad exposure. The environmental conditions of a postulated LOCA transient were then simulated by raising the temperature and pressure to 157°C and 70 psig, respectively. These conditions were maintained for three hours with chemical additives and saturated steam. After return to ambient conditions, a second identical LOCA environment was simulated and maintained for an additional three hours. This second LOCA transient was added for conservatism. The test continued for 14 days after the second LOCA transient to simulate a return to normal ambient conditions.

The connectors under test were checked before and during the test by making electrical measurements that are indicative of the ability of the connectors to perform in safety system circuits. All twelve of the connectors tested failed at some time during the test. Seven connectors failed early in the first LOCA transient, but the other five continued to function through the first LOCA transient.

As a result of the electrical connector tests at Sandia Laboratory, the staff made preliminary efforts to determine the extent of the use of connectors in safety systems. Although it was originally believed that electrical connectors were not generally in use in safety systems inside containment, as a result of discussions in mid-August between IE, NRR, and RES, IE contacted Sandia, connector suppliers and reactor vendors in September to obtain further information on the qualification of the connectors used in the Sandia tests and their use in nuclear power plant service.

From these inquiries it was established that the connector assemblies used in the Sandia tests had not been properly qualified to IEEE Std. 323-1974 and that consequently the failures of the Sandia test connectors did not demonstrate a deficiency in the Staff's qualification criteria. It was further concluded from these inquiries that these connectors were not generally used in safety systems required to be environmentally qualified. Nonetheless, in October IE initiated the preparation of Bulletin 77-05 "Electrical Connector Assemblies" to obtain documented information regarding connectors presently in use in safety systems. The bulletin, issued on November 8, 1977, directed licensees and permit holders to provide information on connectors used in safety related systems located inside containment, subject to a LOCA environment, and required to be operable during a LOCA. A supplemental bulletin 77-05A, issued November 14, 1977, directed licensees also to provide information on all connectors in safety systems inside or outside containment, and required to function to mitigate an accident (not limited to LOCA's) where the accident itself could adversely affect the ability of the system to perform its safety function.

The UCS Petition was received on November 4, 1977, while Bulletin 77-05 was being prepared. After receipt of the UCS Petition, and prior to the issuance of the Bulletin, the Staff initiated a telephone survey to obtain early preliminary information from architect-engineering firms and nuclear steam suppliers on the extent to which connectors are being used in nuclear power plants, the type of connectors in use, and the bases available to support that, where necessary, the connectors had been qualified for an accident environment.*

The results of the staff follow-up meetings with licensees identified during the telephone survey as utilizing connectors indicated that some licensees did not have complete quality assurance data immediately available. The Office of Inspection and Enforcement has this matter under consideration and will take whatever enforcement action may be indicated when the review is completed.

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*The initial results of the telephone survey were reported to the Commission in the Staff's November 9, 1977 report and discussed further at the November 11, 1977 Commission meeting. Additional information was reported to the Commission on November 18, 1977, November 22, 1977, November 25, 1977, December 6, 1977 and was discussed at the December 8, 1977 Commission meeting.

In addition to the preliminary telephone survey, the Staff has reviewed responses to Bulletins 77-05 and 77-05A, received from the licensees of the 65 operating nuclear power plants.

Table B-1 summarizes actions taken and the current status of the nineteen plants identified in the staff survey and in response to the Bulletins as using connectors in safety-related systems inside containments. Table B-1 is different from the table shown by the staff at the December 8, 1977 briefing for the Commission as a result of additional information received from licensees in response to the bulletins. All identified connectors that are inside containment and required to function in the LOCA environment have been qualified as set forth in the Table. The responses to the bulletins have confirmed the staff's understanding that electrical connectors are also used in various safety-related systems that are not required to function in a LOCA environment. The staff is continuing to review the responses as part of its review of the entire subject of electrical equipment qualification discussed in Appendix B to this report.

TABLE B-1
ACTIONS TAKEN AND CURRENT STATUS OF 19 PLANTS
HAVING SAFETY-RELATED ELECTRICAL CONNECTORS INSIDE CONTAINMENT
AND REQUIRED TO FUNCTION IN LOCA ENVIRONMENT

December 15, 1977

PLANT	INITIAL ACTION	CURRENT CONNECTOR STATUS
D. C. Cook 1	Nov. 18 shutdown and confirmatory order	Replaced with splices qualified by tests--restart Dec. 2
Browns Ferry 1, 2, and 3	Nov. 18 letter 10-day response	Qualified by tests and analysis--confirmatory tests ongoing
Nine Mile Point 1	Nov. 18 letter 10-day response	Qualified by analysis and comparison--confirmatory tests ongoing
Oyster Creek	Nov. 23 letter 9-day response	Qualified by analysis and comparison--additional supporting information requested
Maine Yankee	Nov. 18 letter and I&E 77-05	Partially Qualified--awaiting documentation of full qualification
Surry 1, 2	Nov. 18 letter and I&E 77-05	Partially Qualified--awaiting documentation of full qualification
Oconee 1, 2, 3	Nov. 18 letter and I&E 77-05	Qualified by tests--awaiting formal documentation
Hatch 1	Nov. 18 letter and I&E 77-05	Qualified by tests--awaiting formal documentation
Ft. St. Vrain	Nov. 18 letter and I&E 77-05	Qualified by tests--awaiting formal documentation
Pilgrim 1	I&E 77-05	Qualified by tests--awaiting formal documentation
Peach Bottom 2, 3	I&E 77-05	Qualified by tests--awaiting formal documentation
Palisades	I&E 77-05	Identical to Oconee Connectors --awaiting formal documentation
Connecticut Yankee	I&E 77-05	Four connectors-replaced by terminal blocks inside qualified junction boxes

3. Electrical Penetrations

Although the Sandia tests did not involve electrical penetrations, the supplementary UCS affidavits of November 10, 1977 and November 17, 1977 questioned the qualification of penetrations on the basis of the Sandia tests, and noted recent experience with penetrations at Millstone Unit 2.

IE Bulletin 77-06 "Potential Problems with Containment Electrical Penetration Assemblies" was issued on November 22, 1977. That Bulletin was issued as the result of the electrical shorts that had occurred in penetrations at the Millstone Unit 2 facility during normal operation. The Bulletin required licensees of operating reactors to provide oral and written information regarding the use and qualification of certain containment electrical penetration assemblies.

The oral responses to the Bulletin on penetrations were reported to the Commission on December 6, 1977. Written responses to IE Bulletin 77-06 have been received from all of the operating light water reactors. (The bulletin was not issued to Ft. St. Vrain, a gas cooled reactor, because it had earlier been determined that the types of penetrations of concern were not installed in that plant).

The written responses have been consistent with the information obtained from the oral responses and the follow-up conversations related to those responses. The written responses provide no new information of significance, but do reinforce the conclusions drawn from the oral responses and indicate a need for the staff to pursue certain specific qualification information in greater detail with individual licensees.

As reported previously, the responses show differences in operating procedures with regard to maintaining nitrogen pressure on those penetrations having nitrogen addition fittings. Since the penetrations have been qualified both with and without nitrogen pressure, this difference is not surprising. However, in view of the experience with unpressurized penetrations, the staff intends to pursue this matter with individual licensees. Thirty of the sixty-four facilities report that nitrogen pressure is maintained on the penetrations, while eighteen report that normal operation does not include the nitrogen pressurization. Six of the remainder have a penetration design that does not accommodate nitrogen pressurization. The remainder were not clear as to their operating practice in this regard, with three of them reporting nitrogen was used for pressurization for leak testing but indicating that pressure was not maintained during normal operation.

Our review indicates that three facilities, at two power stations, experienced significant penetration failures in the past. Surry Units 1 and 2 reported several failures in 1972 due to undersize transition pins. These penetrations were replaced. The third failure was at Millstone Unit 2, which was the event that led to NRC's concerns in this area. The failed penetrations at Millstone Unit 2 are being replaced. Some licensees have indicated, and the staff is also aware of such events, that there have been instances of malfunctions due to leakage; however, in no case has this led to a loss of function of the penetration.

The Office of Inspection and Enforcement has requested copies of qualification test data from all of the penetration vendors, either directly or through licensees. (In some cases, the original manufacturer no longer makes these devices). This information will be reviewed when received, and any remaining questions will be resolved. Since it has been determined that qualification tests have been performed for all penetration types, the major remaining questions relate to specific test conditions.

In the December 6, 1977 report, the Staff indicated that there were three plants (Dresden 1, Yankee Rowe, and Connecticut Yankee) for which we were awaiting additional information on the qualification of their penetrations. These licensees have reported to the staff and indicated that sufficient qualification information exists to allow continued plant operation. The staff agrees.

4. <u>Other Electrical Equipment</u>

During the course of the surveys on electrical connectors and penetrations, the Staff found that there were a number of operating plants for which complete documentation of the environmental qualification tests of the electrical equipment in question was not immediately available, although other types of information were made available to the staff to provide assurance that the equipment would perform its safety function under accident conditions. The staff review of the documentation of the capabilities of these two electrical components is continuing. The surveys conducted by the staff constitute a preliminary audit of the adequacy and qualification of certain safety-related electrical equipment in operating plants, and it is the staff's belief that the findings would be similar for other electrical components in safety systems located inside containment in these plants. Further, we believe that the results of the surveys indicate that the commitments made by the licensees in their applications to qualify safety-related equipment have generally been satisfied and support our overall position that no immediate action with regard to the question of environmental qualifications of other safety-related electrical equipment in operating reactors is warranted.

Beyond the question of immediate action, the staff has considered whether the recently completed preliminary surveys regarding electrical connectors and containment electrical penetrations in operating plants should be expanded to

consider, on a longer term basis, the safety adequacy and environmental qualification of other electrical equipment in these plants. The Systematic Evaluation Program for Operating Reactors recently approved by the Commission provides a suitable framework for such an expanded effort since it already includes "Environmental Qualification of Safety-Related Equipment" as one of the topics of safety significance (Topic III-12) and further provides that topics considered to be of special safety significance may be considered on a case-by-case basis in advance of completing the overall program.

The staff has determined that it is appropriate to complete its review of this subject as the first topic of the Systematic Evaluation Program (SEP). The licensees of the eleven SEP facilities will be required to evaluate the environmental qualification of all electrical equipment they deem necessary to mitigate the consequences of design basis events. It is expected that in about three months, the staff review effort will be sufficient to assess any safety implications in sufficient detail to decide whether or not additional review of facilities other than those included in the SEP is required. The results of this staff review will be used to determine whether it is necessary to expand the effort from the eleven SEP facilities to other operating nuclear power plants. The staff's bases for this approach are set forth in its report, titled, "Staff Report on Environmental Qualification of Safety-Related Electrical Equipment" dated December 15, 1977 attached as Appendix B.

5. Summary of Public Responses to UCS Petition Concerning Qualification Tests

Forty-four letters were submitted in response to the Nuclear Regulatory Commission's request for public comments on the Union of Concerned Scientists' Petition for Emergency and Remedial Action, noticed in the Federal Register (42 FR 58803) on November 11, 1977. These letters did not provide any new information not previously considered by the NRC staff in the evaluation of the UCS petition. The comments made with respect to the environmental qualification of safety-related electrical cable connectors can be categorized into the following four general areas:

a. Comments Concerning the Applicability of Sandia Tests

The comments in support of the petition are general in nature and reiterate the statements made in the petition without presenting new information.

The comments in opposition to granting the petition made the following specific comments:

1) The majority of the plants in operation and under construction do not use connectors in safety-related systems inside reactor containment that must function in the LOCA environment. In some instances connectors are used in safety-related systems required to function in the LOCA environment; however, in most cases the connectors used are of a different design than those tested at Sandia.

The NRC staff agrees with this comment on the basis of the results to date of the survey on the use of electrical connectors in safety systems inside containment in operating plants. The results are contained in memos from E. Case, Acting Director, NRR to the Commissioners, dated November 18, 1977, November 25, 1977, and December 6, 1977, and are summarized in Section III.B.2 above.

2) The connectors tested by Sandia had not been previously qualified to withstand the LOCA environment in accordance with the test procedures specified by IEEE-323-1974 and required for safety-related systems that must remain operable in a LOCA environment.

The NRC staff addressed this point regarding the qualification of the connectors tested by Sandia in Enclosure (2) to the memo from E. G. Case, Acting Director, NRR to the Commissioners dated November 22, 1977. The staff concluded that the Sandia tested connectors were not qualified to IEEE-323-1974 as required for the LOCA environment.*

3) The improper assembly of the connectors (i.e., the mating of the connector to the cable) was a significant factor in the inability of the cable/ connector assembly to withstand the Sandia qualification environment

The NRC staff notes that the Trip Report of R. Feit dated August 5, 1977, did in fact indicate that there were some problems in the assembly of the connectors that could have led to the failure of the tested cable/connector assemblies.

b. Comments Concerning the Necessity for Connectors to Function in the LOCA Environment

The comments in support of the petition tend to view the failure of the electrical cable connectors in the Sandia tests as indicative that all nuclear power

*A response to the Commission's request for comments on the UCS petition from ITT-Cannon Electric Division, dated December 13, 1977 confirms that the Cannon connectors usd in the Sandia test were not qualified and states, "At no time was it stated that Class IE qualification status was necessary for components in this test program. In addition, no Q.A. documentation was requested either verbally or in writing to support qualification status. As a result, we shipped Sandia off-the-shelf connectors designed and qualified to Military Specification MIL-C-83723 (a non-nuclear aerospace specification)., These connectors were not qualified to IEEE 323 by Cannon. There are no design or fabrication problems with these connectors when used in accordance with the Military Specifications to which they were designed. The connectors/cable assemblies subsequently failed the test, as did assemblies from other manufacturers."

plants are unsafe in that the safety-related accident mitigating systems would not function due to failure of electrical cable connectors.

Comments in opposition to the petition note that the use of electrical cable connectors is plant specific and must be evaluated on an individual case-by-case basis.

The NRC staff believes that blanket extension of the Sandia test results to all safety-related accident mitigating systems in all power plants is not appropriate because the Sandia connectors were not properly qualified for use in nuclear plants. The staff has evaluated the use of electrical cable connectors on a case-by-case basis.

c. Comments Concerning the Qualification of Electrical Cable Connectors in Operating Plants and Plants Under Construction

Commentators in support of the petition raised a question as to whether electrical cable connectors that were used inside containment for applications where such components must function in a LOCA environment were properly qualified in presently operating plants and plants under construction.

The qualification of electrical cable connectors in the 65 operating plants has been addressed by the NRC survey on use of electrical connectors in safety systems inside containment in operating plants, as discussed above. The qualification data for plants under construction will be reviewed in accordance with the staff's Standard Review Plan as part of the reviews for operating licenses.

d. <u>Comments Concerning the Validity of Qualification of Class 1E Equipment to IEEE-323</u>

Comments in support of the petition indicated that the failure of the electrical cable connectors thought to be qualified to IEEE-323 was indicative of a deficiency in the IEEE-323 qualification standard.

Commentators opposed to the petition indicated that the Sandia tests do not reflect on the IEEE-323 document since the connectors tested were not qualified to the IEEE-323 Standard for the LOCA environment.

The NRC staff does not consider the results of the Sandia tests on electrical cable connectors to be indicative of any inadequacies in the IEEE-323 qualification document since the equipment tested was not qualified to the IEEE-323 Standard for the LOCA environment. Data sufficient to demonstrate environmental qualification are required for all safety-related equipment that must be functional in the LOCA environment.

IV. CONFORMANCE WITH COMMISSION REGULATIONS

The UCS petition, and the supplemental materials filed by UCS, assert that the Sandia tests demonstrate that nuclear power plants now operating or under construction do not meet applicable Commission regulations. Specifically, UCS asserts that such plants do not conform to GDC-3 which deals with fire protection, GDC-4 which deals with environmental qualification, with 10 CFR 50.55a(h) as it relates to single failure requirements and with other "single failure" requirement provisions of the General Design Criteria.

A. General Background on Relationship Between Regulations and Staff Guides

The basic Commission regulations establishing design and performance goals for nuclear power plant safety are contained, in the main, in the General Design Criteria in 10 CFR Part 50, Appendix A. These performance standards are generally cast in broad general terms as to the goal to be satisfied by various structures, systems and components important to safety.

Specific methods for implementing the GDC requirements have been developed over the course of time as the licensing process has progressed from its early years. For older plants, the design and performance standards were evaluated in all instances on an ad hoc case-by-case basis.* Starting in 1969, the Staff began development of

*Although the GDC were promulgated as part of 10 CFR Part 50 in 1971 the basic safety considerations embodied in the GDC had been in general use from the early 1960's.

Regulatory Guides (originally called Safety Guides). Today, there is a panoply of regulatory guides, standard format and content guides, and Standard Review Plan provisions, along with associated Branch Technical Positions (BTP's), all aimed at providing specific guidance as to acceptable methods of implementing various facets of the general design and performance goals of the GDC.

