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MFN-102-80

May 23, 1980

Mr. Harold Denton
Director of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Mr. Denton:

SUBJECT: ANTICIPATED TRANSIENTS WITHOUT SCRAM (ATWS) - GENERAL ELECTRIC COMMENTS ON NUREG 0460 (VOLUME 4)

This letter responds to a request in the Federal Register for comment on NUREG-0460 (Volume 4), Anticipated Transients Without Scram for Light Water Reactors.

We are deeply concerned that the major changes in direction proposed in Volume 4 (vis-a-vis Volume 3) of NUREG-0460 were considered without technical justification or adequate interaction with industry. Volume 4 has escalated the ATWS mitigation requirements for the BWR without explanation or a cost impact evaluation. The NRC staff issued acceptance criteria in Volume 3 which, if met, would satisfy the concerns of the NRC for ATWS mitigation. The evaluation by General Electric which demonstrated compliance with the NRC criteria has been seemingly ignored with preference now shown by the NRC for requiring additional mitigation equipment.

The following summarizes the actions regarding ATWS which provide the basis for resolution of the issue for the BWR:

1. The NRC staff developed a position on ATWS in Volume 3 of NUREG-0460 which stated that Alternate 3 would be satisfactory for operating plants and plants under construction if the staff questions could be answered.

2. In February 1979 the staff issued 52 pages of questions and requested that General Electric provide a major ATWS assessment. General Electric responded to all issues and submitted an in-depth analysis in December 1979. General Electric considers that this assessment satisfies the criteria NRC specified.

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3. The NRC staff did not review the General Electric submittal in detail. The staff raised additional questions regarding calculated limit cycle oscillations for a specific BWR product line. These were discussed in a meeting with you and several BWR utility executives on February 7, 1980. General Electric responded to the NRC concern by demonstrating that if the limit cycle oscillation actually occurred, it was not a significant safety consideration; in addition, several alternatives were identified which eliminated even the calculated limit cycle oscillation. Since the NRC staff had agreed to resolve ATWS if this was no longer an issue, there is no basis to escalate BWR requirements from those identified in Volume 3.

Based upon the above, we strongly recommend that the NRC adopt the recommendations of Volume 3 if mitigation is required for the operating an. under construction BWR's. There is no technical justification for escalating to the recommendations of Volume 4. Indeed, the ACRS in their letter to the NRC on April 16, 1980 supported the recommendations of Volume 3 as all that is necessary to resolve ATWS.

We are also concerned that Volume 4 does not adequately reflect consideration of BWR unique capabilities to accommodate the consequences of ATWS. The BWR ATWS assessment was responsive to the NRC staff request and was acknowledged by the staff as providing "excellent analysis information". Yet, Volume 4 continues to hypothesize BWR problems which are contrary to what has been reported in GE assessments. We are concerned that this may indicate a lack of understanding of the BWR capability and a potential overreaction to perceived problems. Clearly the goal should be to establish an acceptable level of protection for all reactor types. We do not believe this has been accomplished by the prescriptive recommendations in Volume 4.

In addition, it is noted that the implementation schedule proposed in Volume 4 is unrealistic. It is unlikely that such a schedule could be achieved even with a highly intensive program to produce plant specific designs and hardware. To avoid adverse effects, the schedule must allow for a disciplined engineering approach for establishing plant modifications and insuring that system interaction effects are properly evaluated. It is strongly recommended that industry inputs be considered when establishing the implementation schedule for any required ATWS modification.

General Electric has several additional specific comments as a result of our review of NUREG-0460 (Volume 4). They are provided in the attachment to this letter.

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In summary, we do not believe that NUREG-0460 (Volume 4) adequately reflects the technical assessments provided your staff by General Electric for the BWR. It establishes unnecessary requirements and requires unachievable schedules, while failing to provide a balanced resolution for the ATWS issue. We believe that the ACRS in their review of the NUREG-0460 concurs with the General Electric observations. We recommend that you not accept the proposals in Volume 4, but rather follow the recommendations of Volume 3 if mitigation is deemed necessary for the operating and under construction BWR's.

Please contact me or R. H. Buchholz [(408) 925-5722] if you have questions or require clarification of this letter.

