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SUBJECT: Forwards response to all open items identified in SER & technical evaluation rept except for deferred mild environ & TMI-related equipment. Installed safety-related equipment capable of performing all required safety functions.
SEE Repts. # 8109100183

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September 4, 1981

Director of Nuclear Reactor Regulation
Attention: Dennis M. Crutchfield, Chief
Operating Reactors Branch No. 5
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Subject: Environmental Qualification of Safety-Related Electrical Equipment
R. E. Ginna Nuclear Power Plant
Docket No. 50-244

Dear Mr. Crutchfield:

This letter is in response to your letter of June 1, 1981, transmitting the "Safety Evaluation Report by the Office of Nuclear Reactor Regulation" and "Technical Evaluation Report, Equipment Environmental Qualification", (TER-C5257-178) by Franklin Research Center.. The attached information responds to all open items identified in the SER and TER, except for deferred "mild environment" and "TMI-related" equipment.

For certain items, qualification information is still being developed. In those cases, a commitment is made to complete the evaluation/qualification of this equipment by June 30, 1982. Rationale for acceptability of the use of the installed equipment in the interim is also provided. It must be pointed out, however, that the June 30, 1982 date may not be achievable for all modifications. Equipment procurement and delivery schedules, testing requirements, and installation days (especially as related to scheduled refueling shutdowns) may require a delay in the installation of certain modifications. Another major source of delay results from new issues, which we consider go beyond the DOR Guidelines, brought up in the SER/TER. These issues include the additional containment temperature margin requirements, the dose rate concern presented in Appendix H of the TER, and the complete resolution of the aging/qualified life concerns. Specific comments are provided in the attachments to this letter. Finally, a resolution of disagreements between RG&E and FRC/NRC relative to system operational considerations (see Appendix E of the TER) is required.

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In a number of instances in the attachments, RG&E notes additions or corrections required to our October 31, 1980 report. This report will be resubmitted in the near future. RG&E also makes reference to additional documentation in responding to certain TER concerns. This documentation (with the exceptions of References 61 and 72, included with this letter) will also be transmitted in the near future.

Based on the information provided in response to the SER/TER, RG&E considers the installed safety-related electrical equipment to be capable of performing all required safety functions, such that the safe, continued operation of Ginna Station is warranted.

Very truly yours,

John E. Maier

John E. Maier

Attachments

Attachment 1: Response to June 1, 1981 SER, attached to June 1, 1981 letter, "Environmental Qualification of Safety-Related Electrical Equipment"

1. In Section 3.2, it is stated that the NRC requires the licensee to verify that the Containment Spray System is not subject to a disabling single component failure. This has been done. The Containment Spray System is a fully-redundant system, not subject to credible disabling single component failures. The single suction line from the RWST, with valves 896A and 896B in series, has been addressed in the Ginna Technical Specification, Section 3.3.1.1.g. RG&E's "ECCS Single Failure" analysis, accepted by the NRC staff in Amendment No. 7, was again reviewed and accepted by letter from Dennis M. Crutchfield, NRC, to John E. Maier, RG&E, "SEP Topics VI-7.C and VI-7.C.2," dated February 20, 1981. The Ginna Containment Spray System thus satisfies the DOR Guideline requirements of Section 4.2.1.

2. In Section 3.3, the NRC concludes that the Ginna specified peak post-LOCA temperature should be the steam saturation temperature corresponding to the total building pressure, rather than the steam saturation temperature corresponding to the partial pressure of steam only. This would include margin to account for higher than average temperatures in the upper regions of the containment that can exist due to stratification, especially following a steam line break.

   RG&E does not intend to modify the Ginna post-accident temperature profile, since we have serious reservations concerning both the method of imposition of this new requirement, and its technical validity for Ginna. As expressed in the attachment to our February 20, 1981 response to the NRC's "Partial Review, Equipment Evaluation Report by the Office of Nuclear Reactor Regulation":

   "a) The purpose of the October 31, 1980 RG&E submittal was to respond to the September 19, 1980 Commission Order, requiring submittal of information to show compliance with the "DOR Guidelines". The "DOR Guidelines" explicitly state that in Section 4.2.1 that "...equipment qualified for a LOCA environment is considered qualified for a MSLB accident environment in plant with automatic spray systems not subject to disabling single component failures." In Appendix A of FRC Project C5257, it is stated that "...the design of the Ginna plant satisfies these criteria." We consider it inappropriate that the staff would modify explicit previous guidance via this "Partial Review". The requirement to meet a 307°F temperature envelope is obviously beyond the requirements of the "DOR Guidelines".

b) The apparent reason for this new requirement is to provide margin to account for higher than average temperatures in the upper regions of containment due to potential stratification. RG&E does not have any safety-related electrical equipment in this area of containment. Therefore, the basis for the staff concern does not apply to the Ginna plant.

c) Since it is not conceivable that all the air in containment would be expelled after a LOCA or MSLB, it does not appear reasonable to determine margin in this manner. The criteria of IEEE-323-1974, together with the margin inherent in the analyses arriving at containment conditions, serve as ample assurance of the determination of conservative environmental conditions.