These guidelines describe acceptable methods for implementing the General Design Criteria. Applicants are permitted to propose other methods (e.g., by design, tests and administrative procedures), of demonstrating acceptable compliance with the General Design Criteria. As new guidelines and staff positions have evolved, however, there has been a general trend for them to address more topics and be more comprehensive. A result is that new applicants find the staff is looking deeper and imposing more specific limitations and more specific functional requirements than it did previously.

The regulatory guides and staff positions are intended to establish reasonable and uniform methods which will satisfy specific aspects of the particular requirements. However, they are not directed toward identifying minimum methods to achieve compliance with the regulatory requirement. Rather, a system or component designed to satisfy an applicable regulatory guide or staff position will satisfy the related aspect of the GDC with margin. The Staff prefers the use of methods described in regulatory guides and staff positions to achieve a more effective and efficient review

process which also provides a degree of uniformity in attention given by industry to specific safety considerations. There is generally a margin, and in some cases a wide margin, between the methods that satisfy a regulatory guide that is generally applicable to all plants or to a class of plants and those methods which could be considered the minimum needed to satisfy a GDC requirement for a specific plant.

Thus, while compliance with regulatory guides is not mandatory, such compliance provides safety margins above the minimum requirements of the applicable GDC. As a corrollary, non-conformance with regulatory guides does not necessarily result in failure to meet applicable regulations or inadequate safety.

B. General Design Criterion 3

The Commission's basic requirement for fire protection for nuclear power facilities is set forth in General Design Criterion 3. In summary, General Design Criterion 3 requires:

1. Structures, systems and components important to safety shall be designed and located to minimize, consistent with other safety requirements ... the probability and effect of fires and explosions;

2. Non-combustible and heat resistant materials shall be used wherever practicable throughout the unit ...;

3. Fire detection and fighting systems of appropriate capacity and capability shall be provided and shall be designed to minimize adverse effects of fires ...;

4. Fire fighting systems shall be designed to assure that their rupture or inadvertent operation does not significantly impair the safety capability of these structures, systems and components.

Prior to the Browns Ferry fire, these safety objectives of General Design Criterion 3 were implemented on a case-by-case basis. No systematic or comprehensive consideration of fire protection was given by the staff in its safety review of facility applications comparable to present practices. In general, the basic fire resistant characteristics inherent in the concrete and steel structures of nuclear power facilities, when considered together with the use of electrical systems conforming to Underwriters Laboratory and National Electric Manufacturer's Association requirements, were generally believed to provide a degree of intrinsic fire resistance such that no additional or special fire protection requirements were necessary for nuclear power facilities. The staff safety review consisted principally of verifying information given in SAR's that good industrial fire protection and fire fighting practices and equipment were available at nuclear power facilities. This followed the general requirements given in the Standard Format Section 9.5.1.

The experience of the Browns Ferry fire in 1975, however, unquestionably changed the direction of the staff review. Following the Special Review Group study of the Browns Ferry fire, the staff developed a comprehensive plan for review of nuclear power facilities to assure the capability of detecting, preventing, fighting, and withstanding effects of serious fires so that the experience of Browns Ferry would not be repeated. These actions have already been discussed at length in Section III.A of this report.

The Staff believes that the general goals expressed in General Design Criterion 3 must properly be read in light of the available knowledge

at that time. After the Browns Ferry fire, the Commission might have modified the regulations to improve more specific fire protection requirements. We believe, however, that this was not necessary; the staff developed a program to substantially enhance fire protection at nuclear plants based on Branch Technical Position 9.5-1 and its Appendix A. This program includes requirements on fire protection systems to assure that no more than one electrical division in the plant can be lost as a result of fire and details of administrative control. We believe that this program provides substantial protection beyond the minimum that might be deemed to satisfy GDC-3. As noted in Section III.A, the NRC has embarked upon a program to assure that fire protection programs at nuclear power plants will conform to this guidance at the earliest practicable date.

For the period until these comprehensive fire protection programs can be implemented, the NRC staff has worked closely with licensees to reduce substantially the potential for serious fire by assuring careful control of combustible materials and sources of flame and by improving the capability of fire fighting equipment and personnel.

For the interim period in which they will be applicable, these measures will satisfy in a minimum fashion the general goals of GDC-3, particularly in light of the very careful attention presently being given to administrative control providing for good "house-keeping" to eliminate potential fire

sources and to control sources of flame (e.g., welding and cutting practices). For the long term, however, the sta*f position is that there be an appropriate combination of permanent "built-in" features providing fire retardancy, barriers, fire detection and fire suspension, as called for by Branch Technical Position 9.5-1 and its Appendix A.

In summary, the NRC's program for implementation of fire protection at nuclear power facilities has two phases: an interim period with an adequate degree of fire protection based on careful administrative controls over potential sources of fire, followed, as promptly as practicable, by the implementation of a comprehensive program of fire protection with extensive use of permanent "built-in" fire prevention and fire suppression systems.

The cable fire test results at Sandia do not significantly affect current knowledge on needed fire protection measures for nuclear plants for the reasons given in Section III.A. Indeed, the ongoing Sandia test program may well provide additional information, that may enable us to develop a wider range of alternative methods to achieve the goals of General Design Criterion 3; that is, to assure that fires at nuclear power facilities do not become a source of radiological danger to public health and safety as a result of adverse effects.

C. <u>General Design Criterion 4</u>

The basic Commission requirement concerning capability of safety systems to accommodate the environmental effects of postulated accidents is GDC-4 which requires that structures, systems, and components be:

- designed to accommodate "... the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents."
- appropriately protected against dynamic effects, including the effects of missiles, pipe whipping and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit.

This requirement is complemented by the general provision of Criterion 1 that such systems be designed and tested to quality standards commensurate with their importance to safety.

The general standards of Criterion 1 have been comprehensively implemented by Appendix B to 10 CFR Part 50 which specifies with particularity the characteristics of QA programs required to assure that systems important to safety have indeed been designed, fabricated, erected, and tested to high quality standards. Together these requirements form the basis for a comprehensive program to assure adequate performance of safety systems under conditions of normal operation, maintenance, testing and postulated accidents.

To the extent that a system or component important to safety is not capable of accommodating the effects of postulated accidents, it does not conform to GDC-4. On the other hand, there may be components capable of accommodating such effects whose ability to withstand environmental conditions under which they are intended to operate has not been adequately demonstrated by methods conforming to the requirements of Appendix B. This would demonstrate, not a failure to satisfy GDC-4, but a failure to conform to the quality assurance requirements of GDC-1, or a failure to conform with a particular provision of Appendix B (e.g., III, VI, and XI).

If the connectors tested at Sandia were representative of components in actual use in safety systems in a nuclear power plant, and if the test conditions were actually representative of the conditions associated with normal operation, maintenance, testing or postulated accidents, including loss-of-coolant accidents, then the failure reported by the tests would appear to be a deviation from GDC-4. The word "appear" is used to indicate it would be necessary to ascertain whether the system was in fact capable of performing its function before the failure occurred and to consider the extent of margin, if any, which exists.

The staff reviews of operating plants conducted in the course of responding to this petition have indicated that qualified electrical components, capable of functioning when required during accidents, apparently are in use. Thus, no failure to conform with GDC-4 has been found. Followup actions are in progress, as described elsewhere in this report, to confirm these indications. However, there have been some cases of inadequate documentary support and some other questionable quality assurance practices with respect to matters of qualification of equipment. These matters are being evaluated for appropriate enforcement action, if indicated, such as whether additional effort should be required in the area of quality assurance practices by specific licensees.

D. Single Failure Requirements

The UCS petition asserts that the exposure cable fire in the Sandia test was able to propagate across redundant divisions and indicates not only a failure to conform to the General Design Criterion 3 but also a

failure to satisfy the single failure criterion of 10 CFR 50.55a(h) requirements for electrical systems contained in IEEE-279. This standard is incorporated by reference into the Commission's regulations in 10 CFR 50.55a(h). The petition also refers to the single failure requirements of General Design Criteria 17 and 21. With regard to the concern on equipment qualification, the UCS petition contends that connector and penetrations could fail if not properly qualified. The petition asserts that this would also constitute a failure to satisfy the single failure criterion of 10 CFR 50.55a(h). The thrust of the UCS assertion appears to be that these single failure requirements for electrical systems constitute a basis separate from GDC 3 and GDC 4 for imposition of fire protection and environmental qualification requirements.

The single failure requirements for electrical systems do not establish an independent set of design basis events. Rather, they establish standards for design and performance of electrical systems to assure that such systems are capable of responding if various design bases events should occur.

In order to determine the adequacy of an electrical system, one must first establish the various overall postulated design basis events to which such systems must respond. Guidance as to the selection of these design basis events is contained in the overall requirements of, inter alia, GDC-5 of 10 CFR Part 50. The application of this guidance results in the selection of various design basis events customarily used in PSARs and FSARs: e.g., design basis earthquakes, floods, tornadoes and hurricanes;

loss of coolant accidents; main steam line breaks; rod ejection and rod withdrawal accidents as well as anticipated operational transients. After the various design basis events are selected, protection systems must be provided to assure capability to respond safely under the various postulated conditions that may result from the design basis events.

In the analysis to determine if a particular electrical, instrumentation or control system meets single failure requirements,the particular design basis event or accident is postulated to occur, along with any related consequential failure that could result from it. Then, the analysis assumes the presence of all identifiable failures that cannot be detected or tested, or which are not in fact subject to surveillance tests as set forth in the Technical Specifications. Finally, the presence of a most adverse single additional failure is assumed in assessing the capability of the system to provide the necessary pretection for the design basis event. It is the last of these steps that is mandated by the various "single failure" requirements.* The first of these steps is mandated by GDC-1-5.

For fires, the requirements to minimize potential for and effects of fires is established by GDC-3. The credible occurrence of cross-divisional fires must be minimized by appropriate combinations or separation, re-tardancy, barriers, fire suppression and fire fighting capability, i.e., defense-in-depth. In instances where cross-divisional fires remain a

*For a more complete discussion of the single failure criteria see Appendix D.

concern, alternative safe shutdown methods or system must be provided which are not subject to damage by any single fire sequence. These are set forth in Appendix A to BTP 9.5-1.

The single failure requirements do not require the postulation of, or protection against, a fire threat more severe than those encompassed by the general fire protection goals of GDC-3.

For electrical equipment qualification, GDC-4 imposes the requirement that such equipment be designed to accommodate the environmental conditions associated with the postulated design basis accident conditions under which it must function. The single failure requirements do not require the imposition of standards more severe than those mandated by GDC-4. Rather, they require that the equipment provided to respond to such events have sufficient redundancy to be capable of providing the required responses despite the occurrence of another single failure. As discussed above, the surveys conducted by the Staff indicate that equipment capable of withstanding such environment is generally in use.

The Sandia tests do not bear upon consideration of "single failure" requirements but rather on the basic question of conformance with overall design goals of GDC-3 and GDC-4. The Staff believes, for the reasons discussed above, that operating facilities satisfy these requirements and that staff review requirements for plants under construction and those under licensing review will assure that such facilities are designed to conform to these requirments.

E. Available Remedial Actions

The Staff report to the Commission of November 18, 1977 contained a substantial legal discussion of the remedies available to the Commission should a violation of Commission regulations be discovered. UCS responded to the Staff's legal position in a memorandum dated November 23, 1977. The Staff believes that in the third filing by UCS, its prior assertions have become tempered and are, in general, now consistent with respect to legal principles with that expressed by the Staff: viz, violation of a regulation does not *ipso facto* result in a requirement that a license or construction permit be suspended; but if there is reason to believe that the public health and safety is threatened as a result of a discovered violation of a regulation, remedial action must be taken. A wide range of remedial actions are available to the Commission, including the shutdown of a reactor, if necessary.

Two areas of controversy remain. First, UCS and the Staff disagree as to whether reactor licensees are currently in non-compliance with Commission regulations. The Staff has addressed this question above, and has concluded that the Commission regulations cited by UCS have not been violated. (Questions concerning possible violation of quality assurance requirments are currently being evaluated). Secondly, and we think much more basic to UCS' petition, is a factual dispute with the Staff as to whether the fire protection program now being implemented by the Staff, and the methodology used for environmental qualification

of reactor components, are sufficient to safeguard the public health and safety. These factual disagreements are really the crux of UCS' petition, and have been fully addressed by the Staff, both in previous submissions to the Commission, and in previous sections of this report.

APPENDIX A

DETAILED CHRONOLOGY OF CORRESPONDENCE RELATED TO UCS PETITION

November 4, 1977	Letter from Ellyn R. Weiss, Attorney for the Union of Concerned Scientists (UCS) to the NRC Commissioners containing a filing of a Petition for Emergency and Remedial Action by the Union of Concerned Scientists. Petition alleges that information from recent Sandia tests, which was withheld from Licensing Boards, establishes that safety equipment may fail to operate because (i) certain connectors in safety related systems cannot withstand a LOCA environment, and (ii) fires can destroy redundant electrical cables. Four attachments to the UCS Petition and an affidavit of Robert Pollard are submitted as the basis to support the allegation that NRC regulations are being violated and, the public health and safety cannot be assured until all reactors are shutdown, all plant construction is ceased, and research programs are accelerated.
November 8, 1977	Memorandum from N. Moseley, Director, Division of Reactor Construction Inspection, Office of Inspection and Enforcement to the five NRC Regional Directors requesting IE Bulletin No. 77-05, Electrical Connector Assemblies, be dispatched to all facilities with an operating license or construction permit. Bulletin requests information be submitted to NRC indicating the use of connectors affected by a LOCA and the documentation supporting their qualification. Attachments to Bulletin 77-05 contain a description of connector test equipment, test scope and test results of the Sandia tests.

November 9, 1977	Memorandum from E. Case to NRC Commissioners stating that UCS request for immediate suspension of issuance of all licensing and the shutting down and stopping of construction on all plants is not warranted. Attachment "NRC Staff Report On the Question of Whether the Petition of the Union of Concerned Scientists Raises Matters that Require Immediate Commission Action" dated 11/9/77, contains staff evaluation of Sandia tests and staff basis for rejecting UCS Petition.
November 10, 1977	Letter from Ellyn R. Weiss of the Union of Concerned Scientists to the NRC Commissioners responding to the staff's November 9 report to the Commissioners. An enclosure entitled "Supplemental Affidavit of Robert D. Pollard In Support of the Union of Concerned Scientists' Petition for Emergency and Remedial Action" provides a refutation of the staff's evaluation of the safety significance of the cable fire and connector tests and expands the concern to containment electrical penetration asssemblies.
November 10, 1977	Memorandum from K. Pedersen, OPE, J. Nelson, OGC to the NRC Commissioners containing a list of questions which the Commission might want to pursue with the staff at the meeting scheduled for November 11 with respect to electrical connectors and cable fires.
November 11, 1977	Staff Briefing of the Commission in open meeting.
November 14, 1977	IE Bulletin 77-05A extends the scope of IE Bulletin 77-05A to include all connectors in safety systems which could be affected by accidents other than LOCAs and to locations outside containment.

November 17, 1977	Letter from Ellyn R. Weiss of the Union of Concerned Scientists to the NRC Commissioners with an enclosure "Second Supplemental Affidavit of Robert D. Pollard in Support of the Union of Concerned Scientists petition for Emergency and Remedial Action" that responds in particular to the Staff's comments made at the Commission's November 11 public briefing. UCS requested that Mr. Pollard be permitted to participate on an equal basis with the staff at future Commission briefings, and stated again that equipment such as electrical penetrations, cables and cable terminations would fail in a LOCA.
November 18, 1977	Memorandum from E. Case, NRR to the NRC Commissioners responding to the Commission's request for (i) the results of the latest surveys on the use of electrical connectors, (ii) OELD's position on UCS's letter of November 10, 1977 and (iii) staff's responses to the November 10 memorandum from OGC and OPE. Enclosures to this memorandum include a Commission order to D. C. Cook Unit 1 to shut down until unqualified connectors are replaced, a summary of the connector survey and OELD's discussion on the interpretation of the Commission's responsibility with the law.
November 22, 1977	IE Bulletin 77-06, "Potential Problems with Containment Electrical Penetration Assemblies" requests information be submitted to NRC on a) whether GE Series 100 type containment electrical penetrations or similar type are used in safety related equipnent, b) have electrical failures been experienced and c) whether a nitrogen pressure is maintained.
November 22, 1977	Memorandum from E. Case, NRR, to the NRC Commissioners containing the staff's responses to UCS's letter of November 10, 1977, and to the questions raised in the November 10, 1977 memorandum from OGC and OPE. Copies of letters to selected licensees requested further connector information are enclosed. Memorandum indicates that additional plants e.g. BWR's having Target Rock valves have been found to contain connectors since the staff's November 18 preliminary survey was issued.

