Docket

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

June 5, 1979

50-335

ALL OPERATING COMBUSTION ENGINEERING PLANTS

Gentlemen:

We are generically reviewing operating plants using Combustion Engineering (CE) designed reactors in light of the Three Mile Island Unit No. 2 (TMI-2) incident. Accordingly, the purpose of this letter is to advise you of several concerns involved with this generic review and to request your active participation in resolving these matters.

We have scheduled a meeting with the owners of all operating plants using CE designed reactors. This meeting is scheduled for 9:30 a.m., June 12, 1979, in Room 6507, Maryland National Bank Building, 7735 Old Georgetown Road, Bethesda, Maryland. You are expected to attend this meeting and to be prepared to discuss the matters identified below.

(1) We are preparing a generic report on TMI-2 matters related to operating plants that use CE designed reactors. Although this report is not yet complete, we expect that it will recommend further analyses of transients and small reactor coolant system breaks, the development of appropriate written procedural guidance to operators as indicated by these analyses, and further training of operators in the use of these new procedures. A clear understanding of the remaining scope of work is expected to be developed during the course of the meeting using the staff's information needs shown in Enclosure 1 regarding small break LOCA analyses and models.

(2) In certain instances, licensees are using fuel which was not provided by CE; therefore, these licensees may be relying on safety analyses which were not provided by CE. Thus, it is not clear to us what the respective roles of the licensees, CE, the fuel suppliers or other parties should be in implementing such activities as the development of small break and transient analyses and the actual system guidance for the preparation of operator procedures. (See item 1, above) We therefore need a clear and concise definition of the respective roles of the above parties in these cases.

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(3) As a result of its review of the TMI-2 incident, the ACRS has issued five letters to the Commission containing recommendations to preclude TMI-2 type events. CE was previously requested to provide the staff with a concise discussion and its position on each of the recommendations contained in the ACRS letters. By letter dated May 25, 1979, attached hereto as Enclosure 2, CE submitted only a general response which did not address individual recommendations. We believe that it would be mutually beneficial for utilities to provide specific comments on those recommendations having a potential impact on plant design and/or operations. A summary of the ACRS recommendations is provided in Enclosure 3.

You should be prepared to discuss the procedure and a tentative schedule regarding the submittal of information needed by the NRC staff to complete its review.

It is clear that a number of technical issues will need to be resolved for plants using CE designed reactors. The difficulty in performing the necessary work with the limited resources available within the NRC is intensified by the need to conduct similar and concurrent activities with owners of Babcock & Wilcox, Westinghouse and General Electric designed operating plants. At the same time, there is need to resolve these matters promptly.

To resolve the issues involving CE designed reactors in an expeditious manner, we believe there is a compelling need to establish an owner's group for CE operating plants. We expect that such a group would be needed for the remainder of calendar year 1979. Since owner's groups have been effective in the past in resolving generic problems with a minimum of staff and industry resources, we strongly urge you to meet with other owners of CE operating plants to consider the formation of such a group prior to our meeting on June 12, 1979. This matter will be one of the principal agenda items at that meeting.

Please note that the investigation of a number of areas related to the TMI-2 accident, including the ACRS recommendations and action items from NUREG-0560, "Staff Report on the Generic Assessment of Feedwater Transients is Pressurized Water Reactors Designed by the Babcock & Wilcox Company," will be specifically included as a part of the "Lessons Learned" staff activity. You can expect additional correspondence in the future on these items.

If you require any clarification of the matters discussed in this letter, please contact I. Villalva, the staff's assigned project manager for these generic activities involving CE designed reactors. Mr. Villalva may be reached on (301) 492-7745.

