ATTACHMENT A

NUCLEAR UTILITY GROUP
ON EQUIPMENT QUALIFICATION

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October 26, 1983

Nunzio J. Palladino
Chairman
U.S. Nuclear Regulatory
Commission
Washington, D.C. 20555

Dear Mr. Chairman:

The Nuclear Utility Group on Equipment Qualification ("Group")[1] has been closely following the potential impact on equipment qualification activities of the NRC Staff's rule-making initiative regarding hydrogen generation (SECY-81-245A). It is our understanding that the Staff recently forwarded to the Commission its recommended final rule regarding this issue, which includes a fundamental shift of position from the proposed rule. The Staff would now require a detailed analysis employing very conservative equipment qualification standards (rather than the equipment survivability standards envisioned by the proposed rule) to demonstrate the operability of essential equipment following a postulated accident involving significant hydrogen generation. As discussed more fully below, we maintain that this shift in position is unsupported by the record, technically unjustified and will inappropriately establish as Commission policy that the full range of equipment qualification requirements is required for events well beyond the design basis.

The proposed hydrogen generation rule, published for public comment on December 23, 1981, would have required that licensees provide an analysis to show that equipment necessary to achieve and maintain safe shutdown could survive a postulated hydrogen burn associated with the release of hydrogen from a 75% clad-water reaction. 46 Fed. Reg. 62281.

1 The Group consists of 23 utilities each of which is a holder of at least one construction permit or operating license for a nuclear power reactor. Since its formation in 1980, the Group has been actively involved in all aspects of equipment qualification.

In responding to Commission questions on this issue, the Staff stated that a survivability standard was proposed instead of the very conservative qualification standard because any accident which would result in significant hydrogen generation would be well beyond the design basis and its probability of occurrence would be extremely remote. (Commission Briefing of September 16, 1981, Subject: Discussion of Rule on Hydrogen Control at pp. 37-44.) In any event, the Staff had noted that all indications reflected that necessary equipment would be able to withstand such an event. See e.g., Attachment to SECY-80-107 at pp. 16-17 (February 22, 1980). See also Commission Briefing of November 19, 1982, Subject: Briefing on Hydrogen Control Program at pp. 48-9 wherein Roger Mattson stated as follows:

> First, it has been fairly easy to show [that, e.g., important to safety equipment would survive] for the small containments, the ice condensers and Mark III. . . . For example, the local temperatures from [a] hydrogen burn turn out to be less than the environmental qualification temperatures for the steamline break used in the normal EQ process. Once you can show that, of course, you have really reduced the regulatory burden imposed upon licensees for the survivability question.

In this regard, in proposing the survivability standard, the Staff referenced the operating license proceeding for Duke Power Company's William B. McGuire Nuclear Generating Facility (an ice condenser plant), the only proceeding at that time where the issue of essential equipment survivability after a hydrogen burn had been litigated and resolved based upon evidence presented in an adjudication. In the McGuire case, the Applicant presented a panel of three eminently qualified experts in the area of hydrogen and hydrogen combustion phenomena who had each been actively involved with investigations of hydrogen combustion specifically at nuclear power facilities. McGuire Operating License Hearing Transcript at Tr. 3162-66, Docket Nos. 50-369-OL and 50-370-OL, ("McGuire Transcript"). With regard to the effect of the combustion of hydrogen on essential equipment, such as typically found in a nuclear power plant, the panel of experts testified as follows:

> Based upon the range of temperatures this equipment can survive and based upon a consideration of the temperature of the flame front, its speed and duration, the heat transfer from the hot burned gases, the time for the hot burned products to return to ambient temperatures, and the total heat energy available, it is concluded that the essential equipment needed for the safe shutdown of the plant will not be so affected by the combustion of the hydrogen so as to impair its ability to safely perform its intended function. [Testimony of Lewis Panel incorporated in the McGuire Transcript following Tr. 3346-7 at 12.]

Significantly, with regard to the issue of equipment survivability in the McGuire case, the NRC Staff conducted an independent analysis of all relevant equipment, thoroughly reviewed Duke's analysis, and agreed with the conclusions stated above. Testimony of Staff Regarding Reactor and Containment Systems Performing Assessment incorporated into McGuire Transcript following Tr. 4535 at 9. See Tr. 4538, 4621-3.

In short, at the time of issuance of the proposed rule for comment, the Staff position regarding use of a survivability instead of qualification standard was appropriately based on the low probability of a significant degraded core accident and on the substantial body of data which indicated that equipment of concern would withstand the effects of a significant hydrogen burn environment (even in a small ice condenser containment).

Subsequent to issuance of the proposed rule nothing has occurred to alter this view. On the contrary, we understand that the preliminary results of additional research support the conclusion that equipment can continue to function during a hydrogen burn environment. Further, the comments of almost one half of those responding to this proposed rule point out that the qualification standard is inappropriate for such a low probability event. We are not aware of any commentators who presented a contrary view. Yet, in the face of all generic research and analysis to date which, if anything, supports the withdrawal of the proposed requirement, the Staff seeks to increase significantly the extent and complexity of the documentation testing and/or analysis required of each licensee. That Staff now would require that all equipment of concern meet the extremely conservative qualification criteria heretofore reserved for analysis of only design basis events. We maintain that such an expansion is technically unwarranted and not supported by the record.

