July 17, 1985

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Docket No. 50-320
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Docket No. 50-320

Mr. F. R. Standerfer
Vice President/Director
Three Mile Island Unit 2
GPU Nuclear Corporation
P.O. Box 480
Middletown, PA 17057

Dear Mr. Standerfer:

Subject: Applicable Generic Letters - January 1, 1985 through June 30, 1985

The Three Mile Island Program Office has reviewed all generic letters issued from January 1, 1985 through June 30, 1985. As a result of our review, we are forwarding to you five (5) generic letters that have been determined to be applicable to your facility. For those generic letters that require some action by a certain date, you should lengthen the "compliance by" date indicated on the letter by the number of days that has elapsed between the date of issuance of the generic letter and the date of this letter.

If there are any questions with regard to applicability or compliance, please contact Michael T. Masnik of my staff.

Sincerely,

/s/ R. A. Weller for

Bernard J. Snyder, Program Director
Three Mile Island Program Office
Office of Nuclear Reactor Regulation

Enclosures:
Generic Letters
(85-1, 85-4, 85-5,
85-8, and 85-11)

cc: T. F. Demmitt
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S. Levin
W. H. Linton
J. J. Byrne
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

JAN 9 1985

TO ALL POWER REACTOR LICENSEES AND ALL APPLICANTS FOR POWER REACTOR LICENSES

Gentlemen:

SUBJECT: FIRE PROTECTION POLICY STEERING COMMITTEE REPORT
(Generic Letter 85-01)

Enclosed is a copy of the NRC Fire Protection Policy Steering Committee Report, dated October 26, 1984. The background and purpose of the Steering Committee is described in the report. A notice will be published in the Federal Register in the near future that will provide an opportunity for public comments on this report. No response to this letter is required.

Darrell G. Eisenhut, Director
Division of Licensing

Enclosure:
As stated

8501100621 150pp.

ENCLOSURE

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

OCT 26 1984

MEMORANDUM FOR: William J. Dircks
Executive Director for Operations

FROM: Fire Protection Policy Steering Committee

SUBJECT: RECOMMENDED FIRE PROTECTION POLICY AND PROGRAM ACTIONS

Introduction

At an August 27, 1984 meeting on fire protection, you directed that a review of current fire protection issues be conducted and that recommendations for resolution of these issues be made within two months. This effort was to examine all current licensing, inspection, and technical issues to develop policy recommendations aimed at expediting Appendix R compliance for older plants and assuring consistent levels of fire protection safety at all plants.

By your memorandum dated September 13, 1984 (Enclosure 1) to the NRR and IE Office Directors and the Regional Administrators, you confirmed this direction and approved a Steering Committee to provide appropriate recommendations. You indicated that, among the issues to be considered, were:

° adequacy of current guidance to industry;

° interpretation of Appendix R requirements;

° treatment of technical and schedular exemptions;

° comparison of Appendix R and current NTOL plants for fire protection safety;

° adequacy of current inspection practices; and

° identification and resolution plan for any outstanding technical issues.

In response to this direction, the Fire Protection Policy Steering Committee (SC) has considered the broad range of fire protection issues necessary to arrive at policy recommendations. The SC has reviewed documents which provide the basis for current fire protection policies and which discuss many of the issues that could significantly delay Appendix R compliance and question consistency in fire protection safety at all plants. The SC held six meetings. These included meetings with the Senior NRC Managers, the NRR and IE Office Directors, and the fire protection engineers from NRR, IE, and the Regions. At the latter meeting, the candid views of the individuals intimately involved in the fire protection issues were solicited and received. A record of the SC meetings is included as

Enclosure 7. Finally, the SC received significant input from Thomas Wambach, who acted as Secretary to the SC, and the Working Group, headed by Faust Rosa, NRR, and composed of V. Benaroya, C. Grimes, and V. Moore of NRR; S. Richardson, IE; W. Shields, OELD; and C. Anderson, T. Conlon, and W. Little of Regions I, II and III respectively.

The recommendations of the SC are provided below. We believe that this responds to your direction, and when fully implemented, represent actions that will indeed expedite Appendix R compliance and assure consistent levels of fire protection safety at all plants. The SC is aware that not all parties will be fully satisfied with these actions. Nor have our recommendations been reviewed by the cognizant Offices or Regions. However, we believe that they represent sound judgments balanced with other competing safety priorities, and that with your approval the plan can and should be initiated promptly.

Recommendations

To expedite Appendix R compliance and assure consistent levels of fire protection safety at all plants, the Fire Protection Steering Committee recommends that the following actions be taken:

1. Promptly issue the enclosed Generic Letter (Enclosure 2) informing all licensees that:

   (a) Extensions to the 50.48(c) schedules will no longer be granted;

   (b) An expedited fire protection inspection program will be instituted;

   (c) Documentation of valid analyses supporting fire protection features must be available for inspection;

   (d) Quality assurance applicable to fire protection systems is that required by GDC-1 of Appendix A to 10 CFR Part 50; and

   (e) The interpretations of Appendix R, (Enclosure 3) which should facilitate industry implementation of Appendix R and the responses to industry questions (Enclosure 6) represent the official agency position on all issues covered. (It should be noted that The Commission requested these documents for their review prior to issuance to industry.)

2. Conduct fire protection inspections within CY 1985 at ORs and NTOLs to include at least one site per licensee not subject to a previous Appendix R fire protection inspection. These inspections will assess the degree of fire safety, steer and promote licensee compliance, and take enforcement action where appropriate. A Temporary Instruction for this program will be issued by 11/15/84. To make this program of inspection most effective:

   (a) A workshop for the inspection teams will be conducted in mid December with SC, NRR, IE and Regional participation to assure common understanding of the objectives, scope and technical issues. Followup workshops will be held as needed;

   (b) The fire protection inspections will utilize new guidance for enforcement actions (Enclosure 4);

   (c) The processing of current fire protection enforcement actions will be expedited; and

   (d) A referee will be established to promptly resolve significant differences between the inspection teams and licensees.

3. Upgrade regulatory documents and procedures to achieve an appropriate level of fire protection safety while maintaining consistency among plants. In particular:

   (a) Impose a standard fire protection condition (Enclosure 5) in each operating license (already being implemented);

   (b) Reevaluate all fire protection guidance for consistency with the SC recommendations and compare fire protection requirements for ORs and NTOLs, both under the auspices of the Working Group;

   (c) Develop appropriate revisions to the Standard Review Plan and Standard Technical Specifications by March 31, 1985; and

   (d) Designate the Director, Division of Engineering, NRR as the central point of contact for interoffice/region fire protection issues.

4. To assure timely and on-track completion of these recommended actions, the SC will review progress at least quarterly, make mid-course corrections if appropriate, and report to the EDO.

Discussion

The recommended actions are grouped into three main areas dealing with (1) guidance to industry, (2) an expedited program of fire protection inspections, and (3) a general upgrading of regulatory documents to reach and maintain consistent fire protection safety. This discussion section will focus broadly on what the SC found during its deliberations to warrant the focus of these recommendations and will indicate how this satisfies the agenda of issues cited in your memo of September 13. Details on these issues are provided in the record of the SC deliberations contained in Enclosure 7.

With regard to guidance to industry, the SC concludes that adequate technical guidance had been issued but that there were areas where confusion could arise. It was not clear where exemptions were needed, for example. However, a diligent reading of Appendix R and other staff documents did provide the basis for the satisfactory implementation of Appendix R at Calvert Cliffs. The SC concluded that it was neither needed nor appropriate to develop new guidance, rather, bringing current technical and implementation guidance together in one Generic Letter and make the SRP, Tech Specs, and licenses consistent would suffice. The Generic Letter makes clear (1) that extensions to the 50.48(c) schedules will no longer be granted, (2) that an expedited inspection program will be instituted to see what fire protection fixes are in place and give licensees the inspection team judgements on the acceptability of future modifications, and (3) that the licensee judgements must be backed by documented and valid analyses. The SC believes that this will demonstrate to the licensee what action he must take and what our inspections will look for. The Generic Letter notes that, although the 50.48(c) schedules will not be extended, the relative safety priroities of fire protection modifications need to be considered in the development of "living schedules." One item of guidance in the Generic Letter that had not been uniformly disseminated is that the QA applicable to fire protection features is that required by GDC-1. This would not attempt to backfit any QA requirements. Rather it would assure that future design, procurement, installation, testing and maintenance of fire protection features would receive high industrial quality attention. The SC believes that this initiative fully responds to the first three issues in your memo of September 13.

Turning now to the inspection program, the SC found that the current inspections are generally satisfactory but that steps must be taken to indicate NRC's view of the importance of expediting implementation of Appendix R. These steps are to (1) speed up the inspection process, (2) develop a sound policy for fire protection enforcement actions, and (3) issue enforcement actions currently pending. These steps, in our view, would help expedite licensee compliance because it would raise industry's

awareness to NRC's resolve in this area and, more importantly, would allow the teams to judge the current direction (for licensees still designing or installing fixes) and advise the licensee on its acceptability. This should save both industry and staff resources in the long run. In the short run, that is 1985, the SC believes that adequate resources exist for the inspection teams (one for R-I, R-II and R-III, and one for R-IV/V) to be taken from regional staff, augmented by contractor, NRR, and IE assistance. While this would take a modest amount of reprogramming in the regional inspection program, we suggest that it's worth the effort to get Appendix R implementation behind us.

Prior to the 1985 inspection, a several-day inspection team workshop would be held to discuss the inspection program, the technical issues, and reach a common understanding on acceptability of various configurations and required documentation. Since this workshop cannot solve all potential problems the inspection teams will encounter, a team at HQ would be set up to promptly resolve significant differences between the inspection personnel and the licensee. This referee team would be headed by NRR (SES level) and would have an NRR, an IE and a regional technical member. Their decision would be issued in one week and would be sent to all teams for their information.

The SC believes that this program of expedited inspections, aimed at reaching ORs and NTOLs and to include at least one site from each licensee not previously subject to a fire protection inspection, coupled with denial of future schedular exemptions and a fire enforcement policy will result in a fair and uniform speed up of Appendix R compliance. Further, since the resource cost is believed to be modestly above the already-programmed fire protection inspections, we believe the cost is well worth it and will even benefit industry by correcting false starts in Appendix R implementation where they are found. Although we found the current inspections adequate (fifth item in your memo of September 13), this program will continue to be focussed on safe shutdown, will be implemented more expeditiously, and will build on the resolution of other initiatives considered by the SC. A Temporary Instruction for this inspection program has been drafted and is undergoing final revisions. It will be in final form by November 15, 1984 and will include the elements discussed above, for example, the team set up to resolve inspection differences with the licensee.

Finally, the SC considered means to assure and maintain consistent levels of fire protection safety at all plants. The Working Group researched the guidance documents currently available and how these are applied to old and new plants. The SC discussed findings of the reviewers and inspectors who are close to the issues. As a result of this work, the SC found that the requirements for old and new plants were generally the same but that discrepancies do exist. The application of guidelines, both in the review process and the inspection process, leaves room for interpretation. The SC concluded that several steps needed to be taken in addition to those described above some of which were to assure and maintain consistency.

These steps are to develop and implement standard fire protection license conditions, Standard Review Plan, and Standard Tech Specs for all plants. The license condition developed for this is along the lines of the security plan and QA program (Enclosure 5). The Standard Review Plan should need minimum revision to assure that Appendix R is fully included. The Tech Specs, however, will require more research and development by the Working Group. We need to assure that the Tech Specs are soundly based to assure functioning of fire protection features but which require only those activities which are commensurate with other Tech Spec items in terms of importance to safety. As part of the above tasks, all fire protection guidance needs review and culling to assure that only a consistent and appropriate set remains. This complete set of guidance will be referenced in the Standard Review Plan revision. A last step in achieving uniform technical requirements is the SC recommendation to designate an office as responsible for awareness and resolution of interoffice/region fire protection issues. This is felt to be needed since current fire protection review is conducted within three divisions within NRR and one in IE. Although there is a lead branch responsibility, it is not always kept informed and involved. Therefore, the SC believes that the Director, Division of Engineering in NRR should be designated as the central point of contact.

Conclusion

The Fire Protection Policy Steering Committee concludes that the actions described above will accomplish the goals set forth in your memo of "expediting Appendix R compliance for older plants and assuring consistent levels of fire protection safety at all plants." We believe that these actions will facilitate industry implementation of Appendix R through the use of the "interpretations" and a consistent set of guidance, yet will provide the necessary regulatory tools to guide, monitor, and, where appropriate, enforce this implementation process. We feel strongly that the actions we propose are synergistic and therefore all need to be completed to be most effective.

The Fire Protection Policy Steering Committee has found the assignment to be challenging and rewarding. We would be pleased to brief you on our efforts at your earliest convenience.

Fire Protection Policy Steering Committee

OCT 2 6 1984

FIRE PROTECTION POLICY STEERING COMMITTEE

J. Nelson Grace, Director
Quality Assurance, Safeguards & Inspection Programs
Office of Inspection and Enforcement

Thomas T. Martin, Director
Division of Engineering and Technical Programs
Region I

FOR: William J. Olmstead, Director and Chief Counsel
Regulations Division
Office of the Executive Legal Director

John A. Olshinski, Director
Division of Reactor Safety
Region II

Richard L. Spessard, Director
Division of Reactor Safety
Region III

Richard H. Vollmer, Director
Division of Engineering
Office of Nuclear Reactor Regulation (Chairman)

Enclosures:
1. Memo to H. Denton et al on Review of NRC Fire Protection Policy and Programs.
2. Generic Letter on Fire Protection
3. Interpretations of Appendix R
4. Guidance for Enforcement Actions Concerning Fire Protection Requirements
5. Fire Protection License Condition
6. Appendix R Questions and Answers
7. Steering Committee Memoranda

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

ENCLOSURE 1

SEP 13 1984

MEMORANDUM FOR: Harold R. Denton, Director, NRR
Richard C. DeYoung, Director, IE
Thomas E. Murley, Regional Administrator, R-I
James P. O'Reilly, Regional Administrator, R-II
James G. Keppler, Regional Administrator, R-III
John T. Collins, Regional Administrator, R-IV
John B. Martin, Regional Administrator, R-V

FROM: William J. Dircks
Executive Director for Operations

SUBJECT: REVIEW OF NRC FIRE PROTECTION POLICY AND PROGRAMS

At our August 27th meeting on fire protection, I directed NRR to review and make recommendations for resolving current fire protection issues within two months. The objective of such an effort is to examine all current licensing, inspection, and technical issues and, based on such a review, develop policy recommendations aimed at expediting Appendix R compliance for older plants and assuring consistent levels of fire protection safety at all plants.

Some of the current issues that should be considered are:

- adequacy of current guidance to industry;
- interpretation of the Appendix R requirements vice staff guidance;
- treatment of expected future technical and schedular exemptions into late 1980s and early 1990s;
- comparison of Appendix R and current NTOL plants for fire protection safety;
- adequacy of current inspection practices; and
- identification and resolution plan for any outstanding technical issues.

To implement the above, I understand that you have set up a Steering Committee composed of Messrs. Grace, IE; Martin, R-I; Olmstead, ELD;

Olshinski, R-II; Spessard, R-III; and Vollmer, NRR, Chairman. This Steering Committee is to decide the scope of issues to be considered, meet with HQ and regional personnel as necessary to consider these issues, make assignments as appropriate to a working group headed by Faust Rosa, NRR, for detailed consideration of certain issues, and make recommendations for actions along with supporting bases to me by October 26, 1984. I concur in these assignments and the general charter of the Steering Committee.

All ongoing regulatory actions in your Offices regarding fire protection should be continued and should not be delayed or deferred pending the outcome of this review.

(Signed) William J. Dircks

William J. Dircks
Executive Director for Operations

cc: V. Stello
G. Cunningham
Steering Committee

GENERIC LETTER ON FIRE PROTECTION

TO ALL LICENSEES AND APPLICANTS FOR OPERATING LICENSES

Gentlemen:

SUBJECT: IMPLEMENTATION OF FIRE PROTECTION REQUIREMENTS

In the Spring of 1984, the Commission held a series of Regional Workshops on the implementation of NRC fire protection requirements at nuclear power plants. At those workshops, a package of recently-developed NRC guidance was distributed to each attendee which included NRC staff responses to industry questions and a document titled "Interpretations of Appendix R." The cover memo for the package explained that it was a draft package which would be issued in final form via Generic Letter following the workshops.

The guidance approved by the Commission is appended to this letter, and is in the same format as the draft package, i.e., "Interpretations of Appendix R" and responses have been modified from the draft package, and a number of industry questions raised at or subsequent to the workshops have been added and answered. This package represents the official agency position on all issues covered, and where this guidance differs from previously issued guidance (including Generic Letter 83-33) on this subject, this guidance takes precedence.

In the lettered sections below, some additional topics are covered which also bear on the interpretation and implementation of NRC fire protection requirements. The topics are: (A) schedular exemptions, (B) revised inspection program, (C) documentation required to demonstrate compliance, (D) applicability of GDC-1 to fire protection systems, and (E) notification of the NRC when deficiencies are discovered.

A. Schedular Exemptions

The Appendix R implementation schedule was established by the Commission in 10 CFR 50.48(c), promulgated together with Appendix R in November of 1980. Allowing time to evaluate the need for alternative or dedicated shutdown systems, which require prior NRC approval before installation, and time for design of and NRC review of such systems, the Commission envisioned that implementation of Appendix R would be complete in four to five years, or approximately by the end of 1985. Many schedule extensions were granted by the staff under the "tolling provision," 50.48(c)(6), and under 10 CFR 50.12, the longest of which now extends into 1987. Some licensees have proceeded expeditiously to implement

Appendix R and are now finished or nearly finished with that effort. Others have engaged in lengthy negotiations with the staff while continuing to file requests for schedule extensions, and thereby have barely begun Appendix R modifications needed to comply with Sections III.G and III.L. Schedule extension requests have been received seeking implementation dates of 1990 or beyond.

As the 50.48(c) schedule was intended to be a one-time schedule commencing in the 1980-1982 time frame and ending in the 1985 time frame, extensions well beyond this schedule (particularly where major modifications remain to be completed) undermine the purpose of the schedule, which was to achieve expeditious compliance with NRC fire protection requirements. The NRC will therefore grant no further extensions to the 50.48(c) schedules. When a licensee's schedule expires, compliance is expected and appropriate enforcement action will be taken. If compliance cannot be achieved by that date, the licensee will be required to submit and justify a minimum schedule for completion of fire protection modifications, and to supply interim measures to compensate for the lack of compliance. In submitting a schedule which goes beyond the current 50.48 deadline*, the licensee will be required to demonstrate that it has endeavored in good faith to complete modifications on schedule. A showing of good faith attempt to complete implementation on schedule may mitigate enforcement action for noncompliance with NRC requirements.

The NRC is currently reviewing all dockets of plants covered by the 50.48 schedule to determine schedule deadlines. When this review is completed, each licensee will be informed of the deadlines.

B. <u>Revised Inspection Program</u>

In 1982, the NRC developed an inspection program to verify compliance with the requirements of 10 CFR 50, Appendix R. This program was primarily oriented towards reviewing safe shutdown features of those pre-1979 licensees that had completed Appendix R modifications and selected NTOL plants. From 1982 to the present, approximately seven Appendix R compliance inspections have been performed. In a number of cases, these inspections have discovered that licensees had made significant errors in implementing a number of Appendix R requirements.

In order to expedite compliance verification and to provide the NRC staff with earlier indication of problems associated with implementation of fire protection features, the NRC will conduct fire protection inspections of operating plants and plants currently undergoing operating license review during 1985 to include at least one

* Licensees submitting "living schedules" for NRC approval should be aware that existing 50.48 schedules continue to apply. Licensees intending to include fire protection modifications within a "living schedule" are expected to assign within such schedules the relative safety priorities of remaining fire protection modifications.

site from each licensee who has not been subject to a previous NRC fire protection inspection. This inspection will review completed modifications and, in the case of incomplete modifications, review licensee plans and schedules for completing such modifications.

C. Documentation Required to Demonstrate Compliance

The "Interpretations" document attached to this letter states that, where the licensee chooses not to seek prior NRC review and approval of, for example, a fire area boundary, an evaluation must be performed by a fire protection engineer (assisted by others as needed) and retained for future NRC audit. Evaluations of this type must be written and organized to facilitate review by a person not involved in the evaluation. Guidelines for what such an evaluation should contain may be found in: (1) Section B of Appendix R and (2) Section C.1.b of Branch Technical Position (BTP) CMEB 9.5-1 Rev. 2 dated July 1981. All calculations supporting the evaluation should be available and all assumptions clearly stated at the outset. Failure to have such an evaluation available for an area where compliance with Appendix R is not readily demonstrated will be taken as prima facie evidence that the area does not comply with NRC requirements, and may result in enforcement action.

D. Quality Assurance Requirements Applicable

Fire protection systems must meet the requirements of General Design Criterion 1 of Appendix A to 10 CFR Part 50. For such systems the licensee is therefore required to have and maintain a quality assurance program adequate to assure that these systems will perform their functions when called upon. Fire protection systems are not "safety-related" and are therefore not within the scope of Appendix B to 10 CFR Part 50, unless the licensee has committed to include these systems under the Appendix B program for the plant. NRC guidance for an acceptable quality assurance program for fire protection systems, given in Section C.4 of Branch Technical Position CMEB 9.5-1 Rev.2 dated July 1981, has generally been used in the review and acceptance of approved fire protection programs.

E. Notification of the NRC When Deficiencies are Discovered

Licensees are reminded of their obligation to notify the NRC of fire protection deficiencies which meet the criteria of 10 CFR 50.72 or 10 CFR 50.73 as applicable.

INTERPRETATIONS OF APPENDIX R

1. Process Monitoring Instrumentation

Section III.L.2.d of Appendix R to 10 CFR Part 50 states that "the process monitoring function shall be capable of providing direct readings of the process variables necessary to perform and control" the reactivity control function. In I&E Information Notice 84-09, the staff provides a listing of instrumentation acceptable to and preferred by the staff to demonstrate compliance with this provision. While this guidance provides an acceptable method for compliance with the regulation, it does not exclude other alternative methods of compliance. Accordingly, a licensee may propose to the staff alternative instrumentation to comply with the regulation. While such a submittal is not an exemption request, it must be justified based on a technical evaluation. The licensee may also propose alternatives to actual compliance with the regulation (e.g., instrumentation which does not provide a direct reading of the process variable) by filing an exemption request with adequate justification.

2. Repair of Cold Shutdown Equipment

Section III.L.5 of Appendix R states that when in the alternative or dedicated shutdown mode, "equipment and systems comprising the means to achieve and maintain cold shutdown conditions shall not be damaged by fire; or the fire damage to such equipment and systems shall be limited so that the systems can be made operable and cold shutdown can be achieved within 72 hours." This is not to be confused with the requirements in Section III.G.1.b of Appendix R.

Section III.G.1.b contains the requirements for normal shutdown modes utilizing the control room or emergency control station(s) capabilities. The fire areas falling under the requirements of III.G.1.b are those for which an alternative or dedicated shutdown capability is not being provided. For these fire areas, Section III.G.1.b requires only the capability to repair the systems necessary to achieve and maintain cold shutdown from either the control room or emergency control station(s) within 72 hours, not the capability to repair and achieve cold shutdown within 72 hours as required for the alternative or dedicated shutdown modes by Section III.L (noted above).

With regard to areas involving normal shutdown, however, Section I of Appendix R states that repairs must be made using only onsite capabilities. After repairs are made, cold shutdown can be achieved on a reasonable schedule using any available power source.

3. Fire Damage

Appendix R to 10 CFR Part 50 utilizes the term "free of fire damage." In promulgating Appendix R, the Commission has provided methods acceptable for assuring that necessary structures, systems and components are free of fire damage (see Section III.G.2a, b and c), that is, the structure, system or component under consideration is capable of performing its intended function during and after the postulated fire, as needed. Licensees seeking exemptions from Section III.G.2 must show that the alternative proposed provides reasonable assurance that this criterion is met. (Note also that Section III.G.2 applies only to equipment needed for hot shutdown. Therefore, an exemption from III.G.2 for cold shutdown equipment is not needed.)

4. Fire Area Boundaries

The term "fire area" as used in Appendix R means an area sufficiently bounded to withstand the hazards associated with the area and, as necessary, to protect important equipment within the area from a fire outside the area. In order to meet the regulation, fire area boundaries need not be completely sealed floor-to-ceiling, wall-to-wall boundaries. However, all unsealed openings should be identified and evaluated. Where fire area boundaries were not approved under the BTP process, or where such boundaries are not wall-to-wall, floor-to-ceiling boundaries with all penetrations sealed to the fire rating required of the boundaries, licensees must perform an evaluation to assess the adequacy of fire boundaries in their plants to determine if the boundaries will withstand the hazards associated with the area and protect important equipment within the area from a fire outside the area. This analysis must be performed by at least a fire protection engineer and, if required, a systems engineer. Although not required, licensees may submit their evaluations for staff review and concurrence. In any event, these analyses must be retained by the licensees for subsequent NRC audits.

5. Automatic Detection and Suppression

Sections III.G.2.b and III.G.2.c of Appendix R state that "In addition, fire detectors and an automatic fire suppression system shall be installed in the fire area..." Other provisions of Appendix R also use the phrase "fire detectors and an automatic fire suppression system in the fire area..." (see e.g., Section III.G.2.e).

In order to comply with these provisions, suppression and detection sufficient to protect against the hazards of the area must be installed. In this regard, detection and suppression providing less than full area coverage may be adequate to comply with the regulation. Where full area suppression and detection is not installed, licensees must perform an evaluation to assess the adequacy of partial suppression and detection to protect against the hazards in the area. The evaluation must be performed

by a fire protection engineer and, if required, a systems engineer. Although not required, licensees may submit their evaluations to the staff for review and concurrence. In any event, the evaluations must be retained for subsequent NRC audits. Where a licensee is providing no suppression or detection, an exemption must be requested.

6. Alternative or Dedicated Shutdown

Section III.G.3 of Appendix R provides for "alternative or dedicated shutdown capability and its associated circuits, independent of cables, systems or components in the area, room, or zone under consideration." While "independence" is clearly achieved where alternative shutdown equipment is outside the fire area under consideration, this is not intended to imply that alternative shutdown equipment in the same fire area but independent of the room or the zone cannot result in compliance with the regulation. The "room" concept must be justified by submission of a detailed fire hazards analysis that demonstrates a single fire will not disable both normal shutdown equipment and the alternative shutdown capability.

GUIDANCE FOR ENFORCEMENT ACTIONS CONCERNING
FIRE PROTECTION REQUIREMENTS

1. General Guidance

   A. Fire protection requirements are delineated by 10 CFR 50 Appendix A General Design Criterion 3, 10 CFR 50.48, 10 CFR 50 Appendix R, Facility License Conditions, facility technical specifications and other legally binding requirements, as applicable. A Notice of Violation will be issued for violation of requirements. However, failure to meet fire protection commitments will be designated as deviations.

   B. Failures to meet regulatory requirements for protecting trains of equipment required for achieving and maintaining safe hot or cold shutdown are serious violations. The specific violations should be reviewed individually and as a group to determine their root cause(s). This guidance gives examples of violations at various severity levels and should be used to determine the appropriate enforcement action. For purposes of this guidance, required structures, systems, and components are those which are necessary to achieve and maintain hot and/or cold safe shutdown and which require the application of fire protection features as described in the licensee's fire hazards analysis report and safety evaluation report.

   C. Fire protection violations may involve inoperable or inadequate: fire barriers, separation, suppression or detection systems, repair parts, procedures or other conditions or items required to protect safe shutdown equipment from fire and/or permit the operation of safe shutdown equipment during a fire or to restore safe shutdown equipment to service following an actual fire.

   D. Numerous violations of fire protection requirements which individually may be classified at lower severity levels may cumulatively be classified at a higher severity level due to inadequate implementation of the fire protection program.

2. Severity Categories

   A. Severity I. Violations of fire protection requirements established to protect or enable operation of safe hot shutdown equipment concurrent with an actual fire which damages that equipment such that safe hot shutdown could not be achieved or maintained using the equipment dedicated for the purpose.

   B. Severity II. Violations of fire protection requirements established to protect or enable operation of safe cold shutdown equipment concurrent with an actual fire which damages that equipment such that safe cold shutdown could not have been achieved and maintained using the equipment dedicated for this purpose in accordance with the applicable requirements.

   C. Severity III. Violations of fire protection requirements established to protect or enable operation of safe shutdown equipment such that a fire in the area could damage that equipment to the extent that safe hot or cold shutdown could not have been achieved and maintained using the equipment dedicated for this purpose in accordance with applicable requirements. Failure to have a written evaluation available for an area where compliance with Appendix R is not readily demonstrated will be taken as prima facie evidence that the area does not comply with NRC requirements and may result in enforcement action at the severity level.

   D. Severity IV. Violations of one or more fire protection requirements that do not result in a Severity Level I, II or III violation and which have more than minor safety or environmental significance.

   E. Severity V. Violations of one or more fire protection requirements that have minor safety or environmental significance.

FIRE PROTECTION LICENSE CONDITION

1. The licensee shall implement and maintain in effect all provisions of the approved fire protection program as described in the Final Safety Analysis Report for the facility and as approved in the SER subject to provisions 2 and 3 below.

2. The licensee may make no change to the approved fire protection program which would decrease the level of fire protection in the plant without prior approval of the Commission. To make such a change the licensee must submit an application for license amendment pursuant to 10 CFR 50.90.

3. The licensee may make changes to features of the approved fire protection program which do not decrease the level of fire protection without prior Commission approval provided (a) such changes do not otherwise involve a change in a license condition or technical specification or result in an unreviewed safety question (see 10 CFR 50.59), and (b) such changes do not result in failure to carry out the fire protection program approved by the Commission prior to licence issuance. The licensee shall maintain, in an auditable form, a current record of all such changes, including an analysis of the effects of the change on the fire protection program, and shall make such records available to NRC inspectors upon request. All changes to the approved program made without prior Commission approval shall be reported annually to the Director of the Office of Nuclear Reactor Regulation, together with supporting analyses.

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1 INTRODUCTION

A major fire damaging safe shutdown equipment occurred at the Browns Ferry Nuclear Station in March 1975. The fire damaged over 1600 electrical cables and caused the temporary unavailability of some core cooling systems. Because this fire did substantial damage, the NRC established a Special Review Group which initiated an evaluation of the need for improving the fire protection programs at all nuclear power plants. The group found serious design inadequacies regarding fire protection at Browns Ferry, and its report, "Recommendations Related to Browns Ferry Fire" (NUREG-0050, February 1976), contained over fifty recommendations regarding improvements in fire prevention and control in existing facilities. The report also called for the development of specific guidance for implementing fire protection regulations, and for a comparison of that guidance with the fire protection program at each operating plant.

NRC developed technical guidance from the technical recommendations in the Special Group's report, and issued those guidelines as Branch Technical Position Auxiliary Power Conversion Systems Branch 9.5-1 (BTP APCSB 9.5-1), 1/ "Guidelines for Fire Protection for Nuclear Power Plants." This guidance did not apply to plants operating at that time. Guidance to operating plants was provided later in Appendix A 2/ to BTP APCSB 9.5-1 which, to the extent practicable, relies on BTP APCSB 9.5-1. The guidance in these documents was also published for public comment as Regulatory Guide 1.120, "Fire Protection for Nuclear Power Plants" (June 1976). In response to public comment, the NRC issued an extensively revised version of Regulatory Guide 1.120 for further public comment.

1/ Rather than serving as inflexible, legal requirements that must be followed by licensees, issuances such as regulatory guides and branch technical positions are meant to give guidance to licensees concerning those methods the staff finds acceptable for implementing the general criteria embodied in the NRC's rules. See, e.g., Petition for Emergency & Remedial Action, CLI-78-6, 7 NRC 400, 406 (1978); Gulf States Utilities Company (River Bend Station, Units 1 and 2) ALAB-444, 6 NRC 760, 772 (1977).

2/ Guidelines for Fire Protection for Nuclear Power Plants Docketed Prior to July 1, 1976.

In May 1976, the NRC asked licensees to compare operating reactors with BTP APCSB 9.5-1, and in September 1976, those licensees were informed that the guidelines in Appendix A would be used to analyze the consequences of a fire in each plant area. In September 1976 the licensees, were also requested to provide a fire hazards analysis that divided the plant into distinct fire areas and show that redundant systems required to achieve and maintain cold shutdown are adequately protected against damage by a fire. Early in 1977 each licensee responded with a Fire Protection Program Evaluation which included a Fire Hazard Analysis. These evaluations and analyses identified aspects of licensees' fire protection programs that did not conform to the NRC guidelines. Thereafter, the staff initiated discussions with all licensees aimed at achieving implementation of fire protection guidelines by October 1980. The staff held many meetings with licensees, conducted extensive correspondence with them, and visited every operating reactor. As a result, many fire protection items were resolved, and agreements were included in Fire Protection Safety Evaluation Reports issued by the NRC. Several fire protection issues remained unresolved with a number of licensees.

By early 1980, most operating plants had implemented most of the guidelines in Appendix A. However, as the Commission noted in its Order of May 23, 1980, the fire protection program has had some significant problems with implementation. Despite the staff's efforts, several licensees had expressed continuing disagreement with, and refused to adopt recommendations relating to several generic issues, including the requirements for fire brigade size and training, water supplies for fire suppression systems, alternate and dedicated shutdown capability, emergency lighting, qualifications of seals used to enclose places where cables penetrated fire barriers, and the prevention of reactor coolant pump lubrication system fires. To establish a definitive resolution of these contested subjects in a manner consistent with the general guidelines in Appendix A to the BTP and to assure timely compliance by licensees, the Commission issued a proposed fire protection rule and its Appendix R, which was described as setting out minimum fire protection requirements for the unresolved issues (45 Fed. Reg. 36082 May 29, 1980). The fire protection features addressed included protection of safe shutdown capability, emergency lighting, fire barriers, associated circuits, reactor coolant pump lubrication system, and alternate shutdown systems. The Commission stated that it expected all modifications (except for alternate and dedicated shutdown capability) to be implemented by November 1, 1980. 4/

3/ 11 NRC 707, 718 (1980)

4/ Id. at 719

As originally proposed (Federal Register Vol. 45. No. 1&5, May 22, 1980), Appendix R would have applied to all plants including those for which the staff had previously accepted other fire protection modifications. After analyzing comments on the rule, the Commission determined that only three of the fifteen items in Appendix R were of such safety significance that they should apply to all plants, including those for which alternative fire protection actions had been approved previously by the staff. These items are protection of safe shutdown capability (including alternate shutdown systems), emergency lighting, and the reactor coolant pump lubrication system. Accordingly, the final rule required all reactors licensed to operate before January 1, 1979, to comply with these three items even if the NRC had previously approved alternative fire protection features in these areas (45 Fed. Reg. 76602 Nov. 19, 1980). However, the final rule is more flexible than the proposed rule because Item III. G now provides three alternative fire protection features which do not require analysis to demonstrate the protection of redundant safe shutdown equipment, and reduces the acceptable distance in the physical separation alternative from fifty feet to twenty feet. In addition, the rule now also provides an exemption procedure which can be initiated by a licensee's assertion that any required fire protection feature will not enhance fire protection safety in the facility or that such modifications may be detrimental to overall safety (10 CFR 50.48(c)(6)). If the Director, Nuclear Reactor Regulation determines that a licensee has made a <u>prima facie</u> showing of a sound technical basis for such an assertion, then the implementation dates of the rule are tolled until final Commission action on the exemption request.

Most licensees requested and were granted additional time to perform their reanalysis, propose modifications to improve post fire shutdown capability and to identify exemptions for certain fire protection configurations. In reviewing some exemption requests, the staff noted that some licensees had made significantly different interpretations of certain requirements. These differences were identified in the staff's draft SER's. These differences were also discussed on several occasions with the cognizant licensee as well as the Nuclear Utility Fire Protection Group. These discussions culminated in the issuance of generic letter 83-33.

