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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

JUN 1 1 1979

Docket No. 50-344

Mr. Lynn Frank, Director
Oregon Department of Energy
Labor and Industries Building
Room 111
Salem, Oregon 97310

Dear Mr. Frank:

This is in response to the Oregon Department of Energy's letter of March 1, 1979, wherein several questions were raised concerning generic safety issues and the seismic, environmental, and radiation qualification of electrical equipment at Trojan Nuclear Plant.

Taking the questions in the order that you posed them, the following five Westinghouse Topical Reports were referenced in the Trojan FSAR and contained information intended to support the issuance of the operating license for Trojan:

WCAP-7821, Seismic Testing of Electrical and Control Equipment (High Seismic Plants), December 1971

WCAP-7744, Environmental Testing of Engineered Safety Features Related Equipment (NSSS - Standard Scope), August 1971

WCAP-7672, Solid State Logic Protection System Description, June 1971

WCAP-7819 Revision 1, Test Report, Nuclear Instrumentation System Isolation Amplifier, January 1972

WCAP-7705, Engineered Safeguards Final Device or Actuator Testing, March 1973

The present generic review status and the Trojan review status of these WCAPs is as follows:

WCAP-7821

Generically, this report is acceptable provided that (1) justification of the Eagle Signal timer used is provided and (2) all output relays in the Solid-State Protection System for all high seismic plants are replaced with the qualified relays.

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The Trojan design does not use a Westinghouse-supplied Eagle Signal timer. Therefore, proviso (1) is not applicable to Trojan. PGE has replaced all output relays in the Solid-State Protection System with the qualified rotary-type relays. Therefore the Trojan design meets proviso (2) and is acceptable.

WCAP 7744*

Generically, this report was found to be acceptable provided that Westinghouse supplied satisfactory answers to the following four items:

(1) Verification that the deviations in accuracy and time of failure noted in the test results are within the specified time and accuracy required in the accident analysis for each specific plant.

(2) Identification of those instruments inside containment required to follow the course of Condition III and IV** events and verification of the capability of each instrument so identified, together with recommendations for a replacement instrument model for those not capable of long-term monitoring.

(3) Westinghouse has indicated that additional instrumentation located outside of containment is available to the operator to follow the course of Condition III and IV events. Westinghouse has been requested to identify this instrumentation and its capability for each specific event and plant referencing this WCAP.

(4) Confirmation that the differential pressure transmitters are temperature-compensated and that deviations are within that required for each specific application.

Westinghouse was requested to provide this information in a letter dated January 15, 1979. Their response is enclosed. Since Westinghouse did not supply answers to these questions and has withdrawn its reliance on WCAP-7410L, we have requested that PGE meet with us to provide answers to these questions as they relate to Trojan and discuss the qualification of replacement transmitters which we understand they have ordered. We expect that this matter will be resolved at this meeting or shortly thereafter.

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*Non-proprietary. The proprietary version of this document is designated WCAP-7410L.

**These are defined in Trojan FSAR, pp. 15.3-1 and 15.4-1.

Additional background regarding environmental qualification of electrical equipment is contained in the enclosed memorandum to the Commissioners dated March 15, 1979, and NUREG-0413, "Staff Report on the Environmental Qualification of Safety-Related Electrical Equipment", February 1978.

WCAP-7672 and WCAP-7819

Generically, the staff has found these topical reports acceptable. However, these reports do not address environmental and seismic qualification of the Solid-State Protection System and the nuclear instrumentation system isolation amplifier. These equipments have been seismically qualified under WCAP-7821 (see status of WCAP-7821 above). In regard to environmental qualification, Westinghouse has documented that these equipments have been qualified to operate at 120°F and 95% humidity.

For Trojan, the Solid-State Protection System and nuclear instrumentation system cabinets are located in the control room. The maximum temperature and humidity of the control room are 110°F and 80% humidity respectively. These values are well within the qualified design limits for these equipments. Therefore, these topical reports are acceptable for Trojan.

WCAP-7705

Generically, this report was found unacceptable. However, this report is not applicable to the Trojan Plant. As stated in the Trojan Safety Evaluation Report (SER), Sections 7.3.1 and 7.3.2 dated October 1974, the design of the ESF actuation system was found acceptable. This conclusion was reached without reliance on Topical Report WCAP-7705, "Engineered Safeguard Final Device or Actuator Testing", which was referenced in the FSAR but was found unacceptable by the staff as a basis for establishing conformance to safety criteria pertaining to ESF final actuator testability. However, our evaluatio of the ESF actuation system was based on our review of the Trojan FSAR, the schematic drawings of the circuitry used to initiate operation of ESF components, and on our prior review (on the D. C. Cook docket) of identical ESF actuation logic. Therefore, the fact that WCAP-7705 was found unacceptable does not constitute an unreviewed safety issue for Trojan.

For the balance-of-plant equipment, we find no indication of any Class 1E equipment that is not environmentally qualified for Trojan. However, PGE did identify unqualified electrical splices found on three pressurizer level transmitters (LER #78-027). These splices were replaced with

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qualified components. This licensee event report was submitted in response to IE Circular 78-08 dated May 31, 1978, which directed all licensees to examine all installed safety-related electrical equipment and determine that proper documentation exists which provides assurance that the equipment will function under postulated accident conditions. In essence, the intent of that circular was to highlight to all licensees important lessons learned from environmental qualification deficiencies reported by individual licensees.

Further, by IE Bulletin No. 79-01, all licensees with operating power reactors including PGE, have been requested to expedite completion of the re-review program described in IE Circular 78-08. This was done because inspections conducted by NRC with respect to responses to Circular 78-08 had identified components which licensees either have found to be unqualified for service within the LOCA environment or which do not have documentation of such qualification. The effect of Bulletin 79-01 was to raise the threshold of IE Circular 78-08 to the level of a Bulletin, which would require licensee response. Bulletin 79-01 in part requires licensees of all operating power reactor facilities to provide written evidence of the qualification of electrical equipment required to function under accident conditions and to complete the re-review program described in Circular 78-08. PGE's response to this Bulletin is expected in June 1979. This response should resolve any concerns regarding environmental qualification of all instrumentation and electrical equipment for Trojan.

With regard to compliance of the Trojan environmental qualification program with IEEE-323-1974 as endorsed and modified by Regulatory Guide 1.89, the acceptance criteria for the qualification of Class IE equipment for Trojan did not include the aging consideration specified in IEEE Standard 323-1974 and Regulatory Guide 1.89 (which endorses IEEE-323-1974).

