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MFN-183-80

October 24, 1980

Honorable John F. Ahearne, Chairman
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Dear Mr. Ahearne:

SUBJECT: NRC REPORT ON ANTICIPATED TRANSIENTS WITHOUT SCRAM

References:

1) Proposed Rulemaking to Amend 10CFR Part 50 Concerning Anticipated Transients Without Scram (ATWS) Events, SECY-80-409, September 4, 1980.
2) NUREG-0460, "Anticipated Transients Without Scram for Light Water Reactors"
3) "Assessment of BWR Mitigation of ATWS, Volume 11," NEDE-24222, General Electric Company, December 1979
4) "Assessment of BWR/3 Mitigation of ATWS," NEDE-24223, General Electric Company, December 1979
5) Letter from G. G. Sherwood to H. R. Denton, "Anticipated Transients Without Scram (ATWS) General Electric Comments on NUREG-0640 (Volume 4) Implementation Schedule," August 22, 1980

This letter is to provide General Electric comments on the anticipated transient without scram (ATWS) proposed rule and regulatory guide (Reference 1). General Electric is pleased to be able to provide its comments on such an important issue.

Our Company shares with the nuclear industry a strong desire for resolution of the long-standing ATWS licensing issue. We believe that ATWS can be resolved on a fair and equitable basis, but without the plant modifications and schedule requirements recommended in the NRC Staff report. We wish to provide some background on ATWS as it relates to the BWRs, make several recommendations for changing the Staff report, and propose a basis for a resolution of ATWS on the BWRs.

The NRC Staff has maintained (Reference 2) that the LWR scram systems do not provide the necessary reliability to eliminate NRC concerns about

ATWS, and accordingly the BWR required a separate and diverse shutdown system. General Electric and our customers argued that the BWR shutdown systems have high reliability and this had been demonstrated in the excess of 300 reactor years of experience and in our analyses. EPRI in an independent analysis reached the same conclusion. In addition, we offered to further improve the reliability of the scram system by incorporating the alternate rod insertion (ARI) system, a diverse means to initiate scram. (The NRC Staff refers to this system in conjunction with recirculation pump trip as Alternative 2a.)

Notwithstanding our arguments, the NRC Staff insisted that a separate, diverse, and automatic shutdown system was needed for the possibility of a control rod drive system failure. As a result of that requirement, General Electric and our customers provided analysis (References 3 and 4) which showed the capability of the existing standby liquid control system, when modified for two-pump operation, to shutdown the BWR. (The NRC Staff calls this system Alternative 2b when automated.) We firmly believe that this capability provides an acceptable diverse shutdown system meeting the additional NRC requirements. This judgment was also shared by the ACRS.

With this background in mind, we wish to provide the following comments on the proposed rule and regulatory guide:

1. ATWS Hardware Requirements

General Electric believes that Alternative 2a, the additional system to prevent ATWS, is sufficient for resolving the ATWS issue as discussed above. We defended these arguments in meetings with the NRC and ACRS, wherein we showed the ATWS risk reduction to be a factor of 100 with Alternative 2a.

However we acknowledge the Staff requirement for ATWS mitigation features. In this regard, it should be noted that for a BWR, the two-pump standby liquid control system mitigates all ATWS events postulated by the NRC Staff. This is based upon an extensive assessment of the BWR mitigation capability (References 3 and 4). Because of the capabilities of the liquid poison system, we believe that if the Commission requires additional mitigation features, modifications beyond Alternative 2b (manually initiated) are not necessary for the BWR and should not be required.

Justification for requiring mitigation beyond Alternative 2b has not been substantiated in the NRC report (Reference 1), nor in any of the earlier analyses. The Staff's value-impact evaluation has greatly overestimated the accrued value. However, even with the Staff's figures, the incremental value is less than the incremental impact for Alternatives 2c or 2d. If these Staff proposals are approved, BWR owners will be required to expend far greater resources than incremental value derived for these improvements. There is

fundamentally no reason for Alternatives 2c and 2d for the BWR in light of the mitigation capability provided by Alternative 2b.