Very truly yours,

RHBuchholz for

Glenn G. Sherwood, Manager
Safety and Licensing Operation

GGS:mm/1685-87

Attachment

cc: D. G. Eisenhut
R. J. Mattson
D. F. Ross
L. S. Gifford (Washington Liaison Office)
A. C. Thadani

SPECIFIC COMMENTS ON NUREG 0460, VOLUME 4

1. Page 8 (Item 1 of Paragraph 1.3.2) - General Electric has not observed power/flow oscillations in any operating plant. Even under special high power/low flow test conditions aimed at studying the threshold of such behavior, large stability margins have been measured. Only in one such test - after removal of all normal inlet orificing in a test assembly - a local hydraulic oscillation condition was observed. General Electric models account for this phenomenon and compare well against available data. The General Electric evaluations in response to Volume 3 and the February 15 NRC letter were not based upon an exact predictive capability, but rather demonstrated that no extreme sensitivities exist. It was also demonstrated that fuel/cladding performance was acceptable for all ranges of potential cycles.

2. Page 9 (Items 2, 3 and 6) - While details of design and equipment capability may need additional review (some are plant unique or BOP) as Alternate 3 is implemented, the bulk of the information has been provided. Based upon the GE review it is concluded that no significant problems will be encountered.

3. Page 9 (Item 4) - Loads have been shown to be within current design bases using techniques agreed upon by the Staff.

4. Page 9 (Item 5) and Page 22 (Paragraph 2.3.1.5) - This does not apply to the BWR since Mark I and II containments are always isolated; Mark III containments already have early, automatic containment isolation.

5. Page 35 (central paragraph) - The BWR analysis does provide solid, full shutdown evaluation.

6. Page 36 (BWR Short-term Behavior) - This asks for comparison of the REDY and ODYN codes; however, that is what was given for several cases (including the non-anticipated T-G trip with bypass failure) in the ATWS submittals provided by GE. Since the basic first-principle simulations of both codes (REDY and ODYN) are identical for integral effects, the fact that they closely match each other (when the detailed timing of scram is not involved) is quite understandable.

7. Page 36 (BWR Long-Term Behavior) - The long-term behavior is quasi-steady state. NEDE-24222 documented that REDY power prediction capability has been successfully cross-checked with a 3D core simulation code. The long-term flow conditions compare well to the BWR design tools. The behavior of water level during ATWS events has been evaluated using LOCA analysis tools. The results of this evaluation shows reasonable comparison and excellent core cooling for all cases. NEDE-24222 results are considered accurate predictions of BWR behavior.

8. Page 43 (Middle Paragraph) - Failure of all fuel rods is not a "reasonable" assumptio It is a totally bounding, worst-case assumption. In mcst uations we believe few, if any, perforation failures would occur

9. Page 50 (Item 3) - General Electric does not agree that this is a "Pro" compared t equiring only Alternate 3A.

10. Page 50 (Item 4) - There is no significant difference for the BWR; little operator action is involved, even with 3A.

11. Page 50 (Item 5) - General Electric does not agree that this is a "Pro" compared to requiring only Alternate 3A.

12. Page 52 (Item 2) - This has been shown for Alternate 3 for BWR/4's when enriched boron is used.

13. Page 52 (Item 3) - These variables are already within limits with Alternate 3.

14. Page 60 (Bottom Paragraph) - The Dresden cleanup cannot be called typical (a partial, gravity-flow injection occuring during shutdown) as implied here. Besides, an unusually large amount of unused storage tank capacity was available at the time which facilitated cleanup operations.

15. Page 62 (Table 3) - Since the evaluations of Alternate 3 demonstrated compliance with all the NRC safety criteria, General Electric considers there is no increase in value from "3A" to "4A" as shown for the BWR.

16. Page 63 (Item 4) - The General Electric ATWS assessments demonstrate the acceptability of Alternate 3A for the BWR in a comprehensive manner. The escalation to Alternate 4A will require even greater resources.

17. Page 65 - Orders to make such modifications are unjustified.

18. Appendix A (Paragraph 4.1.8) - The staff's hypothesis of excessive depressurization with the reactor at significant power is not correct; the addition of subcooled water will simply reduce flow through the S/R valves without dropping pressure below the lowest S/R closure point.

