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

NOV 25 1985

(3)

MEMORANDUM FOR: Robert Bernero, NRR
Edward Jordan, IE
Richard Cunningham, NMSS
Denwood Ross, RES
Clemens Heltemes, AEOD
Joseph Scinto, ELD

THRU: James H. Sniezek, Director
Regional Operations and Generic Requirements Staff

FROM: Merrill A. Taylor, Senior Program Manager
Regional Operations and Generic Requirements Staff

SUBJECT: SUMMARY AND ISSUE IDENTIFICATION FOR CRGR MEETING NO. 83

Enclosed for your information and use is the ROGR staff summary of the Proposed Closeout Actions on Generic Issue B-19, Thermal Hydraulic Stability for BWRs and related actions on Issues B-59 and MPA E-04 concerning PWR and BWR Partial (N-1) Loop Operations.

This matter is scheduled for CRGR review at Meeting No. 83 on Wednesday, November 27, 1985 in Room 6507 MNBB.

Mat Taylor
Merrill A. Taylor
ROGR Staff

Enclosure: As stated

cc: V. Stello

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Original signed by
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SUMMARY AND ISSUE IDENTIFICATION FOR CRGR REVIEW ITEM CRGR MEETING NO. 83

IDENTIFICATION

1. Generic Issue B-19, Thermal Hydraulic Stability for BWRs; Proposed Closeout Actions and Generic Letter for All Licensees of Operating BWRs.

2. Generic Issue B-59, (N-1) Loop Operations in BWRs and PWRs; Proposed Closeout Actions and Generic Letter for All Licensees of Operating BWRs and PWRs and License Applicants. (This also relates to the backlog on multiplant action items, MPA 04 and MPA 05 - see ORLAS.)

DESCRIPTION AND OBJECTIVE

The staff is proposing to closeout Generic Issues B-19 for the BWRs concerning Thermal Hydraulic Stability and the closely related Issue B-59 (MPA 04 and 05) that addresses the NRR approval of partial (N-1) loop operations for the PWRs and BWRs.

These proposed closeout actions would result in licensee Technical Specification revisions particularly for the BWR plants. These revisions to the Technical Specifications could be brought about by the licensees response to the proposed B-19 and B-59 generic letters or could be imposed by the staff at some later time during core reload reviews. The revisions for the BWR would in essence enforce operational recommendations contained within a General Electric Company Service Information Letter (SIL-380, Revision 1, February 10, 1984) by incorporation into Technical Specifications. Such revisions would represent a trade-off for NRC approval of BWR single loop operation. This SIL was issued by General Electric shortly after unusual oscillatory behavior occurred at a foreign reactor during a thermal-hydraulic stability test. Applicable BWR Atomic Safety and Licensing Boards and the Commission were notified of the experience during the Feb-March 1984 time frame. The staff advised that these foreign reactor data were important and confirmed the possibility of local thermal-hydraulic instabilities that had been postulated but not observed in similar BWR stability tests in the United States. Further, the staff advised in this notification that the staff would be working with applicants and the BWROG in review of the standard Technical Specifications to assure that they properly protect against the potential for instabilities. The staff also stated that if changes were to be required, they would follow staff procedures for generic Technical Specification changes including a CRGR review.

Contacts:

L. Phillips, NRR, 492-9472
M. Taylor, ROGR Staff, 492-4356

The ASLBs and Commission were advised that this new information from the foreign BWR reactor did not pose an immediate safety concern for continued BWR operations in large part because of (1) licensee detectability, (2) the large heat flux margins and (3) current BWR Technical Specifications placed restriction on operations under conditions of natural circulation or single loop operation such that the frequency of operation into regions with low stability margin would be very low. The proposed resolution for B-19 when taken together with B-59, would be expected to change (increase) this frequency by some as yet undetermined amount. The staff prioritization analyses of June 12, 1985 on Generic Issue B-59 could bring about changes in the standard review plan and to standard technical specification; this final resolution of B-59 however, hinged on the resolution of B-19. Additionally, these prioritization analyses indicated that resolution of B-59 largely affected single loop operation for the BWRs and had but very slight effect on reducing public risk and occupational exposures. On the other hand, the analyses that could be required of a licensee by the staff to carry out resolution of B-59 is estimated to involve approximately $420,000 per plant for one-half of the 44 BWRs affected. By permitting the BWR single loop operation heretofore restricted (to 24 hours or less before shutdown), the staff projects an outage avoidance savings of $255,000 per plant-year with the present worth savings projected to average roughly $3.8 million per plant over the remaining lifetime.

In his May 21, 1985 memorandum on the matter of B-19 closeout, the Director of NRR advised that he planned the following actions:

1. For all new licensing actions which affect thermal-hydraulic stability, the licensee must show compliance with GDC 10 and GDC 12 by either of the following methods.

    a. Show thermal-hydraulic instabilities are not possible by design. This would be done using a best estimate analysis which includes approved analysis uncertainties.

    OR

    b. Show that proper detection and suppression capabilities (consistent with GE recommendations) are in the plant Technical Specifications.

2. Inform all BWR licensees by letter of the staff's technical resolution of B-19 and its safety significance to evaluation of core reloads.

3. Take <u>no</u> immediate action on operating plants but assure that new core reloads are properly evaluated for conformance to GDC 12 in accordance with item (1).

This memorandum further advised that restrictions on continuous operation in the single loop mode may be removed for the BWR plants having the revised Technical Specifications and stability procedures in place to enforce GE SIL 380, Revision 1. A conclusion was given that these proposed NRR actions did not constitute new requirements; thus, a CRGR review was not needed.

BACKGROUND

See the above description and the package of information transmitted with the CRGR Meeting NO. 83 agenda announcement of November 18, 1985. This pa^kage did not contain information called for by Paragraph IV.B of the CRGR Charter; however, the NRR staff has been requested to be prepared to address this information (memorandum dtd November 14, 1985 from CRGR Chairman). The CRGR information package contained the following:

1. Memorandum dtd May 21, 1985, H. R. Denton to V. Stello, Jr., Thermal-Hydraulic Stability. This transmitted the following:

   a. Schedule for Resolving B-19, memo dtd January 4, 1985, H. Denton to R. Bernero

   b. Staff Safety Evaluation of GE Topical Report NEDE-24011 (GESTAR) Amendment 8. Memorandum dtd April 17, 1985 from L. Rubenstein to D. Crutchfield.

   c. Staff Safety Evaluation of Exxon Nuclear Company Topical Report XN-NF-691(P). Letter dtd May 10, 1984 to Mr. Chandler Exxon Nuclear Co., Inc. from Cecil Thomas, NRR

   d. Proposed Generic Letter to all BWR licensees concerning Resolution of B-19.

2. Memorandum dtd August 20, 1985, R. Bernero to H. Thompson concerning proposed Generic Letter for Issue B-59, (N-1) loop operation in BWRs and PWRs.

3. Memorandum dtd June 12, 1985, H. Denton to R. Bernero; Schedule for Resolving and Completing B-59 including Prioritization Evaluation.

4. Copy of GE SIL-380, Revision 1, dtd February 10, 1984.

5. Memorandum dtd March 23, 1984, D. Eisenhut to the Commission concerning Board Notification on BWR Core Thermal-Hydrualic Stability (transmitted, memorandum dtd February 27, 1984, R. Mattson to D. Eisenhut).

6. Memorandum dtd July 11, 1984, L. Rubenstein to T. Novak concerning Susquehanna Technical Specifications incorporating GE SIL-380, Revision 1, recommendations as being acceptable for plants that do not have permanent single loop operation approval.

7. Letter dtd May 28, 1985 to Mr. Liu, Chairman of the Board/CEO, Iowa Electric Light and Power from M. Thadani NRR. Transmitted Amendment 119 consisting of approval on Technical Specifications for single loop operation and incorporating GE SIL-380, Revision 1. (Included Supporting Safety Evaluation for Amendment 119).

8. Letter dtd July 20, 1985 to Mr. Carey, VP Duquesne Light Co. (DLC) from S. Varga, NRR. Transmitted Staff Safety Evaluation Report approval of two of three loop operations. Responds to October 27, 1978 request by DLC for such Technical Specification changes. (Staff finds final approval of requested technical specifications for Beaver Valley 1 to be contingent on resolving concerns about asymmetric LOCA blowdown loads.)

ISSUES

1. CRGR may wish to determine the bases on which staff believes Resolution of B-19 and B-59 represents no new requirements of generic nature.

2. CRGR may wish to explore the recent operating experience at a foreign reactor that is understood to have experienced an instability even with the recommendations of GE SIL-380, Revision 1 in place. It could be of interest to determine what specific safety gains are expected by the staff from enforcing the GE SIL-380 recommendations via Technical Specifications.

3. CRGR may wish to determine the validity of cost estimates of $420,000 per plant for licensee implementation of B-19 and B-59. The nature of the analyses that could be required by staff involving such an expenditure for the safety benefits involved is not evident to the ROGR staff. For example, would the staff expect these extensive analyses to be required during core reload reviews?

4. CRGR may wish to determine if the staff intends to update its Board Notifications to advise of staff approval of BWR single loop operation since restriction to this mode of operation was a factor of importance cited to the ASLB in favor of permitting continued BWR operations.

5. The proposed resolution of B-59 advises all PWRs of staff approval of N-1 loop operation based on the Beaver Valley 1 analyses and SER. This would seem to be an unusually broad finding generalized to all PWRs from a three loop plant analyses. CRGR may wish to further explore the staff rationale for this approach. Beaver Valley 1 requested approval of this mode of operation in the latter part of 1978 and there has been substantive supporting analyses provided at staff's request since that time. The Beaver Valley 1 has design loop isolation valves that could be used for such N-1 mode of operation similar to a handul of other W designs (e.g., Zion, Surry, etc.). CRGR may wish to determine how staff's findings are affected for PWR plants with or without the loop isolation valve design and whether the staff foresees any other PWRs having a interest in adopting this N-1 mode of operation. (Why has the staff chosen to treat the Beaver Valley 1 plant-specific finding with a generic letter?)

6. In its prioritization analyses for B-59, the staff uses Oconee (PWR) pump data and applies these to analyses of benefits for BWR single loop operation. CRGR may wish to explore reasons why BWR burnup data were not used and the validity of the projected economic benefits to the licensees.

Philosophically, this analyses would imply economic rather than safety regulation.

7. CRGR may wish to explore staff's ECCS findings concerning BWR single loop operations and the resultant sensitivity of ECCS reliability and performance given loop isolation and the use of loop selection logic.

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Enclosure 4 to the Minutes of CRGR Meeting No. 90
Proposed Final Revisions to Property Insurance Rule, 50.54(w)

J. Saltzman and R. Wood (SP) presented for CRGR review, the proposed final rule package that would modify the existing onsite property insurance requirements set forth by 10 CFR 50.54(w), Conditions of Licenses. A copy of the vugraphs used in presenting this issue to CRGR are attached hereto.

As to background on this issue, a 50.54(w) proposed rule modification was published for comment on November 8, 1984, (49 FR 44645) which would increase the amount of onsite property damage insurance that commercial reactor licensees are required to carry pursuant to 10 CFR 50.54(w) and would propose a decontamination priority to be placed on any proceeds from such insurance at the order of the Director of NRR. Operating reactor licensees are currently required to carry both (1) the maximum amount of property insurance offered as primary coverage by either American Nuclear Insurers/Mutual Atomic Energy Reinsurance Pool (ANI/MAERP) or Nuclear Mutual Limited (NML) -- currently $500 million -- and (2) any excess coverage in an amount no less than that offered by either ANI/MAERP -- $85 million -- or Nuclear Electric Insurance Limited (NEIL-II) -- recently increased to $550 million. Thus, the minimum currently required under the rule is $500 million primary coverage and $85 million excess coverage. By buying both excess layers, most utilities are able to purchase a total of $1.135 billion in property insurance (per site) except where State constitution or law prohibits. (The current 50.54(w) onsite property damage insurance requirements were codified by NRC on March 31, 1982 pursuant to the TMI-2 accident experience.)

The proposed final rule package was transmitted for CRGR review on April 25, 1986. As this proposed rule package clearly noted, the NRC is injecting itself explicitly into the regulation of terms and conditions of onsite property damage insurance agreements and into allocations of the insurance proceeds after a reactor accident. This action is said to have been taken in the interest of protecting public health and safety, but it has also raised a number of complex issues and potential conflicts (such as NRC regulation affecting investment risks and capital costs, the abridgement of insurance contractual rights, Federal preemption of traditional State regulatory jurisdiction over insurance practices, Federal preemption of certain State laws and State constitutional prohibitions on public power entities within these States, etc). The matter of the NRC prioritizing insurance proceeds for cleanup and decontamination purposes was viewed as one of the more controversial issues underlying this proposed final rule. In light of these problems, the staff has offered alternative approaches in the rule package that the Commission may wish to take into further consideration prior to issue of a final rule on this matter. The package also contains analyses intended to address the provisions of the existing backfit rule (10 CFR 50.109). These analyses conclude with a staff belief that because of the minimal cost impact on licensees and because such insurance does reduce the potential for public and occupational exposures, the final rule is justified even though a finding of substantial improvement to public health and safety

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cannot be made. The staff further observed in its CRGR transmittal letter for this final rule package that on September 20, 1983, the Commission (as a matter of policy), directed the EDO and staff to revise the property insurance rule to conform to its explicit directions concerning the amount of insurance coverage to be required and the prioritization of insurance proceeds to accident decontamination and cleanup.

The following comments, concerns and recommendations were expressed by the CRGR on the proposed rule package:

1. CRGR expressed concern about the rule structure and the lack of precision in the rule language and its meaning. Specifically, CRGR noted that there was a need to develop a better definition and a staff consensus as to when the reactor would no longer be considered a public health and safety threat. CRGR expressed the view that the proposed rule language was in need of revision in this regard since the impression was being conveyed that the Director of NRR would only order prioritization of the insurance proceeds after the reactor had reached a safe and stable stage and when some considerably lesser, but yet undefined, threat to the public health and safety existed .

2. CRGR expressed a concern about the lack of definition (or distinction) between the decontamination and cleanup phase of the accident versus the start of the decommissioning phase when the NRR prioritization over the property damage insurance proceeds would cease. CRGR suggested that the staff should undertake an effort to more explicitly define these various accident transition points so as to reduce the apparent degree of vagueness between the safe and stable phase, the decontamination and cleanup phase and the decommissioning phase of an accident .

3. CRGR expressed the view that it was important that the staff consider the adequacy of (or any conflicts with) the actual policy language contained in the NEIL-II excess layer insurance policies since the staff advised CRGR that such policies were thought to already have included a decontamination and cleanup priority on the insurance proceeds.

4. CRGR suggested that the staff review and, as appropriate, revise the language used in the Regulatory Analyses for the rule. For example, CRGR noted that staff had referred to the <u>"health and safety problem"</u> at TMI-2 rather than the <u>threat</u> of such problems - the latter being more correct.

5. CRGR noted that the staff's rationale for requiring an insurance amount (i.e., 1.06 billion dollars) based on the worst case accident scenario was unclear and questioned whether such an amount was really needed for purposes of protecting public health and safety. Regarding a requirement for this worst-case accident coverage, CRGR noted that there seemed to be a dichotomy between the Price-Anderson approach and that being taken by the staff for the onsite property damage coverage. CRGR suggested that the staff should be prepared to more fully explain these differences in approach and their rationale for selecting the worst-case economic loss scenario for required insurance coverage.

6. CRGR noted that the proposed rule language regarding the minimum insurance coverage seemed to be in need of revision so as to guard against an apparent loophole. The staff should make it more clear that licensees would be expected to maintain the maximum amount of insurance coverage generally available.

7. CRGR suggested that the staff consider the merits of revising the annual reporting requirements of the rule to reflect that the reporting to NRC need only be done when the amount of insurance coverage is less than that specified by the rule.

8. CRGR noted that the proposed rule language appeared to create a prerequisite that the Director of NRR issue an order for prioritizing the insurance proceeds to decontamination and cleanup regardless of the nature and magnitude of an accident. The CRGR questioned the need for a requirement to be placed on the Director of NRR and, therefore, recommended SP and ELD further review the merits and acceptability of this requirement with the Director of NRR. CRGR noted that the matter of when the Director of NRR is or is not to become involved in the use of the insurance proceeds is also a central issue requiring better definition.

9. CRGR noted that the proposed final rule would essentially be an NRC directive to the insurance companies to alter their policies. CRGR attempted to better understand the NRC's jurisdiction and role in this regard and its rights and relationships with the utility's indenture trustee in the control over the insurance proceeds. CRGR also questioned NRC's standing with respect to a bankruptcy proceeding in this matter. The staff acknowledged that they did not have good answers to some of these questions and that it was entirely possible that legal challenges could arise from NRC's use of the decontamination priority on the insurance proceeds. On the other hand, the staff felt that the pros and cons on these difficult matters had been discussed to a considerable extent in the proposed rule package.

