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July 2 , 1981
LIL 202

Office of Nuclear Reactor Regulation
Attn: John F. Stolz, Chief
Operating Reactors Branch No. 4
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Sir:

Three Mile Island Nuclear Station, Unit 1 (TMI-1)
Operating License No. DPR-50
Docket No. 50-289
Environmental Qualification of Electrical Equipment
(Response to NRC Staff SER)

This letter is in response to your letters dated March 24, 1981 and April 23, 1981 which provided the NRC staff Safety Evaluation Report (SER) concerning Environmental Qualification of Safety-Related Electrical Equipment. This letter provides responses to the general issues identified in the SER, as an attachment, but does not address the component specific concerns identified in the SER. A response which addresses the component specific concerns will be filed by July 20, 1981 and will consist of revised or new component work/evaluation sheets. Due to the extensive number of pages in our previous response which are affected by the changes (all previously issued volumes will be replaced since nearly all pages are affected to some degree) our review of the revised sheets will not be completed until July 20 1981. In addition the up-coming meeting in Bethesda (July 7-10) on Environmental Qualification may have an impact on our responses for individual items.

The July 20, 1981 response will not address the qualifications of TMI Lessons Learned equipment (NUREG-0578) since the installation of this equipment is not complete and the environmental qualification data is not fully assembled.

Sincerely,

H. D. Hukill
H. D. Hukill
Director, TMI-1

HDH:CWS:vjf

Enclosure

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RESPONSE TO ISSUES RAISED IN THE NRC STAFF SAFETY EVALUATION REPORT (SER)
CONCERNING ENVIRONMENTAL QUALIFICATION OF
SAFETY-RELATED ELECTRICAL EQUIPMENT

I. Paragraph 3.1 Completeness of Safety-Related Equipment

"Display instrumentation which provides information for the reactor operators to aid them in the safe handling of the plant was not specifically identified by the licensee. A complete list of all display instrumentation mentioned in the LOCA and HELB emergency procedures must be provided.

Equipment qualification information in the form of summary sheets should be provided for all components of the display instrumentation exposed to harsh environments. Instrumentation which is not considered to be safety related but which is mentioned in the emergency procedure should appear on the list. For these instruments, (1) justification should be provided for not considering the instrument safety related and (2) assurance should be provided that its subsequent failure will not mislead the operator or adversely affect the mitigation of the mitigation of the consequences of the accident."

Comment: A "Master List" of instruments mentioned in the appropriate LOCA/HELB emergency procedures will be provided as part of the July 20, 1981 submittal. Some of the instruments included will not be environmentally qualified. The essential instruments needed to respond to these events are environmentally qualified and appropriate components evaluation work sheets summarizing their qualifications will be provided.

Other instruments included in the list will not mislead the operator nor result in an adverse affect in mitigation of any accident for these reasons:

1. the operator training program stresses the use of alternate/multiple instruments to assess plant conditions.

2. the emergency procedures also specify assessment and confirmation of indication using alternate indications.

3. As discussed above, all instruments essential in responding to LOCA/HELB are environmentally qualified and considered safety related.

II. Paragraph 3.2 Service Conditions

> "The staff assumed, and required the licensee to verify, that the containment spray system is not subjected to a disabling single-component failure and therefore satisfies the requirements of Section 4.2.1 of the DOR quidelines."

Comment: As assumed by the NRC staff, TMI-1 does have a containment spray system not subject to a disabling single component failure. That is, no single failure will result in the inability of the Reactor Building Spray System to deliver spray in sufficient amounts to minimize containment pressure and temperature. The RBSS is currently under review with respect to chemical performance in response to NRC letter dated March 7, 1980. The results of this review will be forwarded by the end of July, 1981 and will demonstrate RBSS performance within the range specified for equipment qualification.

III. Paragraph 3.3 Temperature, Pressure, and Humidity Conditions Inside Containment

A. "The Licensee has provided the results of accident analyses as follows: LOCA Max Temp (°F) 275..."

B. "The staff has concluded that the minimum temperature profile for equipment qualification purposes should include a margin to account for higher-than-average temperatures in the upper regions of the containment that can exist due to stratification, especifally following a postulated MSLB. Use of the steam saturation temperature corresponding to the total building pressure (partial pressure of steam plus partial pressure of air) versus time will provide an acceptable margin for either a postulated LOCA or MSLB, whichever is controlling, as to potential adverse environmental effects on equipment.