In 1979, during the deliberations of the NRC's Regulatory Requirements Review Committee on the implementation of Regulatory Guide 1.89, consideration was given to the incremental improvements to safety it afforded in comparison of the then-current staff review practice. The Committee recommended that the guide be applied only to future CP applications; i.e., it should not be backfitted. The decision was based on the Staff's judgment that the incremental improvements were not significant to safety and that full implementation of IEEE-323-1974 required the further development of other ancillary standards to provide guidance on specific safety-related equipment and components. Subsequent public comments and review by the ACRS did not alter the recommendation concerning implementation of Regulatory Guide 1.89.

We recognize that additional guidance is needed in the area of accelerated aging techniques used to establish a qualified life for electrical equipment and assemblies. Our Category A technical activity on equipment qualification (Task Action Plan A-24) and an NRC extensive research program being carried out at Sandia Laboratories are intended to provide additional guidance for the development of test methods and licensing review procedures on aging. These programs will also allow us to make informal judgments regarding the effects of aging. In addition, as part of the Staff's Systematic Evaluation Program (SEP), the staff is assessing the surveillance and maintenance records for equipment inside and outside of containment of eleven selected older plants. Since this equipment has been effectively "aged", the assessment of these records should provide additional information on the effects of aging.

Following completion of these ongoing activities -- the Task Action Plan A-24, the NRC research program, and the SEP effort -- we will reconsider our position on the need for backfitting the aging requirements. At that time, should we deem it necessary, we will take appropriate steps to ensure that aging effects are considered in assessing the adequacy of Class IE equipment used in the Trojan Plant. It is our judgment that the natural aging that the Class IE equipment will undergo in the period to this reassessment will have little effect on its environmental or seismic capability.

Regarding your question about the asymmetric blowdown load generic issue, acceptance criteria are currently being developed as part of generic Task A-2, "Asymmetric Blowdown Loads on Reactor Primary Coolant System". The program for resolution of this task (Task Action Plan) is contained in NUREG-0371, "Task Action Plans for Generic Activities". Acceptance criteria are scheduled to be established by early summer. The NRC staff has developed guidelines for load combinations. These are described in NUREG-0484, "Methodology for Combining Dynamic Responses". We are currently evaluating break area and break opening time assumptions. The Trojan analysis (PGE-1014) will be reviewed with respect to these generic criteria, and we expect that an evaluation for Trojan will be made later this summer. The basis for continued operation of affected plants is contained in Section 3 of the Task Action Plan.

The NRC staff recommendations for ATWS resolution are described in Volume 3 of NUREG-0460. It is expected that a proposed rule on ATWS will be presented to the Commission for approval this summer after completion of ACRS review. We expect the rule to be effective early next year. The recommended modifications for Trojan class of plants (operating Westinghouse plants) are changes necessary to provide ATWS mitigating system actuation circuitry satisfying the criteria in Appendix C of NUREG-0460 Volume 3. The rulemaking process would ultimately determine the need for these plant changes and provide an implementation plan.

With respect to fire protection, PGE has proposed to provide a safe shutdown capability independent of the cable spreading room and control room. This shutdown capability does not involve a new system. It consists of design modifications to existing safe shutdown systems which allow their use to safely shut down the plant even if cables now located in the control room or cable spreading room were lost in a fire. The independent safe shutdown capability was not a requirement of the NRC; it was chosen by the licensee as an alternative to upgrading the existing fire suppression system in the cable spreading room and the control room to provide the level of defense which the staff considered acceptable for these areas to preserve shutdown capability. The present fire protection is provided by (1) administrative controls over ignition sources and combustibles, (2) automatic sprinkler system (cable spreading room), (3) smoke detection and manual fire suppression with hoses and portable extinguishers, and (4) physical separation and marinite barriers between divisions of redundant safe shutdown cabling.

For most fires, any one of these levels of defense is adequate to prevent loss of safe shutdown capability. However, for added conservatism we postulate that both (1) and (2) fail; and therefore, the fire may become larger, in which case, (3) must be adequate to prevent loss of redundant safe shutdown equipment. For Trojan, the staff had initially required (for the cable spreading room) a more sophisticated prompt-acting automatic system; i.e., directed water spray with open head at intermediate levels to cover all trays and flame retardant coatings and fire barriers, and an automatic halon system for control room cabinets containing redundant safe shutdown systems. As an alternative to upgrading the sprinkler system the licensee proposed, in addition to the existing protection and additional detection in control room cabinets, to provide an alternate shutdown capability independent of these areas. The criteria for these existing systems as modified are the same as that prior to their modification. The modifications to these existing systems are not to degrade the original design but simply to allow the flexibility of operation of at least one shutdown capability independent of damage in these areas.

In response to the second part of the request as to why interim operation is acceptable without the alternate capability, the following basis is provided.

In the report of the Special Review Group on the Browns Ferry Fire (NUREG-0050) dated February 1976, consideration of the safety of operation of all operating nuclear power plants pending the completion of our detailed fire protection evaluation was presented. The following quotations from the report summarize the basis for the Special Review Group's conclusion that the operation of these facilities need not be restricted for public safety.

> "Fires occur rather frequently, however, fires involving equipment unavailability comparable to the Browns Ferry Fire are quite infrequent (see Section 3.3 of NUREG-0050). The Review Group believes that steps already taken since March 1978 (see Section 3.3.2) have reduced this frequency significantly.
>
> "Based on its review of the events transpiring before, during and after the Browns Ferry Fire, the Review Group concluded that the probability of disruptive fires of the magnitude of the Browns Ferry event is small and that there is no need to restrict operation of nuclear power plants for public safety. However, it is clear that much can and should be done to reduce even further the likelihood of disabling fires and to improve assurance of rapid extinguishment of fires that occur. Consideration should be given also to features that would increase further the ability of nuclear facilities to withstand large fires without loss of important functions should such fires occur."

We recognize that the "Risk Assessment Review Group Report to the U. S. Nuclear Regulatory Commission" NUREG/CR-0400 (The Lewis Committee Report) states that this Review Group is unconvinced of the correctness of the WASH-1400 conclusion that fires contribute negligibly to the overall risk of nuclear plant operation.

It is our conclusion that the operation of the Trojan facility, pending the implementation of all facility modifications including the alternate shutdown capability, does not present an undue risk to the health and safety of the public. This is based on our concurrence with the Browns Ferry Special Review Group's conclusions identified above as well as the significant improvements in fire protection already made at the Trojan facility since the Browns Ferry fire. These include estblishment of administrative controls over combustible materials and use of ignition sources; training and staffing of a fire brigade; issuance of technical specifications to provide limiting conditions for operation and surveillance requirements on fire protection systems; and the existing detection, automatic suppression systems and the manual fire suppression means for all areas including those for which an alternate capability is being provided.