In view of the above concerns the CRGR concluded that it could not, at this time, recommend to the EDO approval of the proposed rule for Commission review for publication. In addition to the need for further coordination between SP, ELD and NRR on the matters noted above, CRGR concluded that the rule language was in need of revision. Accordingly, the CRGR recommended that additional CRGR review of this matter be scheduled when the rule package has been revised and resubmitted by SP.

PROPERTY INSURERS

1. AMERICAN NUCLEAR INSURERS/MUTUAL ATOMIC ENERGY REINSURANCE POOL (ANI/MAERP) - TRADITIONAL INSURANCE POOL ALSO PROVIDING PRICE-ANDERSON COVERAGE.

   - OFFERS $500 M PRIMARY PROPERTY COVERAGE AND $85 M EXCESS.

2. NUCLEAR MUTUAL LIMITED (NML) - "CAPTIVE" INSURER FORMED BY UTILITIES. IN ADDITION TO ADVANCE PREMIUMS, UTILITIES ARE LIABLE TO PAY A RETROACTIVE ASSESSMENT IF CLAIMS OCCUR.

   - OFFERS $500 M PRIMARY COVERAGE ONLY

3. NUCLEAR ELECTRIC INSURANCE LIMITED (NEIL-II) - LIKE NML, IS CAPTIVE INSURER FORMED BY UTILITIES AND INCLUDES POSSIBLE PAYMENT OF RETROSPECTIVE PREMIUMS.

   - OFFEKS $550 M EXCESS COVERAGE ONLY.

PROPERTY INSURANCE

CURRENT RULE

(10 CFR 50.54(w))

1. REQUIREMENTS PEGGED TO CAPACITY PROVIDED BY PROPERTY INSURERS.

• PRIMARY CAPACITY FROM EITHER NML OR ANI/MAERP	- $500M
• EXCESS CAPACITY FROM EITHER ANI/MAERP ($85M) OR NEIL-II ($550M)	- $85M
	$585M
	TOTAL MINIMUM CURRENTLY REQUIRED

CURRENT PROPERTY INSURANCE

RULE (CON'T)

2. NO DECONTAMINATION PRIORITY.

3. LICENSEES PROHIBITED BY STATE LAW FROM PURCHASING FULL COVERAGE ARE EXEMPTED FROM NRC REQUIREMENTS (50.54 (w) (3)).

4. LICENSEES REQUIRED TO REPORT ANNUALLY ON AMOUNT OF INSURANCE REQUIRED.

PROPOSED RULE

1. INCREASE REQUIRED COVERAGE TO $1.06 BILLION

   - PEGGED TO PNL REPORT (NUREG/CR-2601)

   - TOTAL CAPACITY CURRENTLY AVAILABLE - $1.135 BILLION

2. INSTITUTE A DECONTAMINATION PRIORITY

3. PREEMPT STATE LAW THAT PROHIBITS LICENSEES FROM BUYING NRC REQUIRED AMOUNT

1983 COMMISSION GUIDANCE

- 1983 STAFF PROPOSAL WOULD HAVE KEPT SAME DOLLAR REQUIREMENTS BUT ENCOURAGED GREATER STATE ROLE TO INDUCE HIGHER COVERAGE

- COMMISSION DISAPPROVED STAFF RECOMMENDATION AND REQUESTED REVISED RULE THAT:

  1. REQUIRES A MINIMUM OF $900 MILLION

  2. INCLUDES A DECONTAMINATION PRIORITY PROVISION

  3. ANALYZES IN DETAIL THE PROBLEM OF STATE PROHIBITIONS

CURRENT COVERAGE

- $1.135 BILLION CURRENTLY AVAILABLE FROM INSURANCE MARKET
- 68 SITES COVERED (102 REACTORS)
  - 49 COVERED FOR AT LEAST $1.06 BILLION
    (41 COVERED FOR $1.135 BILLION)
  - 7 ADDITIONAL MAINTAIN AT LEAST $1.0 BILLION
  - 4 EXEMPT IN PART FROM EXISTING REQUIREMENTS
  - 7 CARRY $585 MILLION CURRENT REQUIREMENT
  - 1 NOT YET REPORTED

ISSUE: BASIS FOR $1.06 BILLION

FROM NUREG/CR-2601

• BASIC ACCIDENT CLEANUP COST FOR REFERENCE PWR FOR SCENARIO 3 ACCIDENT	$404 M
• BASE OPERATIONS & MAINTENANCE	$124 M
• COST ESCALATION DURING CLEANUP	$209 M
• DESIGN DIFFERENCE (FOR COMPARISON WITH TMI-[illegible]	$ 84 M
• ADDITIONAL DECONTAMINATION OF CONTAINMENT BUILDING	$100 M
• NET STABILIZATION COST	$139 M
	[illegible] 06 B

ISSUE: STATE PROHIBITIONS OF THE PURCHASE OF MUTUAL OR RETROSPECTIVE PREMIUM INSURANCE

- ELD'S ORIGINAL POSITION WAS THAT NRC COULD NOT PREEMPT STATE LAW
- ELD REVISED ITS OPINION AND, AFTER CONSULTATION WITH OGC, DETERMINED THAT NRC COULD PREEMPT STATE LAW.
  - WHERE A REACTOR LICENSEE HAS NO REASONABLE ALTERNATIVE TO PURCHASING NUCLEAR PROPERTY DAMAGE INSURANCE, STATE PROHIBITION WOULD CONFLICT WITH FEDERAL LAW.
  - IN FOREGOING SITUATION CASE LAW SUPPORTS FEDERAL PREEMPTION

ISSUE: DECONTAMINATION PRIORITY

- DECONTAMINATION PRIORITY - PROCEEDS FROM INSURANCE SHOULD BE USED TO DECONTAMINATE AND CLEAN UP AFTER ACCIDENT BEFORE ANY OTHER PURPOSE SUCH AS FACILITY RESTORATION OR PAYMENT OF INVESTORS.

- WOULD BE IMPOSED ONLY WHEN ORDERED BY DIRECTOR OF NRR

- WOULD ONLY LAST FOR ONE YEAR AND BE RENEWABLE IN 6-MONTH INCREMENTS

ISSUE: DECONTAMINATION PRIORITY (CON'T)

COMMENT: PRIORITY WOULD CONFLICT WITH EXISTING INDENTURE LANGUAGE THAT PROVIDES THAT ALL PROPERTY INSURANCE PROCEEDS ARE TO BE PAID TO AN INDENTURE TRUSTEE WHO REPRESENTS BONDHOLDERS.

STAFF RESPONSE: ACCORDING TO OTHER COMMENTS RECEIVED (I.E. FROM NEW YORK CITY BAR ASSOCIATION), BOND INDENTURE LANGUAGE TYPICALLY REQUIRES A UTILITY "TO INSURE ITS PROPERTY AGAINST LOSS OR DAMAGE TO THE SAME EXTENT THAT PROPERTY OF A SIMILAR CHARACTER IS USUALLY SO INSURED BY COMPANIES SIMILARLY SITUATED AND OPERATING LIKE PROPERTIES." SUCH LANGUAGE WOULD ALLOW THE NRC TO IMPOSE A DECONTAMINATION PRIORITY BECAUSE ALL UTILITY LICENSEES WOULD FACE SIMILAR CONDITIONS, WOULD BE INSURED 'TO THE SAME EXTENT' AND WOULD THUS COMPLY WITH INDENTURE LANGUAGE.

ISSUE: DECONTAMINATION PRIORITY (CON'T)

COMMENT: PRIORITY WOULD BE REDUNDANT BECAUSE A DECONTAMINATION PRIORITY ALREADY EXISTS IN THE $500 MILLION EXCESS LAYER OFFERED BY NEIL-II.

STAFF RESPONSE: STATEMENT IS PARTIALLY TRUE, BUT BASIC $500 M IS STILL NOT PRIORITIZED. ALSO, THIS ARGUMENT UNDERCUTS UTILITY ARGUMENT THAT PRIORITY CONFLICTS WITH BOND INDENTURE LANGUAGE AND WOULD DISRUPT CAPITAL MARKETS.

ISSUE: DECONTAMINATION PRIORITY (CON'T)

COMMENT: PRIORITY WOULD DISCOURAGE INVESTMENT AND INCREASE COST OF CAPITAL

STAFF RESPONSE: WHETHER COSTS WOULD BE INCREASED IS SPECULATIVE. STAFFS DOUBTS THAT THIS IS TRUE BECAUSE (1) NRC MAY ALREADY HAVE AUTHORITY TO IMPOSE A DECONTAMINATION PRIORITY; (2) NEIL-II ALREADY HAS A PRIORITY WITHOUT CAUSING NOTICEABLE DISRUPTION; (3) UNPRIORITIZED INSURANCE NOW IS ONLY SMALL PORTION OF POTENTIAL LOSS, AND (4) CONTINUING GROWTH IN INSURANCE CAPACITY SHOULD ACT TO EASE LACK OF COVERAGE.

ISSUE: DECONTAMINATION PRIORITY (CON'T)

COMMENT: PRIORITY WOULD BE UNNECESSARY BOTH BECAUSE OF THE INTEREST OF UTILITY MANAGEMENT AND INVESTORS TO CLEAN UP A PLANT AND RETURN IT TO OPERATION AND BECAUSE THE NRC ALREADY HAS AUTHORITY TO IMPOSE SUCH A PRIORITY.

STAFF RESPONSE: STATEMENT MAY BE TRUE, BUT DOES NOT REALLY ARGUE AGAINST EXPLICIT DECONTAMINATION PRIORITY AND UNDERCUTS ARGUMENT THAT AN EXPLICIT DECONTAMINATION PRIORITY IS BURDENSOME.

ISSUE: DECONTAMINATION PRIORITY (CON'T)

COMMENT: PRIORITY WOULD POSSIBLY LEAD TO PROTRACTED HEARINGS THAT COULD IMPEDE THE CLEANUP PROCESS AND ADVERSELY AFFECT PUBLIC HEALTH AND SAFETY.

STAFF RESPONSE: IF THE LICENSEE CHOSE NOT TO CONTEST THE DIRECTOR'S ORDER, NO PROCEEDING WOULD BE INSTITUTED AND THERE WOULD BE NO OPPORTUNITY FOR A MEMBER OF THE PUBLIC TO INTERVENE. NEVERTHELESS, TO EASE LICENSEES' FEARS, STAFF HAS PROPOSED TIME LIMITS AFTER WHICH PRIORITY WOULD ELAPSE. ALSO, IT WOULD NOT BE IN NRC'S INTEREST TO IMPOSE A PRIORITY SO RIGID SO AS TO IMPEDE THE RECOVERY PROCESS.

PROPERTY INSURANCE COVERAGE
FOR COMMERCIAL NUCLEAR REACTORS

Reactor	Total Coverage ($ Million) As of 4/1/83	As of 4/1/84	As of 4/1/85	As of 4/1/86
Arkansas Nuclear 1-2	963.9	1,020.0	1,085.0	1,135.0
Duane Arnold	983.0	1,020.0	1,085.0	1,135.0
Beaver Valley	983.0	1,003.0	1,085.0	1,135.0
Big Rock Point (a)	500.0	500.0	500.0	500.0
Browns Ferry 1-3	983.0	1,020.0	1,085.0	1,135.0
Brunswick 1-2	915.0	935.0	1,000.0	1,050.0
Byron 1	NA	NA	1,085.0	1,085.0
Callaway	NA	NA	1,085.0	1,135.0
Calvert Cliffs 1-2	915.0	935.0	1,085.0	1,135.0
Catawba 1-2	NA	NA	1,085.0	1,135.0
Donald C. Cook 1-2	983.0	1,020.0	1,085.0	1,135.0
Cooper	568.0	585.0	585.0	585.0
Crystal River	982.0	1,003.0	1,070.0	1,120.0
Davis-Besse	983.0	1,020.0	1,085.0	1,135.0
Diablo Canyon 1-2	915.0	935.0	1,000.0	1,050.0
Dresden 1-3	983.0	1,020.0	1,085.0	1,085.0
Joseph M. Farley 1-2	1,028.0	1,020.0	1,085.0	1,135.0
Fermi 2	NA	NA	1,085.0	1,135.0
James A. Fitzpatrick	568.0	585.0	585.0	585.0
Fort Calhoun	568.0	585.0	585.0	585.0
Fort St. Vrain (b)	500.0	585.0	585.0	1,020.0
R. E. Ginna	983.0	1,020.0	1,085.0	1,135.0
Grand Gulf	983.0	935.0	1,085.0	1,135.0
Haddam Neck (Conn. Yankee)	983.0	1,020.0	1,060.0	1,135.0
Edwin I. Hatch 1-2	983.0	1,020.0	1,085.0	1,135.0
Humboldt Bay (c)	103.7	106.2	120.2	113.4
Indian Point 1-2	983.0	1,020.0	1,085.0	1,135.0
Indian Point 3	568.0	585.0	585.0	585.0
Kewaunee	528.0	585.0	585.0	
La Crosse (a)	61.8	500.0	500.0	500.0
La Salle	983.0	1,020.0	1,085.0	1,085.0
Limerick	NA	NA	1,085.0	1,135.0
Maine Yankee	915.0	935.0	975.0	1,050.0
McGuire 1-2	983.0	1,020.0	1,085.0	1,135.0
Millstone 1-3	983.0	1,020.0	1,060.0	1,135.0
Monticello	983.0	1,000.0	1,000.0	1,000.0
Nine Mile Point	568.0	1,020.0	1,085.0	1,135.0
North Anna 1-2	1,064.0	1,030.0	1,088.0	1,135.0
Oconee 1-3	983.0	1,020.0	1,085.0	1,135.0
Oyster Creek	983.0	1,020.0	1,085.0	
Palisades	983.0	1,003.0	1,050.0	1,100.0
Palo Verde 1-2	NA	NA	585.0	1,135.0
Peach Botton 2-3	983.0	1,020.0	1,085.0	1,135.0

Perry 1	NA	NA	NA	
Pilgrim	983.0	1,020.0	1,085.0	1,135.0
Point Beach 1-2	983.0	1,003.0	1,085.0	1,135.0
Prairie Island 1-2	983.0	1,000.0	1,000.0	1,000.0
Quad-Cities 1-2	983.0	1,020.0	1,085.0	1,085.0
Rancho Seco	568.0	585.0	585.0	585.0
River Bend 1	NA	NA	NA	1,135.0
H. B. Robinson	915.0	935.0	1,000.0	1,050.0
St. Lucie 1-2	983.0	1,020.0	1,085.0	1,135.0
Salem 1-2	983.0	1,020.0	1,085.0	1,135.0
San Onofre 1-3	1,028.0	1,020.0	1,085.0	
Sequoyah 1-2	983.0	1,020.0	1,085.0	1,135.0
Shoreham	NA	NA	585.0	585.0
Summer	982.0	993.0	1,085.0	1,110.0
Surry 1-2	1,064.0	1,030.0	1,085.0	1,135.0
Susquehanna 1-2	983.0	1,020.0	1,085.0	1,135.0
Three Mile Island 1-2	983.0	1,020.0	1,085.0	1,135.0
Trojan	983.0	1,020.0	1,085.0	1,135.0
Turkey Point 3-4	983.0	1,020.0	1,085.0	1,135.0
Vermont Yankee	983.0	1,020.0	1,085.0	1,135.0
Waterford	NA	NA	1,085.0	1,135.0
Wolf Creek	NA	NA	1,085.0	1,135.0
WPPSS	NA	585.0	1,020.0	1,135.0
Yankee Atomic (a)	500.0	500.0	500.0	500.0
Zion 1-2	983.0	1,020.0	1,085.0	1,085.0

Notes

(a) exempt from excess layer
(b) request for exemption from excess layer
(c) exempt from insurance exceeding $100 million
(d) Blanks indicate responses not yet received. Responses have been solicited from licensees.

(3)

Enclosure 2 to the Minutes of CRGR Meeting No. 80
Offsite Emergency Medical Services

E. M. Podolak, IE, briefed the CRGR on the staff views and recommendations for clarifying the requirements of 10 CFR 50.47(b)(12) for offsite emergency medical services. The staff positions and recommendations were contained in a draft Commission Paper which had been coordinated with FEMA and prepared in response to a Commission request. IE requested the information briefing to inform the CRGR of the staff position on offsite emergency medical services and to give the CRGR the opportunity to comment on the proposed staff position. A copy of the briefing slides used for the presentation is attached. As a result of the briefing and discussion, the CRGR concluded that IE should formally present the staff position and recommendations to the CRGR for review. A tentative date of September 25, 1985 was set for CRGR review.