2. OVERVIEW

Section 50.48 Fire Protection of 10 CFR Part 50 requires that each operating nuclear power plant have a fire protection plan that satisfies General Design Criterion 3 of Appendix A to 10 CFR 50. It specifies what should be contained in such a plan and lists the basic fire protection guidelines for this plan. It requires that the Fire Protection Safety Evaluation Report which has been issued for each operating plant state how these guidelines were applied to each facility.

Section 50.48 also requires that all plants with operating licenses prior to January 1, 1979 satisfy the requirements of Section III.G, III.J and III.O, and other Sections of Appendix R where approval of similar features had not been obtained prior to the effective date of Appendix R. By a separate action, the Commission approved the staff's requirement that all plants to receive their operating license after January 1, 1979 also satisfy the requirements of Sections III.G, III.J and III.O and that a fire protection license condition be established. Deviations from Appendix R requirements for pre-1979 plants are processed under the exemption process. Deviation from other guidelines are identified and evaluated in the Safety Evaluation Report.

A standard fire protection license condition has been developed and will be included in each new operating license. Present operating licenses will be amended to include the standard license condition.

The Regions initiated inspections of operating plants and identified several significant items of non-compliance. The Nuclear Utility Fire Protection Group requested interpretations of certain Appendix R requirements and provided a list of questions that they thought should be discussed with the industry. The NRC held workshops in each Region to assist the industry in understanding the NRC's requirements and to improve the Staff's understanding of the industry's concerns.

This document presents the NRC's response to the questions posed by the industry and supplemented with additional questions identified at the workshops as being of interest to the industry or the staff. These responses may be used as guidance for design, review and inspection activities. The questons have been reformatted according to their applicability to Sections of Appendix R, BTP CMEB 9.5-1, licensing policy or inspection policy.

3. SECTION III G, FIRE PROTECTION OF SAFE SHUTDOWN CAPABILITY

3.1 Fire Area Boundaries

3.1.1 Fire Area Definition

Question

Section III.G states the fire protection features required for cables and equipment or redundant trains of systems required to achieve and maintain hot shutdown that are located within the same fire area. Is the fire area of Section III.G, the same fire area referred to in BTP APCSB 9.5-1, Appendix A; and the supplementary guidance of September 1976?

Response

Yes. Prior to the issuance of Appendix R. fire area boundaries should have been established using the BTP guidelines. The concept of fire areas was described in BTP APCSB 9.5-1. Also, definitions were given for fire areas, fire barriers and fire ratings. The same fire areas were referred to in our supplementary guidance of 1977 and Appendix A to BTP APCSB 9.5-1. The same fire areas concepts definitions are carried over to BTP CMEB 9.5-1.

During the "Appendix A" reviews, some licensees performed their fire hazards analysis using these definitions, some did not. Licensees sometimes called "fire zones" "fire areas." Section III.G sets forth fire protection alternatives within a fire area. If new fire areas are identified they should be established using the BTP guidelines.

The concept of fire areas was described in BTP APCSB 9.5-1:

"C. Establishment and Use of Fire Areas

The concept of separate fire areas for each division of safety equipment which requires redundancy will facilitate the installation of automatic water extinguishing systems since it will reduce the possibility of water damaging redundant safety-related equipment.

Fire areas should be established based upon the amount of combustible material present and considering suitably chosen design basis fires so that adequate protection can be provided for safety-related systems and equipment. Design basis fires are those fires that result in the most severe exposure

to the area or systems being considered. For this condition, it is assumed that no manual or automatic fire suppression action has been started and the fire has reached its peak burning rate and involves all combustibles present.

Within each area special attention should be given to limiting the amount of combustible material and to providing effective barriers and fire resistive coatings to reduce the spreading of a fire in these areas. A design basis fire should be assumed and provisions should be made to limit the consequence of such a fire by providing fire barriers with suitable separation between redundant systems and components which are provided to carry out required safety functions. This separation is enhanced if the plant is divided into suitable fire areas since redundant safety equipment can then be placed in separate fire areas.

Particular design attention should be given to the use of separate isolated fire areas for redundant cables to avoid loss of redundant safety-related cables.

Provisions should also be made to limit the consequences of a fire by suitable design of the ventilation systems so that the spread of the products of combustion to other areas of the plant is prevented. Means should be provided to ventilate or isolate the area as required. The power supply and controls for the area ventilation system should be from outside the area, and the power and control cables should not pass through the area.

The fire detection systems should be designed using detectors of the right types at locations suitable to detect the particular type of fire expected in each area.

In the design, consideration should be given to provide personnel access to and escape routes from each fire area. The emergency plans for all plants should lay out access and escape routes to cover the event of a fire in critical areas of the plant."

In addition, definitions of the basic fire area components were given:

"Fire Area - that portion of a building or plant that is separated from other areas by boundary fire barriers (walls, floors or roofs) with any openings or penetrations protected with seals or closures having a fire resistance rating equal to that of the barrier.

Fire Barrier - those components of construction (walls, floors and roofs) that are rated by approving laboratories in hours for resistance to fire to prevent the spread of fire.

Fire Rating - refers to the endurance period of a fire barrier or structure and defines the period of resistance to a standard fire exposure elapsing before the first critical point in behavior is observed. (Refer to NFPA 251).

Fire Zones - subdivisions of fire areas in which the fire suppression systems are designed to combat particular types of fires. The concept of fire zone aids in defining to the fire fighter the fire parameters and the actions which would be necessary."

The supplementary guidance, stated information to be provided in the fire hazards analysis for each fire area or fire zone established.

"In order to perform a proper fire hazards analysis, the services of a qualified fire protection engineer should be utilized. To demonstrate the results of the fire hazards analysis the following information must be provided:

1. Provide plan and elevation views of the plant that show the plant as divided into distinct fire areas. Provide a description of the various systems, both safety-related and non-safety-related, which occupy the fire area and could provide cooling to the core to safely shutdown the reactor, including decay heat removal. Provide a description of areas of the plant that contain radioactive material that may be released to the exclusion area or beyond should a fire occur in those areas.

   For each fire area, provide the following:

   a) Describe the fire barrier that defines the fire area; the consequences of the design basis fire for that area; the consequences of the fire if the fire protection system functions as designed.

   b) Identify the safety related equipment and associated cabling. Provide the design criteria for the fire protection related to such equipment. Provide the design criteria for protection of such equipment against inadvertent operation, careless operation or rupture of extinguishing systems.

   c) Provide a list of the type, quantity, and other pertinent characteristics of combustible materials associated with each fire area.

   d) Provide a list of the fire loading which represent the combustibles identified in (c) above for each fire area.

3.1.2 Previously Accepted Fire Area Boundaries

QUESTION

If a fire area boundary was described as a rated barrier in the 1977 fire hazards analysis, no open items existed in this area in the Appendix A SER, and the barriers have not been altered, then need those barriers be reviewed by licensees or the Staff under Appendix R?

RESPONSE

If a fire area boundary was described as a rated barrier in the 1977 fire hazards analysis, and was evaluated and accepted in a published SER, the fire area boundary need not be reviewed as part of the re-analysis for compliance with Section III.G of Appendix R. Openings in the fire barriers, if any, should have been specifically identified and justified in the fire hazards analysis performed in the Appendix A process. If openings in the fire area boundaries were not previously evaluated, such an evaluation should be performed as a basis for assessing compliance with Appendix R. See Item #4 of the "Interpretations of Appendix R."

In BTP CMEB 9.5-1, Fire Barrier is defined as:

"Fire Barrier - those components of construction (walls, floors, and the supports), including beams, joists, columns, penetration seals or closures, fire doors, and fire dampers that are rated by approving laboratories in hours of resistance to fire and are used to prevent the spread of fire."

The term "fire area" as used in Appendix R means an area sufficiently bounded to withstand the hazards associated with the fire area and, as necessary, to protect important equipment within the fire area from a fire outside the area. In order to meet the regulation, fire area boundaries need not be completely sealed with floor-to-ceiling and/or wall-to-wall boundaries. Where fire area boundaries were not approved under the Appendix A process, or where such boundaries are not wall-to-wall or floor-to-ceiling boundaries with all penetrations sealed to the fire rating required of the boundaries, licensees must perform an evaluation to assess the adequacy of fire area boundaries in their plants to determine if the boundaries will withstand the hazards associated with the area and protect important equipment within the area from a fire outside the area. This analysis must be performed by at least a fire protection engineer and, if required, a systems engineer. Although not required, licensees may submit their evaluations for Staff review and concurrence. In any event, these analyses must be retained by the licensees for subsequent NRC audits.

Enclosure 7. Finally, the SC received significant input from Thomas Wambach, who acted as Secretary to the SC, and the Working Group, headed by Faust Rosa, NRR, and composed of V. Benaroya, C. Grimes, and V. Moore of NRR; S. Richardson, IE; W. Shields, OELD; and C. Anderson, T. Conlon, and W. Little of Regions I, II and III respectively.

The recommendations of the SC are provided below. We believe that this responds to your direction, and when fully implemented, represent actions that will indeed expedite Appendix R compliance and assure consistent levels of fire protection safety at all plants. The SC is aware that not all parties will be fully satisfied with these actions. Nor have our recommendations been reviewed by the cognizant Offices or Regions. However, we believe that they represent sound judgments balanced with other competing safety priorities, and that with your approval the plan can and should be initiated promptly.

Recommendations

To expedite Appendix R compliance and assure consistent levels of fire protection safety at all plants, the Fire Protection Steering Committee recommends that the following actions be taken:

1. Promptly issue the enclosed Generic Letter (Enclosure 2) informing all licensees that:

   (a) Extensions to the 50.48(c) schedules will no longer be granted;

   (b) An expedited fire protection inspection program will be instituted;

   (c) Documentation of valid analyses supporting fire protection features must be available for inspection;

   (d) Quality assurance applicable to fire protection systems is that required by GDC-1 of Appendix A to 10 CFR Part 50; and

   (e) The interpretations of Appendix R, (Enclosure 3) which should facilitate industry implementation of Appendix R and the responses to industry questions (Enclosure 6) represent the official agency position on all issues covered. (It should be noted that The Commission requested these documents for their review prior to issuance to industry.)

3.1.5 Fire Zones

QUESTION

Appendix R, Section III.G.3 states "alternative or dedicated shutdown capability and its associated circuits, independent of cables, systems or components in the area room or zone under consideration...." What is the implied utilization of a room or zone concept under Section III.G of Appendix R? The use of the phraseology "area, room or zone under consideration" is used again at the end of the Section III.G.3. Does the requirement for detection and fixed suppression indicate that the requirement can be limited to a fire zone rather than throughout a fire area? Under what conditions and with what caveats can the fire zone concept be utilized in demonstrating conformance to Appendix R?

RESPONSE

Section III.G was written after NRC's multi-discipline review teams had visited all operating power plants. From these audits, the NRC recognized that it is not practical and may be impossible to subdivide some portions of an operating plant into fire areas. In addition, the NRC recognized that in some cases where fire areas are designated, it may not be possible to provide alternate shutdown capability independent of the fire area and, therefore, would have to be evaluated on the basis of fire zones within the fire area. The NRC also recognized that because some licensees had not yet performed a safe shutdown analysis, these analyses may identify new unique configurations.

To cover the large variation of possible configurations, the requirements of Section III.G were presented in three parts:

- Section III.G.1 requires one train of hot shutdown systems be free of fire damage and damage to cold shutdown systems be limited.
- Section III.G.2 provides certain separation, suppression and detection requirements within fire areas; where such requirements are met, analysis is not necessary.
- Section III.G.3 requires alternative dedicated shutdown capability for configurations that do not safisfy the requirements of III.G.2 or where fire suppressants released as a result of fire fighting, rupture of the system or inadvertent operation of the system may damage redundant equipment.

Discussion

The recommended actions are grouped into three main areas dealing with (1) guidance to industry, (2) an expedited program of fire protection inspections, and (3) a general upgrading of regulatory documents to reach and maintain consistent fire protection safety. This discussion section will focus broadly on what the SC found during its deliberations to warrant the focus of these recommendations and will indicate how this satisfies the agenda of issues cited in your memo of September 13. Details on these issues are provided in the record of the SC deliberations contained in Enclosure 7.

With regard to guidance to industry, the SC concludes that adequate technical guidance had been issued but that there were areas where confusion could arise. It was not clear where exemptions were needed, for example. However, a diligent reading of Appendix R and other staff documents did provide the basis for the satisfactory implementation of Appendix R at Calvert Cliffs. The SC concluded that it was neither needed nor appropriate to develop new guidance, rather, bringing current technical and implementation guidance together in one Generic Letter and make the SRP, Tech Specs, and licenses consistent would suffice. The Generic Letter makes clear (1) that extensions to the 50.48(c) schedules will no longer be granted, (2) that an expedited inspection program will be instituted to see what fire protection fixes are in place and give licensees the inspection team judgements on the acceptability of future modifications, and (3) that the licensee judgements must be backed by documented and valid analyses. The SC believes that this will demonstrate to the licensee what action he must take and what our inspections will look for. The Generic Letter notes that, although the 50.48(c) schedules will not be extended, the relative safety priroities of fire protection modifications need to be considered in the development of "living schedules." One item of guidance in the Generic Letter that had not been uniformly disseminated is that the QA applicable to fire protection features is that required by GDC-1. This would not attempt to backfit any QA requirements. Rather it would assure that future design, procurement, installation, testing and maintenance of fire protection features would receive high industrial quality attention. The SC believes that this initiative fully responds to the first three issues in your memo of September 13.

Turning now to the inspection program, the SC found that the current inspections are generally satisfactory but that steps must be taken to indicate NRC's view of the importance of expediting implementation of Appendix R. These steps are to (1) speed up the inspection process, (2) develop a sound policy for fire protection enforcement actions, and (3) issue enforcement actions currently pending. These steps, in our view, would help expedite licensee compliance because it would raise industry's

3.2 Fire Barrier Qualification

3.2.1 Acceptance Criteria

QUESTION

Recently the Staff has applied a 325°F cold side temperature criterion to its evaluation of the acceptability of one-hour and three-hour fire barrier cable tray wraps. This criterion is not in Branch Technical Position (BTP) APCSB 9.5-1, Appendix A as an accceptance criterion for fire barrier cable tray wraps and is not contained in Appendix R. It appears to represent post-Appendix R guidance. What is the origin of this criterion and why is it applicable to electrical cables where insulation degradation does not begin until jacket temperatures reach 450°F to 650°F?

RESPONSE

Fire barriers relied upon to protect shutdown related systems to meet the requirements of III.G.2 need to have a fire rating of either one or three hours. §50.48 references BTP APCSB 9.5-1, where the fire protection definitions are found. Fire rating is defined:

"Fire Rating - the endurance period of a fire barrier or structure; it defines the period of resistance to a standard fire exposure before the first critical point in behavior is observed (see NFPA 251)."

The acceptance criteria contained in Chapter 7 of NFPA 251, "Standard Methods of Fire Tests of Building Construction and Materials," pertains to non-bearing fire barriers. These criteria stipulate that transmission of heat through the barrier "shall not have been such as to raise the temperature on its unexposed surface more than 250°F above its initial temperature." The ambient air temperature at the beginning of a fire test usually is between 50°F and 90°F. It is generally recognized that 75°F represents an acceptable norm. The resulting 325°F cold side temperature criterion is used for cable tray wraps because they perform the fire barrier function to preserve the cables free of fire damage. It is clear that cable that begins to degrade at 450°F is free of fire damage at 325°F.

These steps are to develop and implement standard fire protection license conditions, Standard Review Plan, and Standard Tech Specs for all plants. The license condition developed for this is along the lines of the security plan and QA program (Enclosure 5). The Standard Review Plan should need minimum revision to assure that Appendix R is fully included. The Tech Specs, however, will require more research and development by the Working Group. We need to assure that the Tech Specs are soundly based to assure functioning of fire protection features but which require only those activities which are commensurate with other Tech Spec items in terms of importance to safety. As part of the above tasks, all fire protection guidance needs review and culling to assure that only a consistent and appropriate set remains. This complete set of guidance will be referenced in the Standard Review Plan revision. A last step in achieving uniform technical requirements is the SC recommendation to designate an office as responsible for awareness and resolution of interoffice/region fire protection issues. This is felt to be needed since current fire protection review is conducted within three divisions within NRR and one in IE. Although there is a lead branch responsibility, it is not always kept informed and involved. Therefore, the SC believes that the Director, Division of Engineering in NRR should be designated as the central point of contact.

Conclusion

The Fire Protection Policy Steering Committee concludes that the actions described above will accomplish the goals set forth in your memo of "expediting Appendix R compliance for older plants and assuring consistent levels of fire protection safety at all plants." We believe that these actions will facilitate industry implementation of Appendix R through the use of the "interpretations" and a consistent set of guidance, yet will provide the necessary regulatory tools to guide, monitor, and, where appropriate, enforce this implementation process. We feel strongly that the actions we propose are synergistic and therefore all need to be completed to be most effective.

The Fire Protection Policy Steering Committee has found the assignment to be challenging and rewarding. We would be pleased to brief you on our efforts at your earliest convenience.

Fire Protection Policy Steering Committee

4. The application or "end use" of the fire barrier is unchanged from the tested configuration. For example, the use of a cable tray barrier to protect a cable tray which differs in configuration from those that were tested would be acceptable. However, the use of structural steel fire proofing to protect a cable tray assembly would not be acceptable.

5. The configuration has been reviewed by a qualified fire protection engineer and found to provide an equivalent level of protection.

3.2.3 Fire Door Modifications

QUESTION

Where labeled and rated fire doors have been modified to incorporate security hardware or for flooding protection, is an exemption from Appendix R required?

RESPONSE

Where a door is part of a fire area boundary, and the modification does not effect the fire rating (for example, installation of security "contacts"), no further analysis need be performed. If the modifications could reduce the fire rating (for example, installation of a vision panel), the fire rating of the door should be reassessed to ensure that it continues to provide adequate margin considering the fire loading on both sides.

An exemption is required if fire doors installed in a fire barrier used to satisfy Section III.G.2 are modified such that the labeled rating no longer applies.

3.3 Structural Steel

3.3.1 NFPA Approaches

QUESTION

Does the NRC's definition of structural steel supporting fire barriers completely accomodate approaches described in NFPA guidance documents and standards?

Enclosures:
1. Memo to H. Denton et al on Review of NRC Fire Protection Policy and Programs.
2. Generic Letter on Fire Protection
3. Interpretations of Appendix R
4. Guidance for Enforcement Actions Concerning Fire Protection Requirements
5. Fire Protection License Condition
6. Appendix R Questions and Answers
7. Steering Committee Memoranda

RESPONSE

In general, cable tray supports should be protected, regardless of whether there is a sprinkler system. If (1) the qualification tests were performed on wrapped cable trays with unprotected supports, and the supports are shown to be adequate, or (2) a structural analysis is performed which demonstrates failure of the unprotected support(s) will not cause a loss of the cable tray fire barrier, then they need not be protected.

An exemption is not required; however, the qualification tests and applicability or the structural evaluation should be documented and available for audit.

3.4 Automatic Suppression System

3.4.1 Water Density

QUESTION

Staff guidance provided in Generic Letter 83-33 concerning automatic suppression coverage of fire areas interprets the phrase "in the fire area" in Section III.G as meaning "throughout the fire area." What delivered water density or occupancy standard as specified in NFPA-STD-13 must be achieved to meet this guidance?

RESPONSE

Individual plant areas are diverse in nature. The designer should determine the particular water density or occupancy classification. Those areas which contain a limited quantity of in-situ and anticipated transient combustibles and which feature contents such as tanks and piping, may be considered as "Ordinary Hazard (Group 1)", as defined by NFPA Standard No. 13. For those areas containing large amounts of cables or flammable liquids, an occupancy classification of "Extra Hazard" may be warranted. The decision as to which classification should be applied should be made by a qualified fire protection engineer.

Once the occupancy classification is determined, the minimum water density should be based on the Density Curves in table 2.2.1(B) of NFPA 13. Any density equal to or in excess of the curves would be in conformance with our guidelines as delineated in Section C.6.c of BTP CMEB 9.5-1.

Olshinski, R-II; Spessard, R-III; and Vollmer, NRR, Chairman. This Steering Committee is to decide the scope of issues to be considered, meet with HQ and regional personnel as necessary to consider these issues, make assignments as appropriate to a working group headed by Faust Rosa, NRR, for detailed consideration of certain issues, and make recommendations for actions along with supporting bases to me by October 26, 1984. I concur in these assignments and the general charter of the Steering Committee.

All ongoing regulatory actions in your Offices regarding fire protection should be continued and should not be delayed or deferred pending the outcome of this review.

(Signed) William J. Dircks

William J. Dircks
Executive Director for Operations

cc: V. Stello
G. Cunningham
Steering Committee

3.4.4 Fixed Suppression System In Fire Area

QUESTION

Are fixed suppression systems required by Section III G.3 to be throughout the fire area, room or zone under consideration?

RESPONSE

No, but partial coverage must be properly justified and documented.

See Item #5 of the "Interpretations of Appendix R."

"...suppression less than full area coverage may be adequate to comply with the regulation. Where full area suppression and detection is not installed, licensees must perform an evaluation to assess the adequacy and necessity of partial suppression and detection in an area. The evaluation must be performed by a fire protection engineer and, if required, a systems engineer. Although not required, licensees may submit their evaluations to the staff for review and concurrence. In any event, the evaluations must be retained for subsequent NRC audits..."

3.4.5 Sprinkler Head Location

QUESTION

If stacks of horizontal or vertical cable trays extend from ceiling to floor, are sprinkler heads required (1) under the lowest horizontal trays, near the floor for vertical trays; (2) at some intermediate level between the floor and ceiling, and (3) at the ceiling?

RESPONSE

Sprinkler heads should be located at the ceiling. Sprinkler heads at other locations may be necessary depending upon the hazard and the cumulative effect of the obstructions to the discharge of water from the sprinkler head. The sprinkler system design should meet NFPA 13.

Appendix R and are now finished or nearly finished with that effort. Others have engaged in lengthy negotiations with the staff while continuing to file requests for schedule extensions, and thereby have barely begun Appendix R modifications needed to comply with Sections III.G and III.L. Schedule extension requests have been received seeking implementation dates of 1990 or beyond.

As the 50.48(c) schedule was intended to be a one-time schedule commencing in the 1980-1982 time frame and ending in the 1985 time frame, extensions well beyond this schedule (particularly where major modifications remain to be completed) undermine the purpose of the schedule, which was to achieve expeditious compliance with NRC fire protection requirements. The NRC will therefore grant no further extensions to the 50.48(c) schedules. When a licensee's schedule expires, compliance is expected and appropriate enforcement action will be taken. If compliance cannot be achieved by that date, the licensee will be required to submit and justify a minimum schedule for completion of fire protection modifications, and to supply interim measures to compensate for the lack of compliance. In submitting a schedule which goes beyond the current 50.48 deadline*, the licensee will be required to demonstrate that it has endeavored in good faith to complete modifications on schedule. A showing of good faith attempt to complete implementation on schedule may mitigate enforcement action for noncompliance with NRC requirements.

The NRC is currently reviewing all dockets of plants covered by the 50.48 schedule to determine schedule deadlines. When this review is completed, each licensee will be informed of the deadlines.

B. Revised Inspection Program

In 1982, the NRC developed an inspection program to verify compliance with the requirements of 10 CFR 50, Appendix R. This program was primarily oriented towards reviewing safe shutdown features of those pre-1979 licensees that had completed Appendix R modifications and selected NTOL plants. From 1982 to the present, approximately seven Appendix R compliance inspections have been performed. In a number of cases, these inspections have discovered that licensees had made significant errors in implementing a number of Appendix R requirements.

In order to expedite compliance verification and to provide the NRC staff with earlier indication of problems associated with implementation of fire protection features, the NRC will conduct fire protection inspections of operating plants and plants currently undergoing operating license review during 1985 to include at least one

* Licensees submitting "living schedules" for NRC approval should be aware that existing 50.48 schedules continue to apply. Licensees intending to include fire protection modifications within a "living schedule" are expected to assign within such schedules the relative safety priorities of remaining fire protection modifications.

3.5.2 Floor-to-Floor Separation

QUESTION

Where redundant circuits are separted by floor elevation but are within the same fire area due to open hatchways, stairs, etc., what is the NRC's position with regard to separation criteria? If train A is located twenty feet from an open hatchway on the lower elevation and train B is located ten feet from the same opening on the next elevation, would this be considered adequate separation?

RESPONSE

If a wall or floor/ceiling assembly contains major unprotected openings such as hatchways and stairways, then plant locations on either side of such a barrier must be considered as part of a single fire area. Refer to the staff position on Fire Areas in Generic Letter 83-33.

As to the example provided, if train A was separated by a cumulative horizontal distance of 20 feet from train B, with no intervening combustible materials or fire hazards, and both elevations were provided with fire detection and suppression, the area would be in compliance with Section III.G.2.b.

3.6 Intervening Combustibles

3.6.1 Negligible Quantities of Intervening Combustibles

QUESTION

Twenty feet of separation with absolutely no intervening combustibles is a rare case in most nuclear plants. What is the most acceptable method of addressing intervening combustibles? How are various utilities addressing this subject, and what would be sufficient justification to support an exemption request?

RESPONSE

If more than negligible quantities of combustible materials (such as isolated cable runs) exist between redundant shutdown divisions, an exemption request should be filed. ["Negligible quantity" is an admittedly judgmental criterion, and this judgment should be made by a qualified fire protection engineer and documented for later NRC audit.] Justifications for such exemptions have been based on the following factors:

Enclosure to GL 85-01, Re: Fire Protection Policy

b. Material having a structural base of noncombustible material, as defined in a., above, with a surfacing not over 1/8-inch thick that has a flame spread rating not higher than 50 when measured using ASTM E-84 Test "Surface Burning Characteristics of Building Materials."

In Generic Letter 83-33, we state:

"Staff Position: Section III.G.2.b requires the "separation ... with no intervening combustibles ..." To meet this requirement, plastic jackets and insulation of grouped electrical cables, including those which are coated, should be considered as intervening combustibles."

For fire protection, "no intervening combustibles" means that there is no significant quantities of in-situ materials which will ignite and burn located between redundant shutdown systems. The amount of such combustibles that has significance is a judgmental decision. As with other issues, if the licensees fire protection engineer is concerned that the quantity of combustibles between shutdown divisions may not be considered insignificant by an independent reviewer, an exemption could be requested, or the staff consulted.

Transient materials are not considered as an intervening combustible; however, they must be considered as part of the overall fire hazard within an area.

Cables that are in covered cable tray should also be considered as intervening combustibles.

Coated cables with a fire retardant material are also considered as intervening combustibles.

3.6.3 Unexposed Combustibles

QUESTION

Are unexposed combustibles, such as oil in sumps, closed cans, or sealed drums, or electrical cable in conduits, considered as "intervening combustibles"?

RESPONSE

Only oil in closed containers which are in accordance with NFPA 30 or electrical cables in conduits are not considered as intervening combustibles. In situ oil in sumps is considered to be an intervening combustible.

3. Fire Damage

Appendix R to 10 CFR Part 50 utilizes the term "free of fire damage." In promulgating Appendix R, the Commission has provided methods acceptable for assuring that necessary structures, systems and components are free of fire damage (see Section III.G.2a, b and c), that is, the structure, system or component under consideration is capable of performing its intended function during and after the postulated fire, as needed. Licensees seeking exemptions from Section III.G.2 must show that the alternative proposed provides reasonable assurance that this criterion is met. (Note also that Section III.G.2 applies only to equipment needed for hot shutdown. Therefore, an exemption from III.G.2 for cold shutdown equipment is not needed.)

4. Fire Area Boundaries

The term "fire area" as used in Appendix R means an area sufficiently bounded to withstand the hazards associated with the area and, as necessary, to protect important equipment within the area from a fire outside the area. In order to meet the regulation, fire area boundaries need not be completely sealed floor-to-ceiling, wall-to-wall boundaries. However, all unsealed openings should be identified and evaluated. Where fire area boundaries were not approved under the BTP process, or where such boundaries are not wall-to-wall, floor-to-ceiling boundaries with all penetrations sealed to the fire rating required of the boundaries, licensees must perform an evaluation to assess the adequacy of fire boundaries in their plants to determine if the boundaries will withstand the hazards associated with the area and protect important equipment within the area from a fire outside the area. This analysis must be performed by at least a fire protection engineer and, if required, a systems engineer. Although not required, licensees may submit their evaluations for staff review and concurrence. In any event, these analyses must be retained by the licensees for subsequent NRC audits.

5. Automatic Detection and Suppression

Sections III.G.2.b and III.G.2.c of Appendix R state that "In addition, fire detectors and an automatic fire suppression system shall be installed in the fire area..." Other provisions of Appendix R also use the phrase "fire detectors and an automatic fire suppression system in the fire area..." (see e.g., Section III.G.2.e).

In order to comply with these provisions, suppression and detection sufficient to protect against the hazards of the area must be installed. In this regard, detection and suppression providing less than full area coverage may be adequate to comply with the regulation. Where full area suppression and detection is not installed, licensees must perform an evaluation to assess the adequacy of partial suppression and detection to protect against the hazards in the area. The evaluation must be performed

3.8 Design Bases

3.8.1 Fire Protection Features NFPA Conformance

QUESTION

Should the fire protection features required by Section III.G conform to the NFPA Codes?

RESPONSE

Yes. For example, Section III G.2 requires an automatic suppression system. Our guidelines would recommend that the system be in accordance with an NFPA Code. If deviations are made from the Code, they should be identified in the FSAR or FHA.

3.8.2 Design Basis Fire

QUESTION

Why isn't the industry allowed to design to protect against a design basis fire?

RESPONSE

Neither the industry nor the Staff has been able to develop criteria for establishing design basis fire conditions because the in-situ and potential transient combustibles vary widely in different areas of the plant.

3.8.3 Redundant Trains/Alternate Shutdown

QUESTION

Confusion exists as to what will be classified as an alternate shutdown system and thus what systems might be required to be protected by suppression and detection under Section III.G.3.b. For example, while we are relying upon the turbine-building condensate system for a reactor building fire and the RHR system for a turbine building fire, would one system be considered the alternative to the other. If so, would suppression and detection be required for either or both systems under III.G.3.b? An explanation of alternative shutdown needs to be advanced for all licensees.

1. The reactor is tripped in the control room.

2. Offsite power is lost as well as automatic starting of the onsite a.c. generators and the automatic function of valves and pumps whose control circuits could be affected by a control room fire.

The analysis should demonstrate that capability exists to manually achieve safe shutdown conditions from outside the control room by restoring a.c. power to designated pumps, assuring that valve lineups are correct, and assuming that any malfunctions of valves that permit the loss of reactor coolant can be corrected before unrestoreable conditions occur.

Note that the only manual action usually credited in the control room by this analysis is the reactor trip. Any additional control room actions deemed necessary would have to be justified under the exemption process.

After the fire, the operators could return to the control room when the following conditions have been met:

1. The fire has been extinguished and so verified by appropriate fire protection personnel.

2. The control room has been deemed habitable by appropriate fire protection personnel and the shift supervisor.

3. Damage has been assessed and, if necessary, corrective action has been taken to assure necessary safety, control and information systems are functional (some operators may assist with these tasks) and the shift supervisor has authorized return of plant control to the control room.

4. Turnover procedures which assure an orderly transfer of control from the alternate shutdown panel to the control room has been completed.

After returning to the control room, the operators can take any actions compatible with the condition of the control room. Controls in any area (cabinet) where the fire occurred would not be available. Smoke and fire suppressant damage in other areas (cabinets) must also be assessed and corrective action taken before controls in such cabinets are deemed functional. Controls in undamaged area (cabinets) could be operated as required. Minor modifications inside the control room may be performed to reach cold shutdown.

2. Severity Categories

   A. Severity I. Violations of fire protection requirements established to protect or enable operation of safe hot shutdown equipment concurrent with an actual fire which damages that equipment such that safe hot shutdown could not be achieved or maintained using the equipment dedicated for the purpose.

   B. Severity II. Violations of fire protection requirements established to protect or enable operation of safe cold shutdown equipment concurrent with an actual fire which damages that equipment such that safe cold shutdown could not have been achieved and maintained using the equipment dedicated for this purpose in accordance with the applicable requirements.

   C. Severity III. Violations of fire protection requirements established to protect or enable operation of safe shutdown equipment such that a fire in the area could damage that equipment to the extent that safe hot or cold shutdown could not have been achieved and maintained using the equipment dedicated for this purpose in accordance with applicable requirements. Failure to have a written evaluation available for an area where compliance with Appendix R is not readily demonstrated will be taken as *prima facie* evidence that the area does not comply with NRC requirements and may result in enforcement action at the severity level.

   D. Severity IV. Violations of one or more fire protection requirements that do not result in a Severity Level I, II or III violation and which have more than minor safety or environmental significance.

   E. Severity V. Violations of one or more fire protection requirements that have minor safety or environmental significance.

5. ALTERNATIVE AND DEDICATED SHUTDOWN CAPABILITY

5.1 Safe and Alternative Shutdown

5.1.1 Previously Accepted Alternative Shutdown Capability

QUESTION

As part of the Appendix A review process, some plants had committed to an alternative shutdown system in the form of a remote shutdown panel or remote shutdown system. Footnote 2 to Appendix R describes alternative shutdown capability as being associated with "Rerouting, relocating, or modifying of existing systems." To the extent that an existing remote shutdown system previously reviewed and approved under Appendix A to BTP 9.5-1 does not require modifications, rerouting, or relocating of existing systems, are the requirements of Section III.L of Appendix R backfit?

RESPONSE

Yes. Existing remote shutdown capabilities previously reviewed and approved under Appendix A to BTP APCSB 9.5-1 do not categorically comply with Section III.G.3 of Appendix R. Licensees were requested to re-analyze their plants to determine compliance with Section III.G. If the licensee chooses to use the option of III.G.3 for provision of safe shutdown capability for certain areas, the criteria of Section III.L. are applicable to that capability for that area. See also the response to 5.1.3.

5.1.2 Pre-Existing Alternative Shutdown Capability

QUESTION

Some licensees defined safe shutdown capability for purposes of analysis to Section III.G criteria as being composed of both the normal safe shutdown capability and the pre-existing redundant or remote safe shutdown capability which was previously installed as part of the Appendix A process. This definition often took the form of two "safe shutdown trains" comprising (1) one of the two normal safe shutdown trains, and (2) a second safe shutdown train capability which was being provided by the pre-existing remote shutdown capability. This definitional process, which was undertaken by a number of licensees, makes a significant difference in the implementation of Appendix R. Under such a definition, does Section III.L criteria apply when the Commission did not call out Section III.L as a backfit?

Enclosure to GL 85-01, Re: Fire Protection Policy

RESPONSE

Safe shutdown capabilities including alternative shutdown capabilities are all designed for some maximum level of fire-damage (system unavailabilities, spurious actuations). Since the extent of the fire can not be predicted, it seems prudent to have the post-fire shutdown procedures guide the operator from full system availability to the minimum shutdown capability. As for repair procedure, similar conditions exist. A repair procedure can be written based on the maximum level of damage that is expected. This procedure would then provide shutdown capability without accurately predicting likely fire damage.

5.2.2 Post Fire Operating Procedures

QUESTION

Does the NRC have any requirements regarding whether post-fire operating procedures should be based upon fire areas, systems, or by symptom-based?

RESPONSE

The NRC does not have requirements, nor do we propose any requirements regarding whether post-fire operating procedures should be based upon fire areas, systems or be symptom-based. We suggest that the post-fire shutdown capabilities designs be reviewed with the plant operation staff and procedures written with their input. See also responses to 5.2.1 and 5.2.3.

5.2.3 Alternative Shutdown Capability

QUESTION

Is it acceptable to develop post-fire operating procedures only for those areas where alternative shutdown is required? (For other areas standard, emergency operating procedures would be utilized in the presence of potential fire damage to a single train.)