In recommending this final rule, the Staff also states that it is simply "formalizing regulatory positions" already taken in "individual licensing cases (i.e., Sequoyah, McGuire, D.C. Cook and Grand Gulf)." ACRS Subcommittee Meeting on Class 9 Accidents (June 26, 1983) Attachment at Enclosure G at p. 4. Further, the Staff states that the cost estimates used as a basis for the Staff's recommendation regarding demonstration of qualification were based on actions taken in these licensing cases. Id. at p. 11. Significantly, however, the licensing positions and costs associated with these cases relate to demonstrations of survivability and not the more conservative demonstrations of qualification as proposed by the Staff in this final rule. Accordingly, unless the demonstrations of survivability made in these cases are equated with the demonstrations of qualification as proposed by the Staff in its final rule, the basis for the Staff's position is technically flawed.[2]

In sum, we maintain that the Staff's proposed shift from a survivability to qualification standard for demonstration of operability is unsupported by the record, technically unjustified, and, from a policy perspective, an inappropriate expansion of equipment qualification requirements beyond design basis events. Accordingly, we strongly encourage the Commission to reject the Staff's proposed requirement regarding this issue. We believe that all generic research and analysis to date supports our view, and that the diversion of significant industry resources which this shift would occasion is not warranted. Rather, if any new requirement is needed regarding demonstration of equipment operability for a degraded core accident resulting in significant hydrogen generation, we recommend that it be based on a survivability standard and clearly provides for full use and reliance on generic testing and analysis.

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2 If the Staff is using the terms survivability and qualification interchangeably, then our concerns are somewhat lessened. Thus, if the demonstrations of survivability as provided in these licensing cases are, without more (e.g., testing, analysis, documentation), equivalent to demonstrations of qualification as proposed here, this should be clearly stated.

We appreciate the opportunity to provide you with our comments on this crucial issue.

Sincerely,

Nicholas S. Reynolds
Counsel to Nuclear Utility Group
on Equipment Qualification

cc: Commissioner Asselstine
Commissioner Bernthal
Commissioner Gilinsky
Commissioner Roberts

bcc: Richard H. Vollmer
William J. Dircks
Victor Stello, Jr.

ATTACHMENT B

IN THE UNITED STATES COURT OF APPEALS FOR THE DISTRICT OF COLUMBIA CIRCUIT

No. 82-2000

UNION OF CONCERNED SCIENTISTS

Petitioner

v.

NUCLEAR REGULATORY COMMISSION, ET AL.

Respondent

NUCLEAR UTILITY GROUP
ON EQUIPMENT QUALIFICATION

Intervenor

ON PETITION FOR REVIEW OF A RULE OF THE NUCLEAR REGULATORY COMMISSION

SUPPLEMENTAL BRIEF FOR INTERVENOR NUCLEAR UTILITY GROUP ON EQUIPMENT QUALIFICATION PURSUANT TO THE COURT'S ORDER DATED OCTOBER 19, 1983

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October 31, 1983

For Intervenor Nuclear
Utility Group on
Equipment Qualification

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TABLE OF AUTHORITIES

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* Cases and authorities principally relied upon are marked by an asterisk.

IN THE UNITED STATES COURT OF APPEALS
FOR THE DISTRICT OF COLUMBIA CIRCUIT

No. 82-2000

Union of Concerned Scientists

<u>Petitioner</u>

v.

Nuclear Regulatory Commission, <u>et</u> <u>al</u>.

<u>Respondent</u>

Nuclear Utility Group on
Equipment Qualification

<u>Intervenor</u>

ON PETITION FOR REVIEW OF A RULE
OF THE NUCLEAR REGULATORY COMMISSION

SUPPLEMENTAL BRIEF FOR INTERVENOR NUCLEAR
UTILITY GROUP ON EQUIPMENT QUALIFICATION
PURSUANT TO THE COURT'S OCTOBER 19, 1983 ORDER

<u>STATEMENT</u>

The Court's June 30, 1983 decision in this case vacated the Commission's interim rule on equipment qualification on three procedural grounds: failure to observe the procedural requirements of section 189(a) of the Atomic Energy Act, 42 U.S.C. §2239(a), as amended; failure to conform to the procedural requirements of the NRC's Rules of Practice and

Procedure; and failure to show "good cause" for dispensing with the notice and comment requirements of the Administrative Procedure Act.

The NRC, supported by Intervenor, has moved the Court to vacate that portion of its opinion discussing section 189(a). As the Commission observed, the issue was not briefed by the parties, and the Court's decision appears inconsistent with longstanding NRC practice.

The Court's order directed simultaneous briefs "on the question of the interpretation of Section 189(a) of the Atomic Energy Act and the good cause provision of the Administrative Procedure Act."