The draft Commission paper responds to a May 17, 1985 Commission request for staff views on three options being considered to clarify the requirements of 50.47(b)(12) for offsite emergency medical services. The three options addressed in the Commission paper are:

1. Reinterpret the phrase "contaminated injured individuals" to eliminate exposed individuals from the scope of the planning standard (b)(12).

2. Determine what additional arrangements are necessary for exposed individuals.

3. Amend planning standard (b)(12) to make a list of medical facilities the sole planning requirement for exposed individuals.

The staff favors option #2 with the recommendation that the licensees, states, local governments, and community medical services (hospitals, clinics, etc.) execute letters of agreement for responding to offsite, general public, medical service needs in the event of an emergency (similar to arrangements for onsite emergency medical service needs). A number of questions and issues were raised during the meeting and the CRGR requested IE to address the following points at the September 25, 1985 briefing:

1. Whether the proposed policy would change the environment/society of the area surrounding the nuclear power plant or merely assure that the existing capabilities of the surrounding environment/society are best used.

2. The staff should review and determine if NRC should specify criteria to be met in providing offsite emergency services, for example, number of beds, capability and quality of decontamination equipment, number of trained and qualified personnel. The staff should clearly identify licensee, state, local government and medical service actions required to satisfy the revised criteria.

K/2

3. The staff should identify the geographic area surrounding the nuclear power plant to which the revised criteria are applicable (for example, 10-mile EP zone).

4. In the draft proposed guidance for medical services, the staff should reassess the need for "accreditation" of radiological capabilities (criteria, program, accredited by whom, costs, etc.) and justify the need for data on ambulatory/nonambulatory capacity of medical services.

5. A cost analysis should be performed for the three options considered in the draft paper. Costs should include licensee, state, local government, and local medical services costs to implement the revised criteria (include for example, training, equipment, exercise costs).

6. The Commission paper should clearly identify if any new requirements or new interpretations of existing requirements are contained in the discussion of the three options.

MEDICAL SERVICES PAPER

1980	COMMISSION REGULATION	° ONSITE AND OFFSITE PLANS: ARRANGEMENTS ARE MADE FOR MEDICAL SERVICES FOR CONTAMINATED INJURED INDIVIDUALS
1981	FEMA POSITION	° SPECIAL ARRANGEMENTS NEED TO BE MADE FOR MEDICAL SERVICES FOR EXPOSURE, CONTAMINATION OR BOTH
1982	ASLBP POSITION	° CONTAMINATED INJURED = EXPOSED
1982	ASLAP POSITION	° CONTAMINATED INJURED ≠ EXPOSED
1982	STAFF POSITION	° CONTAMINATED INJURED ≠ EXPOSED ° <u>AD</u> <u>HOC</u> ARRANGEMENTS FACILITATED BY LIST ARE SUFFICIENT
1983	COMMISSION POLICY	° GREAT WEIGHT TO FEMA'S VIEW ° CONTAMINATED INJURED = EXPOSED ° <u>AD</u> <u>HOC</u> ARRANGEMENTS FACILITATED BY LIST ARE SUFFICIENT
1985	COURT DECISION	° LIST ≠ "ARRANGEMENTS" ° NO TIGHT RESTRAINT ON NRC REGULATORY AUTHORITY
1985	STAFF POSITION	° CONTAMINATED INJURED = EXPOSED ° MINIMAL GENUINE ARRANGEMENTS ARE NECESSARY

MEDICAL SERVICES PAPER

OPTION (1): ELIMINATE EXPOSED INDIVIDUALS FROM THE PLANNING STANDARD.

- CLINICAL COURSE FOR EXPOSURE UNFOLDS OVER TIME (LINNEMANN)
- TRAUMATIC INJURY TAKES PRECEDENCE (TRIAGE)
  - ∴ EMERGENCY MEDICAL SERVICES FOR EXPOSURE ARE NOT WARRANTED
- NUREG-0654 LIST FACILITATES AD HOC ARRANGEMENTS

OPTION (2): ADDITIONAL ARRANGEMENTS.

- NEED FOR BASELINE AND FOLLOW UP BLOODWORK
- NEED TO TREAT PRECURSORY SYMPTOMS
- PLANNING WOULD FACILITATE DEFINITIVE (LONG-TERM) TREATMENT OF EXPOSURE
- PLANNING WOULD FACILITATE TREATMENT OF CONTAMINATED/INJURED
- PLANNING WOULD FACILITATE TRANSPORTING VICTIMS

OPTION (3): LIST OF MEDICAL FACILITIES

- PROVIDES A PLANNING BASE FOR AD HOC ARRANGEMENTS
- DOES NOT ENSURE COOPERATION
- DOES NOT ENSURE PROPER TRAINING
- DOES NOT ENSURE AVAILABILITY OF TRANSPORTATION
- DOES NOT DEMONSTRATE CAPABILITY

DRAFT PROPOSAL FOR MEDICAL SERVICES

The NRC staff and FEMA would issue a guidance memorandum that would contain the following points from NUREG-0654/FEMA-REP-1, Rev. 1:

1. For Planning Standard A, "Assignment of Responsibility (Organization Control)," require a letter or "signature page" agreement between licensees and state and local governments and the service organizations concerning their ability and willingness to transport or treat contaminated injured and exposed individuals. This already has been applied to licensees for the medical services in the onsite plans.

2. For Planning Standard L, "Medical and Public Health Support."

   (a) Emphasize Evaluation Criterion 1 for state and local governments, which requires that they arrange for local and backup hospital facilities for medical services having the capability for evaluation of radiation exposure and uptake, including assurance that persons providing these services are adequately prepared to handle contaminated individuals.

   (b) Continue emphasis on Evaluation Criterion 3 (list of hospitals, capacity and special radiological capabilities) with the possible addition of pertinent accreditation such as an NRC or state license to posesss radioactive materials, and an indication of their ambulatory/nonambulatory capacity to treat radioactively contaminated injured (including exposed) individuals.* Emphasize the need for general statements defining special radiological capabilities for the evaluation of exposed and contaminated individuals.

   (c) Continue emphasis on Evaluation Criterion 4 for licensees and state and local governments, which requires that they arrange for transporting victims of radiological accidents to medical support facilities.

3. Emphasize that Planning Standard N, "Exercises and Drills," Criterion 2(c), "Medical Emergency Drills" includes local governments and their primary support hospitals.

4. For Planning Standard O, "Radiological Emergency Response Training," emphasize Evaluation Criterion 4(h) for state and local governments to ensure training of medical support personnel.

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*These two items, as indication of accreditation and capacity to treat contaminated injured (or exposed) individuals, are the only items not found in the existing guidance.

3

Attachment 2
to Enclosure 2

Compilation of CRGR Member Recommendations for Changes to Proposed Policy Statement

1. There are serious legal issues. OGC must review and approve.

2. The proposal should contain a discussion of why staff concludes that tech specs as established by new criteria would satisfy the current statutory standards.

3. The proposed program to improve tech spec language and basis should be recognized as risk beneficial, not risk neutral.

   Staff paper should better support the proposed need for any NRC staff review and approval of plant or procedure changes.

4. Objectives (targets) of policy statement both long- and short-term should be more clearly and separately identified. NRC justification should be more explicit on the problems and safety benefits of tech spec reform. Should better treat the selection of this approach among alternatives.

5. The proposition that paper processing is substantially reduced is flawed. If 50.59 reviews are improved as intended by TSIP, difference in current versus future paper work is minimal. The paper generally attempts to claim great value in the wrong areas. Should correct this.

K/3

6. The paper does not adequately treat the 50.59 problem. A Regulatory Guide or similar guidance document will not alone compensate for basic flaws in the current 50.59 regulation. Believes that revision to 50.59 is necessary.

7. Policy statement grammar and diction are weak and ambiguous. Paper should be improved by OGC's review.

8. If program is voluntary, the plant owners who most need it are the least likely participants. This should be clearly presented to the Commission.

9. Proposed policy statement should more clearly state that the staff would strongly pursue long-range tech spec improvements as well as the short-term relocations of tech specs.

10. The five recommendations of NUREG-1024 should each be addressed explicitly in the proposal. To the extent that the policy statement falls short of the five recommendations of NUREG-1024, the policy statement is not responsive to the SECY-86-10 commitment.

11. Commenter believes that CRGR should endorse a policy statement to Commission for public comment, with certain caveats:

    a. CRGR should monitor implementation of policy statement to determine whether plants known to have problems related to tech specs are participating in the program.

b. NRR should assure that each of five recommendations in NUREG-1024 is implemented by owners' groups proposed tech spec revisions.

c. Commission should receive periodic progress reports in which the need for further regulatory remedies is evaluated.

(3)

Enclosure 3 to the Minutes of CRGR Meeting No. 98
Review of Proposed Resolution for USI A-46
Seismic Qualification of Equipment in Operating Plants

T. Speis, R. Bosnak, N. Anderson, and T.Y. Chang (NRR) presented for CRGR review the proposed resolution for Unresolved Safety Issue (USI) A-46. This USI is concerned with the seismic qualification of equipment in older operating plants whose designs were not reviewed to current seismic qualification criteria in their initial licensing reviews. The actions proposed at this time by the NRC staff for resolving USI A-46 represent the final recommendations of the staff in this matter; they reflect public comments on the draft USI A-46 resolution package issued for comment by the staff in September 1985. (That draft USI A-46 package was considered by CRGR at Meeting No. 78 on July 8, 1985.) The proposed resolution for USI A-46 relies heavily on the use of seismic experience data gathered from non-nuclear power plants and other facilities which have experienced strong motion earthquakes, and on seismic test experience data developed in connection with the licensing of newer nuclear power plants. It is proposed that seismic qualification of equipment in the older nuclear power plants be established by suitable comparison with these types of experience data, rather than by application of more formal and detailed current licensing criteria. Copies of the briefing slides used by the NRR presenters of this proposal in discussions at this meeting are attached. (See Attachment 1 to this Enclosure.)

The package submitted by NRR for CRGR review in this matter was transmitted by memorandum dated October 1, 1986, Harold R. Denton to James Sniezek; that package included the following documents:

1. NUREG-1030, circa August, 1986, "Seismic Qualification of Equipment in Operating Nuclear Power Plants."

2. NUREG-1211, undated, "Regulatory Analysis for Proposed Resolution of Unresolved Safety Issue A-46, Seismic Qualification of Equipment in Operating Plants," with attachments:

   a) Appendix A - Draft Generic Letter, undated, "Verification of Seismic Adequacy of Mechanical and Electrical Equipment in Operating Reactors, Unresolved Safety Issue (USI) A-46," and enclosure "Seismic Adequacy Verification Procedure."

   b) Enclosure - "Operating Plants To Be Reviewed to USI A-46 Requirements."

3. ACRS Letter, dated September 17, 1986, "ACRS Comments on the Resolution of USI A-46..."

K/4

Additional related documents provided to the CRGR in connection with their review of the proposed resolution for USI A-46 included the following:

4. Summary of NRC staff justification for proposed action as required by CRGR Charter, "Background Information for CRGR Review of USI A-46 Resolution." (See Attachment 2 to this Enclosure.)

5. Letter, dated September 23, 1986, Nuclear Utility Group on Equipment Qualification to James Sniezek, CRGR Chairman, re: NRC Staff Proposed Resolution of Unresolved Safety Issue A-46, Seismic Qualification of Equipment in Operating Nuclear Power Plants. (See Attachment 3 to this Enclosure.)

Major points of discussion at this meeting regarding the resolution proposed by the NRC staff for USI A-46 were as follows:

1. The Committee commended the staff for the innovative approach taken in developing the proposed resolution for USI A-46, and cited the work done by the staff and the utilities in addressing this complex safety issue as an excellent example of the type of cooperative effort encouraged by the Commission.

2. J. Richardson (RES) gave a brief presentation to the Committee regarding the Seismic Margins Research Program. He noted specifically how that program and the USI A-46 program complemented each other, and how the two efforts were carefully coordinated so as not to duplicate efforts in the collection of data needed for completing successfully those programs. Both programs are aimed at better understanding how well existing nuclear plants can sustain seismic events. But whereas the A-46 program is concerned specifically with plants surviving an SSE, the Seismic Margins program is focusing on the regime of seismic events greater than SSE. In particular, Mr. Richardson noted that the work done in collecting the seismic experience data that forms a principal basis for the proposed resolution for USI A-46 contributed significantly, and has provided high confidence, in establishing lower bound fragility limits for nuclear power plant equipment.

3. A recently-issued report on AEOD's evaluation of operating experiences that involved discoveries of potentially serious seismic vulnerabilities in existing plants was given considerable discussion at this meeting. The report was only made available to the staff and CRGR the day of the meeting (see Attachment 4 to this Enclosure); so there had not yet been the opportunity to fully evaluate its possible implications in the context of USI A-46 and the proposed resolution being reviewed by the Committee at this meeting. Neil Thomasson (AEOD), the author of the report, noted to the Committee, however, that its findings and conclusions were consistent with and supportive of those of the NRC staff in NUREG-1030. Specifically, although the inherent seismic ruggedness of equipment in operating plants appears adequate if properly installed, enough examples of construction defects that result in unacceptable seismic vulnerability

have been found in existing plants to justify the walkdowns/anchorage reviews required by the staff in the proposed USI A-46 resolution. Both AEOD and RES supported NRR on this point; and this was an important point to be made, because this recommendation has been challenged by some in the industry who contend that reasonable assurance of seismic qualification of the older plants is provided by what has already been done in the A-46 program. (See #4 and #5 below for further detail.) In conclusion on this point, the staff agreed to evaluate further the AEOD report in question to determine if there were any significant implications to the proposed USI A-46 resolution, beyond the general support already recognized and discussed at this meeting. To some, the configuration control-type problems indicated by the examples cited in the AEOD report, and other past programs such as SEP, raised the question of whether additional requirements were necessary for newer plants as well as older plants (analogous to the proposed new requirements for older plants in the USI A-46 resolution, but not necessarily restricted to seismic qualification).

4. The Committee noted receipt of a letter from the Nuclear Utility Group on Equipment Qualification (EQ Group) in which the EQ Group challenged imposition of the proposed USI A-46 resolution as an information request under the provisions of 10 CFR 50.54. The Committee agreed with the EQ Group's position on that point, and expressed strongly their view that the proposed resolution is a backfit and ought to be treated as such in a straightforward manner under the backfit rule, 10 CFR 50.109. The basis for this view was that under the proposed resolution the adequacy of the design of a licensee's facility would be judged against significantly different criteria than were used by the staff in licensing the facility initially (i.e., the staff's position regarding what had to be done by the licensee to satisfy the requirements of the regulations had changed significantly and become more stringent). These were clearly the type of circumstances contemplated by the Commission in approving the backfit rule. Secondly, the time and expense involved in developing the seismic experience and seismic testing data bases, and the analyses involved in qualifying the equipment in the older plants by comparison with that data, is clearly greater than the "information request" contemplated by the Commission in approving the section of 10 CFR 50.54 allowing such requests without resort to the complete working of the process called for in the the backfit rule. (The work already done by the industry in that regard has covered several years and cost several tens of millions of dollars; and the final cost of implementation of the proposed resolution is in the 50-100 million dollar range.) The Committee view was supported by an OGC representative in attendance at this meeting specifically for the purpose of addressing this point.

   In view of these considerations, and in further view of the fact that, in its earlier consideration of the A-46 matter, the Committee had supported the staff's proposed USI A-46 resolution as a backfit, the Committee questioned why the staff had chosen now to call this action not a backfit. The staff simply responded that, in the view of some, in circumstances such as this (where a reasonable doubt exists regarding the adequacy of a facility and something must be done to resolve the doubt, and it is to the

advantage of licensees to do as much as possible by analysis rather than by facility modification toward resolving the doubt, as has been done in this case), application of the 10 CFR 50.54 information request provision seems appropriate. In retrospect, the staff agreed this determination should be reexamined and committed to obtain formal OGC review of the package as a whole. The backfit analysis (if necessary) will not, however, be substantively different than the regulatory analysis already provided; specifically, it will not be any more quantitative or extensive. The Committee agreed that it is not necessary to prepare a substantively different analysis to satisfy the backfit analysis requirement. The Committee believes that the extensive analyses done independently by the staff, and the cooperative effort of the staff with the utilities and industry over the past several years in further exploring more cost effective resolution of this complex issue, represent an appropriate effort by the staff not to impose an unnecessary burden on licensees in resolving this difficult issue, and is in full accord with the spirit and intent of the backfit rule and evaluation process that has been put in place by the Commission. In this context, the Committee reemphasized to NRR the importance of obtaining formal OGC review and concurrence on packages coming to CRGR for review (as required by the Committee Charter), particularly in the case of proposed actions at the final stage.