Further, the Ginna-specific post-accident pressure and temperature profiles are being evaluated by the NRC as part of the Systematic Evaluation Program, Topics VI-2.D and VI-3. Proper post-accident profiles will be available following conclusion of these SEP topics for Ginna."

3. In Section 3.5, the NRC states that RG&E provided information relative to maximum submergence in containment (7 feet), but not the elevation level. The elevation of the containment basement floor is 235'8", the maximum submerged elevation level is thus 242'8". This information will be added to Table 4 of RG&E's report.

4. In Section 3.5, it is stated that the licensee should provide an assessment of the failure modes associated with submergence of equipment, and ensure that subsequent failures of the equipment will not adversely affect the safety functions or mislead an operator. RG&E provides such information for each potentially affected item of equipment, as well as time of flooding if applicable, in the discussion of the individual equipment items.

5. In Section 3.7, "Aging", it is stated that, although a qualified life need not be established for all safety-related equipment, certain actions relative to materials evaluation, surveillance and maintenance, and replacement scheduling are required. RG&E addresses this in Reference 72, included in this response.

6. In Table C of Appendix D, a number of instruments appear which RG&E does not consider as being required for accident mitigation and/or safe shutdown. These instruments were not addressed in our October 31, 1980 submittal. For the reasons given below, the following instruments should be deleted from Table C:

a) Chemical and volume control flow - This instrument is normally used only for safe shutdown to provide inventory control. However, pressurizer level information could also determine if inventory control was not being maintained.

b) Diesel generator, emergency AC power, emergency DC power - The effects of failure of these systems, resulting in failure of the powered equipment, would be apparent. There is thus no need for monitoring of the status of the electrical systems themselves.

The Component Cooling Water flow and Service Water System indication were not addressed in the October 31 report, but will be added to the list of safety-related equipment. Since this equipment is in a mild environment, discussion of its environmental qualification requirements will be deferred.

Attachment 2: Response to March 18, 1981 TER-C5257-178, FRC report included with June 1, 1981 letter from NRC concerning "Environmental Qualification of Safety-Related Electrical Equipment"

1. In Section 4.1.1 of the TER, it is stated that it was unclear if the criteria used by RG&E for selection of systems to be evaluated were in accordance with the DOR Guidelines. This confusion arises from the NRC review of SEP Topic III-5.B, "Pipe Break Outside Containment". In the resolution of this topic, RG&E committed to make modifications to the steam heating lines, or provide leak detection in the vicinity of required safety-related electrical equipment, such that a harsh environment would be prevented from occurring. This modification, when complete, will negate the need to evaluate the effects of a harsh environment on the electrical equipment. Justification for the present arrangement, until modifications are made, are presented in the SEP topic review documentation.

2. In Section 4.1.1 of the TER, it is noted that RG&E did not attempt to qualify all equipment mentioned in the Emergency Procedures, and no criteria for selection was presented. The following is RG&E's rationale:

   RG&E has selected for qualification those items which are safety-related, and perform a necessary post-accident mitigation function. Other items specified in the Emergency Procedures provide additional flexibility to the operator in performing post-accident functions; they are not required, and no credit for the operation of these items is taken in the plant safety analyses.

   The potential for operator confusion, based on erroneous indications of non-qualified equipment, is accounted for in the Emergency Procedures. Specific requirements are detailed in the discussion of the particular instrumentation, and will be discussed later in this attachment.

3. In Section 4.1.1, it is stated that certain identified safety-related equipment was not included in the list of equipment to be qualified. Each item is discussed below:

   a) solenoids controlling air-operated containment isolation valves - Only those solenoids in a "harsh" environment were identified in our previous submittals. Other solenoids will be addressed in future submittals. Since these are located in a "mild" environment, discussion can be deferred at this time.

   b) solenoids controlling air-operated valves 4561 and 4562 - same as a) above.

c) motorized valve actuators for valves 313, 813, and 814 - As noted for solenoids controlling air-operated valves, RG&E did not generally address MVA's in a mild environment. This will be done in future submittals. Discussion of these items can be deferred in the interim, since they are located in a mild environment.

d) other motorized valve actuators for other containment isolation valves - same as c) above.

e) motorized valve actuators for 704 A, B - These MVA's will see a harsh radiation environment during post-LOCA sump recirculation. Their qualification is addressed in RG&E's Reference 53; however, they were not specifically identified in our October 31, 1980 submittal. These MVA's will be added to item 8E of Table 3 of that submittal.