5.3.2 "Hot Short" Duration

QUESTION

If one mode of fire damage involves a "hot short" how long does that condition exist as a result of fire damage prior to terminating in a ground or open circuit and stopping the spurious actuation?

RESPONSE

We would postulate that a "hot short" condition exists until action has been taken to isolate the given circuit from the fire area, or other actions as appropriate have been taken to negate the effects of the spurious actuation. We do not postulate that the fire would eventually clear the "hot short."

5.3.3 Hot Shutdown Duration

QUESTION

Since hot shutdown cannot be maintained indefinitely, hot shutdown equipment needs to be protected for only a limited period of time. How long must a plant remain in that condition in order to meet the requirement for achieving hot shutdown with a single train of equipment?

RESPONSE

Section III.G.1 requires that the one train of systems needed to achieve and maintain hot shutdown be free of fire damage. Thus, the systems needed are to be completely protected from the fire regardless of time. If the intent of the question concerns how long these systems must operate, these systems must be capable of operating until the systems needed to achieve and maintain cold shutdown are available.

5.3.4 Cooldown Equipment

QUESTION

Certain equipment is necessary only in the cooldown phase when the plant is neither in hot nor cold shutdown condition as defined by technical specifications. Is this equipment considered hot or cold shutdown in nature?

RESPONSE

These statements are meant to indicate that the alternative shutdown capability should be powered from an onsite power system independent (both electrically and physically) from the area under consideration. Further, if the normal emergency onsite power supplies (diesel generators) are not available because of fire damage, then a separate and independent onsite power system shall be provided. As an example, some plants are utilizing a dedicated onsite diesel generator or gas turbine to power instrumentation and control panels which are a part of the alternative shutdown capability.

5.3.7 Torus Level Indication

QUESTION

For BWRs, I&E Information Notice 84-09 suggests that licensees need to have torus level indication post-fire. If an analysis shows that a level does not change significantly during any operational modes or worse case conditions, is level indication still required? Is an analysis in file adequate or is an exemption request required?

RESPONSE

It continues to be our position that torus (suppression pool) level indication is the preferred post-fire monitoring instrumentation in order to confirm the availability of the torus (suppression pool) as a heat sink. We recognize that existing analyses indicate that suppression pool level is not significantly changed during emergency shutdown conditions. However, we believe the operator should be able to confirm that spurious operations or other unanticipated occurrences have not affected the torus function. An analysis of torus level change by itself is not considered an acceptable basis.

5.3.8 Short Circuit Coordination Studies

QUESTION

Should circuit coordination studies consider high impendance faults?

b. The safe shutdown capability should not be adversely affected by a fire in any plant area which results in the loss of all automatic function (signals, logic) from the circuits located in the area in conjunction with one worst case spurious actuation or signal resulting from the fire; and

c. The safe shutdown capability should not be adversely affected by a fire in any plant area which results in simultaneous spurious actuation of all valves in high-low pressure interface lines.

5.3.11 Alternative/Dedicated Shutdown vs. Remote Shutdown Systems

QUESTION

What is the difference between the alternate/dedicated shutdown systems required for fire protection and the remote shutdown systems recommended under Chapter 7 of the SRP?

RESPONSE

The remote shutdown systems recommended under Chapter 7 of the SRP are needed to meet GDC 19. These remote shutdown systems need to be redundant and physically independent of the control room in order to meet GDC 19. For GDC 19, damage to the control room is not considered. Alternate shutdown systems for Appendix R need not be redundant but must be both physically and electrically independent of the control room.

In May 1976, the NRC asked licensees to compare operating reactors with BTP APCSB 9.5-1, and in September 1976, those licensees were informed that the guidelines in Appendix A would be used to analyze the consequences of a fire in each plant area. In September 1976 the licensees, were also requested to provide a fire hazards analysis that divided the plant into distinct fire areas and show that redundant systems required to achieve and maintain cold shutdown are adequately protected against damage by a fire. Early in 1977 each licensee responded with a Fire Protection Program Evaluation which included a Fire Hazard Analysis. These evaluations and analyses identified aspects of licensees' fire protection programs that did not conform to the NRC guidelines. Thereafter, the staff initiated discussions with all licensees aimed at achieving implementation of fire protection guidelines by October 1980. The staff held many meetings with licensees, conducted extensive correspondence with them, and visited every operating reactor. As a result, many fire protection items were resolved, and agreements were included in Fire Protection Safety Evaluation Reports issued by the NRC. Several fire protection issues remained unresolved with a number of licensees.

By early 1980, most operating plants had implemented most of the guidelines in Appendix A. However, as the Commission noted in its Order of May 23, 1980, the fire protection program has had some significant problems with implementation. Despite the staff's efforts, several licensees had expressed continuing disagreement with, and refused to adopt recommendations relating to several generic issues, including the requirements for fire brigade size and training, water supplies for fire suppression systems, alternate and dedicated shutdown capability, emergency lighting, qualifications of seals used to enclose places where cables penetrated fire barriers, and the prevention of reactor coolant pump lubrication system fires. To establish a definitive resolution of these contested subjects in a manner consistent with the general guidelines in Appendix A to the BTP and to assure timely compliance by licensees, the Commission issued a proposed fire protection rule and its Appendix R, which was described as setting out minimum fire protection requirements for the unresolved issues (45 Fed. Reg. 36082 May 29, 1980). The fire protection features addressed included protection of safe shutdown capability, emergency lighting, fire barriers, associated circuits, reactor coolant pump lubrication system, and alternate shutdown systems. The Commission stated that it expected all modifications (except for alternate and dedicated shutdown capability) to be implemented by November 1, 1980. 4/

3/ 11 NRC 707, 718 (1980)

4/ Id. at 719

points would be safely collected and drained to the sump. The sump should be shown capable of safely containing all of the anticipated oil leakage. The analysis should verify that there are no electric sources of ignition.

2. OVERVIEW

Section 50.48 Fire Protection of 10 CFR Part 50 requires that each operating nuclear power plant have a fire protection plan that satisfies General Design Criterion 3 of Appendix A to 10 CFR 50. It specifies what should be contained in such a plan and lists the basic fire protection guidelines for this plan. It requires that the Fire Protection Safety Evaluation Report which has been issued for each operating plant state how these guidelines were applied to each facility.

Section 50.48 also requires that all plants with operating licenses prior to January 1, 1979 satisfy the requirements of Section III.G, III.J and III.O, and other Sections of Appendix R where approval of similar features had not been obtained prior to the effective date of Appendix R. By a separate action, the Commission approved the staff's requirement that all plants to receive their operating license after January 1, 1979 also satisfy the requirements of Sections III.G, III.J and III.O and that a fire protection license condition be established. Deviations from Appendix R requirements for pre-1979 plants are processed under the exemption process. Deviation from other guidelines are identified and evaluated in the Safety Evaluation Report.

A standard fire protection license condition has been developed and will be included in each new operating license. Present operating licenses will be amended to include the standard license condition.

The Regions initiated inspections of operating plants and identified several significant items of non-compliance. The Nuclear Utility Fire Protection Group requested interpretations of certain Appendix R requirements and provided a list of questions that they thought should be discussed with the industry. The NRC held workshops in each Region to assist the industry in understanding the NRC's requirements and to improve the Staff's understanding of the industry's concerns.

This document presents the NRC's response to the questions posed by the industry and supplemented with additional questions identified at the workshops as being of interest to the industry or the staff. These responses may be used as guidance for design, review and inspection activities. The questons have been reformatted according to their applicability to Sections of Appendix R, BTP CMEB 9.5-1, licensing policy or inspection policy.

Our guidelines on the seismic design of hydrogen lines is delineated in BTP CMEB 9.5-1 C.5.d(5):

(5) Hydrogen lines in safety-related areas should be either designed to seismic Class I requirements, or sleeved such that the outer pipe is directly vented to the outside, or should be equipped with excess flow valves so that in case of a line break, the hydrogen concentration in the affected areas will not exceed 2%.

All PWR's have a hydrogen line going to the Volume Control Tank (Make-up Tank) that needs to be protected.

To identify plant specific situations in which seismic events could initiate a fire in a specific plant area, the fire protection engineer and systems engineer performing the fire hazards analysis should be concerned with in-situ combustible materials which can be released in a manner such that they could contact in-situ ignition sources by a seismic event. An example of this would be the rupture of the RCP lube oil line directly above the hot reactor coolant piping. The fire protection engineer should also be concerned with seismic induced ignition sources, electrical or mechanical, which could contact nearby in-situ combustible materials.

7.2 Random Fire and Seismic Events

QUESTION

Is a random fire to be postulated concurrent with a seismic event?

RESPONSE

Our position, as stated in Section C.1.6 of BTP CMEB 9.5-1, is "Worst case fire need not be postulated to be simultaneous with nonfire-related failures in safety systems, plant accidents, or the most severe natural phenomena."

Where plant systems are designed to prevent the release of combustible materials caused by a seismic event, such as a dike around a fuel oil tank transformer, or seismic supports for hydrogen lines, then no fire need to be arbitrarily assumed to take place in the fire hazards analysis.

Because it is impossible to completely preclude the occurrence of a seismically induced fire, Section C.6.c(4) of CMEB 9.5-1 states:

"Provisions should be made to supply water at least to standpipes and hose connections for manual firefighting in areas containing equipment required for safe plant shutdown in the event of a safe shutdown earthquake. The piping system serving such hose stations should be analyzed for SSE loading and should be provided with supports to ensure system pressure integrity. The piping

to the area or systems being considered. For this condition, it is assumed that no manual or automatic fire suppression action has been started and the fire has reached its peak burning rate and involves all combustibles present.

Within each area special attention should be given to limiting the amount of combustible material and to providing effective barriers and fire resistive coatings to reduce the spreading of a fire in these areas. A design basis fire should be assumed and provisions should be made to limit the consequence of such a fire by providing fire barriers with suitable separation between redundant systems and components which are provided to carry out required safety functions. This separation is enhanced if the plant is divided into suitable fire areas since redundant safety equipment can then be placed in separate fire areas.

Particular design attention should be given to the use of separate isolated fire areas for redundant cables to avoid loss of redundant safety-related cables.

Provisions should also be made to limit the consequences of a fire by suitable design of the ventilation systems so that the spread of the products of combustion to other areas of the plant is prevented. Means should be provided to ventilate or isolate the area as required. The power supply and controls for the area ventilation system should be from outside the area, and the power and control cables should not pass through the area.

The fire detection systems should be designed using detectors of the right types at locations suitable to detect the particular type of fire expected in each area.

In the design, consideration should be given to provide personnel access to and escape routes from each fire area. The emergency plans for all plants should lay out access and escape routes to cover the event of a fire in critical areas of the plant."

In addition, definitions of the basic fire area components were given:

"<u>Fire Area</u> - that portion of a building or plant that is separated from other areas by boundary <u>fire barriers</u> (walls, floors or roofs) with any openings or penetrations protected with seals or closures having a fire resistance rating equal to that of the barrier.

<u>Fire Barrier</u> - those components of construction (walls, floors and roofs) that are rated by approving laboratories in hours for resistance to fire to prevent the spread of fire.

8. LICENSING POLICY

8.1 Fire Hazard Analysis/Fire Protection Plan Updating

QUESTION

What constitutes the fire protection plan required by 50.48(a)? Should licensees have programs to maintain the fire hazards analysis and the fire protection plan current or updated periodically? How often should the plan be updated? Must revisions be provided to the NRC?

RESPONSE

The basic elements required in the fire protection plan are described in 10 CFR 50.48(a). The fire protection program that implements that plan should include the details of the fire hazards analysis. The plan and program may be separate or combined documents and must be kept current with the fire hazards analysis updated prior to making modifications. We would expect that for most plants licensed after January 1, 1979, the fire protection plan and program would be part of the FSAR and therefore, would be updated and submitted to the NRC in conformance with the requirements of 10 CFR 50.71(e). For plants whose fire protection plans and programs are not part of the FSAR, we would expect that they would be updated prior to making modifications and kept at the site in an auditable form for NRC inspection.

8.2 Fire Protection License Condition

QUESTION

What is the significance of the fire protection license condition?

RESPONSE

For those plants licensed prior to January 1, 1979 (Appendix R plants), the license condition is the legally enforceable requirement for the fire protection features other than those required by III.G, III.J. and III.O that were accepted by the NRC staff as satisfying the provisions of Appendix A to Branch Technical Position BTP APCSB 9.5-1.

For those plants licensed after January 1, 1979, the license condition is the legally enforceable requirement for all fire protection features at the facility.

Appendix R is only enforceable on Post 1979 plants through the license condition. 10 CFR 50.48 makes Appendix R applicable only to plants licensed prior to January 1, 1979. Refer to 10 CFR 50.48(e).

e) Describe all the extinguishing and detection capabilities within each fire area. Discuss all means for containing and inhibiting the progress of a fire, e.g., the use of fire stops, coatings, curbs, walls, etc. Describe the extinguishing equipment outside an area which has access to the area.

NOTE: If large fire areas are divided into fire zones for the purpose of fire protection, the above information should be provided for each zone.

2. Where redundant safety related equipment or cabling is located in a given fire area, describe the design features which prevent the loss of both redundant trains in a common fire, e.g., the separation provided by distance, physical barriers, and electrical isolation. Where control, power or instrument cables of redundant systems used for bringing the reactor to safe, cold shutdown are located in the same cable trays, either provide a bounding analysis demonstrating that the worst consequences as a result of a fire in the cable trays are acceptable or show that redundant systems required to achieve and maintain a cold shutdown are adequately protected against damage by the fire."

The guidelines for the fire rating of fire area boundaries and their penetrations were given in Appendix A to BTP 9.5-1.

"APPLICATION DOCKETED BUT CONSTRUCTION PERMIT NOT RECEIVED AS OF 7/1/76	PLANTS UNDER CONSTRUCTION AND OPERATING PLANTS
(j) Floors, walls and ceilings enclosing separate fire areas should have minimum fire rating of three hours. Penetrations in these fire barriers, including conduits and piping, should be sealed or closed to provide a fire resistance rating at least equal to that of the fire barrier itself. Door openings should be protected with equivalent rated doors, frames and hardware that have been tested and approved by a nationally recognized laboratory. Such doors should be normally closed and locked or alarmed with alarm and annunciation in the control room. Penetrations for ventilation system should be protected by a standard "fire door damper" where required. (Refer to NFPA 80, "Fire Doors and Windows")	(j) SAME. The fire hazard in each area should be evaluated to determine barrier requirements. If barrier fire resistance cannot be made adequate, fire detection and suppression should be provided, such as: (i) water curtain in case of fire, (ii) flame retardant coatings, (iii) additional fire barriers."

8.4 Future Changes

QUESTION

Will future changes (no matter how minor) to approved configurations be required to be reviewed by the Staff in an exemption request? At what point may the process of 10 CFR 50.59 be invoked?

RESPONSE

If a future modification involves a change to a license condition or technical specification, a license amendment request must be submitted. When a modification not involving a technical specification or license condition is planned, the evaluation made in conformance with 10 CFR 50.59 to determine whether an unreviewed safety question is involved must include an assessment of the modification's impact on the existing fire hazards analysis for the area. This part of the evaluation must be performed by the person responsible for the fire safety program for the plant. The assessment must include the effect on combustible loading and distribution and the consideration of whether circuits or components, including associated circuits, for a train of equipment needed for safe shutdown are being affected or a new element introduced in the area. If this evaluation concludes that there is no significant impact, this conclusion and its basis must be documented as part of the 50.59 evaluation and be available for future inspection and reference. If the evaluation finds that there is an impact that could result in the area either not being in conformance with Appendix R, or some other aspect of the approved fire protection program, or being outside the basis for an exemption that was granted for the area involved, the licensee must either make modifications to achieve conformance or justify and request exemption (or, for the post 1979 plants, approval) from the NRC. See also responses to Questions 8.1 and 8.2.

8.5 Schedular and Blanket Exemptions

QUESTION

If an exemption is warranted and at the same time the provisions of the rule indicate that the appropriate schedular deadlines have passed, should a scheduler exemption be filed at the same time as the technical exemption request?

If as part of the exemption request the utility is proposing to make modifications to achieve a reasonable level of conformance with Appendix R, and if the associated "clock" has run out for that type of modification, should the technical exemption request and the description of the modification be filed with a schedular exemption?

3.1.3 Exterior Walls

QUESTION

Must exterior walls to buildings and their penetrations be qualified as rated barriers?

RESPONSE

Exterior walls and their penetrations should be qualified as rated barriers when (1) they are required to separate a shutdown-related division(s) inside the plant from its redundant (alternate) counterpart outside the plant in the immediate vicinity of the exterior wall, (2) they separate safety related areas from non-safety related areas that present a significant fire threat to the safety related areas, or (3) they are designated as a fire barrier in the FSAR or FHA.

Usually exterior walls are designated as a fire area boundary; therefore, they are evaluated by the guidelines of Appendix A. A FHA should be performed to determine the rating of exterior walls, if required by the above criteria.

3.1.4 Exterior Yards

QUESTION

How should a utility define the boundaries of fire areas comprising exterior yards?

RESPONSE

An exterior yard area without fire barriers should be considered as one fire area. The area may consist of several fire zones. The boundaries of the fire zones should be determined by a FHA.

The protection for redundant/alternate shutdown systems within a yard area would be determined on the bases of the largest credible fire that is likely to occur and the resulting damage. The boundaries of such damage would have to be justified with a fire hazards analysis. The analysis should consider the degree of spatial separation between divisions; the presence of in-situ and transient combustibles, including vehicular traffic; grading; available fire protection; sources of ignition; and the vulnerability and criticality of the shutdown related systems.

8.8 Smallest Opening in a Fire Barrier

QUESTION

What is the smallest opening allowed in a fire area barrier for which an exemption request is not needed?

RESPONSE

Unsealed openings in the configuration for which approval was obtained by an approved laboratory or the NRC staff would be acceptable.

Our position on openings is given in Section 5.a(3) of BTP CMEB 9.5-1:

"(3) Openings through fire barriers for pipe, conduit, and cable trays which separate fire areas should be sealed or closed to provide a fire resistance rating at least equal to that required of the barrier itself. Openings inside conduit larger than 4 inches in diameter should be sealed at the fire barrier penetration. Openings inside conduit 4 inches or less in diameter should be sealed at the fire barrier unless the conduit extends at least 5 feet on each side of the fire barrier and is sealed either at both ends or at the fire barrier with non-combustible material to prevent the passage of smoke and hot gases. Fire barrier penetrations that must maintain environmental isolation or pressure differentials should be qualified by test to maintain the barrier integrity under such conditions."

The unsealed opening(s) allowed in a fire area boundary or a barrier which separates redundant shutdown divisions should not permit flame, radiant energy, smoke and hot gases to pass through the barrier and cause damage to redundant shutdown divisions on the other side. The licensee should assess the adequacy of existing protection and should determine the minimum size based on a fire hazards analysis and conservative fire protection engineering judgment. If the significance of openings in fire barriers is marginal, a formal exemption request could be submitted or the staff consulted. The basis for the lack of significance should be available for review by NRC Inspectors.

Our acceptance of unprotected openings in fire barriers would depend upon the quantity and nature of combustible materials on either side of the barrier; the location of the opening(s) in relation to the ceiling (for openings in walls); the location, vulnerability and importance of shutdown systems on either side of the barrier; and compensating fire protection.

Section III.G recognizes that the need for alternate or dedicated shutdown capability may have to be considered on the basis of a fire area, a room or a fire zone. The alternative or dedicated capability should be independent of the fire area where it is possible to do so (See Supplementary Information for the final rule Section III.G). When fire areas are not designated or where it is not possible to have the alternative or dedicated capability independent of the fire area, careful consideration must be given to the selection and location of the alternative or dedicated shutdown capability to assure that the performance requirement set forth in Section III.G.1 is met. Where alternate or dedicated shutdown is provided for a room or zone, the capability must be physically and electrically independent of that room or zone. The vulnerability of the equipment and personnel required at the location of the alternative or dedicated shutdown capability to the environments produced at that location as a result of the fire or fire suppressant's must be evaluated. These environments may be due to the hot layer, smoke, drifting suppressants, common ventilation systems, common drain systems or flooding. In addition, other interactions between the locations may be possible in unique configurations.

If alternate shutdown is provided on the basis of rooms or zones, the provision of fire detection and fixed suppression is only required in the room or zone under consideration. Compliance with Section III.G.2 cannot be based on rooms or zones.

3.1.6 Documentation

QUESTION

In Generic Letter 83-33 at pg. 2, the NRC Staff referred to the guidance in Appendix A to BTP 9.5-1 to establish the rating of the barrier. What level of documentation must be provided to verify that the fire area meets the requirements of Appendix R?

RESPONSE

The documentation required to verify the rating of a fire barrier should include the design description of the barrier and the test reports that verify its fire rating. Reference can be made to UL listed designs.

- The guidelines identified in the footnotes to 50.48(a)
- Guidelines documents issued after January 1, 1979.
- Commitments made to meet the requirements of Appendix R or specific sections such as III.G, III.J and III.O.
- BTP CMEB 9.5-1, which includes the requirements of Appendix R and the previous guidance documents incorporated into the Branch Technical Position.

The license for each plant licensed after January 1, 1979 contains a license condition which identifies by reference the approved fire protection program for that plant.

8.12 Cold Shutdown Equipment Availability

QUESTION

A. Can a licensee achieve compliance with III.G.1(b) by demonstrating that one train of cold shutdown equipment will remain free of fire damage?

B. In demonstrating that one train of cold shutdown equipment will remain free of fire damage, is a licensee limited to the three alternatives in III.G.2?

RESPONSE

A. Yes.

B. No.

8.13 Guidance Documents

QUESTION

Please list all NRR guidance documents and position papers issued since Appendix R was promulgated.

RESPONSE

Fire Protection Guidance Issued Since January 1, 1975:

IE Information Notices

No. 83-41: Actuation of fire suppression systems causing inoperability of safety related equipment.

No. 83-69: Improperly installed fire dampers at nuclear power plants.

During the Appendix A review, licensees began to propose fire barriers to enclose cable trays, conduit, fuel lines, coolant lines, etc. Industry did not have standard rating tests for such components or for electrical, piping or bus duct penetrations. The NRC issued a staff position giving acceptance criteria for electrical penetration tests. These criteria require an analysis of any temperature on the unexposed side of the barrier in excess of 325°F. In the past, manufacturers designed their own qualification tests. Nuclear Insurers, and the Institute of Electrical and Electronic Engineers have issued tests for some of these components. These tests usually exposed the component to the ASTM E-119 time temperature curve, but all had different acceptance criteria. Conduit and cable tray enclosure materials accepted by the NRC as 1 hour barrier prior to Appendix R (e.g. some Kaowool and 3M materials) and already installed by the licensee need not be replaced even though they may not have met the 325°F criteria. However, new material should meet the 325°F criterion. Justification of temperatures which exceed 325°F is required.

3.2.2 Deviations from Tested Configurations

QUESTION

Due to obstructions and supports, it is often impossible to achieve exact duplication of the specific tested configuration of the one-hour fire barriers which are to be placed around either conduits or cable trays. For each specific instance where exact replication of a previously tested configuration is not and cannot be achieved, is an exemption necessary in order to avoid a citation for a violation?

RESPONSE

No. Where exact replication of a tested configuration cannot be achieved, the field installation should meet all of the following criteria:

1. The continuity of the fire barrier material is maintained.

2. The thickness of the barrier is maintained.

3. The nature of the support assembly is unchanged from the tested configuration.

7. Letter dated 10/31/80 - Enclosing new 10 CFR 50.48 regarding fire protection schedules for operating nuclear power plants.

8. Letter dated 11/24/80 - Enclosing a copy of revised 10 CFR 50.48 and new App. R to 10 CFR 50, and a summary of open items from the SER for the BTP APCSB 9.5-1 review.

9. Letter dated 2/20/81 - Generic Letter 81-12 identifying information needed for NRC review of modifications for alternative shutdown capability.

10. Letter dated 4/7/82 - Provided clarification to Generic Letter 81-12 and guidance on information needed for NRC review of exemption requests.

11. Letter dated 10/6/82 - Generic Letter 82-21; provided criteria for annual, biennial, and triennial audits required by Technical Specifications.

*12. Letter dated 10/19/83 - Generic Letter 83-33; NRC Positions on Certain Requirements of Appendix R to 10 CFR 50.

Staff Generic Positions

1. Letter, Denton to Bernsen, dated 4/20/82 - Control room fires.

*2. SECY 83-269, dated July 5, 1983 - Attachments B and C.

3. Memo, Eisenhut to Olshinski, dated 12/20/83 - Physical independence of electrical systems.

4. Memo, Eisenhut to Jordan, dated 10/24/83 - Bullet resistant fire doors.

*Staff positions regarding the need for certain exemptions delineated in this guidance document have been revised per the "Interpretations of Appendix R".

8.14 Deviation From Guidance Documents

QUESTION

If a utility determines that a deviation from a guidance document exists, does an exemption request need to be filed? If so, what is the legal basis for this requirement?

RESPONSE

No.

RESPONSE

The NRC does not define the structural steel supporting fire barriers. This steel is identified by the licensee. Our position regarding the need to protect the structural steel, which forms a part of or supports fire barriers, is consistent with sound fire protection engineering principles as delineated in both NFPA codes and standards, and The Fire Protection Handbook.

3.3.2 Previously Accepted Structural Steel

QUESTION

Is it necessary to protect structural steel in existing fire barriers where those barriers were approved in an Appendix A SER?

RESPONSE

No.

3.3.3 Seismic Supports

QUESTION

Does structural steel whose sole purpose is to carry dynamic loads from a seismic event require protection in accordance with Section III.G.2a of Appendix R?

RESPONSE

No, unless the failure of any structural steel member due to a fire could result in significant degradation of the fire barrier. Then it must be protected.

3.3.4 Cable Tray Support Protection

QUESTION

Should cable tray supports be protected if there is a sprinkler system in the fire area? Under what conditions may cable tray supports be unprotected? Do unprotected supports require an exemption?

8.18 Coordination Study Update

QUESTION

Circuit modifications are an ongoing process. How recent must a coordination study be in order to be valid in protecting circuits associated by common power source?

RESPONSE

We would expect that as circuit modifications are made, the design package would address the electrical protection required and the effects of this protection on the coordination of the protection for the power distribution system. This type of consideration should be included in the evaluation required by 10 CFR 50.59 Changes, Tests and Experiments. The design package and modification evaluation could not be complete without consideration of the coordination study. Therefore, we would expect that the coordination studies would be current with the last circuit modification made.

8.19 Exemption Request Threshold

QUESTION

(a) What is the threshold for exemption requests? (b) Is it necessary to file a request for each and every possible deviation from Appendix R?

RESPONSE

Typical examples are discussed in the response to Questions 8.21.1 through 8.21.6.

(a) The licensee must develop its criteria for an exemption request threshold.

(b) No.

8.19.1 Penetration Designs Not Laboratory Approved

QUESTION

Where penetration designs have been reviewed and approved by NRC but have not been classified by an approval laboratory, will it be necessary to submit an exemption request?

RESPONSE

No.

3.4.2 NRC Consultation

QUESTION

Section 4.1.2 of NFPA-STD-13 allows for "partial installations" or partial coverage. The standard states that "the authority having jurisdiction shall be consulted in each case." With the NRC as authority in this instance, must consulation occur only through the exemption process?

RESPONSE

No. The staff is always available to consult with utility representatives and provide guidance as to the acceptability of a particular fire protection configuration in individual plant areas.

3.4.3 Sprinkler Location

QUESTION

How does a suppression system designer know whether the term "throughout the area" means that sprinkler heads must be above or below cable trays when, in his judgment, the hazard of concern is a floor based fire?

RESPONSE

Section C.6.c(3) of BTP CMEB 9.5-1 states:

"(3) Fixed water extinguishing systems should conform to requirements of appropriate standards such as NFPA-13, "Standard for the Installation of Sprinkler Systems," and NFPA-15, "Standard for Water Spray Fixed Systems"."

This question pertains to those sprinkler systems covered by NFPA-13. Chapter 4 of NFPA-13 provides guidance as to the location of sprinkler heads in relation to common obstructions. In general, to achieve complete area-wide coverage, sprinklers should be located at the ceiling, with additional sprinklers provided below significant obstructions such as wide HVAC ducts and "shielded" or solid bottom stacked cable trays. To the extent that an existing or proposed sprinkler system design deviates from this concept, the design would have to be justified by a fire hazards analysis.

8.21 NRC Approval for BTP CMEB 9.5-1 Deviations

QUESTION

Do future deviations from BTP CMEB 9.5-1 guidelines require approval by the NRC? Do such deviations constitute a violation of license conditions?

RESPONSE

Compliance with guidelines in the BTP is only required to the extent that they were icorporated in the approved Fire Protection Program as identified in the license condition.

When the new license condition is in place (See Response 8.2), future deviations may be made in accordance with the procedure stated therein. With present nonuniform license conditions, such deviations may or may not require a license amendment. In the absence of a license amendment, a violation may exist.

3.4.6 Previously Approved Suppression Systems

QUESTION

Must suppression systems approved and installed under BTP APCSB 9.5-1, Appendix A be extended or altered to meet the total area requirements of Section III.G (as interpreted by the Staff) or does this "requirement" only apply to new installations?

RESPONSE

Suppression systems installed in connection with Appendix A may or may not have to be extended as a result of III.G. The licensee must analyze each area where suppression is required by III.G, and where only partial suppression has been provided, determine if the coverage is adequate for the fire hazard in the area. The licensee may consult with the staff during this review. In any event, the Appendix R analysis showing that the suppression provided is adequate must be retained and available for NRC audit.

3.5 Separation of Redundant Circuits

3.5.1 Twenty-Foot Separation Criteria

QUESTION

Assuming that a licensee is utilizing the 20-foot separation for circuit protection, could an exemption request be granted for a portion of the circuit that did not maintain the 20-foot minimum separation if that portion was protected by one-hour barrier until 20-foot was achieved? This barrier would not be firewall-to-firewall, and the circuit protection would not be claimed under the one-hour barrier rule.

RESPONSE

With the erection of a partial qualified one-hour rated barrier for portions of the circuits with less than 20 ft. separation, if 20 feet of horizontal separation existed between the redundant unprotected portions of the circuits without intervening combustibles or fire hazards, and if the fire area was protected by automatic fire detection and suppression, compliance with Section III.G.2.b would be achieved.

These types of configuration have to be evaluated on a case-by-case basis.

RESPONSE

Yes, NTOLs will be subject to the Appendix R audit; the TI 2515/62 is being revised to reflect the appropriate requirements for NTOLs' and it is our intent to conduct such inspections prior to issuing the operating license.

10 CFR 50.48 requires each such plant to have a fire protection plan. Their operating license will contain a specific license condition to implement their approved fire protection program which must identify deviations from Appendix R. The fire protection inspections will be against the particular license conditions.

9.4 Future TI 2515/62 Revisions

QUESTION

Does the NRC plan to issue a new or revised version of Temporary Instruction 2515/62 for future Appendix R audits?

RESPONSE

Yes.

9.5 Documentation Supplied by Licensee

QUESTION

Temporary Instruction 2515/62 provided a list of documentation that the NRC needs to review as part of the audit process. In past audits, the NRC has requested additional information other than that contained on the list. Will a new list of documentation be developed?

RESPONSE

The documentation listing provided in TI-2515/62 does not restrict the inspection team from enhancing inspection efficiency by requesting a licensee to provide additional relevant documentation. A new listing of documentation for TI-2515/62 is not being developed.

9.6 Subsequent Inspections

QUESTION

To what extent will Appendix R issues be raised at future Regional I&E Fire Protection audits after a successful Appendix R audit? For example, if an area has already been reviewed and no noncompliance found, will it be subject to later review and reinterpretation by the Staff?

1. A relatively large horizontal spatial separation between redundant divisions; all cables qualified to IEEE-383.

2. The presence of an automatic fire suppression system over the intervening combustible (such as a cable tray fire suppression system);

3. The presence of fire stops to inhibit fire propagation in intervening cable trays;

4. The likely fire propagation direction of burning intervening combustibles in relation to the location of the vulnerable shutdown division;

5. The availability of compensating active and passive fire protection.

Any future changes in the cable configuration due to modifications could be handled under 50.59. See the provisions of the license condition in the response to question 8.2.

3.6.2 In-Situ Exposed Combustibles

QUESTION

Within Appendix R, Section III.G.2.b, the phrase "twenty feet with no intervening combustible or fire hazards" is utilized. What is the definition of "no intervening combustible"? Is the regulation focused predominantly on the absence of fixed combustibles?

RESPONSE

There is no specific definition of "no intervening combustible." The regulation is focused on the absence of in-situ exposed combustibles. Non combustible materials would not be considered as intervening combustibles.

In BTP CMEB 9.5-1, noncombustible material is defined as:

"Noncombustible Material

a. A material which in the form in which it is used and under the conditions anticipated, will not ignite, burn, support combustion, or release flammable vapors when subjected to fire or heat.

RESPONSE

To the extent that a licensee's submittal to NRR is comprehensive and sufficiently detailed, the basis for the OI&E Appendix R inspection will be the assumptions, shutdown paths and equipment selections approved by NRR. If the inspection results in new information that casts doubt upon the approved configuration, the Regional inspectors have the responsibility to resolve such doubts.

9.9 List of Shutdown Equipment

QUESTION

What lists of shutdown equipment will be used by the Regional inspectors, if the shutdown analysis has not been reviewed and approved by NRR?

RESPONSE

Regional Inspectors will use the lists of shutdown equipment the licensee has identified in his fire protection plan.

Generic Letter 81-12 and its clarification documents expect licensees to show how they will shutdown if a fire area is not provided with redundant train separation. Inherent within this expectation is the assumption that the licensee will identify the equipment to be used. It is because the licensees have not had fire hazard analyses at all for non-alternative shutdown fire areas that the inspectors to date have resorted to using the only lists available (the alternative shutdown equipment list used by NRR in their reviews).

It is unlikely there would not be a list of at least those systems to be used for alternate shutdown, since 10 CFR 50.48 requires NRR review and approval of the means of alternate shutdown.

3.7 Radiant Energy Shield

3.7.1 Fire Rating

QUESTION

Recently, the NRC Staff indicated that non-combustible radiant energy shields should be tested against ASTM-TD-E-119 based, apparently, on the requirements of BTP CMEB 9.5-1, Rev. 3, a document issued after Appendix R was promulgated. This new requirement would not appear to be required by Appendix R or BTP APCSB 9.5-1 Appendix A. Could the Staff clarify the requirements in this area?

ANSWER

During the Appendix A reviews, we observed that inside some containments, there were large concentrations of cables converging at electrical penetration areas. In some cases, where the penetrations were grouped by division, shields were placed between the divisions so that radiant energy from a fire involving the cables of one division would not degrade or ignite cables of the other divisions. These shields also directed the convective energy from the fire away from the surviving division. These shields were usually constructed of 1/2-inch marinite board in a metal frame. Appendix R, Section III.G.f refers to these shields as "a noncombustible radiant energy shield." The guidelines in BTP CMEB 9.5-1, Section C.7.a(1)b. indicate that these shields should have a fire rating of 1/2 hour. In our opinion any material with a 1/2 hour fire rating should be capable of performing the required function.

The guidelines of BTP CMEB 9.5-1 relating to a firerated radiant energy shield are being considered in our current reviews of NTOL plants. However, to the extent that an applicant can justify that a proposed radiant energy shield can achieve an equivalent level of safety, we have been accepting shields that have not been tested against the acceptance criteria of ASTM E-119.

In our Appendix R reviews, we have accepted non-fire-rated radiant energy shields that have been demonstrated by fire hazards analysis to provide an acceptable level of protection against the anticipated hazard of a localized fire within the containment. We have also accepted fire-rated metal-sheathed mineral insulated cables, as a radiant energy shield in specific configurations.