Sincerely,

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Operating Reactors

Enclosures:
1. Small Break LOCA Analysis Information Needs
2. CE May 25, 1979 General Response
3. Summary of ACRS Recommendations

cc w/enclosures: See next page

ENCLOSURE 1

REQUEST FOR ADDITIONAL INFORMATION
REGARDING SMALL BREAK LOCA ANALYSIS

The response of the primary system of a given plant to small break LOCA's will differ greatly depending upon the break size, the location of the break, mode of operation of the reactor coolant pumps, numbers of ECCS systems functioning, and the availability of secondary side cooling. In addition, this response may differ for different plants designed by the same NSSS vendor because of differences in loop configuration (i.e., 2-loops, 4-pumps; 3-loops, 3-pumps) or different ECCS designs. In order for the staff to complete its evaluation of the response of currently operating C-E PWR designs to postulated small break LOCA's, the following information is needed:

(1) Provide a qualitative description of expected system behavior for (a) a range of postulated small break LOCA's, including the zero break case, and (b) feedwater-related limiting transients combined with a stuck-open power operated relief valve. These cases should include situations where auxiliary feedwater is both assumed available and not available. The cases considered should also include breaks large enough to (a) depressurize the primary system, (b) maintain the primary system at some intermediate pressure, and (c) repressurize the primary system to the safety and/or relief valve setpoint pressure. Various break locations in the primary system should be considered, including the pressurizer.

(2) Provide a qualitative description of the various natural circulation modes of expected system behavior following a small break LOCA. Discuss any ways in which natural circulation can be interrupted. In particular, discuss the applicability of the concerns in the Michelson reports (reports on B&W 205 FA plants and CE System 80 plants) identified in Annex 1 to this Enclosure. Assess the possible effects of non-condensible gases contained in the primary system.

The following questions pertain to your small break LOCA analysis methods:

(3) Demonstrate that your current small break LOCA analysis methods are appropriate for application to each of the cases identified in Items (8) through (12) below. This demonstration should include an assessment of the adequacy of the pressurizer and steam generator noding, and the pressurizer surge line representation. This may be accomplished by verifying the methods with the use of data (e.g., comparison with experiments, TMI-2 evaluation).

If, as a result of the above assessment, you modify your analysis methods (e.g., pressurizer and steam generator noding), provide justification for any such modification.

(4) Verify the break flow model used for each break flow location analyzed in the response to Item (8) below.

(5) Verify the analytical model used to calculate natural circulation heat removal under two-phase flow conditions.

(6) Provide justification for your treatment of non-condensible gases following discharge of the safety injection tanks.

(7) Verify your analytical calculation of fluid level in the reactor pressure vessel for small break LOCA's and feedwater transients.

For each of the analyses requested in Items (8) through (12) below,

(i) Provide plots of the output parameters specified in Annex 2 to this Enclosure.

(ii) Indicate when the pressurizer safety and/or relief valves would open.

(iii) Include appropriate information about the role of control systems in the course of the transient. Describe how the system response would be affected by control systems.

(iv) If the scenario is different for different classes of plants (two-loop, three-loop, four-loop, different ECCS designs), provide an example of each kind.

(8) Provide the results of a sample analysis of each type of small break behavior discussed in the response to item (1) (e.g., depressurization, pressure hangup, repressurization).

(9) Provide the results of an analysis of the worst break size and location in terms of core uncovering. This may be a break which does not result in HPI initiation. This may require more than one calculation.

(10) Provide the results of a complete analysis of feedwater-related limiting transients combined with a stuck-open power operated relief valve. These cases should include situations where auxiliary feedwater is both assumed available and not available.

(11) Provide the results of a small break LOCA analysis assuming loss of feedwater and auxiliary feedwater. The case with the worst break location which affords the least amount of time for operator action should be analyzed. Single failure of the ECCS should be considered.

(12) Provide the results of a small break LOCA analysis assuming that one steam generator is lost either due to isolation or due to loss of auxiliary feedwater.

(13) Provide the results of an analysis of the effect of reactor coolant pump operation (tripping all RCP's, keeping all and some RCP's running) on the course of small break LOCA's.

(14) Provide the results of an analysis of the effects of different HPI termination criteria on the course of small LOCA's. Specifically, for each small break LOCA analyzed in response to Item (8) above, compare the effects of the NRC HPI termination criteria (as stated in I&E Bulletin No. 79-06B, Item 6(b)) to those for the HPI termination criteria which C-E has recommended to licensees with C-E designed operating plants. Provide plots of significant parameters of interest, such as system pressures, temperatures, and subcooling, on a common time axis. Indicate on the plot when the operator would terminate HPI injection for both sets of criteria.