4. In Section 4.1.1, FRC notes that certain electrical equipment was identified in the DITER, but not addressed in RG&E's October 31, 1980 submittal. The equipment, and reason for not including it, follows:

   a) junction boxes and terminal boards located outside containment - Most of these items are located in a "mild" environment. An evaluation of the effects of a steam line break in the intermediate building on such items is still under review. Since the Standby AFW System, used to mitigate a steamline break, is located outside the intermediate building, it is not expected that this would be a concern. However, the evaluation will be provided when completed.

   b) control stations - this equipment is located in a "mild" environment. Discussion can thus be deferred.

   c) Class 1E medium voltage switchgear - same as b) above.

   d) Class 1E motor control centers - same as b) above.

   e) inverters - same as b) above.

   f) battery chargers - same as b) above.

   g) hydrogen monitors - This equipment is being procured and installed as a "TMI modification". Qualification documentation will be provided in accordance with the NRC's Generic Letter 81-05 dated January 19, 1981.

   h) charcoal filter deluge valves - These valves are "additional protection", useful but not required for protection of the charcoal filters. Although it is expected that these valves would operate, no credit is taken for these valves in the plant safety analysis. Qualification is thus not required.

5. In 4.1.1, it is noted that certain other items should be added to the list of safety-related electrical equipment requiring environmental qualification. These are discussed below:

   a) steam dump to atmosphere valves. These valves are required only for eventual cold shutdown of the reactor, not for accident mitigation. As discussed in Section 3.2 of the NRC's "SEP Review of Safe Shutdown Systems for the R. E. Ginna Nuclear Power Plant," manual operation of the atmospheric dump valves is acceptable. There is thus no need to qualify this equipment.

   b) actuators for valves in the hot leg injection paths (878A, 878C). A discussion of these valves was inadvertently omitted. These should have been discussed together with item 8G of the October 31, 1980 submittal (878B, 878D). The 878A and 878C valves are locked in position (closed) with power removed. No credit is taken for hot leg Safety Injection flow in the Ginna Safety Analysis. These valves should thus be classified in NRC Category III, equipment that is exempt from qualification.

   c) I/P converters in the RHR heat exchanger discharge valves. As discussed in our October 31, 1980 submittal, these controllers have no post-accident function (these RHR system valves are normally open, and need only remain open; the I/P controller fails open). It is thus not considered that qualification is required.

   d) limit switches. RG&E reference 44, "Limit Switches," includes a January 16, 1979 letter from Leon D. White, Jr. to Boyce H. Grier. This letter provides our rationale that limit switches need not be environmentally qualified for Ginna. In the case of containment isolation valve limit switches for position indication in particular, the NRC specifically exempted them from the list of safety-related electrical equipment during the May 5, 1980 site visit to Ginna. Note that containment isolation valve position indication is not listed in Table C of Appendix D of the SER.

   e) connectors. In general, RG&E does not use connectors in safety-related equipment applications, where subject to a harsh environment.

6. In Section 4.1.2, FRC judges that RG&E has not provided acceptable justification for ignoring the "minimum of one hour" guideline. It is then stated that the NRC's rationale for this requirement is presented in Section 2.2.4 of this report. However, Section 2.2.4 merely reiterates, with no elaboration or discussion as to its basis, the statement that ". . . the [NRC] staff has indicated that time is the

most significant factor in terms of the margins required to provide an acceptable confidence level that a safety-related function will be completed. The 1-hour qualification requirement is based on the acceptance of a type test for a single unit and the spectrum of accidents (small and large breaks) bounded by a single test."

RG&E accounts for margin by specifying a conservatively calculated qualification profile, accounting for the potential occurrence of small and large breaks. Additional margin is not required.

Nevertheless, except for the reactor trip switchgear, RG&E did not specifically take credit for operating times of less than one hour in our October 31, 1980 submittal, other than for several specific pieces of equipment in the area of submergence. The guidelines of Section 2.2.2 are considered in the discussion of the specific items affected.

7. In Section 4.1.2, High Energy Line Breaks, it is noted that additional equipment might be subject to a harsh environment due to a HELB. This was discussed in item 1 of Attachment 2.

8. In Section 4.1.2, Nuclear Radiation Dose, reference is made to Appendix H of the TER, entitled, "Effects of Nuclear Radiation Dose Rate on Cable Performance During a LOCA." This appendix makes reference to the fact that greater degradation may occur at higher dose rates than were used in qualification testing. It should be noted that this "finding" has not been previously addressed in NRC criteria, including the DOR Guidelines. Furthermore, no specific references in the technical literature are cited, which makes it impossible for the RG&E technical staff to review the validity or applicability of the data. In general, RG&E testing and analysis related to dose rate in insulating materials is consistent with IEEE Std. 278, "USIA and IEEE Guide for Classifying Electrical Insulating Materials Exposed to Neutron and Gamma Radiation." Since failure of electrical cables to perform their required safety functions due to higher dose rates is still the subject of research, and clearly beyond present-day regulatory criteria, we consider that it is inappropriate for discussion in this review. RG&E considers that the radiation qualification requirements specified for Ginna are very severe, and that equipment shown operable after exposure to this dose would operate in a post-accident environment. RG&E has reviewed several other plant TER's not prepared by FRC, and did not find this to be an area of concern. These other reviews appear to be more consistent with the DOR Guidelines. A consistent approach should be used for Ginna.