ARGUMENT

I. The APA's "Good Cause" Exceptions Apply To NRC Rulemaking

The Court's opinion states that the notice and hearing requirement of Atomic Energy Act section 189(a) "may not be circumvented through the 'good cause' exception of the APA" (opinion, p. 22). On the contrary, the good cause exceptions do apply to NRC rulemaking, as we show below.

A. Longstanding And Consistent NRC Interpretation Has Applied The "Good Cause" Provision In APA Section 553(b) To Rulemaking

The Court's conclusion that "[t]he hearing and notice requirements of the Atomic Energy Act are not subject to the 'good cause' exception of the APA" (opinion, p. 22) contradicts twenty-eight years of practice by the NRC and its predecessor, the Atomic Energy Commission. In fact, a number of AEC/NRC rules issued as long ago as 1955 and continuing to the present time have been adopted without notice and comment. See JOINT COMMITTEE ON ATOMIC ENERGY, IMPROVING THE AEC REGULATORY PROCESS, 87th Cong., 1st Sess., Vol. II, Appendix at 151 (1961); see also the NRC's August 3, 1983 motion to amend the Court's opinion (at 3-4).

With reference to the NRC, this Court has observed, "the Supreme Court has instructed us to give particular deference to the [Commission's] construction of [its] statute * * *." Natural Resources Defense Council, Inc. v. NRC, 666 F.2d 595, 603 (D.C. Cir. 1981). This is particularly true where, as in this case, the interpretation is consistent and longstanding. United States v. Rutherford, 442 U.S. 544, 553-54 (1979); Natural Resources Defense Council, Inc. v. NRC, 582 F.2d 166, 171 (2d Cir. 1978). Indeed, under these circumstances a court is loath to reverse the agency's interpretation even though the court's own construction, if it were writing on a clean slate, might be a different one. Udall v. Tallman, 380 U.S. 1, 16 (1965); see also National Ass'n of Postal Supervisors v. USPS, 602 F.2d 420, 432-33 (D.C. Cir. 1979).[1]

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[1] One additional point should be noted regarding the NRC's interpretation of §189. That interpretation has been brought to Congress' attention--specifically, the Joint Committee on Atomic Energy--and not disturbed. See JOINT COMMITTEE ON ATOMIC ENERGY, IMPROVING THE AEC REGULATORY PROCESS, 87th Cong., 1st Sess., Vol. II, Appendix at 151 (1961). While not dispositive, this further supports the conclusion that the NRC's approach has been legally correct. See Power Reactor Development Co. v. International
(footnote continued)

Surely a case such as this one, where interpretation of section 189(a) is not necessary to the Court's decision, is not an appropriate proceeding in which to jettison the agency's longstanding practice.

B. The Court's Discussion Of The APA's "Good Cause" Exception Is Incomplete And Erroneous

APA section 553(b)'s proviso states that the exceptions to the notice requirement in that statute are not available "when notice or hearing is required by [another] statute". Based on the premise that Atomic Energy Act section 189(a) imposes a notice and hearing requirement on all NRC rulemaking, the Court concludes that APA section 553(b)'s "good cause" exception is not available to the NRC. This construction, however, overlooks the provisions of APA section 553(d) and misconstrues the intent of section 189(a) of the Atomic Energy Act.

1. Section 553(d) Provides An Exception To The Notice Requirement

Section 553(d) of the APA authorizes an agency like the NRC to make a regulation effective immediately and without prior notice. See United States Steel Corp. v. USEPA, 605 F.2d 283, 286, 289 (7th Cir. 1979), cert. denied, 444 U.S. 1035 (1980);[2] Cf., Connecticut Light and Power Co. v. NRC,

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(footnote continued from previous page)
Union of Electrical, Radio & Machine Workers, 367 U.S. 396, 408 (1961); Kay v. FCC, 443 F.2d 638, 646-47 (D.C. Cir. 1970).

673 F.2d 525, 534 (D.C. Cir.), cert. denied, 103 S. Ct. 79 (1982). Of the three circumstances in which section 553(d) authorizes immediately effective rules, two are directly applicable to this case. This is because the interim rule at issue here is "a substantive rule which * * * relieve[d] a restriction," and because the agency found "good cause" to make the rule effective immediately. See APA §§553(d)(1) and (3).

In this regard it should be noted that there is an important distinction between APA sections 553(b) and 553(d). Although section 553(b) grants relief from the prior notice requirement "except when notice or hearing is [otherwise] required," section 553(d) contains no such limitation. Thus, section 553(d) clearly authorizes the NRC to make immediately effective a rule that "relieves a restriction" imposed on licensees.[3]

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2 This Court has distinguished the decision in United States Steel on grounds not relevant here. See State of New Jersey v. USEPA, 626 F.2d 1038 (D.C. Cir. 1980).

3 It is true that §553(d) dispenses only with comments prior to the rule's effective date, and not the opportunity to comment at a later time. See United States Steel Corp. v. USEPA, 605 F.2d at 289-90. Intervenors submit that this additional step was unnecessary in this case because "the Commission ha[d] already solicited comments on the proposed rule's schedule delaying implementation beyond June 30." See J.A. 1.