19. Appendix A (Paragraph 4.1.9) - Not even the hottest channel "voids out" and then refloods for the postulated dryout/rewet sequence. If rewetting occurs at the low power condition, boiling transition will not reoccur. It is not like reflooding after a LOCA; the channel is always full of two-phase coolant.

20. Appendix A (Paragraph 4.3.1) - It is indicated that large SRV vibratory loads occur simultaneously with the ATWS (RPV) pressure loading. This is not true. SRV's lift at their pressure setpoints

which are typically 1060 to 1120 psi. Within 1 second, the oscillatory SRV loads are finished and the vessel is still below ~1200 psi (within upset limits). The same comment applies to Section 4.3.2 for the BWR 3 plants.

21. Appendix A (Section 4.4.1) - It is indicated that SRV loads are more severe during ATWS compared with normal operating transients because reactor pressure, pool temperature and containment pressure are higher. This is not true. First, reactor pressure is the same for ATWS as for normal transients because the driving force is the RPV pressure at the time the valve lifts, i.e. the setpoint. In fact the RPV pressure expected during ATWS is lower than that typically used for design because the design basis includes margin to account for pressure switch failure, setpoint drift, and maximum rated ASME flow. Pool temperature is no higher for ATWS than it is for power isolation events or small break accidents, which are already part of the design basis. In addition, extensive testing of full-scale quenchers (NEDE-21078) has shown that suppression pool temperature has a small effect on load magnitude and frequency. The effect of pool temperature has been included in the predicted ATWS SRV loads. Wetwell pressure during ATWS events is only slightly higher than during normal operation. The resultant increase in SRV load and frequency is bounded by design values.

22. Appendix A (Section 4.4) - The NRC states that the SRV loads were predicted by an analytical model not available to them. The SRV loads in fact were predicted by the same empirical model the NRC reviewed and approved in GESSAR and the DFFR (approved by NUREG-0487). The magnitudes were multiplied by a factor of ~1/2 to account for the Caorso data. The requests in Items (1) and (2) were already addressed in a meeting during September 1979 with the NRC in Denver, Colorado.

23. Appendix A (Section 4.4.2) - The NRC maintains that the quencher limit must be maintained at 200°F. This ignores test data which in several cases goes above 212°F and in one run exceeded 225°F. New data presented in NEDE-24222 shows stable quencher performance up to and including bulk boiling conditions (Appendix A.1). During ATWS events, the subcooling will never drop below 40°F due to pressurization of the containment as the pool temperature rises. Condensation stability at this degree of subcooling (40°F) has been amply demonstrated by subscale and full scale quencher tests.

24. Appendix A (Section 4.4.2) - The NRC requests confirmation of the local to bulk ΔT. This verification was shown to the NRC in a September 1979 meeting and appears in Appendix A.2 of NEDE-24222. The values used for the ATWS assessment are adequately supported by an analytical model, as well as in-plant test data.

25. Appendix E (Figure E-1 and Table E-1)

    a. No mention or credit is given for the BWR ARI feature.

b. BWR scenario "ATWS-U" (loss of high pressure makeup) does not produce a "likely core melt". From the NRC's viewpoint it may be an unanalyzed case, but recent work has shown that the BWR regulates itself to decay heat, and that only a small amount of inventory supply is needed (and is readily available).

c. BWR scenario "ATWS-C2" (loss of boron injection) does not lead "obviously" to core melt. If loss of 1 of 2 pumps (or simply slower initiation) is meant, high containment temperatures may result (depending on the event) but not core melt.

d. BWR scenario "ATWS-P" (stuck open S/R valve with pool cooling) is not a "high containment temperature" problem. A small incremental increase may occur, but it is clearly out of the range of any significant containment concern.

e. BWR scenario "ATWS-R" (loss or delay of pool cooling) does not produce a significant pool temperature increase if the delay is extended or only 1 of 2 heat exchangers function.

f. Table E.1 for present designs is very misleading. All BWR's already have (or are implementing) RPT.

g. Table E.1 (Note 2) - We do not agree. Note 3 - No reason to state makeup water depleted. Note 4 - Frequency of isolation events is less than 2/RY even before the setpoint changes. Note 5 - Not justifiable in light of NEDE-24222 report.

NOTE: These comments obviously change the BWR probability tables significantly.