9. In Section 4.1.3, Aging and Qualified Life, it is stated that the ". . . DOR Guidelines require that the licensee

establish (numerically) the qualified life of all equipment items containing components susceptible to degradation produced by heat and nuclear radiation . . ." This contradicts the statement in 3.7 of the SER, which states that "The DOR Guidelines, section 7, does not require a qualified life to be established for all safety-related electrical equipment . . ." This contradiction should be corrected.

RG&E, in Reference 72, provides information relative to a maintenance/ surveillance program implemented at Ginna to check for, among other things, potential failures due to aging. This program should satisfy Section 7 of the DOR Guidelines.

10. In Section 4.2.1, the Crouse-Hinds electrical penetrations were classified as NRC Category I.a. No additional information is thus required.

11. In Section 4.3.1, equipment is judged to meet all applicable requirements of the DOR Guidelines except qualified life. As noted earlier, Reference 72 provides information about RG&E's surveillance/ maintenance program to meet Section 7.0 of the DOR Guidelines. Specific equipment items are addressed below:

    a) 4.3.1.1 - Steam Line Pressure Transmitters - these are to be replaced by June 30, 1982.

    b) 4.3.1.2 - BAST Level Transmitters - see reference 72 for "aging" program.

    c) 4.3.1.3 - RWST Level Transmitter - to be replaced by June 30, 1982.

    d) 4.3.1.4 - Various motorized valve actuators (Note: the MVA's for MOV's 704A and 704B should be added to this list). It is not clear why the TER requests that RG&E determine a "qualified life," since the DOR Guidelines specifically state that a qualified life need not be determined for all electrical equipment. Aging degradation was assessed in the Limitorque test report B0003 (200 hours at 165°F). Finally these MVA's will be considered in the maintenance/surveillance program noted in Reference 72.

12. In Section 4.3.3.1, it is stated that "a thorough review of the AFW system(s) at this plant (see Item E.1 in Appendix E) has led to the conclusion that the present configuration, with remote-manual initiation of the standby AFW system, is not satisfactory." RG&E must dispute this conclusion. In fact, the NRC has reviewed and approved the present configuration of the AFW system. In the NRC's "SEP Review of Safe Shutdown Systems for the R. E. Ginna Nuclear Power Plant," it was determined that:

[illegible]

[illegible]

> "The AFS and SAFS conform to GDC 19, "Control Room," GDC 44, "Cooling Water," GDC 45, "Inspection of Cooling Water Systems," GDC 46, "Testing of Cooling Water Systems," and Regulatory Guide 1.62, "Manual Initiation of Protective Actions." GDC 5, "Sharing of Structures, Systems and Components," is not applicable."

Further, the NRC reviewed the RG&E arrangement during the review of SEP Topic III-5.B, "Pipe Break Outside Containment." In Table 1 of the assessment, it is noted that the Intermediate Building elevation 253', where the pumps are located, is adequately protected from pipe break effects because ". . . there exists another system (SAFS) to supply auxiliary feed to the steam generators . . ."

Finally, the NRC has reviewed the RG&E accident analyses during the "Design Basis Event" topic assessments, transmitted by letter of June 29, 1981. The limiting case of a Feedwater Line Break was evaluated as SEP Topic XV-6. In that assessment, acceptability of the Ginna system was based, in part, on approval of a feedline break analysis which took no credit for the Auxiliary Feedwater System, but only remote-manual actuation of the Standby Auxiliary Feedwater System in 10 minutes.

Based on these numerous reviews and approvals by the NRC of our present AFW system configuration, the TER conclusion that our present system is unacceptable should be withdrawn.

13. Item 4.3.3.2 addresses MVA's for motor operated valves 852A and 852B. Contrary to the FRC conclusion, a qualified life does not need to be established, in accordance with the DOR Guidelines. However, Reference 72 concerning maintenance/ surveillance does apply to these items.

14. Item 4.3.3.3 addresses Kerite power cable outside containment. The FRC "Conclusion" notes that a conservative qualified life should be determined. Yet the FRC "Evaluation" states that thermal aging does not simulate all aging conditions to which the cable would be subjected, and thus disagrees with the RG&E establishment of a 93.3 year life at 140°F based on Arrhenius data. Since the DOR Guidelines address only thermal and radiation aging, and since both of these concerns have been addressed for this cable, no additional information should be required for resolution of this item.

    Since this cable is safety-related electrical equipment, Reference 72 concering maintenance/surveillance applies.

15. Item 4.3.3.4 addresses various MVA's in the auxiliary building. Inclusion of these valves in the maintenance/ surveillance program described in Reference 72 should relieve the concerns relative to aging.

[illegible]

[illegible]

16. Item 4.3.3.5 addresses the MVA's actuating MOV's 825A and 825B. Inclusion of these valves in the maintenance/surveillance program described in Reference 72 should relieve the concerns relative to aging.