2. Nor Does APA Section 553(b) Mandate The Court's Conclusion

Furthermore, although the preceding discussion of APA section 553(d) is dispositive of the question here, analysis of section 553(b) also shows that the NRC has authority to make rules effective immediately. Thus, in interpreting section 553(b)'s proviso "except when notice or hearing is required by statute," both sections 181 and 189 of the Atomic Energy Act, 42 U.S.C. §§2231 and 2239, must be read together. In section 181, Congress clearly states that NRC proceedings are to be governed by the APA's procedural requirements--including those in section 553(b). In other words, Congress did not intend that NRC rulemakings would include procedures other than those that section 553 requires in the circumstances of a given case. And APA section 553(b) does not require prior notice and opportunity to comment where in a given case good cause is shown for not affording those procedures. There is no basis for engrafting additional procedures where Congress did not clearly intend them. See Vermont Yankee Nuclear Power Corp. v. Natural Resources Defense Council, Inc., 435 U.S. 519, 543-44 (1978).

II. Any Doubt Concerning The Applicability Of Section 189(a) Should Be Resolved In The First Instance By The NRC, And Not By The Court

Furthermore, aside from the merits of the Court's interpretation of section 189(a), it is important to emphasize that the question whether that statute applied to the rulemaking in this case was not addressed by the Commission and is unnecessary to the Court's decision. Indeed, as noted earlier, the Court's decision rejected the NRC's reasons for dispensing with notice and comment. The panel concluded (at 24-25): "The two reasons given by the Commission as constituting good cause--its inability to promulgate a final rule on time and its reluctance to place licensees in jeopardy of enforcement action pending issuance of the final rule--are insufficient."

The Court's opinion, however, went much further than simply rejecting the sole ground on which the Commission based its decision. Referring to section 189(a), the Court reached the following conclusions, none of which had been mentioned, let alone addressed, by the Commission:

-- the NRC's June 30 rule was "plainly a 'proceeding' for the 'amending of [a] license' within the meaning of section 189a" (opinion, p. 20);[4]

-- the Commission's rule "runs afoul of the express terms of section 189a, which unequivocally requires notice and opportunity to comment" (id.);

-- the conclusion in the final rule (J.A. 1) that "continued operation of these plants pending completion of the equipment qualification program will not present undue risk to the public health and safety" cannot serve as a "no significant hazards" determination within the meaning of section 189(a)[5] because of a statement in the NRC's brief to the Court (id. at 21); and

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4 If read to require that any rulemaking which affects outstanding licenses must employ adjudicatory-type procedures, the Court's opinion would be a departure from Siegel v. AEC, 400 F.2d 778 (D.C. Cir. 1968).

5 The "no significant hazards" provision has never been interpreted as applicable to rulemaking. See also in this connection n. 4, above.

-- the "hearing and notice requirements of the Atomic Energy Act are not subject to the 'good cause' exception of the APA" (id. at 22).

Not only did the Commission's rule not discuss these issues, they were not fully briefed or argued by the parties on appeal.

We suggest that any doubts as to the applicability of section 189(a) be resolved by the Commission in the first instance. "[I]t is a uniform course of appellate review procedure to decline to review questions not necessary to a decision by an appellate court." United Automobile, Aerospace & Agricultural Implement Workers v. NLRB, 462 F.2d 298, 300 (D.C. Cir. 1972). Normally, moreover, the reviewing court will not consider a question "unless the agency has been given a prior opportunity * * * to consider the point at issue." Coffey v. Jordan, 275 F.2d 1, 2 (D.C. Cir. 1959) (per curiam). See also California Interstate Telephone Co. v. FCC, 328 F.2d 556, 559 (D.C. Cir. 1964) ("Judicial review of administrative action is limited to matters upon which the agency has had an opportunity to pass"). Indeed, where portions of the Court's opinion are "unnecessary with respect to [its] decision", the appropriate course is to delete the unnecessary discussion. See Pennzoil Producing Co. v. FPC, 558 F.2d 816, 817 (5th Cir. 1977).

There is, moreover, one further reason for deleting the Court's discussion of section 189(a). On January 3, 1983, Congress responded to the Court's decision in Sholly v. NRC, 651 F.2d 780 (D.C. Cir. 1980), vacated, 75 L.Ed.2d 423 (1983), by amending section 189(a) to redefine the Commission's authority to issue immediately effective license amendments. The so-called Sholly amendments (see the Appendix to this brief) authorize the Commission to amend licenses without prior hearing "notwithstanding the pendency before the Commission of a request for a hearing from any person" upon an NRC determination "that such amendment involves no significant hazards consideration." The amendments also authorize the Commission to dispense with prior notice of and opportunity to comment on proposed "no significant hazards" determinations in defined emergency situations.