November 23, 1977	Letter from Ellyn R. Weiss of the Union of Concerned Scientists to the NRC Commissioners enclosing a legal memorandum in response to the OELD memorandum of November 18, 1977.
November 25, 1977	Memorandum from E. Case to the NRC Commissioners supplements the staff's November 18, 1977 report relating to the use of electrical connectors at Oyster Creek, and BWRs with Target Rock Valves. Enclosure contains IE Bulletin 77-06 which requests information from operating reactors by 12/5/77 on the use of electrical penetration assemblies.
December 6, 1977	Memorandum from E. Case to the NRC Commissioners supplements the Staff's November 18 and 25, 1977 reports, specifically addressing a) staff's preliminary survey on the use of connectors in operating plants, b) staff's preliminary survey on the use of containment electrical penetrations in operating plants, c) public comments on the UCS petition, d) R. Pollard's second supplemental affidavit, and e) summary of staff present and future actions with regard to equipment qualification.
December 8, 1977	Staff briefing of the Commission in open meeting. Mr. R. Pollard of the Union of Concerned Scientists, and Mr. K. Ellis of Conner-Moore also made presentations to the Commission.

APPENDIX B

STAFF REPORT ON THE ENVIRONMENTAL QUALIFICATION OF SAFETY-RELATED ELECTRICAL EQUIPMENT

December 15, 1977

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ENVIRONMENTAL QUALIFICATION OF SAFETY-RELATED ELECTRICAL EQUIPMENT

1.0 Introduction

The current NRC safety review process for nuclear power plants includes criteria related to the qualification of certain electrical equipment. These criteria require that electrical equipment important to safety must be qualified to function in the environment that might result from various accident conditions. Although such criteria have been applied since the early days of commercial nuclear power, the details of these criteria have been changed over the years. The evolution of environmental qualification of safety-related electrical equipment is described in Appendix A.

Changes to these criteria have raised some questions as to:

(1) the degree to which electrical equipment used in older plant designs (those operating) is capable of withstanding the environmental conditions (pressure, temperature, humidity, steam, chemicals, vibration, and radiation) of various accident conditions under which it must function (i.e., the "qualification of equipment" in these older plants), and

(2) the adequacy of tests or analyses conducted for equipment used in newer plants to "qualify" such equipment as capable of withstanding the conditions of the environment created by various accidents during which the equipment must function (i.e., the "adequacy" of qualification tests).

As noted in Appendix A, both of these items are the subjects of ongoing generic programs of the NRR staff. The first subject, the "qualification of equipment review", and the safety of operating plants is the principal subject of this status report. This subject will be discussed in Section 2.0.

The second item, the "adequacy" of qualification tests, is the subject of two NRR Category A Technical Activities. An allied concern, whether tests for certain parameters (temperature, pressure, humidity, steam and chemicals) and radiation which are conducted sequentially accurately reflect electrical component performance under accident conditions in which all conditions would be imposed simultaneously, is the subject of an ongoing NRC confirmatory research program. These subjects will be discussed in Sections 3.0 and 4.0.

For both the "qualification" and the "adequacy" issues, it is our conclusion that there is reasonable assurance that public health and safety is adequately protected during the period of time necessary to achieve their systematic, complete resolution and that no additional immediate action is warranted at this time.

The bases for this conclusion, supporting information, and our established programs for the resolution of these issues are described in later sections of this report.

2.0 Equipment Qualification - Older Plants

2.1 Technical Safety Concern

Despite the conservative design, construction and operating practices and quality assurance measures required for nuclear power plants, safety systems are installed at nuclear plants to mitigate the consequences of postulated accidents. It is the equipment associated with these safety systems that is of principal concern with respect to the environmental qualification issue. Equipment needed for normal operation and equipment needed to respond to plant transients is not of primary concern. This is because past operating experience, especially that from frequent testing requirements, has shown that safety-related equipment can perform in a normal operating environment. When problems are encountered during normal operation they are easily identified and corrected. Additionally, the principal safety systems required to function during anticipated transients (those events that are likely to occur during a plant's lifetime) would not be subjected to environmental conditions as severe as those of postulated accidents before performing their safety function.

The postulated accidents that have been identified as creating severe environmental conditions inside of containment are breaks of high energy pipes. The most limiting of these accidents are

the loss of coolant accident (LOCA) and main steam line break (MSLB). In each of these cases, hot pressurized water and steam can create a high temperature environment (250 to 400°F) at high humidity (including steam) and pressure (as high as about 50 psig). For some applications, chemicals are included in sprays that are used to reduce the pressure in the containment. Additionally, some electrical equipment is predicted to be submerged following the postulated accident. Therefore, for these applications, chemicals and submergence are included in the environmental qualification in addition to the temperature, radiation, humidity and pressure conditions.

While such accidents are judged to have a low likelihood of occurrence, less than once per thousand reactor years for smaller pipe breaks, and much less for the large design basis breaks, the NRC has required that safety systems, principally the ECCS and containment isolation and cleanup systems, be environmentally qualified to mitigate these accidents. For PWRs, the ECCS is often supplemented with additional instrumentation and controls to isolate a steam generator with broken pipes to mitigate the consequences of a postulated MSLB. To assure that these systems would perform their required function, the NRC has required not only redundancy in this equipment, but that it be designed with the capability to perform in the environment associated with such an accident.

The electrical equipment of concern during postulated accident conditions includes (1) the instrumentation needed to initiate the safety systems and provide diagnostic information to the plant operators (e.g., electrical penetrations into containment, any electrical connectors to cabling which transmits signals, and the instruments themselves), and (2) control power to motor operators for certain valves (e.g., ECCS and containment isolation valves located inside containment). Because of the Commission's requirements for redundancy of electrical equipment and "active" components (e.g., those valves that must change position), nuclear power plants have backup instruments, penetrations, connectors, cables, control devices and valves in addition to the primary safety system components. Therefore, to prevent a safety system from adequately performing its function, a substantial amount of equipment, both the primary and backup, must fail due to environmental conditions. In some cases the redundant equipment is outside containment and not subject to the same hostile environmental conditions as the primary equipment (e.g. containment isolation valves).

2.2 Operating Experience

Several events have occurred at operating nuclear power plants which provided a level of independent verification of the environmental qualification of safety related equipment. Each

of these events involved a release of steam to the containment. Although the radiological releases were negligible, sufficient steam was released to create adverse enviornmental conditions within containment. The most severe event of this type involved an inadvertent discharge of steam through primary system safety valves directly into containment at Dresden Nuclear Power Station.

2.2.1 Boiling Water Reactors

On June 5, 1970 an unexpected reactor scram at Dresden Unit 2 was followed by a main steam line safety valve discharge to the containment. About 250,000 pounds of primary coolant were released to containment causing an increase in the primary containment (drywell) pressure to an estimated maximum value of 20 psig at 320°F. Although some equipment damage did occur during the event, the ECCS and containment isolation systems functioned properly.

On December 8, 1971, a similar event occurred at Dresden Unit 3. In this case the containment reached a maximum pressure of 20 psig and a maximum temperature of 295°F. Some equipment damage occurred, although much less than at the earlier Dresden Unit 2 event. Again, all systems necessary for safety remained operable.

Steam discharge events of less severe consequences have occurred at several other BWR facilities. In these occurrences the containment was only slightly pressurized and temperatures remained below 200°F. No damage to any electrical systems was reported for any of these events.

2.2.2 Pressurized Water Reactors

Several events have also occurred at PWR facilities which resulted in steam release to containment.

On September 24, 1977, an event occurred at Davis-Besse Unit 1 which resulted in a partial depressurization of the primary system and the release of about 11,000 gallons of water in the form of steam into containment through the pressurizer quench tank rupture disk. The steam release began about 4.5 minutes after the reactor was tripped and was terminated about 20 minutes after the trip. The only equipment damage occurred within the vicinity of steam discharge. No equipment damage occurred in safety systems needed to mitigate the event.

On November 13, 1973, following a reactor trip at Indian Point Unit 2, a break occurred in the feedwater line to a

steam generator. The break was attributed to water-hammer. During the event, containment temperature reached a maximum of 110°F and relative humidity reached a maximum of about 50%. No containment pressurization was reported. One cable tray was partially submerged in water. However, the cable within this tray was designed and tested for submersion. No adverse effects on this cable or on any other electrical equipment in containment were found.

On May 1, 1975, while H. B. Robinson Unit 2 was in hot shutdown condition, a failure of a reactor coolant pump seal resulted in discharge of about 130,000 gallons of reactor coolant to the containment floor. The containment was pressurized to a maximum of 3 psig with the maximum temperature estimated to be about 140 to 150°F. High humidity conditions within containment remained for several hours. No adverse effects on electrical equipment were reported.

Other releases of high or moderate energy coolant into containment have occurred at other PWRs. However, no pressure rise or equipment damage resulting from an adverse environment were reported.

2.3 Qualification of Equipment in Older Plants

2.3.1 Background

For older nuclear power plants (those licensed for operation prior to about 1967) specific nuclear environmental qualification requirements had not been established in the industry or by the Commission. Licensing conclusions at the time were based upon overall knowledge of the nature of systems and components used and on the types of accidents of interest, and upon the awareness that nuclear components, including electrical system components were, and are, of high industrial quality. At the time, design and purchase specifications for electrical equipment adhered to applicable industry standards, such as the National Electrical Manufacturers Association Standards and existing IEEE Standards (e.g., IEEE Std. 98-1957). In addition, applicants referenced various testing programs, such as those conducted at the Franklin Institute Research Laboratories and those conducted in the AEC's Naval Reactors Program in support of their plant designs.

While these standards are not uniformly comparable to the more specific criteria currently used for nuclear facilities, they nonetheless are of high quality.

2.3.2 Previous Backfitting

The staff has continued to review the degree to which licensed operating reactors comply with NRC regulations as significant new safety information becomes available or new regulations are established. This effort, generally termed upgrading or "backfitting", has had the effect of increasing the degree to which older operating reactors have improved the documentation of environmental qualification of safety systems. Generally, this is accomplished by staff discussions convincing licensees of the desirability of plant upgrading. However, the NRC has directed licensees to upgrade systems in the past. In one case, that of Dresden Unit 1, on June 23, 1976, the staff ordered that the reactor protection system be upgraded to meet IEEE Std. 279-1968 and that other safety related equipment not previously environmentally qualified be qualified or replaced. Dresden Unit 1 is the oldest U. S. operating reactor (licensed in 1959) and for that reason documentation of conformance to current licensing requirements including 10 CFR Part 50, Appendix A (GDC 4) and Appendix B which were issued in 1971 is lacking.

A second, more extensive effort, involved the review of operating reactors to the new ECCS requirements issued

in 1974 (10 CFR Part 50.46 and Appendix K). In conducting these reviews the staff audited the degree to which certain operating reactors met requirements for environmental qualification of ECCS equipment. For those plants reviewed in detail, certain areas were discovered as not meeting licensing criteria for equipment qualification. Most of these deficiencies involved those plants licensed prior to the introduction of the General Design Criteria (Appendix A to 10 CFR Part 50). These plants were licensed before systems to mitigate the consequences of a LOCA were required. As would be expected, some of the safety-related equipment in these older facilities was found not be qualified for LOCA environmental conditions. Environmental qualification can be established by either suitable analyses or tests on equipment to substantiate that the equipment can withstand the postulated environmental conditions. Licensees were required to upgrade existing systems and components in their ECCS in order to ensure reliable performance if subjected to adverse LOCA environmental conditions. Several examples of these staff reviews are discussed below.

A case in point is the LaCrosse boiling water reactor which was licensed in 1967. Extensive modifications were required to satisfy the ECCS requirements. An integral part of the staff review was the environmental qualification of safety related equipment. Solenoid operated valves in the ECCS were provided with uninterruptible power sources, the solenoids were housed in water free junction boxes connected by sealed mineral insulated cables. Environmentally qualified valves were added in series to existing valves where qualification had not been established. Analyses were performed and tests were conducted to verify that electrical equipment inside containment would operate in an accident environment. Containment electrical penetrations were reviewed and found to be capable of withstanding temperature, pressure, radiation, chemical attack, and submergence in excess of that which is postulated for the Design Basis Event.

In May 1976, Consumers Power Company was granted, by Order from the NRC, an exemption to a portion of the FCCS criteria set forth in 10 CFR 50.46 Appendix K for Big Rock Point (licensed in 1962). The order set forth several conditions which had to be satisfied prior to the Big Rock Point facility's return to operation. Condition 2.(iii) stated that prior to further operation of Big Rock Point, Consumers Power Company shall:

Protect the controls, indication and annunciation circuitry associated with the ECCS, including the core spray valves, against the consequences of flooding following a LOCA which affects the ability of the ECCS or plant operator to take corrective action during the course of a LOCA.

During the course of the exemption review the staff and licensee identified other areas where environmental qualification including submergence was suspect. Corrective measures (relocation of equipment, augmented procedures, etc.) proposed for environmental qualification were found acceptable.

San Onofre (licensed in 1967) installed a new onsite power system in order to comply with the ECCS requirements. This modification included diesel generators, circuitry, and associated switch gear. All of which were designed and implemented in accordance with the IEEE-279 standard and are required to be environmentally qualified. In addition to these major modifications, in 1974 San Onofre proposed an environmental qualification program which has been implemented and approved by the NRC staff. ECCS reviews presently ongoing at San Onofre involve again the environmental qualifications of equipment related to safety.

2.3.3 Connectors and Penetrations

As a result of the recent staff surveys of environmental qualifications for electrical connectors and penetration assemblies, the staff has concluded that operating plants are safe but questions related to documentation remain. In the case of electrical connectors, as part of the staff's followup of experimental results, a staff survey showed that there were several cases in which a licensee asserted that components were qualified but adequate documentation of qualification was unavailable. In only one case[1/] was there an absence of both documentation and other evidence of capability to perform in the LOCA environmental conditions.

As a result of certain operating experience with electrical penetrations, the staff conducted a survey of all licensees since all plants have these components. In this case, all operating facilities could provide some assurance that their penetrations had the capability to perform in the LOCA environment, although several did not have adequate documentation of qualification. Our experience in following through with these instances of questionable qualification indicates that

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1/ In the one facility (D. C. Cook Unit 1) where connectors were used without adequate documentation of qualification, there was a temporary period during which the licensee could not assure itself and the NRC that there was reasonable basis for concluding that the systems could nevertheless safely function. Because of this uncertainty, and after discussions with the NRC, the licensee agreed to shut down the facility while connectors could be replaced with qualified splices.

there are sufficient alternatives to assure safety while the qualifications of the components are established.

It is noteworthy that, although these surveys addressed only two types of components that need to function in accident environments, the survey results do not indicate that unqualified components are in widespread use in operating reactors.

2.3.4 Cabling

Wire and cable manufacturers have been aware for many years of the potentially harsh and extreme environments to which electrical circuits may be exposed. Cable has been developed and classified not only on the basis of its electrical capabilities but also for its mechanical integrity, i.e., physical strength and ability to survive extreme temperature, moisture, steam, harsh chemicals, and to varying degrees, fires. Of course the adequacy of cabling for a particular environment is predicated upon the selection of cable with the properties required to survive that environment. When cable is manufactured, not all

electrical and mechanical attributes can be practically incorporated into one cable design. For example, the very best in electrical properties may not allow the ultimate in mechanical strength or fire resistant properties to be attained. Compromises are made and designs include the important design features and properties to ensure a high reliability electrical insulation resistance system for a given installation.

IEEE Std. 98-1957 outlines the preparation of test procedures for determining experimentally the probable life of insulating materials and systems. Test procedures for specific materials and systems incorporating insulating materials are given in a number of other older and updated versions of IEEE publications. Insulation life test procedures are a part of ASTM (American Society for Testing Materials) and NEMA (National Electrical Manufacturers Association) standards and proposed procedures and constitute a frequent and continuing part of current technical literature.

Military and commercial nuclear applications of cabling have required the additional consideration of the effects of exposure to radiation as well as other environmental conditions These environmental effects were studied on insulating materials, as well as insulating systems which are a combination of insulation materials used in the manufacturing of cable, during the early portion of the research and development phase of nuclear energy applications, both military and commercial. Several professional societies, some of which are mentioned above, participated in the studies and developed industry standards for guidance on radiation effects.

IEEE Std. 279-1968, which was begun in 1964, contained industry thinking that type-test data or reasonable engineering extrapolation based on type test data should be available to verify that safety related components or equipment would perform their function during an accident condition in nuclear power plant designs. These early standards provided documentation of what was existing industry practice and provide reasonable assurance that the cable used in the older nuclear power plants was carefully specified and procured for the service conditions to which it would be exposed.

One purpose of the Systematic Evaluation Program, discussed in Section 5, is to determine whether cabling built to these earlier standards provides sufficient safety margin or whether upgrading, discussed in Section 2.3.2, is needed.

2.4 Other Considerations

2.4.1 Timing Considerations

In considering the potential effects of severe LOCA conditions on performance of electrical systems, it is important to consider the equipment in two separate classes - that which must function essentially instantaneously to transmit a protective signal in the event of an accident and that which must function for an extended time.