At the time of the Fire Protection SER issuance (March 9, 1978) the Trojan operating license was amended to require the implementation of the alternate shutdown capability during the refueling outage prior to return to power for Cycle 3 operation. It was estimated at that time that return to power for Cycle 3 operation would be about June 1979. Due to unanticipated outages at Trojan, the actual refueling outage will be later than June 1979. This refueling outage is presently scheduled for Spring 1980. The modifications for alternate shutdown capability will be implemented at that time. For the reasons stated above, we continue to hold that this implementation schedule is acceptable.

Unresolved safety issues are identified in NUREG-0510, "Identification of Unresolved Safety Issues Relating to Nuclear Power Plants", Report to Congress, January 1979. Justification for continued plant operation is discussed in this report and in NUREG-0371.

Copies of all NUREG documents mentioned (except for enclosure 3 below) in this letter have previously been made availble to you.

Sincerely,

for Victor Stello, Jr., Director
Division of Operating Reactors

Enclosures:
1. W response to Question I.3
2. Memo to the Commissioners dated March 15, 1979
3. NUREG-0413

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cc: Mr. H. H. Phillips
Portland General Electric Company
121 S.W. Salmon Street
Portland, Oregon 97204

Warren Hastings, Esquire
Counsel for Portland General
Electric Company
121 S.W. Salmon Street
Portland, Oregon 97204

Mr. J. L. Frewing, Manager
Generation Licensing and Analysis
Portland General Electric Company
121 S.W. Salmon Street
Portland, Oregon 97204

Columbia County Courthouse
Law Library, Circuit Court Room
St. Helens, Oregon 97501

Director, Oregon Department of Energy
Labor and Industries Building, Room 111
Salem, Oregon 97310

Richard M. Sandvik, Esquire
Counsel for Oregon Energy Facility
Siting Counsel and Oregon Department
of Energy
500 Pacific Building
520 S.W. Yamhill
Portland, Oregon 97204

Michael Malmrose
U. S. Nuclear Regulatory Commission
Trojan Nuclear Plant
P. O. Box O
Rainier, Oregon 97048

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~~I.3~~ (Continued)

RESPONSE

Beginning in mid-1976, Westinghouse initiated a retesting program for the instrument transmitters utilized by the plants in the program (Table 1) at the request of the NRC, to demonstrate their capability to perform their required functions, either trip or long-term monitoring, under more severe environmental conditions than previously employed. The retesting of the original models of transmitters proved unsuccessful for long-term monitoring functions and as a consequence, between August and September 1977, Westinghouse issued detailed, transmitter replacement recommendations to those customers within the scope of this program (Table 1).

On more result license applications (D. C. Cook and North Anna) the staff has introduced the additional requirements that sequential testing be employed and that a minimum of one hour operability be demonstrated for transmitters employed for short-term reactor trip and/or safety injection automatic protective functions in a high energy line break environment. Prior to this change in NRC Requirements Westinghouse had considered that transmitters qualified for shorter periods than one hour were capable of performing short-term functions.

As a consequence of these additional staff requirements, the information contained in WCAP-7410L justifying the qualification of Barton, Foxboro and Fisher-Porter pressure and differential pressure transmitters for short-term safety related high energy line break applications is hereby withdrawn from consideration within the Supplemental Program. Westinghouse will issue additional transmitter replacement recommendations, as appropriate, to the plants within this program (Table 1) to reflect this change of position. Since the transmitter qualification test results contained in WCAP-7410L will not be relied upon for instrument qualification for the plants in this program, Question I.3, in total, requires no further response.

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March 15, 1979

SECY-79-112A

FOR: The Commissioners

THRU: Lee V. Gossick, Executive Director for Operations

FROM: Harold R. Denton, Director, Office of Nuclear Reactor Regulation

SUBJECT: UNION OF CONCERNED SCIENTISTS' PETITION FOR RECONSIDERATION - ENVIRONMENTAL QUALIFICATION OF ELECTRICAL EQUIPMENT

In a memorandum dated December 12, 1978, B. Snyder of the Office of Policy Evaluation requested that we respond to several questions regarding environmental qualification of electrical equipment in operating plants. Our response is enclosed.

H R Denton

H. R. Denton, Director
Office of Nuclear Reactor
Regulation

Enclosure:
Response to B. Snyder's
memorandum dated
December 12, 1978

cc: NRC PDR
Union of Concerned Scientists
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RESPONSE TO B. SNYDER'S

MEMORANDUM OF DECEMBER 12, 1978

Question 1

"To what degree has the staff relied on probability analysis when it states on page 36, Appendix B to the December 15, 1977 staff memo, that one of the reasons that no immediate action is required is that "the likelihood of a major accident requiring the performance of this equipment is very low."

Response

The staff has not conducted an analysis of the probability of a major accident as part of its consideration of the adequacy of environmental qualification of safety related equipment. However, in reaching judgments on this issue as to the type of action (i.e., immediate or otherwise) that should be taken to ensure no undue risk to the public, the level of protection provided in the facilities to prevent sudden pipe breaks ("major accident requiring the performance of this equipment") is considered. Experience in commercial power reactors alone is sufficient to demonstrate that the likelihood of such events is low. In addition, data developed from similar piping system designs in other industries is in agreement with this experience.

As initially outlined in the staff memoranda of December 15, 1977 and March 1978 and subsequently in Item 11 of Enclosure 1 of the July 6, 1978 response, the scope and timing of staff programs to provide additional confidence that adequate environmental qualification of equipment exists are based on several factors, including the likelihood of a major accident requiring the performance of this equipment.

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The degree to which this factor has shaped the staff's actions is difficult to quantify. In cases in which the licensing staff had insufficient confidence that equipment important to safety would function in a major accident, the plants were required to shut down and remedy the problem (e.g., D. C. Cook Unit 1 and Pilgrim Unit 1). These decisions were reached with little or no consideration as to the likelihood of such a "major accident." In other cases, the staff judgment was that the equipment could be demonstrated to be qualified and additional time was allowed for such demonstration in part because the likelihood of an accident environmentally challenging this equipment during the time required to confirm or further document its qualification was low.

In continuing to recommend that no immediate action need be taken, the staff does not rely solely on the low likelihood of a major accident, but rather is guided primarily by its judgment, as discussed in the response to question 6 below, that equipment required for safety will not fail before performing its safety function when exposed to design basis accident conditions.

Question 2

Item 17 (page 40) of Enclosure 1 of the staff's August 31, 1978 memo does not appear to address the UCS statements as requested. It appears from the staff's response that D.C. Cook was permitted to operate prior to complete demonstration for all environmental qualification. If this is so, on what basis was this decision made? Was such operation without complete environmental justification consistent with the Commission's regulations? Is the following statement in the April 13 Order on page 25, footnote 25 correct:

> "As a pre-condition for initial operation, the staff required the licensee to document adequate environmental qualification of numerous electrical components, including connectors and terminal blocks."