The Sholly amendments, which postdate the interim rule in this case, substantially alter the procedures by which the Commission gives public notice and responds to hearing requests respecting license amendments (when it proceeds by adjudication rather then by rulemaking). The Commission has promulgated detailed criteria defining a "no significant hazards" determination pursuant to amended section 189(a). Interim Final Rule, Standards for Determining Whether License Amendments Involve No Significant Hazards Considerations, 48 Fed. Reg. 14864 (April 6, 1983). These considerations further support the need to vacate the Court's discussion of section 189(a). Failure to do so would preempt the Commission's prerogative to interpret the amended statute in the first instance, and may create confusion in the wake of the *Sholly* amendments.

CONCLUSION

The Court should vacate section III.A of its June 30, 1983 opinion in this case.

Respectfully submitted,

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October 31, 1983

Appendix

Section 189(a) of the Atomic Energy Act, 42 United States Code § 2239(a), as amended January 4, 1983, by Public Law 97-415, § 12(a), provides as follows:

§ 2239. Hearings and judicial review

(a)(1) In any proceeding under this chapter, for the granting, suspending, revoking, or amending of any license or construction permit, or application to transfer control, and in any proceeding for the issuance or modification of rules and regulations dealing with the activities of licensees, and in any proceeding for the payment of compensation, an award or royalties under sections 2183, 2187, 2236(c) or 2238 of this title, the Commission shall grant a hearing upon the request of any person whose interest may be affected by the proceeding, and shall admit any such person as a party to such proceeding. The Commission shall hold a hearing after thirty days' notice and publication once in the Federal Register, on each application under section 2133 or 2134(b) of this title for a construction permit for a facility, and on any application under section 2134(c) of this title for a construction permit for a testing facility. In cases where such a construction permit has been issued following the holding of such a hearing, the Commission may, in the absence of a request therefor by any person whose interest may be affected, issue an operating license or an amendment to a construction permit or an amendment to an operating license without a hearing, but upon thirty days' notice and publication once in the Federal Register of its intent to do so. The Commission may dispense with such thirty days' notice and publication with respect to any application for an amendment to a construction permit or an amendment to an operating license upon a determination by the Commission that the amendment involves no significant hazards consideration.

(2)(A) The Commission may issue and make immediately effective any amendment to an operating license, upon a determination by the Commission that such amendment involves no significant hazards consideration, notwithstanding the pendency before the Commission of a request for a hearing from any person. Such amendment may be issued and made immediately effective in advance of the holding and completion of any required hearing. In determining under this section whether such amendment involves no significant hazards consideration, the Commission shall consult with the State in which the facility involved is located. In all other respects such amendment shall meet the requirements of this chapter.

(B) The Commission shall periodically (but not less frequently than once every thirty days) publish notice of any amendments issued, or proposed to be issued, as provided in subparagraph (A). Each such notice shall include all amendments issued, or proposed to be issued, since the date of publication of the last such periodic notice. Such notice shall, with respect to each amendment or proposed amendment (i) identify the facility involved; and (ii) provide a brief description of such amendment. Nothing in this subsection shall be construed to delay the effective date of any amendment.

(C) The Commission shall, during the ninety-day period following the effective date of this paragraph, promulgate regulations establishing (i) standards for determining whether any amendment to an operating license involves no significant hazards consideration; (ii) criteria for providing or, in emergency situations, dispensing with prior notice and reasonable opportunity for public comment on any such determination, which criteria shall take into account the exigency of the need for the amendment involved; and (iii) procedures for consultation on any such determination with the State in which the facility involved is located.

CERTIFICATE OF SERVICE

I hereby certify that two copies of the foregoing Supplemental Brief of Intervenor were served by first class mail, postage prepaid, this 31st day of October, 1983, on the following:

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Scott M. DuBoff
Scott M. DuBoff

ATTACHMENT C

October 14, 1983

Mr. D. K. Porter/NES File PT8.1.1:

TRIP REPORT ON MEETING WITH NRC STAFF
TO RESOLVE EQ SER DEFICIENCIES

On October 14, 1983 Mr. Roger Newton and I met with members of the NRC Staff at the NRC offices in Bethesda, Maryland to resolve the deficiencies listed in the NRC's Safety Evaluation Reports (SERs) regarding "Environmental Qualification of Safety-Related Electrical Equipment" at Point Beach Nuclear Plant, Units 1 and 2. Members of the NRC Staff included Tim Colburn, our NRC Project Manager, Bob LaGrange and Paul Shemanski from the Equipment Qualification Branch, and Jose Calvo from the Operating Reactors Assessment Branch. I presented detailed proposed technical resolutions for each of the deficiencies listed in the SERs (see attached agenda and detailed agenda used for the meeting). An acceptable resolution was reached between the NRC Staff and ourselves (essentially the ones I proposed) for each deficiency.

The NRC Staff appeared strongly interested in resolving the SER deficiencies and in fact resolving the entire environmental qualification issue for all equipment items within the scope of 10 CFR 50.49. It was agreed that Wisconsin Electric would submit a letter stating the status of qualification of each Point Beach Nuclear Plant equipment item within the scope of 10 CFR 50.49 including a summary of the resolution reached for each SER deficiency. An updated justification for continued operation (JCO) for each item not yet qualified would also have to be provided unless a previously submitted JCO which is still valid could be referenced. We were also requested to confirm that the list of Point Beach Nuclear Plant equipment within the scope of 10 CFR 50.49 (b) (1) (i.e., our May 20, 1983 submittal) addressed all equipment relied upon for all design-basis events including "flooding outside containment." The Staff expects us to reference previous analysis of flooding and not to begin a new study.