For those systems and components whose primary purpose is to promptly transmit a signal in the event of an accident,[2] there is a high likelihood that they will successfully perform their function well before the onset of environmental conditions (temperature, pressure, etc.) which could cause deterioration of the equipment and, thereby, interfere with their function. Typically such signals are transmitted in less than a few seconds, often in fractions of a second after the event (e.g., scram signals which actuate control rods and permit their insertion). It is unlikely that exposure to accidental environmental conditions would cause equipment deterioration in such a limited period of time.

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[2] These include such items as the reactor control rods, sensors, that indicate reactor parameters such as flow rate and neutron flux, associated cables, connectors and penetrations.

In any event, once a signal is generated (even if only by the environmental conditions themselves), the initiation of the remaining safety-related equipment is accomplished and the signal [illegible]ly is "sealed in" and maintained even if the equipment providing the actuation signal were to subsequently fail.

Once the safety systems are initiated, few components located inside the containment are required to function. Such components include: valves that change position, usually within the first minute, to align the flow paths of the safety systems; isolation valves that quickly close to seal the containment from leakage; and the few instrument signals and valves that would normally be required to diagnose the type of accident and institute long term actions (usually required several hours after an accident). Given the requirements for redundance for all "active" components in safety-related systems and their quick acting nature (fractions of a second to several tens of seconds), we would expect that the adverse environmental conditions would not prevent the essential functioning of the safety-related equipment. However, because this a potential common cause for equipment failure, the staff requires environmental qualification of safety equipment. In place of equipment that

would need to function as long as several hours after an accident, other equipment can often be used to perform many of the same functions.

2.4.2 Equipment Response

Analyses of postulated design basis accidents inside the reactor containment have identified transient conditions in the containment atmosphere that may be severe but are often of short duration. Generally, the licensing approach of the NRC requires that a conservative determination of containment atmosphere parameters (e.g., temperature) be calculated and that these conservative parameters then be used as the basis for qualification of electrical equipment located inside containment. Historically, equipment was qualified to the conditions predicted to exist during and subsequent to a large LOCA since they were believed to be limiting. As discussed in Appendix A, the staff has determined that a postulated main steam line break (MSLB) in PWR type plants with dry containments could result in predicted temperatures higher than that of a LOCA, but only for a short

period of time (i.e., 60 to 100 seconds). In order to better understand the thermal response of selected typical electrical equipment inside containment, the staff has performed "best estimate" evaluations of typical components to determine the surface temperature of such components for a particular calculated temperature in the containment that might result from a MSLB. This evaluation[3/] indicates that because of the short duration of the predicted MSLB environmental conditions that exceed typical predicted LOCA conditions, the thermal response of the type of equipment in question would generally not exceed the conditions under which the equipment was originally qualified.

2.4.3 IE Inspection Program

Another program in the NRC that provides additional confidence in the environmental qualification of electrical equipment, is the inspection program of the Office of Inspection and Enforcement. The Office of Inspection and Enforcement in its routine inspection program has emphasized review of environmental qualification test results for engineered safety systems. The emphasis has been placed primarily on the larger components, such as motors, switchgear, breakers, controls, transmitters, cables, and emergency diesels. These activities have been carried out by review of documentation at the licensee's facilities and by inspectors accompanying the licensee at inspection of vendor facilities or

[3/] Memorandum to R. Boyd, V. Stello and R. Mattson from R. Tedesco dated December 2, 1977.

testing laboratories. These IE practices, add to the with engineered safety systems will perform their required functions in the accident environments for which they were designed.

2.4.4 Routine Experience Review

As the number of nuclear power facilities licensed to operate continues to grow (67 at present), the amount of operating experience at these facilities also increases. The NRC has a thorough mechanism for the reporting of operating experience. Over the past year there have been about 3000 Licensee Event Reports (LERs) submitted to the NRC. These LERs are routinely reviewed by IE and the Division of Operating Reactors. A complete file is maintained by the Office of Management Information and Program Control (OMIPC).

The NRC's review of LERs helps to identify, first, whether any electrical equipment is degrading under normal operation and secondly, whether operational transients and occurrences have degraded performance of electrical equipment under these conditions.

3.0 Adequacy of Qualification Tests

The second issue identified in Section 1.0 relates to the "adequacy" of present environmental qualification testing. The question is basically one of whether qualification tests that have been and are being performed are adequate to demonstrate that safety-related electrical equipment will perform in accident environments. One aspect of this question is whether successive exposure to certain parameters (temperature, humidity, pressure, etc.) and radiation is adequate to reflect performance of electrical components under accident conditions in which the exposure to these conditions is concurrent. The present industry standard, IEEE Std 323-1974 and its predecessors are based on the assumption that sequential testing is adequate. At the present time, the staff believes that the successful qualification of electrical equipment for these conditions, albeit sequentially, nonetheless provides a reasonable assurance that such equipment will be able to perform successfully under combined accident conditions. However, there is an absence of rigorous testing under concurrent exposure conditions, and while it is likely that sequentially tested components will successfully perform under accident conditions, the staff believes it prudent to confirm this judgment. The confirmatory research efforts in this regard are discussed in Section 4.0

NRR has a Technical Activities Program involving generic technical activities judged by the staff to warrant priority attention in terms of resources to attain early resolution. These are designated as Category A Technical Activities. The Technical Activities Program was developed to provide a basic framework of policy, organizational structure, priority, and procedures for the effective management of the major technical activities within NRR. Two activities in this program are directly related to the environmental qualification of safety-related mechanical and electrical equipment. These activities, designated A-21 and A-24, are related to main steam line break considerations inside containment and to qualification of safety-related equipment, respectively. These efforts are described in Appendix A.

4.0 NRC Confirmatory Research Programs

The current NRC Qualification Testing Evaluation Program is directed toward providing a confirmatory assessment of current environmental qualification testing procedures for LOCA conditions and includes the following specific program elements:

1. An assessment to determine if sequential (as opposed to simultaneous) environmental qualification testing is conservative, i.e., an investigation of synergistic effects.
2. Confirmation that accelerated aging methodology can be utilized. for qualification testing of safety-related equipment.
3. Definition of the nuclear radiation source based on the Regulatory Guide 1.89 accident assumptions and evaluation of the adequacy of radiation simulators.

4.1 Synergistic Testing

The tests were to confirm that the sequential test sequence recommended in IEEE Std 323-1974 conservatively simulates the combined radiation and steam environment to which safety-related equipment would be exposed in the unlikely event of a LOCA. A research program to investigate potential synergistic effects was initiated at Sandia Laboratories in FY 1975. LOCA qualification tests are being conducted using the same experimental test chamber and identical test samples (1) sequentially

as recommended in IEEE-Std-323-74, with radiation exposure preceding exposure to steam and chemical environments and (2) simultaneously with radiation, steam, and chemical environments imposed together. A qualitative comparison of the performance of the test specimens in both tests will be made, using on-line measurements and post-test evaluation.

Preliminary evaluation of the Sandia tests completed to date does not indicate a significant functional synergism for electrical cables; however, with respect to connectors, it was not possible to determine whether synergism exists because of the failures that occurred.

4.2 Aging Effects

Considerations of aging in environmental qualification test programs are important because of the potential to create a weakened condition in a safety-related component through some aging mechanism that may not be detected through routine periodic testing. A research program to develop a methodology that can be utilized for simulation of the natural aging process

of safety-related materials on an accelerated basis was initiated in FY 1976 and is continuing. The current effort includes the following elements:

a. Single environment aging tests on polymeric electric cable materials are being conducted to obtain data from the separate effects of radiation, temperature and humidity. From these tests, single environment acceleration functions of damage versus time will be obtained at relatively low stress levels using a test cycle of about one year. Cable elongation is being used as a relative damage indicator to verify the aging methodology and for comparison with naturally aged cable samples.

b. Combined environment aging tests will be conducted to obtain data on the synergistic aging effect of temperature and radiation. Synergisms with other aging parameters are planned later in the program.

c. Tests to determine rate effects are underway. Of particular interest are the rate effects associated with oxygen diffusion and radiation.

d. A study of alternate damage indicators is underway that could be utilized in addition to the material elongation criterion which is currently being used as the reference damage indicator for aging damage to electrical cable.

e. Naturally aged samples are being collected so that the aging methodology being developed can be checked with naturally aged material.

4.3 Source Term Equivalences

One of the exceptions taken to IEEE Std-323-1974 by the NRC is the required radiation environment to be utilized for environmental qualification testing. LOCA radiation releases are defined in Regulatory Guide 1.89 which is to be used by an applicant in establishing the nuclear radiation environment for type testing. The accident conditions are defined in terms of the percentage of halogens and solid fission products contained in the coolant, the percentage of noble gases and halogens released to the containment atmosphere and the percentage of halogens plated out on surfaces inside containment.

The adequacy of currently used radiation simulators to duplicate the accident radiation environment requires additional experimental evaluation. A research program to assist in this evaluation was initiated in FY 1976 and is continuing. Progress to date has consisted of analysis to determine the time relationship following a LOCA of dose, dose rate, energy spectra and particle type. These data show that current industry practice with regard to radiation

simulation testing may be significantly different in terms of dose rate, spectrum and particle type than that described in Regulatory Guide 1.89. The ongoing work in this area is aimed at determining the importance of these differences in terms of damage to safety related equipment. The current effort consists of the following three tasks:

a. Additional source term calculations will be made, based on Regulatory Guide 1.89 assumptions, taking into account new codes and test data developed in other programs. Also, calculations based on proposed regulatory guide modifications will be performed. These calculations will be based on the proposed revision to Regulatory Guide 1.89 which allows for reduced release assumptions for certain classes of safety-related equipment. In addition, source term calculations will be made with best estimate LOCA release assumptions as required.

b. An evaluation is being made of the adequacy of currently utilized radiation simulators to duplicate the hypothetical environment following the radioactive release postulated in Regulatory Guide 1.89. An initial assessment will be made comparing dose rates and energy spectra resulting from the conservative accident assumptions in Regulatory Guide 1.89 to the dose rates and energy spectra obtainable with practical simulators.

c. Studies will be conducted to determine the damage to safety-related equipment materials as a function of the gamma and beta dose-rates and to determine how close the dose-rate profiles resulting from the Regulatory Guide 1.89 assumptions must be simulated during qualification testing. Materials studies will be conducted utilizing radiation damage data available, and additional experimental data will be obtained as needed.

5.0 SEP Program

5.1 Scope of Technical Review

Very recently NRR embarked on a program to re-evaluate selected safety considerations for eleven older operating facilities. That program called the Systematic Evaluation Program is described in a recently issued NRC report.*

One of the topics included in this program, which is directed at a determination and documentation of the degre^ to which these older facilities meet current licensing requirements, concerns the environmental qualification of safety-related equipment.

The objective of the SEP's review of this topic is to evaluate the degree to which the mech..ical and Class IE electrical equipment of safety-related systems has been qualified for the environments associated with design basis events. As such, the SEP will be directed toward the determination of existing safety margins and the evaluation of the adequacy of such safety margins to determine if any backfitting or facility upgrading is necessary.

Because of recent operational occurrences at the Milestone plant, and in view of the results of the recent surveys regarding connectors and penetrations, this review topic will be completed

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* Report on the "Systematic Evaluation of Operating Reactors", dated November 25, 1977.

as the first topic of the SEP. While the overall Systematic Evaluation Program is scheduled to be completed in about three years, the review of this topic will be accelerated. It is expected that within about 90 days the review effort will be sufficient to assess any safety implications in sufficient detail to decide whether or not additional review of facilities other than those included in the SEP is required. The adequacy of the environmental qualification of mechanical equipment will follow that of the electrical equipment. The review plan for this effort for the eleven SEP facilities is set forth in Appendix B.

5.2 Extent of Present Program

The present Systematic Evaluation Program includes the review of eleven of the older operating nuclear reactors. These eleven include plants licensed before 1969 and those facilities which require a review for conversion of a Provisional Operating License (POL) to a Full-Term Operating License (FTOL).

Following the 90-day review of the environmental qualification of electrical equipment for these eleven older facilities, the staff will determine whether any plant modifications or followup actions are required for those facilities and will also consider

whether the environmental qualification review should be extended to include the remainder of the operating facilities. We have concluded that the eleven older facilities can be used as a basis to make such a decision because as noted in Appendix A, they represent a grouping of plants that would likely have a lesser degree of environmental qualification for the safety-related electrical equipment than more recent plant designs.

6.0 Conclusions

The NRC staff has concluded that no immediate action Commission is needed on the question of environmental qualification of safety-related electrical equipment in operating reactors is warranted.

Beyond the question of immediate action, the staff has considered whether the recently completed preliminary surveys regarding electrical connectors and containment electrical penetrations in operating plants should be expanded to consider, on a longer term basis, the safety adequacy and environmental qualification of other electrical equipment in these plants. The Systematic Evaluation Program for Operating Reactors recently approved by the Commission provides a suitable framework for such an expanded effort since it already includes "Environmental Qualfication of Safety-Related Equipment" as one of the topics considered. The SEP further provides that topics considered to be of special safety significance may be considered on a case-by-case basis in advance of completing the overall program.

The staff has determined that it is appropriate to complete the review of this subject as the first topic of the Systematacic Evaluation Program. It is expected that within about 90 days the review effort will be sufficient to assess any safety implications in sufficient detail to decide whether or not additional review of facilities other than those included in the SEP is required.

The results of the detailed staff review of these topics for these facilities, the eleven of the older reactors, will indicate whether further action is needed on the other operating reactors.

In reaching the judgment that no immediate action is required on operating reactors, the staff, as discussed elsewhere in this report, considered the following:

1. Nuclear power plants include provisions, such as redundancy and diversity, to cope with equipment failures without affecting the public health and safety.
2. Operating experience indicates that electrical equipment has performed adequately under both normal operating environmental conditions and on the few occasions where severe environmental conditions have existed.
3. Even the older operating reactors used conservative design and construction practices and many improvements have been made in the area of environmental qualification.
4. A preliminary audit of the environmental qualification of electrical connectors and penetrations in operating reactors has indicated that there is reasonable assurance that this equipment would perform its safety function under accident conditions even though complete documentation is not readily available in all cases. It is the staff's belief that these findings would be essentially the same for other safety-related equipment.

5. The likelihood that essential safety-related equipment or other non-safety equipment would not perform the necessary safety function prior to failure due to environmental reasons[4/] coupled with the likelihood of a major accident requiring the performance of this equipment is very low.

6. The regulations have included requirements for environmental qualification and a comprehensive quality assurance program since 1971. The requirement for environmental qualification was included in initial versions of these regulations in the mid 1960s. The NRC compliance effort by the Office of Inspection and Enforcement has emphasized review of environmental qualification test results for safety systems in its routine inspection program.

4/ It should be noted that even in the Sandia tests, under the conditions of Reg. Guide 1.89, which envelope DBA conditions and are thus conservative, particularly as to radiation and steam temperature conditions, a number of unqualified connectors, survived for periods in excess of several hours.

Appendix A

REPORT ON THE HISTORICAL EVOLUTION OF ENVIRONMENTAL QUALIFICATION REQUIREMENTS FOR SAFETY-RELATED ELECTRICAL EQUIPMENT

December 15, 1977

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ABSTRACT

Since the early days of the commercial nuclear power industry , the NRC (formerly AEC) criteria for the licensing of nuclear facilities have undergone an evolutionary process. This document traces the development of Commission and industry requirements for environmental qualifications of safety-related electrical equipment from the late 1950s to the present time. It also addresses the expanding role of related Inspection and Enforcement activities over this period of time.

1.0 INTRODUCTION

The purpose of this report is to describe the evolution of NRC licensing requirements for the environmental qualification of safety-related electrical equipment. The scope of the report has been limited to criteria which relate to normal, abnormal, accident, and post-accident environmental conditions, i.e., temperature, pressure, relative humidity, steam, radiation, chemicals, and vibration.

As has been the case with virtually all NRC licensing criteria, the licensing criteria for the environmental qualification of safety-related electrical equipment have evolved over the years as the design of reactor systems has changed and as regulatory and operating experience have accumulated. The evolution of NRC licensing criteria for environmental qualification of safety-related electrical equipment has occurred simultaneously and at essentially the same pace as the evolution of the overall NRC licensing criteria, as summarized below.

2.0 EVOLUTION OF OVERALL NRC LICENSING CRITERIA

In the early days of the civilian nuclear power industry, the Commission's licensing review of the acceptability of proposed nuclear plant designs was based on much less documented design information than is presently required by the NRC licensing process. In addition, as early reactor designs evolved, the

Commission's "standards of acceptability" were established on an ad hoc basis unique to each new licensing review. Many of these "standards of acceptability" were not formalized; rather they evolved during plant specific licensing reviews, thereby establishing precedents for subsequent reviews. As the number of applications for Construction Permits (CPs) submitted to the Commission began to grow, it became evident that more uniform and consistent guidelines ("standards for acceptability") were necessary. In 1966, the Commission issued a "Guide to the Organization and Contents of Safety Analysis Reports."1/ Historically, from this time on, the amount of documentation required by the Commission's licensing process began to significantly increase. That guide identified the areas of NRC staff safety concern and identified the degree of detailed information and analyses required from applicants to permit the staff to complete its review.