17. Items 4.4.1, 4.4.2, 4.4.3, and 4.4.4 address items exempt from qualification. RG&E notes that the MVA's for MOV's 878A and 878C, located in the hot leg safety injection lines, should be included in 4.4.2, as discussed previously.

18. Item 4.5.2.1 addresses containment pressure transmitters located in the Auxiliary Building. All five of these transmitters are scheduled to be replaced by June 30, 1982.

19. Item 4.5.2.2 addresses the RWST level switch. When required to perform its safety function, this item does not experience a "harsh" environment. Environmental qualification testing is thus not required. FRC has also concluded that qualification is not required. However, the maintenance/surveillance program in Reference 72 will apply to this item.

    Although RG&E had originally thought to replace this item by June 30, 1982, it does not appear necessary because of its location in a "mild" environment.

20. Item 4.5.2.3 addresses Kerite cable located inside containment. The FRC evaluation of this cable is in error. Contrary to the statement that this cable was "laid in the bottom of the box," these cables were actually the "main specimens" being tested together with the Raychem sleeves. The Kerite cable should be evaluated, accounting for this correction. RG&E considers that this cable is fully qualified to meet the DOR Guidelines, and should be classified as I.a.

    As stated earlier, RG&E believes that the dose rate concern expressed in Appendix H goes beyond the scope of the DOR Guidelines, and is inappropriate for review in this document.

    RG&E also notes that the cable testing performed in FIRL Report F-C4020-1 was at peak conditions greatly in excess of the Ginna post-accident environmental conditions. Testing performed at 340°F/113 psig should not be used to make judgments for cables installed at Ginna, whose peak environmental conditons are 286°F/60 psig. RG&E cannot make any definitive arguments relative to F-C4020-1, since it was not provided to us.

21. Item 4.5.2.4 addresses the Reactor Containment Fan Coolers Located Within Containment. Several concerns were addressed, which are discussed below:

    a) FRC notes that RG&E did not completely address LOCA parameter effects on the lubrication system, insulation, and other components of the motor. Actually, Reference 2.18, WCAP-7410-L, provides this information.

[illegible]

[illegible]

RG&E has added a small amount of ARCO Rotanium lubricant to the original Westinghouse lubricant. Although the Rotanium lubricant does not have any specific testing for post-LOCA radiation, the small amount added (estimated by review of plant records of the fan cooler motor maintenance history at no more than 10%) is not expected to adversely affect the performance of the lubrication system. As a result of this review, however, RG&E will no longer use the Rotanium lubricant, but will use a Chevron product which is equivalent to the original Westinghouse lubricant, with a radiation resistance of 5×10^8 rads. Qualification information regarding this lubricant is being obtained. It will be submitted following receipt by RG&E. RG&E, therefore, considers the lubrication system acceptable for post-accident operation.

FRC also noted that splice material is still in question. This will be addressed in 21.c) below. RG&E has confirmed that Okonex tape and Elastimould No. 86 was not used for splice material in the fan cooler motors.

FRC also notes the confusion created by RG&E references 2.67 and 2.20. The fan cooler motor insulation is "Thermalastic Epoxy." This is consistent with the qualification documentation provided by WCAP-9003 and WCAP-7410-6. Reference 2.67 will be corrected when the Ginna Environmental Qualification Report is resubmitted.

b) A review was made of plant maintenance records to determine if the fan cooler motor bearings, or other motor components had had any noted failures or degradation. No such evidence was found. RG&E will include the fan cooler motor bearings in the maintenance/surveillance program described in Reference 72.

c) Although Westinghouse drawings 206C391, referenced via transmittal of supplemental information for Reference 2.64, is not an "as-built" for Ginna, RG&E has every reason to expect this to be the proper splice material. Reference 2.64 indicates that the splice met all required test parameters, including pressure, chemical spray, and radiation. RG&E believes that it was an oversight in not specifying accident temperatures along with the accident pressures of 60 psig for 2 hours and 20 psig for 18 hours; however, it is expected that at least 286°F was used together with 60 psig, and 219°F with 20 psig, since this corresponds well to the general Westinghouse qualification profile in use at that time (see WCAP-7410-L), and because the specified test conditions are identical in every respect to those used in WCAP-7410-L, except for the omission of these temperatures.

Based on the testing performed as shown in Reference 2.64, RG&E concludes that the splice is acceptable for use inside containment. If during our Spring 1982 refueling shutdown it is found that the splice shown in Westinghouse drawing 206C391 is not the same as that used at Ginna, the lead-to-cable splice will be replaced with a fully qualified Raychem sleeve.

It should be noted that this information pertains only to the lead-to-cable splice. The motor-to-lead splice was tested together with the motor in WCAP-9003, and is thus qualified by virtue of the acceptability of the motor qualification test.

22. Item 4.5.2.5 addresses Raychem cable splices located inside containment. The only FRC finding concerned the omission of a qualified life. In fact, RG&E provided a qualified life of 40 years at 91°C, based on reference 2.63. It is not clear what the concern is.