Have other plants been permitted to initiate operation prior to full environmental qualification? If so, please identify those plants and explain. Also, has full environmental qualification now been provided?

RESPONSE

A. Summary

Item 17 of Enclosure 1 of our August 31 submittal summarized the actions taken by us in licensing D. C. Cook Unit 2 to begin operation. We described the bases for our conclusion that safety related electrical equipment was adequately qualified at the time the decision was made to allow Unit 2 to operate at significant power levels. That decision was made by us on the basis of data supplied by the licensees to support their statement that safety related electrical equipment would perform its safety function if a design basis event were to occur. However, we also required at that time that the applicant perform in the near future certain confirmatory tests to provide

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additional information. Most of those tests have been completed and, for the most part, support our original judgement. Therefore we believe that the footnote at the bottom of page 25 of the April 13 Order is correct.

The UCS petition, together with the results of the Sandia tests referenced in that petition, identified a need to intensify our review of environmental qualification information for some equipment. Since that time, staff review in this area has been intensified. Only four plants -- North Anna Unit 1, Cook Unit 2, Hatch Unit 2 and Arkansas Nuclear One, Unit 2 -- have been licensed to initiate operation since the petition was filed. The staff found the environmental qualification program for both the Hatch and Arkansas plants to be acceptable. The North Anna and D. C. Cook plants were also acceptable with one exception. The exception concerned a need for confirmatory data on several kinds of pressure transmitters used in both plants. For both Cook Unit 2 and North Anna Unit 1, we permitted operation based on our judgement that the transmitters would perform their safety function if called upon, but we required that additional tests be performed to confirm that judgement. The response to question 6 addresses the status of equipment qualification for plants licensed prior to D. C. Cook and North Anna.

As noted above, almost all confirmatory tests in connection with the Cook Unit 2 OL have now been completed. There are still some unresolved questions regarding tests performed on Barton pressure transmitters which were one of several transmitters requiring confirmatory testing. We will require that additional tests be performed to resolve the questions remaining

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on the Barton transmitters. With that exception, all equipment qualification issues have been resolved for D. C. Cook Unit 2 and North Anna Unit 1.

More details on each of these points are presented below. This information is intended to supplement the response provided in our August 31, 1978 submittal.

B. Significance of Adequate Qualification

Some clarification of our position on the adequacy of equipment qualification may be helpful. The UCS statements refer to "complete" demonstration of equipment qualification. We believe that the word "complete" connotes an achievement of perfection that is neither possible in an engineering sense nor required by the Commission's regulations. The staff conclusions with regard to the adequacy of environmental qualification are a matter of engineering judgement based on a technical evaluation of the available experimental and analytical data. Further, the details of exactly what information is needed to reach such a finding are continuing to evolve as we acquire and analyze more data. Today we believe there is sufficient technical information available to make judgements that equipment is adequately qualified, but there is always the possibility that, as we expand our equipment qualification data base, new information may be developed that would require reassessment of some equipment.

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C. Licensing of D. C. Cook, Unit 2

On March 7, 1978, near the end of our review for licensing Cook Unit 2 initial operation, we met with the licensees. They described both test results and test procedures for transmitters, terminations, cables, and penetrations to demonstrate that this equipment was adequately qualified. We concluded that most of this information, once documented, would be sufficient to ensure that all electrical equipment would perform its safety function if a design basis event were to occur. However, the licensees justified the operability of Barton and Foxboro pressure transmitters on the basis of separate effects test. In separate effects tests, different pieces of equipment or components are exposed separately to the different environmental conditions (e.g., radiation, temperature, humidity) produced by design basis events. Engineering judgement and analysis are used to assess the effects of combining these conditions. While we found the results of these separate effects tests acceptable, we believed that full sequential tests should be performed to confirm our judgment regarding transmitter operability in the event of a design basis accident.

Following receipt of the required documentation and other review matters being satisfactorily completed, Amendment 2 of the D. C. Cook Unit 2 operating license, which authorized operation at or below 5% power, was issued on March 8, 1978. This amendment included four license conditions related to equipment qualification. The conditions are summarized as follows:

- Condition 3.B.1 required that the licensees provide within 90 days (June 8, 1978) results of full sequential qualification tests, in accordance with IEEE-323-1971, to demonstrate the qualification of Foxboro and Barton transmitters and, within 2 weeks, provide a basis for continued operation of the facility during the time required to complete these tests.
- Condition 3.B.2 required that the licensees provide, within two weeks, qualification test procedures and results for all electrical cable connections and terminations in safety related circuits within containment.
- Condition 3.B.3 required that the licensees provide within two weeks the documentation of test results and analysis that demonstrate the environmental qualification of safety related cables.
- Condition 3.B.4 required that the licensees provide within two weeks documentation demonstrating comparability of electrical penetrations installed at D.C. Cook Unit 2 to prototype penetrations which had undergone testing under steamline break environmental conditions.

We required that these conditions be resolved prior to operation above 20% of rated power. This requirement was selected somewhat arbitrarily, but was derived from the staff's general knowledge that accidents at this power level would result in much less severe environmental stress on safety equipment than accidents at higher power levels. Because the conditions called for the licensees to supply most of the documentation within two weeks, the power limit of 20%

was expected practically to apply only to the transmitters. As noted above, it was our belief that these transmitters would perform their safety function if required and it was expected that the confirmatory sequential tests would confirm that judgement. The licensee believed in March that these tests could be completed by the time the plant was expected to reach 20% power, although we also considered the fact that, up until that point, the fission product inventory would be inconsequential even if the transmitters were to fail and safety equipment did not function as designed.

Amendment 3, issued on March 16, 1978, allowed power operation above 5% and up to 20% of full rated power. None of the qualification related conditions added to the license as a consequence of Amendment 2 were altered by Amendment 3.

On March 20, 1978, the licensees met with the staff and provided information which satisfied Conditions 3.B.3 and 3.B.4 of that Amendment. They also provided information in accordance with Condition 3.B.1 to justify their request for permission to operate above 20% power before the tests on the Foxboro and Barton transmitters were completed. This information included additional results of testing performed on these transmitters. We concluded that these results provided additional assurance that these transmitters would perform their function in the unlikely event of a desgin basis accident. This additional information was placed on the Cook docket at that time.