(15) Provide a list of transients expected to lift the PORVs; identify the assumed steam and two-phase flow rates through the valves for these transients. Provide justification for your assumptions, including the time at which two-phase flow discharge would be experienced.

(16) Provide guidelines for the preparation of operational procedures for the recovery of plants following small LOCA's. This should include both short-term and long-term situations and follow through to a stable condition. The guidelines should include recognition of the event, precautions, actions, and prohibited actions.

If RC pump operation is assumed under two-phase conditions, a justification of pump operability should be provided. Discuss instrumentation available to the operator and any instrumentation that might not be relied upon during these events (e.g., pressurizer level). What would be the effect of this instrumentation on automatic protection actions?

ANNEX 1

TVA (C. Michelson) Concerns

1. Pressurizer level is an incorrect measure of primary coolant inventory.

2. The isolation of small breaks (e.g., letdown line; PORV) not addressed or analyzed.

3. Pressure boundary damage due to loadings from a) bubble collapse in subcooled liquid and 2) injection of ECC water in steam-filled pipes.

4. In determining need for steam generators to remove decay heat, consider that break flow enthalpy is not core exit enthalpy.

5. Are sources of auxiliary feedwater adequate in the event of a delay in cooldown subsequent to a small LOCA?

6. Is the recirculation mode of operation of the HPSI pumps at high pressure an established design requirement?

7. Are the HPSI pumps and RHR pumps run simultaneously? Do they share common piping?/suction? If so, is the system properly designed to accommodate this mode of operation (i.e., are any NPSH requirements violated, etc...?)

8. Mechanical effects of slug flow on steam generator tubes needs to be addressed. (transitioning from solid natural circulation to reflux boiling and back to solid natural circulation may cause slug flow in the hot leg pipes).

9. Is there minimum flow protection for the HPSI pumps during the recirculating mode of operation?

10. The effect of the accumulators dumping during small break LOCAs is not taken into account.

11. What is the impact of continued running of the RC pumps during a small LOCA?

12. During a small break LOCA in which offsite power is lost, the possibility and impact of pump seal damage and leakage has not been evaluated or analyzed.

13. During transitioning from solid natural circulation to reflux boiling and back again, the vessel level will be unknown to the operators, and emergency procedures and operator training may be inadequate. This needs to be addressed and evaluated.

NOTE: Items 1 through 4 are taken from "Decay Heat Removal During A Very Small Break LOCA for a B&W 205-Fuel Assembly PWR," C. Michelson, Draft Report, January 1978.

Items 5 through 15 are taken from "Decay Heat Removal Problem Associated with Recovery from a Very Small Break LOCA for CE System 80 PWR," C. Michelson, Draft Report, May 1977.

(continued next page)

14. The effect of non-condensible gas accumulation in the steam generators and its possible disruption of decay heat removal by natural circulation needs to be addressed.

15. Delayed cooldown following a small break LOCA could raise the containment pressure and activate the containment spray system. Impact and consequences need addressing.

ANNEX 2

Plotted Output Parameters

Core: L, X, T_{CL}

Reactor Vessel:

Upper Head: L, X

Downcomer: L, X

Piping:

Hot Leg: X, T, W, L (Pressurizer Leg)

Cold Leg: X, T, W, L, W_{HPI}, $\int W_{HPI} dt$ (Break Leg)

Pressurizer: Win, Xin, L, X, P, T

Steam Generator:

Primary: X, L, T, h

Secondary: P, L, X, T, W_{REL}, W_{AFW}, h

Leak:

PORV, W, X

or

Break, W, X, $\int W dt$

Pump Loop Seal: X, L

Nomenclative:

P - Pressure
L - Mixture Level
X - Quality
T - Temperature
W - Mass Flow Rate
h - film heat transfer coefficient
HPI - High Pressure Injection
REL - Relief Valve
AFW - Auxiliary Feedwater

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ENCLOSURE 2

POWER SYSTEMS

May 25, 1979
LD-79-031

Dr. D. F. Ross, Jr., Deputy Director
Division of Project Management
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Subject: ACRS Recommendations Relating to TMI-2 Accident

References: (A) NRC letter from D. F. Ross, Jr. to A. E. Scherer, dated May 17, 1979.

(B) NRC letter from D. F. Ross, Jr. to A. E. Scherer, received May 23, 1979.