It was suggested that this letter be submitted in about a month or sooner. The NRC Staff would then issue supplemental SERs, with our letter as an attachment, which would officially resolve the environmental qualification issue on the Point Beach Nuclear Plant dockets. I will, of course, be preparing the above letter.

R. K. Hann

R. K. Hanneman

/bjm

Attachments

AGENDA FOR
WISCONSIN ELECTRIC MEETING WITH NRC STAFF TO
RESOLVE ENVIRONMENTAL QUALIFICATION SER DEFICIENCIES

1. Generic EQ Deficiencies

2. Specific Equipment EQ Deficiencies

    Pressure, D/P, and Level Transmitters

    Solenoid Valves for Air-Operated Valves

    Motors (including splices and bearing/lubricant)

    Temperature Measurement Devices

    Electro-Pneumatic (I/P) Transducers

    Electrical Distribution Devices

    Limit Switches for Air-Operated Valves/Acoustical Monitors

    Motor Operated Valves (including lubricants)

3. May 20, 1983 Response to 10 CFR 50.49 and Extension Requests

4. September 1, 1983 Response to Regulatory Guide 1.97

DETAILED AGENDA
WISCONSIN ELECTRIC MEETING WITH NRC STAFF TO RESOLVE ENVIRONMENTAL QUALIFICATION SER DEFICIENCES

I. Generic EQ Deficiences:

	Deficiency	Reference	Proposed Resolution
A.	Lists of Safety-Related Systems "Control habitability and safety equipment area ventilation should be part of the general equipment listed with the supporting systems." "Subject to the above verification by the Licensee, this item is considered resolved."	TER, App. C. p.C-5	This equipment was evaluated as part of the heating and ventilating system but is located in a mild environment.
B.	Installation Date of TMI Action Plan Equipment "The approximate installation date for the TMI Action Plan equipment items is requested so that the appropriate qualification criteria (NUREG-0588 or DOR Guidelines) can be used in the EEQ evaluation."	TER, App. E, p. E-10	Approximate installation dates were provided in our May 20, 1983 response (revised October 10, 1983) to 10 CFR 50.49 and our September 1, 1983 response to NRC Generic Letter No. 82-33 (i.e., Reg. Guide 1.97).

II. Specific Equipment EQ Deficiencies:

A. Pressure, D/P, and Level Transmitters

Item No.	Description	NRC Category	Deficiencies	Proposed Resolution
1	1&2-PT922&923 (SI Pump Discharge Pressure)	I.B	Evaluation of Aging Degradation; Qualified Life	Replacement with qualified Foxboro N-E10 Series
2, 3, 4	1&2-LT931 (Containment Spray Additive Tank Level) 1&2-FT928 (Low-Head SI [Train B] Flow)	I.B	Documentation	Replacement with qualified Foxboro N-E10 Series
5	1&2-LC942A&B, 943A&B (Containment Sump B Level)	I.B	Documentation	Replacement with qualified Gems DeLaval level transmitters
6, 7, 8	1&2-FT619 (Component Cooling Flow) 1&2-FT626 (Low-Head SI [Train A] & RHR Flow) 1&2-PT628 & 629 (RHR Pump Discharge Pressure)	I.B	Similarity; Evaluation of Aging Degradation	Replacement with qualified Foxboro N-E10 Series
9	1&2-PT945 thru 950 (Containment Narrow and Intermediate-Range Pressure)	I.B	Documentation	Replacement with qualified Foxboro N-E10 Series
10, 11	1&2-PT429, 430, 431, & 449 (Pressurizer Narrow-Range Pressure) 1&2-LT426, 427, 428, & 433 (Pressurizer Water Level)	I.B	Similarity; Evaluation of Aging Degradation; Radiation; Test Sequence	Replacement with qualified Foxboro N-E10 Series
12	1&2-LT106, 172, 190; LT102, 171, & 189 (BAST Water Level)	I.B	Similarity; Evaluation of Aging Degradation	Replacement with qualified Foxboro N-E10 Series

A. Pressure, D/P, and Level Transmitters (continued)

Item No.	Description	NRC Category	Deficiencies	Proposed Resolution
13, 14	1&2-PT420 (RCS Wide-Range Pressure) 1&2-LT461, 462, 463, 471, 472, & 473 (S/G Narrow-Range Water Level) 1&2-LT460 & 470 (S/G Wide-Range Water Level)	I.B	Similarity; Evaluation of Aging Degradation; Radiation; Test Sequence	Replacement with qualified Foxboro N-E10 Series
15	1&2-FT464, 465, 474, & 475 (Main Steam Line Flow)	I.B	Documentation	Replacement with qualified Foxboro N-E10 Series
16, 17	1&2-PT468, 469, 478, 479, 482, & 483 (Main Steam Line Pressure) LT4025 & 4031 (CST Water Level)	I.B	Similarity; Evaluation of Aging Degrdation	Replacement with qualified Foxboro N-E10 Series