At the March 20th meeting, the licensee also provided test data for safety-related connections and terminations used in D. C. Cook Unit 2 (Condition 3.B.2). Nearly all of these connections and terminations successfully passed tests which, for the most part, simulated main steam line break accident conditions. Power cable terminal blocks failed the tests and were replaced with qualified splices. Based on this information, we concluded that the safety-related terminations and connections would perform as designed in the event of a design basis accident and thus that Condition 3.B.2 was satisfied. However, we concluded that full sequential confirmatory tests on these connections and terminations should be performed to confirm that conclusion.

Amendment 4 to the license, which authorized operation at 50% of rated power, was issued on March 24, 1978. It called for the full sequential confirmatory tests, for both the Barton and Foxboro transmitters and for the connections and terminations, to be completed by June 6, 1978. (The 50% power limit was imposed for reasons not related to equipment qualification). The decision to allow operation at power levels above 20% prior to receiving results of full sequential tests on the transmitters (instead of limiting power to 20% as stated in Amendment 2) was based on the additional information we received at the March 20 meeting and our judgement, based on that information, that the equipment would perform its safety function. Also the licensees indicated that the confirmatory, sequential tests were to begin shortly and that we would be made aware immediately of any results which indicated that the transmitters were inadequately qualified to perform their safety functions.

Subsequent to the issuance of Amendment 4, the licensees met with the staff in April and May and provided additional information on both the transmitters and safety related terminations inside containment.

Foxboro Transmitters

Additional documentation of separate effects tests was provided for staff review. The licensees asked the staff to reconsider the need for sequential tests. However, the staff's requirement for full sequential tests was based on the limitations of separate effects testing. Therefore, this documentation was not sufficient to change our judgement that additional confirmatory tests were necessary. Rather than pursue these confirmatory tests, the licensees committed to replacing these transmitters with other qualified instruments prior to startup following the first regularly scheduled refueling. We found this proposal acceptable based on our judgement that the Foxboro instruments were qualified to perform their function if required. (As noted in our August 31, 1978 submittal to the Commission, one model of the Foxboro transmitters survived the separate effects test while the other model failed after 4 minutes. It was our judgement that 4 minutes was adequate to ensure that the transmitter would perform its safety function. However, it could not be assured that it would perform its post accident monitoring function. The applicant provided an evaluation which demonstrated that, by modifying plant operating procedures, proper operator action during the post accident period could be assured. These modified procedures were to be utilized only until the instruments were replaced.)

Barton Transmitters

We reviewed the program proposed by the licensees to sequentially test the Barton Transmitters and concluded that it was consistant with the conditions specified in Amendment 4. However, we were told that the test program could not be completed until July 1978, with a formal report to be submitted by October 1978. It was our judgement that continued operation of Unit 2 until October 1978 was acceptable because we viewed these tests as being confirmatory in nature and because we would be informed promptly of any deficiencies revealed by the test in July.

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Safety-Related Terminations Inside Containment

The licensees provided documentation which satisfied the condition 3.B.2 specified in Amendment 4.

Safety-Related Cables

The applicant also informed the staff that one type of cable previously found acceptable in accordance with condition 3.b.3 of Amendment 3 had failed further tests being performed by the applicant. These additional tests were conservative in that the test conditions were selected to envelop both the high radiation conditions associated with a postulated LOCA and the high temperatures associated with a steam line break accident. Rather than retest to more realistic conditions, the applicant elected to replace this cable with cable that had survived qualification tests.

Amendment 6, issued on June 16, 1978, reflected our decisions regarding the Foxboro and Barton transmitters and called for completion of sequential testing of the Barton transmitters by October 1978. As noted in Section E below, those tests have been completed and the results have been reviewed by the staff.

In summary, there was an intensive review of environmental qualifications by the staff in the process of granting D.C. Cook Unit 2 permission to operate. I was our judgement prior to granting permission to allow operation at reduced power that equipment was adquately qualified. To confirm that judgement, we required that the licensee provide additional documentation and, in some instances, the results of additional tests. That process took about seven months to complete but, for the most part, it confirmed that our initial judgements were correct. As noted in our August 31 submittal, in two instances (for certain terminal blocks and one type of cable) the tests did not show that equipment was adequately qualified and the applicant replaced it.

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D. Licensing of North Anna, Unit 1

The staff licensed the operation of North Anna Unit 1 at about the same time that D. C. Cook Unit 2 was undergoing licensing review. On April 1, 1978 North Anna was licensed to begin power operation conditioned, in part, on the future performance of full sequential confirmatory tests on Barton and Foxboro transmitters installed in the plant. These transmitters were of the same design as those used in D. C. Cook. These tests were to be completed within 90 days. Subsequent to that date the licensee decided that, rather than test both the Foxboro and Barton transmitters, the Foxboro transmitters would be replaced with Bartons.

Amendment 7 to the North Anna Unit 1 license, dated July 3, 1978, reflected that change. It also allowed the licensee an additional 3 months (from July 1 to October 1, 1978) to complete the confirmatory sequential tests on the Barton instruments. These were the same tests which were to be used to confirm the adequacy of the qualification of the D. C. Cook instruments. This delay was found acceptable for North Anna for the same reason as for D. C. Cook.

E. Current Status of the D. C. Cook/North Anna Environmental Qualification Review

We noted in our August 31, 1978, memorandum to the Commission that problems had developed during the sequential tests performed on the Barton transmitters and that we had requested additional information. These sequential tests were performed by Westinghouse and, on the 29th of September, they provided the staff with a report describing all tests performed and the results obtained. This report showed that the instruments performed properly during the time they would be required to perform their safety function of initiating protective actions. However, at a later stage in the test, which corresponded to the post accident monitoring phase, the transmitters exhibited larger than expected errors that fluctuated with time. At the time the test was completed, all transmitters were again functioning properly.

The environmental conditions for these sequential tests were established by selecting a conservative radiation/steam/temperature/pressure profile that bounded those conditions expected for both the steamline break (SLB) and loss of coolant accidents (LOCA). When the accuracy problems were detected, additional tests were performed to help identify the cause. From their analysis of these tests, Westinghouse deduced that the problem was due to the combined effects of high radiation and high temperature produced by attempting to test for

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both LOCA and SLB conditions simultaneously. This approach was employed for testing efficiency - it was not required for meeting NRC licensing requirements. In addition, Westinghouse believed that these additional tests demonstrated that the transmitters would perform properly when subjected to the less conservative conditions (but acceptable under NRC requirements) for LOCA and SLB accidents considered separately.

We have concluded that the tests demonstrate that the instruments will perform their safety function (i.e. initiate protective action) in the early stages of either a LOCA or SLB. We also agree with Westinghouse that the errors in instrument readings which occurred during the post accident monitoring phase of the sequential tests may be due to the unusual way the sequential test was performed. (combining LOCA and SLB). However, we plan to request that full sequential confirmatory tests be completed to confirm our judgement.