The licensee's specified temperature (service condition) of 275°F does not satisfy the above requirement. A saturation termperature corresponding to the peak profile (298°F peak temperature at 50.6 psig) should be used instead. The Licensee should update his equipment summary tables to reflect this change. If there is any equipment that does not meet the staff position, the licensee must provide either justification that the equipment will perform its intended function under the specified conditions or propose corrective action."

Comment: The temperature profile provided in our response indicated a peak temperature of 286°F not 275°F as assumed by the NRC staff. This value was conservatively calculated as discussed in the FSAR Sections 14.2.2.3 and 6.1.2.12. The conservatism inherent in the calculations are ample assurance that the worst containment conditions have been bounded. Furthermore, neither the IE Bulletin 79-01B nor the Commissions Order (CLI-80-21) require going beyond the DOR Guidelines which state in part "...equipment qualified for a LOCA environment is considered qualified for MSLB accident environment in plants with automatic spray systems not subject to a disabling single component failure." (The TMI-1 system is such a system as discussed in 3.2 above.) This recent NRC staff criteria revision clearly goes beyond the DOR Guidelines without adequate justification.

IV. Paragraph 3.5 Submergence

> "The licensee's value for maximum submergence is 5.94 ft."
>
> "The licensee states that flood levels associated with breaks where steam generator water level is needed will not subject the level transmitters to submergence. However the staff does not agree with the licensee, because the flood level should be the same across the spectrum of postulated events.

Comment: The maximum containment sump level has been recalculated using a more realistic but still conservative model of the steam generator exterior surface (for determination of non-floodable volume). The revised flood level is 5.66 ft above the containment floor. For MSLB a lower flood level was calculated. The LOCA flood level was nevertheless conversatively used to determine equipment qualification. The Steam Generator level transmitters have been relocated such that the lowest instrument is greater than 5.80 ft above the containment floor.

V. Paragraph 3.6 Chemical Spray

"The licensee's FSAR value for the chemical concentration is 2270 ppm boron and sodium hydroxide (NaOH) to raise the pH to 9.5. Only the work sheet for Raychem heat shrink tubing provides enough information to show qualification for chemical spray. Therefore, for the purpose of this review the effects of chemical spray will be considered unresolved. The staff will review the licensee's response when it is submitted and discuss the resolution in a supplemental report."

Comment: Justification for the pH valves used for equipment qualification will be provided on the component evaluation sheets to be provided with our July 20, 1981 submittal. It should be noted that ability of a component to withstand the effects of chemical spray depends on more than pH alone. Temperature, time, material, thickness, and material properties are amoung some of the more important considerations. Research and testing, such as that reported in ORNL Report TNI-24-12 show the significance of some of these factors.

VI. Paragraph 3.7 Aging

"Section 7 of the DOR guidelines does not require a qualified life to be established for all safety-related electrical equipment. However, the following actions are required.

(1) Make a detailed comparison of existing equipment and the materials identified in Appendix C of the DOR quidelines. The first supplement to IEB-79-01B requires licensees to utilize the table in Appendix C and identify any additional materials as the result of their effort.

(2) Establish an ongoing program to review surveillance and maintenance records to identify potential age-related degradations.

(3) Establish component maintenance and replacement schedules which include considerations of aging characteristics of the installed components.

For this review, however, the staff requires that the licensee submit supplemental information to verify and identify the degree of conformance to the above requirements. For equipment for which a material evaluation has already been performed, items (2) and (3) above should be addressed. The response should include all the equipment identified as required to maintain functional operability in harsh environments."

Comment: As described in our letter dated June 12, 1981, we have improved our preventitive maintenance program (procedures) to require replacement of items in accordance with established replacement schedules. All items with a qualified life less than the life of the plant will be replaced according to the established schedules. These procedures will be in place prior to restart criticality. Any item with a known qualified life which has been exceeded will be replaced before criticality. The only such components which have exceeded that qualified life identified to date are neoprene cover gaskets on certain pressure switches and transmitters. A program has been established to capture for evaluation the surveillance maintenance history for all safety related equipment. The collected information will be periodically reviewed to determine and identify age-related degradation. (The offsite engineering resources of GPU Nuclear will conduit these reviews.) The reviews will be ongoing but may be at distinct intervals in lieu of continuous reviews.