As for the symbiotic classification of the cable and splice sleeves - we consider that this to be inappropriate. The cable and sleeves are purchased separately, and reviewed separately. Qualification documentation can be and should be addressed separately. RG&E notes again that the beta dose rate concerns for the cable goes beyond the DOR Guidelines, and should thus not be discussed in the context of this review, but should be the subject of a separate independent review.

23. Item 4.5.2.6 addresses solenoid valves in the turbine building. RG&E has committed to replace this equipment by June 30, 1982.

24. Item 4.5.2.7 addresses Coleman cable located inside containment. The major concern expressed in the TER is that the radiation dose to which this cable was exposed during radiation testing could not be quantified. Although this observation may be correct, it is apparent that a lower limit on the amount of radiation received can be easily estimated.

The radiation source during the test consisted of a plaque array of Cobalt-60 elements approximately 60 inches by 72 inches wide. The only significant attenuation of the gamma flux was due to a 1/2" steel plate. For 1.33 MeV gamma, the attenuation factor for 1/2" steel is conservatively (collimated beam without buildup) calculated as about 0.60. The actually measured integrated dose at the back of the crate top was 6.93×10^7 rads. Thus, a conservative estimate of the lower bound of the cable radiation dose received is 4.1×10^7 rads. This is appreciably higher than the 2×10^7 rads required by the DOR Guidelines. The DOR Guidelines are thus met.

As noted earlier, RG&E believes that the beta dose rate concerns of Appendix H go beyond the requirements of the DOR Guidelines, and are inappropriate for review in this document.

Based on the above information, RG&E considers that the DOR Guidelines are met for this cable.

25. Item 4.5.2.8 and 4.5.2.9 address electrical cables located outside containment. A preliminary evaluation of the expected post-accident dose has been made, based on the information provided in RG&E October 31, 1980 submittal, Accident Reference [TMI-3]. It is expected that the integrated dose three feet from the recirculation piping would be less than 3×10^5 rads. The cables in question are further from the source.

    Information presented in draft EPRI Report 1707-3, "Radiation Effects on Organic Materials in Nuclear Plants," cites a radiation threshold for PVC (the materials of the cables in question) of 5×10^5 rads. It is thus not expected that any damage potential exists.

    RG&E has not yet completed confirmation of this issue. When completed and verified, proper documentation will be made available for NRC staff review.

26. Item 4.5.2.10 addresses solenoid valves for the main steam isolation valves. RG&E has committed to replace these valves by June 30, 1982.

27. Item 4.5.2.11 addresses solenoid valves for the containment recirculation system dampers. RG&E has committed to replace these solenoid valves by June 30, 1982.

28. Item 4.6.1 addresses Medium Voltage switchgear, the reactor coolant pump breakers and reactor trip breakers. These will be discussed separately.

    a) Reactor Coolant Pump Trip Breakers - Appendix E.2 states that the RCP Motors should be maintained operable during a HELB in the turbine building. Actually, the RCS design is such that natural circulation cooling will remove sufficient decay heat. The reactor coolant pumps switchgear provides only one safety-related function - to open and remove power to the pumps in the event of a small cold leg break LOCA. For this particular accident, there will be no adverse environment in the turbine building. Thus, environmental qualification documentation is not required.

    b) Reactor Trip Breakers - As correctly noted by FRC in the DITER, this equipment fails safe (open) upon loss of actuating power. This was also noted by NRC in their review of SEP Topic III-5.B, "Pipe Break Outside Containment," transmitted to RG&E by letter dated

June 24, 1980 (see Table 1 of that report). It is thus not anticipated that a failure to trip when required will be a concern.

RG&E does not, however, have documentation to definitively demonstrate the functional capability of these trip breakers under HELB conditions. RG&E is thus planning to make an equipment evaluation to determine the potential degradation of this switchgear under such conditions. RG&E does not agree that documentation is required to "... demonstrate equipment functional capability for a time period of one hour plus the length of time it is required to operate." The occurrences of an HELB requiring reactor trip would be detectable by use of Reactor Protection System instrumentation. If automatic tripping did not occur, an operator could trip the electrical bus providing power to the control rods, either from the control room or locally in the turbine building. Upon this action, the reactor will trip, regardless of conditional trip breakers. It is expected that this action could be taken within 5 minutes.

Since the only function of these breakers is to open, and they are designed to fail-open in loss of power, RG&E does not consider the one-hour requirement to be at all appropriate for this situation.

29. Item 4.6.2 addresses the auxiliary feedwater pumps. As noted in item E.1 of Appendix E, FRC had concluded that the present configuration of the auxiliary feedwater system is not satisfactory. As noted earlier in this letter (#12), the NRC has consistently reviewed and approved this system.

RG&E also provided an evaluation to determine if the Class 1E AFW circuits could be degraded by an HELB in the intermediate building, thus negating the operability of the Standby AFW system. This was provided in a letter from John E. Maier, RG&E, to Dennis M. Crutchfield, NRC, dated December 15, 1980. This evaluation concluded that no degradation would occur.