Question 3

"Please furnish an updated response to Item 2 of Enclosure 1 of the staff's August 31, letter addressing the further results of the staff's ongoing review. Can the staff now fully respond to the question?"

Response

The question responded to in Item 2 of Enclosure 1 of the staff's August 31, 1978 memorandum for the Commissioners was as follows:

> "Are there any plants which cannot demonstrate environmental qualification for electrical connectors, splices, penetrations, and terminal blocks with full documentation such that they meet the minimum requirements of the Commission's regulations? If there are, please explain the legal and regulatory basis for permitting their operation."

In its response to this question, the staff was able to verify that adequate documentation existed for connectors, penetrations, and unprotected terminal blocks. With regard to protected terminal blocks, splices, and other electrical equipment, the staff is still unable to fully respond.

In the August 31, 1978 response the staff stated that it would continue to pursue the question of environmental qualification and documentation in two principal ways: (1) the Systematic Evaluation Program (SEP); and (2) IE inspections of licensee actions in response to IE Circular 78-08. The preliminary results of the SEP were previously reported in NUREG-0458 and continuing SEP efforts are discussed in the response to Question 6 below. IE inspections related to Circular 78-08 are continuing. The results of this effort since the August 31, 1978 response and subsequent followup actions are summarized in the following paragraphs.

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As a result of IE inspections conducted to date, items of safety related equipment located inside containment have been identified where documentation to demonstrate environmental qualification was not available at the time of the inspection. Licensee efforts to produce documentation for this equipment are continuing.

In addition to the apparent documentation problems identified, at least one plant has been identified which had unqualified splices in safety related circuits (see Trojan LER No. 78-027). These splices have been replaced and the licensee has expanded its inspection program to check for similar problems in other electrical circuits.

The only other components that have been identified as being unqualified are certain specific models of valve position indication limit switches. These limit switches are different from those identified in an earlier IE Bulletin (IEB 78-04). A failure of these switches would not directly result in a loss of valve function, only the position indication would be affected. The switches are being replaced with qualified components at each of the plants where their incorrect application has been identified.

The above findings are preliminary since many of the licensees' reviews of qualification documentation initiated by the Circular are only partially completed. However, based on these preliminary findings, the staff concluded that IE Circular 78-08 is not receiving the level of attention from all licensees that the staff believes is warranted. To expedite completion of the licensee's re-review programs the staff has issued IE Bulletin 79-01 (see attachment). This bulletin requires that the licensees' complete their re-reviews within 120 days and report to the staff, in writing, the documented basis for qualification of electrical equipment required to function under accident conditions.

NRC inspections of licensees' programs for review of component qualification documentation will continue in conjunction with IE Circular 78-08 and IE Bulletin 79-01. The basis for continued reactor operation while the staff and licensees continue to pursue the question of environmental qualification and documentation remains as stated in the August 31, 1978 response.

Attachment To
Response No. 3

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF INSPECTION AND ENFORCEMENT
WASHINGTON, D.C. 20555

February 8, 1979

IE Bulletin No. 79-01

ENVIRONMENTAL QUALIFICATION OF CLASS IE EQUIPMENT

Description of Circumstances:

The intent of IE Circular 78-08 was to highlight to all licensees important lessons learned from environmental qualification deficiencies reported by individual licensees. In this regard, licensees were requested to examine installed safety-related electrical equipment and determine that proper documentation existed which provided assurance that this equipment would function under postulated accident conditions. The scope of IE Circular 78-08 was much broader than other previously issued Bulletins and Circulars (such as IEB 78-04 and IEB 78-02) which addressed specific component failures. The intent of this Bulletin is to raise the threshold of IE Circular 78-08 to the level of a Bulletin; i.e., action requiring a licensee response.

Inspections conducted to date by the NRC of licensees' activities in response to IE Circular 78-08 have identified one component which licensees have found to be unqualified for service within the LOCA environment. Specificially, unqualified stem mounted limit switches (SMLS), other than those identified in previously issued IE Bulletin 78-04, were found to be installed on safety-related valves inside containment at both Duane Arnold and Quad Cities 1 and 2 Nuclear Generating Stations. The unqualified switches are identified as NAMCO Models SL2-C-11, S3CML, SA1-31, SA1-32, D1200j, EA-700 and EA-770 switches. According to the manufacturer, these switches are designed only for general purpose applications and are not considered suitable devices for service in the LOCA environment. Consequently, switches are being replaced at the above power plants with qualified components.

Also, NRC inspection of component qualification has identified equipment which does not have documentation indicating it is qualified for the LOCA environment. The inspections have also identified that the licensees' re-review and resolution of problem areas are not receiving the level of attention from all licensees which the NRC believes is warranted. Because of the protracted schedule for completion of the re-review, we are now requesting the power reactor facilities with operating licenses to expedite completion of their re-review program originally requested by IE Circular 78-08 dated May 31, 1978.

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Action to Be Taken By Licensees of All Power Reactor Facilities (Except Those 11 SEP Plants Listed on Enclosure 3) With An Operating License:

1. Complete the re-review program described in IE Circular 78-08 within 120 days of receipt of this Bulletin.

2. Determine if the types of stem mounted limit switches described above are being used or planned for use on safety-related valves which are located inside containment at your facility. If so, provide a written report to the NRC within the time frame specified and to the address specified in Item 4 below.

3. Provide written evidence of the qualification of electrical equipment required to function under accident conditions.* For those items not having complete qualification data available for review, identify your plans for determining qualification, either by testing or engineering analysis, or combination of these, or by replacement with qualified equipment. Include your schedule for completing these actions and your justification for continued operation.

   Submit this information to the Director, Division of Reactor Operations Inspection, Office of Inspection and Enforcement, Nuclear Regulatory Commission, Washington, D.C. 20555 with a copy to the appropriate NRC Regional Office within 120 days of receipt of this Bulletin.

4. Report any items which are identified as not meeting qualification requirements for service intended to the Director, Division of Operating Reactors, Office of Nuclear Reactor Regulation, Nuclear Regulatory Commission, Washington, D.C. 20555 with a copy to the appropriate NRC Regional Office within 24 hours of identification. If plant operation is to continue following identification, provide justification for such operation. Provide a detailed written report within 14 days of identification to NRR, with a copy to the appropriate NRC Regional Office.

No additional written response to this IE Bulletin is required other than those responses described above. NRC inspectors will continue to monitor the licensees' progress in completing the requested action described above. If additional information is required, contact the Director of the appropriate NRC Regional Office.