2. ATWS Models

The NRC Staff has suggested that BWR evaluation models used for ATWS may require upgrading. More than 25 man-years have been expended preparing our recent BWR ATWS mitigation analysis (References 3 and 4) in conformance with Staff guidance and responding to the many Staff requests for ATWS scenarios. The GE reports, which total more than 600 pages, demonstrate that the BWR satisfies the Staff's ATWS mitigation criteria with Alternative 2b. The Staff concluded in an ACRS meeting that GE's analysis was the most complete of all industry submittals. The BWR models used in these analyses are qualified for transient analysis for all BWR design basis events. The postulated ATWS events are similar to these design events and do not create conditions which invalidate the use of these models.

Accordingly, General Electric believes that BWR models have already demonstrated the capability to analyze ATWS events. These models and ATWS analysis should be accepted by the NRC without further modification or demonstration tests. Agreement on the part of the Staff that GE's models satisfy ATWS criteria would eliminate confusion on this issue and permit design activities to proceed by General Electric and its customers.

3. Schedule and ATWS Exemptions

We believe that the NRC Staff has asked for unrealistic schedules for the implementation of ATWS modifications (Reference 5). Although 1982 may seem reasonable on the surface for completing ATWS modifications, this does not take into consideration the substantial amount of work by General Electric, our customers, and their architect engineers needed to do the detailed design work on a plant-by-plant basis. Therefore we recommend that these mandatory schedule requirements be relaxed in favor of more realistic schedules which could be decided in discussions between the NRC and its licensees, after the latter determine engineering and hardware schedule requirements. We also recommend that the schedule requirements be removed from the rule so that adequate time can be devoted to their development.

Reference 1 also identifies other exemptions to the requirements for early operating plants and similar plants at the same site. These exemptions are presently not contained in a proposed rule and we believe they warrant inclusion. The acceptance criteria for the plants should be contained in the rule while the regulatory guide should include detailed clarification of the requirements along with the proposed schedules.

We have provided a number of more detailed recommended changes to the rule and regulatory guide in the enclosure to this letter. Within the next few weeks, we will provide additional, specific comments on the wording contained in Reference 1. We are willing to work with the NRC Staff to resolve these concerns.

General Electric strongly recommends that the Commission request the NRC Staff to address the issues identified in this letter and thereby modify their proposed rule and regulatory guide. We believe that the resolution of ATWS as an unresolved safety issue can be effectively achieved by requiring the Staff's Alternative 2a. Such a resolution would significantly reduce the already small ATWS risks in a short period of time, and without major diversion of resources from other safety programs on the part of industry and the NRC.

If the Commission in its ultimate evaluation of the ATWS issue for the BWR requires more than Alternative 2a, the Commission should seriously consider limiting this requirement to a manually-initiated Alternative 2b, as this modification clearly satisfies the NRC stated concerns on ATWS. In addition, the Commission should insist on a manageable and achievable resolution of the ATWS issue considering the schedule and other concerns described in this letter. We do not believe that SECY 80-409 provides this capability, and therefore needs to be modified.

We would pleased to discuss this matter with you or members of your staff. Please feel free to call me (408) 925-5040 or Mr. R. H. Buchholz at (408) 925-5722.

Very truly yours,

Glenn G. Sherwood, Manager
Safety & Licensing Operation

GGS:ggo/104E

Enclosure

cc: Commissioner Bradford
Commissioner Gilinsky
Commissioner Hendrie
Secretary of the Commission, S. C. Chilk
M. S. Plesset (ACRS)
H. R. Denton (NRC)
C. D. Gibbs (AIF)

ENCLOSURE

ADDITIONAL BWR RELATED COMMENTS ON SECY-80-409

Enclosure A, Page A-11, Evaluation Model Conservatisms. In the past the Staff's position has been to require realistic analyses for ATWS events. We suggest that this section be rewritten to indicate that the ATWS evaluation models should be based upon realistic assumptions as specified in the first reference to this enclosure.