In a further effort to provide guidance to the industry and to increase staff review efficiency and effectiveness, the Commission began issuing Safety Guides in 1970. These guides present methods acceptable to the Commission for implementing specific parts of the Regulations, including the General Design Criteria of 10 CFR Part 50 Appendix A. In 1971, The Commission began issuing Information Guides to list needed information which was frequently omitted from applications. In 1972 the Safety and Information Guides were replaced by the broader based NRC Regulatory Guide program which

continues today. Regulatory Guides are not substitutes for Regulations and compliance with them is not in itself a legal requirement. Methods and solutions different from those set forth in the Guides may be, and have been found to be, acceptable by the NRC staff.

The next major improvement in guidance to applicants was provided in a document entitled "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants",2/ initially issued in 1972 and subsequently revised in 1975. These documents reflect the increased scope and detail of information required to support license applications.

In a similar way, the licensing criteria and requirements used by the NRC staff to determine acceptability have evolved over the years. In 1965, the Commission published, for public comment, 27 proposed General Design Criteria (GDC) for nuclear power plants. Those criteria established minimum requirements for the principal design criteria for commerical nuclear plants. These GDC were refined and formally adopted in the Regulations, as Appendix A to 10 CFR Part 50 in 1971 in the form of 55 General Design Criteria. These GDC have been used by the Commission as guidance in reviewing plant applications since they were originally drafted. The GDC were specifically written in general terms such that a variety of alternative design techniques may be utilized to satisfy them.

As the knowledge of reactor designs increased and operating experience accumulated, additional licensing requirements were also developed. In an effort to document such requirements and thus

increase regulatory effectiveness, efficiency and predictability, the Commission developed and published the Standard Review Plan (SRP) in 1975.3/ That document provides guidance for staff reviewers to improve the quality and uniformity of staff reviews. It also improves communication and understanding of the staff review process with interested members of the public and the nuclear power industry and helps to standardize the licensing pr[illegible]ess.

In general, the detailed acceptance criteria published in the SRP did not represent new licensing requirements; rather, they reflected current staff review practices and standards of acceptability which had evolved during previous licensing reviews. However, in many cases, these criteria had not previously been published in any regulatory document. The SRP will be periodically revised to incorporate new or modified requirements as they are developed and approved.

Because of the continual evolution of reactor designs and associated licensing requirements, operating nuclear power plants that were reviewed and approved in the past have a broad spectrum of design characteristics. Each of these reactors was found to be in conformance with the licensing "requirements" in effect at the time of licensing. As noted above, however, these requirements have become more detailed over the years. Consequently, if some older licensed facilities were reevaluated using current licensing procedures, they would likely be at variance in some respects; the older the plant the more at variance it is likely to be. Although such variances may not necessarily

represent significant safety deficiencies, the current Systematic Evaluation Program (SEP), which is described in Section 4.1 of this report, is directed toward the determination of existing safety margins at older operating reactors and the evaluation of such safety margins to determine if any backfitting or facility upgrading for safety is necessary.

3.0 EVOLUTION OF NRC ENVIRONMENTAL QUALIFICATION REQUIREMENTS

3.1 Evolution of Applicable National Standards

Over the past ten years, many national Standards have been prepared to describe the methods commonly used to demonstrate the environmental qualification of safety-related electrical equipment utilized in nuclear power generating stations. These Standards reflect the elements of good engineering practice which have evolved in the development of reactor system designs and regulatory licensing requirements. The NRC staff has, for many years, participated with representatives of industry in the development of these Standards and, after independent review, has often incorporated these Standards into its Regulations and Regulatory Guides with appropriate supplemental material. Each of the pertinent Standards is described in the following sections of this report. A graphical presentation of the sequence of the development of these Standards is provided in Table 1.

As noted previously, however, these Standards, as endorsed by NRC Regulatory

Guides, are not substitutes for Regulations. Consequently, compliance with these Standards (with the exception of IEEE Standard 279 which is incorporated by reference in 10 CFR Part 50, §50.55a) is not a legal requirement. During the course of the licensing review for a particular facility, methods or solutions different from those set forth in these Standards may be found to be acceptable by the staff, but usually on some basis of comparability with the provisions of the Standards.

3.1.1 Section 4.4 of IEEE Standard 279-1968, "Proposed IEEE Criteria for Nuclear Power Plant Protection Systems," and its revision dated 1971, require that either type test data or reasonable extrapolations based on test data be available to demonstrate the environmental qualification of protection system equipment at nuclear power plants. This Standard has been incorporated as part of the Commission's regulations by reference in 10 CFR Part 50.55a, "Codes and Standards".

More detailed qualification guidance for electrical equipment has been developed in several later IEEE Standards and, where appropriate, these Standards have been endorsed by NRC Regulatory Guides (sometimes with supplementary material) as acceptable methods for qualifying electric equipment in general or specific kinds of electric equipment.

3.1.2 IEEE Standard 323-1971, "General Guide for Qualifying Class I Electrical Equipment for Nuclear Power Generating Stations," and its revision dated 1974, describe the basic requirements for qualifying Class IE equipment and interfaces that are to be used in nuclear power generating sta' ns, in partial support of the requirements of GDC 4 and 21 (Appendix A to 10 CFR Part 50) and Sections 4.4 and 4.5 of IEEE Standard 279-1971. The 1974 Revision of this Standard included criteria which establish requirements for qualification procedures, methods, and documentation and incorporated new or improved guidelines related to aging, testing margins, and the sequence for testing of different environmental parameters. This Standard recognizes that environmental qualification of safety-related electrical equipment can be accomplished by several different methods (e.g., type testing, operating experience, analysis utilized separately or in combination. Furthermore, this Standard recognizes that, while fulfillment of its requirements does not necessarily fully establish the adequacy of the environmental qualification of electrical equipment, omission of any of its requirements will, in most instances, be an indication of inadequate qualification.

IEEE 323-1974, nas been endorsed by NRC Regulatory Guide 1.89. The Standard and Regulatory Guide 1.89 have been written such that they can be followed by successive ancillary Standards and Guides which reference the parent documents for common qualification techniques and specify the additional requirements pertinent to a specific component. For example, IEEE 334-1974 (motor qualification) is ancillary to IEEE 323-1974. Respective endorsement guides have the same relationship. A listing of applicable qualification Standards, including ancillary Standards, follows:

a. IEEE Standard 383-1974, "Type Test of Class IE Electric Cables, Field Splices, and Connections for Nuclear Power Generating Stations," provides direction for establishing type tests which may be used in qualifying Class IE electric cables, field splices, and connections for service in nuclear power generating stations in conjunction with of the general guidelines for qualification which are given in IEEE Std. 323-1974 and in GDC 3. Regulatory Guide 1.131, endorsing this ancillary IEEE Standard, was issued for public comment in August 1977.

b. IEEE Standard 317-1971, "Standard for Electrical Penetration Assemblies in Containment Structures for Nuclear Fueled Power Generating Stations," and its revisions dated

1972 and 1976, provide guidance for qualifying electrical penetrations and include testing requirements. The 1976 version of this Standard includes additional design and testing requirements and is ancillary to the 1974 version of IEEE Standard 323. Revision 1 to NRC Regulatory Guide 1.63, in turn, endorses this Standard.

c. IEEE Standard 334-1971, "Type Tests of Continuous Duty Class I Motors Installed Inside the Containment of Nuclear Power Generating Stations," and its revision dated 1974, specify acceptable methods for qualifying electric motors. The 1974 version of the Standard reflects the 1974 update of IEEE Standard 323 and will be endorsed by Revision 1 to NRC Regulatory Guide 1.40. Regulatory Guide 1.40 endorsed IEEE Standard 334-1971.

d. IEEE Standard 382-1972, "Type Tests of Class I Electric Valve Operators for Nuclear Power Generating Stations" specifies acceptable methods for qualifying electric valve operators. NRC Regulatory Guide 1.73, in turn, endorses this Standard. (This is not an ancillary Standard).

e. IEEE Standard 381-1977, "Type Tests of Class IE Modules Used in Nuclear Power Generating Stations," is an ancillary Standard which specifies acceptable methods for qualifying electric modules. An NRC Regulatory Guide endorsing this Standard is being considered for development.

1972 and 1976, provide guidance for qualifying electrical penetrations and include testing requirements. The 1976 version of this Standard includes additional design and testing requirements and is ancillary to the 1974 version of IEEE Standard 323. Revision 1 to NRC Regulatory Guide 1.63, in turn, endorses this Standard.

c. IEEE Standard 334-1971, "Type Tests of Continuous Duty Class I Motors Installed Inside the Containment of Nuclear Power Generating Stations," and its revision dated 1974, specify acceptable methods for qualifying electric motors. The 1974 version of the Standard reflects the 1974 update of IEEE Standard 323 and will be endorsed by Revision 1 to NRC Regulatory Guide 1.40. Regulatory Guide 1.40 endorsed IEEE Standard 334-1971.

d. IEEE Standard 382-1972, "Type Tests of Class I Electric Valve Operators for Nuclear Power Generating Stations" specifies acceptable methods for qualifying electric valve operators. NRC Regulatory Guide 1.73, in turn, endorses this Standard. (This is not an ancillary Standard).

e. IEEE Standard 381-1977, "Type Tests of Class IE Modules Used in Nuclear Power Generating Stations," is an ancillary Standard which specifies acceptable methods for qualifying electric modules. An NRC Regulatory Guide endorsing this Standard is being considered for development.

3.2 Evolution of NRC (ONRR) Licensing Requirements

The NRC's licensing requirements for the environmental qualification of safety-related electrical equipment have evolved over the years as the design of reactor systems has changed, as operating and regulatory experience have been accumulated, and as testing facility capabilities and testing techniques have expanded and improved. In a very general sense, the evolution of these NRC licensing requirements can be characterized by three stages: the evaluation of facilities licensed prior to 1967, facilities licensed after 1967 up to facilities with Construction Permit (CP) applications tendered prior to July 1974, and facilities with CP applications tendered after July 1974. A listing of the facilities which fall within each of these three groupings is provided in Table 2. A graphical presentation of the sequence of development of licensing requirements and/or guidance for the environmental qualification of safety-related electrical equipment is provided in Table 3.

For all plants in the first two of the above-mentioned three groupings, the licensing review included an evaluation of the environmental qualification of safety-related electrical equipment located inside containment. As noted, the scope and depth of such reviews have increased with time. In addition, equipment environmental qualification has been considered by the staff for subsequent plant design modifications proposed by licensees of these facilities and for modifications required by changes in the Regulations; e.g., modifications associated with the demonstration of compliance with Appendix K of 10 CFR Part 50.

The staff review of plants falling within each of the three above-mentioned groupings is characterized in the following subsections.

3.2.1 PRIOR TO 1967

For facilities licensed prior to 1967, the information initially submitted to the regulatory staff for review of the environmental qualification of safety-related electrical equipment included, as a minimum, design specifications and demonstration of the adherence of such design specifications to appropriate industry standards, such as the National Electrical Manufacturers Association Standards, existing IEEE Standards which were not specifically developed for application by the nuclear industry (e.g., IEEE Standard 117-1956, "IEEE Standard Test Procedures for Evaluation of Systems of Insulating Materials for Random-Wound AC Electrical Machinery"), and other IEEE Standards under development. In addition, applicants referenced various environmental testing programs such as those conducted at the Franklin Institute Research Laboratories and those conducted in the Naval Reactors Program.

3.2.2 1967 - 1974

The staff's review of the second grouping of facilities utilized the initial criteria of IEEE Standard 279-1968 and IEEE Standard 323-1971. These criteria, and others that became available subsequent have been utilized, as they became available, for subsequent evaluations. It should be noted that several of the test programs to demonstrate the qualification of safety-related electrical equipment were initiated during the development of the criteria

established in IEEE Standard 323-1971 prior to its issuance. In some cases during the course of its reviews, the staff determined that the test plan procedures or other available documentation for particular facilities did not meet all of the IEEE 323-1971 guidelines. In such cases, the staff required confirmatory programs to be implemented, additional analyses to be performed, and/or additional documentation related to previous environmental testing programs to be supplied. Consideration of the plant specific design and site conditions were included in the staff's evaluation of the adequacy of equipment to perform its safety-related functions during normal, abnormal, accident, and post accident environmental conditions. These evaluations were performed on a selected component basis, i.e., the review treated components judged by the staff to be representative of all safety-related components in the facility.

As indicated in Section 3.1, the IEEE Standards continued to evolve during this time frame and, consequently, the licensing reviews of the plants that preceded the implementation of IEEE Standard 323-1971 and other Standards that became available subsequently had varying degrees of qualification, testing, analysis, and associated documentation. The evolution of the NRC and AEC Regulatory process and the Standards development process have markedly increased the amount of documentation and the scope and depth of the reviews performed by the staff since 1971.

Appendix X to the "Reactor Safety Study"4/ provides a detailed summary of the environmental qualification of safety-related equipment installed in four of the facilities within this grouping, i.e., Surry Units 1 and 2 and Peach Bottom Units 2 and 3.

3.2.3 CPs Tendered After July 1974

The staff's reviews of the environmental qualification of safety-related electrical equipment for plants tendering CPs after July 1974 reflect the more comprehensive guidelines specified in IEEE Standard 323-1974 and the successive ancillary Standards. As discussed in Section 4.2.2 of this report, the staff is currently reviewing the concepts, methods, test procedures, and acceptance criteria proposed by the major NSSS vendors and Balance-of-Plant equipment suppliers to meet the guidelines for environmental qualification of safety-related electrical equipment provided in IEEE Standard 323-1974 and its ancillary Standards. The staff has not completed its review of these programs.

The methods and procedures which are reviewed for facilities within this grouping are documented in Section 3.11 of the Standard Review Plan (SRP) which was issued in September 1975. The Standard Review Plan is written so as to cover a variety of site conditions and plant designs. For any given appli-

cation, the staff selects and emphasizes particular aspects of the Standard Review Plan as is appropriate for that application. In some cases, a plant feature may be sufficiently similar to that of a previous plant so that a detailed re-review is not needed. Therefore, the staff has not and does not expect to perform in detail all of the review steps listed in the Standard Review Plan for the review of each application. This approach is typical for most other review areas.

The SRP-type environmental qualification review includes an information audit review to determine if the following information is included in the application:

a. All safety-related mechanical and electrical equipment must be identified. The equipment tabulations provided should be checked for completeness against the descriptions of safety-related systems. Definitions of the three categories of safety related systems are contained in Section 7.1 of the SRP.

b. The location of each item of safety-related equipment, both inside and outside the containment, must be identified. Location of the equipment is required in order to establish accurate definitions of both the normal, abnormal, and accident environments.

c. Both the normal, abnormal, and accident environmental conditions must be explicitly defined for each item of equipment. These definitions must include the following parameters: temperature, pressure, relative humidity, steam, radiation, chemicals, and vibration. For the normal environment, specific values should be provided. For the abnormal and accident environments these parameters should be presented as functions of time for the particular cause of the postulated environment, (e.g., pipe break, or other).

d. The length of time that each item of equipment is required to operate in an abnormal or accident environment must be provided.

e. The qualification report should contain a complete description of the design bases and environmental qualification tests and/or analyses that have been performed on each item of safety-related equipment. This should include qualification for the accident environments, qualification for extreme normal operating environments, and qualification to assure that loss of environmental control systems that are not classified as safety-related will not adversely affect the operability of safety-related equipment, particularly electrical equipment located in the control room or other rooms housing control equipment.

The staff review involves an evaluation of the completeness and adequacy of the information presented to assure that an adequate demonstration of the required environmental capabilities of safety-related equipment has been provided. This phase of the review is performed after it has been established (by means of the information audit phase of the review previously described) that the information content requirements for Section 3.11 of the Standard Review Plan have been satisfied. An essential part of this evaluation is the formulation of questions by the licensing reviewers and responses by applicants to document their qualification programs. Typical of these requests for additional information are:

a. A statement that the staff requires that the following qualification test program information be provided for a specified list of Class IE equipment: (1) equipment design specification requirements, (2) test plan, (3) test set up, (4) test procedures, (5) acceptability goals and requirements, and (6) test results.

b. A requirement that the applicant demonstrate that the sequence of the environmental conditions which were imposed during the qualification testing is at least as severe as the actual environment sequence during a postulated accident.

c. A requirement that the applicant provide supporting analyses, operating experience, or other information that demonstrates the adequacy of specific equipment to perform its required safety function in normal, abnormal, accident, and post-accident environments.

The Regulations, current IEEE Standards, and the NRC Regulatory Guides which are identified in Section 3.11 in the Standard Review Plan are used as guidelines and as the acceptance criteria for the staff reviews to provide assurance that the equipment can perform its safety-related function. The staff reviews the proposed concepts, methods, and test procedures that will be utilized to demonstrate compliance with the current criteria and evaluates the results of analyses, tests, experience, or other methods (including combinations of the above) for acceptability when they are submitted at the FSAR stage of the licensing process.