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

ENCLOSURE 7

SEP 2 1 1984

MEMORANDUM FOR: Harold R. Denton, Director, ONRR
Richard C. DeYoung, Director, IE

FROM: Richard H. Vollmer, Chairman
Fire Protection Policy Steering Committee

SUBJECT: FIRST MEETING OF FIRE PROTECTION POLICY STEERING COMMITTEE

Summary

At its first meeting, the Fire Protection Policy Steering Committee (SC) discussed the issues identified in the EDO memo of September 13, 1984. The SC made the following recommendations and assignments:

- the SC recommended that Appendix R implementation policy follow the "Interpretations" discussed at the Regional Workshops, as modified by details discussed below, rather than requiring prior staff review or exemptions for deviations from Generic Letter 83-33;

- the SC recommended that the issuance of the fire protection enforcement policy and enforcement actions be expedited;

- the SC recommended that QA for fire protection be clearly defined as that required by GDC-1; and

- the SC assigned to the Working Group tasks dealing with the adequacy of available guidance, comparison of fire protection requirements for ORs and NTOLs, the adequacy of current inspection practices, and identification of outstanding technical issues.

Introduction

As a result of the EDO memo of September 13, 1984, regarding a review of NRC fire protection policy and programs, the SC held its first meeting* on September 13 and 14. The objectives of this meeting were to:

*Attended by: T. Martin, J. Olshinski, L. Spessard, N. Grace, F. Rosa, S. Richardson, W. Shields, W. Little, T. Wambach, W. Olmstead, K. Cyr and R. Vollmer.

RESPONSE

If the system is being used to provide its design function, it generally is considered redundant. If the system is being used in lieu of the preferred system because the redundant components of the preferred system does not meet the separation criteria of Section III.G.2, the system is considered an alternative shutdown capability. Thus, for the example above, it appears that the condensate system is providing alternative shutdown capability in lieu of separating redundant components of the RHR System. Fire detection and a fixed fire suppression system would be required in the area where separation of redundant components of the RHR system is not provided. However, in the event of a turbine building fire, the RHR system would be used for safe shutdown and is not considered an alternative capability. However, one train of the RHR system must be separated from the turbine building.

3.8.4 Control Room Fire Considerations

QUESTION

What considerations should be taken into account in a control room fire? What is the damage that is considered? What actions can the operators take before evacuating the CR? When can the control room be considered safe after a fire for the operator to return?

RESPONSE

The control room fire area contains the controls and instrumental redundant shutdown systems in close proximity (i.e. usually separation is a few inches). Because it is possible to provide shutdown capability that is physically and electrically independent of the fire area, it is our opinion that alternative or dedicated shutdown capability and its associated circuits for the control room be independent of the cables system and components in the control room fire area.

The damage to the system in the control room for a fire that causes evacuation of the control room cannot be predicted. A bounding analysis should be made to assure that safe conditions can be maintained from outside the control room. This analysis is dependent to the specific design. The usual assumptions are:

The SC will take advantage of the Working Group chaired by Faust Rosa and make assignments to that group and to certain individuals. It was our belief that use of currently available resources, including the Working Group, would be sufficient to achieve our objectives and schedule.

Issues

1. Adequacy of current guidance to industry.

The SC felt that enough guidance has been generated but that it needed to be made consistent with our recommendations and cleared up technically in some areas. The mechanism for this would be a generic letter, superceding previous guidance, to be sent to all licensees as promised at the regional workshops. This generic letter would be sent to the Commission, as requested at the Commission's May 30 meeting, before being issued to industry. The SC agreed that it would be best to utilize guidance already available, to the extent possible, to minimize possible confusion both within industry and the NRC. It was also agreed that we should utilize either Generic Letter 83-33 or the "Interpretations" rather than a third option for dealing with the exemption issue. This is discussed below.

The Working Group was tasked with reviewing all current guidance and outstanding technical questions and to revise the Regional Workshop package to incorporate in one place a comprehensive set of guidance that is consistent with the SC's policy recommendations and the approved technical recommendations of the Working Group.

2. Interpretation of the Appendix R requirements vice staff guidance.

The basic issue is whether industry can deviate from the contents of Generic Letter 83-33 without prior staff review and approval. This issue is fully developed by the current Staff DPO. The SC had the benefit of ELD's views and Faust Rosa's recommendations resulting from his assignment by Mr. Denton to make an independent assessment of this DPO. The SC decided that it could only support the contents of 83-33 as guidance to industry, consistent with the "Interpretations" drafted for the Regional Workshops since ELD advised that Generic Letter 83-33 stated requirements which went beyond the terms of Appendix R itself.

However, the SC also felt that some of the clarifying language contained in the DPO should be utilized and that specific guidance should be supplied to clearly indicate the level of fire protection to be achieved and the documentation necessary to demonstrate it. In addition, the SC felt that, at an appropriate time prior to the Appendix R inspection, the licensee should be requested to provide information necessary to

4. EMERENCY LIGHTING

4.1 Illumination Levels

QUESTION

What is the requisite intensity level for emergency lighting for egress routes and areas where shutdown functions must be performed? What are the bases for determining these levels of lighting?

ANSWER

The level of illumination provided by emergency lighting in access routes to and in areas where shutdown functions must be performed is a level that is sufficient to enable an operator to reach that area and perform the shutdown functions. At the remote shutdown panels the illumination levels should be sufficient for control panel operators.

The bases for estimating these levels of lighting are the guidelines contained in Section 9.5.3 of the Standard Review Plan, which are based on industry standards (i.e., Illuminating Engineering Society Handbook).

Where a licensee has provided emergency lighting per Section III.J of Appendix R, we would expect that the licensee verify by field testing that this lighting is adequate to perform the intended tasks.

6. identification and resolution plan for outstanding technical issues.

The Working Group was requested to identify any outstanding technical issues that impede Appendix R compliance. These will be discussed with the SC and a plan and schedule for resolution will be developed. One issue that was discussed is the QA standard to be applied to fire protection compliance. The SC felt strongly that the requirements of GDC-1 apply to fire protection features, recognizing that the activities are underway to resolve the GDC-1/Appendix B and safety-related/important-to-safety issues. Without trying to muddy the water in these areas, the SC felt that, at minimum, licensees be made aware that the fire protection program falls under GDC-1. The Working Group should look into this also and review the QA commitments made by some licensees in their pre-Appendix R SERs.

The next meeting of the Fire Protection Steering Committee will be held on September 27 at 9:00 a.m. in P-202A.

R H Vollmer

Richard H. Vollmer, Chairman
Fire Protection Policy Steering Committee

cc: W. Dircks
V. Stello
R. Minogue
T. Murley
J. O'Reilly
J. Martin
E. Case
J. Taylor
D. Eisenhut
R. Bernero
G. Arlotto
F. Rosa (10)
SC Members

RESPONSE

The definitional process mentioned considers an alternative shutdown capability provided under the Appendix A review as a redundant shutdown capability under the Appendix R review. This definitional process is incorrect. For the purpose of analysis to Section III.G.2 criteria, the safe shutdown capability is defined as one of the two normal safe shut-down trains. If the criteria of Section III.G.2 are not met, an alternative shutdown capability is required. The alternative shutdown capability may utilize existing remote shutdown capabilities and must meet the criteria of Sections III.G.3 and III.L of Appendix R. See also the response to 5.1.3.

5.1.3 III.L Backfit

QUESTION

Why do the Staff interpretive memoranda regarding the criteria for satisfaction of Section III.L form the auditable basis for determining compliance to Appendix R when the Commission failed to backfit this section to all plants?

RESPONSE

Although 10 CFR 50.48(b) does not specifically include Section III.L. with Sections III.G., J., and O. of Appendix R as a requirement applicable to all power reactors licensed prior to January 1, 1979, the Appendix, read as a whole, and the Court of Appeals decision on the Appendix, Connecticut Light and Power, et al. v. NRC, 673 F2d. 525 (D.C. Cir., 1982), demonstrate that Section III.L. applies to the alternative safe shutdown option under Section III.G. if and where that option is chosen by the licensee.

5.2 Procedures

5.2.1 Shutdown and Repair Basis

QUESTION

With regard to the term "post-fire procedures" the Commission states that it is impossible to predict the course and extent of a fire. Given this, how does one write post-fire shutdown and repair procedures that are both symptomatic and usable to an operator?

UNITED STATES
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WASHINGTON, D. C. 20555

SEP 2 6 1984

MEMORANDUM FOR: Fire Protection Policy Steering Committee

FROM: Richard H. Vollmer, Chairman
Fire Protection Policy Steering Committee

SUBJECT: AGENDA FOR SECOND MEETING

The second meeting of the Fire Protection Policy Steering Committee will be held at 9:00 a.m. on September 27, 1984 in Room P-202 A. At this meeting we will be briefed on the Working Group activities and on current inspection content and schedules and have the benefit of the IE and NRR Office Director's views on our activities. Our tentative schedule is as follows:

9:00 a.m.	Executive Session
9:30	Harold Denton and Ed Case
10:00	Dick DeYoung and Jim Taylor
10:30	Faust Rosa will discuss Working Group activities
1:00 p.m.	Steve Richardson will discuss Appendix R inspections
2:00	Committee work

In addition to our consideration of issues raised by the above agenda items, we need to consider policy approaches to deal with technical and schedular exemptions which put implementation of Appendix R into the distant future. Any approaches you have to deal with this issue will be welcomed.

R.H. Vollmer
Richard H. Vollmer, Chairman
Fire Protection Policy Steering Committee

cc: W. Dircks
V. Stello
H. Denton
R. DeYoung
T. Murley
R. Minogue
J. O'Reilly
J. Keppler
J. Collins
J. Martin
G. Cunningham
E. Case
J. Taylor
D. Eisenhut
R. Bernero
G. Arlotto
F. Rosa

RESPONSE

Yes. The only requirement for post-fire operating procedures is for those areas where alternative shutdown is required. For other areas of the plant, shutdown would be achieved utilizing one of the two normal trains of shutdown system. Shutdown in degraded modes (one train unavailable) should be covered by present operator training and abnormal and emergency operating procedures. If the degraded modes of operation are not presently covered, we would suggest that the operation staff of the plant determine whether additional training or procedures are needed.

5.2.4 Post Fire Procedures Guidance Documents

QUESTION

Do any NRC Staff guidance documents exist relative to the extent, form, nature, etc. of Appendix R post-fire operating procedures?

RESPONSE

No. Other than the criteria of Section III.L, no specific post-fire shutdown procedure guidance has been developed. See also responses to 5.2.1, 5.2.2 and 5.2.3.

5.3 Safe Shutdown and Fire Damage

5.3.1 Circuit Failure Modes

QUESTION

What circuit failure modes must be considered in identifying circuits associated by spurious actuation?

RESPONSE

Sections III.G.2 and III.L.7 of Appendix R define the circuit failure modes as hot shorts, open circuits, and shorts to ground. If the concern is spurious actuation of equipment, actual circuit failure modes could be bypassed by assuming all possible failure states for the equipment (valves could fail either open or closed).

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WASHINGTON, D. C. 20555

SEP 27 1984

MEMORANDUM FOR: Harold R. Denton, Director, ONRR
Richard C. DeYoung, Director, OIE
Thomas E. Murley, Regional Administrator, R-I
James P. O'Reilly, Regional Administrator, R-II
James G. Keppler, Regional Administrator, R-III
John T. Collins, Regional Administrator, R-IV
John B. Martin, Regional Administrator, R-V

FROM: Richard H. Vollmer, Chairman
Fire Protection Policy Steering Committee

SUBJECT: NRC FIRE PROTECTION MEETING

The Fire Protection Policy Steering Committee requests that the fire protection engineers in your organization attend a meeting in Bethesda on October 2, 1984, to give their views on licensing and inspection fire protection issues. This is an information gathering session to help the Committee formulate its policy recommendations. A candid discussion of how these issues are viewed by the fire protection reviewers and inspectors, and recommendations they may have, would be welcome. A meeting agenda is enclosed.

RHVollmer

Richard H. Vollmer, Chairman
Fire Protection Policy Steering Committee

Enclosure: Agenda

cc: W. Dircks
V. Stello
R. Minogue
G. Cunningham
E. Case
J. Taylor, IE
D. Eisenhut
R. Bernero
G. Arlotto, RES
F. Rosa
N. Grace, IE
L. Spessard, R-III
W. Olmstead, ELD
J. Olshinski, R-II
T. Wambach

RESPONSE

As stated in Section III.G.1, one train of systems needed to achieve and maintain hot shutdown conditions must be free of fire damage. Systems necessary to achieve and maintain cold shutdown can be repaired within 72 hours. Thus, if this certain equipment necessary only in the cooldown phase, is used to achieve cold shutdown, it can be repaired within 72 hours. If the certain equipment is maintaining hot shutdown while repairs are being made, one train must be free of fire damage.

5.3.5 Pressurizer Heaters

QUESTION

Most PWRs do not require pressurizer heaters to maintain stable conditions. In fact, the Commission does not consider heaters to be important to safety and they are not required to meet Class IE requirements. Are they required for hot shutdown under Appendix R? If yes, then how does a plant meet the separation requirements of Section III.G.2.d, e. or f without major structural alterations to the pressurizer?

RESPONSE

One train of systems necessary to achieve and maintain hot shutdown conditions must be free of fire damage. PWR licensees have demonstrated the capability to achieve and maintain stable hot shutdown conditions without the use of pressurizer heaters by utilizing the charging pump and a water solid pressurizer for reactor coolant pressure control.

5.3.6 On-Site Power

QUESTION

Appendix R, Section III.L.4 states in part, "If such equipment and systems will not be capable of being powered by both on-site and off-site electrical power systems because of fire damage, an independent on-site power system shall be provided." Again, in Appendix R, Section III.L.5, the statement is made "If such equipment and systems used prior to 72 hours after the fire will not be capable of being powered by both onsite and offsite electrical power systems because of fire damage, an independent onsite power system shall be provided." An interpretation is needed of the meaning and the applicability of these two quotes relative to alternative shutdown capabilities.

NRC Fire Protection Meeting

Date: October 2, 1984 at

Location: P-422 Phillips Building

9:00 a.m.	Opening Remarks - R. Vollmer
9:10	Region I Remarks
9:40	Region II Remarks
10:10	Region III Remarks
10:40	Region IV Remarks
11:40	Brookhaven Remarks
1:00 p.m.	NRR Remarks
2:00	I&E Remarks
2:30 to 4:00	Group Discussion
4:00 to 5:00	Each group given opportunity to provide any additional comments.
5:00	Adjourn

RESPONSE

Yes. To meet the separation criteria of Section III.G.2 and II.G.7 of Appendix R, high impedance faults should be considered for all associated circuits located in the fire area of concern. Thus, simultaneous high impedance faults (below the trip point for the breaker on each individual ciruit) for all associated circuits located in the fire area should be considered in the evaluation of the safe shutdown capability. Clearing such faults on non-essential circuits may be accomplished by manual breaker trips governed by written procedures.

5.3.9 Diagnostic Instrumentation

QUESTION

What is diagnostic instrumentation?

RESPONSE

Diagnostic instrumentation is instrumentation, beyond that previously identified in Attachment 1 to I&E Information Notice 84-09, needed to assure proper actuation and functioning of safe shutdown equipment and support equipment (e.g., flow rate, pump discharge pressure). The diagnostic instrumentation needed depends on the design of the alternative shutdown capability. Diagnostic instrumentation, if needed, will be evaluated during the staff's review of the licensee's proposal for the alternative shutdown capability.

5.3.10 Design Basis Plant Transients

QUESTION

What plant transients should be considered in the design of the alternative or dedicated shutdown systems?

RESPONSE

Per the criteria of Section III.L of Appendix R. a loss of offsite power shall be assumed for a fire in any fire area concurrent with the following assumptions:

a. The safe shutdown capability should not be adversely affected by any one spurious actuation or signal resulting from a fire in any plant area; and

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WASHINGTON, D. C. 20555

October 3, 1984

MEMORANDUM FOR: Faust Rosa, Chairman
Fire Protection Working Group

FROM: Richard H. Vollmer, Chairman
Fire Protection Policy Steering Committee

SUBJECT: ASSIGNMENTS FOR FIRE PROTECTION WORKING GROUP

At the first meeting of the Fire Protection Policy Steering Committee, the Committee assigned action tasks to the Working Group. The purpose of this memo is to provide better focus on the Working Group assignments as a result of the second and third Committee meetings.

The Working Group's first priority should be the revision of the enclosure to the Generic Letter discussed at the fire protection workshops. This package should include answers to the questions posed by industry and revision of the "interpretations" section to utilize clarifying aspects of the DPO and to emphasize the level of fire protection expected by staff guidance, e.g., Generic Letter 83-33, and the documentation necessary to demonstrate equivalence to staff guidance. This package should not contain any new guidance and should be internally consistent and consistent with the Committee's proposed policy on the "interpretations." To the extent possible criteria acceptable to the staff should be clearly identified. This package should be available for Committee review by October 12 and whatever efforts are necessary to meet this date should be expended.

Bill Shields will prepare a draft of the Generic Letter in which he will lay out the elements for expediting Appendix R implementation. It will discuss the proposed policy on schedular exemptions, the proposed augmentation of plant inspections, and indicate generally what is expected of licensees and what enforcement action may occur. In addition, it was agreed that the 50.48 schedule expiration for each plant would be indicated.

The item of next priority is the development of an inspection plan which would accomplish the objectives of a "helpful" inspection. That is, in a one week, four-man inspection, we need to find out how the licensee is approaching his Appendix R implementation, the status of the plant with respect to compliance, and plans and schedules for complete implementation. This inspection should identify improper paths being taken by the licensee, and allow the staff to assess current fire protection safety and need for any plant specific enforcement action.

Finally, the Working Group should compare fire protection requirements for ORs and NTOLs and work on any outstanding technical issues. The outline

6. OIL COLLECTION SYSTEMS FOR REACTOR COOLANT PUMP

6.1 Lube Oil System Seismic Design

QUESTION

If the reactor coolant pump lube oil system and associated appurtenances are seismically designed, does the lube oil collection system also require seismic design? Is an exemption required?

RESPONSE

Where the RCP lube oil system is capable of withstanding the safe shutdown earthquake (SSE), the analysis should assume that only random oil leaks from the joints could occur during the lifetime of the plant. The oil collection system, therefore, should be designed to safely channel the quantity of oil from one pump to a vented closed container. Under this set of circumstances, the oil collection system would not have to be seismically designed.

An exemption is required for a non-seismically designed oil collection system. The basis for this exemption would be that random leaks are not assumed to occur simultaneously with the seismic event, since the lube oil system is designed to withstand the seismic event. However, the Rule, as written, does not make this allowance.

6.2 Container

QUESTION

It would appear that a literal reading of Section III.O regarding the oil collection system for the reactor coolant pump could be met by a combination of seismically designed splash shields and a sump with sufficient capacity to contain the entire lube oil system inventory. If the reactor coolant pump is seismically designed and the nearby piping hot surfaces are protected by seismically designed splash shields such that any spilled lube oil would contact only cold surfaces, does this design concept conform to the requirements of the rule?

RESPONSE

If the reactor coolant pump, including the oil system, is seismically designed and the nearby hot surfaces of piping are protected by seismically designed splash shields such that any spilled lube oil would contact only cold surfaces, and it could be demonstrated by engineering analysis that sump and splash shields would be capable of preventing a fire during normal and design basis accident conditions, the safety objective of Section III.O would be achieved. Such a design concept would have to be evaluated under the exemption process. The justification for the exemption should provide reasonable assurance that oil from all potential pressurized and unpressurized leakage

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WASHINGTON, D. C. 20555

OCT 4 1984

MEMORANDUM FOR: Fire Protection Policy Steering Committee

FROM: Richard H. Vollmer, Chairman
Fire Protection Policy Steering Committee

SUBJECT: AGENDA FOR THE FOURTH MEETING

The fourth meeting of the Fire Protection Policy Steering committee will be held at 9:00 a.m. on October 10, 1984 in the Region II offices in Atlanta. At this meeting we need to reach agreement on the enforcement policy guidance and develop our decision, reached at the second meeting and fortified in the third meeting, on a FY 85 inspection for most plant types. I request that John prepare a rewrite of the enforcement policy and Nelson a general plan for inspections for Committee consideration.

Other items we need to discuss are:

- Focus of inspections relative to fire protection and safe shutdown;
- Working group assignments;
- What should be included in technical specifications;
- Resolution of licensee/inspection team disputes; and
- Disputes of issues raised at third meeting.

I hope at this meeting we can agree on our overall approach to make our recommendations to Dircks coherent and consistent so that we can begin writing to have a draft report ready at our fifth meeting.

R. H. Vollmer /BM

Richard H. Vollmer, Chairman
Fire Protection Policy Steering Committee

cc: See next page

7. BRANCH TECHNICAL POSITION CMEB 9.5-1

7.1 Fire Protection and Seismic Events

QUESTION

For which situations other than the reactor coolant pump lube oil system are seismic events assumed to be initiators of a fire?

RESPONSE

The guidelines for the seismic design of fire protection systems which cover other general situations is delineated in BTP CMEB 9.5-1 C.1.C(3) and (4):

"(3) As a minimum, the fire suppression system should be capable of delivering water to manual hose stations located within hose reach of areas containing equipment required for safe plant shutdown following the safe shutdown earthquake (SSE). In areas of high seismic activity, the staff will consider on a case-by-case basis the need to design the fire detection and suppression systems to be functional following the SSE.

(4) The fire protection systems should retain their original design capability for (a) natural phenomena of less severity and greater frequency than the most severe natural phenomena (approximately once in 10 years) such as tornadoes, hurricanes, floods, ice storms, or small-intensity earthquakes that are characteristic of the geographic region, and (b) potential manmade site-related events such as oil barge collisions or aircraft crashes that have a reasonable probability of occurring at a specific plant site. The effects of lightning strikes should be included in the overall plant fire protection program."

We have considered California as being a high seismic activity area.

For those plants reviewed under Appendix A, our position is (A.4):

"Postulated fires or fire protection system failures need not be considered concurrent with other plant accidents or the most severe natural phenomena"

Our guidelines on the seismic design of fire protection systems installed in safety related areas are delineated in Regulatory Guide 1.29 "Seismic Design Classification", paragraph C.2. The failure of any system should not affect a system from performing its safety function.

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October 12, 1984

MEMORANDUM FOR: Fire Protection Policy Steering Committee

FROM: Richard H. Vollmer, Chairman
Fire Protection Policy Steering Committee

SUBJECT: AGENDA FOR THE FIFTH MEETING

The fifth meeting of the Fire Protection Policy Steering Committee will be held at 9:00 a.m. on October 17, 1984 in Room P-202A. At this meeting you should be prepared to discuss the generic letter and its attachments relative to the ability of these documents to satisfy Steering Committee decisions. We should also be prepared to settle on the enforcement policy guidance and the inspection module revision if these are available. Finally we will need to discuss format, content, and writing assignments for our October 26 report to the EDO.

Richard H. Vollmer, Director
Fire Protection Policy Steering Committee

cc: W. Dircks
V. Stello
H. Denton
R. DeYoung
R. Minogue
T. Murley, R-I
J. O'Reilly, R-II
J. Keppler, R-III
J. Collins, R-IV
J. Martin, R-V
G. Cunningham
E. Case
J. Taylor
D. Eisenhut
R. Bernero
G. Arlotto
F. Rosa

> and valves for the portion of hose standpipe system affected by this functional requirement should, as a minimum, satisfy ANSI B31.1, 'Power Piping.' The water supply for this condition may be obtained by manual operator actuation of valves in a connection to the hose standpipe header from a normal seismic Category I water system such as the essential service water system. The cross connection should be (a) capable of providing flow to at least two hose stations (approximately 75 gpm per hose station), and (b) designed to the same standards as the seismic Category I water system; it should not degrade the performance of the seismic Category I water system."

The post-seismic procedures should include a damage survey, and a determination of whether any fires were initiated as a result of the seismic event. See also the response to Question 7.1.

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October 12, 1984

MEMORANDUM FOR: Harold R. Denton, Director, NRR
Richard C. DeYoung, Director, OIE

FROM: Richard H. Vollmer, Chairman
Fire Protection Policy Steering Committee

SUBJECT: SECOND MEETING OF THE FIRE PROTECTION POLICY STEERING COMMITTEE HELD ON SEPTEMBER 27, 1984

Summary

At its second meeting, the Fire Protection Steering Committee (SC) considered policy approaches to deal with technical and schedular exemptions which threaten to put Appendix R implementation into the distant future, discussed SC activities with the IE and NRR Office Directors and Deputies, and was briefed on Working Group activities. As a result of this meeting, the SC made the following decisions:

- the SC decided to hold a meeting with all HQ and Regional fire protection engineers, as a body, to candidly discuss their views on fire protection issues, problems, and possible future actions;
- the SC decided that the most promising way to expedite Appendix R compliance is to initiate an aggressive inspection program which would steer and promote licensee compliance, assess the degree of fire safety, and exercise enforcement policy where appropriate; and
- the SC decided that no further schedular exemptions should be granted.

Discussion

1. Meeting with Office Directors and Deputies.

The SC met with Messrs. Denton, DeYoung, Case and Taylor to discuss their views on the fire protection problems, potential solutions, and SC activities. They viewed the problem as a lack of staff cohesiveness in a highly judgemental area exacerbated by industry reluctance to meet Appendix R requirements. They felt that a meeting between the HQ and Regional fire protection staff to air any problems, issues, and possible solutions, would benefit the SC's work. We agreed and set up such a meeting for October 2nd.

The NRC has drafted new language for this license condition which delineates the circumstances under which the fire protection plan may be revised. We are now including this language in all new licenses and are considering amending present licenses. The revised language is as follows.

"9.5 Fire Protection Program (Section 9.5, SER)

a. The licensee shall maintain in effect all provisions of the approved fire protection program as described in the Final Safety Analysis Report for the facility through Amendment __ and as approved in the SER through Supplement __, subject to provisions b & c below.

b. The licensee may make no change to the approved fire protection program which would decrease the level of fire protection in the plant without prior approval of the Commission. To make such a change the licensee must submit an application for license amendment pursuant to 10 CFR 50.90.

c. The licensee may make changes to features of the approved fire protection program which do not decrease the level of fire protection without prior Commission approval after such features have been installed as approved, provided such changes do not otherwise involve a change in a license condition or technical specification or result in an unreviewed safety question (see 10 CFR 50.59). However, the licensee shall maintain, in an auditable form, a current record of all such changes including an analysis of the effects of the change on the fire protection program and shall make such records available to NRC inspectors upon request. All changes to the approved program made without prior Commission approval shall be reported annually to the Director of the Office of Nuclear Reactor Regulation, together with supporting analyses."

8.3 III G, J and O Exemptions for Future Modifications

QUESTION

Is an exemption required from Appendix R Sections other than III.G, III.J and III.O for future modifications that do not comply with such sections?

RESPONSE

Yes. The exclusion of the applicability of Sections of Appendix R other than III.G., III.J., and III.O is limited to those features "accepted by the NRC staff as satisfying the provisions of Appendix A to Branch Technical Position BTP APCSB 9.5-1 reflected in staff fire protection safety evaluation reports issued prior to the effective date of the rule."

Reference: 10 CFR 50.48(b).

3. Technical issues.

Two technical issues were identified to be pursued by the Working Group: (1) the Dow Corning ascertion that penetration material was not being properly installed; and (2) whether our criteria on control room fires are appropriate and consistent.

4. Working Group discussion.

Faust Rosa discussed the Working Group activities, in particular, the outline of the Working Group report. All items of interest to the SC were covered in the outline of this report. However, it appeared that completion of the work as outlined was too ambitious for the time and resources provided. The SC indicated that priority attention be given to the completion of the generic letter package. The SC also indicated in its first meeting that the Working Group should compare fire protection requirements for ORs and NTOLs, review the adequacy of current inspection practices, and identify outstanding technical issues. These tasks should proceed except for the review of inspection practices which is superceded by the development of the program described in item 2 above.

Richard H. Vollmer, Chairman
Fire Protection Policy Steering Committee

cc: W. Dircks
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J. Keppler, R-III
J. Collins, R-IV
J. Martin, R-V
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R. Bernero
G. Arlotto
F. Rosa

When filing a schedular exemption under §50.12, it is not always clear from what specific paragraphs of §50.48 an exemption should be sought. Is it acceptable to request a blanket exemption from the schedular provisions of 10 CFR §50.48 without a specification by paragraph?

If an exemption request is submitted to meet newly published interpretations of Appendix R, when does the licensee need to be in compliance? Is the schedule presented in Appendix R still the guideline or must a new schedule be developed under a different criteria?

RESPONSE

We do not intend to issue any further extensions of the 50.48(c) schedules. When a licensee determines that a 50.48(c) schedule cannot be met, the appropriate NRC Region must be notified. This policy is further explained in the generic letter transmitting this package.

8.6 Trivial Deviations

QUESTION

What guidance can the NRC Staff give the industry regarding when a deviation from the literal interpretation of Appendix R is sufficiently trivial as to not require a specific exemption?

RESPONSE

The significance of a deviation must be judged as part of a fire hazards analysis. The conclusion of this analysis is always subject to review by the NRC inspector.

8.7 Revised Modifications

QUESTION

What is the process for altering configurations not yet implemented for plants with Appendix R SERs?

RESPONSE

If licensees propose changes to their NRC approved modifications, they must submit their new proposal and revised schedule for implementation for NRC approval.

This exchange must be justified as to (1) the reason for the change, (2) the basis for the revised schedule, and (3) the interim measures that will be provided to assure post fire shutdown capability until the final modifications are implemented. Whether or not enforcement action will be taken based upon continued noncompliance with Appendix R will be decided by the NRC Regional Administrator in consultation with NRC Headquarters.

UNITED STATES
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WASHINGTON, D. C. 20555

OCT 1 9 1984

MEMORANDUM FOR: Harold R. Denton, Director, ONRR
Richard C. DeYoung, Director, OIE

FROM: Richard H. Vollmer, Chairman
Fire Protection Policy Steering Committee

SUBJECT: THIRD MEETING OF THE FIRE PROTECTION POLICY STEERING COMMITTEE, HELD ON OCTOBER 2, 1984

Summary

At its third meeting the Fire Protection Policy Steering Committee (SC) met with the fire protection engineers from HQ and the Regional Offices to obtain their candid views on licensing and inspection fire protection issues. Comments were presented by fire protection engineers from NRR, IE, and Regions I, II, III, and IV. In addition, Jane Axelrad discussed the proposed fire protection enforcement policy with the SC. Highlights of these comments and ensuing discussions are as follows:

- The responsibility for fire protection was viewed as fragmented since CMEB, ASB, LQB, and QUAB in HQ are involved in addition to Regions. Commentors believed a central point of contact is needed in HQ;

- The fire protection guidelines and scope of Tech Specs were considered by some to be inconsistent and inadequate, and a list of "minimum requirements" for fire protection was requested;

- Regional inspectors indicated need for an enforcement policy and for policy on QA for fire protection;

- Need for control room electrical review policy was indicated; and

- Comments were voiced for and against use of the "interpretations."

This meeting was very helpful to the SC in better defining the issues and clarifying where action was most needed. However, the SC indicated it would not be able to resolve or even address all of the issues raised. Of particular note was the attitude expressed by the fire protection engineers of a desire to resolve the issues promptly and of a willingness to support the recommendations of the SC. A list of attendees is provided in the enclosure.

8.9 NFPA Code Deviation

QUESTION

Is an exemption/deviation required for deviations from NFPA Codes?

RESPONSE

Deviations from the codes should be identified and justified in the FSAR or FHA.

An exemption is not required for NFPA codes. NRC guidelines reference certain NFPA codes as guidelines to the systems acceptable to the staff, and therefore such codes may be accorded the same status as Regulatory Guides.

When the applicant/licensee states that its design "meets the NFPA codes" or, "meets the intent of the NFPA Codes" and does not identify any deviations from such codes, NRR and the Regions expect that the design conforms to the code and the design is subject to inspection against the NFPA codes.

8.10 "ASTM E-119" Design Basis Fire

QUESTION

Is an exemption/deviation required, if components are designed to withstand an "ASTM E-119" fire?

RESPONSE

Some cables are being developed for high temperature (e.g., 1700°F) applications. An exemption would be required if such cable is used in lieu of the alternatives of III.G.2 or III.G.3 in a pre-1979 plant. A deviation from the guidelines would be required for similar applications in a post 1979 plant.

8.11 Plants Licensed After January 1, 1979

QUESTION

What fire protection guidelines and requirements apply to the plants licensed after January 1, 1979?

RESPONSE

Post-1979 plants are subject to:

- GDC 3
- 10 CFR 50.48(a) and (e)

Region II also indicated that guidance for inspectors needed revising and expansion and that an inspection module was needed for NTOLs. It was also pointed out that the fire protection inspection must be done early in the inspection phase when the licensee has the opportunity to make changes. It was suggested that Regional inspectors accompany NRR reviewers in their site visits and that a general improvement in communication and understanding of SER commitments was needed. Finally, Region II voiced the view that the definition of fire areas in 83-33 must be retained, that guidance is needed for suppression systems and intervening combustibles, reiterated that the inspection module needs improvement by supplying minimum acceptance criteria, and stressed the need for an enforcement policy in this area.

Region III generally endorsed the comments of Region I and II. In addition, they pointed out the need for QA guidance in the area of fire protection. They stated that deficiencies in Tech Specs resulted from omission of fire damper surveillance, and inconsistancies of Code requirements. It was suggested that the present inspection modules be combined into one for all plants. Region III indicated that they felt the need to explain the rule requirements to industry; for example, 20 feet separation. They requested that RES be tasked to supply the technical basis. They felt that inspectors needed such information to guide them in making judgments and evaluations. The SC pointed out that the items in the rule were based on the best information at the time and that inspectors needed not feel obligated to explain rule obligations to licencees. If there are areas where the inspector feels safety is not well served by meeting rule provisions, such concerns should be elevated to management but that the rule, including its defense indepth provisions, seemed adequate. Finally, Region III indicated that the three things most needed were: (1) enforcement policy, (2) minimum requirements, and (3) consistent levels of inspection. To take care of (3), a training program would be needed. When asked, Region III cited the following as their three biggest frustrations: (1) the adequacy of licensee analyses, (2) the adequacy of regulatory requirements, and (3) the inconsistent reviews and inspection criteria.

Region IV has inspected Fort Calhoun, Fort St. Vrain and some NTOLs. They endorsed most of the comments of the previous Regions. In particular, they felt the need for acceptance criteria, enforcement policy, and-up-to-date Tech Specs.

Brookhaven National Laboratory (BNL) commented on problems with specific compliance vice meeting the "intent" of Appendix R. In particular, NTOLs allege that they meet the intent of Appendix R through a number of

No. 83-83: Use of portable radio transmitters inside nuclear power plants.

*No. 84-09: Lessons Learned From NRC Inspections of Fire Protection Safe Shutdown Systems (10 CFR 50, Appendix R)

Standard Review Plan

9.5.1, Rev. 1 Fire Protection System, dated 5/1/76

9.5.1, Rev. 2 Fire Protection Program, dated 03/78

9.5.1, Rev. 3 Fire Protection Program, July 1981.

Regulations

10 CFR Part 50: Proposed fire protection program for nuclear power plants operating prior to January 1, 1979, dated May 29, 1980. Federal Register Vol. 45, No. 105, 36082.

10 CFR Part 50: Fire protection program for operating nuclear power plants, dated November 19, 1980. Federal Register Vo. 45, No. 225, 76602.

10 CFR Part 50: Fire protection rule corrections, dated September 8, 1981. Federal Register Vo. 46, No. 173, 44734.

Generic Letters

Note: The following documents were obtained from the Palisades file Docket No. 50-255. Similar documents should be in the file for other operating facilities. The dates may vary slightly.

1. Letter dated 9/28/76 - Enclosing App. A to BTP APCSB 9.5-1 and supplementary guidance on information needed for fire protection program evaluation.