* This written evidence should include: 1) component description; 2) description of the accident environment; 3) the environment to which the component or equipment is qualified; 4) the manner of qualification which should include test methods such as sequential, synergistic, etc., and 5) identification of the specific supporting qualification documentation.

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LISTING OF IE BULLETINS
ISSUED IN LAST TWELVE MONTHS

Bulletin No.	Subject	Date Issued	Issued To
78-03	Potential Explosive Gas Mixture Accumulations Associated with BWR Offgas System Operations	2/8/78	All BWR Power Reactor Facilities with an OL or CP
78-04	Environmental Qualification of Certain Stem Mounted Limit Switches Inside Reactor Containment	2/21/78	All Power Reactor Facilities with an OL or CP
78-05	Malfunctioning of Circuit Breaker Auxiliary Contact Mechanism-General Model CR105X	4/14/78	All Power Reactor Facilities with an OL or CP
78-06	Defective Cutler-Hammer, Type M Relays With DC Coils	5/31/78	All Power Reactor Facilities with an OL or CP
78-07	Protection afforded by Air-Line Respirators and Supplied-Air Hoods	6/12/78	All Power Reactor Facilities with an OL, all class E and F Research Reactors with an OL, all Fuel Cycle Facilities with an OL, and all Priority 1 Material Licensees
78-08	Radiation Levels from Fuel Element Transfer Tubes	6/12/78	All Power and Research Reactor Facilities with a Fuel Element transfer tube and an OL.

LISTING OF IE BULLETINS
ISSUED IN LAST TWELVE MONTHS

Bulletin No.	Subject	Date Issued	Issued To
78-09	BWR Drywell Leakage Paths Associated with Inadequate Drywell Closures	6/14/78	All BWR Power Reactor Facilities with an OL or CP
78-10	Bergen-Paterson Hydraulic Shock Suppressor Accumulator Spring Coils	6/27/78	All BWR Power Reactor Facilities with an OL or CP
78-11	Examination of Mark I Containment Torus Welds	7/21/78	BWR Power Reactor Facilities for action: Peach Bottom 2 and 3, Quad Cities 1 and 2, Hatch 1, Monticello and Vermont Yankee
78-12	Atypical Weld Material in Reactor Pressure Vessel Welds	9/29/78	All Power Reactor Facilities with an OL or CP
78-12A	Atypical Weld Material in Reactor Pressure Vessel Welds	11/24/78	All Power Reactor Facilities with an OL or CP
78-13	Failures In Source Heads of Kay-Ray, Inc., Gauges Models 7050, 7050B, 7051, 7051B, 7060, 7060B, 7061 and 7061B	10/27/78	All general and specific licensees with the subject Kay-Ray, Inc. gauges
78-14	Deterioration of Buna-N Components In ASCO Solenoids	12/19/78	All GE BWR facilities with an OL or CP

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Enclosure No. 3

SEP Plants

Plant	Region
Dresden 1	III
Yankee Rowe	I
Big Rock Point	III
San Onofre 1	V
Haddam Neck	I
LaCrosse	III
Oyster Creek	I
R. E. Ginna	I
Dresden 2	III
Millstone	I
Palisades	III

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Question 4

"Page 12 of the Commission's April 13, 1978 Memorandum and Order states:

> "Fundamental to NRC regulation of nuclear power reactors is the principle that safety systems must perform their intended functions in spite of the environment which may result from postulated accidents. (The controlling regulation here is 10 CFR 50, Appendix A, General Design Criterion 4.) For example, if an electrical component is required to function in a safety system which was designed to mitigate the consequences of certain accidents, that component must perform its intended function for postulated accidents such as: (a) loss-of-coolant accident (LOCA), (b) main steam line break (MSLB), or (c) failure of any other high-energy confining system."

Is there any inconsistency between that statement and the staff's conclusion that no regulation was violated by the licensees which installed unqualified equipment because the licensees were under no such requirement? Please elaborate."

Question 5

"Are all licensees now under a duty or commitment to have full environmental qualification for their electrical equipment so that any failure to have such qualification could result in an enforcement action?"

Response (Questions 4 and 5)

Questions 4 and 5 are related in that they both deal with the question of a licensee's obligation under past and present NRC regulations to install qualified equipment in plants licensed by the NRC. Before the question of whether a licensee is, or has been, in violation of a Commission regulation can be answered (Question 4), the extent of a licensee's obligation to comply with the regulation must be defined (Question 5). Therefore, a response to Question 5 will be provided first, followed by a response to Question 4.

With regard to Question 5, all licensees are now and have in the past been required to assure themselves that equipment important to safety is environmentally qualified for its service environment. The regulatory basis for this requirement are General Design Criteria 1 and 4 which articulate long standing regulatory practices of the NRC and the AEC before it. Simply stated, Criterion 4 requires that structures, systems, and components important to safety be designed to function in both normal and accident environments. Criterion 1 requires that a quality assurance program be established to assure that these structures, systems, and components are designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. To the extent that licensees are not in compliance with these general requirements, they are in violation of the Commission's regulations and are subject to corrective enforcement action.

These requirements have been fundamental to the development and regulation of the nuclear industry and are generally understood by individuals familiar with the industry. The issue in question is what constitutes "full" or adequate environmental qualification and the extent to which the qualification must be documented.

This question has been addressed by the staff in several previous submittals to the Commission.[1/] In summary, industry practices and commitments which have been found acceptable by the NRC staff for satisfying qualification

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[1/] 1) Enclosure 5 of the staff's memorandum for the Commissioner's dated March 23, 1978, subject: Union of Concerned Scientists Petition.

2) "Report on the Historical Evolution of Environmental Qualification Requirements for Safety-Related Electrical Equipment", dated December 15, 1977. This report was part of the overall "NRC staff Report on Union of Concerned Scientist's Petition for Emergency and Remedial Action", also dated December 15, 1977.

requirements in the past have ranged from a simple specification of the highest industrial quality components available at the time for the oldest plants, to a licensing commitment to a comprehensive program of environmental qualification for newer plants in the CP stage of licensing which is in accordance with industry standards such as IEEE Std. 323-1974[2/] and documented in accordance with 10 CFR 50, Appendix B requirements for quality assurance programs.

General Design Criteria 1 and 4 contain no detailed systematic requirements for either quality assurance methods or for systematically maintaining quality records. The specific qualification and documentation requirements applicable to individual plants are set forth in the license applications approved by the staff at the time of initial licensing and in all current conditions of the licenses, including technical specifications. For older plants the information contained in license applications and technical specifications tends to be general in nature, consistent with the more generalized provisions of Criteria 1 and 4. Consequently, it is often very difficult to identify a "violation" of these generalized specifications. However, enforcement action in the form of Orders can be taken even when no specific regulatory requirement exists wherever potentially hazardous conditions are identified. As discussed in response to Question 6 below, the staff is not aware of any such conditions.