3.2.4 Other Considerations

Further evidence of the evolutionary nature of the licensing considerations for environmental qualification of safety-related electrical equipment has been the recent recognition

that the environment associated with a postulated main steam line break (MSLB) accident in PWR facilities may, in some respects, be more demanding on electrical equipment installed inside containment than the environmental conditions associated with the design basis loss of coolant accident (LOCA). Prior to 1976, the accident environment against which safety-related electrical equipment located inside containment was qualified was bounded by the environment produced by the loss of coolant accident. However, in 1976, information became available that indicated that the calculated temperature inside the containment associated with the MSLB accident could be as much as 100 - 150°F higher, for a short time duration (i.e., 60-100 seconds), than that associated with a LOCA. As indicated in Section 4.2.1 of this report, further efforts are presently underway to establish environmental envelope requirements for MSLB accidents inside containment.

3.3 Evolution of OI&E Inspection Practices

The NRC Office of Inspection & Enforcement's involvement in the environmental qualification testing of safety-related electrical equipment has evolved in step with the NRC licensing (ONRR) re-

quirements. During the late 1960's and early 1970's, the Office of Inspection & Enforcement's inspectors periodically visited equipment vendors for the purpose of auditing vendor practices with respect to qualification testing. Inadequacies identified during such inspections were brought to the attention of appropriate NRC licensing personnel. In a number of instances, generic problems and/or design deficiencies were identified and corrected.

Since the early 1970's the involvement of the Office of Inspection and Enforcement's inspectors in this area has increased substantially. In the 1972-1973 time frame, the current Office of Inspection and Enforcement's licensee contractor and vendor inspection program was initiated. Included in that program are provisions for the inspection, on a sampling basis, of suppliers of all types of equipment (including electrical and instrumentation components). While that program focuses on the review of vendor quality assurance programs and fabrication activities, witnessing of actual qualification tests is occasionally performed.

At the reactor construction sites, inspectors review selected qualification test results for safety-related components and systems. These reviews are based on specified requirements and/or test conditions. During the early 1970's inspectors reviewed testing and quality

control requirements, material certification, and test results. Since about 1975, the inspection requirements have been expanded and have been further clarified in areas relating to equipment qualification tests. For example, existing Office of Inspection and Enforcement inspection procedures provide for (1) reviews of selected vendor supplied documents which identify the environmental qualification testing performed for safety-related electrical equipment, and (2) inspections to assess the adequacy of the licensee's or applicant's quality assurance programs which are designed to assure that safety-related electrical equipment installed at the facility have been qualified in accordance with the vendor's testing programs.

Additionally, increased emphasis has been placed on the inspector's involvement in the pre-operational testing phase of facility systems. The inspectors witness an increased number of testing activities and perform a more thorough review of safety-related test results.

4.0 CURRENT ACTIVITIES

The nuclear industry and the NRC have various programs currently in progress related to the environmental qualification of safety-related electrical equipment. Activities of each of these programs are discussed in the following sections.

4.1 Systematic Evaluation Program (NRC/ONRR)

The Division of Operating Reactors, ONRR, presently has underway Phase II of the Systematic Evaluation Program (SEP). This Program consists of the systematic review of eleven older nuclear power facilities (plants licensed for operation before 1969 and those which require a review for conversion of a Provisional Operating License to a Full-Term Operating License) to determine and document the degree to which they meet current licensing requirements for new plants. Phase II of the SEP was approved by the Commission on November 9, 1977. The results of this systematic evaluation, which is scheduled for completion within the next three years, will be considered by the Commission in deciding whether the program should be extended to include other operating facilities. One of the specific review topics included in this program is titled "Environmental Qualification of Safety-Related Equipment," (SEP Topic List No. III-12).

The objective of the SEP review of this topic is to evaluate the degree to which the mechanical and Class IE electrical equipment of safety-related systems have been qualified for the most severe environment (i,.e., temperature, pressure, humidity, steam, chemistry, and radiation) of design basis accidents. As such, the SEP will be directed toward the determination of existing safety margins and the evaluation of the adequacy of such safety margins to determine if any backfitting or facility upgrading is necessary.

4.2 NRR Category A Technical Activities

As part of NRR's Technical Activities Program, which was developed to provide a basic framework of policy, organizational structure, priority, and procedures for the effective management of the major technical activities within NRR, those generic technical activities judged by the staff to warrant priority attention in terms of resources to attain early resolution were designated as Category A Priority Activities. Of those so designated, two are directly related to the environmental qualification of safety-related mechanical and electrical equipment. A brief description of the applicable portions of each of these generic technical activities is provided below.

4.2.1 Category A Technical Activity No. A-21, "Main Steam Line Break Inside Containment"

One of the subtasks of this technical activity is to perform an evaluation of the procedures and content of analyses performed to establish environmental envelope requirements for main steam line break accidents inside containment. These environmental envelope requirements will then be utilized to assess the adequacy of the environmental qualification of safety-related

equipment inside containment. In addition, criteria for methodology of environmental simulation will be evaluated to determine if the important environmental parameters have been properly simulated during testing. The target date for completion of this task is December 1978.

4.2.2 Category A Technical Activity No., A-24, "Qualification of Class IE Safety Related Equipment"

The objective of this technical activity is to perform a generic review and evaluation of methods developed by industry to qualify safety-related equipment to the requirements established in IEEE Standard 323-1974, "IEEE Standard for Qualifying Class IE Equipment for Nuclear Power Generating Stations." Certain concepts and methods proposed by industry in addressing equipment qualification, such as testing margins, aging effects on materials and equipment, and adequacy of testing simulators (which simulate the worst case environment for the testing of equipment) have yet to be reviewed and accepted by the staff. In order to expedite the review of the adequacy of the proposed qualification methods, a generic review

of the qualification methodology and associated acceptance criteria used by the major NSSS vendors and Balance-of-Plant equipment suppliers will be conducted. The Target Date for completion of this task is early 1979.

4.3 Confirmatory Research Programs (NRC/RES)

The current NRC Qualification Testing Evaluation Program is directed towards providing a confirmatory assessment of current environmental qualification testing procedures for LOCA conditions and includes the following specific program elements:

1. Assessment to determine if sequential (as opposed to simultaneous) environmental qualification testing is conservative, i.e., an investigation of synergistic effects.
2. Confirmation that accelerated aging methodolgy can be utilized for qualification testing of safety-related equipment.
3. Definition of the nuclear radiation source based on the Regulatory Guide 1.89 accident assumptions and evaluation of the adequacy of radiation simulators.

4.3.1 Synergistic Tests

The tests were to confirm that the sequential test sequence recommended in IEEE Std. 323-1974 conservatively simulates the combined radiation and steam environment to which safety-related equipment would be exposed in the unlikely event of a

LOCA. A research program to investigate potential synergistic effects was initiated at Sandia Laboratories in FY 1975.

Preliminary evaluation of the Sandia tests which have been completed to date does not indicate a significant functional synergism for electrical cables; however, with respect to connectors, it was not possible to determine whether synergism exists because of the failures that occurred.

4.3.2 Aging Effects

Considerations of aging in enviromental qualification test programs are important because of the potential to create a weakened condition in a safety-related component through some aging mechanism that may not be detected through routine periodic testing. A research program to develop a methodology that can be utilized for simulation of the natural aging process of safety-related materials on an accelerated basis was initiated in FY 1976 and is continuing.

4.3.3 Source Term Equivalences

One of the exceptions taken to IEEE Std 323-1974 by the NRC is the required radiation environment to be utilized for enviromental qualification testing. LOCA radiation releases to be used by an applicant in establishing the nuclear radiation environment for type testing are defined in Regulatory Guide 1.89.

An assessment of the adequacy of currently used radiation simulators to duplicate the accident radiation environment requires additional experimental evaluation. A research program to assist in this evaluation was initiated in FY 1976 and is continuing. Progress to date has consisted of analysis to determine the post-LOCA time relationship of dose, dose rate, energy spectra and particle type. These data show that current industry practice with regard to radiation simulation testing may be significantly different in terms of dose rate, spectrum and particle type than that described in Regulatory Guide 1.89. The ongoing work in this area is aimed at determining the importance of these differences in terms of damage to safety related equipment.

4.4 Standards Development Programs

IEEE Standards related to the environmental qualification of the following specific safety-related electrical equipment are currently under development:

(a) Fire stops
(b) Fire breaks
(c) Storage batteries
(d) Switchgear
(e) Circuit breakers
(f) Battery chargers
(g) Transformers
(h) Motor control centers

A general Standard addressing the environmental qualification of safety-related mechanical and electrical equipment is being developed jointly by IEEE and ASME.

It is anticipated that NRC Regulatory Guides endorsing the above-mentioned Standards will be developed through FY 1979.

NRC Regulatory Guide 1.40 is currently being revised to reflect the updated requirements of IEEE Standard 334-1974. NRC Regulatory Guides 1.89 and 1.131 are being revised subsequent to their issuance for public comment.

5.0 REFERENCES

1. "Guide to the Organization and Contents of Safety Analysis Reports," U.S. Atomic Energy Commission, June 30, 1976.

2. "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," Regulatory Guide 1.70, Revision 1, U.S. Atomic Energy Commission, October 1972; Revision 2, NUREG-75/094, U.S. Nuclear Regulatory Commission, September, 1975.

3. "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," NUREG-75-087, U.S. Nuclear Regulatory Commission, September, 1975.

4. "Reactor Safety Study: An Assessment of Accident Risks In U.S. Commerical Nuclear Power Plants," WASH-1400 (NUREG-75/014), U.S. Nuclear Regulatory Commission, October 1975.

TABLE 1

Time Phasing of the Development of
IEEE Standards Addressing the
Environmental Qualification of
Safety-Related Electrical Equipment
For Nuclear Power Generating Stations

1968	1969	1970	1971	1972	1973	1974	1975	1976	1977	1978
	IEEE Std. 279-1968		IEEE Std. 279-1971							
			IEEE Std. 323-1971			IEEE Std. 323-1974				
			IEEE Std. 317-1971	IEEE Std. 317-1972				IEEE Std. 317-1976		
			IEEE Std. 334-1971			IEEE Std. 334-1974				
				IEEE Std. 382-1972						
									IEEE Std. 381-1977	
						IEEE Std. 383-1974				

TABLE 2

Facility Groupings by the Evolutionary Stage
of NRC Licensing Requirements For
Environmental Qualification of Safety-Related Electrical Equipment

Facility Name	Construction Permit Issued	Operating License Issued
A. Pre-1967		
Dresden Unit No. 1	5/56	9/59
Yankee-Rowe	11/57	7/60
Humboldt Bay Unit No. 3	11/60	8/62
Big Rock Point	5/60	8/62
Indian Point Unit No. 1	5/56	3/62
San Onofre Unit No. 1	3/64	3/67
Connecticut Yankee (Haddam Neck)	5/64	6/67
LaCrosse	3/63	7/67
B. 1967-1974		
Oyster Creek	12/64	4/69
Nine Mile Point Unit No. 1	4/65	8/69
Dresden Unit No. 2	1/66	12/69
Ginna	4/66	9/69
Millstone Unit No. 1	5/66	10/70
Dresden Unit No. 3	10/66	1/71
Indian Point Unit No. 2	10/66	10/71
Quad Cities Units Nos. 1 & 2	2/67	10/71/3/72
Palidades	3/67	3/71
Robinson Unit No. 2	4/67	7/70

Facility Name	Construction Permit Issued	Operating License Issued
Turkey Point Units Nos. 3 & 4	4/67	7/72/4/73
Browns Ferry Units Nos. 1 & 2	5/67	6/63/6/74
Monticello	6/67	9/70
Point Beach Unit No. 1	7/67	10/70
Oconee Units 1, 2 & 3	11/67	2/73/10/73/7/74
Vermont Yankee	12/67	3/72
Peach Bottom Units 2 & 3	1/68	8/73/7/74
Diablo Canyon Unit 1	4/68	
Three Mile Island Unit 1	5/68	4/74
Cooper	6/68	1/74
Ft. Calhoun	6/68	5/73
Prairie Island Units 1 & 2	6/68	8/73/10/74
Surry Units 1 & 2	6/68	5/72/1/73
Point Beach Unit No. 2	7/68	11/71
Browns Ferry Unit No. 3	7/68	7/76
Kewaunee	8/68	12/72
Pilgrim Unit No. 1	8/68	6/72
Ft. St. Vrain	9/68	12/73
Crystal River Unit No. 3	9/68	12/76
Salem Unit Nos. 1 & 2	9/68	8/76
Rancho Seco	10/68	8/74
Maine Yankee	10/68	9/72
Arkansas 1	12/68	5/74
Zion Units Nos. 1 & 2	12/68	4/73/11/73

Facility Name	Construction Permit Issued	Operating License Issued
D. C. Cook Units 1 & 2	3/69	10/74
Calvert Cliffs Units 1 & 2	7/69	7/74/8/76
Indian Point Unit 3	8/69	12/75
Hatch Unit 1	9/69	8/74
Three Mile Island 2	11/69	
Brunswick Units 1 & 2	2/70	9/76/12/74
Fitzpatrick	5/70	10/74
Sequoyah 1 & 2	5/70	
Duane Arnold	6/70	2/74
Beaver Valley Unit 1	6/70	1/76
Diablo Canyon 2	12/70	
St. Lucie Unit 1	7/70	3/76
Millstone Unit 2	12/70	8/75
Trojan	2/71	11/75
North Anna 1 & 2	2/71	
Davis-Besse	3/71	4/77
Farley 1 & 2	8/72	6/77
Fermi 2	9/72	
Zimmer 1	10/77	
Arkansas 2	12/72	
Midland 1 & 2	12/72	
Hatch 2	12/72	
Watts Bar 1 & 2	1/73	

Facility Name	Construction Permit Issued	Operating License Issued
McGuire 1 & 2	3/73	
Washington Nuclear 2	3/73	
Summer 1	3/73	
Shoreham	4/73	
Forked River	7/73	
LaSalle 1 & 2	7/73	
San Onofre 2 & 3	10/73	
Susquehanna 1 & 2	11/73	
Bailly 1	5/74	
Beaver Valley 2	5/74	
Limerick 1 & 2	6/74	
Nine Mile Point 2	6/74	
Vogtle 1 & 2	6/74	
C. Post-July 1974		
North Anna 3 & 4	7/74	
Millstone 3	8/74	
Grand Gulf 1 & 2	9/74	
Hope Creek 1 & 2	11/74	
Waterford 3	11/74	
Comanche Peak 1 & 2	12/74	
Surry 3 & 4	12/74	
Bellefonte 1 & 2	12/74	
Catawba 1 & 2	8/75	

Facility Name	Construction Permit Issued	Operating License Issued
South Texas 1 & 2	8/75	
Washingtin Nuclear 1	12/75	
Byron 1 & 2	12/75	
Braidwood 1 & 2	12/75	
Clinton 1 & 2	2/76	
Seabrook 1 & 2	2/76	
Callaway 1 & 2	4/76	
Palo Verde 1, 2 & 3	5/76	
Hartsville 1, 2, 3, & 4	5/77	
Perry 1 & 2	5/77	
Wolf Creek 1	5/77	
River Bend 1 & 2	3/77	
St. Lucie 2	5/77	
Sterling 1	9/77	

And all other CP applications currently under staff review

TABLE 3

Time Phasing of the Development of AEC/NRC Requirements and/or Guidance for the Environmental Qualification of Safety-Related Electrical Equipment

1965	1966	1967	1968	1969	1970	1971	1972	1973	1974	1975	1976	1977
Draft General Design Criteria Published for Comment 11/65 (GDC-16)		Revised General Design Criteria Published for Comment 7/67 (GDC-4, 23)				General Design Criteria Incorporated into Regulations, Appendix A to 10 CFR Part 50 5/71 (GDC-4, 23)						
				Revision to 10 CFR Part 50 (§50.55a), "Codes and Standards," Published for comment 11/69			10 CFR 50.55a "Codes and Standards," incorporated into Regulations 7/71 (Endorsed IEEE Std. 279)					
								AEC R.G. 1.89 (Endorsed IEEE Std. 323-1974) 11/74				
	AEC Published a Guide for Preparation of Safety Analysis Reports 6/66						AEC R.G. 1.70, Rev. 1 (Standard Format and Content of Safety Analysis Reports) 10/72				R.G. 1.70 Rev. 2 9/75 NRC Standard Review Plan 9/75	NRC R.G. 1.131 (endorsing IEEE Std. 383-1974) Published for comment 8/77

1965	1966	1967	1968	1969	1970	1971	1972	1973	1974	1975	1976	1977
						AEC I.G.2 Published 10/71						
									AEC R.G. 1.63 (endorsing IEEE Std. 317-1972) 10/73			Revision 1 to R.G. 1.63 (endorsing IEEE Std. 317-1976) published for comment 5/77
								AEC R.G. 1.40 (endorsing IEEE Std. 334-1971) 3/73				
									AEC R.G. 1.73 (endorsing IEEE Std. 382-1972) 1/74			

PROCEDURE FOR EVALUATION OF ENVIRONMENTAL QUALIFICATION OF SAFETY-RELATED ELECTRICAL EQUIPMENT AND COMPONENTS

The staff has determined that it is appropriate to complete the review of this subject as the first topic of the Systematic Evaluation Program. The licensees of the eleven SEP facilities will be required to evaluate the environmental qualification of all electrical equipment they deem necessary to mitigate the consequences of Design Basis Events. It is expected that within about 90 days the review effort will be sufficient to assess any safety implications in sufficient detail to decide whether or not additional review of facilities other than those included in the SEP is required. The results of this staff review will be used to make a determination of whether it is necessary to expand this effort from the eleven SEP facilities to other operating nuclear power plants. The staff's bases for this approach are set forth in its report, titled, "Staff Report on Environmental Qualification of Safety-Related Electrical Equipment" dated December 15, 1977.