Enclosure A, Page A-12, Consideration of Single Failure in the ATWS Mitigating Equipment. The requirement to include consideration of single failure in the ATWS mitigating equipment for plants with OLs after 1/1/84 is not consistent with the mitigation system criteria in the proposed regulatory guide (Page E-13). The regulatory guide criteria allow the use of one of the three following options: overall unavailability of 10^{-2}/demand, meet mitigation acceptance criteria with the most limiting single failure, or an optimization to enhance reliability within the limitations imposed by the existing plant design. We recommend the rule be modified to permit the options specified in the regulatory guide.

Enclosure A, Page A-13, Primary System Pressure Criterion. This criterion allows steam generator tubes to be exempted from the Service Level C requirement. We propose that Service Level C exemptions be allowed for any LWR pressure boundary component provided that there is sufficient empirical data and analysis to justify the exemption.

Enclosure A, Page A-15, Oscillation Criterion. The criterion of no damaging neutron flux oscillations presently applies only to BWRs; it either should not be a criterion or be applied to all LWRs.

Enclosure A, Page A-17, Capability to Attain Natural Circulation. This item appears to be a collation error since the concern for natural circulation capability is only mentioned in Enclosure E as a PWR oper item [e.g., Page E-2, Paragraph titled Void Tests (PWR)]. This natural circulation requirement should be removed from the BWR.

Enclosure E, Page E-3, Separate Effects Tests For BWRs. It has been General Electric's practice to qualify its transient computer models against actual data from operating reactors, since these data provide the most valid conditions for model qualification. Routine and special startup tests have been used in addition to special tests such as the turbine trip performed at Peach Bottom-2 and the KKM reactors. These tests include all the relevant phenomena (rapid pressurization, depressurization, water level change, reactivity changes) and exercise all the plant systems under expected operating conditions. Comparison of model predictions with these data validate the model for conditions expected during an ATWS. These separate effects tests are not needed for verification of the modelling assumptions and should be removed from the regulatory guide.

Enclosure E, Page E-6, Local Pool Temperature Limit for BWRs. A temperature limit of 200°F is imposed on the suppression pool. Extensive testing of the quencher discharge device in Germany and in Italy over the past seven years has demonstrated that quenchers ensure smooth condensation up to bulk boiling conditions. We propose no temperature limit need be imposed for SRV discharge bevond that specified for containment design.

Enclosure E, page E-6, Fuel Behavior Analysis. The statements about "bounding estimates" of 100% fuel failure for a BWR are unrealistic. Calculated BWR failures are zero. A more appropriate bounding estimate should be established.

Enclosure E, page E-7, Coolable Geometry Criterion. The proposed local fuel enthalpy limit of 267 cal/gram is not justified as a criterion for coolable geometry for the following reasons:

a. Comparison of an ATWS event to a reactivity insertion accident (RIA) for the purpose of deriving a coolable geometry criterion is inappropriate. In the limiting rod for a RIA, the fuel starts with an isothermal room temperaure and receives an energy impulse in the order of 0.2 seconds. The preponderance of the impulse energy is toward the outside of the rod, due to the flux distribution within the rod. Because of this rapid energy deposition, it has been convenient to correlate the data on the basis of the radial average enthalpy that is input during the insertion. For total energy depositions up to about 230 cal/gram in an RIA, the failure mechanism is normally a small cladding split resulting from a low pressure rupture.[2] This is substantiated by (i) high speed photographs taken during the SPERT/TREAT test series, and (ii) data taken from the SPERT tests which indicate that the rod internal pressure is sufficient to produce rupture at the high temperatures encountered during an RIA event, i.e., the pressure/temperature data correlate well with burst test results. Coolable geometry is assured, even though there may be a fuel clad failure. For total energy deposition above ~250 cal/gram the failure mechanism changes from low pressure rupture to cladding melt. For the limiting RIA, a conservative coolable geometry criterion of 280 cal/gram (radial average, not maximum local) has been adopted, even though the test data show no evidence of clad rupture due to gross fuel melting until ~330 cal/gram.