5. The Committee took note of a letter submitted to CRGR by the Seismic Qualification Utility Group (SQUG) at the time of the Committee's earlier consideration of the A-46 matter at the draft resolution stage. The basic position stated clearly by SQUG in that letter, and reaffirmed recently in conversations with the ROGR staff, on the A-46 matter is that reasonable assurance of the seismic adequacy of equipment in older operating plants is already provided on the basis of the work already completed in the A-46 program. The staff's position in this matter, as given in the package provided to CRGR for review, is stated somewhat more equivocally. For example, the Committee noted that at some points in the package (e.g.,NUREG-1030, Sec.1.1, p.1-1; NUREG-1211, Sec.I, p.1; and in the Draft Generic Letter, Paragraph 1, p.1) the staff states that the seismic adequacy of equipment must be reassessed (as prescribed in the proposed USI A-46 resolution) to ensure its safe shutdown capability. At other points in the package (e.g., NUREG-1030, Paragraph 1, p.iii; and NUREG-1211, Paragraph 1, p.iii), the staff says that their seismic adequacy should be reassessed. Noting that the terms "must" and "should" have significantly different connotations in the regulatory context, the Committee inquired whether these seeming equivocations reflected doubts on the part of the staff that the substantial backfit involved in implementing the proposed USI A-46 resolution was really necessary for reasonable assurance of safety, and that it could be justified as such in accordance with 10 CFR 50.109.

The staff responded unequivocally that it is their judgment that the measures prescribed in the proposed resolution are necessary to provide assurance of the seismic adequacy of equipment in the older operating plants to which this action is directed, and that those measures are fully justified and are cost effective in view of what has been learned from all

of the A-46 work taken together (i.e., both by the staff and the industry).

6. The Committee asked the staff to address comments on the proposed USI A-46 resolution that were provided by the ACRS in a letter dated September 17, 1986 (see Background Item #3 above). The staff was somewhat surprised by those comments, because the ACRS had generally expressed support for the approach being taken by the staff to resolution of this issue in earlier discussions. The staff felt that the recommendations made by the ACRS at this stage represented a considerable broadening of the scope that had been contemplated and discussed with them all along, and in some cases involved treatment of issues already addressed reasonably in other contexts (e.g., flooding, fires, and inadvertent actuation of fire protection systems). The staff has not dismissed the ACRS' concerns, however; and another meeting is planned in early November to discuss staff's plans for addressing their additional concerns. Generally, the staff does not intend to expand the scope of A-46 to accommodate the ACRS' concerns; rather, those concerns will be evaluated and prioritized on their own merits, and will be addressed further if, on the basis of that process, it is judged necessary and cost effective to do so.

   In exploring further some specific concerns of ACRS, there was much discussion regarding precisely what is included in the current scope of USI A-46. With regard to seismic induced physical interactions, the wording of the review package at some points seems to encompass at least some of the areas that ACRS indicated as lacking. After much discussion on this point, it was established that the scope of equipment and support services to be examined for seismic-induced physical interactions during the planned walkdown inspections prescribed under the proposed resolution includes equipment needed for hot shutdown and all connected services (e.g., air, electric power, hydraulics) out to the first anchor point. But the difficulty experienced in arriving at this understanding prompted the CRGR to wonder if some of the ACRS concerns did not simply result from similar difficulties (and frustration) that the ACRS may have experienced in similar discussions with the staff in trying to fully understand the intended scope. The Committee encouraged the staff to make another effort to work out differences with ACRS on just that basis; and they further recommended that descriptions of scope throughout the package be reviewed and revised as necessary to clarify specifically the scope questions discussed with CRGR at this meeting. The staff agreed to do so.

   One concern raised by the ACRS that clearly did not appear to be the result of any misunderstanding is the exclusion of long piping runs between individual pieces of equipment needed for safe shutdown from the intended scope of USI A-46 walkdown inspections. In response to direct questions by CRGR, the staff acknowledged that this piping will not be included in the walkdown inspections to look for seismic-induced physical interactions (e.g., due to one pipe, or other heavy object, falling on another or impacting another due to seismic-induced failure or vibration) The Committee inquired regarding the staff's basis for excluding such piping and such interactions from the A-46 scope. The staff responded that it was a matter of attempting to reasonably bound what is otherwise

potentially an unbounded and unboundable effort (i.e., attempting to treat all possible interactions). It was noted that the Indian Point seismic systems interaction study did treat such interactions within its scope; and a number of such interactions were found. None of the interactions of this type found at that facility, however, was ultimately determined to have major risk significance.

7. The Committee inquired regarding the status of the various procedures, guidance, and other information still being developed or gathered (e.g., for verifying equipment anchorage, for identifying data base exclusions and caveats, and for equipment types not included in the available experience data bases) that will be needed by licensees in responding to the staff regarding the schedule for completion of the reassessment prescribed under the proposed resolution. It appears that all such information and guidance may not be available to licensees at the time projected by the staff for issuance of the Generic Letter that will be used in implementing the proposed resolution. This could be the limiting consideration (especially for licensees who have not participated in the SQUG effort all along) in determining whether licensees can respond on the schedule specified in the proposed Generic Letter. The Committee recommended, therefore, that the wording of the proposed Generic Letter (in the final paragraph on p.2) be changed to indicate that schedules for implementation of the (proposed) USI A-46 resolution be provided within 60 days of the final availability of all associated guidance. The staff agreed to make the recommended revision.

8. The Committee questioned the staff's attempt to quantify the benefits to be gained from implementation of the proposed resolution, as given in Background Item #4 above. Some of the specific criticisms were as follows:

   a) The basis for the staff's estimate of a 2350 man-rem/reactor risk benefit is not clear; and it is also not clear that that estimate is consistent with the BNL conclusions regarding risk benefit of the proposed action as summarized at p.29 of NUREG-1211.

   b) The conclusion that equipment in the older operating plants is inherently rugged seems inconsistent with the estimate of risk benefit given in Background Item #4 (which appears to correspond roughly to an order of magnitude improvement in core melt frequency).

   The staff response to b) above was that the inherent ruggedness finding is not necessarily inconsistent with the expectation of significant improvement in safety and/or risk to be realized from implementing the proposed resolution, because that finding involves an important caveat i.e., the inherent ruggedness of equipment in the older plants, as demonstrated by experience data on similar equipment in nonnuclear facilities, provides reasonable assurance of seismic adequacy of those plants, if anchorage deficiencies, relay chatter problems, etc. are identified and corrected in accordance with the proposed resolution. The staff noted that the additional attempts at quantification in this package

were included only because the staff felt considerable pressure to do so in preparing their case for office review prior to submittal to CRGR. The Committee reiterated that it is not considered necessary by CRGR that the case for a proposed action be made solely or even principally on the basis of quantitative arguments, if the factors that must be considered do not lend themselves to quantification, as is the case in treatment of A-46 issues.

RECOMMENDATION TO THE EDO

On the basis of their review of the USI A-46 matter and the discussions with the staff at this meeting, the Committee recommended that the proposed resolution for USI A-46 be approved for implementation, subject to:

1. Legal review and concurrence by OGC (required by the CRGR Charter), as noted in the preceding;

2. Review by NRR of the AEOD report noted in the preceding and evaluation of its possible implications in the USI A-46 context.

3. Evaluation and prioritization of ACRS concerns, as noted in the preceding;

4. Revisions/clarifications of the wording of the package, as noted in the preceding, regarding the scope, schedule for implementation, and basis in the regulations of the measures to be imposed by the staff in this action.

5. Review by the CRGR Chairman of the changes made to the package.

ATTACHMENT 1 to
ENCLOSURE 3

USI A-46
SEISMIC QUALIFICATION OF EQUIPMENT IN OPERATING PLANTS
PRESENTATION TO CRGR
OCTOBER 16, 1986

THEMIS SPEIS
ROBERT BOSNAK
NEWTON ANDERSON
T. Y. CHANG

Speis

USI A-46 OVERVIEW

- o TECHNICAL RESOLUTION COMPLETE
- o ALL PUBLIC COMMENTS ~~RESOLVED~~ ADDRESSED
- o IMPLEMENTATION PROCEDURES UNDER DEVELOPMENT BY SQUG/EPRI
- o NRC OFFICE LEVEL (NRR, IE, RES, AEOD) CONCURRENCE WITH PROPOSED RESOLUTION
- o ACRS IN AGREEMENT WITH APPROACH, RAISED CONCERNS ON SCOPE
- o NEED CRGR RECOMMENDATION TO ISSUE FINAL RESOLUTION

5 pris

BACKGROUND

- o TASK A-46 DESIGNATED AS USI IN DECEMBER 1980
- o LOOK FOR ALTERNATIVES TO CURRENT SEISMIC QUALIFICATION REQUIREMENT
- o 5 TASKS IDENTIFIED, USE OF SEISMIC EXPERIENCE DATA PROVED TO BE THE MOST REASONABLE ALTERNATIVE
- o SQUG PILOT PROGRAM (1982)
- o SSRAP FORMED 1983
- o ADDITIONAL SEISMIC EXPERIENCE DATA COLLECTED BY SQUG Now 20 classes
- o COLLECTION OF TEST EXPERIENCE DATA BY EPRI TO SUPPLEMENT SEISMIC EXPERIENCE DATA
- o PROPOSED STAFF POSITION BASED ON USE OF SEISMIC EXPERIENCE AND TEST EXPERIENCE
- o MET WITH CRGR IN MEETINGS NO. 71 (12/3/84) AND NO. 78 (7/8/85). HAD CONCERNS ON RELAY EVALUATION.
- o CRGR RECOMMENDED (7/8/85) ISSUANCE OF A-46 PROPOSED RESOLUTION FOR PUBLIC COMMENT

Speis

THREE MAJOR CONCERNS

1. EQUIPMENT ANCHORAGES

2. RELAY FUNCTIONAL CAPABILITY

3. OUTLIERS

PROPOSED RESOLUTION

- o OPERATING PLANTS
  - \- PERFORM WALK THROUGH INSPECTION:
    VERIFY SEISMIC ADEQUACY OF ANCHORAGES
    REVIEW SEISMIC SYSTEMS INTERACTION
    IDENTIFY & ADDRESS DEFICIENCIES AND OUTLIERS
  - \- VERIFY FUNCTIONAL CAPABILITY OF EQUIPMENT (RELAYS)
- o NEW LICENSEES
  - \- NO REQUIREMENTS
- o IMPLEMENT BY GENERIC LETTER - REPLY IN 60 DAYS
- o PROVISION FOR GENERIC IMPLEMENTATION AND FOR INDIVIDUAL UTILITY IMPLEMENTATION
- o PROVISION FOR REPLACEMENT PARTS

SCOPE OF SEISMIC ADEQUACY REVIEW

- o ASSUMPTIONS
  - SSE SEISMIC INPUT
  - SSE DOES NOT CAUSE LOCA, HELB OR MSLB
  - LOCA, HELB OR MSLB DOES NOT OCCUR SIMULTANEOUSLY WITH OR DURING SSE
  - OFFSITE POWER MAY BE LOST DURING OR FOLLOWING SSE
- o MAINTAIN HOT SHUTDOWN FOR A MINIMUM OF 72 HOURS
- o EQUIPMENT SCOPE
  - ACTIVE ELECTRICAL AND MECHANICAL COMPONENTS INCLUDING INSTRUMENTATION AND CONTROLS NEEDED TO ACHIEVE AND MAINTAIN HOT SHUTDOWN
  - ANCHORAGES ON TANKS, HEAT EXCHANGERS REQUIRED TO ACHIEVE AND MAINTAIN HOT SHUTDOWN
  - CABLE TRAYS AND CONDUITS
  - SEISMIC SYSTEMS INTERACTION
- o PLANTS AFFECTED
  - OPERATING PLANTS NOT REVIEWED TO CURRENT CRITERIA AS DOCUMENTED BY SER'S
  - ABOUT 70 UNITS
  - MOST SEP PLANTS WILL BE REVIEWED FOR FUNCTIONAL CAPABILITY ONLY

5pics

ACRS COMMENTS

7/83, 5/84, 8/85

COMMENTS - ENDORSED THE APPROACH, ASKED TO BE KEPT INFORMED

1/86

PRESENTED RESOLUTION OF PUBLIC COMMENTS

8/86, 9/86

CONCERNS ON SCOPE

- o ADD SEISMICALLY INDUCED FIRES
- o EXTEND SEISMIC SYSTEMS INTERACTION TO FLOODING
- o INADVERTENT ACTUATION OF FIRE PROTECTION SYSTEMS
- o CONSIDER MULTIPLE EVENTS
- o ADD LOW ENERGY AND SMALL BORE PIPING

STAFF RECOMMENDATION

- o PROCEED WITH PROGRAM WITH CURRENT SCOPE
- o ADDRESS ACRS CONCERNS SEPARATELY

GENERAL IMPLEMENTATION GUIDELINES

- o SQUG/EPRI DEVELOP GIP
- o SSRAP/NRC REVIEW AND APPROVE
- o TRIAL PLANT REVIEWS
- o FINE TUNE PROCEDURES
- o WORKSHOPS FOR PARTICIPATING UTILITIES AND NRC REVIEWERS
- o PLANT SPECIFIC REVIEWS

NRC RELAY REVIEW GUIDELINES

- o IDENTIFY ESSENTIAL FUNCTIONS, SYSTEMS, CIRCUITS AND RELAYS
- o RELAYS WHICH MUST FUNCTION DURING STRONG SHAKING:
  - VERIFY WITH TEST DATA
  - REPLACE WITH QUALIFIED RELAYS
  - QUALIFY BY TEST
  - OTHER ALTERNATIVES
- o RELAYS WHICH MUST FUNCTION AFTER STRONG SHAKING:
  - VERIFY, REPLACE OR QUALIFY AS ABOVE
  - SHOW CHATTER OR CHANGE OF STATE DOES NOT AFFECT PLANT SHUTDOWN
  - SHOW OPERATOR CAN RESET
  - OTHER ALTERNATIVES

SQUG RELAY EVALUATION PROCEDURE

- METHODOLOGY FOR PLANT-SPECIFIC IDENTIFICATION OF THOSE SYSTEMS AND ASSOCIATED RELAYS WHICH MUST REMAIN FUNCTIONAL DURING AND/OR AFTER EARTHQUAKE

- STEP-BY-STEP PROCEDURE TO:

  - SCREEN SYSTEMS, CIRCUITS AND RELAYS TO IDENTIFY RELAYS WHICH SERVE ESSENTIAL EQUIPMENT

  - ELIMINATE FROM FURTHER EVALUATION THOSE CIRCUITS/RELAYS (1) WHOSE OPERABILITY IS NOT NEEDED TO ACHIEVE ESSENTIAL FUNCTIONS, (2) WHOSE MALFUNCTION WILL NOT PREVENT PERFORMING THE ESSENTIAL FUNCTION, AND (3) WHOSE ACTUATION CAN BE CORRECTED BY OPERATOR

  - IDENTIFY THOSE CIRCUITS/RELAYS WHOSE ACTUATION COULD LEAD TO OPERATION OF EQUIPMENT WHICH SHOULD NOT OPERATE

  - FOR REMAINING ESSENTIAL RELAYS, DETERMINE TYPE, LOCATION AND SEISMIC REQUIREMENTS

- ASSESS THE SEISMIC ADEQUACY OF THE IDENTIFIED ESSENTIAL RELAYS BY COMPARISON WITH TEST DATA (GERS) AND EXPERIENCE DATA

ON-GOING DEVELOPMENT OF IMPLEMENTATION PROCEDURES

SQUG

- SEISMIC ADEQUACY OF EQUIPMENT
- GENERIC IMPLEMENTATION PROCEDURE (GIP)
- RELAY SEISMIC EXPERIENCE DATA
- CABLE TRAY/CONDUIT EXPERIENCE DATA

EPRI

- EQUIPMENT ANCHORAGE REVIEW GUIDELINES
- COLLECT TEST EXPERIENCE DATA AND PRESENT IN THE FORM OF GENERIC EQUIPMENT RUGGEDNESS SPECTRA (GERS)
- RELAY EVALUATION PROCEDURES

STAFF IMPLEMENTATION PLANS

- o RELAY REVIEW TEAM
- o ANCHORAGE REVIEW TEAM
- o LICENSING REVIEW BRANCHES
- o PARTICIPATION IN IMPLEMENTATION WORKSHOPS
- o SQUG, SSRAP AND NRC AUDIT
- o ISSUE PLANT SPECIFIC SERs