2. Letter dated 12/1/76 - Enclosing sample Technical Specifications and an errata sheet.

3. Letter dated 8/19/77 - Enclosing "Nuclear Plant Fire Protection Functional Responsibilities, Administrative Controls and Quality Assurance."

4. Letter dated 6/8/78 - Re: Manpower requirements for operating reactors.

5. Letter dated 9/7/79 - Re: Minimum fire brigade shift size.

6. Letter dated 9/14/79 - Enclosing staff positions - safe shutdown capability.

OCT 19 1984

meet Appendix R as well as deviations or exemptions, there would be fewer inspection and enforcement problems, fewer citations and a better overall fire protection image. They pointed out that, if the licensee's evaluation is kept in house and not docketed under oath, it could be inaccurate.

The NRR representatives stated that the practical effect of the interpretations would be to relax requirements because an additional burden is placed on reviewers and inspectors that changes to licensee fixes are needed. They also asked for the agency to characterize the priority of fire protection in plant safety.

2. Views of IE enforcement staff.

Jane Axelrad discussed the current policy and indicated that it has not yet been issued because of a lack of general policy on what constitutes compliance with the rule. She gave background on the enforcement policy and comments on efforts to apply policy consistently across the regions. The SC indicated that it would provide its revision of the enforcement policy guidance for EDO approval.

RHVollmer

Richard H. Vollmer, Chairman
Fire Protection Policy Steering Committee

Enclosure: As stated

cc: W. Dircks
V. Stello
R. Minogue
T. Murley, R-I
J. O'Reilly, R-II
J. Keppler, R-III
R. Martin, R-IV
J. Martin, R-V
G. Cunningham
E. Case
J. Taylor
D. Eisenhut
R. Bernero
G. Arlotto
F. Rosa
SC Committee

8.15 Staff Interpretation of Appendix R

QUESTION

How does the Staff initiate interpretations of Appendix R in a manner which ensures their technical adequacy and consistency with the rule's objectives (e.g., presentation to ACRS, issue for comment as in draft regulatory guides, etc.)?

RESPONSE

Staff positions are initiated when our experience shows that generic issues are identified that require clarification. These positions are reviewed for accuracy and consistency by the cognizant Division Directors. Usually, they are not issued for comment. However, Generic Letter 83-33 was commented on by the NUFPG since it was initiated, in part, at their request.

8.16 Dissemination of New Staff Positions

QUESTION

Will licensees be automatically sent a copy of new Staff position papers as they are developed?

RESPONSE

The Staff positions on generic subjects are considered for issuance in Generic Letters from ONRR and Information Notices or Bulletins from OI&E. Staff positions issued for specific questions on specific plants are not given generic promulgation because they normally involve plant specific deisgn considerations.

8.17 Equivalent Alternatives

QUESTION

How does a licensee demonstrate that alternative measures are equivalent to the measures of Section III.G.2 in order to obtain an exemption lacking a formal definition of the term "free of fire damage"?

RESPONSE

See Item #3 of "Interpretations of Appendix R."

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

October 19, 1984

MEMORANDUM FOR: Harold R. Denton, Director, NRR
Richard C. DeYoung, Director, OIE

FROM: Richard H. Vollmer, Chairman
Fire Protection Policy Steering Committee

SUBJECT: FOURTH MEETING OF THE FIRE PROTECTION POLICY STEERING COMMITTEE

Summary

At its fourth meeting, the Fire Protection Policy Steering Committee (SC) again considered the generic letter to licensees, the enforcement policy guidance and the scope, timing and resources for the expedited fire protection inspections. The SC focussed on some of the peripheral issues which would be important to the success of the expedited inspections and other SC initiatives. At this meeting, the SC decided that:

- Prior to the fire protection inspections, a workshop would be held with IE, NRR, and Regional participants in the inspections to assure understanding in the objectives, scope, and technical issues and to help provide consistency between inspection teams;
- A team would be established to promptly handle disputes between licensees and the inspection teams; and
- That a standard condition should be incorporated into all licenses, requiring maintenance of the fire protection commitments but allowing change under 50.59 which do not decrease the level of fire protection with annual reporting to the Commission of such changes.

Discussion

1. Expedited inspections.

The SC discussed the scope, timing, and resources for the expedited inspections. There was a discussion of whether this should be a review or an inspection. The SC felt that the concept was one of an inspection rather than a review and that the availability of enforcement was important to the process. The SC discussed resources and concluded that an adequate pool existed; however it was not clear if or how the expedited

8.19.2 Individual vs. Package Exemptions

QUESTION

How do we submit future modification exemption requests, etc.? Would NRC prefer them individually, or developed and submitted in packages for review and approval?

RESPONSE

Future exemptions should be submitted individually, if they are independent of each other.

8.19.3 Exemption Request Supporting Detail

QUESTION

When an exemption request is filed, what criteria are used to determine the level of detail needed to support the request?

RESPONSE

See Enclosure 2 of NRC's letter to all licensees dated April-May 1982.

8.19.4 50.12 vs. 50.48 Exemption Requests

QUESTION

With regard to exemption requests for future modifications, will they be submitted under 50.12 or 50.48?

RESPONSE

10 CFR 50.12.

8.20 Post January 1, 1979 Plants and Exemption Requests

QUESTION

Do plants licensed after January 1, 1979 which have committed to meet the requirements of Section III.G, III.J and III.O and are required to do so as a license condition, need to request exemptions for alternative configurations?

RESPONSE

No; however, deviations from the requirements of Section III.G, III.J and III.O should be identified and justified in the FSAR or FHA and the deviation would probably require a license amendment to change the license condition. See responses 8.1 and 8.2.

cc: W. Dircks
V. Stello
R. Minogue
T. Murley, R-I
J. OReilly, R-II
J. Keppler, R-III
R. Martin, R-IV
J. Martin, R-V
G. Cunningham
E. Case
J. Taylor
D. Eisenhut
R. Bernero
G. Arlotto
F. Rosa
SC Committee

9. INSPECTION POLICY

9.1 Safety Implications

QUESTION

Since the Commission states that fire damage cannot be defined and fire spread cannot be predicted, how does the Commission determine which Appendix R violations have "important safety implications?"

RESPONSE

III.G.2 provides alternatives to ensure that one of the redundant trains is free of fire damage. Fire spread within one area cannot be predicted, but damage is limited to ona fire area.

Determination of the Appendix R violations that have "important safety implications" are based on the equipment, components, and systems that are located in the same fire area that are needed for safe shutdown or can adversely affect safe shutdown, and are not protected by the features of III.G.2, III.G.3 or an approved alternative.

9.2 Uniform Enforcement

QUESTION

How does the Commission ensure that violations of the rule are uniformly treated between regions?

RESPONSE

Each Region evaluates violations in accordance with the NRC Enforcement Policy, 10 CFR 2, Appendix C. The Policy provides guidance for the determination of appropriate enforcement sanctions for violations. The Office of Inspection and Enforcement provides guidance for and monitors Regional implementation of the Policy to ensure a uniform application. In addition, the policy requires that all escalated enforcement actions be approved by the Director of the Office of Inspection and Enforcement.

9.3 NTOL Inspections

QUESTION

Will NTOLs be subject to an Appendix R audit now being performed on plants licensed to operate prior to January 1, 1979? Or, will the current review and analysis being performed by the Staff be satisfactory?

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

OCT 19 1984

MEMORANDUM FOR: Fire Protection Policy Steering Committee

FROM: Richard H. Vollmer, Chairman
Fire Protection Policy Steering Committee

SUBJECT: AGENDA FOR SIXTH MEETING

The sixth meeting of the Fire Protection Policy Steering Committee will be held at 9:00 a.m. on October 22, 1984 in the Region III offices. At this meeting you should be prepared to discuss and finalize versions of:

- Enforcement Policy
- Generic Letter and Interpretations
- Technical Issues Package
- Standard License Condition
- Inspection Module

In addition, we need to prepare our final report to the EDO. I will FAX an outline to you for your consideration today. In addition, there are a number of issues and suggestions still left hanging. For example: the central point of contact for fire protection issue; the status of NFPA codes; what to do about Tech Specs; and the format of the workshop in advance of the expedited inspections.

Richard H. Vollmer, Chairman
Fire Protection Policy Steering
Committee

cc: See next page.

RESPONSE

The Appendix R inspections are conducted on a sample basis. These inspections do not certify that all possible items of noncompliance with Appendix R have been identified. The inspection results do provided a basis for a determination of the adequacy of a licensee's Appendix R reanalysis, modification and preparation. When a noncompliance with Appendix R requirements is identified, a notice of violation will be issued to ensure adequate corrective action. In those cases in which the licensee believes that the staff has invoked a reinterpretation of adequacy in areas which had previously been reviewed, NRC's procedures for appeal would be applicable.

9.7 NRC List of Conforming Items

QUESTION

At the end of the audit, will the NRC provide a list of items that had been reviewed and found in conformance with Appendix R? To date, only areas of nonconformance have been specifically identified in exit interviews.

RESPONSE

Subsequent to an Appendix R inspection, the NRC will not provide a list of items reviewed and found to be in conformance with Appendix R.

We do list the areas inspected and where non-compliances were not found.

9.8 Inspection Re-review

QUESTION

Where assumptions are made and clearly stated within the analysis submitted to NRR for review, will such assumptions be subject to a second review by OI&E during the inspection process?

Where assumptions are made in conjunction with the analysis, should exemption requests be filed just to provide protection for the licensee?

If NRR accepts a licensee's selection of equipment and shutdown paths as being sufficient to meet the Appendix R shutdown criteria, will OI&E review and have the right to challenge the approved shutdown paths and approved equipment selection? Or will they only check the shutdown paths and equipment in question to see that they meet the Appendix R requirements, i.e., separation?

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

OCT 26 1984

MEMORANDUM FOR: Harold R. Denton, Director, ONRR
Richard C. DeYoung, Director, OIE

FROM: Richard H. Vollmer, Chairman
Fire Protection Policy Steering Committee

SUBJECT: FIFTH AND SIXTH MEETINGS OF THE FIRE PROTECTION POLICY STEERING COMMITTEE

At the fifth and sixth meetings of the Fire Protection Policy Steering Committee (SC), held in Bethesda on October 17 and the Region III offices on October 22, respectively, the final version of (1) enforcement policy guidance, (2) Generic Letter, (3) standard license condition, (4) temporary instruction for fire protection inspections, and (5) technical issues package of questions and answers were discussed, edited, and put into final form. No new initiatives were discussed but the impact and consistency of all initiatives developed by the SC were reviewed. The SC also assured that all issues included in the EDO memo of September 13 had been fully addressed and that all issues raised to the SC's attention by other parties had been fully considered.

Richard H. Vollmer, Chairman
Fire Protection Policy Steering Committee

cc: See next page.

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

ENCLOSURE 7

SEP 21 1984

MEMORANDUM FOR: Harold R. Denton, Director, ONRR
Richard C. DeYoung, Director, IE

FROM: Richard H. Vollmer, Chairman
Fire Protection Policy Steering Committee

SUBJECT: FIRST MEETING OF FIRE PROTECTION POLICY STEERING COMMITTEE

Summary

At its first meeting, the Fire Protection Policy Steering Committee (SC) discussed the issues identified in the EDO memo of September 13, 1984. The SC made the following recommendations and assignments:

- the SC recommended that Appendix R implementation policy follow the "Interpretations" discussed at the Regional Workshops, as modified by details discussed below, rather than requiring prior staff review or exemptions for deviations from Generic Letter 83-33;

- the SC recommended that the issuance of the fire protection enforcement policy and enforcement actions be expedited;

- the SC recommended that QA for fire protection be clearly defined as that required by GDC-1; and

- the SC assigned to the Working Group tasks dealing with the adequacy of available guidance, comparison of fire protection requirements for ORs and NTOLs, the adequacy of current inspection practices, and identification of outstanding technical issues.

Introduction

As a result of the EDO memo of September 13, 1984, regarding a review of NRC fire protection policy and programs, the SC held its first meeting* on September 13 and 14. The objectives of this meeting were to:

*Attended by: T. Martin, J. Olshinski, L. Spessard, N. Grace, F. Rosa, S. Richardson, W. Shields, W. Little, T. Wambach, W. Olmstead, K. Cyr and R. Vollmer.

- discuss the background leading to the formation of the SC;
- establish a general charter, objectives, schedule, and working arrangements; and
- discuss issues currently in hand and make decisions for resolution if appropriate.

The discussion of background included the events leading up to and including the fire protection regional workshops, the May 30, 1984 meeting with the Commission, the August 27 meeting with the EDO, and the events surrounding the DPO signed by three fire protection reviewers and two inspectors. As part of this background, ELD representatives discussed the Rule and the distinction between its legally enforceable requirements and staff guidance issued subsequent to the Rule. The background discussion also focussed on the issues identified in the EDO memo of September 13.

Charter and Schedule

The charter of the SC is the review of NRC fire protection policy and programs leading to policy recommendations which would expedite compliance with Appendix R at older plants and assure consistent levels of fire protection safety at all plants. To implement this Charter the SC agreed that current licensing, inspection, legal, and technical issues needed to be examined. The SC's objective would be to make specific recommendations to the EDO which could be carried out (1) within the existing framework of 50.48 and Appendix R, and (2) without making a disruption in the effort already underway to implement fire protection requirements. The SC would attempt to make recommendations that could be immediately effective. However, we recognize that there may be some instances in which further study was needed. In such cases, we agreed to recommend a specific assignment and end date for such study. Finally, the SC agreed that its work would be completed through issuance of its report to the EDO by October 26, 1984.

Working Arrangements

The SC discussed how it could use the available resources most effectively. It was decided that the SC would not need to meet at this time with HQ and Regional people since their views have been expressed extensively in written and transcribed material. The SC did not feel the need to meet with industry for the same reason. However, the SC does wish to meet with Vic Stello and hereby offers the opportunity of a meeting with the recipients of this memo.

The SC will take advantage of the Working Group chaired by Faust Rosa and make assignments to that group and to certain individuals. It was our belief that use of currently available resources, including the Working Group, would be sufficient to achieve our objectives and schedule.

Issues

1. Adequacy of current guidance to industry.

The SC felt that enough guidance has been generated but that it needed to be made consistent with our recommendations and cleared up technically in some areas. The mechanism for this would be a generic letter, superceding previous guidance, to be sent to all licensees as promised at the regional workshops. This generic letter would be sent to the Commission, as requested at the Commission's May 30 meeting, before being issued to industry. The SC agreed that it would be best to utilize guidance already available, to the extent possible, to minimize possible confusion both within industry and the NRC. It was also agreed that we should utilize either Generic Letter 83-33 or the "Interpretations" rather than a third option for dealing with the exemption issue. This is discussed below.

The Working Group was tasked with reviewing all current guidance and outstanding technical questions and to revise the Regional Workshop package to incorporate in one place a comprehensive set of guidance that is consistent with the SC's policy recommendations and the approved technical recommendations of the Working Group.

2. Interpretation of the Appendix R requirements vice staff guidance.

The basic issue is whether industry can deviate from the contents of Generic Letter 83-33 without prior staff review and approval. This issue is fully developed by the current Staff DPO. The SC had the benefit of ELD's views and Faust Rosa's recommendations resulting from his assignment by Mr. Denton to make an independent assessment of this DPO. The SC decided that it could only support the contents of 83-33 as guidance to industry, consistent with the "Interpretations" drafted for the Regional Workshops since ELD advised that Generic Letter 83-33 stated requirements which went beyond the terms of Appendix R itself.

However, the SC also felt that some of the clarifying language contained in the DPO should be utilized and that specific guidance should be supplied to clearly indicate the level of fire protection to be achieved and the documentation necessary to demonstrate it. In addition, the SC felt that, at an appropriate time prior to the Appendix R inspection, the licensee should be requested to provide information necessary to

support the inspection for HQ and regional review. In addition, the licensees should be encouraged to meet with the staff to discuss plans before extensive hardware modifications are initiated.

The Working Group was tasked to work on language to support this recommendation as part of its work on item 1.

3. Treatment of expected future technical and schedular exemptions into late 1980s and early 1990s.

This item was discussed extensively and tabled when no clear direction was apparent. The concern is that some licensees may unduly request staff reconsideration of technical findings and/or implementation schedules which would in effect defer compliance. This item will receive priority consideration at the next meeting and no assignments were made. However, the SC felt strongly that the fire protection enforcement policy and current enforcement packages consistent with that policy should be issued promptly to demonstrate NRC resolve in this area and that the backlog in NRR be processed expeditiously.

4. Comparison of Appendix R and current NTOL plants for fire protection safety.

Based on statements made at the May 30 Commission meeting and correspondence from the Regions, there may be differences in the licensing evaluation and inspection practices for Appendix R plants and NTOLs. The Working Group was tasked with investigating if such differences do exist and how the goal of consistent levels of fire protection safety at all plants might be achieved. In pursuing this, it was suggested that the Working Group meet with representatives from HQ and the Regions active in fire protection reviews and inspections. It was also acknowledged that the existing Temporary Instruction for Appendix R safe shutdown inspections must be revised to be consistent with the new interpretations. The Working Group must also verify that the guidance documents referenced in the Temporary Instruction for use by the inspectors have been officially sent to all licensees and applicants.

5. Adequacy of current inspection practices.

The SC was informed that consistent inspection practices and schedules have been set up for Appendix R plants but that the NTOLs have been handled on a somewhat ad hoc basis. Some regions have instituted a fairly extensive review of the capability to achieve a safe shutdown following a fire and others look more at SER-specific hardware items. Steve Richardson of IE was tasked with reporting to the SC, at the next meeting on current inspection content and schedules for all classes of plants, suggestions for modifying current practices, and, specifically, when the NTOL's should be inspected to assure effective compliance with NRC's fire protection requirements.

6. identification and resolution plan for outstanding technical issues.

The Working Group was requested to identify any outstanding technical issues that impede Appendix R compliance. These will be discussed with the SC and a plan and schedule for resolution will be developed. One issue that was discussed is the QA standard to be applied to fire protection compliance. The SC felt strongly that the requirements of GDC-1 apply to fire protection features, recognizing that the activities are underway to resolve the GDC-1/Appendix B and safety-related/important-to-safety issues. Without trying to muddy the water in these areas, the SC felt that, at minimum, licensees be made aware that the fire protection program falls under GDC-1. The Working Group should look into this also and review the QA commitments made by some licensees in their pre-Appendix R SERs.

The next meeting of the Fire Protection Steering Committee will be held on September 27 at 9:00 a.m. in P-202A.

RH Vollmer

Richard H. Vollmer, Chairman
Fire Protection Policy Steering Committee

cc: W. Dircks
V. Stello
R. Minogue
T. Murley
J. O'Reilly
J. Martin
E. Case
J. Taylor
D. Eisenhut
R. Bernero
G. Arlotto
F. Rosa (10)
SC Members

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

SEP 2 6 1984

MEMORANDUM FOR: Fire Protection Policy Steering Committee

FROM: Richard H. Vollmer, Chairman
Fire Protection Policy Steering Committee

SUBJECT: AGENDA FOR SECOND MEETING

The second meeting of the Fire Protection Policy Steering Committee will be held at 9:00 a.m. on September 27, 1984 in Room P-202 A. At this meeting we will be briefed on the Working Group activities and on current inspection content and schedules and have the benefit of the IE and NRR Office Director's views on our activities. Our tentative schedule is as follows:

9:00 a.m.	Executive Session
9:30	Harold Denton and Ed Case
10:00	Dick DeYoung and Jim Taylor
10:30	Faust Rosa will discuss Working Group activities
1:00 p.m.	Steve Richardson will discuss Appendix R inspections
2:00	Committee work

In addition to our consideration of issues raised by the above agenda items, we need to consider policy approaches to deal with technical and schedular exemptions which put implementation of Appendix R into the distant future. Any approaches you have to deal with this issue will be welcomed.

R.H. Vollmer

Richard H. Vollmer, Chairman
Fire Protection Policy Steering Committee

cc: W. Dircks
V. Stello
H. Denton
R. DeYoung
T. Murley
R. Minogue
J. O'Reilly
J. Keppler
J. Collins
J. Martin
G. Cunningham
E. Case
J. Taylor
D. Eisenhut
R. Bernero
G. Arlotto
F. Rosa

Enclosure to GL 85-01, Re: Fire Protection Policy

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

SEP 27 1984

MEMORANDUM FOR: Harold R. Denton, Director, ONRR
Richard C. DeYoung, Director, OIE
Thomas E. Murley, Regional Administrator, R-I
James P. O'Reilly, Regional Administrator, R-II
James G. Keppler, Regional Administrator, R-III
John T. Collins, Regional Administrator, R-IV
John B. Martin, Regional Administrator, R-V

FROM: Richard H. Vollmer, Chairman
Fire Protection Policy Steering Committee

SUBJECT: NRC FIRE PROTECTION MEETING

The Fire Protection Policy Steering Committee requests that the fire protection engineers in your organization attend a meeting in Bethesda on October 2, 1984, to give their views on licensing and inspection fire protection issues. This is an information gathering session to help the Committee formulate its policy recommendations. A candid discussion of how these issues are viewed by the fire protection reviewers and inspectors, and recommendations they may have, would be welcome. A meeting agenda is enclosed.

RHVollmer

Richard H. Vollmer, Chairman
Fire Protection Policy Steering Committee

Enclosure: Agenda

cc: W. Dircks
V. Stello
R. Minogue
G. Cunningham
E. Case
J. Taylor, IE
D. Eisenhut
R. Bernero
G. Arlotto, RES
F. Rosa
N. Grace, IE
L. Spessard, R-III
W. Olmstead, ELD
J. Olshinski, R-II
T. Wambach

NRC Fire Protection Meeting

Date: October 2, 1984 at

Location: P-422 Phillips Building

9:00 a.m.	Opening Remarks - R. Vollmer
9:10	Region I Remarks
9:40	Region II Remarks
10:10	Region III Remarks
10:40	Region IV Remarks
11:40	Brookhaven Remarks
1:00 p.m.	NRR Remarks
2:00	I&E Remarks
2:30 to 4:00	Group Discussion
4:00 to 5:00	Each group given opportunity to provide any additional comments.
5:00	Adjourn

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

October 3, 1984

MEMORANDUM FOR: Faust Rosa, Chairman
Fire Protection Working Group

FROM: Richard H. Vollmer, Chairman
Fire Protection Policy Steering Committee

SUBJECT: ASSIGNMENTS FOR FIRE PROTECTION WORKING GROUP

At the first meeting of the Fire Protection Policy Steering Committee, the Committee assigned action tasks to the Working Group. The purpose of this memo is to provide better focus on the Working Group assignments as a result of the second and third Committee meetings.

The Working Group's first priority should be the revision of the enclosure to the Generic Letter discussed at the fire protection workshops. This package should include answers to the questions posed by industry and revision of the "interpretations" section to utilize clarifying aspects of the DPO and to emphasize the level of fire protection expected by staff guidance, e.g., Generic Letter 83-33, and the documentation necessary to demonstrate equivalence to staff guidance. This package should not contain any new guidance and should be internally consistent and consistent with the Committee's proposed policy on the "interpretations." To the extent possible criteria acceptable to the staff should be clearly identified. This package should be available for Committee review by October 12 and whatever efforts are necessary to meet this date should be expended.

Bill Shields will prepare a draft of the Generic Letter in which he will lay out the elements for expediting Appendix R implementation. It will discuss the proposed policy on schedular exemptions, the proposed augmentation of plant inspections, and indicate generally what is expected of licensees and what enforcement action may occur. In addition, it was agreed that the 50.48 schedule expiration for each plant would be indicated.

The item of next priority is the development of an inspection plan which would accomplish the objectives of a "helpful" inspection. That is, in a one week, four-man inspection, we need to find out how the licensee is approaching his Appendix R implementation, the status of the plant with respect to compliance, and plans and schedules for complete implementation. This inspection should identify improper paths being taken by the licensee, and allow the staff to assess current fire protection safety and need for any plant specific enforcement action.

Finally, the Working Group should compare fire protection requirements for ORs and NTOLs and work on any outstanding technical issues. The outline

you provided of the Working Group's program review appears far too ambitious in view of current time and resource constraints. We should discuss this further at our October 10 meeting in Atlanta.

RHVollmer

Richard H. Vollmer, Chairman
Fire Protection Policy Steering Committee

cc: Steering Committee
W. Dircks
H. Denton
R. DeYoung
E. Case
J. Taylor
W. Shields

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

OCT 4 1984

MEMORANDUM FOR: Fire Protection Policy Steering Committee

FROM: Richard H. Vollmer, Chairman
Fire Protection Policy Steering Committee

SUBJECT: AGENDA FOR THE FOURTH MEETING

The fourth meeting of the Fire Protection Policy Steering committee will be held at 9:00 a.m. on October 10, 1984 in the Region II offices in Atlanta. At this meeting we need to reach agreement on the enforcement policy guidance and develop our decision, reached at the second meeting and fortified in the third meeting, on a FY 85 inspection for most plant types. I request that John prepare a rewrite of the enforcement policy and Nelson a general plan for inspections for Committee consideration.

Other items we need to discuss are:

- Focus of inspections relative to fire protection and safe shutdown;
- Working group assignments;
- What should be included in technical specifications;
- Resolution of licensee/inspection team disputes; and
- Disputes of issues raised at third meeting.

I hope at this meeting we can agree on our overall approach to make our recommendations to Dircks coherent and consistent so that we can begin writing to have a draft report ready at our fifth meeting.

R. H. Vollmer / [illegible]
Richard H. Vollmer, Chairman
Fire Protection Policy Steering Committee

cc: See next page

cc: W. Dircks
V. Stello
H. Denton
R. DeYoung
R. Minogue
T. Murley, R-I
J. O'Reilly, R-II
J. Keppler, R-III
J. Collins, R-IV
J. Martin, R-V
G. Cunningham
E. Case
J. Taylor
D. Eisenhut
R. Bernero
G. Arlotto
F. Rosa

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

October 12, 1984

MEMORANDUM FOR: Fire Protection Policy Steering Committee

FROM: Richard H. Vollmer, Chairman
Fire Protection Policy Steering Committee

SUBJECT: AGENDA FOR THE FIFTH MEETING

The fifth meeting of the Fire Protection Policy Steering Committee will be held at 9:00 a.m. on October 17, 1984 in Room P-202A. At this meeting you should be prepared to discuss the generic letter and its attachments relative to the ability of these documents to satisfy Steering Committee decisions. We should also be prepared to settle on the enforcement policy guidance and the inspection module revision if these are available. Finally we will need to discuss format, content, and writing assignments for our October 26 report to the EDO.

RHVollmer

Richard H. Vollmer, Director
Fire Protection Policy Steering Committee

cc: W. Dircks
V. Stello
H. Denton
R. DeYoung
R. Minogue
T. Murley, R-I
J. O'Reilly, R-II
J. Keppler, R-III
J. Collins, R-IV
J. Martin, R-V
G. Cunningham
E. Case
J. Taylor
D. Eisenhut
R. Bernero
G. Arlotto
F. Rosa

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

October 12, 1984

MEMORANDUM FOR: Harold R. Denton, Director, NRR
Richard C. DeYoung, Director, OIE

FROM: Richard H. Vollmer, Chairman
Fire Protection Policy Steering Committee

SUBJECT: SECOND MEETING OF THE FIRE PROTECTION POLICY STEERING COMMITTEE HELD ON SEPTEMBER 27, 1984

Summary

At its second meeting, the Fire Protection Steering Committee (SC) considered policy approaches to deal with technical and schedular exemptions which threaten to put Appendix R implementation into the distant future, discussed SC activities with the IE and NRR Office Directors and Deputies, and was briefed on Working Group activities. As a result of this meeting, the SC made the following decisions:

- the SC decided to hold a meeting with all HQ and Regional fire protection engineers, as a body, to candidly discuss their views on fire protection issues, problems, and possible future actions;
- the SC decided that the most promising way to expedite Appendix R compliance is to initiate an aggressive inspection program which would steer and promote licensee compliance, assess the degree of fire safety, and exercise enforcement policy where appropriate; and
- the SC decided that no further schedular exemptions should be granted.

Discussion

1. Meeting with Office Directors and Deputies.

The SC met with Messrs. Denton, DeYoung, Case and Taylor to discuss their views on the fire protection problems, potential solutions, and SC activities. They viewed the problem as a lack of staff cohesiveness in a highly judgemental area exacerbated by industry reluctance to meet Appendix R requirements. They felt that a meeting between the HQ and Regional fire protection staff to air any problems, issues, and possible solutions, would benefit the SC's work. We agreed and set up such a meeting for October 2nd.

It was also noted that, although industry was less than enthusiastic about fire protection, many problems have been solved and there have been significant improvements in fire protection of plants. To assist in defining fire protection problems and assessing safety significance, it was also suggested that we sort out fire protection and safe shutdown issues. The inclusion of fire protection features in Tech Specs was also discussed; namely, that they are not currently consistent, that augmenting of Tech Specs in this area has been proposed, and that such activities must be considered relative to general goals of simplifying Tech Specs. Finally, it was suggested that we need to better integrate the disciplines involved in this licensing and inspection area and identify a central point of responsibility.

2. Exemptions.

The SC discussed what, if anything, could be done to keep schedular and technical exemptions from dragging Appendix R implementation into the next decade. ELD stated that there was little we could do on technical exemptions since, if the utility has a valid reason for requesting one, then NRC must review it. On schedular exemptions, however, we can make a policy decision not to grant any more. Such a decision would have a legitimate basis since the Commission's Appendix R record viewed implementation in four or five years. Further, many 50.48 schedules have or are near running out. The SC decided that no further schedular exemptions should be granted.

The discussion turned to means of assuring that licensees recognized and could implement their responsibility for Appendix R implementation. The SC felt that the staff was taking on too much of the burden and that a well defined set of technical criteria, coupled with a program of inspection, and an enforcement policy would provide the best incentives. The SC felt that about five teams each consisting of a team leader and a fire protection, electrical, and systems engineer, should be set up. Each team, beginning in February 85, should go to one plant per month for a one-week inspection. These inspections would target plants of each type, each utility, and each A/E including NTOLs. Recognizing that plants in varying degrees of compliance would be inspected, the inspections should focus on safe shutdown. Where Appendix R implementation is still being engineered, the team should steer and promote licensee compliance in a technically supportable way. These inspections would also establish where each plant stands vis a vis Appendix R, and would use enforcement action where appropriate in a prompt fashion. It was suggested that there should be central and prompt resolution of any licensee/team disputes. This could be handled by a team consisting of a member of NRR, IE and the appropriate Regional management. The decision of this team would state the NRC position followed by a confirmatory letter or order as appropriate. Following such an inspection program, decisions on the long term fire protection inspections, both in extent and timing, would follow.

3. Technical issues.

Two technical issues were identified to be pursued by the Working Group: (1) the Dow Corning ascertion that penetration material was not being properly installed; and (2) whether our criteria on control room fires are appropriate and consistent.

4. Working Group discussion.

Faust Rosa discussed the Working Group activities, in particular, the outline of the Working Group report. All items of interest to the SC were covered in the outline of this report. However, it appeared that completion of the work as outlined was too ambitious for the time and resources provided. The SC indicated that priority attention be given to the completion of the generic letter package. The SC also indicated in its first meeting that the Working Group should compare fire protection requirements for ORs and NTOLs, review the adequacy of current inspection practices, and identify outstanding technical issues. These tasks should proceed except for the review of inspection practices which is superceded by the development of the program described in item 2 above.

Richard H. Vollmer, Chairman
Fire Protection Policy Steering
Committee

cc: W. Dircks
V. Stello
H. Denton
R. DeYoung
R. Minogue
T. Murley, R-I
J. O'Reilly, R-II
J. Keppler, R-III
J. Collins, R-IV
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R. Bernero
G. Arlotto
F. Rosa

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

OCT 1 9 1984

MEMORANDUM FOR: Harold R. Denton, Director, ONRR
Richard C. DeYoung, Director, OIE

FROM: Richard H. Vollmer, Chairman
Fire Protection Policy Steering Committee

SUBJECT: THIRD MEETING OF THE FIRE PROTECTION POLICY STEERING COMMITTEE, HELD ON OCTOBER 2, 1984

Summary

At its third meeting the Fire Protection Policy Steering Committee (SC) met with the fire protection engineers from HQ and the Regional Offices to obtain their candid views on licensing and inspection fire protection issues. Comments were presented by fire protection engineers from NRR, IE, and Regions I, II, III, and IV. In addition, Jane Axelrad discussed the proposed fire protection enforcement policy with the SC. Highlights of these comments and ensuing discussions are as follows:

- The responsibility for fire protection was viewed as fragmented since CMEB, ASB, LQB, and QUAB in HQ are involved in addition to Regions. Commentors believed a central point of contact is needed in HQ;

- The fire protection guidelines and scope of Tech Specs were considered by some to be inconsistent and inadequate, and a list of "minimum requirements" for fire protection was requested;

- Regional inspectors indicated need for an enforcement policy and for policy on QA for fire protection;

- Need for control room electrical review policy was indicated; and

- Comments were voiced for and against use of the "interpretations."

This meeting was very helpful to the SC in better defining the issues and clarifying where action was most needed. However, the SC indicated it would not be able to resolve or even address all of the issues raised. Of particular note was the attitude expressed by the fire protection engineers of a desire to resolve the issues promptly and of a willingness to support the recommendations of the SC. A list of attendees is provided in the enclosure.

The SC believes that its previous decision on expediting plant inspections was reinforced by the comments in this meeting. The SC indicated its view that inspections should go forward rapidly to get on with the identification and resolution of problems. To require more documentation in areas not specifically required by the Rule would slow compliance down.

Discussion

1. Views expressed by fire protection engineers.

Region I discussed the inspections of Vermont Yankee, Salem and Calvert Cliffs. It was indicated that confusion generated was caused by differences between Generic Letter 83-33 and "interpretations" but that the Region endorsed the interpretations because they would expedite the process and not create inspection problems. The implementation of Appendix R at Calvert Cliffs was successful because the licensee did a very thorough evaluation of his alternate shutdown needs, had substantial communication with the licensing staff, had some unit-specific features which benefitted shutdown, and had support of licensee management. Other comments made by Region I were that: (a) Vermont Yankee was confused by the Appendix R implementation letter, (b) at Salem the inspection was complicated because many exemptions were needed just prior to the inspection as a result of 83-33, and (c) all Region I licensees appear to be taking Appendix R seriously and making good faith efforts.

Region II discussed their experience with Appendix A and Appendix R. They saw Appendix R and fire protection as a moving target in particular since different utilities take different approaches and when these are accepted in licensing, confusion in the inspection process results. They also noted that utilities were concerned that NRC was going beyond reactor safety and getting into loss prevention. Region II raised an issue, generally endorsed by others in the meeting, that the responsibility for fire protection is fragmented because of all the disciplines responsible. In particular, in NRR responsibility lies in engineering, systems interaction and human factors safety. Licensing work also resides in the Quality Assurance Branch in IE. This, along with different Regional views and inspector approaches, results in confusion and inconsistency. It was suggested that a central contact was needed at HQ to provide central authority for fire protection. It was also indicated that reviewers and inspectors need additional guidelines, in particular, minimum acceptance criteria. Region II also pointed out differences in license requirements and Tech Specs dealing with fire protection. A discussion evolved concerning the need for augmenting Tech Specs in relation to other safety significant items. The general concensus of the fire protection engineers was that Tech Specs needed to be expanded in this area.

Region II also indicated that guidance for inspectors needed revising and expansion and that an inspection module was needed for NTOLs. It was also pointed out that the fire protection inspection must be done early in the inspection phase when the licensee has the opportunity to make changes. It was suggested that Regional inspectors accompany NRR reviewers in their site visits and that a general improvement in communication and understanding of SER commitments was needed. Finally, Region II voiced the view that the definition of fire areas in 83-33 must be retained, that guidance is needed for suppression systems and intervening combustibles, reiterated that the inspection module needs improvement by supplying minimum acceptance criteria, and stressed the need for an enforcement policy in this area.