Dear Dr. Ross:

In response to the subject letters, you are no doubt aware that Combustion Engineering has been actively involved in evaluating the concerns listed therein and has participated in several discussions on these subjects with the NRC staff as well as a recent ACRS meeting. We have been concerned, however, that by simply evaluating the individual items listed in the ACRS letters, we would not adequately assess the total impact of these recommendations on overall plant safety. As a result, rather than addressing items on a case-by-case basis, we are addressing them as part of an overall design review program.

Combustion Engineering (C-E) has recently formed a task force to evaluate the safety implications for nuclear power plant design and operation of the TMI-2 accident. In parallel, this task force has begun a review of the sensitivity of the C-E design to abnormal operating conditions such as those which occurred at TMI-2. The task force will use the Three Mile Island-2 accident as an indication of areas for consideration and will not restrict itself solely to the consideration of TMI-2 specific problems. Furthermore, the task force is not restricting itself to traditional approaches to safety evaluation (such as considering only fixed scenarios, single failure philosophy, plant protection and recovery using only safety-grade systems or past design decisions). We believe that this effort will present the best evaluation of TMI-2 related problems and will consider all aspects of the C-E design.

We have, nevertheless, reviewed the specific concerns listed in the enclosures to your letters. Our initial review has indicated that while many of these recommendations appear to be desirable with respect to plant safety, the actual implementation of certain changes should await completion of our engineering evaluation. On the other hand several of the recommendations are being implemented. A natural circulation test is already conducted during the power ascension test program in every C-E test plant and in fact a complete natural circulation cooldown has been conducted at a C-E plant. Combustion Engineering has also provided recommendations to the utilities operating C-E designed plants in response to I&E Bulletin 79-06B which you have already seen. Other recommendations, such as reactor vessel level indication are under active consideration but will require additional evaluation.

Combustion Engineering intends to continue its evaluation of TMI-2 related problems. When the evaluation of each individual design change has been completed we will provide recommendations to our owners. If you should have any questions regarding this matter, please feel free to contact me or Mr. R. R. Mills, Jr. of my staff at (203)688-1911, extension 4738.

Very truly yours,

COMBUSTION ENGINEERING, INC.

A. E. Scherer
Licensing Manager

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ENCLOSURE 3

ACRS RECOMMENDATIONS

A. Letter, M. Carbon to Chairman Hendrie, dated April 7, 1979

Recommendation 1 - Perform further analyses of small break transients and accidents.

Recommendation 2 - Provide operator additional information and means to follow the course of an accident; as a minimum, consider expeditiously:

(a) unambiguous RV level indication

(b) remotely controlled vent for RCS high points

Recommendation 3 - Item 4b of Bulletin 79-05A considered unduly prescriptive in view of uncertainties in predicting course of anomalous small break transients/accidents.

B. Letter, R. Fraley to Commissioners, dated April 18, 1979

Recommendation 1 - Natural Circulation-related Items

a. Detailed analyses of natural circulation mode, supported as required by experiment, by licensees and NSSS vendors.

b. Develop procedures for initiating natural circulation.

c. Provide operator means for assurance that natural circulation has been established, e.g., by installation of instructions to indicate flow at low velocities.

d. Expeditiously survey operating PWR's to determine whether suitable arrangements of PZR heaters and reliable on-site power distribution can be provided to assure this vital aspect of natural circulation capability.

e. Operator should be adequately informed concerning RCS conditions which affect natural circulation capability, e.g.,

(1) indication that RCS is approaching saturation condition by suitable display to operator of T_c & T_h and PZR pressure in conjunction with steam tables

(2) use of flow exit temperature indicator by fuel assembly thermocouples, where available.

ACRS Recommendations

Recommendation 2 - Thermocouples used to measure fuel assembly exit temperatures to determine core performance should be used, where currently available, to guide operator concerning core status (full range capability).

Recommendation 3 - Operating reactors should be given priority regarding definition and implementation of instrumentation to diagnose and follow the course of a serious accident, including

(a) improved sampling procedures under accident conditions

(b) improved techniques to provide guidance to offsite authorities.