VII. Paragraph 3.8 Radiation (Inside and Outside Containment)

A. "The value required by the licensee inside containment is an integrated dose of 2×10^7 rads. The radiation service condition provided by the licensee is lower than provided in the DOR quidelines for gamma and beta radiation. The licensee is requested to either provide justification for using the lower service condition or use the service condition provided in the DOR quidelines for both gamma and beta radiation."

B. "For some equipment, the licensee has used a calculated integrated radiation dose at 1 hour after a LOCA as the required radiation environment. Without additional justification (as discussed in Section 4.2 of this SER), specifying these lower radiation levels is not acceptable and, therefore, radiation has been listed as a deficiency for these components."

C. "Radiation has also been identified as a deficiency if the licensee used other than the values listed in Appendix C of the DOR quidelines unless the Appendix C values also enveloped the specified value.

Comment: A. The DOR quidelines permit the use of a dose of 2×10^7 Rads integrated dose for gamma radiation in lieu of a detailed plant specific calculation. We have elected, for convenience, to use this conservative integrated dose for TMI-1. We have not included the integrated dose associated with beta radiation since the safety related equipment in the TMI-1 containment is in metal enclosures or conduit. The conduit and enclosures while not necessarily gas tight do shield the sensitive materials from essentially all of the beta source term which is external to the conduit/enclosures. The insignificant beta dose from gases potentially within the enclosures/conduit was considered too trival to effect equipment qualification (i.e. several orders of magnitude lower than 2×10^7 Rads used for equipment qualification.)

B. Justification for use of an integrated dose at one hour was previously provided, however, it was not presented in the format suggested in section 4.2 of the SER. Our July 20, 1981 submittal will restate the justification in the suggested format.

C. Appendix C of the DOR guidelines did not indicate the specific source reference or criteria associated with the values listed. Other source documents were used to determine radiation values and included a review of the criteria used to determine that value and the properties required in the specific application.

In determining the post accident radiation qualification of safety-related electrical equipment, the calculated post accident integrated exposures were compared to the radiation tolerance values found in the literature for each of the component's constituent materials. The documents reviewed were:

Paragraph 3.8 Radiation (Inside and Outside Containment) cont.

1. "The Use of Plastics and Elastomers in Nuclear Radiation:, by W. W. Parkinson and O. Sisman, Oak Ridge National Laboratory, October 19, 1970, appearing in "Nuclear Engineering and Design 17", (1971), pp. 247-280.

2. "Effects of Radiat. .n on Materials and Components", M.H. Van de Voorde, CERN 70-5, European Organization for Nuclear Research, February 26, 1970.

3. "Nuclear Engineering Handbook", Harold Etherington - editor, McGraw-Hill, Inc., First Edition, Section 10, 1958.

4. "A Review of Equipment Aging Theory and Technology", Franklin Research Center, EPRI Report NP-1558, September, 1980.

5. "Radiation Stability of Plastics and Elastomers", by C.D. Bopp and O. Sisman, Oak Ridge National Laboratory, appearing in "Nucleonics", Volume 13, No. 7, July 1955, pp. 28-33.

In addition to providing documentable references, many of these reports are based on actual tests conducted by the sponsoring organization. The Parkinson-Sisman study (Item 1), the most detailed and complete reference work available, including substantiating tests, was utilized as the basis for radiation tolerance values whenever the equipment manufacturers did not provide an adequately referenced value. In addition to supplying limiting values for onset of degradation (threshold values), this report provided the range of degradation as a function of total exposure for several material properties (i.e.; elongation, tensile strength, compression set....). This allows one to choose not only the material property most critical for continued operation of the component in question, but also to assess the allowable degradation prior to failure.

VIII. Paragraph 4.1 Equipment Requiring Immediate Corrective Action

> "Appendix A identifies equipment (if any) in this category. The licensee was asked to review the facility's safety-related electrical equipment. The licensee's review of this equipment has identified six Limitorque motor operatiors requiring immediate corrective action; therefore, licensee event report (LER) 80-17 was submitted. The licensee states that these motor operators will either be qualified or replaced by the end of the first refueling after restart. The staff does not find this schedule acceptable and, therefore, the licensee must commit to either showing that these motor operators are qualified or replace them by June 30, 1982."

Comment: It is our intent to modify the subject limitorque motor operators by June 30, 1982 so that they will be qualified. It should be recognized, however, that we, like other licensees, are competing for qualified equipment from a single source and may not be able to obtain the necessary equipment on this schedule.