Region III generally endorsed the comments of Region I and II. In addition, they pointed out the need for QA guidance in the area of fire protection. They stated that deficiencies in Tech Specs resulted from omission of fire damper surveillance, and inconsistancies of Code requirements. It was suggested that the present inspection modules be combined into one for all plants. Region III indicated that they felt the need to explain the rule requirements to industry; for example, 20 feet separation. They requested that RES be tasked to supply the technical basis. They felt that inspectors needed such information to guide them in making judgments and evaluations. The SC pointed out that the items in the rule were based on the best information at the time and that inspectors needed not feel obligated to explain rule obligations to licencees. If there are areas where the inspector feels safety is not well served by meeting rule provisions, such concerns should be elevated to management but that the rule, including its defense indepth provisions, seemed adequate. Finally, Region III indicated that the three things most needed were: (1) enforcement policy, (2) minimum requirements, and (3) consistent levels of inspection. To take care of (3), a training program would be needed. When asked, Region III cited the following as their three biggest frustrations: (1) the adequacy of licensee analyses, (2) the adequacy of regulatory requirements, and (3) the inconsistent reviews and inspection criteria.

Region IV has inspected Fort Calhoun, Fort St. Vrain and some NTOLs. They endorsed most of the comments of the previous Regions. In particular, they felt the need for acceptance criteria, enforcement policy, and-up-to-date Tech Specs.

Brookhaven National Laboratory (BNL) commented on problems with specific compliance vice meeting the "intent" of Appendix R. In particular, NTOLs allege that they meet the intent of Appendix R through a number of

different ways. BNL indicated that there were serious problems with consistency and interpretation of control room fires and that we lack the rationale or basis for these views. They question in particular how long is the control room habitable, what action can be taken, where two units share a control room are both units affected, and must both units shut down outside the control room. BNL also stated that we needed specific guidelines for associated circuit analysis and indicated that the SER was not always a reliable indicator of licensing commitments for inspection.

A representative of ASB indicated the scope of review for alternate shutdowns and that the criteria used were consistent and had been in use for most plant reviews. The criteria were not well documented however. He expanded on the systems used for safe shutdown, the requirements for physical separation and electrical separation for safe shutdown. With respect to the associated circuits analysis it was indicated that the evaluation assured, assuming offsite power loss, that safety could be demonstrated assuming one spurious signal, a loss of all automatic signals, and spurious operation of motor-operated valves in the high/low pressure interface. It was indicated that this included review of licensee's summary of operator actions and that, during inspection, the actual procedures are walked down.

IE's discussion focused on item 3 of the "interpretations" which states that licensees must show equipment must be "free of fire damage" before, during and after a fire. He was concerned that although Section III.G.2 specifies free of fire damage, the interpretation would allow less than this, in particular, scorched and severely heated equipment which are still barely sufficient to perform their intended functions. He said that the rule language would not allow this and that it is not appropriate and conservative.

In many of the above comments from Region and HQ representatives the SC detected a belief of bad faith by the licensees and practices which would subvert the spirit and the technical intent of the Commission's requirements. The SC pursued this to some extent but noted that there seemed to be a lack of specifics. Since the importance and safety significance of each requirement was somewhat judgmental the SC felt that the NRC needed to shoulder some responsibility for lack of compliance because of the evolution of Appendix R.

NRR representatives indicated their belief that the NRC should stick with the Generic Letter 83-33 approach, which in their view has been working, and issue enforcement policy. They felt that if the licensees were required to submit for review their entire program, both how they

OCT 1 9 1984

meet Appendix R as well as deviations or exemptions, there would be fewer inspection and enforcement problems, fewer citations and a better overall fire protection image. They pointed out that, if the licensee's evaluation is kept in house and not docketed under oath, it could be inaccurate.

The NRR representatives stated that the practical effect of the interpretations would be to relax requirements because an additional burden is placed on reviewers and inspectors that changes to licensee fixes are needed. They also asked for the agency to characterize the priority of fire protection in plant safety.

2. Views of IE enforcement staff.

Jane Axelrad discussed the current policy and indicated that it has not yet been issued because of a lack of general policy on what constitutes compliance with the rule. She gave background on the enforcement policy and comments on efforts to apply policy consistently across the regions. The SC indicated that it would provide its revision of the enforcement policy guidance for EDO approval.

RHVollmer

Richard H. Vollmer, Chairman
Fire Protection Policy Steering
Committee

Enclosure: As stated

cc: W. Dircks
V. Stello
R. Minogue
T. Murley, R-I
J. O'Reilly, R-II
J. Keppler, R-III
R. Martin, R-IV
J. Martin, R-V
G. Cunningham
E. Case
J. Taylor
D. Eisenhut
R. Bernero
G. Arlotto
F. Rosa
SC Committee

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

October 19, 1984

MEMORANDUM FOR: Harold R. Denton, Director, NRR
Richard C. DeYoung, Director, OIE

FROM: Richard H. Vollmer, Chairman
Fire Protection Policy Steering Committee

SUBJECT: FOURTH MEETING OF THE FIRE PROTECTION POLICY STEERING COMMITTEE

Summary

At its fourth meeting, the Fire Protection Policy Steering Committee (SC) again considered the generic letter to licensees, the enforcement policy guidance and the scope, timing and resources for the expedited fire protection inspections. The SC focussed on some of the peripheral issues which would be important to the success of the expedited inspections and other SC initiatives. At this meeting, the SC decided that:

- Prior to the fire protection inspections, a workshop would be held with IE, NRR, and Regional participants in the inspections to assure understanding in the objectives, scope, and technical issues and to help provide consistency between inspection teams;

- A team would be established to promptly handle disputes between licensees and the inspection teams; and

- That a standard condition should be incorporated into all licenses, requiring maintenance of the fire protection commitments but allowing change under 50.59 which do not decrease the level of fire protection with annual reporting to the Commission of such changes.

Discussion

1. Expedited inspections.

The SC discussed the scope, timing, and resources for the expedited inspections. There was a discussion of whether this should be a review or an inspection. The SC felt that the concept was one of an inspection rather than a review and that the availability of enforcement was important to the process. The SC discussed resources and concluded that an adequate pool existed; however it was not clear if or how the expedited

inspections would affect other programs. The Working Group/IE was asked to complete a module to be used in these inspections and to plan out which licensees would be inspected to best achieve the objectives. A one-year schedule of plants and resources was requested by the SC.

The SC also discussed how to assure a common understanding of the objectives, scope, and technical issues by all participants in these inspections. To be effective, such inspections need to be uniform. The SC decided that a workshop would be held prior to initiation of the inspections wherein the SC would lay out the philosophy and intent, and technical and procedural issues would be discussed. The SC also decided that it would be important to have a central authority available to promptly and uniformly resolve disputes between the licensees and inspection teams. This was felt to be necessary so that major issues would not be left unsettled and so that licensees would not pursue fire protection solutions that would be unlikely to be acceptable to NRC. The Chairman of the SC was assigned to draft a charter and membership for this authority for SC's consideration. It was also agreed that the resolution of disputes by this authority would be promptly published to all inspection teams.

2. Generic Letter.

The need for a standard fire protection license condition was discussed. Problems resulting from inconsistent license conditions have been raised during the past year. It was agreed that a standard condition, similar to that drafted by ELD and NRR for current NTOLs, should be prepared. The SC discussed the tim ing for reporting of fire protection changes conducted under 50.59 and it was decided that annual reporting was appropriate based on other important 50.59 issues that utilize annual reporting. A discussion ensued on how to get such a license condition applied across the board to all licensees. The use of a Generic Letter or a 50.54(f) letter was discussed. ELD was tasked with developing a recommendation.

3. Enforcement Policy.

The SC discussed in detail a proposed fire protection enforcement policy. Some issues were settled and it was decided that the SC would complete this activity at its next meeting.

Finally, the SC decided to hold its next meeting on October 17 in Bethesda and to hold its final session in Chicago, for two days or more if needed, to complete activities and prepare a report to the EDO.

Richard H. Vollmer, Chairman
Fire Protection Policy Steering Committee

cc: See next page

cc: W. Dircks
V. Stello
R. Minogue
T. Murley, R-I
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SC Committee

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

OCT 19 1984

MEMORANDUM FOR: Fire Protection Policy Steering Committee

FROM: Richard H. Vollmer, Chairman
Fire Protection Policy Steering Committee

SUBJECT: AGENDA FOR SIXTH MEETING

The sixth meeting of the Fire Protection Policy Steering Committee will be held at 9:00 a.m. on October 22, 1984 in the Region III offices. At this meeting you should be prepared to discuss and finalize versions of:

- Enforcement Policy
- Generic Letter and Interpretations
- Technical Issues Package
- Standard License Condition
- Inspection Module

In addition, we need to prepare our final report to the EDO. I will FAX an outline to you for your consideration today. In addition, there are a number of issues and suggestions still left hanging. For example: the central point of contact for fire protection issue; the status of NFPA codes; what to do about Tech Specs; and the format of the workshop in advance of the expedited inspections.

Richard H. Vollmer, Chairman
Fire Protection Policy Steering
Committee

cc: See next page.

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

OCT 2 6 1984

MEMORANDUM FOR: Harold R. Denton, Director, ONRR
Richard C. DeYoung, Director, OIE

FROM: Richard H. Vollmer, Chairman
Fire Protection Policy Steering Committee

SUBJECT: FIFTH AND SIXTH MEETINGS OF THE FIRE PROTECTION POLICY STEERING COMMITTEE

At the fifth and sixth meetings of the Fire Protection Policy Steering Committee (SC), held in Bethesda on October 17 and the Region III offices on October 22, respectively, the final version of (1) enforcement policy guidance, (2) Generic Letter, (3) standard license condition, (4) temporary instruction for fire protection inspections, and (5) technical issues package of questions and answers were discussed, edited, and put into final form. No new initiatives were discussed but the impact and consistency of all initiatives developed by the SC were reviewed. The SC also assured that all issues included in the EDO memo of September 13 had been fully addressed and that all issues raised to the SC's attention by other parties had been fully considered.

Richard H. Vollmer, Chairman
Fire Protection Policy Steering
Committee

cc: See next page.

cc: W. Dircks
V. Stello
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F. Rosa
SC Committee

Applicable
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- - -

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

January 29, 1985

TO ALL POWER REACTOR LICENSEES AND APPLICANTS FOR AN OPERATING LICENSING

Gentlemen:

SUBJECT: OPERATOR LICENSING EXAMINATIONS (Generic Letter 85-04)

This letter is to request your best estimate of your need for operator licensing examinations for the remainder of this fiscal year (February 1, 1985 to September 30, 1985) and fiscal years FY 1986, FY 1987, and FY 1988 (October 1 to September 30 of each year). This information is needed to update the schedules you provided in response to Generic Letter 83-40. We are also requesting that you provide requalification examination schedules for this same time period. Please identify the dates you have scheduled your requalification examination and your anticipated requests for licensing examination site visits and the number of examinations for each visit.

Your best estimates are needed to plan for NRC resources to meet your operator licensing needs. Please be aware that in response to budget reductions the NRC has resources for only two visits to each site per year for administering licensing examinations. To meet this goal in FY 1986 and beyond, the regional offices may be required to redistribute the requested facility operator examinations visits across the entire fiscal year to even out the examination workload and eliminate high demand periods. Therefore, your submittal of this schedule does not guarantee the number or date of examinations requested. However, an accurate estimate of the need for examinations will allow us to propose budget modifications, if necessary. You should also keep us informed of significant changes in your estimates as they occur, so that we can keep our data base current.

Your schedules, in the enclosed suggested format, should be returned to Mr. Don Beckham, Chief, Operator Licensing Branch, AR-5221, Washington, D.C., 20555, with a courtesy copy to the appropriate Regional Administrator 30 days from the receipt of this letter. We appreciate your assistance. If you have any questions, please call Mr. Don Beckham, Chief, Operator Licensing Branch, at (301) 492-4868.

~~8502010365~~ 6 pp.

This request was approved by the Office of Management and Budget (OMB) under clearance OMB-3150-0018, which expired December 31, 1987.

Sincerely,

Darrell G. Eisenhut, Director
Division of Licensing

Enclosures:
1. Operating Licensing Examination Schedule
2. Requalification Examination Schedule
3. List of Recently Issued Generic Letters

OPERATOR LICENSING EXAMINATION SCHEDULE

Facility ____________ NRC Region ____________

	FY 1985 (February 1, 1985 - September 30, 1985)	FY 1986	FY 1987	FY 1988
1. Date	____	Date ____	Date ____	Date ____
#RO	____	____	____	____
#SRO	____	____	____	____
#SRO Upgrade	____	____	____	____
#Instructor Certification	____	____	____	____
#SRO Limited to Fuel Handling	____	____	____	____
2. Date	____	Date ____	Date ____	Date ____
#RO	____	____	____	____
#SRO	____	____	____	____
#SRO Upgrade	____	____	____	____
#Instructor Certification	____	____	____	____
#SRO Limited to Fuel Handling	____	____	____	____
3. Date	____	Date ____	Date ____	Date ____
#RO	____	____	____	____
#SRO	____	____	____	____
#SRO Upgrade	____	____	____	____
#Instructor Certification	____	____	____	____
#SRO Limited to Fuel Handling	____	____	____	____

Please indicate initial cold license examinations by placing an asterisk (*) by the date. Please indicate examinations intended to extend an operator's license to a second or subsequent unit with two asterisks (**) (e.g., Unit One is in operation and Unit Two is approaching fuel load). Three RO candidates with no previous license are to be examined on both Units One and Two and five RO candidates with licenses on Unit One are to be examined to extend their licenses to Unit Two. Indicate (2/15/85, RO 3,5**).

ENCLOSURE 2

REQUALIFICATION EXAMINATION SCHEDULE

Facility________________ NRC Region________________

FY 1985 (February 1, 1985 - September 30, 1985)	FY 1986	FY 1987	FY 1988
1. Date______	Date______	Date______	Date______
2. Date______	Date______	Date______	Date______
3. Date______	Date______	Date______	Date______
4. Date______	Date______	Date______	Date______

LIST OF RECENTLY ISSUED GENERIC LETTERS

GENERIC LETTER NO.	SUBJECT	DATE
84-15	Proposed Staff Actions to Improve and Maintain Diesel Generator Reliability	7/2/84
84-16	Adequacy of On-Shift Operating Experience for Applicants	6/27/84
84-17	Annual Meeting to Discuss Recent Developments Regarding Operator Training, Qualifications and Examinations	7/3/84
84-18	Filing of Applications for Licenses and Amendments	7/6/84
84-19	Availability of Supplement 1 to NUREG-0933 "A Prioritization of Generic Safety Issues"	8/6/84
84-20	Scheduling Guidance for Licensee Submittals of Reloads that Involve Unreviewed Safety Questions	8/20/84
84-21	Long Term Low Power Operation in PWR's	10/16/84
84-22	Not used	
84-23	Reactor Vessel Water Level Instrumentation in BWRs	10/26/84
84-24	Clarification of Compliance to 10 CFR 50.49 Environmental Qualification of Electrical Equipment Important to Safety for Nuclear Power Plants	12/27/84
85-01	Fire Protection Policy Steering Committee Report	1/9/85
85-03	Clarification of Equivalent Control Capacity For Standby Liquid Control Systems	1/28/85
85-04	Operator Licensing Examinations	1/29/85
85-05	Inadvertent Boron Dilution Events	1/31/85

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

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

January 31, 1985

TO ALL PRESSURIZED WATER REACTOR LICENSEES

Gentlemen:

SUBJECT: INADVERTENT BORON DILUTION EVENTS (Generic Letter 85-05)

The purpose of this letter is to inform each licensee of operating pressurized water reactors of the staff position resulting from the evaluation of Generic Issue 22, "Inadvertent Boron Dilution Events" regarding the need for upgrading the instrumentation for detection of boron dilution events in operating reactors.

A boron dilution event is considered as an anticipated operational occurrence which may occur at moderate frequency. The staff has performed analyses of unmitigated boron dilution events for a typical plant for each pressurized water reactor (PWR) vendor. The staff determined that while power excursions during boron dilution events are possible if the operator does not take any action and sufficient volume of dilution water is available, the excursion should be self-limiting. The staff analyses indicate that these type of boron dilution transients should not exceed the staff's acceptance criteria. However, our analyses also show that a few plants may experience slight overpressurization in excess of the 110% overpressure limit in the Residual Heat Removal system if the event occurs during a particular mode of operation.

In addition, the staff recognizes that many operating plants do not have distinct, positive alarms to alert the operators to boron dilution events but rely on other devices such as audible count rate meters. Other problems include lack of alarm redundancy and lack of technical specifications which would prevent operators from taking alarming devices out of service. The staff also does not consider it prudent to credit operators with the ability to recognize a boron dilution event and take the proper mitigative action within specified time limits in the absence of positive boron dilution alarms.

Considering all of the above factors and possible consequences of boron dilution events, the staff has concluded that the criteria in Section 15.4.6 of the Standard Review Plan are adequate and should continue to be applied to plants currently undergoing licensing review. However, the consequences are not severe enough to jeopardize the health and safety of the public and do not warrant backfitting requirements for boron dilution events at operating reactors. The staff will continue to review the analyses of the Boron Dilution Event in reload applications to assure that reasonable confidence is provided that operators can be expected to take the right corrective action using the installed systems.

In summary, while the NRC will not require operating plant backfits for boron dilution events at this time, the staff would regard an unmitigated boron dilution event as a serious breakdown in the licensee's ability to control its plant, and strongly urges each licensee to assure itself that adequate protection against boron dilution events exists in its plants.

~~8502010366~~ 4pp.

This generic letter is provided for information only, and does not involve any reporting requirements. Therefore, no clearance from the Office of Management and Budget is required.

Darrell G. Eisenhut

Darrell G. Eisenhut, Director
Division of Licensing

Enclosure:
List of Generic Letters

LIST OF RECENTLY ISSUED GENERIC LETTERS

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84-18	Filing of Applications for Licenses and Amendments	7/6/84
84-19	Availability of Supplement 1 to NUREG-0933 "A Prioritization of Generic Safety Issues"	8/6/84
84-20	Scheduling Guidance for Licensee Submittals of Reloads that Involve Unreviewed Safety Questions	8/20/84
84-21	Long Term Low Power Operation in PWR's	10/16/84
84-22	Not used	
84-23	Reactor Vessel Water Level Instrumentation in BWRs	10/26/84
84-24	Clarification of Compliance to 10 CFR 50.49 Environmental Qualification of Electrical Equipment Important to Safety for Nuclear Power Plants	12/27/84
85-01	Fire Protection Policy Steering Committee Report	1/9/85
85-03	Clarification of Equivalent Control Capacity For Standby Liquid Control Systems	1/28/85
85-04	Operator Licensing Examinations	1/29/85
85-05	Inadvertent Boron Dilution Events	1/31/85

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

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

May 23, 1985

TO ALL HOLDERS OF CONSTRUCTION PERMITS AND OPERATING LICENSES

SUBJECT: 10 CFR 20.408 TERMINATION REPORTS - FORMAT
GENERIC LETTER NO. 85-08

Pursuant to 10 CFR 20.408, licensees are required to submit to the NRC a report of each individual's exposure to radiation and radioactive material when the individual terminates employment or work assignment at their facility. These exposure reports are commonly referred to as §20.408 termination reports.

Previously, we have not specified a preferred reporting format for compliance with this regulation. However, the NRC is currently receiving approximately 100,000 termination reports each year, and this number is steadily increasing. For purposes of efficient automatic data processing, it is important to use a standard format. Processing in a more timely fashion will make the data more useful to the NRC and others in their performance of various duties (see Enclosure 1).

For future §20.408 termination reports we request that you voluntarily use the attached Standard NRC Form-439. Instructions for completing the form are attached to the form. Questions regarding these instructions should be directed to Barbara G. Brooks, Office of Nuclear Regulatory Research, Washington, D. C. 20555, (301) 427-4577.

The NRC is also conducting a pilot program for the electronic transmission of the termination data to the NRC via computer tapes or by direct linkup to the NRC's computing facility, and we would like to encourage you to consider participating. Should you desire more information about this program please contact Ms. Brooks.

The form is intended for use in connection with the information collection requirement established in Section 20.408, 10 CFR Part 20 and approved under OMB Clearance Number 3150-0014.

Hugh L. Thompson, Jr.

Hugh L. Thompson, Jr., Director
Division of Licensing
Office of Nuclear Reactor Regulation

Enclosure:
1. Uses of Radiation Exposure Data
2. Five copies of NRC Form 439
(With instructions)
3. List of Generic Letters

~~8505210106~~ 17pp.

Uses of Radiation Exposure Data

A number of NRC licensees have inquired as to how occupational radiation exposure data (from reports required by the NRC) are used by the NRC staff. This is a very appropriate inquiry that may be of importance to many affected licensees. In combination with other sources of information, the principal uses of the data are to provide facts regarding routine occupational exposures to radiation and radioactive material that occur in connection with NRC-licensed activities, including individual and collective radiation doses from external sources as well as pertinent information on the inhalation of radioactive material (nuclides involved, bioassay results, exposure magnitude, etc.) These facts are used by the NRC staff as indicated below:

1. The external-dose data permit evaluation of the radiological risk associated with NRC-licensed activities, including the size of the workforce and the collective dose.

2. The data permit evaluation, from the viewpoint of trends, of the effectiveness of the overall NRC/licensee radiation protection and ALARA efforts. They also provide for the identification (and subsequent correction) of unfavorable trends.

3. The data provide for governmental monitoring of the potential transient-worker problem.

4. The data are used in the establishment of priorities for the utilization of NRC health physics resources: research, standards development, regulatory program development.

5. The data are considered in reviews of inspection frequencies that are programmed for various categories of licensees.

6. Licensing action decisions are often influenced by the data.

7. The data are used for comparative analyses of radiation protection performance: US/foreign, BWR's/PWR's, civilian/military, plant by plant, nuclear industry with other industries, etc.

8. The data permit analysis of annual dose distribution changes which can trigger investigations as to the cause.

9. The data are used for purposes of justification in the annual budget process.

10. The data provide facts for evaluating the adequacy of the current risk-limitation system (e.g., are individual lifetime dose limits, worker population collective dose limits, requirements for optimization, etc., needed).

11. The effectiveness of dose-reduction measures is evaluated using the data (e.g., methods for reducing individuals doses that may increase the collective dose).

12. The data provide facts for answering Congressional and Administration inquiries and for responding to questions raised by public interest groups, special interest groups, labor unions, etc.

13. The data permit comparisons of occupational radiation risks with potential public risks when action for additional protection of the public involves worker exposures.

14. The data provide information which can be used in the planning of epidemiological studies.

With regard to routine work-place conditions, the annual statistical summary reports required by 20.407, the termination reports required by §20.408, and the annual dose data reported by work function in accordance with Subsection 6.9.1.5 of the standard technical specifications for nuclear power plants provide the only centralized data base available to assist the staff in the performance of its duties as listed above. It is to everyone's advantage if these duties are performed by a well-informed staff in the light of factual information.

NRC FORM 439
(4-86)
10 CFR §20.408

U.S. NUCLEAR REGULATORY COMMISSION

REPORT OF TERMINATING INDIVIDUAL'S OCCUPATIONAL EXPOSURE

SEE THE ATTACHED INSTRUCTIONS

1. DATE OF REPORT

2. NRC LICENSE NUMBER(S)

PART I. LICENSEE AND INDIVIDUAL IDENTIFICATION DATA

3. NAME AND ADDRESS OF REPORTING LICENSEE

4. NAME OF INDIVIDUAL *(first, middle initial, last)* AND ADDRESS *(optional)*

5. NAME AND ADDRESS OF EMPLOYER, IF DIFFERENT FROM ABOVE *(Optional)*

6. SOCIAL SECURITY NUMBER

7. DATE OF BIRTH | MONTH | DAY | YEAR

PART II. EXTERNAL DOSE DATA

☐ 8. PERSONNEL MONITORING FOR EXTERNAL EXPOSURE TO RADIATION WAS NOT PROVIDED

9. PERIOD(S) OF EXPOSURE *(earliest date first)*	10. WHOLE BODY DOSE *(rems)* DEEP a. TOTAL	b. NEUTRON	SHALLOW *(skin)* c. TOTAL	d. BETA	11. EXTREMITY DOSE *(rems)* SHALLOW *(skin)*

PART III. INTERNAL EXPOSURES TO RADIOACTIVE MATERIAL

☐ 12. PERSONNEL MONITORING FOR EXPOSURE TO RADIOACTIVE MATERIAL WAS NOT PROVIDED

13. PERIOD(S) OF EXPOSURE *(earliest date first)*	14. NUCLIDE	15. FORM (S, I)	16. BIOASSAY RESULTS a. IN VIVO (nCi) (1) BURDEN	(2) ORGAN	b. URINALYSIS RESULTS (L)	17. DOSE ESTIMATES *(rems)* a. COMMITTED DOSE	b. ANNUAL DOSE	c. ORGAN	18. INTAKE (MPC Hrs.)

19. OTHER BIOASSAY RESULTS

20. IF THIS REPORT IS BEING USED TO SATISFY THE NOTIFICATION REQUIREMENTS OF 10 CFR 19.13, CHECK THE FOLLOWING BOX

☐ YES *(This report is furnished to you under the provision of the Nuclear Regulatory Commission's regulation 10 CFR Part 19. You should preserve this report for further reference.*

INSTRUCTIONS FOR COMPLETING NRC FORM 439,
Report of Terminating Individual's Occupational Exposure

If you are licensed by the U.S. Nuclear Regulatory Commission (NRC) as specified in §20.408(a), 10 CFR Part 20, you are required to submit termination radiation exposure reports on certain individuals to the Director, Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission, Washington, DC 20555. This information is to be taken from dose records that must be maintained under §20.401 for individuals likely to receive exposure to radiation that exceeds a certain percentage of the NRC dose standards for the whole body, skin or extremities—25% for workers of age 18 years or more, 5% for workers younger than 18. The term "individual" is used below to represent the worker for whom this report is submitted. The term "dose" as used in Form 439 and in these instructions refers to the dose in rems as defined in §20.4(a) and subsequently designated "dose equivalent" in ICRU Report II (1968). The time to be covered by this report is that period of employment, or work assignment in your facility(s), which ended with the most recent termination and was not interrupted by any previous termination during which personnel monitoring was required by §20.202(a) and/or bioassays were required by your license. "Termination" is defined in §20.3(a)(19). Parts II and III of this form reflect regulatory requirements as well as requests intended to standardize reporting methods; requests are clearly identified as such.

PART I. LICENSEE AND INDIVIDUAL IDENTIFICATION DATA

This part identifies the licensee submitting the report and the terminating individual. It must be completed even if only one of the remaining Parts of this form is applicable. Enter the following data:

ITEM NUMBER

1 Date that the report was prepared.

2 Current NRC license number assigned to the facility(s) in which the individual received the reported dose. If more than one license is involved, enter the license number for the facility or activity under which most of the dose was incurred as the first number. If this is not practical, enter the license numbers in the order of original issuance.

3 Name and address of your facility as it appears on your NRC license.

4 The individual's first name, middle initial, and surname. (Address of the individual may be included, but it is not entered into the NRC records system.)

5 The name and address of the individual's employer, if it is different from the reporting licensee. (Optional, not entered into the NRC records system.)

6 The individual's social security number; if not available, enter the word "unknown."

7 The individual's date of birth.

PART II. EXTERNAL DOSE DATA

For the purpose of this form, the deep dose is defined as the dose assessed at a tissue depth of 300 or 1,000 mg/cm² (or less), the shallow dose is defined as the dose to the skin of any part of the body, and the extremities are defined as hands and forearms, feet and ankles.

Item Number 8 If the individual was not monitored for external exposure to radiation, you are requested to check the box to the left and go to Part III.

COLUMN NUMBER

9 Specify the reporting intervals (periods of exposure) that the individual was monitored at your facility(s) pursuant to §20.202. You are requested to use annual increments up to the year of termination and increments not to exceed one quarter for the year in which the individual terminated.
ANNUAL: Indicate the month and year of the beginning date of exposure when showing annual increments (e.g., May 1979) and indicate the year only for subsequent annual increments;
QUARTER: For each completed quarter of the year of termination, indicate the quarter and year by date;
CURRENT QUARTER: Specify the beginning and ending dates of the actual exposure period (month, day, year);
Enter the following data:

10a Unless the eyes are shielded, enter the deep dose assessed at a tissue depth of 300 mg/cm² (lens depth) or less. If the eyes are protected by shielding which has a tissue equivalent thickness of 700 mg/cm² or more, the deep dose may be assessed at 1000 mg/cm² (gonad depth) or less. Enter the total dose of record, i.e., the highest dose received at the selected depth, from all types of external radiation sources, at any location on the body except the skin and the extremities (hands and forearms, feet and ankles).

10c For all skin areas, except that of the extremities, enter in column 10c the shallow dose of record. Record the total dose to the skin, i.e., the highest dose delivered by all radiation incident on the skin, including non-trivial doses from skin contamination, which penetrates to the depth at which the shallow dose is determined. The dose at a depth of 7 mg/cm² or less, averaged over 1 cm², is acceptable. If Column 10c is left blank, it will be assumed that the entry in 10a is applicable also for the shallow dose. Therefore, an entry for shallow dose is required only if it exceeds the deep dose.

10b & d You are requested to enter in column 10b the contribution made by neutron radiation to the dose reported in Column 10a, and to enter in Column 10d the contribution made by beta radiation to the dose reported in Column 10c. Enter XXX if it is known that there was no exposure to radiation of the type specified in the column heading. Enter UNK if a detectable exposure is reported in 10a or 10c which could have included a beta or neutron contribution of unknown magnitude.

10 & 11 You are requested to enter m or zero (in each column of 10, or in 11) if the dose was undetectable, i.e., the radiation to which the worker's dosimeter was exposed produced a response that you considered to be statistically indistinguishable from the response caused by inherent variabilities of the dosimeter system. Note: It is sometimes required to add m (or its equivalent) to a real number; although NRC regulations do not specify a summation procedure, the NRC staff arbitrarily assigns 10 mrems to be a value of m (assuming 0.5 L ≤ 10 ≤ L, where L is the detection limit) for the purposes of statistical analyses.

11 Reporting of the extremity dose is required. You are requested to comply in the following manner. Enter the dose of record, i.e., the highest dose, averaged over any area of 1 cm², determined for the skin of the hands and forearms or feet and ankles during the reported period. It is unnecessary to specify the extremity that received the dose; doses to different extremities should not be added together. The dose is to include that delivered by all radiation incident on the skin, including non-trivial doses from skin contamination, which penetrates to the depth at which the shallow dose is determined. The dose at a depth of 7 mg/cm² or less is acceptable. If Column 11 is left blank, it will be assumed that the entry in 10c is applicable also for the extremity dose; an entry in Column 11 is required only if the shallow dose exceeded the deep dose.

PART III. INTERNAL EXPOSURES TO RADIOACTIVE MATERIAL

If you are licensed by the NRC as specified in §20.408(a), 10 CFR Part 20, and if your license requires bioassay services for workers at your facility, you are required to submit termination reports on personnel exposures to radioactive material, containing information that you have obtained in compliance with the license and recorded in compliance with §20.401. You are requested to include in each termination report information that you have obtained in compliance with §20.103(a)(3) and 20.103(c)(2) and recorded in compliance with §20.401. This part provides for the reporting of internal monitoring procedures in terms of bioassay results, dose estimates, or intake. Any one (or more) of these reporting methods may be used. The term "individual" is used below to represent the worker for whom this report is submitted. The term "exposures to radioactive material" is used in connection with these termination reports to represent the entry of radioactive material into the body.

Item Number 12 If the individual was not monitored for exposure to radioactive material, you are requested to check the box to the left; otherwise, enter the following data:

Column Number 13 If bioassay results are reported (Column 16), you are requested to use the following format. Summarize by year, separately listing the number of measurements which indicated quantities or concentrations that were undetectable, i.e., in the detection system used, the radionuclide present (if any) produced a response that you considered to be statistically indistinguishable from its background. In Column 13 enter each year bioassay was performed, including the year of termination. In Columns 14 through 16, use two lines for each year, as in the example shown below, the upper line for detectable results and the lower line for those undetectable. In 16a and/or 16b, upper line, enter the number (including zero) of detectable measurements followed in parenthesis by the highest verified result, if any; lower line, enter the number (including zero) of measurements indicating undetectable amounts.

INSTRUCTIONS FOR COMPLETING NRC FORM 439 (Continued)

COLUMN NUMBER

13 (Continued)

Column 13	Column 14	Column 15	Column 16 16a(1)	16a(2)	16b (pCi/L)
1982	U-nat	"I"	0	lung	2(1)
			2	lung	10
1983	U-nat(Th 234)	"I"	1(7)	lung	4(6)
			1	lung	8
1984	U-nat(Th 234)	"I"	2(14)	lung	12(13)
			0	lung	0

Units for the numbers in parentheses shown in Column 16b are to be specified in the heading for Column 16b. If Columns 17 or 18 are completed, notations in Column 16 are unnecessary.

If the dose commitment (50-year integrated dose) is reported, indicate in Column 13 by beginning and ending dates (month, day, year) the period during which the associated radioactive material was taken into the body.

If annual doses are reported, enter in Column 13 the calendar year over which each dose was integrated, including the first and any succeeding years of this employment or work assignment and the year following the termination date.

For entries in Column 18 (intake), specify the reporting intervals (periods of exposure) during which the individual was exposed to concentrations of radioactive material, using annual increments up to the year of termination and increments not to exceed one quarter for the year in which the individual terminated. The periods of exposure for intakes should appear as follows:
ANNUAL. Indicate the month and year of the beginning date of exposure when showing annual increments (e.g., June 1983) and indicate the year only for subsequent annual increments.
QUARTER. For each completed quarter of the year of termination, indicate the quarter and year by date.
CURRENT QUARTER. Specify the beginning and ending dates of the actual exposure period (month, day, year).
Reported intakes which include only the quantities required to be assessed in accordance with §20.103(a)(3) are acceptable.

14 — Identify the symbol used in 10 CFR Part 20, Appendix B, for the radionuclide or mixture of radionuclides for which in vivo and/or urinalysis measurements were performed (e.g., Co 60, U 235). If the measured quantity of activity for one radionuclide is also used to estimate other radionuclide quantities, identify the radionuclide actually measured in parentheses immediately after the radionuclide listed in Column 14. See the example given in the directions for Column 13 where U-nat(Th 234) is entered in Column 14 indicating that the uranium lung burden was determined from measurements of Th 234 photons.

15 — Enter the form, S for soluble or I for insoluble, of the radionuclide to which the worker was exposed. If unknown, use quotes around the letter, thus indicating which concentration value in Part 20, Appendix B, Table I, Column 1, was assumed to apply.

16, 17 & 18 — These columns allow for the reporting of the results of the internal monitoring procedures in terms of bioassay results, or dose estimates, or intake. You may use one or more of these methods.

16a(1) & a(2) — For each year during which in vivo measurements were performed, as shown in Column 13, enter in Column 16a(1) the number of detectable measurements followed by the highest verified result (in nanocuries) in parentheses. On the next line in this column, enter the number of measurements that indicated undetectable amounts. Specify in Column 16a(2) the organ in which the indicated radionuclide was found. See the example given in the directions for Column 13.

16b — First, enter the gravimetric or radiometric unit in which the urinalysis results are reported (e.g., micrograms per liter, nanocuries per liter) in the blank space of the heading for Column 16b. In Column 16b, for each year during which urinalyses were performed, enter the number of detectable results followed by the highest numerical value of the concentration in urine of the radionuclide listed in Column 14 for the year specified in Column 13. On the next line in this column, enter the number of measurements indicating undetectable amounts. See the example given in the directions for Column 13.