Recommendation 4 - Reiterates previous recommendations that high priority be given to "research to improve reactor safety"

(a) research on behavior of LWR's during anomalous transients

(b) NRC to develop capability to simulate wide range of postulated transients and accident conditions.

Recommendation 5 - Consideration should be given to additional monitoring of ESF equipment status, and to supporting services, to help assure availability at all times.

C. Letter, M. Carbon to Acting Chairman Gilinsky dated April 20, 1979

Recommendation 1 - Initiate immediately a survey of operating procedures for achieving natural circulation, including:

(a) event involving loss of offsite power

(b) consideration of role of PZR heaters.

ADDITIONAL RECOMMENDATIONS RELATING TO TMI-2 ACCIDENT IN MAY 16, 1979 ACRS LETTERS

A. Interim Report No. 3 dated May 16, 1979

Recommendation 1 - Examine operator qualifications, training and licensing, and requalification training and testing.

Recommendation 2 - Establish formal procedures for the use of LER information:

(a) in training supervisory and maintenance personnel

(b) in licensing and requalification of plant operating personnel

(c) in anticipating safety problems

Recommendation 3 - Consider formal review of operating procedures for severe transients by inter-disciplinary team, and develop more standardized formats for such procedures.

Recommendation 4 - Re-examine comprehensively the adequacy of design, testing and maintenance of offsite and onsite AC and DC power supplies with emphasis on:

(a) failure modes & effects analyses
(b) more systematic testing of power system reliability
(c) improved quality assurance and status monitoring of power supply systems

Recommendation 5 - Make a detailed evaluation of current capability to withstand station blackout, including:

(a) examination of natural circulation capability under such circumstances
(b) continuing availability of components needed for long-term cooling under such circumstances
(c) potential for improvement in capability to survive extended blackout

Recommendation 6 - Examine a wide range of anomalous transients and degraded accidents which might lead to water hammer, with emphasis on:

(a) controlling or preventing such conditions
(b) research to provide a better basis for control or prevention of such conditions

Recommendation 7 - Plan and define NRC role in emergencies, including consideration of:

(a) assurance that formal emergency plans, procedures and organizations are in place
(b) designation of emergency technical advisory teams (names and alternates)
(c) compilation of an inventory of equipment and materials needed in unusual conditions or situations

Recommendation 8 - Review and revise within three months:

(a) licensees' bases for obtaining offsite advice and assistance in emergencies from within and outside company
(b) licensees' current bases for notifying and providing information to offsite authorities in emergencies

Recommendation 9 - Examine the lessons learned at TMI-2, including consideration of the following:

(a) behavior, failure modes, survivability and other aspects of TMI-2 components and systems as part of the long-term recovery process
(b) determine if design changes are necessary to facilitate decontamination and recovery of major nuclear power plant systems

Recommendation 10 - Expedite resolution of unresolved safety issues by the following means:

(a) suitable studies on a timely basis by licensees to augment NRC staff efforts
(b) use of consultant and contractor support by NRC staff

Recommendation 11 - Augment expeditously the NRC staff capability to deal with problems in reactor and fuel cycle chemistry in the following areas:

(a) behavior of PWR & BWR coolants and other materials under radiation conditions
(b) generation, handling & disposal of radiolytic (or other) H_2 at nuclear facilities
(c) performance of chemical additives in containment sprays
(d) processing and disposal techniques for high and low level radioactive wastes

(e) chemical operations in other parts of nuclear fuel cycle
(f) chemical treatment operations involved in recovery, decontamination or decommissioning of nuclear facilities

Recommendation 12 - Reconsider whether or not use of the Single Failure Criterion establishes an appropriate level of reliability for reactor safety systems

Recommendation 13 - With respect to safety research:

(a) consideration should be given to augmentation of the FY80 NRC safety research budget
(b) consider orienting a larger part of the safety research budget toward exploratory (as opposed to confirmatory) research

Recommendation 14 - Perform design studies of a filtered venting or purging option for containments for possible use in the event of a serious accident.

Interim Report No. 2, dated May 16, 1979

Amplified many of the recommendations included in earlier ACRS letters dated April 7, April 18, and April 20, 1979, including ACRS views on relative priorities to be assigned a number of those earlier recommendations. (Address amplifications and suggested priority assignments as appropriate.)

cc:
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