ORIGINATORS OF PUBLIC COMMENTS

- o 8 UTILITIES: 7 SQUG MEMBERS
- o 2 INDUSTRY GROUPS: - SQUG

  - NUCLEAR UTILITY GROUP ON EQUIPMENT QUALIFICATION
- o 1 NATIONAL LABORATORY: - SANDIA
- o EPRI
- o AIF

CATEGORIZATION OF COMMENTS

- APPLICABILITY OF BACKFIT RULE
- JUSTIFICATION FOR A-46 REVIEW (10 COMMENTS)
- IMPLEMENTATION SCHEDULE (12)
- RELAY REVIEW GUIDELINES (10)
- SCOPE OF REVIEW (10)
- SCOPE OF WALK-THROUGH INSPECTION (8)
- REQUIREMENT FOR JUSTIFICATION FOR CONTINUED OPERATION (5)
- COST ESTIMATE (5)
- GUIDELINES FOR REPLACEMENT EQUIPMENT (4)
- SAFE SHUTDOWN REQUIREMENT (4)
- EQUIPMENT SEISMIC DEMAND AND SEISMIC CAPACITY (6)
- MAKE-UP OF WALK-THROUGH INSPECTION TEAM (4)
- EXPANSION OF SEISMIC EXPERIENCE DATA BASE (3)

- ROLE OF SQUG IN GENERIC IMPLEMENTATION (3)
- ACCESSIBILITY OF SQUG RESULTS TO NON-SQUG MEMBERS (2)
- PLANT SPECIFIC SERS (2)
- APPLICABILITY OF A-46 TO NEW PLANTS/NEW EQUIPMENT (1)
- APPLICABILITY OF A-46 TO SPECIFIC PLANTS (2)

<u>APPLICABILITY OF BACKFIT RULE AND JUSTIFICATION FOR A-46 REVIEW</u>

o COMMENTS:

(1) THE NEW BACKFIT RULE SHOULD BE APPLIED TO THE RESOLUTION OF A-46

(2) REGULATORY ANALYSIS DOES NOT PROPERLY <u>QUANTIFY</u> COST BENEFIT

(3) EARTHQUAKE EXPERIENCE SHOWS THAT EQUIPMENT IS INHERENTLY RUGGED THEREFORE NO NEED TO DO REVIEW

o RESPONSE:

(1), (2) - STAFF POSITION IS THAT BACKFIT RULE DOES NOT SPECIFICALLY APPLY TO REQUIREMENTS TO REVIEW

- BACKFIT RULE DOES APPLY TO CORRECTING ANY DEFICIENCIES UNCOVERED DURING REVIEW

(3) - ACKNOWLEDGE EQUIPMENT IS INHERENTLY RUGGED, STILL HAVE AREAS OF CONCERN:

(A) EQUIPMENT ANCHORAGE

(B) FUNCTIONAL CAPABILITY OF ESSENTIAL RELAYS

(C) OUTLIERS

WE BELIEVE SQUG IN AGREEMENT

USI A-46 SCHEDULE

- o FINAL ISSUANCE OF RESOLUTION AND GENERIC LETTER IN NOVEMBER 1986

- o SQUG PROPOSED IMPLEMENTATION SCHEDULE:

  - ANCHORAGE GUIDELINES COMPLETED
  - COMPLETE SEISMIC AND TEST EXPERIENCE DATA BASE DEVELOPMENT - LATE 1986
  - RELAY EVALUATION PROCEDURE ABOUT 80% COMPLETE, TRIAL USE - LATE 1986
  - GIP UNDER DEVELOPMENT, TRIAL USE - LATE 1986/1987
  - TRAINING WORKSHOPS - LATE 1987/EARLY 1988
  - PLANT SPECIFIC IMPLEMENTATION - START 1988

NRC SEISMIC MARGINS RESEARCH PROGRAM

CRGR BRIEFING

OCTOBER 16, 1986

J. E. RICHARDSON

ISSUES

- o NEW SEISMOLOGICAL INFORMATION
  - USGS CLARIFICATION OF CHARLESTON EARTHQUAKE
  - NEW BRUNSWICK EARTHQUAKE
  - NRC CHARACTERIZATION OF EASTERN U.S. SEISMICITY
  - EPRI/INDUSTRY SEISMICITY PROGRAM

- o NEED TO ESTIMATE PLANT MARGIN TO WITHSTAND LARGER EARTHQUAKES
  - ACCOMMODATE CHANGES IN SITE SPECIFIC HAZARD
  - IDENTIFY PLANT VULNERABILITIES TO EARTHQUAKE HAZARD
  - ADDRESS ACRS CONCERNS REGARDING SEISMIC MARGINS
  - EVALUATE NEED FOR SEISMIC CRITERIA BACKFIT OR DEFICIENCY CORRECTIONS

RESEARCH OBJECTIVES

- DEVELOP SIMPLIFIED PROCEDURES TO ESTIMATE SEISMIC MARGINS
- DEVELOP FRAGILITY DATA BASE ←
- DEMONSTRATE METHODS BY ESTIMATING MARGINS FOR ACTUAL PLANTS
- SUPPORT IMPLEMENTATION OF SEVERE ACCIDENT POLICY

STATUS

- PROCEDURES DEVELOPED FOR PWRS NUREG/CR-4334 NUREG/CR-4482
- REVIEW OF MAINE YANKEE IN PROGRESS (COMPLETED SPRING 1987)
- DEVELOP PROCEDURES FOR BWR (FY 1987)
- REVIEW OF A BWR (FY 1987-1988)
- CONTINUE TO IMPROVE DATA BASE (FY 1987-1989)

Background Information for CRGR Review of USI A-46 Resolution

The following information is provided in the format specified in the CRGR Charter, Revision 3, dated September, 1986. For each item, the request for information is given followed by a discussion of the response or a reference as to where the information is provided. Further supporting information is contained in the enclosed "Regulatory Analysis for Resolution of USI A-46 (NUREG-1211)."

(1) The proposed generic requirement or staff position as it is proposed to be sent out to licensees.

The proposed generic requirements are set forth in the proposed generic letter (See Appendix A of NUREG-1211).

(2) Draft staff papers or other underlying staff documents supporting the requirements or staff positions.

The relevant technical information is contained in the enclosed NUREG-1030 and related references listed therein. Copies of any references will be provided upon request.

(3) Each proposed requirement or staff position shall contain the sponsoring office's position as to whether the proposal would increase requirements or staff positions, implement existing requirements or staff positions, or would relax or reduce existing requirements or staff positions.

Depending on the licensing basis of the particular operating plant in consideration, the proposed requirement (or staff position) would either increase existing requirements (or staff positions) or impelement existing requirements (on staff positions). In general, the staff believes the plants in consideration will fall in the former category.

(4) The proposed method of implementation along with the concurrence (and any comments) of OGC on the method proposed.

The proposed method of implementation is through the issuance of a generic letter under the provisions of 10 CFR 50.54f.

OGC has been provided with a copy of the final CRGR package for review on August 25, 1986. The proposed generic letter is included as Appendix A of the Regulatory Analysis for Proposed Resolution of USI A-46 (NUREG-1211). The resolution of public comments has been incorporated in both NUREG-1211 and NUREG-1030 of the package as appropriate.

(5) Regulatory analyses generally conforming to the directives and guidance of NUREG/BR-0058 and NUREG/CR-3568.

The Regulatory Analysis for Proposed Resolution of USI A-46 (NUREG-1211) is enclosed.

(6) Identification of the category of reactor plants to which the generic requirement or staff position is to apply.

The A-46 review applies to all LWR operating plants that have not been reviewed to the current equipment seismic qualification requirements, as documented by plant Safety Evaluation Reports (SERs). The current requirements for seismic qualification of equipment used in licensing plants are defined in Regulatory Guide 1.100, Institute of Electrical and Electronics Engineers (IEEE) Standard 344-1975, and Standard Review Plant (SRP) Section 3.10 (NUREG-0800).

For most SEP plants, the structural integrity of equipment was generally covered under the SEP review; however, the evaluation of functional capability of equipment in all the SEP plants was left to the resolution of USI A-46. The scope of review should be established in accordance with each plant's Integrated Safety Assessment Report and related Evaluation Reports.

(7) For each such category of reactor plants, an evaluation which demonstrates how the action should be prioritized and scheduled in light of other ongoing regulatory activities. The evaluation shall document for consideration information available concerning any of the following factors as may be appropriate and any other information relevant and material to the proposed action:

(a) Statement of the specific objectives that the proposed action is designed to achieve;

The specific objectives of the proposed A-46 action is to develop viable, cost-effective alternatives to current seismic qualification licensing requirements to be applied to operating nuclear power plants, in order to verify the seismic adequacy of mechanical and electrical equipment which is required to safely bring the reactor and plant to a safe shutdown condition and to maintain it in a safe condition.

(b) General description of the activity that would be required by the licensee or applicant in order to complete the action;

The resolution of USI A-46 is based mainly on the use of seismic experience and test experience data.

The general verification procedure for plant-specific review is described in Section IV.4 of the enclosed Regulatory Analysis (NUREG-1211). Briefly, it includes the following steps:

- o development of an equipment list
- o comparison of site spectra with appropriate bounding spectra
- o walk through inspection, which includes anchorage review, seismic system interaction review, identification and review of deficiencies and outliers.

- o review of equipment functional capability
- o review of equipment unique to nuclear plants
- o replacement (and/or modification) of equipment and/or equipment supports.

(c) Potential change in the risk to the public from the accidental offsite release of radioactive material;

The safety benefit of verifying the seismic adequacy of equipment in operating plants was not quantified in terms of risk reduction. Quantifying the net safety benefit in terms of risk resulting form "qualifying" or verifying the seismic adequacy of equipment proved to be impractical. However, to demonstrate the potential value/impact, the staff has estimated the potential safety benefit in terms of the risk and the cost benefit potentially achievable (see Section V of the enclosed Regulatory Analysis, NUREG-1211). These estimates consider only the risk potential associated with failure of equipment anchorages. The analysis shows that, for a typical 1120 MWe PWR, meteorology typical of the Byron (mid-west) site, the public risk (R) from the initiating SSE and subsequent events is calculated to be

R = 2350 man-rem/reactor

This is also the risk reduction potential (safety benefit) from the A-46 verification program.

(d) Potential impact on radiological exposure of facility employees and other onsite workers.

No estimate was performed. However, it should be larger than the public risk of 2350 man-rem/reactor indicated in (c) above.

(e) Installation and continuing costs associated with the action, including the cost of facility downtime or the cost of construction delay;

The estimated costs to licensees for complying with the requirement of A-46 are presented in Section V.4 of the enclosed Regulatory Analysis (NUREG-1211). The estimated cost per plant ranges from $401,000 to $840,000. The cost of facility downtime is not included in this estimate. The implementation schedule will be negotiated with the licensees in accordance with the NRC policy on integrated schedules for plant modifications stated in Generic Letter 83-20 dated May 9, 1983. The proper integration of the proposed work scope into each plant's living schedule for plant modification will be considered.

(f) The potential safety impact of changes in plant or operational complexity, including the relationship to proposed and existing regulatory requirements and staff positions;

None.

(g) The estimated resource burden on the NRC associated with the proposed action and the availability of such resources;

The cost to the NRC for the implementation of A-46 requirement is estimated in Section V.5 of the enclosed Regulatory Analysis (NUREG-1211).

The principal cost to NRC (for utilities not participating in a generic implementation) would be the cost to review the reports submitted by individual licensees and for participation in the plant audits. About 70 plants would be required to submit reports. It would require 0.6 staff months to review each report and 0.5 staff months to prepare an SER, for a total expenditure of 77 staff months. At an estimated rate of $100,000 per staff year, the total cost would be $640,000.

If a generic program is implemented by SQUG or a similar utility group, the cost to the NRC would be substantially reduced.

(i) Whether the proposed action is interim or final, and if interim, the justification for imposing the proposed action on an interim basis.

The proposed action represents the final staff position on USI A-46.

(8) For each evaluation conducted pursuant to 10 CFR 50.109. the proposing Office Director's determination, together with the rationale for the determination based on the considerations of paragraphs (1) through (7) above, that

(a) there is a substantial increase in the overall protection of public health and safety or the common defense and security to be derived from the proposal; and

The staff position on 10 CFR 50.109 is that the new backfit rule does not apply to the requirement to perform a review. The backfit rule, however, does apply to any plant changes or modifications made as a result of the review (see Appendix D, of NUREG-1030).

In the case of equipment anchorage deficiency discovered and modified as a result of the review, the potential for overall protection of public health and safety is the reduction of risk of 2350 man-rem/reactor indicated in (7) (c) above.

(b) the direct and indirect costs of implementation, for the facilities affected, are justified in view of this increased protection.

The estimated value/impact ratio (V/I) is the ratio of the estimated potential risk reduction (2350 man-rem/reactor) and the estimated average costs of $600,000 (see Section V.1 of enclosed NUREG-1030, Page 17), which yields an estimated V/I ratio of

$$V/I = 3.9 \frac{\text{man-rem/reactor}}{\$1000 \text{ costs}}$$

Based on the stated assumptions in NUREG-1030 and a rule-of-thumb of [illegible] man-rem/$1000 as an acceptable measure of cost effectiveness, the A-46 program would be cost effective and justified.

(9) For each evaluation conducted for proposed relaxations or decreases in current requirements or staff positions, the proposing Office Director's determination, together with the rationale for the determination based on the considerations of paragraphs (1) through (7) above, that

(a) the public health and safety and the common defense and security would be adequately protected if the proposed reduction in requirements or positions were implemented, and

This question does not apply to A-46.

(b) the cost savings attributed to the action would be substantial enough to justify taking the action.

This question does not apply to A-46.

ATTACHMENT 3 to
ENCLOSURE 3

J.C.
relative to the A-46 CRGR meeting 9/23

NUCLEAR UTILITY GROUP
ON EQUIPMENT QUALIFICATION

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September 23, 1986

Mr. James Sniezek, Chairman
Committee to Review Generic Requirements
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Re: NRC Staff Proposed Resolution of Unresolved Safety Issue A-46, Seismic Qualification of Equipment in Operating Nuclear Power Plants

Dear Mr. Sniezek:

On August 14, 1986 representatives of the Nuclear Utility Group on Equipment Qualification ("Group") met with your Staff to discuss its July 22, 1986 letter, in which the Group questioned the adequacy of the Staff's Regulatory Impact Analysis prepared in connection with the proposed resolution of USI A-46. As a follow-up to the August 16, 1986 meeting, this letter will provide you with the Group's comments on whether that Regulatory Impact Analysis satisfies the Commission's amended regulations governing backfitting and information requests. See, respectively, 10 C.F.R. §§ 50.109 and 50.54(f).

The Staff had previously suggested that its proposed resolution of USI A-46, including the information request, constituted a backfit for which it was required to prepare an analysis pursuant to Section 50.109(a)(2). It has now become evident that the Staff believes (1) it is not required at this time to prepare a backfitting analysis pursuant to Section 50.109 because in its view the initial actions required of licensees by its proposed resolution of USI A-46 encompass only responding to an information request pursuant to Section 50.54(f) and (2) it has adequately justified that request.

The Group maintains that the proposed information request is a backfit and may not be issued until the completion or a backfitting analysis, as required by Section 50.109(a)(2). We understand that through the information request the Staff will require licensees to redetermine the seismic adequacy of plants already licensed for operation. Licensees will be expected to perform for a second time extensive analyses that will be evaluated against criteria differing from those

~~8611190068~~ 12pp
8pp.

applied during initial licensing, after which the Staff will render a new finding as to seismic adequacy. Depending on this finding, facility modifications may be required.

For the Staff to retract its previous acceptance of licensee analyses and to require the preparation of new analyses that will be reviewed against criteria differing from those previously applied constitutes a changed Staff position. This changed Staff position at the minimum will require licensees to use new and different procedures in order to assure that the requested analyses are properly completed for their plants. More importantly, the Staff is in effect asking licensees to re-establish the design adequacy of their facilities. Accordingly, the information request constitutes a backfit within the meaning of Section 50.109(a)(1).

The Group recognizes that the Staff has already prepared a Regulatory Impact Analysis regarding the proposed resolution of USI A-46. However, this analysis does not satisfy Section 50.109(a)(2) of the backfitting rule. Among other inadequacies, the Regulatory Impact Analysis fails to articulate any underlying bases on which the Staff's conclusions regarding the benefits of the proposal rest.