17a, b, & c — Specify in Column 17c the organ or tissue receiving doses estimated in Column 17a or 17b. (Note that it is not necessary to provide both the committed and annual doses.) For Columns 17a and 17b you are requested to follow the procedures below; if any alternative procedures are used, describe them on the back of this form. In 17a, enter the dose integrated from t_o to 50 years, where t_o is the beginning date shown in Column 13. In 17b enter the dose integrated over each calendar year shown for this purpose in Column 13. Include the first and any succeeding years of this employment or work assignment and the year following the termination date. Base dose estimates on the quantity (as a minimum) of the radionuclide, Column 14, taken into the body at your facility(s) during this employment or work-assignment period.

18 — Reporting of radionuclide intakes, as determined by air sampling, is not required by 10 CFR 20.408. However, should this option be chosen, indicate the time-weighted concentrations of radioactive material (i.e., MPC-hours) to which the individual was exposed during the time periods indicated in Column 13. Refer to the last paragraph of the instructions for Column 13 for the time intervals to be used. Complete Columns 13, 14, and 15 for each entry in Column 18.

Item number 19 — Any bioassay results that cannot be reported as described above should be entered here.

Item number 20 — If you wish to send a copy of this report to the terminating individual to satisfy the notification requirements of 10 CFR 19.13, check the "Yes" box.

PRIVACY ACT STATEMENT

Pursuant to 5 U.S.C. 552a(e)(3), enacted into law by section 3 of the Privacy Act of 1974 (Public Law 93-579), the following statement is furnished to individuals and persons who supply information to the Nuclear Regulatory Commission on NRC Form 439. This information is maintained in a system of records designated as NRC-27 and described at 40 Federal Register 45344 (October 1, 1975).

1. AUTHORITY. Sections 53, 63, 65, 81, 103, 104, 161(b), and 161(o) of the Atomic Energy Act of 1954, as amended (42 U.S.C. 2073, 2093, 2095, 2111, 2133, 2134, 2201(b), and 2201(o)). The authority for soliciting the social security number is 10 CFR Part 20.

2. PRINCIPAL PURPOSE(S). The information is used by the NRC in its evaluation of the risk of radiation exposure associated with the licensed activity and in exercising its statutory responsibility to monitor and regulate the safety and health practices of its licensees. The data permit a meaningful comparison of both current and long-term exposure experience among types of licensees and among licensees within each type. Data on your exposure to radiation are available to you upon request.

3. ROUTINE USES. The information may be used to provide data to other Federal and State agencies involved in monitoring and/or evaluating radiation exposure received by individuals employed as radiation workers on a permanent or temporary basis and exposure received by monitored visitors. The information may also be disclosed to an appropriate Federal, State, or local agency in the event the information indicates a violation or potential violation of law and in the course of an administrative or judicial proceeding.

4. WHETHER DISCLOSURE IS MANDATORY OR VOLUNTARY AND EFFECT OF NOT PROVIDING INFORMATION ON INDIVIDUAL OR PERSON. It is voluntary that you furnish the requested information, including name, date of birth, and social security number. The social security number is used to assure that NRC has an accurate identifier not subject to the coincidence of similar names or birth dates among the large number of persons on whom data is maintained. Please note, however, that the licensee must file a termination report containing certain required information, such as social security number, for each individual whose employment or work assignment has terminated and for whom personnel monitoring was required under 10 CFR 20.202. Failure of the licensee to provide the information under 10 CFR §20.202 and 20.408 may subject the licensee to enforcement action under 10 CFR 20.601.

5. SYSTEM MANAGER(S) AND ADDRESS. Director, Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555

NRC FORM 439
(4-85)
10 CFR §20.408

U.S. NUCLEAR REGULATORY COMMISSION

REPORT OF TERMINATING INDIVIDUAL'S OCCUPATIONAL EXPOSURE

SEE THE ATTACHED INSTRUCTIONS

1. DATE OF REPORT

2. NRC LICENSE NUMBER(S)

PART I. LICENSEE AND INDIVIDUAL IDENTIFICATION DATA

3. NAME AND ADDRESS OF REPORTING LICENSEE

4. NAME OF INDIVIDUAL *(first, middle initial, last)* AND ADDRESS *(optional)*

5. NAME AND ADDRESS OF EMPLOYER, IF DIFFERENT FROM ABOVE *(Optional)*

6. SOCIAL SECURITY NUMBER

7. DATE OF BIRTH — MONTH | DAY | YEAR

PART II. EXTERNAL DOSE DATA

☐ 8. PERSONNEL MONITORING FOR EXTERNAL EXPOSURE TO RADIATION WAS NOT PROVIDED

9. PERIOD(S) OF EXPOSURE *(earliest date first)*	10. WHOLE BODY DOSE *(rems)* — DEEP: a. TOTAL	DEEP: b. NEUTRON	SHALLOW *(skin)*: c. TOTAL	SHALLOW *(skin)*: d. BETA	11. EXTREMITY DOSE *(rems)* — SHALLOW *(skin)*

PART III. INTERNAL EXPOSURES TO RADIOACTIVE MATERIAL

☐ 12. PERSONNEL MONITORING FOR EXPOSURE TO RADIOACTIVE MATERIAL WAS NOT PROVIDED

13. PERIOD(S) OF EXPOSURE *(earliest date first)*	14. NUCLIDE	15. FORM (S, I)	16. BIOASSAY RESULTS — a. IN VIVO (nCi): (1) BURDEN	a. IN VIVO (nCi): (2) ORGAN	b. URINALYSIS RESULTS (L)	17. DOSE ESTIMATES *(rems)* — a. COMMITTED DOSE	b. ANNUAL DOSE	c. ORGAN	18. INTAKE (MPC-Hrs.)

19. OTHER BIOASSAY RESULTS

20. IF THIS REPORT IS BEING USED TO SATISFY THE NOTIFICATION REQUIREMENTS OF 10 CFR 19.13, CHECK THE FOLLOWING BOX

☐ YES *(This report is furnished to you under the provision of the Nuclear Regulatory Commission's regulation 10 CFR Part 19. You should preserve this report for further reference.)*

INSTRUCTIONS FOR COMPLETING NRC FORM 439,
Report of Terminating Individual's Occupational Exposure

If you are licensed by the U.S. Nuclear Regulatory Commission (NRC) as specified in §20.408(a), 10 CFR Part 20, you are required to submit termination radiation exposure reports n certain individuals to the Director, Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission, Washington, DC 20555. This information is to be taken from se records that must be maintained under §20.401 for individuals likely to receive exposure to radiation that exceeds a certain percentage of the NRC dose standards for the whole ody, skin or extremities—25% for workers of age 18 years or more, 5% for workers younger than 18. The term "individual" is used below to represent the worker for whom this report is submitted. The term "dose" as used in Form 439 and in these instructions refers to the dose in rems as defined in §20.4(a) and subsequently designated "dose equivalent" in ICRU Report II (1968). The time to be covered by this report is that period of employment, or work assignment in your facility(s), which ended with the most recent termination and was not interrupted by any previous termination during which personnel monitoring was required by §20.202(a) and/or bioassays were required by your license. "Termination" is defined in §20.3(a)(19). Parts II and III of this form reflect regulatory requirements as well as requests intended to standardize reporting methods; requests are clearly identified as such.

PART I. LICENSEE AND INDIVIDUAL IDENTIFICATION DATA

This part identifies the licensee submitting the report and the terminating individual. It must be completed even if only one of the remaining Parts of this form is applicable. Enter the following data:

ITEM NUMBER

1	Date that the report was prepared.
2	Current NRC license number assigned to the facility(s) in which the individual received the reported dose. If more than one license is involved, enter the license number for the facility or activity under which most of the dose was incurred as the first number. If this is not practical, enter the license numbers in the order of original issuance.
3	Name and address of your facility as it appears on your NRC license.
4	The individual's first name, middle initial, and surname. (Address of the individual may be included, but it is not entered into the NRC records system.)
5	The name and address of the individual's employer, if it is different from the reporting licensee. (Optional; not entered into the NRC records system.)
6	The individual's social security number; if not available, enter the word "unknown."
7	The individual's date of birth.

PART II. EXTERNAL DOSE DATA

For the purpose of this form, the deep dose is defined as the dose assessed at a tissue depth of 300 or 1,000 mg/cm² (or less); the shallow dose is defined as the dose to the skin of any part of the body, and the extremities are defined as hands and forearms, feet and ankles.

Item Number 8 — If the individual was not monitored for external exposure to radiation, you are requested to check the box to the left and go to Part III.

COLUMN NUMBER

9	Specify the reporting intervals (periods of exposure) that the individual was monitored at your facility(s) pursuant to §20.202. You are requested to use annual increments up to the year of termination and increments not to exceed one quarter for the year in which the individual terminated. ANNUAL: Indicate the month and year of the beginning date of exposure when showing annual increments (e.g., May 1979) and indicate the year only for subsequent annual increments. QUARTER: For each completed quarter of the year of termination, indicate the quarter and year by date. CURRENT QUARTER: Specify the beginning and ending dates of the actual exposure period (month, day, year). Enter the following data:
10a	Unless the eyes are shielded, enter the deep dose assessed at a tissue depth of 300 mg/cm² (lens depth) or less. If the eyes are protected by shielding which has a tissue equivalent thickness of 700 mg/cm² or more, the deep dose may be assessed at 1000 mg/cm² (gonad depth) or less. Enter the total dose of record, i.e., the highest dose received at the selected depth, from all types of external radiation sources, at any location on the body except the skin and the extremities (hands and forearms, feet and ankles).
10c	For all skin areas, except that of the extremities, enter in column 10c the shallow dose of record. Record the total dose to the skin, i.e., the highest dose delivered by all radiation incident on the skin, including non-trivial doses from skin contamination, which penetrates to the depth at which the shallow dose is determined. The dose at a depth of 7 mg/cm² or less, averaged over 1 cm², is acceptable. If Column 10c is left blank, it will be assumed that the entry in 10a is applicable also for the shallow dose. Therefore, an entry for shallow dose is required only if it exceeds the deep dose.
10b & d	You are requested to enter in column 10b the contribution made by neutron radiation to the dose reported in Column 10a, and to enter in Column 10d the contribution made by beta radiation to the dose reported in Column 10c. Enter XXX if it is known that there was no exposure to radiation of the type specified in the column heading. Enter UNK if a detectable exposure is reported in 10a or 10c which could have included a beta or neutron contribution of unknown magnitude.
10 & 11	You are requested to enter m or zero (in each column of 10, or in 11) if the dose was undetectable, i.e., the radiation to which the worker's dosimeter was exposed produced a response that you considered to be statistically indistinguishable from the response caused by inherent variabilities of the dosimeter system. Note: It is sometimes required to add m (or its equivalent) to a real number; although NRC regulations do not specify a summation procedure, the NRC staff arbitrarily assigns 10 mrems to be a value of m (assuming $0.5\,L \leq 10 \leq L$, where L is the detection limit) for the purposes of statistical analyses.
11	Reporting of the extremity dose is required. You are requested to comply in the following manner: Enter the dose of record, i.e., the highest dose, averaged over any area of 1 cm², determined for the skin of the hands and forearms or feet and ankles during the reported period. It is unnecessary to specify the extremity that received the dose; doses to different extremities should not be added together. The dose is to include that delivered by all radiation incident on the skin, including non-trivial doses from skin contamination, which penetrates to the depth at which the shallow dose is determined. The dose at a depth of 7 mg/cm² or less is acceptable. If Column 11 is left blank, it will be assumed that the entry in 10c is applicable also for the extremity dose; an entry in Column 11 is required only if the shallow dose exceeded the deep dose.

PART III. INTERNAL EXPOSURES TO RADIOACTIVE MATERIAL

If you are licensed by the NRC as specified in §20.408(a), 10 CFR Part 20, and if your license requires bioassay services for workers at your facility, you are required to submit termination reports on personnel exposures to radioactive material, containing information that you have obtained in compliance with the license and recorded in compliance with §20.401. You are requested to include in each termination report information that you have obtained in compliance with §20.103(a)(3) and 20.103(c)(2) and recorded in compliance with §20.401. This part provides for the reporting of internal monitoring procedures in terms of bioassay results, dose estimates, or intake. Any one (or more) of these reporting methods may be used. The term "individual" is used below to represent the worker for whom this report is submitted. The term "exposures to radioactive material" is used in connection with these termination reports to represent the entry of radioactive material into the body.

Item Number 12 — If the individual was not monitored for exposure to radioactive material, you are requested to check the box to the left; otherwise, enter the following data:

Column Number 13 — If bioassay results are reported (Column 16), you are requested to use the following format. Summarize by year, separately listing the number of measurements which indicated quantities or concentrations that were undetectable, i.e., in the detection system used, the radionuclide present (if any) produced a response that you considered to be statistically indistinguishable from its background. In Column 13 enter each year bioassay was performed, including the year of termination. In Columns 14 through 16, use two lines for each year, as in the example shown below, the upper line for detectable results and the lower line for those undetectable. In 16a and/or 16b, upper line, enter the number (including zero) of detectable measurements followed in parenthesis by the highest verified result, if any; lower line, enter the number (including zero) of measurements indicating undetectable amounts.

INSTRUCTIONS FOR COMPLETING NRC FORM 439 (Continued)

COLUMN NUMBER

13 (Continued)

Column 13	Column 14	Column 15	Column 16		
			16a(1)	16a(2)	16b (pCi/L)
1982	U-nat	"I"	0	lung	2(1)
			2	lung	10
1983	U-nat(Th 234)	"I"	1(7)	lung	4(6)
			1	lung	8
1984	U-nat(Th 234)	"I"	2(14)	lung	12(13)
			0	lung	0

Units for the numbers in parentheses shown in Column 16b are to be specified in the heading for Column 16b. If Columns 17 or 18 are completed, notations in Column 16 are unnecessary.

If the dose commitment (50-year integrated dose) is reported, indicate in Column 13 by beginning and ending dates (month, day, year) the period during which the associated radioactive material was taken into the body.

If annual doses are reported, enter in Column 13 the calendar year over which each dose was integrated, including the first and any succeeding years of this employment or work assignment and the year following the termination date.

For entries in Column 18 (intake), specify the reporting intervals (periods of exposure) during which the individual was exposed to concentrations of radioactive material, using annual increments up to the year of termination and increments not to exceed one quarter for the year in which the individual terminated. The periods of exposure for intakes should appear as follows:
ANNUAL. Indicate the month and year of the beginning date of exposure when showing annual increments (e.g., June 1983) and indicate the year only for subsequent annual increments.
QUARTER. For each completed quarter of the year of termination, indicate the quarter and year by date.
CURRENT QUARTER. Specify the beginning and ending dates of the actual exposure period (month, day, year).
Reported intakes which include only the quantities required to be assessed in accordance with §20.103(a)(3) are acceptable.

14 — Identify the symbol used in 10 CFR Part 20, Appendix B, for the radionuclide or mixture of radionuclides for which in vivo and/or urinalysis measurements were performed (e.g., Co 60, U 235). If the measured quantity of activity for one radionuclide is also used to estimate other radionuclide quantities, identify the radionuclide actually measured in parentheses immediately after the radionuclide listed in Column 14. See the example given in the directions for Column 13 where U-nat(Th 234) is entered in Column 14 indicating that the uranium lung burden was determined from measurements of Th 234 photons.

15 — Enter the form, S for soluble or I for insoluble, of the radionuclide to which the worker was exposed. If unknown, use quotes around the letter, thus indicating which concentration value in Part 20, Appendix B, Table I, Column 1, was assumed to apply.

16, 17 & 18 — These columns allow for the reporting of the results of the internal monitoring procedures in terms of bioassay results, or dose estimates, or intake. <u>You may use one or more of these methods.</u>

16a(1) & a(2) — For each year during which in vivo measurements were performed, as shown in Column 13, enter in Column 16a(1) the number of detectable measurements followed by the highest verified result (in nanocuries) in parentheses. On the next line in this column, enter the number of measurements that indicated undetectable amounts. Specify in Column 16a(2) the organ in which the indicated radionuclide was found. See the example given in the directions for Column 13.

16b — First, enter the gravimetric or radiometric unit in which the urinalysis results are reported (e.g., micrograms per liter, nanocuries per liter) in the blank space of the heading for Column 16b. In Column 16b, for each year during which urinalyses were performed, enter the number of detectable results followed by the highest numerical value of the concentration in urine of the radionuclide listed in Column 14 for the year specified in Column 13. On the next line in this column, enter the number of measurements indicating undetectable amounts. See the example given in the directions for Column 13.

17a, b, & c — Specify in Column 17c the organ or tissue receiving doses estimated in Column 17a or 17b. (Note that it is not necessary to provide both the committed and annual doses.) For Columns 17a and 17b you are requested to follow the procedures below; if any alternative procedures are used, describe them on the back of this form. In 17a, enter the dose integrated from t_o to 50 years, where t_o is the beginning date shown in Column 13. In 17b enter the dose integrated over each calendar year shown for this purpose in Column 13. Include the first and any succeeding years of this employment or work assignment and the year following the termination date. Base dose estimates on the quantity (as a minimum) of the radionuclide, Column 14, taken into the body at your facility(s) during this employment or work-assignment period.

18 — Reporting of radionuclide intakes, as determined by air sampling, is not required by 10 CFR 20.408. However, should this option be chosen, indicate the time-weighted concentrations of radioactive material (i.e., MPC-hours) to which the individual was exposed during the time periods indicated in Column 13. Refer to the last paragraph of the instructions for Column 13 for the time intervals to be used. Complete Columns 13, 14, and 15 for each entry in Column 18.

Item number 19 — Any bioassay results that cannot be reported as described above should be entered here.

Item number 20 — If you wish to send a copy of this report to the terminating individual to satisfy the notification requirements of 10 CFR 19.13, check the "Yes" box.

PRIVACY ACT STATEMENT

Pursuant to 5 U.S.C. 552a(e)(3), enacted into law by section 3 of the Privacy Act of 1974 (Public Law 93-579), the following statement is furnished to individuals and persons who supply information to the Nuclear Regulatory Commission on NRC Form 439. This information is maintained in a system of records designated as NRC-27 and described at 40 Federal Register 45344 (October 1, 1975).

1. AUTHORITY. Sections 53, 63, 65, 81, 103, 104, 161(b), and 161(o) of the Atomic Energy Act of 1954, as amended (42 U.S.C. 2073, 2093, 2095, 2111, 2133, 2134, 2201(b), and 2201(o)). The authority for soliciting the social security number is 10 CFR Part 20.

2. PRINCIPAL PURPOSE(S). The information is used by the NRC in its evaluation of the risk of radiation exposure associated with the licensed activity and in exercising its statutory responsibility to monitor and regulate the safety and health practices of its licensees. The data permit a meaningful comparison of both current and long-term exposure experience among types of licensees and among licensees within each type. Data on your exposure to radiation are available to you upon request.

3. ROUTINE USES. The information may be used to provide data to other Federal and State agencies involved in monitoring and/or evaluating radiation exposure received by individuals employed as radiation workers on a permanent or temporary basis and exposure received by monitored visitors. The information may also be disclosed to an appropriate Federal, State, or local agency in the event the information indicates a violation or potential violation of law and in the course of an administrative or judicial proceeding.

4. WHETHER DISCLOSURE IS MANDATORY OR VOLUNTARY AND EFFECT OF NOT PROVIDING INFORMATION ON INDIVIDUAL OR PERSON. It is voluntary that you furnish the requested information, including name, date of birth, and social security number. The social security number is used to assure that NRC has an accurate identifier not subject to the coincidence of similar names or birth dates among the large number of persons on whom data is maintained. Please note, however, that the licensee must file a termination report containing certain required information, such as social security number, for each individual whose employment or work assignment has terminated and for whom personnel monitoring was required under 10 CFR 20.202. Failure of the licensee to provide the information under 10 CFR §20.202 and 20.408 may subject the licensee to enforcement action under 10 CFR 20.601.

5. SYSTEM MANAGER(S) AND ADDRESS. Director, Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555

NRC FORM 438
(4-85)
10 CFR §20.408

U.S. NUCLEAR REGULATORY COMMISSION

REPORT OF TERMINATING INDIVIDUAL'S OCCUPATIONAL EXPOSURE

SEE THE ATTACHED INSTRUCTIONS

1. DATE OF REPORT

2. NRC LICENSE NUMBER(S)

PART I. LICENSEE AND INDIVIDUAL IDENTIFICATION DATA

3. NAME AND ADDRESS OF REPORTING LICENSEE

4. NAME OF INDIVIDUAL *(first, middle initial, last)* AND ADDRESS *(optional)*

5. NAME AND ADDRESS OF EMPLOYER, IF DIFFERENT FROM ABOVE *(Optional)*

6. SOCIAL SECURITY NUMBER

7. DATE OF BIRTH — MONTH | DAY | YEAR

PART II. EXTERNAL DOSE DATA

☐ 8. PERSONNEL MONITORING FOR EXTERNAL EXPOSURE TO RADIATION WAS NOT PROVIDED

9. PERIOD(S) OF EXPOSURE *(earliest date first)*	10. WHOLE BODY DOSE *(rems)* DEEP a. TOTAL	DEEP b. NEUTRON	SHALLOW *(skin)* c. TOTAL	SHALLOW *(skin)* d. BETA	11. EXTREMITY DOSE *(rems)* SHALLOW *(skin)*

PART III. INTERNAL EXPOSURES TO RADIOACTIVE MATERIAL

☐ 12. PERSONNEL MONITORING FOR EXPOSURE TO RADIOACTIVE MATERIAL WAS NOT PROVIDED

13. PERIOD(S) OF EXPOSURE *(earliest date first)*	14. NUCLIDE	15. FORM (S, I)	16. BIOASSAY RESULTS a. IN VIVO (nCi) (1) BURDEN	a. IN VIVO (nCi) (2) ORGAN	b. URINALYSIS RESULTS (L)	17. DOSE ESTIMATES *(rems)* a. COMMITTED DOSE	b. ANNUAL DOSE	c. ORGAN	18. INTAKE (MPC-Hrs.)

OTHER BIOASSAY RESULTS

20. IF THIS REPORT IS BEING USED TO SATISFY THE NOTIFICATION REQUIREMENTS OF 10 CFR 19.13, CHECK THE FOLLOWING BOX

☐ YES *(This report is furnished to you under the provision of the Nuclear Regulatory Commission's regulation 10 CFR Part 19. You should preserve this report for further reference.)*

INSTRUCTIONS FOR COMPLETING NRC FORM 439,
Report of Terminating Individual's Occupational Exposure

If you are licensed by the U.S. Nuclear Regulatory Commission (NRC) as specified in §20.408(a), 10 CFR Part 20, you are required to submit termination radiation exposure reports on certain individuals to the Director, Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission, Washington, DC 20555. This information is to be taken from dose records that must be maintained under §20.401 for individuals likely to receive exposure to radiation that exceeds a certain percentage of the NRC dose standards for the whole body, skin or extremities—25% for workers of age 18 years or more, 5% for workers younger than 18. The term "individual" is used below to represent the worker for whom this report is submitted. The term "dose" as used in Form 439 and in these instructions refers to the dose in rems as defined in §20.4(a) and subsequently designated "dose equivalent" in ICRU Report 11 (1968). The time to be covered by this report is that period of employment, or work assignment in your facility(s), which ended with the most recent termination and was not interrupted by any previous termination during which personnel monitoring was required by §20.202(a) and/or bioassays were required by your license. "Termination" is defined in §20.3(a)(19). Parts II and III of this form reflect regulatory requirements as well as requests intended to standardize reporting methods; requests are clearly identified as such.

PART I. LICENSEE AND INDIVIDUAL IDENTIFICATION DATA

This part identifies the licensee submitting the report and the terminating individual. It must be completed even if only one of the remaining Parts of this form is applicable. Enter the following data:

ITEM NUMBER

1 Date that the report was prepared.

2 Current NRC license number assigned to the facility(s) in which the individual received the reported dose. If more than one license is involved, enter the license number for the facility or activity under which most of the dose was incurred as the first number. If this is not practical, enter the license numbers in the order of original issuance.

3 Name and address of your facility as it appears on your NRC license.

4 The individual's first name, middle initial, and surname. (Address of the individual may be included, but it is not entered into the NRC records system.)

5 The name and address of the individual's employer, if it is different from the reporting licensee. (Optional; not entered into the NRC records system.)

6 The individual's social security number; if not available, enter the word "unknown."

7 The individual's date of birth.

PART II. EXTERNAL DOSE DATA

For the purpose of this form, the deep dose is defined as the dose assessed at a tissue depth of 300 or 1,000 mg/cm² (or less), the shallow dose is defined as the dose to the skin of any part of the body, and the extremities are defined as hands and forearms, feet and ankles.

Item Number 8 If the individual was not monitored for external exposure to radiation, you are requested to check the box to the left and go to Part III.

COLUMN NUMBER

9 Specify the reporting intervals (periods of exposure) that the individual was monitored at your facility(s) pursuant to §20.202. You are requested to use annual increments up to the year of termination and increments not to exceed one quarter for the year in which the individual terminated.
ANNUAL: Indicate the month and year of the beginning date of exposure when showing annual increments (e.g., May 1979) and indicate the year only for subsequent annual increments.
QUARTER: For each completed quarter of the year of termination, indicate the quarter and year by date.
CURRENT QUARTER: Specify the beginning and ending dates of the actual exposure period (month, day, year).
Enter the following data:

10a Unless the eyes are shielded, enter the deep dose assessed at a tissue depth of 300 mg/cm² (lens depth) or less. If the eyes are protected by shielding which has a tissue equivalent thickness of 700 mg/cm² or more, the deep dose may be assessed at 1000 mg/cm² (gonad depth) or less. Enter the total dose of record, i.e., the highest dose received at the selected depth, from all types of external radiation sources, at any location on the body except the skin and the extremities (hands and forearms, feet and ankles).

10c For all skin areas, except that of the extremities, enter in column 10c the shallow dose of record. Record the total dose to the skin, i.e., the highest dose delivered by all radiation incident on the skin, including non-trivial doses from skin contamination, which penetrates to the depth at which the shallow dose is determined. The dose at a depth of 7 mg/cm² or less, averaged over 1 cm², is acceptable. If Column 10c is left blank, it will be assumed that the entry in 10a is applicable also for the shallow dose. Therefore, an entry for shallow dose is required only if it exceeds the deep dose.

10b & d You are requested to enter in column 10b the contribution made by neutron radiation to the dose reported in Column 10a, and to enter in Column 10d the contribution made by beta radiation to the dose reported in Column 10c. Enter XXX if it is known that there was no exposure to radiation of the type specified in the column heading. Enter UNK if a detectable exposure is reported in 10a or 10c which could have included a beta or neutron contribution of unknown magnitude.

10 & 11 You are requested to enter m or zero (in each column of 10, or in 11) if the dose was undetectable; i.e., the radiation to which the worker's dosimeter was exposed produced a response that you considered to be statistically indistinguishable from the response caused by inherent variabilities of the dosimeter system. Note: It is sometimes required to add m (or its equivalent) to a real number, although NRC regulations do not specify a summation procedure; the NRC staff arbitrarily assigns 10 mrems to be a value of m (assuming $0.5\ L \leq 10 \leq L$, where L is the detection limit) for the purposes of statistical analyses.

11 Reporting of the extremity dose is required. You are requested to comply in the following manner. Enter the dose of record, i.e., the highest dose, averaged over any area of 1 cm², determined for the skin of the hands and forearms or feet and ankles during the reported period. It is unnecessary to specify the extremity that received the dose; doses to different extremities should not be added together. The dose is to include that delivered by all radiation incident on the skin, including non-trivial doses from skin contamination, which penetrates to the depth at which the shallow dose is determined. The dose at a depth of 7 mg/cm² or less is acceptable. If Column 11 is left blank, it will be assumed that the entry in 10c is applicable also for the extremity dose; an entry in Column 11 is required only if the shallow dose exceeded the deep dose.

PART III. INTERNAL EXPOSURES TO RADIOACTIVE MATERIAL

If you are licensed by the NRC as specified in §20.408(a), 10 CFR Part 20, and if your license requires bioassay services for workers at your facility, you are required to submit termination reports on personnel exposures to radioactive material, containing information that you have obtained in compliance with the license and recorded in compliance with §20.401. You are requested to include in each termination report information that you have obtained in compliance with §20.103(a)(3) and 20.103(c)(2) and recorded in compliance with §20.401. This part provides for the reporting of internal monitoring procedures in terms of bioassay results, dose estimates, or intake. Any one (or more) of these reporting methods may be used. The term "individual" is used below to represent the worker for whom this report is submitted. The term "exposures to radioactive material" is used in connection with these termination reports to represent the entry of radioactive material into the body.

Item Number 12 If the individual was not monitored for exposure to radioactive material, you are requested to check the box to the left; otherwise, enter the following data.

Column Number 13 If bioassay results are reported (Column 16), you are requested to use the following format. Summarize by year, separately listing the number of measurements which indicated quantities or concentrations that were undetectable, i.e., in the detection system used, the radionuclide present (if any) produced a response that you considered to be statistically indistinguishable from its background. In Column 13 enter each year bioassay was performed, including the year of termination. In Columns 14 through 16, use two lines for each year, as in the example shown below, the upper line for detectable results and the lower line for those undetectable. In 16a and/or 16b: upper line, enter the number (including zero) of detectable measurements followed in parenthesis by the highest verified result, if any; lower line, enter the number (including zero) of measurements indicating undetectable amounts.

INSTRUCTIONS FOR COMPLETING NRC FORM 439 (Continued)

COLUMN NUMBER

13 (Continued)

Column 13	Column 14	Column 15	Column 16 16a(1)	16a(2)	16b (pCi/
1982	U-nat	"I"	0	lung	2(1)
			2	lung	10
1983	U-nat(Th 234)	"I"	1(7)	lung	4(6)
			1	lung	8
1984	U-nat(Th 234)	"I"	2(14)	lung	12(13)
			0	lung	0

Units for the numbers in parentheses shown in Column 16b are to be specified in the heading for Column 16b. If Columns 17 or 18 are completed, notations in Column 16 are unnecessary.

If the dose commitment (50-year integrated dose) is reported, indicate in Column 13 by beginning and ending dates (month, day, year) the period during which the associated radioactive material was taken into the body.

If annual doses are reported, enter in Column 13 the calendar year over which each dose was integrated, including the first and any succeeding years of this employment or work assignment and the year following the termination date.

For entries in Column 18 (intake), specify the reporting intervals (periods of exposure) during which the individual was exposed to concentrations of radioactive material, using annual increments up to the year of termination and increments not to exceed one quarter for the year in which the individual terminated. The periods of exposure for intakes should appear as follows:
ANNUAL. Indicate the month and year of the beginning date of exposure when showing annual increments (e.g., June 1983) and indicate the year only for subsequent annual increments.
QUARTER. For each completed quarter of the year of termination, indicate the quarter and year by date.
CURRENT QUARTER. Specify the beginning and ending dates of the actual exposure period (month, day, year).
Reported intakes which include only the quantities required to be assessed in accordance with §20.103(a)(3) are acceptable.

14 Identify the symbol used in 10 CFR Part 20, Appendix B, for the radionuclide or mixture of radionuclides for which in vivo and/or urinalysis measurements were performed (e.g., Co 60, U 235). If the measured quantity of activity for one radionuclide is also used to estimate other radionuclide quantities, identify the radionuclide actually measured in parentheses immediately after the radionuclide listed in Column 14. See the example given in the directions for Column 13 where U-nat(Th 234) is entered in Column 14 indicating that the uranium lung burden was determined from measurements of Th 234 photons.

15 Enter the form, S for soluble or I for insoluble, of the radionuclide to which the worker was exposed. If unknown, use quotes around the letter, thus indicating which concentration value in Part 20, Appendix B, Table I, Column 1, was assumed to apply.

16, 17 & 18 These columns allow for the reporting of the results of the internal monitoring procedures in terms of bioassay results, or dose estimates, or intake. You may use one or more of these methods.

16a(1) & a(2) For each year during which in vivo measurements were performed, as shown in Column 13, enter in Column 16a(1) the number of detectable measurements followed by the highest verified result (in nanocuries) in parentheses. On the next line in this column, enter the number of measurements that indicated undetectable amounts. Specify in Column 16a(2) the organ in which the indicated radionuclide was found. See the example given in the directions for Column 13.

16b First, enter the gravimetric or radiometric unit in which the urinalysis results are reported (e.g., micrograms per liter, nanocuries per liter) in the blank space of the heading for Column 16b. In Column 16b, for each year during which urinalyses were performed, enter the number of detectable results followed by the highest numerical value of the concentration in urine of the radionuclide listed in Column 14 for the year specified in Column 13. On the next line in this column, enter the number of measurements indicating undetectable amounts. See the example given in the directions for Column 13.

17a, b, & c Specify in Column 17c the organ or tissue receiving doses estimated in Column 17a or 17b. (Note that it is not necessary to provide both the committed and annual doses.) For Columns 17a and 17b you are requested to follow the procedures below; if any alternative procedures are used, describe them on the back of this form. In 17a, enter the dose integrated from t_o to 50 years, where t_o is the beginning date shown in Column 13. In 17b enter the dose integrated over each calendar year shown for this purpose in Column 13. Include the first and any succeeding years of this employment or work assignment and the year following the termination date. Base dose estimates on the quantity (as a minimum) of the radionuclide, Column 14, taken into the body at your facility(s) during this employment or work-assignment period.

18 Reporting of radionuclide intakes, as determined by air sampling, is not required by 10 CFR 20.408. However, should this option be chosen, indicate the time-weighted concentrations of radioactive material (i.e., MPC-hours) to which the individual was exposed during the time periods indicated in Column 13. Refer to the last paragraph of the instructions for Column 13 for the time intervals to be used. Complete Columns 13, 14, and 15 for each entry in Column 18.

Item number 19 Any bioassay results that cannot be reported as described above should be entered here.

Item number 20 If you wish to send a copy of this report to the terminating individual to satisfy the notification requirements of 10 CFR 19.13, check the "Yes" box.

PRIVACY ACT STATEMENT

Pursuant to 5 U.S.C. 552a(e)(3), enacted into law by section 3 of the Privacy Act of 1974 (Public Law 93-579), the following statement is furnished to individuals and persons who supply information to the Nuclear Regulatory Commission on NRC Form 439. This information is maintained in a system of records designated as NRC-27 and described at 40 Federal Register 45344 (October 1, 1975).

1. AUTHORITY. Sections 53, 63, 65, 81, 103, 104, 161(b), and 161(o) of the Atomic Energy Act of 1954, as amended (42 U.S.C. 2073, 2093, 2095, 2111, 2133, 2134, 2201(b), and 2201(o)). The authority for soliciting the social security number is 10 CFR Part 20.

2. PRINCIPAL PURPOSE(S). The information is used by the NRC in its evaluation of the risk of radiation exposure associated with the licensed activity and in exercising its statutory responsibility to monitor and regulate the safety and health practices of its licensees. The data permit a meaningful comparison of both current and long-term exposure experience among types of licensees and among licensees within each type. Data on your exposure to radiation are available to you upon request.

3. ROUTINE USES. The information may be used to provide data to other Federal and State agencies involved in monitoring and/or evaluating radiation exposure received by individuals employed as radiation workers on a permanent or temporary basis and exposure received by monitored visitors. The information may also be disclosed to an appropriate Federal, State, or local agency in the event the information indicates a violation or potential violation of law and in the course of an administrative or judicial proceeding.

4. WHETHER DISCLOSURE IS MANDATORY OR VOLUNTARY AND EFFECT OF NOT PROVIDING INFORMATION ON INDIVIDUAL OR PERSON. It is voluntary that you furnish the requested information, including name, date of birth, and social security number. The social security number is used to assure that NRC has an accurate identifier not subject to the coincidence of similar names or birth dates among the large number of persons on whom data is maintained. Please note, however, that the licensee must file a termination report containing certain required information, such as social security number, for each individual whose employment or work assignment terminated and for whom personnel monitoring was required under 10 CFR 20.202. Failure of the licensee to provide the information under 10 CFR §20.202 and 20.408 n subject the licensee to enforcement action under 10 CFR 20.601.