The evaluation of these eleven older operating reactor facilities will be conducted in accordance with the following procedures:

1. Objective of Evaluation

   To confirm that electrical equipment necessary to mitigate the consequences of DBE have been demonstrated, by test, analysis, or operating experience, to have the capability to perform its design

safety function under the environmental conditions of Design Basis Events.

To determine actions that need to be taken to qualify appropriate equipment in accordance with current requirements.

2. <u>Information To Be Provided By Licensees</u>

Licensees will be requested to provide the following information:

A. Identification of safety-related systems and associated electrical equipment located both inside and outside containment which are required to perform a safety function under the environmental conditions resulting from each DBE. Briefly describe the safety function provided by each item of equipment identified. Describe the location of the equipment. Identify any non-safety system, equipment or components, which, if subjected to the environmental conditions associated with a DBE, could affect the safety function of any safety-related system. Identify non-safety grade systems which could perform the function of safety systems by ameliorating the consequences of a DBE and specify electrical components required to assure function of such non-safety grade systems.

B. Definition of the limiting service environmental conditions for operation of the equipment and components identified above. The environmental parameters to be included are pressure, temperature, radiation, submergence, steam, humidity, chemicals, vibration or any combination of the above (seismic conditions are not included in this evaluation but will be considered elsewhere in the SEP). These environmental parameters should be presented as a function of time and the DBE producing the conditions should be identified. The time period during which each item of equipment would be required to operate in a DBE environment should also be identified.

C. Determination of the current status of environmental qualification for safety-related electrical equipment and identification of the supporting documentation. Any evidence of environmental qualification for any environmental condition should be considered and provided.

3. <u>Staff Review of Previously Documented Environmental Qualification</u>

The staff will re-examine any environmental qualifications previously accomplished on these facilities, such as the information submitted by licensees on modifications performed to demonstrate compliance with Appendix K of 10 CFR Part 50.

4. Staff Determination of Plant Environmental Conditions

   The staff will review and verify the environmental conditions provided by licensees for main steam line breaks inside containment, for the limiting loss of coolant accident, and for other DBEs.

5. Site Visit by Staff Review Team

   Early in the review process, staff members will visit the facility to discuss with the licensee the status of his response and to discuss alternatives considered by the licensee for satisfying the environmental qualification acceptance criteria.

6. Identification of Significant Safety Problems

   Following the site visit, the staff will determine if there is inadequate evidence of environmental qualification for any safety-related electrical equipment which must function in a severe environment to mitigate the consequences of a DBE. If such inadequacies are found, appropriate action will be taken to assure no undue risk to the health and safety of the public.

7. Staff Evaluation

   The staff will evaluate the information submitted by the licensee in accordance with Item 2. If necessary, another site visit may be made at this time to clarify or amplify the licensee's submittal.

In the event that documentation of environmental testing does not exist, or is insufficient to assure environmental qualification, the following alternatives will be considered:

A. Evidence of qualification of identical or similar equipment, either in another nuclear facility or by another industry.

B. Importance of the safety function associated with any questionable equipment will be considered. Demonstration of adequate facility response to all DBEs without credit for the function of an unqualified component may be justification for not requiring environmental qualification of a specific component.

C. If important safety equipment is not environmentally qualified, consideration will be given to alternate ways of performing the safety function by using different systems, including the use of non-safety grade systems.

D. Consideration will be given to possible means of protecting components from adverse environmental conditions, such as by enclosing it, coating it or providing other protective features.

E. Other alternatives that may be proposed by the licensee will be considered.

8. Staff Report

A staff report will document the evaluation of this safety topic for each facility. The associated information will be provided

in a format that is compatible with the SEP evaluation procedure, such that the evaluation results can easily be incorporated into the integrated SEP evaluation report for each facility.

APPENDIX C

December 15, 1977

LIST OF PUBLIC RESPONSES

NO.	RESPONDING ORGANIZATION	RECOMMENDATION
1	South Carolina Nuclear Advisory Council	Deny
2	I. Youngheir	Approve
3	Conner, Moore & Corber	Deny
4	Leboeuf, Lamb, Leiby & MacRae	Deny*
5	Natural Resources Defense Council	Approve
6	Arthur L. Reuter, Attorney at Law	Approve
7	Commonwealth Edison Company	Deny
8	Consumers Power Company	Deny
9	North Anna Environmental Coalition	Approve
10	North Anna Environmental Coalition	Approve
11	Power Authority of the State of New York	Deny
12	Carboline (Protective Coatings)	No Position
13	Commonweath of Virginia	Req. Add. Time
14	Center for Law in the Public Interest	Approve
15	Farm Legal Service	Approve
16	Zelia M. Jensen R. N.	Approve
17	SNUPPS	Deny
18	Baltimore Gas and Electric Company	Deny
19	Offshore Power Systems	Deny
20	Aquidneck Island Ecology	Approve
21	Factory Mutual Research	No Position

* Also requested additional time to file additional information.

22	Louise Grenflo	No Position
23	Debevoise & Liberman	Deny
24	Debevoise & Liberman	Deny
25	Portland General Electric Company	Deny
26	Thomas M. Dallito, Attorney at Law	Approve
27	Minnesota Public Interest Research Group	Approve
28	GPU Service Corporation	Deny
29	Concerned Citizens of Tennessee	Approve
30	Sacramento Municipal Utility District	Deny
31	Consolidated Edison Company	Deny
32	Yankee Atomic Electric Company	Deny
33	Tennessee Valley Authority	Req. Add. Time
34	Mid-America Coalition for Energy Alternatives	Approve
35	Toledo Edison	Deny
36	David Winship	Approve
37	Commonwealth Edison Company	Deny
38	Day, Berry & Howard	Deny
39	Arizona Public Service Company	Deny
40	Alabama Power Co.	Deny
41	Gibbs & Hill, Inc.	Deny
42	Gulf States Utilities Company	Deny
43	Rochester Gas & Electric Corp.	Deny
44	ITT - Cannon Electric Division	No position

Appendix D August 17, 1977

INFORMATION REPORT

BY THE

OFFICE OF NUCLEAR REACTOR REGULATION

ON THE

SINGLE FAILURE CRITERION

1. INTRODUCTION

The Single Failure Criterion is just one of several tools applied in systems design and analysis to promote reliability of the systems which are needed in a nuclear power plant for safe shutdown and cooling, and for mitigation of the consequences of postulated accidents. It is not sufficient by itself. Rules of good design practice, such as those required by the ASME Boiler and Pressure Code, IEEE standards, quality assurance requirements and conservatively stipulated design conditions must also be utilized to ensure that high quality and highly reliable systems, components and structures are provided.

The Single Failure Criterion, as a design and analysis tool, has the direct objective of promoting reliability through the enforced provision of redundancy in those systems which must perform a safety-related function. Simply stated, application of the Single Failure Criterion requires that a system which is designed to perform a defined safety function must be capable of meeting its objectives assuming the failure of any major component within the system or in an associated system which supports its operation.

The Single Failure Criterion was developed without the benefit of numerical assessments on the probabilities of component or system failure. However, in applying the Criterion, it is not assumed that any conceivable failure could occur. For example, reactor vessels or certain types of structural elements within systems, when combined with other unlikely events, are not assumed to fail because the probabilities of the resulting scenarios of events are deemed to be sufficiently small that they need not be considered. In general only those systems or components which are judged to have a credible chance of failure are assumed to fail when the Single Failure Criterion is applied. Such failures would include, for example, the failure of a valve to open or close on demand, the failure of an emergency diesel generator to start or the failure of an instrument channel to function. A single failure can also be a short circuit in an electrical bus that results in the failure of several electrically operated components to function.

The Single Failure Criterion, through enforced provision of redundancy, does not give absolute assurance of reliability. The Reactor Safety Study (WASH-1400) indicates that application of the Single Failure Criterion to the plants that were studied did provide an acceptable degree of hardware redundancy for most systems. However, the Reactor Safety Study also pointed out that factors such as systems interactions, multiple human errors, and maintenance and testing requirements also have an influence on reliability. Such factors fall outside the scope of the Single Failure Criterion, and supplementary methods must be utilized in their study.

At the present time, the Single Failure Criterion is codified in Appendix A to 10 CFR 50 (General Design Criteria) and in Appendix K (ECCS Evaluation Models); in addition, 10 CFR 50.55a (Codes and Standards) makes mandatory the use of the ASME Code and of IEEE Std 279 which contains the Single Failure Criterion. Further interpretation and guidance on the application of the Single Failure Criterion is given in the Standard Review Plan and Regulatory Guides (e.g., Standard Review Plan Section 3.6.1 describe its application in the event of postulated piping failures outside containment, and Regulatory Guide 1.53 endorses IEEE Std 379 which describes in detail how the Single Failure Criterion defined in IEEE Std 279 is applied to electrical and instrumentation systems).

2. IMPORTANT ELEMENTS OF THE SINGLE FAILURE CRITERION

A. The Concept

In principle, the Single Failure Criterion is straightforward. Simply stated it is a requirement that a system which is designed to carry out a defined safety function (e.g., an Emergency Core Cooling System) must be capable of carrying out its mission in spite of the failure of any single component within the system or in an associated system which supports its operation. Application of the concept is complicated by the interrelationships between the various fluid and electrical systems and their supporting auxiliaries in a nuclear power plant. Furthermore, there is a need to stipulate the events and associated assumptions which must be considered during application of the Single Failure Criterion.

Application of the Single Failure Criterion involves a systematic search for potential single failure points and their effects on prescribed missions (i.e., Failure Modes and Effects Analysis). Such a search is required by our Standard Review Plan and the Standard Format for the Content of Safety Analysis Reports for specified safety systems and components. The objective is to search for design weaknesses which could be overcome by increased redundancy, use of alternate systems or use of alternate procedures.

B. Definition of Single Failure

Single failure is defined in 10 CFR 50 Appendix A As follows:

> "A single failure means an occurrence which results in the loss of capability of a component to perform its intended safety functions. Multiple failures resulting from a single occurrence are considered to be a single failure. Fluid and electric systems are considered to be designed against an assumed single failure if neither (1) a single failure of any active component (assuming passive components function properly) nor (2) a single failure of a passive component (assuming active components function properly), results in a loss of capability of the system to perform its safety functions."

A footnote to this definition states that "single failures of passive components in electric systems should be assumed in designing against a single failure." This means that for electric systems no distinction is made between failures of active and passive components and all such failures must be considered in applying the Single Failure Criterion. For example, short circuits in electrical cables must be considered even though a short circuit could be regarded as a failure of a passive component.

With regard to passive components in fluid systems, the footnote further states, "The conditions under which a single failure of a passive component in a fluid system should be considered in designing the system against a single failure are under development."

While considerable progress has been made in defining the nature of passive component failures which should be considered in the licensing review process, no change to the regulation has been made since 1969. In application of the Single Failure Criterion to fluid systems, Section 6.3 of the Standard Review Plan requires consideration of passive failures in the Emergency Core Cooling System during the recirculation cooling mode following emergency coolant injection, but does not define the nature of such failures. Other interpretations of the Criterion for passive components have been made on the basis of detailed engineering evaluations conducted during licensing reviews, but with some staff disagreement. For example, NUREG-0138 (Issue 7) has a detailed discussion of passive failures following a Loss of Coolant Accident, and NUREG-0153 (Issue 17) has a detailed discussion of passive type valve failures. This subject is also summarized in Section 4 below and the status of standards development pertinent to this subject is summarized in Section 6. The following definitions of single active and passive failures in fluid systems important to safety are pertinent to the discussion of the Single Failure Criterion.

C. Active Failure in a Fluid System

An active failure in a fluid system means (1) the failure of a component which relies on mechanical movement for its operation to complete its intended function on demand, or (2) an unintended movement of the component. Examples include the failure of a motor- or air-operated valve to move or to assume its correct position on demand, spurious opening or closing of a motor- or air-operated valve, or the failure of a pump to start or to stop on demand. In some instances such failures can be induced by operator error.

D. Passive Failure in a Fluid System

A passive failure in a fluid system means a breach in the fluid pressure boundary or a mechanical failure which adversely affects a flow path. Examples include the failure of a simple check valve to move to its correct position when required, the leakage of fluid from failed components, such as pipes and valves--particularly through a failed seal at a valve or pump--or line blockage. Motor-operated valves which have the source of power locked out are allowed to be treated as passive components.

In the study of passive failures it is current practice to assume fluid leakage owing to gross failure of a pump or valve seal during the long-term cooling mode following a LOCA (24 hours or greater after the event) but not pipe breaks. No other passive failures are required to be assumed because it is judged that compounding of probabilities associated with other types of passive failures, following the pipe break associated with a LOCA, results in probabilities sufficiently small that they can be reasonably discounted without substantially affecting overall systems reliability.

It should be noted that components important to safety are designed to withstand hazardous events such as earthquakes. Nevertheless, in keeping with the defense in depth approach, the staff does consider the effects of certain passive failures (e.g., check valve failure, medium or high energy pipe failure, valve stem or bonnet failure) as potential accident initiating events.

3. APPLICATION OF THE SINGLE FAILURE CRITERION

As noted previously, the events and associated assumptions which are considered in connection with application of the Single Failure Criterion must be defined for specific systems. The basic events and assumptions are defined in the General Design Criteria.

A variety of design basis events which initiate a requirement for safety system action must be considered in the overall safety evaluation of a plant. In general, each of these initiating events requires an assessment of the equipment damage that could occur as a direct consequence of the event. The Single Failure Criterion is applied to those systems which must function after consequential equipment failures have been taken into account.

The General Design Criteria make it clear that for electrical, instrumentation and control systems, application of the Single Failure Criterion to systems evaluation depends not only on the initiating event that invokes safety action of these systems, together with consequential failures, but also on active or passive electrical failures which can occur independent of the event. Thus, evaluation proceeds on the proposition that single failures can occur at any time.

In contrast, for various fluid systems the General Design Criteria require that the safety function be accomplished in the face of certain conservative assumptions in addition to application of the Single Failure Criterion. In general, these assumptions involve (1) the unavailability of offsite or onsite power and (2) the postulated initiating failure. In the case of a loss of coolant accident, for example, it is first assumed that a primary system pipe rupture occurs with consequential blowdown of primary coolant.

Simultaneous with the pipe rupture, it is assumed that only the offsite power source or the onsite emergency power source is available.(1) These assumptions are applied in addition to the Single Failure Criterion which is applied to the aggregate of systems required to fulfill each specific safety function.

The manner in which the Single Failure Criterion is currently applied to various specific classes of safety related systems is outlined below.

A. Electrical, Instrumentation and Control Systems

The general interpretation and application of the Single Failure Criterion to electrical, instrumentation and control systems is stated in IEEE Std 379 as follows:

> "The system shall be capable of performing the pro[illegible]ive actions required to accomplish a protective function in the presence of any single detectable failure within the system [this is the "single failure"] concurrent with all identifiabl., but non-detectable failures, all failures occurring as a result of the single failure, and all failures which would be caused by the design basis event requiring the protective function."

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(1) Successful emergency systems performance must be demonstrated with either offsite or onsite power, assuming a single failure.

Therefore, in the analysis to determine if a particular electrical, instrumentation or control system meets the Single Failure Criterion the following postulates are made:

(1) First, the particular design basis event or accident is postulated to occur, along with any related or consequential failures that could result from it.

(2) Then, the analysis assumes the presence of all identifiable failures which cannot be detected or tested in the design or which are not in fact subject to surveillance tests as set forth in the Technical Specifications.

(3) Finally, the presence of a single additional detectable failure is assumed in assessing the capability of the system to provide the necessary protection for the design basis event.

Analyses are performed in this manner to demonstrate the adequacy of the electrical, instrumentation and control systems design over the full range of postulated design basis events or accidents and worst case single failures.