[2/] Institute of Electrical and Electronics Engineers Standard 323-1974, "IEEE Standard for Qualifying Class IE Equipment for Nuclear Power Generating Stations."

With regard to Question 4, in view of the above discussion, the staff believes that there is no inconsistency between the statement on page 12 of the Commission's April 13, 1978, Memorandum and Order and the staff's conclusion that no regulation was violated by the licensees who installed equipment that was later determined to be of questionable qualification. This conclusion was based on the staff's finding that the equipment was installed in compliance with the general requirements of General Design Criteria 1 and 4 and no identified violation of Appendix B. The statement that the "the licensees were under no such requirement" was intended to mean that some of the licensees were under no specific regulatory requirements to use a specific method of qualification (e.g., testing) or to maintain detailed qualification documentation.

As stated above, all plants are required to be in compliance with General Design Criteria 1 and 4. If for any reason a licensee determines that a plant is not in compliance with these regulations, the licensee must take appropriate action to bring the plant into compliance and to seek any required regulatory approvals if any changes are evolved. The licensee can alternatively seek an exemption from such requirements.

Even though many of the licensees are not under a regulatory requirement to maintain specific qualification documentation, the staff has initiated a program to determine the extent to which the qualification of safety-related equipment in all plants can be documented. The preliminary results of this program are discussed in the response to Question 5.

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At this time the emphasis in the staff effort is being placed on determining the status of qualification and on identifying possible deficiencies. Formal enforcement action will be used if necessary to assure that any needed corrective actions are taken in a timely manner.

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Question 6

"The July 6, 1978 response to Item 11, Enclosure 1 stated that the staff continues to believe that adequate protection for the public health and safety exists despite the six plant shutdown. Does the staff believe that some plants in operation use equipment which will fail when exposed to design basis event conditions? What is the basis for the staff's judgment?"

Response

The staff does not believe that equipment required for safety installed in operating reactors will fail before performing its safety function when exposed to design basis event (DBE) conditions. In making this statement the staff does not mean to imply that test results are available which demonstrate that all safety related equipment would remain functional for the full time duration of all tests that have been devised to envelope DBE conditions. To the contrary, as the Commission is aware, there have been tests performed in which safety related equipment did not remain functional for the full duration of the test. In each of these cases the staff has reviewed the particular circumstances involved and appropriate action was taken.[1/] In some cases equipment was replaced; in others it was not. In those cases where the equipment was not replaced, the staff concluded that the equipment would have performed its safety function prior to failure in actual DBE conditions. The staff conclusion was based on factors such as the length of time the equipment remained functional during the test, the severity of the test conditions as compared to expected DBE conditions, and the actual installed location of the equipment.

The staff acknowledges that if the equipment discussed above had remained

1/ See the staff's memorandum for the Commissioners dated March 23, 1978, Subject: Union of Concerned Scientist's Petition, for examples of staff reviews and actions in connection with recent test results.

functional during the entire test , there would be greater assurance that the equipment would perform its safety function in actual DBE conditions. This statement speaks to the fundamental issue in question, i.e., the adequacy of the level of assurance that exists. The staff believes that there is adequate assurance for the present that safety related equipment is qualified to perform its safety function. This belief is not based on a rigorous component-by-component review and evaluation by the staff of specific test results for all safety-related equipment. Test results do not exist for many components, nor were they required at the time of initial licensing of the currently operating reactors. This belief is a matter of technical judgment based on the staff's consideration of all the available test results, including test failures as discussed above, the previous staff reviews conducted at the time the plants were initially licensed to operate,[2/] subsequent operating experience, and reviews connected with staff backfitted requirements (e.g., 10 CFR 50.46).

Although the staff believes that the level of assurance that equipment is qualified is adequate for the present (i.e., in the short term), it also believes, as stated in its July 6, 1978, response to Item 11, that it is desirable to increase its level of assurance for the future (i.e., in the long term). The objective of the ongoing staff actions discussed in the response is to increase the existing level of assurance. The following paragraphs summarize and update earlier staff discussions of these activities.

[2/] Appendix A to NUREG-0413 provides a detailed discussion of the NRC environmental qualification and documentation requirements that formed the basis for the initial licensing of all currently operating reactors.

The Systematic Evaluation Program (SEP) is an important part of the ongoing staff actions. The results of the initial phase of the SEP for equipment qualification are reported in NUREG-0458. As a result of this activity the staff instituted a program of augmented IE inspections and issued IE Circular 78-08 to feed back lessons learned and initiate action by the licensees to examine the installation and qualification documentation that exists for equipment located inside containment. (See response to question 3 for a summary of the results of IE Circular 78-08 to date and proposed followup actions.)

Subsequent to the issuance of NUREG-0458, the SEP evaluations have continued and additional information is becoming available. The staff has requested each of the eleven SEP licensees to identify the specific conditions and method by which each equipment was qualified. The licensees are providing the information using a tabular format recommended by the staff that facilitates a cross comparison of the qualification information for the same equipment used in different facilities.

The major staff effort under the SEP program is being directed to more precisely define adequate methods of environmental qualification. Type testing of equipment used in the SEP facilities was not performed to the extent that it is employed in more recent facilities. However, an evaluation of the environmental qualification, frequently based on materials or partial testing of the equipment, has been performed by the licensees in many cases. The staff is currently reviewing these evaluations and plans

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to perform an onsite audit of the appropriate documentation for the SEP facilities to determine the adequacy and the margin of the licensees' qualification methods.

With respect to the environmental conditions associated with a postulated main steam line break accident, the staff is continuing its effort to improve the analytical models used to establish these conditions (Generic Technical Activities Program Task A-21). Particular attention is being given to facilities without automatic containment spray initiation because the operation of containment sprays decreases the duration of the peak containment temperature transient.

In addition ongoing staff reviews of operating license applications, which are being conducted in considerably more detail that in the past, are, in general, confirming the adequacy of the qualification of equipment installed in these reactors. The few instances of questionable qualification have resulted in the requalification or replacement of equipment. The fact that these more detailed reviews have generally confirmed the adequacy of qualification of equipment also contributes ot our confidence that equipment installed in operating reactors can perform its safety function in an accident environment.

The staff actions already underway discussed above will result in the development of additional and more detailed information on the environmental

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qualifications of existing safety-related electrical equipment in operating plants, and will provide the basis for a staff judgment of the longer-term actions that are needed to increase the level of assurance of the adequacy of environmental qualifications of such equipment.