30. Item 4.6.3 addresses the Westinghouse electrical penetration. New reference 2.61, included with this report, provides detailed comprehensive qualification data for post-accident qualification testing of the penetrations, as well as a detailed aging evaluation. Please note that even though PEN-TR-81-45 specifies only a 268°F peak temperature, with a radiation dose of 1.5 x 10^8 rads, the actual testing was performed at 340°F, with an integrated dose of 2 x 10^8 rads.

The question of short-circuit currents damaging these penetrations was discussed during a telephone conversation between RG&E personnel and Messrs. DiBenedetto and Lee of the NRC staff. At that time, we stated that only LVDT and

television camera cables use this penetration. These are low voltage and current circuits, which could not damage the penetration even in the event of a short-circuit fault. An assessment of similarly-used penetrations (used as parts of instrumentation loops), is made in Section 3.4 of the "SEP Technical Evaluation, Topic VIII-4, Electrical Penetrations of Reactor Containment," transmitted by letter from Dennis M. Crutchfield to John E. Maier dated March 30, 1981. It was concluded that no mechanical failures would be expected for these penetrations.

31. Item 4.6.4 addresses terminal blocks. These terminal blocks will be elevated, together with the pressurizer pressure and pressurizer level instrumentation, during the Spring 1982 refueling shutdown. Submergence will thus no longer be of concern.

    To reduce the possibility of conductance between the terminals during an HELB, the terminal blocks will be periodically cleaned. At that time the blocks will be inspected for possible aging degradation. The steam test (FIRL #F-C4911-1) indicates that satisfactory performance may be expected for protected terminal blocks. The terminal blocks at Ginna are protected from direct spray impingement by their location in the pressurizer instrumentation cabinets. Also, resilient washers will be installed under the blocks to preclude any cracking of the block while mounting or connecting to the block.

    Additional information is still being developed relative to qualification of these terminal blocks. If full documentation cannot be provided, either a type test will be conducted before June 30, 1982, or the terminal blocks will be replaced with fully qualified Raychem sleeves.

32. Item 4.6.5 and 4.6.6 address pressurizer pressure and pressurizer level transmitters located inside containment. As noted previously, RG&E has committed to replace and elevate these instruments by June 30, 1982. Emergency procedures at Ginna specify the operator is not to terminate SI flow unless all of the following are in evidence: Safety Injection flow is zero, pressurizer pressure is increasing and greater than 2000 psig, pressurizer level is greater than 50%, RCS subcooling is greater than 50°F, and steam generator water level narrow range is greater than 25% in one generator. RG&E considers this to be a very conservative set of SI termination criteria.

33. Item 4.6.7 addresses steam line flow transmitters inside containment. In RG&E's October 31, 1980 submittal, we stated that we planned to replace these transmitters by June 30, 1982. However, it is not clear that replacement is required. As stated by RG&E previously, these transmitters are not required to perform a safety function during an

HELB, since the non-return check valves will isolate the intact steam generator from the break. The check valves are the primary isolation means, since they operate more quickly (1-2 seconds) than the 5 second MSIV's. For a large break inside containment, the containment pressure transmitters will signal the MSIV's to close. A steam line break analysis presently being performed by the NRC for SEP Topics VI-2.D and VI-3 will be completed soon. RG&E will at that time be able to define if the containment pressure transmitters will serve as a suitable backup for the steam line flow transmitters, in the event of a steam line break.

34. Item 4.6.8 addresses the containment pressure transmitters located in the intermediate building. The five transmitters being replaced are those located in the auxiliary building, where the potential for higher radiation levels as a result of post-LOCA sump recirculation dictated their replacement. The intermediate building will not experience a harsh environment for an HELB inside containment, which is the only time these transmitters are required to operate. Although they would see a harsh environment as the result of an HELB in the intermediate building, they have no function to perform during such an event. It is thus considered that these two transmitters need only function in a "mild" environment.

35. Item 4.6.9 addresses the purge and depressurization valve solenoid valves. As previously stated, RG&E has committed to replace these solenoid valves by June 30, 1982.

36. Item 4.6.10 addresses the RTD's. As previously noted, RG&E has committed to replace the RTD's required as input to the subcooling meter, as a "TMI-related" modification. RG&E does not consider the RTD's as necessary for the mitigation of a LOCA. Although FRC refers to E.5 of Appendix E, no discussion is provided there, or anywhere else throughout Appendix E.

RG&E is also aware that the environmental service conditions for the RTD's include both RCS conditions and containment conditions. This will be accounted for in the replacement units.

37. Item 4.6.11 addresses the steam generator level transmitters. As previously committed, these instruments will be replaced by June 30, 1982.

38. Item 4.6.12 addresses the solenoid valves for the pressurizer PORV's. The original specification provided as Reference 2.48 provided information relative to the environmental qualfication of these valves; the test report is not at this time available to RG&E. This will be obtained prior to June 30, 1982, to provide confirmation of the environmental qualification of these solenoid valves. The September 24, 1980 information is withdrawn as a reference for our submittal; it was incorrectly submitted.