There is a special interpretation of the Criterion (Section 4.7 of IEEE Std 279) which specifically addresses designs in which safety-related instrumentation or controls are also used to provide inputs to non-safety related plant control systems. In such a design it is required that where a single random failure in the safety-related system can cause a control system action that results in a generating station condition requiring protective action and can also prevent proper action of a protection system channel designed to protect against the condition, the remaining redundant protection channels shall be capable of providing the protective action even when degraded by a second random failure. This special interpretation of the Single Failure Criterion is specific for the design cited above, and it is not applied to safety-related electric power systems.

The general interpretation of the Single Failure Criterion is applicable to safety-related electric power systems. However, the offsite power system is an exception. The specific requirements of General Design Criterion 17 take precedence over the rigorous application of the Single Failure Criterion; i.e., an offsite power system comprised of one delayed access circuit and one immediate access circuit is deemed acceptable. The basis for this position is that a second immediate access circuit would not significantly improve the availability of offsite power at the emergency buses. This has been established by an analysis using reliability data and not the Single Failure Criterion.

B. Emergency Core Cooling Systems

In applying the Single Failure Criterion to Emergency Core Cooling Systems which must function following postulated loss of coolant accidents, the requirements of General Design Criterion 35 - Emergency Core Cooling - are followed. Therein it is stipulated that following a postulated loss of coolant accident, suitable redundancy in equipment shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the ECCS safety function can be accomplished, assuming the most limiting additional single failure. Appendix K to 10 CFR 50 requires that the only ECCS subsystems to be assumed available are those operable after the most damaging additional single failure of ECCS equipment has taken place. Selection of the single failure to be applied to the emergency core cooling system is made independent of the size or location of the postulated pipe break in the reactor coolant system. Thus, for each postulated pipe break, that single failure which results in minimum emergency core cooling performance is considered in judging the adequacy of the system. For example, this could be failure of a component in a redundant ECCS subsystem or the loss of an emergency diesel generator in addition to the loss of all offsite power.

During the short-term ECCS coolant injection mode immediately following a loss of coolant accident, the most limiting single active failure is considered in evaluating systems performance capability.

During the long-term ECCS recirculation cooling mode the most limiting active failure, or a single passive failure equal to the leakage that would occur from a valve or pump seal failure, is assumed. The basis for not including other passive failures during the long term is based on engineering judgment that such failures (pipe or valve breaks) have an acceptably low likelihood of occurrence during the long-term phase of a loss-of-coolant accident. Analyses of ECCS performance in WASH-1400 indicate that passive failures of valves and piping are relatively small contributors to ECCS unavailability during both the injection and recirculation modes of operation.

C. Containment Heat Removal and Cleanup Systems

General Design Criterion 38 - Containment Heat Removal - requires the provision of a system to rapidly reduce containment pressure and temperature following any LOCA. While current practice is to apply only an active component failure to the evaluation of the performance of these systems, component redundancy ensures their availability even in the presence of some possible passive failures.

General Design Criterion 41 - Containment Atmosphere Cleanup - requires systems to control fission products, hydrogen, oxygen, and other substances which may be released into containment. These systems must be capable of functioning with either onsite or offsite power. Contaminants can enter the containment due to a variety of events, such as a LOCA. The Single Failure Criterion is applied subsequent to the postulated event and, in evaluating these systems, only active failures are considered, except in instances where components may be shared with ECCS systems. In such cases, the possibility of seal leakage is considered in the long-term ECCS recirculation mode.

D. Residual Heat Removal System

The capability for residual heat removal must be available using onsite or offsite power, assuming an additional single failure. To accommodate certain single failures, for the older class of plants, the staff has accepted use of the auxiliary feedwater system as a backup to the residual heat removal system. For current designs, the residual heat removal system has been modified to include additional piping and valves such that the system now has additional flexibility to perform its function even after a wide range of possible single failures. Also, as part of current staff reviews, certain initiating events have been postulated which are related to the Single Failure Criterion. These events involve application of the pipe break criteria for moderate energy lines located outside of containment as described in Standard Review Plan 3.6.1. Thus, the staff applies a limited passive failure as an initiating event for the residual heat removal system. For this event, no additional single failure is applied to the Residual Heat Removal System.

E. Ultimate Heat Sink

General Design Criterion 44 - Cooling Water - requires a system to transfer heat from systems, structures, and components important to safety to an ultimate heat sink under normal operating and accident conditions. The system must be capable of carrying out its function using either onsite or offsite power assuming any single failure. The requirements of the Single Failure Criterion are applied in a manner similar to that which is applied to residual heat removal systems.

F. Containment Piping Penetrations

Requirements for isolation valves on containment penetrations are defined in the General Design Criteria. The requirements anticipate the possibility of single active failure of isolation valves in each line by requiring double barriers. The Single Failure Criterion is also applied to the plant protection devices which initiate automatic closure of such isolation devices.

4. PROBLEMS THAT HAVE BEEN ENCOUNTERED IN THE APPLICATION OF THE SINGLE FAILURE CRITERION

A. Additional Passive Failures

As stated previously, there is a footnote in the General Design Criteria that the conditions under which single passive failures should be considered in applying the Single Failure Criterion to fluid systems are under development. That footnote was included when the Criteria were published in 1969. During subsequent years staff assumptions regarding the nature of passive failures which should be considered have not been completely consistent and there has been some disagreement. However, on the basis of the licensing review experience accumulated in the period since 1969, it has been judged in most instances that the probability of most types of passive failures in fluid systems is sufficiently small that they need not be assumed in addition to the initiating failure in application of the Single Failure Criterion to assure safety of a nuclear power plant. This opinion appears to have been verified by the Reactor Safety Study. Nevertheless, it is receiving further study.

In some licensing review areas, the staff does impose a passive failure in addition to the initiating event, while in others it does not. As previously mentioned, an example of the application of a passive failure requirement is the approach to long-term recovery subsequent to a loss-of-coolant accident. Applicants are required to consider degradation of a pump or valve seal and resulting leakages in addition to the initiating failure (LOCA). The rationale for applying this type of failure is a recognition of the relatively extended periods of required operation of systems that are expected to be on a standby status throughout the plant life. The likelihood of accelerated wear of such components as pump and valve seals would be increased after the adverse conditions following a LOCA. Extended operation during the long term (up to months) requires that these types of failures be considered in designing the plant. The basis for excluding additional passive piping failures is elaborated in detail in NUREG-0138, Issue 7. Other examples of passive failure considerations are presented in Section 4.B.

B. Valve Failures

A variety of valve functions and valve types exist in each nuclear plant. Valve functions include isolating flow, controlling flow, admitting flow, and preventing flow reversal. Valve types include those that are electrically controlled and operated, electrically controlled and air operated, manually controlled and operated, manually controlled and electrically operated, spring operated, and self actuated (check valves).

Accordingly, a variety of failure modes can be postulated for valves within the application of the Single Failure Criterion. Certain passive-type valve failure modes have occurred (for example, dropping of a valve disc). This has resulted in a reevaluation of postulated valve failures. NUREG-0153 (Issue 17) concludes that while the staff does not consider that changes in safety criteria are warranted at this time, ongoing efforts regarding the probability and effects of various valve failure modes will seek to compile a more rigorous data base and will apply such information to plant safety analyses. This effort has been classed as a Category B generic task.

C. Electrical Failures

In order to provide an electrical, instrumentation and control system design to satisfy the Single Failure Criterion, redundancy is included. The degree of redundancy (i.e., the number of "independent" divisions of equipment) depends on many design considerations. Provisions are typically included to prevent the initiating event from affecting the electrical, instrumentation and control systems.

If it is postulated that the failure of a portion of the safety-related electrical, instrumentation and control systems is the initiator of a design basis event, then the general interpretation of the Single Failure Criterion, discussed in Section 3.A, is not applicable to the remaining portions of the system. In such cases supplementary analyses are relied upon to evaluate the reliability of the systems in question.

In the case of the current issue on the reliability of the safety-related direct current power systems as raised by an ACRS consultant, the postulated initiating event is failure of one division of a two division system. However, this DC power system design does meet the general interpretation of the Single Failure Criterion, but it is not covered by the special interpretation noted in Section 3.A for specific safety-related instrumentation and control systems. Therefore, the staff evaluation of this issue, summarized in NUREG-0305, was based upon reliability data and not the Single Failure Criterion. It was concluded that the likelihood of occurrence of the postulated sequence of events is low enough to permit continued operation and licensing of plants pending further assessments. It is possible that new requirements to assure greater reliability of DC power systems may result from the ongoing study. It is a Category A generic task.

D. Classification of Events

Recent staff work related to issues raised in dissent or pertaining to reactor transient event classifications and consequence criteria has disclosed some confusion on how to handle certain infrequent transients which do not have public consequences as severe as "accidents". The confusion stems primarily from the differences in event classification from vendor to vendor, among standards writing bodies and within NRC. A study is underway within the Reactor Systems Branch to develop a "unified" event classification scheme. It is expected to be completed in early 1978. While this study is not aimed at application of the Single Failure Criterion, it is expected that for some events it will bring into sharper definition the circumstances under which the Criterion should or should not be applied. For example, a moderate frequency transient such as a feedwater malfunction is routinely analyzed in Safety Analysis Reports. An additional single failure concurrent with the feedwater malfunction may result in a compound event which, because of the multiple failures, has a lower probability and therefore a different classification. Less stringent acceptance criteria may then be appropriate. The above study will examine such additional single failures as they apply to acceptance criteria for transients and accidents. This study has been classed as a Category B generic task.

E. Operator Error

An operator error could cause an active single failure, such as an inadvertent valve closure. In many instances consideration of such single operator errors is given in licensing reviews; however, the degree to which any given operator error is considered reasonably equivalent to the likelihood of a single active failure is based on judgments made concerning the situation. For example, in studying the effects of an operator error of "omission" (failure to perform an action), if there is time to bring a system on line through remedial operator action, reliance on such action is permitted. On the other hand, in cases where rapid actuation of engineered safety systems is required, the actuation is required to be automatic and operator independent.

Increasing attention is being given to human reliability in an effort to adopt more definitive criteria for the role of the operator in mitigating the consequences of transients or accidents. A Regulatory Guide is currently being developed in conjunction with staff review of the proposed Standard ANSI-N660, "Proposed ANS Criteria for Safety-Related Operator Actions." Increasing activities in human reliability will assist the staff in developing a more rigorous basis for assessing operator involvement in plant safety.

5. INSIGHTS OF THE REACTOR SAFETY STUDY RELATIVE TO THE SINGLE FAILURE CRITERION

The Reactor Safety Study (WASH-1400) assessed a pressurized water and a boiling water reactor design. The Single Failure Criterion had been applied in the design and Regulatory review processes for these plants, generally as outlined in the preceding sections. Although the Single Failure Criterion is not a quantitative design and analysis tool, the numerical assessments in the Reactor Safety Study indicate that its application, through enforced provision of component and systems redundancy, has made an important and necessary contribution to the overall reliability of nuclear plant safety systems. The assessments in the Reactor Safety Study also indicate that supplementary methods of analysis must be utilized to study effects on reliability which are beyond the scope of the Single Failure Criterion. The principal insights gained from this study are briefly summarized below:

(1) Application of the Single Failure Criterion has led to a suitable level of hardware redundancy in most systems. The level of redundancy thus provided has, for many safety systems, resulted in systems reliability being controlled by such factors as human and operational interactions (i.e., human errors, test and maintenance downtimes, test intervals) rather than potential single design failures as defined in the Single Failure Criterion.

Quantitative optimization of reliability in terms of such non-hardware factors would require the review of information beyond that now considered in the licensing process.

(2) The Single Failure Criterion must be supplemented by methods and criteria in the area of common mode assessments if improved reliability characteristics for safety systems are necessary. Although the effects of common mode failure are not now quantitatively considered in licensing safety reviews, considerable attention is given to reducing the potential for the occurrence of common mode failures through stringent application of high-quality design and quality assurance requirements to various components. For example, considerable attention is given to reducing the potential for multiple electrical relay failures such as might arise from a generic design defect in components supplied by a single manufacturer.

(3) The probability of accident sequences resulting in core meltdown were found by the RSS to be importantly influenced by system to system interactions and by functional dependencies between systems. These functional dependencies can be considered as a class of interactions where the functioning of one system depends on satisfactory functioning of another system. Redundancy of components within systems, mandated by the Single Failure Criterion, does not ameliorate the functional dependence. Thus, application of the Single Failure Criterion requires supplemental methods and use of an integrated systems approach to identify such functional dependencies if it is desired to further reduce accident risk.

6. ACTIVITIES RELATED TO CLARIFYING AND IMPROVING APPLICATION OF THE SINGLE FAILURE CRITERION

A number of technical activities by various nuclear industry groups and by the Offices of Standards Development and Nuclear Reactor Regulation are underway, which will have an effect on system reliability requirements and the use of the Single Failure Criterion. These are summarized in this section.

In late 1971 the American Nuclear Society initiated a standards writing effort with the objective of setting forth a clear, detailed set of criteria for application of the Single Failure Criterion to fluid systems. In 1975 the resulting Standard was issued as "ANSI N658 - Single Failure Criteria for PWR Fluid Systems." In November of 1976, the Office of Standards Development initiated a task to draft a Regulatory Guide endorsing the Standard, with appropriate exceptions, for both PWRs and BWRs. The staff review of this Standard disclosed several deficiencies which relate primarily to inconsistencies with current regulatory practice and to areas in which staff application of the Single Failure Criterion is not yet fully defined. For example: (1) literally applied to "postulated pipe breaks outside containment," the Standard would make no exception for certain dual purpose moderate energy systems (e.g., service water systems) as presently provided in Standard Review Plan 3.6.1; (2) some passive failures would be treated as active failures (e.g., check valves) contrary to staff practice; and, (3) event categorization is not consistent with current staff interpretation. Nevertheless, ANSI-N658 represents a significant step toward achieving satisfactory criteria for application of the Single Failure Criterion to fluid systems, and it is expected that a Regulatory Guide could be issued in mid-1978.

IEEE Std 379 was issued in 1972 as a Trial-Use Guide for the Application of the Single Failure Criterion to Electrical, Instrumentation and Control Systems and its application was endorsed in Regulatory Guide 1.53. IEEE Std 379 was recently updated and reissued. The subcommittee which prepared the Standard is currently working to develop definitive guidance on application of the Single Failure Criterion to shared systems and to single operator errors. When this work is completed it is expected that Regulatory Guide 1.53 will be revised to endorse these added requirements.

Earlier this year, the Office of Nuclear Reactor Regulation initiated a formal system providing for continuing management oversight and attention to generic safety-related technical activities. A number of these generic activities may include clarification of the conditions under which the Single Failure Criterion should be applied. The Category A activities expected to include single failure considerations are:

(1) Anticipated Transients Without Scram;

(2) Non-Safety Loads on Class IE Power Supplies;

(3) Adequacy of Safety-Related d.c. Power Supplies;

(4) Reactor Vessel Pressure Transient Protection;

(5) Steam Line Breaks;

(6) RHR Shutdown Requirements;

(7) Systems Interaction; and

(8) Generic Accident Risk Study

(9) Snubbers

The Category B activities expected to include single failure considerations are:

(1) Event Categorization;

(2) ECCS Reliability;

(3) Locking Out of ECCS Power Operated Valves;

(4) Protection Against Postulated Piping Failures in Fluid Systems Outside Containment;

(5) Criteria for Safety-Related Operator Actions;

(6) Passive Mechanical Failures; and

(7) Allowable ECCS Equipment Outage Periods

In some cases these activities are being conducted to evaluate adequacy of previous staff positions, while in others some new provisions may result. The single failure aspects of these activities will be utilized as appropriate in connection with improving application of the Single Failure Criterion.

The NRR staff is developing a plan for incorporating risk assessment methodology into the licensing process. Because of manpower limitations, and the need to train an initial cadre in risk assessment methodology and to carefully weigh impacts of its application, it is expected that application of risk assessment methodology to the licensing process would necessarily increase gradually over a period of several years. It is not expected that risk assessment methodology will come into large-scale systematic use in the near future as a replacement for the Single Failure Criterion as it is now applied. It is expected, however, that reliability engineering and probabilistic methodologies, together with an expanding data base on component and systems failure rates, will be applied to specific studies pertaining to reliability requirements and evaluations that go beyond the Single Failure Criterion. The current study of the adequacy of DC power supplies is an example of such an application.

7. SUMMARY CONCLUSIONS

Application of the Single Failure Criterion as it is presently defined in the regulations, Standard Review Plan, and various Regulatory Guides and industry standards has led to a generally acceptable level of hardware redundancy in most electrical, control and instrumentation systems and in fluid systems important to safety. As indicated by the Reactor Safety Study, systems unavailabilities are controlled to a large extent by factors such as operator errors, systems interactions, and maintenance and testing requirements, rather than by inadequate hardware redundancy. Some problems exist in specific interpretations and applications of the Single Failure Criterion and these are receiving staff attention. It is the considered judgment of the staff that the Single Failure Criterion should continue to be applied subject to resolution of specific problem areas currently defined and under study, pending any long-term wide-scale incorporation of reliability and risk assessment methodology into the licensing process.