In any event, the Group believes that should the information request itself not constitute a backfit, the Regulatory Impact Analysis does not satisfy 10 C.F.R. §50.54(f). This information request involves considerable licensee (and NRC) resources and cannot be characterized as a routine request for information. The Staff has failed to justify this burden to be imposed on licensees in light of the importance of the safety issue to be addressed.

I. BACKGROUND

By letter dated November 15, 1985, the Group provided comments to the Staff on draft NUREG-1030 (August 1985) and its attachment, the Regulatory Impact Analysis and draft generic letter on USI A-46 (August 1985). The Group expressed its view that the proposed resolution of USI A-46 constituted a backfit and that the Staff should prepare a backfitting analysis. The Group stated that the Staff's Regulatory Impact Analysis failed to provide any underlying justification for its conclusion that the proposed inspection and verification program would result in significant safety improvements and that the backfitting rule required more than the preparation of a conclusory analysis. The Group recommended that the Staff perform a quantitative analysis but did not state that such an analysis was required by the backfitting rule.

On January 15, 1986 the Staff addressed this comment at a meeting of the ACRS Subcommittee on the Qualification Program for Safety-Related Equipment. Instead of responding directly to the Group's comment that its Regulatory Impact Analysis did not satisfy the backfitting rule, the Staff explained that "Quantitative PRA analysis for evaluating [the risk-reduction aspects of the proposal] has very high uncertainty and is highly dependent on assumptions . . . ." Accordingly, the Staff concluded that "we don't think a quantitative analysis is very meaningful here." Transcript at 5.1/

In its July 22, 1986 letter to you, the Group explained that it did not concur with the Staff's January 15 statement. The Group stated that (1) the backfit rule applies to plant modifications which result from the resolution of USI A-46; (2) the Staff must give meaningful consideration in its Regulatory Impact Analysis to the potential change in risk as a result of its proposed resolution of USI A-46; and (3) the Staff has not meaningfully addressed the risk reduction potential of the proposal, as required by Section 50.109. The Group requested CRGR review of this issue during its forthcoming deliberations on USI A-46.2/

On August 5, 1986 the Staff made a presentation on USI A-46 to the ACRS Subcommittee on Reliability Assurance. At the meeting, the Staff challenged directly and apparently for the first time the Group's comment that the backfitting rule applies to its proposed resolution of USI A-46. The Staff asserted, as follows:

> We believe the requirement to perform plant review is not subjected to the backfit rule. It is subject to § 50.54(f) [information request rule].

1/ Contrary to the Staff presentation to the ACRS on January 15, 1986, the Staff is not freed from its obligation under the backfitting rule to address the risk-reduction aspects of its proposal simply because it believes that PRA analysis would involve "high uncertainty," is "highly dependent on the assumptions" and is not "meaningful." Transcript at 6. The backfitting rule does not even mention, let alone require, the use of PRAs when preparing a backfit analysis. Moreover, the NRC has for years made judgments as to the need for new safety requirements without using PRAs.

2/ It is our understanding the CRGR is scheduled to review USI A-46 in mid-to-late September.

> However, during the review if some deficiencies are found, to fix the deficiencies, it's under the backfit rule.

Transcript at 32 (statements by Mr. Chang, Task Manager of USI A-46). In taking this new position, the Staff now claims that its Regulatory Impact Analysis is sufficient to meet the less stringent standards of Section 50.54(f).

Subsequently, on August 14, 1986, the Group met with members of your Staff to discuss the views expressed in the Group letter dated July 22, 1986, including the application of Sections 50.54(f) and 50.109 to the Staff's proposed resolution of USI A-46. This letter, submitted as a follow-up to the August 14 meeting, addresses (1) whether the "information request" under consideration constitutes a backfit within the meaning of Section 50.109(a)(1); (2) whether, if it does, the Staff has justified the backfit as required by Section 50.109(a)(2); and (3) assuming for the sake of argument that the information request is just that (and not a backfit), whether the the Staff has adequately justified the information request under Section 50.54(f).

II. DISCUSSION

A. The Information Request Under Consideration Is A De Facto Backfit

Section 50.109(a)(1) defines a backfit to include "the modification of [the] design of a facility; or the procedures . . . . required to design, construct or operate a facility; any of which may result from . . . . the imposition of a regulatory staff position interpreting the Commission rules that is either new or different from a previously applicable Staff position . . . ." This definition encompasses requests for information in which licensees are expected to resubmit analyses to justify previously accepted plant design. As explained during the Region II NRC Backfit Workshop with Industry, held on May 14, 1986 in Atlanta:

> MR. SPARKS: What I'm asking is, if you make or put out a staff position which requires us to spend a great deal of time, effort and money to redocument something you've previously approved in our plant design, is that a backfit or not?
>
> MR. SNIEZEK: If you have an operating license -- I'll use the terms you used -- that means we wrote off on your analysis. Whether we reviewed it or not, you submitted your [FSAR]. We wrote off on your analysis. If we later

> change our staff position because we don't think we want to accept that type of analysis anymore and ask you to re-do your analysis using other considerations, that would be a backfit, because we have changed the staff position from a previous staff position.

Workshop Transcript at 31-32.

> **MR. SPARKS:** [I]'d like to clarify my earlier question. I think you've answered part of it in your discussion on undue risk, and I want to make sure I have a good understanding of what you said. I guess I'd pose this question. If an NRC request for proof of the acceptability of an already approved plant design is requested and such request requires an extensive additional analysis -- in other words, it's going to take a lot of engineering man-hours and time and effort to re-approve something that's already been approved by the NRC, using a backfit -- Or I think I understand that it is a backfit unless an exception is taken because of a fear of undue risk to the public health and safety. Did I understand that correctly?
>
> **MR. SNIEZEK:** That's correct.

Workshop Transcript at 47-48.

Although the Staff disavows any intent to upgrade site design bases (See Transcript of the August 5, 1986 ACRS Meeting at 73), it is clear that the Staff does intend to apply criteria when evaluating licensee responses to the proposed information request that differ materially from those in effect when many older plants were granted operating licenses. It also appears that as part of the information request the Staff will require licensees to submit "justifications for continued operation" ("JCOs") if these new criteria are not met. Thus, the proposed information request asks for nothing less than the rejustification, in accordance with new criteria, of plant designs previously accepted by NRC.

Specifically, the proposed information request will apparently mandate that licensees "verify that the appropriate data base spectra envelope the site free field spectra at the ground surface defined for the plant." Appendix A to Regulatory Impact Analysis at 4 (Draft Generic Letter). The Staff suggests that where floor response spectra do not already exist at a given plant, such spectra are to be determined using a method developed

for the Staff by Brookhaven National Laboratory (BNL) and detailed in NUREG/CR-3266 (September, 1983). As discussed below, licensees will have little choice but to use the BNL method in order to identify "deficiencies" in equipment that would have to be replaced.

The BNL method adopted by the Staff is a new way to develop floor response spectra. It has never before been applied to many operating plants and it is not reflected in the approved design basis for those plants. Thus, the use of the generic spectra at these plants is part of a new requalification process, rather than a confirmation of whether exiting licensing bases are satisfied.

> The equipment in operating nuclear plants was designed at different times and to perhaps different standards. The task is to establish that this equipment is nevertheless conservatively designed and should therefore qualify when examined by a uniform procedure which follows present day acceptance practice.

NUREG/CR-3266 at 16 (emphasis added).

NUREG-1030 confirms that the generic spectra differ from the criteria against which the design of many current operating plants were evaluated and that the phrase "verification of seismic adequacy" used by the Staff in connection with A-46 is actually a euphemism for "seismic requalification".

> Floor response spectra, therefore, are essential elements for the qualification of equipment in nuclear power plants.
>
> * * *
>
> For many operating nuclear power plants, the information on floor response spectra may not have been developed according to the current requirements. In other cases, the information is simply no longer available. The objective of this task was to develop a set of generic floor response spectra which can be utilized for qualifying equipment.

NUREG-1030 at 2-86 (Staff background on NUREG/CR-3266)(emphasis added).

It appears from the Regulatory Impact Analysis that the Staff expects licensees to complete the following steps when responding to the proposed information request. Licensees first

will have to identify "deficiencies" by applying the A-46 criteria (i.e., NUREG-1030). As the Group understands the proposed information request, "deficient" equipment is that which is not enveloped by the new seismic experience data base, applied in conjunction with the new BNL methodology. In this sense the data base and BNL methodology will be used as screening criteria to assess initially the acceptability of present plant design. Licensees will not be able to determine deficiencies solely by reference to their licensing design basis.

Licensees will then be expected to provide JCOs for deficient equipment. The criteria for such JCOs is likely to include the new seismic experience data base, rather than the licensing basis of the facilities for which JCOs are provided. It is unclear how licensees will prepare JCOs regarding equipment for which the data base and the BNL methodology is not readily applicable. Nor is it clear how the Staff will evaluate such justifications. Based on Staff review of the JCOs, plant modifications could be required.

In sum, under the guise of an information request the Staff is about to require licensees to adopt elaborate inspection procedures and engineering reviews to demonstrate that the seismic design for their plants is adequate when measured against new criteria, and to provide JCOs where the design deviates from the new criteria. This constitutes a changed Staff position that will result in modified plant procedures and design. Therefore, the proposed information request amounts to a backfit within the meaning of Section 50.109(a)(1).

B. The Regulatory Impact Analysis Performed By the Staff Does Not Satisfy Section 50.109(a)(2).

In order to determine whether a proposed backfit provides a substantial increase in the overall protection of the public health and safety and whether the direct and indirect costs of its implementation are justified, the Staff is to perform a systematic and documented analysis. That analysis is to take into account all relevant and material information bearing on the proposed backfit and it is to address to the extent applicable nine specific factors set forth in the backfitting rule. 10 C.F.R. §§ 50.109 (a) and (c). The Commission explained the importance of preparing this analysis when it promulgated the backfitting rule:

> Because there must be safety reasons for the agency to impose any changes to a regulatory requirement or a staff position applicable to the licensee, because the safe[ty]

> consequences are unknown until analyzed, and because the Commission should fully understand the effects of a proposed backfit before its imposition, it is of little consequence how a backfit is imposed. Safety and sound management require that analysis precede imposition of a new or modified regulatory requirement or staff position.

50 Fed. Reg. 38101 (1985).

The Group believes that to the extent the Staff relies on its Regulatory Impact Analysis prepared in connection with USI A-46 to justify the proposed backfit, the Staff has failed to comply with Section 50.109(a)(2). In the Group's view the analysis is conclusory and does not adequately address several critical areas that should be considered before proceeding with the backfit under consideration. See generally the Group's July 22, 1986 letter.

Perhaps most significantly, the Staff has not adequately addressed the "potential change in the risk to the public from the accidental off-site release of radioactive material." There is no basis for concluding that the Staff believes that the proposed backfit is necessary to assure public health and safety. To the contrary, the Staff has never stated that the seismic design for existing plants is inadequate in comparison to current standards. On August 5, 1986 the following exchange took place during an ACRS meeting:

> DR. MICHELSON: We are saying now, I thought, the older requirement, must be saying the older requirements aren't good enough somehow.
>
> MR. ANDERSON: No.

ACRS Transcript at 74. The Staff also stated that "we of course are not intending to upgrade any of the site design bases" (Transcript at 73), and its Regulatory Impact Analysis states at 21 that "equipment installed in nuclear plants is inherently rugged and not susceptible to seismic damage."

In light of these statements, it is significant that the Staff has provided no underlying basis for its conclusion that the proposed backfit will have safety benefits. At page 35 of the Regulatory Impact Analysis the Staff simply states without elaboration that, "the safety benefit of the proposed seismic verification program is reduced likelihood of core melt and radiation release due to seismic failure of equipment. . . ." However, the Staff provides no basis for this statement. Apparently as explanation the Staff advises that:

> [t]he safety benefit of verifying the seismic adequacy of equipment in operating plants was not quantified in terms of risk reduction. . .. It proved impractical to quantify the result in a manner which would show the net safety benefit in terms of risk.

The Staff then asserted that "although the incremental risk has not been quantified . . . . the potential for safety improvement exists." Regulatory Impact Analysis at 36.

The Group does not mean to imply that the Staff is required under the backfitting rule to prepare a quantitative analysis of the reduction in risk to the public from the accidental off-site release of radioactive material. However, it does believe strongly that the Staff is required to make a good faith inquiry into the reduction in risk to be achieved by a proposed backfit. To date, the proposed backfit has been justified by no more than statements by the Staff that it will result in a potential reduction of risk. Clearly, the backfitting rule mandates a more rigorous inquiry than that which has been undertaken.

This is particularly the case where, as here, the backfit under consideration will be costly for both licensees preparing the analyses and related JCOs and the Staff reviewing these submittals. The 1985 (and now outdated) Regulatory Impact Analysis states that the estimated costs for each plant, exclusive of the costs associated with actual hardware modifications, range from \$191,000 to \$420,000 for each licensee that participates in a generic program (e.g., the program developed with the Seismic Qualification Utility Group). According to the Staff, licensees that do not participate in such programs can expect to pay an additional \$50,000 to \$100,000 to complete the initial design activities contemplated by the Staff in its proposed resolution of USI A-46. Finally, the Staff estimates that the costs to industry of developing the generic program range from \$28 to \$59 million. Regulatory Impact Analysis at 33-35 (1985).3/

In sum, the efforts contemplated by the Staff to resolve USI A-46 are resource intensive, not just in terms of dollars but also in terms of available engineering talent both inside and

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3/ The Group believes that current estimates of these costs are closer to \$100 million. In addition, they are exclusive of expenses accrued by utilities participating in the SQUG effort, that will each spend from \$200,000 to \$500,000 to develop the seismic experience data base for USI A-46 (Id. at 34).

outside of NRC. Before these efforts are undertaken, they should be subjected to careful scrutiny. The Commission had such scrutiny in mind when it promulgated the backfitting rule. The Group does not believe that the proposed Staff resolution of USI A-46 has been subjected to the rigorous scrutiny mandated by the backfitting rule.

C. Even as an Information Request, The Staff Has Not Justified its Proposed Action As Mandated By Section 50.54(f)

We believe that the foregoing convincingly shows that the Staff's request of licensees is a backfit under Section 50.109. However, even if for the sake of argument the Staff's request is considered to be only an information request, the Staff has not justified its proposed action. Section 50.54(f) prescribes that unless information is sought to verify compliance with a facility's current licensing basis, "the NRC must prepare the reason or reasons for each information request prior to issuance to ensure that the burden to be imposed on respondents is justified in view of the potential safety significance of the issue to be addressed in the requested information."

Much like the requirement to prepare a backfitting analysis, this obligation reflects the unequivocal Commission view that licensee and NRC resources be used effectively to address issues of greatest safety concern first. "The amendment of § 50.54(f) [to require the justification of information requests] should be read as indicating a strong concern on the part of the Commission that extensive information requests be carefully scrutinized by staff management prior to initiating such requests." 50 Fed. Reg. 38102 (1985).

Because of the broad scope and unusual nature of the information request proposed by the Staff, the Group does not believe that the Staff has justified the request as mandated by Section 50.54(f). The broad scope of this information request is clear:

- "Each licensee will be required to determine the systems, subsystems, components, and instrumentation and controls needed during and following a safe shutdown earthquake . . . . " Regulatory Impact Analysis at 4.

- "Each licensee must show practical means of staying at hot shutdown for a minimum of 72 hours [following a safe shutdown earthquake]." Id at 7.

- "Each licensee will be required to develop an equipment list that includes all items identified as necessary to perform functions related to plant hot shutdown." Id. at 8.

- "The licensee will verify [by comparison] that the appropriate data base bounding spectra envelope the site free field spectra at ground surface defined for the plant." Id. at 11.

- "Each licensee will be required to conduct a plant walk-through and visual inspection of all equipment items in the equipment scope." Id. at 12.

- "[F]unctional capability of the equipment or components [required to function during the strong ground motion part of an SSE] must be established." Id. at 14.

- "Each individual utility should submit to NRC an inspection report which should include: certification of completion of review, identification of deficiencies and outliers, justification for continued operation for identified deficiencies, modifications and replacements of equipment/anchorages (and supports) made as a result of the reviews, and the proposed schedule for required modifications and replacements not completed at the time of the report submitted." Id. at 18. 4/

In view of the resource intensive nature of the proposed information request, the Group submits that the request should be subjected to careful and searching management scrutiny. Doing so would certainly further the Commission management initiatives and in our view is mandated by Section 50.54(f). Clearly, the greater the burden of an information request on licensees (and the greater the amount of information to be considered by the

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4/ The Group does not believe that the proposed information request is intended to verify compliance with the current licensing bases of those facilities required to respond to the request because, as discussed above, the Staff does not intend to verify compliance with the current licensing bases of such facilities. Under Section 50.54(f), information requests to verify compliance with current bases are not subject to the same type of management scrutiny as are other information requests.

