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May 15, 1980

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

Subject: Anticipated Transients Without Scram (ATWS)

Dear Mr. Denton:

This letter is writt. on behalf of the owners of Combustion Engineering designed and operating NSSS power plants. We fear that the ATWS issue is about to be resolved by the NRC without considering many of the relevant views pertinent to this issue. We find the need to express our concerns at this time because we are in fundamental disagreement with many of the positions expressed in NUREG-0460, Volume 4.

In Volume 4 of NUREG-0460, the Staff states that their recommendations for resolving ATWS are necessitated by the failure of the early verification program. We disagree with this conclusion. We have attached brief comments on the eleven items which the Staff summarized as areas of inadequacy in the industry submittals on early verification of the PWR designs for ATWS. More detailed discussion of NUREG-0460, Volume 4 and additional information on CE NSSS's ATWS mitigation capabilities are being prepared for submittal in July, 1980.

We urge that you consider these comments and our forthcoming detailed review in your deliberations on the ATWS issue. To this end, we stand ready to meet with you and your Staff at your convenience to discuss this matter.

Sincerely,

Arthur E. Lundvall, Jr.

For CE NSSS Owners

cc: Commissioner Ahearne, Chairman
Commissioner Bradford
Commissioner Gilinsky
Commissioner Hendrie
Commissioner Kennedy
Dr. Milton Plesset, ACRS
Dr. William Kerr, ACRS Subcommittee on ATWS

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COMMENTS ON
NUREG-0460 VOLUME 4

Based upon the industry submittals as part of the ATWS early verification efforts, the NRC staff, in NUREG-0460, Volume 4, concluded that the adequa... had not been demonstrated for the proposed Alternative 3 modifications of NUREG-0460, Volume. The areas which the NRC staff judged to be inadequately addressed in these industry submittals were summarized, for the PWR designs, in Section 1.3.1 of Volume 4. In that Section, the Staff identified eleven specific items. The following comments address those items by number:

ITEM (1): Not all significant anticipated transients were analyzed. The stuck-open power-operated relief valve (PORV) anticipated transient has not been correctly analyzed.

COMMENT: The more recent report, CENPD-263-P, provides analysis of the complete loss of feedwater transient because this transient was shown to lead to the highest peak primary system pressures. Analyses of other less severe transients were presented by referencing other submittals, including CENPD-158 and CEN-114-P. The analysis of the failure of a PORV to close, discussed in Section 2.3.2 of CENPD-263, provides an overall assessment of this event.

ITEM (2): Long-term shutdown has not been adequately addressed. In particular, the impact of voids in the primary system after the initial pressure peak has passed, the timing of the reactor coolant pump trip, and the plants with low high-pressure safety injection (HPSI) shutoff head have not been addressed. The PWR transient codes used in these analyses are unacceptable for situations where significant voids are calculated to be present in the primary.

COMMENT: Although this aspect of ATWS has not received detailed attention in previous submittals, long-term shutdown following ATWS could be evaluated utilizing the engineering and analytical expertise gained in work on post-LOCA long term cooling and small break LOCA analysis. The effect of reactor coolant pump operation was addressed in Section 2.3.4 of CENPD-263-P. The impact of lower shutoff head of some HPSI designs is being considered elsewhere, and does not appear to be an issue specific to ATWS. The available analytical codes can analyze ATWS transients in sufficient detail by utilizing different codes during the different portions of the ATWS event, rather than developing new transient codes.

ITEM (3): Combustion Engineering (CE) information reveals that some instrument capability will be lost due to high primary pressure; this is likely to be the case for the other PWRs also. Ability of the instruments and equipment needed for safe shutdown to withstand the pressure peak are only partially addressed by C-E and not addressed at all by Babcock & Wilcox (B&W) and Westinghouse (W).

COMMENT: The information provided on instrument pressure capabilities was based upon the limits of the instrument vendor's specification and qualification. Although the instruments may not have certification for the highest predicted ATWS pressures, there is no evidence available to indicate these instruments would fail. Following the events at Three Mile Island-Unit 2, many instruments survived and functioned well beyond the conditions of their design qualification.