B. Solenoid Valves for Air Operated Valves

Item No.	Description	NRC Category	Deficiencies	Proposed Resolution
18,19,20 21,22,23	1&2-SV966C (Solenoid Valves for RCS Hot Leg Sample AOVs) 1&2-RC430 & 431C (Solenoid Valves for pressurizer PORVs) 1&2-HV3213 & 3245 (Solenoid Valves for Containment Purge Supply & Exhaust AOVs) 1&2-HV3200C (Solenoid Valves for Containment RMS Sampling Line AOVs) 1&2-CV1296 (Solenoid Valves for Auxiliary Charging Line AOVs) 1&2-SV951, 953, & 955 (Pressurizer Steam & Liquid Space and RCS Hot Leg Sample Line AOVs)	I.B	Documentation	Replacement with qualified ASCO NP Series

C. Motors

Item No.	Description	NRC Category	Deficiencies	Proposed Resolution
24, 50	1&2-P15A & B (SI Pump Motors, Splices and Bearing/Lubricant)	II.A	Documentation; Similarity; Evaluation of Aging Degradation; Qualified Life/Replacement Schedule; Aging Program; Aging Simulation; Peak Temperature; Radiation	Additional analysis and documentation
25,52,55	1&2-P14A & B (Containment Spray Pump Motors, Splices, and Bearing/Lubricant)	II.A	Documentation; Similarity; Evaluation of Aging Degradation; Qualified Life/Replacement Schedule; Aging Program; Aging Simulation; Peak Temperature; Radiation	Additional analysis and documentation
26,51,55	1&2-P11A & B (Component Cooling Pump Motors, Splices, and Bearing/Lubricant)	II.A	Documentation; Similarity; Evaluation of Aging Degradation; Qualified Life/Replacement Schedule; Aging Program; Aging Simulation; Peak Temperature; Radiation	Additional analysis and documentation
27,52,55	1&2-P10A & B (RHR Pump Motors, Splices, and Bearing/Lubricant)	II.A	Documentation; Similarity; Evaluation of Aging Degradation; Qualified Life/Replacement Schedule; Aging Program; Aging Simulation; Peak Temperature; Aging Degradation Program; Radiation	Additional analysis and documentation
28,29,53	1&2-W1A1, B1, C1, & D1 (Containment Emergency Fan Cooler Motors, Splices, and Bearing/Lubricants)	II.A	Documentation; Similarity; Evaluation of Aging Degradation; Qualified Life/Replacement Schedule; Aging Simulation; Peak Temperature; Radiation; Beta-emitter Plateout	Additional analysis and documentation

D. Temperature Measurement Devices

Item No.	Description	NRC Category	Deficiencies	Proposed Resolution
30	1&2-TE621, 627, & 630 (Component Cooling HX Outlet and RHR HX Outlet and Inlet RTDs)	II.C	Documentation; Evaluation of Aging Degradation	Replacement with qualified Conax RTDs for TE621. Substitution of TE622 & 623 for TE627. TE630 is only required for cold shutdown and not in scope of 10 CFR 50.49.
31, 37, 46, 47	1&2-TE1-39 (RCS Core Exit Thermocouples, Connectors, Extension Cables, and Reference Junction Boxes)	I.B	Documentation	Additional analysis and documentation of T/Cs; Replacement of connectors, cables, splices, penetrations with qualified components; Replacement of reference junction boxes and relocation in a mild environment.
32, 69 (Unit 2 only)	1&2-450A & B and 451A & B (RCS Hot and Cold Leg Wide-Range Loop RTDs)	I.B	Documentation; Evaluation of Aging Degradation; Qualified Life/Replacement Schedule; Functional Testing; Instrument Accuracy	Replacement with qualified Conax dual-element RTDs

E. Electro-Pneumatic (I/P) Transducers

Item No.	Description	NRC Category	Deficiencies	Proposed Resolution
33	1&2-AC624, 625, & 626; (Electro-pneumatic Transducers for RHR HX Outlet and Bypass AOVs) 1&2-SI836A & B (Electro-pneumatic Transducers for CS Additive Line AOVs)	I.A	None	N/A