Staff), the greater should be the significance of the safety issue addressed. While the Group recognizes that the justification required by Section 50.54(f) should not be as detailed as a backfitting analysis under Section 50.109, the analysis prepared by the Staff to date is nevertheless not adequate to satisfy Section 50.54(f). As explained previously, this analysis does not provide any underlying justification to support the Staff's conclusion that the proposed resolution of USI A-46 would result in a reduction in risk.

III. CONCLUSION

The Group remains fully committed to addressing and resolving USI A-46 promptly. Further, the Group does not mean to imply that procedural requirements should be erected as a barrier to legitimate regulatory actions necessary to enhance safety. The top priority of licensees (and the NRC) must always be the protection of public health and safety.

The Staff has proposed a resource intensive program to address USI A-46. We share the desires of the Staff to see that this issue is promptly resolved. As shown above, because the Staff action is a backfit under Section 50.109, the Staff is compelled by law to perform the backfitting analysis called for in that regulation. Further, even if no backfitting rule existed, we believe that prudential management of both NRC and industry resources dictates that the Staff proposal be subjected to intensive scrutiny to assure that it addresses a significant safety concern in an effective manner and that it does not have any unintended and unanticipated consequences.

We appreciate this further opportunity to provide you with our views.

Respectfully submitted

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Counsel to Nuclear Utility
Group on Equipment
Qualification

ATTACHMENT 4 TO
ENCLOSURE 3

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

October 16, 1986

AEOD/E611

MEMORANDUM FOR: Richard H. Vollmer, Deputy Director
Office of Nuclear Reactor Regulation

Richard W. Starostecki, Acting Deputy Director
Office for Inspection and Enforcement

FROM: Frederick J. Hebdon, Deputy Director
Office for Analysis and Evaluation
of Operational Data

SUBJECT: DEFICIENCIES IN SEISMIC ANCHORAGE FOR ELECTRICAL AND CONTROL PANELS

Enclosed is an engineering evaluation report concerning several recent events involving inadequate anchorage of electrical and control panels. The report specifically addresses recent events at Cooper, Dresden 2/3 and Davis-Besse. However, we also identified other similar events uncovered by inspections by Sandia for the USI A-45 (Decay Heat Removal) program and in the SCSS database. Some of the events found by Sandia (e.g., at ANO-1 and Cooper) are equally as significant as the ones in the recent LERs.

In each of the recent events, although the plant design and installation specifications correctly specified the seismic anchorage, construction deficiencies occurred that resulted in inadequate anchorages. The Cooper deficiencies were related to both emergency diesel generator switchgear electrical cabinets; at Davis-Besse the Class 1E equipment electrical cabinets were missing anchor bolts; and at Dresden 2/3 none of the control panels in the control room were anchored.

The USI A-46 (seismic adequacy of operating reactors) is nearing resolution. As documented in the draft final reports for USI A-46 (NUREG-1030 and NUREG-1211), the staff is proposing a detailed reinspection program for verifying the seismic design margin of all equipment necessary to bring a plant to a safe, hot shutdown and to maintain it for 72 hours. The reinspection effort would be required by a generic letter and would be conducted after utility personnel receive training on reinspection procedures and evaluation techniques as developed by the Seismic Qualification Utility Group.

We have made the following suggestions for bringing the USI A-46 program to completion:

1. It is suggested that the Office of Inspection and Enforcement (IE) consider issuing an IE Information Notice to all licensees and possibly owners of NTOL plants concerning the recent evidence that inadequacies continue to

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exist in the seismic anchorage of equipment essential for accomplishing a safe, hot shutdown. Issuance of an Information Notice characterizing the recently identified problems in spite of previous, related reinspections and walk-throughs would provide details of the extent and variation of the continuing deficiencies and would emphasize this regulatory concern. This seems justified in light of the evidence of the inadequate response to IN 80-21. The Information Notice should discuss the recent experience, the potential safety implications, past and present regulatory attention to this concern (i.e., IN 80-21 and USI A-46) and reference the generic letter to be issued to resolve USI A-46.

2. The fundamental assumption underlying the proposed resolution of USI A-46 is that gross deficiencies do not exist in the seismic anchorage of operating nuclear plant safety equipment. The operational experience seems to indicate that this assumption may not be valid. Thus, we suggest that the proposed generic letter be modified to require early walk-through inspection of operating reactors to verify the absence of gross seismic anchorage defects before the detailed reinspections are conducted.

3. We suggest that, prior to issuance of the generic letter, the scope of the applicability of the letter, which would exclude reactors that have been designed to the IEEE-344/1975 standard, be reconsidered in light of recent experience that demonstrates construction and QA deficiencies are the root cause of inadequate anchorage rather than primarily design deficiencies.

4. We suggest that the generic, reinspection procedures be reviewed (e.g., reviewed by experts who are familiar with previous reinspections and walk-throughs) and/or documentation be developed by NRR to illustrate why the new program will overcome the limitations of the previous efforts to assure that limitations in walk-through inspections associated with IN 80-21, the Systematic Evaluation Program (SEP), the Seismic Qualification Utility Group (SQUG) and the Senior Seismic Review Advisory Panel (SSRAP) activities are remedied by the generic procedures.

5. We suggest that NRR consider whether the proposed USI A-46 generic letter requirement to identify equipment categories essential for achieving a safe, hot shutdown could be identified prior to availability of the final reinspection procedures rather than serially after the procedures and training of inspection personnel have been completed. Parallel development of equipment lists would somewhat expedite the reinspection efforts and would aid in accomplishing preliminary walk-downs for identification of gross deficiencies.

If you or your staff have any questions concerning the enclosed engineering evaluation report, please contact Neill Thomasson of my staff on extension x24431.

Frederick J. Hebdon

Frederick J. Hebdon, Deputy Director
Office for Analysis and Evaluation
of Operational Data

Enclosure:
As Stated

cc w/enclosure:
D. Humenansky, OCM
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E. Jordan, IE
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R. Gilbert, NRR
W. Long, NRR
A. DeAgazio, NRR
J. O'Brien, RES
D. Dubois, SRI (Cooper)
W. Rogers, SRI (Davis-Besse)
L. McGregor, SRI (Dresden)
W. Kane, RI
R. Walker, RII
C. Norelius, RIII
E. Johnson, RIV
D. Kirsch, RV

AEOD ENGINEERING EVALUATION REPORT*

UNITS: Dresden Units 2 and 3
Cooper Nuclear Station
Davis-Besse Unit 1

DOCKET NOS: 50-237/249
50-298
50-346

LICENSEES: Commonwealth Edison Company
Nebraska Public Power District
Toledo Edison Company

NSSS/AE: General Electric/Sargent & Lundy
General Electric/Burns and Roe
Babcock & Wilcox/Bechtel

EE REPORT NO.: AEOD/E611
DATE: October 16, 1986
EVALUATOR/CONTACT: N. Thomasson

SUBJECT: SEISMIC ANCHORS FOR ELECTRICAL AND CONTROL PANELS

SUMMARY

Three recent Licensee Event Reports (LERs) indicate that problems continue to exist in the adequacy of seismic anchorage of critical safety equipment in operating nuclear power plants. This has been a generic regulatory concern since 1980, when Information Notice (IN) 80-21 was published, and the issue was designated Unresolved Safety Issue (USI) A-46, "Seismic Qualification of Equipment in Operating Nuclear Power Plants." The 1980 Information Notice specifically suggested that licensees should perform walk-through inspections of the anchorage systems of certain specified categories of equipment, including electrical cabinets and control panels. In spite of IN 80-21 and independent walk-through inspections related to seismic anchorage of equipment at certain plants by different groups, the lack of seismic anchorage for electrical and control panels continues to be identified, including in some cases gross inadequacies. The near-term resolution of USI A-46 is aimed in part at remedying these residual defects. A generic letter is to be issued requiring walk-through inspections of operating nuclear plants using detailed generic checklists and procedures to verify the adequacy of seismic anchorage for all equipment necessary to accomplish and maintain safe, hot shutdown. Consequently, that solution appears to be adequate to assure that the issue is thoroughly addressed. However, in light of the evidence of continuing deficiencies, we suggest (1) that IE issue an updated IN characterizing recently identified deficiencies in order to emphasize the need to respond expeditiously to the generic letter to complete the reinspections, and to correct the deficiencies thus identified; and (2) that NRR modify the generic letter to require near-term walkdown of critical equipment to verify the absence of gross deficiencies before the planned detailed inspections.

INTRODUCTION

The Systematic Evaluation Program (SEP) addressed the seismic adequacy of 12 older plants vis-a-vis current licensing requirements. Based on the subsequent findings of possible significant deficiencies in seismic anchorages and support of safety-related electrical equipment at several of these plants,

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* This document supports ongoing AEOD and NRC activities and does not represent the position or requirements of the responsible NRC program office.

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an IE Information Notice (IN 80-21) was issued in May 1980 with specific focus on performing walk-down inspections of equipment anchorages in nuclear plants designed prior to 1971. Then, in 1980 NRR established the concern of seismic qualification of equipment in operating nuclear plants as an Unresolved Safety Issue (USI A-46).

Between late March 1986 and May 1986, three Licensee Event Reports (LERs) were received that documented inadequate seismic anchorage of electrical equipment in operating nuclear power plants. The initial case was at Davis-Besse 1 (LER 86-011) where cabinet doors on Cyberex class 1E equipment for essential instrument 120 VAC power were found to lack the required door bolts. The second LER concerned emergency diesel generator switchgear cabinets at Copper that were not fastened to embedded channels beneath the cabinets (LER 86-009). The third and most extensive deficiency was found at Dresden 2, where it was determined that the control room control panels did not have positive anchorage to the floor (LER 86-009). In each instance, the deficiency had existed since plant construction and was the result of installation errors, since the design drawings had specified seismic anchorage. As a consequence of these events, a review was initiated by the Office for Analysis and Evaluation of Operational Data (AEOD) to determine the extent of other similar deficiencies, regulatory requirements, history of addressing the question of seismic anchorage for electrical equipment, and the safety considerations of the deficiencies.

DISCUSSION

Regulatory Requirements

The fundamental regulatory requirements for seismic design of nuclear power plants are given in Appendix A to 10 CFR Part 100, "Seismic and Geologic Siting Criteria for Nuclear Power Plants" and General Design Criterion 2 of Appendix A to 10 CFR Part 50, "Design Basis for Protection Against Natural Phenomena." The regulations require nuclear power plants to be designed to assure functionality of certain structures, systems and components in the event of a design basis earthquake [a safe shutdown earthquake (SSE)]. One reason for such a requirement is to ensure the capability to shut down the reactor and maintain it in a safe shutdown condition during and following an SSE.

Specific guidance related to electrical cabinets, relays and motor control centers is also given in the Standard Review Plan (SRP) (NUREG-0800) Section 3.10, Regulatory Guide (RG) 1.100, and IEEE Standards 344/1975, 501/1978, and 649/1980. These requirements and guidance have evolved over the history of commercial nuclear power development. Thus, criteria and methods for the seismic qualification of mechanical and electrical equipment have experienced considerable change. Consequently, for older operating reactors** the capability of the plant and equipment to resist induced seismic loads and to perform essential safety functions following an SSE may vary significantly.

** Those not reviewed against current design and equipment seismic qualification requirements, in particular IEEE 344/1975.

- The proposed resolution of USI A-46, scheduled for the Fall of 1986, is to issue a generic letter (GL) requiring reinspection using specified generic checklists and inspection procedures*** for the reinspection of seismic anchorage and mounts for all equipment necessary to bring the older power reactors (i.e., those not designed in accordance with IEEE 344/1975) to a safe, hot shutdown and to maintain it for 72 hours. If such reinspection demonstrates the absence of significant deficiencies in the seismic anchorage of the critical equipment, the plant will be acknowledged to have adequate seismic design margins.

Regulatory Reviews

All nuclear power plants have been designed and constructed to the seismic design requirements operative at the time of their construction permits. Similarly, the plants have implemented quality assurance (QA) inspections complemented by inspections by NRC staff during construction. Although all plants were reviewed during their construction permit and operating licensing proceedings against prevailing seismic and QA criteria and requirements, regulatory guidance and requirements in these areas have experienced significant upgrading since the early 1970's.

In 1978, the NRC established the Systematic Evaluation Program (SEP) to review the designs of older operating plants to reconfirm and document their margins of safety. The SEP review assessed the significance of the differences between current technical safety requirements and guidance and those operative when a given reactor was licensed by comparing the as-built design to the Standard Review Plan (SRP), NUREG-0800, in 137 review areas. Seismic qualification of plant safety equipment was one of the review areas. As a result of the SEP reviews, IN 80-21 was issued to alert the nuclear industry of NRC's concern regarding the lack of engineering supports of safety-related electrical equipment in older plants, including motor control centers, switchgear, control room control panels and instrument panels.

In support of developing a resolution for USI A-46, an industry group, designated as the Seismic Qualification Utility Group (SQUG), was formed to provide input on seismic experience in the nuclear and non-nuclear industry. Another group the Senior Seismic Review Advisory Panel (SSRAP) was created by SQUG to provide an independent review of the SQUG contractor work. A key element of the SQUG effort was to identify (1) seismic design practices in the nuclear and non-nuclear industries and (2) the extent of comparability of critical equipment in both types of facilities so that the extensive database of non-nuclear industry equipment experience in response to severe seismic motion might be translated to expected nuclear power plant response. In conducting these activities, SQUG contractors and SSRAP performed walk-through inspections of Dresden 2, surveying eight classes of critical equipment and the seismic anchorages of the equipment.

The NRC research program to develop equipment fragility information also included some plant inspections of critical safety equipment seismic anchorage, as has the NRR program to address decay heat removal (USI A-45).

Recent Operational Events

Davis-Besse 1 On February 28, 1986, Toledo Edison determined that, since plant licensing (1977), the cabinet doors on Cyberex class 1E equipment (battery chargers, inverters, rectifiers, and distribution panels) for

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*** NUREG-1030 and its Regulatory Analysis--NUREG-1121.

essential instrument AC power were not bolted closed as required for seismic qualification. The deficiency was attributed to failure to develop adequate procedures for equipment installation. Apparently, installers during original construction did not recognize the seismic function of the bolts, given that the cabinet doors were hinged and latched; thus, during equipment installation the bolts were apparently removed on the belief that the bolts were for shipping purposes. The deficiency was recognized when Cyberex Corporation confirmed that the missing bolts were added during seismic qualification testing.

Each of the four 120 VAC distribution panels with the missing bolts supply one channel of the plant's (1) engineered safety features actuation system; (2) the Steam and Feedwater Rupture Control System; (3) the Reactor Protection System; and (4) the Anticipatory Reactor Trip System. According to Toledo Edison, loss of power to these systems results in a fail-safe condition since each system actuates on loss of two distribution panels, causing a reactor trip. However, as noted by the licensee, control and stabilization of the plant following such a trip become increasingly complicated as additional control panels are lost. The licensee's corrective action was to install the required bolts on the cabinet doors and to revise plant maintenance procedures.

<u>Cooper</u> On March 18, 1986, the Emergency Diesel Generator (EDG) 2 switchgear generator output breaker cabinet at Cooper Nuclear Station was determined to lack anchorage to embedded channels beneath the cabinets. Although this reactor had been operating since 1974, the construction defect went unrecognized until plant staff found the deficiency while preparing for future modifications. In response to the deficiency, utility personnel inspected other floor-mounted switchgear and, on a sampling basis, inspected the anchorage of other safety-related equipment. A similar deficiency was found for EDG-1, but no other like problems with other equipment were found. The corrective action was to weld the cabinets to the embedded channels.

<u>Dresden 2 and 3</u> On April 1, 1986, Commonwealth Edison determined that the Dresden Unit 2 control room control panels had not been positively anchored to the floor as required by the original design and installation instructions. The deficiency was found during seismic evaluation of the control panels for addition of future instrumentation. The affected General Electric control panels are supported on structural steel channel sections, which were anchored to the concrete floor. The control panels were supposed to have been bolted to the steel channels via mounting holes in the front and rear of the control panels. However, the bolts had never been installed. A similar condition was subsequently identified for the Unit 3 control panels.

The corrective actions at Dresden 2 and 3 were to install ½" x ½" SAE Grade 5 (ASTM A449) anchor bolts through the control panels to the structural steel channels or, in cases where interferences existed, to weld the control panel to the steel channel. Also, in order to prevent overturning, braces were added to the top of the control panel for the Standby Gas Treatment System.