5. SYSTEM MANAGER(S) AND ADDRESS. Director, Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555

NRC FORM 439
(4-85)
10 CFR §20.408

U.S. NUCLEAR REGULATORY COMMISSION

REPORT OF TERMINATING INDIVIDUAL'S OCCUPATIONAL EXPOSURE

SEE THE ATTACHED INSTRUCTIONS

1. DATE OF REPORT

2. NRC LICENSE NUMBER(S)

PART I. LICENSEE AND INDIVIDUAL IDENTIFICATION DATA

3. NAME AND ADDRESS OF REPORTING LICENSEE

4. NAME OF INDIVIDUAL *(first, middle initial, last)* AND ADDRESS *(optional)*

5. NAME AND ADDRESS OF EMPLOYER, IF DIFFERENT FROM ABOVE *(Optional)*

6. SOCIAL SECURITY NUMBER

7. DATE OF BIRTH — MONTH | DAY | YEAR

PART II. EXTERNAL DOSE DATA

☐ 8. PERSONNEL MONITORING FOR EXTERNAL EXPOSURE TO RADIATION WAS NOT PROVIDED

9. PERIOD(S) OF EXPOSURE *(earliest date first)*	10. WHOLE BODY DOSE *(rems)*				11. EXTREMITY DOSE *(rems)*
	DEEP		SHALLOW *(skin)*		
	a. TOTAL	b. NEUTRON	c. TOTAL	d. BETA	SHALLOW *(skin)*

PART III. INTERNAL EXPOSURES TO RADIOACTIVE MATERIAL

☐ 12. PERSONNEL MONITORING FOR EXPOSURE TO RADIOACTIVE MATERIAL WAS NOT PROVIDED

13. PERIOD(S) OF EXPOSURE *(earliest date first)*	14. NUCLIDE	15. FORM (S, I)	16. BIOASSAY RESULTS			17. DOSE ESTIMATES *(rems)*			18. INTAKE (MPC Hrs.)
			a. IN VIVO (nCi)		b. URINALYSIS RESULTS (L)	a. COMMITTED DOSE	b. ANNUAL DOSE	c. ORGAN	
			(1) BURDEN	(2) ORGAN					

OTHER BIOASSAY RESULTS

20. IF THIS REPORT IS BEING USED TO SATISFY THE NOTIFICATION REQUIREMENTS OF 10 CFR 19.13, CHECK THE FOLLOWING BOX.

☐ YES *(This report is furnished to you under the provision of the Nuclear Regulatory Commission's regulation 10 CFR Part 19. You should preserve this report for further reference.)*

INSTRUCTIONS FOR COMPLETING NRC FORM 439,
Report of Terminating Individual's Occupational Exposure

If you are licensed by the U.S. Nuclear Regulatory Commission (NRC) as specified in §20.408(a), 10 CFR Part 20, you are required to submit termination radiation exposure reports on certain individuals to the Director, Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission, Washington, DC 20555. This information is to be taken from dose records that must be maintained under §20.401 for individuals likely to receive exposure to radiation that exceeds a certain percentage of the NRC dose standards for the whole body, skin or extremities—25% for workers of age 18 years or more, 5% for workers younger than 18. The term "individual" is used below to represent the worker for whom this report is submitted. The term "dose" as used in Form 439 and in these instructions refers to the dose in rems as defined in §20.4(a) and subsequently designated "dose equivalent" in ICRU Report II (1968). The time to be covered by this report is that period of employment, or work assignment in your facility(s), which ended with the most recent termination and was not interrupted by any previous termination during which personnel monitoring was required by §20.202(a) and/or bioassays were required by your license. "Termination" is defined in §20.3(a)(19). Parts II and III of this form reflect regulatory requirements as well as requests intended to standardize reporting methods; requests are clearly identified as such.

PART I LICENSEE AND INDIVIDUAL IDENTIFICATION DATA

This part identifies the licensee submitting the report and the terminating individual. It must be completed even if only one of the remaining Parts of this form is applicable. Enter the following data:

ITEM NUMBER

1 Date that the report was prepared.

2 Current NRC license number assigned to the facility(s) in which the individual received the reported dose. If more than one license is involved, enter the license number for the facility or activity under which most of the dose was incurred as the first number. If this is not practical, enter the license numbers in the order of original issuance.

3 Name and address of your facility as it appears on your NRC license.

4 The individual's first name, middle initial, and surname. (Address of the individual may be included, but it is not entered into the NRC records system.)

5 The name and address of the individual's employer, if it is different from the reporting licensee. (Optional, not entered into the NRC records system.)

6 The individual's social security number; if not available, enter the word "unknown."

7 The individual's date of birth.

PART II EXTERNAL DOSE DATA

For the purpose of this form, the deep dose is defined as the dose assessed at a tissue depth of 300 or 1,000 mg/cm² (or less); the shallow dose is defined as the dose to the skin of any part of the body, and the extremities are defined as hands and forearms, feet and ankles.

Item Number 8 If the individual was not monitored for external exposure to radiation, you are requested to check the box to the left and go to Part III.

COLUMN NUMBER

9 Specify the reporting intervals (periods of exposure) that the individual was monitored at your facility(s) pursuant to §20.202. You are requested to use annual increments up to the year of termination and increments not to exceed one quarter for the year in which the individual terminated.
ANNUAL: Indicate the month and year of the beginning date of exposure when showing annual increments (e.g., May 1979) and indicate the year only for subsequent annual increments.
QUARTER: For each completed quarter of the year of termination, indicate the quarter and year by date.
CURRENT QUARTER: Specify the beginning and ending dates of the actual exposure period (month, day, year).
Enter the following data:

10a Unless the eyes are shielded, enter the deep dose assessed at a tissue depth of 300 mg/cm² (lens depth) or less. If the eyes are protected by shielding which has a tissue equivalent thickness of 700 mg/cm² or more, the deep dose may be assessed at 1000 mg/cm² (gonad depth) or less. Enter the total dose of record, i.e., the highest dose received at the selected depth, from all types of external radiation sources, at any location on the body except the skin and the extremities (hands and forearms, feet and ankles).

10c For all skin areas, except that of the extremities, enter in column 10c the shallow dose of record. Record the total dose to the skin, i.e., the highest dose delivered by all radiation incident on the skin, including non-trivial doses from skin contamination, which penetrates to the depth at which the shallow dose is determined. The dose at a depth of 7 mg/cm² or less, averaged over 1 cm², is acceptable. If Column 10c is left blank, it will be assumed that the entry in 10a is applicable also for the shallow dose. Therefore, an entry for shallow dose is required only if it exceeds the deep dose.

10b & d You are requested to enter in column 10b the contribution made by neutron radiation to the dose reported in Column 10a, and to enter in Column 10d the contribution made by beta radiation to the dose reported in Column 10c. Enter XXX if it is known that there was no exposure to radiation of the type specified in the column heading. Enter UNK if a detectable exposure is reported in 10a or 10c which could have included a beta or neutron contribution of unknown magnitude.

10 & 11 You are requested to enter m or zero (in each column of 10, or in 11) if the dose was undetectable, i.e., the radiation to which the worker's dosimeter was exposed produced a response that you considered to be statistically indistinguishable from the response caused by inherent variabilities of the dosimeter system. Note: It is sometimes required to add m (or its equivalent) to a real number; although NRC regulations do not specify a summation procedure, the NRC staff arbitrarily assigns 10 mrems to be a value of m (assuming $0.5 L \leq 10 \leq L$, where L is the detection limit) for the purposes of statistical analyses.

11 Reporting of the extremity dose is required. You are requested to comply in the following manner. Enter the dose of record, i.e., the highest dose, averaged over any area of 1 cm², determined for the skin of the hands and forearms or feet and ankles during the reported period. It is unnecessary to specify the extremity that received the dose; doses to different extremities should not be added together. The dose is to include that delivered by all radiation incident on the skin, including non-trivial doses from skin contamination, which penetrates to the depth at which the shallow dose is determined. The dose at a depth of 7 mg/cm² or less is acceptable. If Column 11 is left blank, it will be assumed that the entry in 10c is applicable also for the extremity dose; an entry in Column 11 is required only if the shallow dose exceeded the deep dose.

PART III INTERNAL EXPOSURES TO RADIOACTIVE MATERIAL

If you are licensed by the NRC as specified in §20.408(a), 10 CFR Part 20, and if your license requires bioassay services for workers at your facility, you are required to submit termination reports on personnel exposures to radioactive material, containing information that you have obtained in compliance with the license and recorded in compliance with §20.401. You are requested to include in each termination report information that you have obtained in compliance with §20.103(a)(3) and 20.103(c)(2) and recorded in compliance with §20.401. This part provides for the reporting of internal monitoring procedures in terms of bioassay results, dose estimates, or intake. Any one (or more) of these reporting methods may be used. The term "individual" is used below to represent the worker for whom this report is submitted. The term "exposures to radioactive material" is used in connection with these termination reports to represent the entry of radioactive material into the body.

Item Number 12 If the individual was not monitored for exposure to radioactive material, you are requested to check the box to the left; otherwise, enter the following data.

Column Number 13 If bioassay results are reported (Column 16), you are requested to use the following format. Summarize by year, separately listing the number of measurements which indicated quantities or concentrations that were undetectable, i.e., in the detection system used, the radionuclide present (if any) produced a response that you considered to be statistically indistinguishable from its background. In Column 13 enter each year bioassay was performed, including the year of termination. In Columns 14 through 16, use two lines for each year, as in the example shown below, the upper line for detectable results and the lower line for those undetectable. In 16a and/or 16b: upper line, enter the number (including zero) of detectable measurements followed in parenthesis by the highest verified result, if any; lower line, enter the number (including zero) of measurements indicating undetectable amounts.

INSTRUCTIONS FOR COMPLETING NRC FORM 439 (Continued)

COLUMN NUMBER

13 (Continued)

Column 13	Column 14	Column 15	Column 16		
			16a(1)	16a(2)	16b (pCi/L)
1982	U-nat	"I"	0	lung	2(1)
			2	lung	10
1983	U-nat(Th 234)	"I"	1(7)	lung	4(6)
			1	lung	8
1984	U-nat(Th 234)	"I"	2(14)	lung	12(13)
			0	lung	0

Units for the numbers in parentheses shown in Column 16b are to be specified in the heading for Column 16b. If Columns 17 or 18 are completed, notations in Column 16 are unnecessary.

If the dose commitment (50-year integrated dose) is reported, indicate in Column 13 by beginning and ending dates (month, day, year) the period during which the associated radioactive material was taken into the body.

If annual doses are reported, enter in Column 13 the calendar year over which each dose was integrated, including the first and any succeeding years of this employment or work assignment and the year following the termination date.

For entries in Column 18 (intake), specify the reporting intervals (periods of exposure) during which the individual was exposed to concentrations of radioactive material, using annual increments up to the year of termination and increments not to exceed one quarter for the year in which the individual terminated. The periods of exposure for intakes should appear as follows:
ANNUAL. Indicate the month and year of the beginning date of exposure when showing annual increments (e.g., June 1983) and indicate the year only for subsequent annual increments.
QUARTER. For each completed quarter of the year of termination, indicate the quarter and year by date.
CURRENT QUARTER. Specify the beginning and ending dates of the actual exposure period (month, day, year).
Reported intakes which include only the quantities required to be assessed in accordance with §20.103(a)(3) are acceptable.

14 — Identify the symbol used in 10 CFR Part 20, Appendix B, for the radionuclide or mixture of radionuclides for which in vivo and/or urinalysis measurements were performed (e.g., Co 60, U 235). If the measured quantity of activity for one radionuclide is also used to estimate other radionuclide quantities, identify the radionuclide actually measured in parentheses immediately after the radionuclide listed in Column 14. See the example given in the directions for Column 13 where U-nat(Th 234) is entered in Column 14 indicating that the uranium lung burden was determined from measurements of Th 234 photons.

15 — Enter the form, S for soluble or I for insoluble, of the radionuclide to which the worker was exposed. If unknown, use quotes around the letter, thus indicating which concentration value in Part 20, Appendix B, Table I, Column 1, was assumed to apply.

16, 17 & 18 — These columns allow for the reporting of the results of the internal monitoring procedures in terms of bioassay results, or dose estimates, or intake. You may use one or more of these methods.

16a(1) & a(2) — For each year during which in vivo measurements were performed, as shown in Column 13, enter in Column 16a(1) the number of detectable measurements followed by the highest verified result (in nanocuries) in parentheses. On the next line in this column, enter the number of measurements that indicated undetectable amounts. Specify in Column 16a(2) the organ in which the indicated radionuclide was found. See the example given in the directions for Column 13.

16b — First, enter the gravimetric or radiometric unit in which the urinalysis results are reported (e.g., micrograms per liter, nanocuries per liter) in the blank space of the heading for Column 16b. In Column 16b, for each year during which urinalyses were performed, enter the number of detectable results followed by the highest numerical value of the concentration in urine of the radionuclide listed in Column 14 for the year specified in Column 13. On the next line in this column, enter the number of measurements indicating undetectable amounts. See the example given in the directions for Column 13.

17a, b, & c — Specify in Column 17c the organ or tissue receiving doses estimated in Column 17a or 17b. (Note that it is not necessary to provide both the committed and annual doses.) For Columns 17a and 17b you are requested to follow the procedures below; if any alternative procedures are used, describe them on the back of this form. In 17a, enter the dose integrated from t_i to 50 years, where t_i is the beginning date shown in Column 13. In 17b enter the dose integrated over each calendar year shown for this purpose in Column 13. Include the first and any succeeding years of this employment or work assignment and the year following the termination date. Base dose estimates on the quantity (as a minimum) of the radionuclide, Column 14, taken into the body at your facility(s) during this employment or work-assignment period.

18 — Reporting of radionuclide intakes, as determined by air sampling, is not required by 10 CFR 20.408. However, should this option be chosen, indicate the time-weighted concentrations of radioactive material (i.e., MPC-hours) to which the individual was exposed during the time periods indicated in Column 13. Refer to the last paragraph of the instructions for Column 13 for the time intervals to be used. Complete Columns 13, 14, and 15 for each entry in Column 18.

Item number 19 — Any bioassay results that cannot be reported as described above should be entered here.

Item number 20 — If you wish to send a copy of this report to the terminating individual to satisfy the notification requirements of 10 CFR 19.13, check the "Yes" box.

PRIVACY ACT STATEMENT

Pursuant to 5 U.S.C. 552a(e)(3), enacted into law by section 3 of the Privacy Act of 1974 (Public Law 93-579), the following statement is furnished to individuals and persons who supply information to the Nuclear Regulatory Commission on NRC Form 439. This information is maintained in a system of records designated as NRC-27 and described at 40 Federal Register 45344 (October 1, 1975).

1. AUTHORITY: Sections 53, 63, 65, 81, 103, 104, 161(b), and 161(o) of the Atomic Energy Act of 1954, as amended (42 U.S.C. 2073, 2093, 2095, 2111, 2133, 2134, 2201(b), and 2201(o)). The authority for soliciting the social security number is 10 CFR Part 20.

2. PRINCIPAL PURPOSE(S): The information is used by the NRC in its evaluation of the risk of radiation exposure associated with the licensed activity and in exercising its statutory responsibility to monitor and regulate the safety and health practices of its licensees. The data permit a meaningful comparison of both current and long-term exposure experience among types of licensees and among licensees within each type. Data on your exposure to radiation are available to you upon request.

3. ROUTINE USES: The information may be used to provide data to other Federal and State agencies involved in monitoring and/or evaluating radiation exposure received by individuals employed as radiation workers on a permanent or temporary basis and exposure received by monitored visitors. The information may also be disclosed to an appropriate Federal, State, or local agency in the event the information indicates a violation or potential violation of law and in the course of an administrative or judicial proceeding.

4. WHETHER DISCLOSURE IS MANDATORY OR VOLUNTARY AND EFFECT OF NOT PROVIDING INFORMATION ON INDIVIDUAL OR PERSON: It is voluntary that you furnish the requested information, including name, date of birth, and social security number. The social security number is used to assure that NRC has an accurate identifier not subject to the coincidence of similar names or birth dates among the large number of persons on whom data is maintained. Please note, however, that the licensee must file a termination report containing certain required information, such as social security number, for each individual whose employment or work assignment has terminated and for whom personnel monitoring was required under 10 CFR 20.202. Failure of the licensee to provide the information under 10 CFR §20.202 and 20.408 may subject the licensee to enforcement action under 10 CFR 20.601.

5. SYSTEM MANAGER(S) AND ADDRESS: Director, Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555

LIST OF RECENTLY ISSUED GENERIC LETTERS

GENERIC LETTER NO.	SUBJECT	DATE
84-20	Scheduling Guidance for Licensee Submittals of Reloads that Involve Unreviewed Safety Questions	8/20/84
84-21	Long Term Low Power Operation in PWR's	10/16/84
84-22	Not used	
84-23	Reactor Vessel Water Level Instrumentation in BWRs	10/26/84
84-24	Clarification of Compliance to 10 CFR 50.49 Environmental Qualification of Electrical Equipment Important to Safety for Nuclear Power Plants	12/27/84
85-01	Fire Protection Policy Steering Committee Report	1/9/85
85-02	Staff Recommended Actions Stemming From NRC Integrated Program for the Resolution of Unresolved Safety Issues Regarding Steam Generator Tube Integrity	4/15/85
85-03	Clarification of Equivalent Control Capacity For Standby Liquid Control Systems	1/28/85
85-04	Operator Licensing Examinations	1/29/85
85-05	Inadvertent Boron Dilution Events	1/31/85
85-06	Quality Assurance Guidance for ATWS Equipment that is not Safety-Related	4/16/85
85-07	Implementation of Integrated Schedules for Plant Modifications	5/02/85
85-08	10 CFR 20.408 Termination Reports - Format	5/23/85
85-09	Technical Specifications for Generic Letter 83-28, Item 4.3	5/23/85
85-10	Technical Specifications for Generic Letter 83-28, Items 4.3 and 4.4	5/23/85

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JUN 2 6 1985

TO ALL LICENSEES FOR OPERATING REACTORS

Gentlemen:

SUBJECT: COMPLETION OF PHASE II OF "CONTROL OF HEAVY LOADS AT NUCLEAR POWER PLANTS" NUREG-0612. (GENERIC LETTER 85-11)

On December 22, 1980, NRC issued a generic letter (unnumbered) which was supplemented February 3, 1981 (Generic Letter 81-07) regarding NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants". This letter requested that you implement certain interim actions and provide the NRC information related to heavy loads at your facilities. Your submittals were requested in two parts; a six month response (Phase I) and a nine month response (Phase II).

All licensees have completed the requirement to perform a review and submit a Phase I and a Phase II report. Based on the improvements in heavy loads handling obtained from implementation of NUREG-0612 (Phase I), further action is not required to reduce the risks associated with the handling of heavy loads (See enclosed NUREG-0612 Phase II). Therefore, a detailed Phase II review of heavy loads is not necessary and Phase II is considered completed. However, while not a requirement, we encourage the implementation of any actions you identified in Phase II regarding the handling of heavy loads that you consider appropriate.

For each plant which has a license condition requiring commitments acceptable to the NRC regarding Phase II, an application for license amendment may be submitted to the NRC to delete the license condition citing this letter as the basis. If you have any questions, contact your Project Manager or Don Neighbors (301) 492-4837.

Sincerely,

Hugh L. Thompson, Jr., Director
Division of Licensing

Enclosure:
As Stated

8506270216 14 pp.

NUREG-0612, "CONTROL OF HEAVY LOADS AT NUCLEAR POWER PLANTS"

RESOLUTION OF PHASE II

Generic Technical Activity A-36 was established to systematically examine the staff's licensing criteria, adequacy of measures in effect at operating plants and recommend necessary changes to assure the safe handling of heavy loads. The task involved review of licensee information, evaluation of historical data, performance of accident analyses and criticality calculations, development of guidelines for operating plants, and review of licensing criteria. The review indicated that the major causes of load handling accidents include operator errors, rigging failures, lack of adequate inspection and inadequate procedures. The results of the review culminated in the issuance of NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants" in July 1980. NUREG-0612 described a resolution of Task A-36.

NUREG-0612 presents an overall philosophy that provides a defense-in-depth approach for controlling the handling of heavy loads. The approach is directed to preventing load drops. The following summarizes this defense-in-depth approach:

1. Assure that there is a well designed handling system.

2. Provide sufficient operator training, load handling instructions, and equipment inspection to assure reliable operation of the handling system.

3. Define safe load travel paths and procedures and operator training to assure to the extent practical that heavy loads are not carried over or near irradiated fuel or safe shutdown equipment.

4. Provide mechanical stops or electrical interlocks to prevent movement of heavy loads over irradiated fuel or in proximity to equipment associated with redundant shutdown paths.

5. Where mechanical stops or electrical interlocks cannot be provided, provide a single-failure-proof crane or perform load drop analyses to demonstrate that unacceptable consequences will not result.

By Generic Letters dated December 22, 1980, and February 3, 1981 (Generic Letter 81-07), all utilities were requested to evaluate their plants against the guidance of NUREG-0612 and to provide their submittals in two parts; Phases I (six month response) and Phase II (nine month response). Phase I responses were to address Section 5.1.1 of NUREG-0612 which covers the following areas:

1. Definition of safe load paths
2. Development of load handling procedures
3. Periodic inspection and testing of cranes
4. Qualifications, training and specified conduct of operators
5. Special lifting devices should satisfy the guidelines of ANSI N14.6 6.
6. Lifting devices that are not specially designed should be installed and used in accordance with the guidelines of ANSI B30.9
7. Design of cranes to ANSI B30.2 or CMAA-70

Phase II responses were to address Sections 5.1.2 thru 5.1.6 of NUREG-0612 which cover the need for electrical interlocks/mechanical stops, or alternatively, single-failure-proof cranes or load drop analyses in the spent fuel pool area (PWR), containment building (PWR), reactor building (BWR), other areas and the specific guidelines for single-failure-proof handling systems.

We have completed our review of the utilities' submittals for Phase I for nearly all operating reactors. Only one plant still remains to be reviewed. During our review we verified that the seven guidelines listed above were providing the desired level of safety indicated in NUREG-0612. By way of the utilities' responses to the criteria of NUREG-0612, Section 5.1.1 and through discussions with our consultants based on their experiences from the reviews, we have concluded that the Phase I guidelines have provided an increased awareness by the utilities of the importance of heavy load handling.

Our review has indicated that satisfaction of the Phase I guidelines assures that the potential for a load drop is extremely small. We have noted

improvements in heavy load handling procedures and training and crane and handling tool inspection and testing. These changes have been geared to limiting the handling of heavy loads over safety-related equipment and spent fuel to the extent practical, but where this can not be avoided, to accomplishing it with the operational and other features of the program implemented in Phase I. We therefore conclude that the guidelines of Phase I are adequately providing the intended level of protection against load drop accidents.

To date we have received Phase II submittals from all licensees. We interpret Phase II of NUREG-0612 as an enhancement to Phase I. Thus, prior to undertaking a review of the utilities' Phase II response for all of the operating reactors, and as a test of the adequacy of the Phase I program, we decided to undertake a pilot program with a limited number of plants. The findings from the pilot program would then provide a basis for a decision on whether to proceed with the review of the Phase II submittals for all operating reactors, to reduce the scope of the review, or to totally eliminate the review.

The pilot program involved the review of operating reactors at 12 sites, a total of 20 reactors (eight BWRs and 12 PWRs). Of the 20 reactors, 5 BWRs (Browns Ferry 1, 2 and 3 and Peach Bottom 2 and 3) have single-failure-proof cranes for all heavy load lifts. "Single-failure-proof" is used to mean a crane which meets the guidelines of NUREG-0554, "Single-Failure-Proof Cranes for Nuclear Power Plants." Three BWR units (Dresden 2 and 3 and Big Rock Point) have taken credit for a combination of single-failure-proof cranes in some plant areas and load drop analyses in others. Five PWR reactors (Millstone 2, Prairie Island 1 and 2, and Surry 1 and 2) have utilized the load drop analysis approach. One plant (Kewaunee) has taken credit for a combination of electrical interlocks in some plant areas and load drop analyses in others. The remaining six reactors (Davis Besse, Indian Point 2, Arkansas 1 and 2 and Calvert Cliffs 1 and 2) chose to take credit for a combination of administrative controls, procedures and Technical Specification restrictions in conjunction with some type of load drop analysis. This approach does not meet the criteria of Sections 5.1.2 to 5.1.6 of NUREG-0612. Rather, it is an amplification of the guidelines of the Phase I effort, reflecting Section 5.1.1 of NUREG-0612.

It should also be noted that we have completed our review of Phase II for five operating license applicants. Of these, two (WNP-2 and Fermi-2) have single-failure-proof cranes. The remaining three (Callaway, Wolf Creek and Catawba 1 and 2) employ a combination of electrical interlocks, mechanical stops, limit switches and load drop analyses.

In addition to the detailed reviews of the Phase II reports in the pilot program and in connection with the five operating license applications, we have performed a sufficient review of all other Phase II reports to flag any outstanding plant-specific concerns reported.

From our pilot program and OL Phase II reviews, together with the above-mentioned reviews of the other Phase II reports, we have concluded that the risks associated with damage to safe shutdown systems are relatively small because:

1. nearly all load paths avoid this equipment
2. most equipment is protected by an intervening floor
3. of the general independence between crane failure probability and safety-related systems which has been observed
4. redundance of components

We did not identify any outstanding plant specific safety concern associated with heavy loads handling.

Therefore, most of the risk appears to be associated with carrying heavy loads over or in a location where spent fuel could be damaged. The single most important example of this concerns loads handled over the open reactor vessel during refueling (such as the reactor vessel head). However, as previously mentioned, this is limited to the extent practical and where necessary, is performed with a specifically implemented program in conformance with the Phase I guidelines.

From the pilot program and OL reviews, we noted that nine of the twenty reactors, all PWRs, do not have single-failure-proof cranes. To date, we have not identified any PWRs with single-failure-proof cranes. Further, since electrical interlocks and mechanical stops are not possible for PWR polar

cranes, these reactors would be required to perform costly detailed load drop analyses. If satisfactory results could not be demonstrated from these analyses, NUREG-0612 would call for installation of a single-failure-proof crane.

Based on the above, since a single failure proof crane becomes the only solution for satisfying the NUREG-0612 criteria, the cost/benefit should be examined. Because we are dealing primarily with PWRs, the cost for modification of a polar crane to meet single failure criteria (NUREG-0554) guidelines) is approximately $30 million. This includes, as the dominant cost element, the cost of the extended shutdown which is required in order to gain access to containment. On the benefit side, given the improvements obtained from the Phase I implementation and the information obtained in the course of the pilot program and OL Phase II reviews, we cannot perceive a significant enough benefit in conversion to single-failure-proof polar cranes to warrant the high costs. (See Attachment I for a cost-benefit analysis.) We believe that the cost/benefit analysis in NUREG-0612 is no longer valid because of the benefits realized by Phase I implementation.

We believe the above assessment is further borne out by the industry experience with handling of heavy loads over the years. Precautions have been and are being taken such that no heavy load drop accidents affecting any features of the defense-in-depth against severe core-damage accidents have occurred.* This determination is also supported by the recommendation of our contractor for the pilot program reviews (Franklin Research Center) and our benefit-cost analysis suggesting that we accept other, less strigent but less costly means for Phase II compliance as an alternative to the criteria of NUREG-0612 with respect to conversion to single-failure-proof cranes.

Conclusion and Recommendation

Based on the above, we believe the Phase I implementation has provided sufficient protection such that the risk associated with potential heavy load

*There have, however, been recent occurrences of lesser severity. (See for example, IE Information Notice No. 85-12: Recent Fuel Handling Events; LER 84-015, Fort Calhoun 1, Load Over the RCS; and LER 84-006, San Onofre 2, Polar Crane Malfunction). Accordingly, nothing in this determination should be regarded as a basis for any de-emphasis of continued attention to the safe handling of heavy loads.

drops is acceptably small. We further conclude that the objective identified in Section 5.1 of NUREG-0612 for providing "maximum practical defense in depth" is satisfied by the Phase I compliance, and that the Phase II analyses did not indicate the need to require further generic action at this time.

This conclusion has been confirmed by the results obtained from the Phase II pilot program and additional Phase II reviews, which identified no residual heavy loads handling concerns of sufficient significance to demand further generic action. All plants have examined their load handling practices against the recommendations of Phase II and submitted the Phase II report. In this way, the utilities were required to identify any unexpected problems to the staff.

ATTACHMENT I

SUMMARY OF COST-BENEFIT ANALYSIS OF PWR POLAR CRANE CONVERSION TO SINGLE-FAILURE-PROOF FEATURES

SCOPE

The safety benefit of converting the polar crane in the containment of an operating or completed or nearly completed PWR to single-failure-proof features and the cost of the conversion were estimated and compared.

The safety benefit was estimated in terms of the resulting reduction in the risk of a severe accident, involving major radioactive material release, during the remaining plant life. The risk was expressed as the product of the accident probability and the population radiation dose from the release, should the accident occur.

The cost estimate included the cost of shutdown (or extension of a non-operating period) needed to accomplish the conversion.

ACCIDENT FREQUENCY ESTIMATES

Crane Failure Frequency

There were 32 crane LER events in the approximately 400 reactor-years of U.S. power-reactor operation in the 10-year period July 1969 to July 1979 (NUREG-0612, p. 4-6). None resulted in radioactive release. Of the 32 events, 17 (i.e., just over half) were apparently due to hardware design or fabrication causes, the other 15 to human factors. (Navy crane statistics, cited in NUREG-0612, for 40 load-drop or potential load-drop events in 1974-77 show 80% of the events to be due to human factors.)

It may be assumed, as a rough approximation, that Phase I of NRC's heavy-loads generic program is addressed to all the human factors causes and one-half of the hardware causes and succeeds in reducing the affected part of the failure frequency to a quite small fraction of the frequency originally present. Since human factors and hardware each contribute about one-half of the failures, approximately 3/4 of the total crane failures can be expected to be eliminated by the Phase I program. Single-failure-proof (SFP) cranes should substantially reduce the remaining 1/4 of the failure frequency, though those failures would not be eliminated altogether, since the SFP feature (as defined in NUREG-0554) does not protect against all types of possible failure (e.g., the bridge is not SFP and the SFP feature itself is subject to defeat by some types of human error). On the other hand, the SFP feature would make the cranes more "forgiving" of imperfections in the Phase I implementation. Accordingly, one may reasonably assume that the SFP feature would have a net effect of eliminating 1/4 of the pre-Phase I failure frequency.

Frequency of Accidents Involving Radioactive Release

Not all LER events involve radioactive release. In over 600 reactor-years of U.S. power-reactor operation to date [1982] there have, to our knowledge, been no radioactive releases due to load drops. The 10-year period covered by the survey in NUREG-0612, which included 32 crane LER events, all without release, represents about 60% of all U.S. power-reactor operating time to date. An assumption of a pre-fix frequency of some radioactive release once in 1,000 reactor-years appears consistent with the LER-reflected failure experience, taken together with the absence of releases to date. With 1/4 of these releases averted by an SFP crane feature, the pertinent release frequency reduction would be 1 in 4,000 RY. For the most part, these can be assumed to be minor releases due to limited fuel damage in the spent-fuel storage pool or in the reactor.

Frequency of Accidents Involving Major Releases

For a load-drop event to cause a major accident, with major radioactive release, special circumstances need to be present -- circumstances that Phase I is intended to make much less likely to occur. A highly damaging heavy load drop, such as one that could destroy a core cooling feature through violation of -- or imperfections in -- Phase I provisions combined with other failures, should be unlikely, and very unlikely to lead to major release, because of back-up safety provisions (e.g., independent additional core cooling provisions).

Review of typical load paths and associated crane-operation frequencies suggests that of all load drops in a typical PWR plant that could have radiological consequences, some 1/4 could involve equipment with a role in safe reactor shutdown, including primary-system piping. If one assumes that there is typically a 1% probability that back-up revisions would also fail, then the pertinent major-release frequency is 1 in 1,600,000 reactor-years.

Frequency Reduction

Single-failure-proof cranes may reasonably be expected to eliminate most, perhaps 90%, of the residual load-drop probability after the Phase I improvements. Thus, the frequency reduction for major release is approximately 1 in 1,800,000 RY (90% of 1/1,600,000).

It should be noted that these estimates are sensitive to plant layout. Plant-specific evaluations could, depending on case specifics, point to a much higher or lower major-release frequency estimate for a specific case. For example, should layout of a specific plant be such that a particularly unfortunate load drop could destroy all means of core cooling or incapacitate the control room (possibilities suggested by the situations at Montecello and Arkansas Nuclear 1, respectively, before remedial actions were taken at those plants), the above generic analysis could be wide of the mark

for such a plant. The major-release accident frequency could well be an order of magnitude higher for such a plant (i.e., of the order of 1 in 100,000 reactor-years) -- or even higher, depending on plant and crane features, load paths, and operating practices.

CONSEQUENCES ESTIMATE

Potential radiological consequences of load-drop accidents encompass a wide range of possibilities, depending on specific features of plant design, operating practices, and the nature and location of the specific load-drop event. We assume that some -- though very rough -- indication of the severity of the load-drop accident risks may be gained by using in these simplified calculations certain selected release categories described in WASH-1400, Appendix VI, pp. 2-1 to 2-4. Category PWR 4 was selected for the major-release estimates for pressurized water reactors.

In PWR 4 core cooling and containment both fail. Core melt occurs. This release category is used to explore consequences of a load drop that incapacitates core cooling (during or promptly after reactor operation), with containment open.

The release estimates, stated as resulting public dose, based on representative generic estimates, for a hypothetical site with a projected Year 2000 mean U.S. power-reactor-site population density, developed by Battelle Pacific Northwest Laboratories (NUREG/CR-2800) is 2,700,000 person-rem.

COST ESTIMATE

Costs of change-over to single-failure-proof cranes are subject to wide plant-specific variation, depending on the number of features of the specific cranes involved and other aspects of plant design and status.

Based on advice from the Auxiliary Systems Branch, DSI, and limited vendor and utility contacts, we take the following estimates as representative (as of 1982, when the estimates were made).

For future plants, the cost differential for original inclusion of SFP features is estimated at about $250,000 for PWRs (based on information from Ederer Crane Co.).

At the pre-operating-license stage, with no startup delay, the costs -- including planning, engineering, hardware, installation, and testing -- are estimated at $2 million per plant. This is based on the Monticello experience (1 M in 1976, adjusted for inflation). (The Monticello information was obtained from the licensee through the NRC resident inspector.)

For operating PWRs the estimated costs are dominated by plant shutdown during modifications of the polar crane located inside the containment building. (The shutdown may be an extension of a shutdown for refueling or other purposes.) The cost effect of a startup delay for a completed or nearly completed plant would be similar. With a 3-month shutdown and with shutdown costs taken as determined by the cost of replacement power at $300,000 per day, representative total change-over costs for operating PWRs are estimated at about $30 million.

RISK REDUCTION

Based on the foregoing frequency and consequences estimates, the "expected value" of the risk subject to being affected by the possible Phase II SFP feature, i.e., the magnitude of release times the frequency of its

occurrence, integrated for the remaining plant life taken as 20 years, is as follows:

$$\text{Major release risk} = 20 \times \frac{2{,}700{,}000}{2{,}800{,}000} = 30 \text{ person-rem/reactor}$$

COST-BENEFIT RATIO

The cost-benefit ratio indicated by the foregoing estimates is approximately \$1,000,000/person-rem. This estimate is subject to wide plant-to-plant variation as well as large uncertainties in the underlying estimates of accident frequency and consequences. Nevertheless, it is possible to conclude with reasonable confidence that the benefit-cost ratio for the crane conversion would fail to meet a \$1,000/person-rem worthwhileness criterion by a large margin.

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