39. Most of the items reviewed in Section 4.7, which are NRC Category VI, Equipment for Which Qualification is Deferred, will be addressed at a later time following review by the NRC. However, the items noted in 40 below should be addressed at this time.

40. A number of items listed as NRC Category VI, "Equipment for Which Qualification is Deferred," should also be included in this review. The specific items and reasons are given below:

    a) Items 4.7.3 and 4.7.18 address portions of the Hydrogen Recombiner unit. This is a "TMI-related" item; however, deferral of review is not necessary, since the unit is already installed. The FRC review directed that an overall life of the motor should be established, and that a review of the motor's bearing, splices, and lubrication system should be made. Based on the intermittent use of the recombiner, a 40-year life would be expected. However, period maintenance and surveillance, per Reference 72, will account for potential failures, including age-related causes. As for the lead-to-cable splice, Westinghouse has noted that the information in Reference 2.64 is applicable. At the next scheduled refueling shutdown of Spring 1982, RG&E will verify that the proper splice material was used. If not, the splices will be replaced with qualified Raychem sleeves.

       FRC also noted additional concerns in item 4.7.18, relative to other recombiner system components. Additional information from Westinghouse states that the ignition lead and thermocouple leads are completely housed in a pressure-tight system with the connections field-brazed. There are no blower damper control solenoids for the RG&E recombiner; the dampers are manually set. Splices have been addressed already. There are no terminal blocks on the RG&E recombiner.

       The only items not fully qualified are the Barton pressure switches. Qualification data will be established for these switches, or qualified replacements will be provided by June 30, 1982.

    b) In item 4.7.7, various Westinghouse pumps motors are addressed. The Containment Spray, RHR, and Safety Injection pump motors could see a high radiation environment during the mitigation of a LOCA. This is a "harsh" environment; qualification of these pump motors should be reviewed for this parameter.

       As properly pointed out by FRC, the Safety Injection pump motor is wound with Westinghouse Thermalastic Epoxy, and is qualified for a radiation environment by

WCAP-8754. The effects of Mobilux lubricant on the bearing system will be evaluated, among other things, by inclusion in the maintenance/surveillance program of Reference 72.

The other motors use PMR Class B insulation. As stated in Reference 2.69, these motors are qualified for 1 x 10^7 rads and an operating-life of 20 years. Since these motors see only intermittent service, an operational capability of 40 years is provided. These pump motors are also included in the maintenance/ surveillance program of Reference 72. We are not able to make this entire proprietary report available at this time. Salient facts have been transmitted in Reference 2.69; the report itself is available for audit at RG&E.

c) Item 4.7.12 addresses cooling fan motors located in the auxiliary building. These motors are subject to a radiation environment during post-LOCA sump recirculation, and should thus be reviewed for qualification in the "harsh" environment. As stated in b) above, Reference 2.69 is applicable to these motors.

Maintenance/Surveillance Program to Detect Failures (Including Age-Related) of Electrical Equipment

A form similar to the attached form is to be used to evaluate causes of failure of all safety-related electrical equipment, both in a "harsh" and in a "mild" environment. This form has not yet been formally approved for Ginna. Implementation wil be effected by September 30, 1981. All failures during operation, testing, or calibration will be evaluated for failure mechanism. If aging of a component or material type is noted to be a potential concern, an evaluation of other components using this component or material will also be made.

This form is an update of the Ginna administrative measures described in Reference 2.47 to detect aging effects in Ginna electrical components.

I&C/ELECTRICAL EQUIPMENT FAILURE (SAFETY RELATED)

EQUIPMENT ______________ USAGE ______________ S/N ______________

FAILURE [ ] YES [ ] NO   OUT OF CALIBRATION [ ] YES [ ] NO   OTHER ______________

DIRECTION OF FAILURE ______________

A-25.1 WRITTEN [ ] YES [ ] NO   A-52.4 WRITTEN [ ] YES [ ] NO

IMMEDIATE CORRECTIVE ACTION [ ] REPAIR [ ] REPLACE   OTHER ______________

I&C SUPERVISOR NOTIFIED ______________ DATE ______________

SEND COPY TO I&C SUPERVISOR

FAILURE INVESTIGATION

IDENTIFY FAILED PARTS OF COMPONENT ______________

______________ [ ] NONE FOUND

I&C SUPERVISOR ASSESSMENT

[ ] PROBABLE IMPROPER OPERATION OR OPERATION FAILURE

____

[ ] PROBABLE AGE RELATED FAILURE
IDENTIFY OTHER COMPONENTS TO BE CHECKED ______________

____

[ ] NORMAL DRIFT OR FAILURE

[ ] OTHER (EXPLAIN) ______________

I&C SUPERISOR ______________ CAR/PAR NEEDED [ ] YES [ ] NO # ______

PORC REVIEW (IF AGE RELATED ONLY) ______________