F. Electrical Distribution Devices

Item No.	Description	NRC Category	Deficiencies	Proposed Resolution
34	1&2-B32 (Motor Control Centers)	I.B	Documentation (Radiation only)	Radiation Shielding Installed
35	Containment Electrical Penetration Assembly (Westinghouse/Crouse-Hinds Welded Canister Type)	I.A	None	N/A
36	Electrical Cable Splice Inside Containment (Bechtel Dwg. SK-E-165/Raychem Type SFR)	I.A	None	N/A
39	2-RC515, 1&2-RC516 (Rome 600 V. Control Cable for PORV Blocking Valves Inside Containment)	I.B	Documentation	PORV Block Valves are not safety-related (i.e., not required for safe shutdown or design-basis accident mitigation)
	1&2-LC942A & B 943A & B (Rome 600 V. Control Cable for Containment Sump Level Switches)	I.B	Documentation	Replacement of level switches and cable with qualified components.
	Various Plant IDs (Rome 600 V. Control Cable Outside Containment)	I.B	Documentation	Additional Analysis and Documentation
40	Various Plant IDs (BIW Bostrad 7 TSP Instrument Cable Located Inside and Outside Containment)	I.A	None	N/A
41	1&2-P15A & B (Okonite Okonex Insulated, Okoprene-Jacketed 5 KV Power Cable for SI Pump Motors)	I.A	None	N/A
42	Various Plant IDs (Okonite Okotherm-Insulated, Okoseal-Jacketed TSP Instrument Cable)	II.A	Similarity	Additional documentation
43	Various Plant IDs (Kerite HTK-Insulated, FR-Jacketed 600 V. Power and FR-Insulated and Jacketed 600 V Control Cables Located Inside and Outside Containment)	I.A	None	N/A
44, 45	1&2-T1C & D (Pressurizer Safeguards-Powered Backup Heaters and Cable Connectors)	III.A & B	None	Cold Shutdown Equipment Only and not within scope of 10 CFR 50.49

G. Limit Switches for Air Operated Valves/Acoustical Monitors

Item No.	Description	NRC Category	Deficiencies	Proposed Resolution
38,48,49	1&2-PCV434 & 435 (Acoustical Monitors, Signal Cable and Connectors for Fluid Leak Detection on Pressurizer Code Safety Valves)	I.B	Documentation	Replacement with qualified Crosby Lift Indicating Switch Assemblies
57	1&2-SV966C, AC624, 625, & 626, & SI836A & B (Limit Switches on AOVs Outside Containment)	I.B	Documentation	Replacement with qualified NAMCO EA-180 limit switches and Conax ECSAs
58,59,60	1&2-RC430 & 431C; 1&2-SV951, 953, & 955 1&2-CV1296; 1&2-HV3200C, 3213, & 3245 (Limit switches on AOVs Inside Containment)	I.B	Documentation; Similarity	Replacement with qualified NAMCO EA-180 limit switches and Conax ECSAs

H. Motor-Operated Valves

Item No.	Description	NRC Category	Deficiencies	Proposed Resolution
61,54,56	1&2-RC515 & 516 (Limitorque MOVs and Lubricants for PORV Blocking Valves Inside Containment)	II.A	Documentation; Similarity; Evaluation of Aging Degradation; Qualified Life/ Replacement Schedule; Aging Program; Aging Simulation; Peak Temperature; Radiation	Additional Analysis and Documentation
62,54,56	1&2-SI878A & C (Limitorque MOVs and Lubricants for Reactor Vessel SI Line Valves Inside Containment)	II.A	Documentation; Similarity; Evaluation of Aging Degradation; Qualified Life/ Replacement Schedule; Aging Program; Aging Simulation; Peak Temperature; Radiation	Additional Analysis and Documentation
63,54,56	1&2-SI852A & B and 878B & D (Limitorque MOVs and Lubricants for Low-Head SI and Cold-Leg SI Line Valves Inside Containment)	II.A	Documentation; Similarity; Evaluation of Aging Degradation; Qualified Life/ Replacement Schedule; Aging Program; Aging Simulation; Peak Temperature; Radiation	Additional Analysis and Documentation
64,54,56	1&2-SI871A & B and SI860A, B, C, & D (Limitorque MOVs and Lubricants for RHR/CS X-connect and CS Discharge Line Valves Outside Containment)	II.A & C	Documentation; Similarity; Evaluation of Aging Degradation; Qualified Life/ Replacement Schedule; Aging Program; Aging Simulation; Peak Temperature; Radiation	Additional Analysis and Documentation
65,54,56	1&2-MS2019 & 2020 (Limitorque MOVs and Lubricants for AFW Pump Turbine Steam Supply Line Valves Outside Containment)	II.A	Documentation; Similarity; Evaluation of Aging Degradation; Qualified Life/ Replacement Schedule; Aging Program; Aging Simulation; Peak Temperature; Radiation	Additional Analysis and Documentation

H. Motor-Operated Valves (continued)

Item No.	Description	NRC Category	Deficiencies	Proposed Resolution
66, 70 (Unit 2 only) 54, 56	1&2-AC738A & B and SI851A & B (Limitorque MOVs and Lubricants for CC/RHR HX Line & RHR Containment Sump Suction RHR Backup Valves Outside Containment)	II.A & C	Documentation; Similarity; Evaluation of Aging Degradation; Qualified Life/ Replacement Schedule; Aging Program; Aging Simulation; Peak Temperature; Radiation	Additional Analysis and Documentation
67, 68 54, 56	1&2-AC700, 701, & 720 (Limitorque MOVs and Lubricants for for RHR Suction and Discharge Lines Isolation Valves Inside Containment)	III.B	None	Cold Shutdown only and not within scope of 10 CFR 50.49.