ITEM (4): The impact of isolated PORVs on plant response to ATWS has not been adequately addressed.

COMMENT: The impact of a closed PORV is provided in Section 2.3.1 of CENPD-263-P. Additional sensitivity to pressurizer relief area is provided in Figures 2-35 and 2-36. This assessment appears to indicate that isolation of a PORV does not significantly change the predicted peak RCS pressure.

ITEM (5): The calculated peak pressure for operating C-E plants would exceed 4000 psi even with the vessel head lifting as calculated to relieve the primary pressure. Also, many components exceed service level "C" stress limit.

COMMENT: The analytical ground rules in NUREG-0460, Volume 3 were utilized for the ATWS verification analyses submittal. Consistent with Alternative 3, C-E plants were to provide "Demonstration of the integrity of the primary coolant system boundary and functionability of valves needed for long-term cooling following conditions calculated for specified ATWS events." CENPD-263-P provided that demonstration. For existing PWR designs, establishing an upper pressure limit of 4000 psi or service level "C" stress appears to be arbitrary and does not have a clear technical basis.

ITEM (6): No stress evaluation has been provided for balance-of-plant components.

COMMENT: The stress analyses provided in CENPD-263-P were generic and were intended to typify the components used within the reactor coolant system pressure boundary. However, the direct applicability of these stress analyses to all BOP components could be demonstrated.

ITEM (7): Overly optimistic assumptions were used in the B&W peak pressure calculation. The staff believes that some components would be calculated to exceed service level "C" limit if more appropriate assumptions are used.

COMMENT: Not applicable to C-E plants.

ITEM (8): Many questions remain on radiological evaluations if the containment structure is not isolated soon after the initiation of an ATWS event.

COMMENT: The analyses provided indicate no fuel damage to occur and, thus, the radiological consequences would remain within the guidelines of 10CFR100. Containment isolation requirements are being addressed within the efforts associated with NUREG-0578, which should preclude additional requirements in this area to resolve ATWS.

ITEM (9): Design information on preventive and mitigative systems has been inadequately addressed.

COMMENT: The objective stated for the ATWS early verification was "to confirm that the mitigation capability we (NRC staff) believe to exist in those designs (C-E and B&W) does in fact exist." CENPD-263 was written with this objective in mind. Until the inherent mitigation capability was confirmed, we could not establish the need or design criteria for preventive and mitigative systems. Prior to expending significant resources on the design of preventive and mitigative systems, we feel that the NRC and industry should reach agreement on the inherent mitigation capability of existing designs.

ITEM (10): If HPSI is actuated early (automatically or manually) while the primary system pressure is above the HPSI design pressure, its operability and integrity are questionable.

COMMENT: If HPSI is actuated early while primary system pressure is above the HPSI pump shutoff head, the HPSI discharge injection isolation check valves will remain seated by primary system pressure. These isolation valves and the associated piping within the reactor coolant pressure boundary were considered in the evaluation provided in Section 3.3 of CENPD-263. As stated, no plasticity is predicted in the active valve bodies and no loss of ability to function is predicted for these valves. Since all stress levels are still well within service level D limits, there appears to be no reason to question the system integrity.

ITEM (11): The effect of pressures substantially above the 3400-3500 psi range considered in Volume 3 is not well understood. In particular, the integrity and performance on safety and relief valves has not been assured; the TMI-related industry testing program is not expected to encompass this extreme pressure range.

COMMENT: Although extensive test data on the effect of pressure above the 3400-3500 psi range are not available, the methodology for stress analysis at stresses corresponding to these pressures is well understood. As stated in Section 3.3.5 of CENPD-263, the inlet flange and bolting of the pressurizer safety valves satisfy Service Level C stress limits. The EPRI valve test program will provide full scale test data at pressures well above the pressures for which data are now available. When the EPRI valve test data are available, much less extrapolation will be necessary to give reasonable assurance of the performance of the safety and relief valves at the slightly higher pressures expected during an ATWS event.