   In the limiting ATWS event, the peak fuel rod starts with a rated-power temperature distribution and receives a much lower energy impulse than the RIA for about 3 seconds. This is sufficent time for heat transfer to remove a significant amount of the energy added to the fuel rod. It has been demonstrated in previous ATWS submittals[3] that (i) the cladding hoop stress remains below the Zircaloy rupture stress and (ii) the peak clad temperature is ~50% below the Zircaloy melting temperature. Therefore, maintenance of coolable geometry due to a rapid energy insertion is not a concern for ATWS.

b. It is assumed that one of the reasons that the 267 cal/gram local limit is being proposed is to prevent melting of the fuel center. General Electric considers this criterion inappropriate. There is

a wealth of information available[4-8] to indicate that fuel centerline melting is a relatively mild event which does not necessarily equate with fuel cladding failure, let alone loss of coolable geometry. This is particularly relevant in an ATWS event wherein the cladding experiences rapid heating as the result of DNB and expands away from the fuel. The small increase in fuel volume as a result of localized centerline melting would not result in any significant fuel-to-cladding mechanical interation and, thus, coolable geometry would be maintained.

c. There are events which result in coolable geometry concerns independent of fuel melting, e.g., LOCA. For these events, we have developed the conservative criteria of less than 17% local cladding oxidation and less than 2200°F peak clad temperature as evidence of maintaining coolable geometry. General Electric believes that the LOCA criteria are sufficiently conservative to assure a coolable geometry in a limiting ATWS event, and are the appropriate measures to use in ATWS licensing evaluations.

Enclosure E, Page E-8, Mitigation System Actuation Times. It is required that mitigation system actuation times must be consistent with the technical specification limits unless substantial data supports use of a different value. We recommend that the requirement be for nominal actuation times which can be supported by data or analysis.

Enclosure E, page E-11 Mitigating System Actuation Circuitry Reliability/Availability Criteria. Criterion number 9, which requires the circuitry to meet either IEEE-279 requirements or an unavailability of less than 10^{-3}/demand, is being applied to all ATWS mitigating systems. This is considerably beyond any of the previous Staff proposals and is not necessary, nor cost-effective. We recommend circuitry reliabilities not be required to be superior to the reliability of the function which they actuate in mitigating an ATWS, e.g., the control-grade feedwater runback feature.

Enclosure E, page E-14, Use of Approved RPT Designs. While the identified recirculation pump trip designs have been accepted by the NRC Staff, other RPT designs may also be proposed and acceptance of these new designs should not be excluded.

Enclosure E, page E-14, SLCS Actuation Circuitry Reliability. It is stated that for plants beginning operation before January 1, 1984, the automatic actuation cricuitry may have a reliability equivalent to the mechanical portion of the SLCS. However, the reliability of the existing mechanical portion of the SLCS is inadequately addressed, and therefore, the automatic actuation circuitry is in effect not exempted. We suggest that this criterion be reworded to require that the unavailability per demand of the SLCS actuation circuitry be equal the unavailability of the existing mechanical portion of the SLCS.

Enclosure H, Browns Ferry Event. It should be noted that the Browns Ferry 3 partial scram failure had no safety impact, and even if it had happened during a full power transient, no adverse consequences would have resulted. The case presented in Enclosure H assumes much further

degradation (total ½ scram failure without the successful subsequent insertion that was achieved in the actual situation). The assumption of core uncovery in 30 minutes without "considerable operator actions" is totally unrealistic. We recommend that this Enclosure be removed from the SECY since the Browns Ferry 3 event has practically no significance to an ATWS event.

REFERENCES

1. Letter, R. J. Mattson to G. G. Sherwood, February 15, 1979.
2. "LWR Fuel Response During RIA Experiments," NUREG/CR-0269, August 1978.
3. Letter, E. A. Hughes to D. F. Ross, General Electric ATWS Program, September 9, 1976.
4. "Post Irradiation Examination Results for the Irradiation Effects Test IE-1, TREE-NUREG-1199," February 1978.
5. "Post Irradiation Examination Results for the Irradiation Effects Test IE-3, TREE-NUREG-1200," March 1978.
6. "Post Irradiation Examination Results for the Irradiation Effects Test IE-5, TREE-NUREG-1201," March 1978.
7. "Fuel Rod Behavior During Test PCM-4," NUREG-CR-0903, August 1979.
8. "An Assessment of Fuel Melting, Radial Extrusion and Cladding Thermal Failure During a PCM Event in Light Water Reactors," NUREG-CR-0500, May 1979.