<u>Other Events</u> The Sequence Coding and Search System (SCSS) LER database contained four other instances since 1981 of inadequate seismic mounts of safety-related electrical equipment and other related seismic inadequacies regarding seismic restraints other than anchorages. Each of these LERs is summarized in Table 1.

Table 1

LERs Involving Inadequate Seismic Anchorage for Electrical and Control Cabinets

DATE	PLANT	LER	EVENT	ROOT CAUSE	PLANT STATUS	COMMENT
3/18/86	Cooper	86-009	EDG-2 Switchgear generator output breaker cabinet not fastened to embedded channels beneath the cabinet. Same deficiency was found upon inspection of EDG-1.	Apparent installation deficiency during original construction.	Reactor at 85% Power.	Found in preparing for future modification; other floor mounted switchgear inspected as well as sampling of other floor mounted safety-related equipment. Bases were welded to channels embedded in the floor.
4/1/86	Dresden 2/3	86-009	Control panels in control room lacked positive anchorage to the floor--bolts had never been installed as specified in original installation instructions during plant construction. Panels rested on structural steel channel section which was anchored to concrete floor.	Anchor bolts never installed.	Unit 2 - 99% Power Unit 3 - Shutdown fuel removed.	Found during seismic evaluation of control room panels for mounting future instrumentation bolts; bolts installed and tack welds were added along with top braces.
2/28/86	Davis-Besse	86-011	Cyberex class 1E equipment cabinet doors for essential instr. AC power not bolted closed as required by seismic design (battery chargers, inverters, recifiers, distribution panels).	Bolting detail missing during development of procedures.	Plant shutdown	A condition outside the design basis existing since plant startup. Bolts installed during seismic qual. test were believed to be shipping bolts.

Table 1 (continued)

DATE	PLANT	LER	EVENT	ROOT CAUSE	PLANT STATUS	COMMENT
12/13/82	Davis-Besse	82-066	Auxiliary shutdown panel terminal board not seiscally mounted.	Design and fabrication error.	Shutdown.	Mechanics trouble-shooting a problem found seismic mount missing--mounting screws had been removed.
11/19/85	Browns Ferry 1, 2, 3	85-016 85-020	Eight local reactor protection panels on each unit not constructed per design drawings.	Construction Configuration never documented.	Units 1-2 Refueling; 3 - shutdown	Analysis showed Unit 3 as-built was seismically qualified. Unit 2 had 2/8 panels not qualified since anchor bolt material could not be determined. Status of Unit 1 to be determined.
4/18/85	Browns Ferry	85-013	Configuration of mounts for startup test instrumentation panels relative to seismic qualifications is indeterminate. Panels are plywood with sheet metal cabinet, fastened to the masonary wall with the masonary anchors. Cables lay unsecured on main control room panels and could adversely affect safety-related panels/functions.	Change in intended use of panels from temporary use during startup testing to operational use.	Unspecified	Panels used for startup testing following refueling and for special testing. Installation field improvised for it was originally intended to be removed after initial startup.

Table 1 (continued)

DATE	PLANT	LER	EVENT	ROOT CAUSE	PLANT STATUS	COMMENT
8/14/81	D.C. Cook	81-033	Switchgear cabinets, reactor trip and bypass breaker cabinets lacked sufficient seismic restraints to prevent overturning during SSE.	Design deficiency in cabinet anchors.	Not Specified	Response to GL 81-14 found problem (Aux. Feedwater System Seismic Qual.)
8/15/83	D.C. Cook-2	83-073	Safety-related battery chargers not adequately seismically anchored - anchor bolt nuts not installed as required by design. (LER 82-047 pevious similar occurrence.)	Nuts not installed during construction.	Not Specified	

In each case the deficiency was related to construction installation errors. The original designs correctly specified the seismic anchorages or other seismic support. The deficiencies had gone undetected since construction inspite of QA inspections, IN 80-21, and the high visibility for such concerns as indicated by its designation as USI A-46.

Other Evidence of Deficiencies In conjunction with staff efforts to resolve USI A-45 (Decay Heat Removal), NRC contractor staff from Sandia National Laboratory have conducted six walk-through inspections regarding seismic fragility of equipment essential for accomplishing decay heat removal. During these inspections, additional evidence was found of inadequate and missing seismic anchorage. For example, two plants lacked anchorage for 4Kv switchgear busses; two plants lacked anchorage on bus switchgear and motor control centers; in three plants the battery racks were inadequately supported or lacked anchors; and one plant had a deficiency like Dresden 2/3 in that its control room control panels had never been anchored to the floor. Cooper, one of the plants inspected by Sandia in 1984, reported a similar deficiency to the one described in the 1986 LER; the utility had performed inspections for such problems after the Sandia audit in 1984 and concluded that no other deficiences existed. Further, the Sandia audits illustrate a need for a careful well-thought out inspection procedure. For example, in one plant, Sandia determined that even though the cabinets appeared to be anchored at their base, the bolts did not penetrate through the steel channels to the floor.

Safety Significance of the Operational Events

Each of the three recent LERs involving inadequate seismic mounts for critical electrical or control cabinets resulted in plant conditions that were associated with unanalyzed safety concerns. Each instance involved significant departures from the plant design basis and assumed conditions for the accident analyses in the plants' Final Safety Analysis Reports (FSARs). In those analyses, only a single failure is assumed as well as the availability of redundant or backup systems required to perform the essential safety functions.

In these events, redundant safety systems could be assumed to have failed in the event of an SSE since common-mode deficiencies were found (Cooper, switchgear for both EDGs; Davis-Besse, electrical cabinet doors for four redundant 120 VAC instrument channels; and Dresden's control room panels). Further, since multiple reactor safety systems would have been affected, the resulting incidents could have been more severe than those analyzed in the FSARs. However, final conclusions regarding the risk significance of these or any other cases of inadequate seismic anchorage of critical safety equipment must be determined by site/plant-specific design and as-built conditions. For example, lack of seismic mounts does not necessarily constitute deficiencies leading to failure of function because much of the equipment is laterally supported by other cabinets and the connecting cables will tend to prevent overturning. Nevertheless, because of the potential for common-mode failures of redundant systems and for simultaneous

failures of multiple safety systems, there is a strong basis to expedite resolution of this issue. As an indication of the urgency of this problem, in each case where plants were at power when the deficiencies were found (Cooper and Dresden 2), the licensee demonstrated recognition of a potentially serious safety concern by taking immediate actions to bring the plant to a safe shutdown condition. (Davis-Besse and Dresden-3 were already shutdown.)

Analysis of Regulatory Activities

The adequacy of the seismic design, in particular the anchorage of critical safety-related equipment at older operating plants, has been the focus of numerous regulatory programs and reinspections since 1980. Some plants (e.g., Dresden 2 and 3) have been the focus of multiple inspections and walk-through inspections by independent groups focusing on the adequacy of seismic anchorage. For example, during construction, utility QA/QC and NRC regional staff performed audits and inspections; then the NRC and utility conducted audits with regard to the SEP program, which were followed by more detailed audits in response to IE IN 80-21. Then the SQUG conducted a walk-through to identify the types of electrical and control equipment used in nuclear plants and to ascertain the characteristics of the anchorage used for eight classes of the equipment. Finally, the SSRAP conducted a walk-through to verify the SQUG findings. Nevertheless, in April 1986, Commonwealth Edison identified the fact that the control room control panels had never been anchored to the floor, even though their response to IN 80-21 indicated that there was no deficiency in these anchors.

At least one other plant (Cooper) has been the subject of an earlier review (USI A-45 study) regarding as-built seismic fragility of electrical equipment. Although this walk-through and evaluation occurred in 1984 when the utility conducted some related reinspections after Sandia found a switchgear mounting deficiency, Cooper found additional deficiencies associated with its emergency diesel generator switchgear cabinets in 1986.

In no case were we able to develop information regarding the specific procedures for or scope of the previous inspections and walk-throughs. Nevertheless, it is evident that significant deficiencies were missed. Therefore, in order to develop greater confidence in the proposed reinspection procedures and checklists for use in the generic resolution of USI A-46, we suggest that before they are implemented, the specifics of the procedures be reviewed by experts who are familiar with the previous reinspections and walk-throughs, and/or that documentation be developed to illustrate why the new program will overcome the limitations of the previous efforts.

The schedule projected by NRR for resolution of USI A-46 and its full implementation is to issue a generic letter in October 1986 followed by completion of generic reinspection procedures and checklists by 1987, trial applications by early 1987, and training seminars for utility personnel in the first quarter of 1987. Then, the utilities would develop lists of safety equipment essential to achieve a safe, hot shutdown following an SSE and conduct plant-specific walk-through reinspections. Final implementation of USI A-46 fixes would be scheduled as part of each plant's living schedule, with a target for completion within 28 months of the generic letter. It is important to

note that, as indicated in NUREGs-1030 and 1211, the fundamental assumption underlying the proposed generic resolution of USI A-46 is that no gross deficiencies exist in the seismic anchorage of critical equipment. Further, the SSRAP recommended that prior to detailed reinspections, walk-throughs be conducted to verify the absence of gross mounting deficiencies (see NUREG-1030, p. 2-30). We agree with the SSRAP recommendation. Thus, we suggest that the proposed generic letter be modified to require early walk-through inspection of operating reactors to verify the absence of gross seismic anchorage defects before the detailed reinspections are conducted.

The generic letter will provide the bases for regulatory action and utility reinspections. However, it only refers to continuing deficiencies and does not provide the details of specific, recently identified deficiencies in seismic anchorage of critical safety equipment that would clearly define the scope and clear presence of current problems, or the limitations of walk-throughs and reinspections. Thus, it is suggested that an updated Information Notice be issued as a supplement to the generic letter to illustrate the continuing inadequacies in the seismic anchorage of critical safety equipment.

The adequacy of the reinspection procedures is critical to a successful resolution of this generic issue. Thus, reinspection without sound guidance and QA controls would be a waste of time and resources. However, the implementation phase of detailed reinspection could be expedited by identification of the equipment essential for achieving a safe, hot shutdown while the reinspection procedures are completed. Thus, once procedures are available and staff trained, reinspections can proceed expeditiously. The results of the reinspections could provide a basis for decisions regarding the need for further interim safety measures and scheduling of long-term corrective actions.

USI A-46 was conceived and its proposed resolution is directed, based on an explicit assumption that modern seismic design and equipment qualification requirements (namely, IEEE 344/1975), underpinned by prelicensing review by the staff's Seismic Qualification Review Team (SQRT), provide reasonable assurance of seismic safety for those plants. Thus, the generic letter is proposed to include only the 70 operating units that were not designed in accordance with IEEE 344/1975. Whether or not this assumption is valid depends on the overall thoroughness of the utility's QA efforts, NRC prelicensing inspection and the scope of the SQRT reviews. In light of the fact that the deficiencies discussed herein relate to construction, rather than design defects, we suggest that the proposed limitation of the reinspection to 70 operating reactors be reconsidered.

FINDINGS

As a consequence of this engineering evaluation the following findings may be made:

- Deficiencies, in some cases gross defects, in seismic anchorage due to absence of anchors exist in operating nuclear power plants despite a 1980 Information Notice on the same subject.

- Deficiencies in seismic anchorage of electrical cabinets and related control panels exist in redundant safety systems and could result in disabling of redundant and multiple reactor safety systems.

° Anchorage deficiencies continue to be identified by multiple, independent programs (LER, USI A-46, USI A-45 and fragility research program) and during reinspections by licensee and other parties.

° The safety significance of the events involving inadequate seismic anchorage is dependent on site/plant-specific design and as-built conditions. However, when redundant equipment and/or multiple safety systems would be susceptible to loss of function due to a single seismic event, the potential for exceeding the design basis of the plant is increased.

° The safety significance of the adequacy of the seismic anchorage is clearly indicated by the conclusions of the SQUG, SSRAP and staff that given adequate seismic anchorage, the seismic margins of the older operating nuclear power plants is deemed adequate. Conversely, without adequate anchorage, there could be serious concern with the seismic margins for the plants.

° The identified deficiencies in seismic anchorages are construction related rather than design related.

CONCLUSIONS

1. There is continuing evidence of continued inadequacy, in some cases gross deficiencies, of seismic anchorage for critical safety equipment due to missing anchorage.

2. The resolution of USI A-46 generic issue is expected to result in a reinspection requirement and procedures that should lead to closure of this concern at older operating reactors.

SUGGESTIONS

1. It is suggested that the Office of Inspection and Enforcement (IE) consider issuing an IE Information Notice to all licensees and possibly owners of NTOL plants concerning the recent evidence that inadequacies continue to exist in the seismic anchorage of equipment essential for accomplishing a safe, hot shutdown. Issuance of an Information Notice characterizing the recently identified problems in spite of previous, related reinspections and walk-throughs would provide details of the extent and variation of the continuing deficiencies and would emphasize this regulatory concern. This seems justified in light of the evidence of the inadequate response to IN 80-21. The Information Notice should discuss the recent experience, the potential safety implications, past and present regulatory attention to this concern (i.e., IN 80-21 and USI A-46) and reference the generic letter to be issued to resolve USI A-46.

2. The fundamental assumption underlying the proposed resolution of USI A-46 is that gross deficiencies do not exist in the seismic anchorage of operating nuclear plant safety equipment. The operational experience seems to indicate that this assumption may not be valid. Thus, we suggest that the proposed generic letter be modified to require early walk-through inspection of operating reactors to verify the absence of gross seismic anchorage defects before the detailed reinspections are conducted.

3. We suggest that, prior to issuance of the generic letter, the scope of the applicability of the letter, which would exclude reactors that have been designed to the IEEE-344/1975 standard, be reconsidered in light of recent experience that demonstrates construction and QA deficiencies are the root cause of inadequate anchorage rather than primarily design deficiencies.

4. We suggest that the generic, reinspection procedures be reviewed (e.g., reviewed by experts who are familiar with previous reinspections and walk-throughs) and/or documentation be developed by NRR to illustrate why the new program will overcome the limitations of the previous efforts to assure that limitations in walk-through inspections associated with IN 80-21, SEP, SQUG and SSRAP activities are remedied by the generic procedures.

5. We suggest that NRR consider whether the proposed USI A-46 generic letter requirements to identify equipment categories essential for achieving a safe, hot shutdown could be identified prior to availability of the final reinspection procedures rather than serially after the procedures and training of inspection personnel have been completed. Parallel development of equipment lists would somewhat expedite the reinspection efforts and would aid in accomplishing preliminary walk-downs for identification of gross deficiencies.

REFERENCES

1. Licensee Event Report (LER) 86-011, Docket Number 50-346, Davis-Besse 1, March 27, 1986.

2. Licensee Event Report (LER) 86-009, Docket Number 50-298, Cooper Nuclear Station, March 18, 1986.

3. Licensee Event Report (LER) 86-009, Docket Number 50-237/249, April 29, 1986.

4. SECY-85-277, "Unresolved Safety Issue A-46, 'Seismic Qualification of Equipment in Operating Nuclear Power Plants,'" August 20, 1985.

5. Final Draft NUREG-1030, "Seismic Qualification of Equipment in Operating Nuclear Power Plants Unresolved Safety Issue A-46."

6. Final Draft NUREG-1211, "Regulatory Analysis for Proposed Resolution of Unresolved-Safety Issue A-46 Seismic Qualification of Equipment of Operating Plants."

7. IE Information Notice Number 80-21, May 16, 1980.

8. Letter from R. F. Janecek, Commonwealth Edison Company, to D. M. Crutchfield, NRR, USNRC Subject: "Dresden Station Unit 2 SEP-Positive Anchorage of Safety-Related Electrical Equipment NRC Docket No. 50-237."

APPROACH

- o SELECTION OF A REFERENCE OR REVIEW EARTHQUAKE
- o FIRST PLANT WALKDOWN
- o PLANT SYSTEM AND ACCIDENT EVALUATION
- o COMPONENT FRAGILITY OR CAPACITY EVALUATION (HCLPF)
- o SECOND PLANT WALKDOWN
- o DETERMINE LOWER BOUND PLANT SEISMIC CAPACITY (HIGH CONFIDENCE OF LOW PROBABILITY OF FAILURE-HCLPF)

FRAGILITY DATA BASE

- CATEGORY I STRUCTURES PROGRAM (LANL)
  - EXPERIMENTS ON SMALL SCALE SHEAR WALL STRUCTURES
  - RESULTS TO DATE SHOW CAPACITIES GREATER THAN 2g
  - FREQUENCY SHIFT PROBLEM UNRESOLVED
- COMPONENT FRAGILITIES PROGRAM (BNL AND LLNL)
  - TEST DATA FROM LABS AND VENDORS
  - EARTHQUAKE EXPERIENCE DATA
  - COOPERATION WITH SQUG AND EPRI