NRC PDR

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

AUG 2 3 1979

Mr. Thomas D. Keenan, Chairman
General Electric Operating Plant Owners Group
Vermont Yankee Nuclear Power Corporation
Seventy-Seven Grove Street
Rutland, Vermont 05701

SUBJECT: ADDITIONAL INFORMATION REQUIRED FOR NRC STAFF GENERIC REPORT ON BOILING WATER REACTORS

Dear Mr. Keenan:

In early July we sent separate letters to each licensee with a boiling water reactor requesting additional information required for the NRC staff generic report on boiling water reactors. This was followed by my letter to you dated July 26, 1979 which requested certain additional information which was required in order to continue our review. Subsequently, two concerns have been identified that bear on our generic review of the boiling water reactors.

The first concern involves a direct current power source failure which was identified in a General Electric Company letter to the staff dated November 1, 1978 as the limiting single failure for a small break loss of coolant accident for certain classes of reactors. Therefore we believe it is necessary for us to address this failure explicitly in our generic evaluation. Responses to the requests for additional information contained in Enclosure 1 are required in order for us to assess the significance of this event.

The other concern involves a letter dated August 16, 1979 from the Advisory Committee on Reactor Safeguards to Chairman Hendrie. The Committee, in its letter, notes that the relatively high frequency of boiling water reactor pipe cracking suggests that there may be a significant probability of a loss coolant accident, particularly a small break loss of coolant accident and that it may be relevant to examine, in greater depth than usual, a range of matters including the following:

1. The reliability of the safety features needed to cope with such an event.

2. The possibility of determining the location of a leak or break more rapidly and more directly than is now the practice.

3. The adequacy of operational procedures for such loss of coolant accidents, including combinations of circumstances that could arise in connection with such an event.

751117

7909200282

We intend to address the issues raised by the Committee in our generic review of boiling water reactors. Therefore, we request that you explicitly address the above three matters for each class of General Electric Company boiling water reactor. The August 16, 1979 Committee letter is provided as Enclosure 2 to the letter for your information and use in developing your responses to the ACRS concerns.

In order for us to maintain our schedule we request that you provide by September 14, 1979 clear and complete responses to each of the requests contained in this letter. If you cannot meet this schedule, or if you require any clarification of these matters, please contact William F. Kane who may be reached at (301) 492-7745.

Denwood F Ross

Denwood F. Ross, Jr., Director
Bulletins & Orders Task Force
Office of Nuclear Reactor Regulation

Enclosures:
As stated

751118

Request for Additional Information

1. By letter dated November 1, 1978 GE presented an analysis to the staff that showed the worst single failure for a small break was a direct current (DC) power source failure. It is reques.ed that this failure be included in your postulated small LOCA analys. , and feedwater-related limiting transients combined with a stuck-open relief valve.

2. For the small break analysis, diversity of initiating signals does not exist for breaks which require ADS operation. Provide rationale for the adequacy of the initiating signals, considering the lack of diversity and increased calculated PCT.

3. Expand the time duration of the small break study to include long term control of reactor pressure, water level and heat removal. Identify the mitigating systems assumed to operate and the length of time ADS operation is assumed. If ADS is assumed to be unavailable, identify the backup system(s) assumed to be available considering a failure of each of the following:

   1) DC power source with loss of offsite power, and
   2) Loss of onsite and offsite AC power (station blackout)

   At what point can LPCI be safely diverted to another function such as pool cooling or containment spray?

4. For both LPCI modified and non-LPCI modified plants, compare the PCT consequenc stuck open and partially open safety relief valves to small LOCA s analyzed for power source failures assumed above. For this comparison describe the study performed and discuss assumptions made, including the following:

   1) systems assumed to provide mitigation,
   2) safety signals available to actuate the ECCS,
   3) containment pressure increase up to the initiation of the high pressure containment signal which actuates ECCS, considering containment coolers and suppression pool condensing which limit

the containment pressure increases, and

4) diversion of LPCI to pool cooling and containment spray, including the time when such diversion is assumed.

5. In your single failure analyses of November 1, 1978, you assumed "1/2h" when only 1 core spray was operating. Provide a basis for assuming "1/2h".

6. Given a DC power source failure and a line break in the operable core spray (CS) system, no CS or HPCI would be available to mitigate the accident. For both LPCI-mod and non-LPCI mod plants, it is possible that only 2 LPCI + ADS would be available. Accordingly, compare the PCT for breaks in the CS line and a DC power source failure to the PCTs of the break location/DC failure cases already analyzed for LPCI-mod and non-LPCI mod plants.

7. In the small break model, at what water level does the heat transfer coefficient (HTC) go to zero. The curves of level and temperature don't show HTC = 0 at hot node uncovery.

8. Provide a table listing the single failure, limiting break size and location, and operating ECCS for each operating BWR. Consider diversion of ECCS to containment or pool cooling and breaks in ECCS lines.

751120

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

August 16, 1979

Honorable Joseph M. Hendrie
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

SUBJECT: PIPE CRACKING IN LIGHT WATER REACTORS

Dear Dr. Hendrie:

There have been a significant number of occurrences of pipe cracking in boiling water reactors (BWRs), the Duane Arnold incident in 1978 representing the most severe example thus far. For pressurized water reactors (PWRs), leaks and deterioration of steam generator tubing have been significant problems and recently, cracking of a related but unpredicted type has been found in PWR steam generator feedwater nozzles.

The Nuclear Regulatory Commission's Pipe Crack Study Group issued a report in 1975 (NUREG-75/067) which reviewed BWR pipe cracking and made recommendations to reduce the incidence and severity of cracking. A second report (NUREG-0531) was issued in early 1979 which again examined the status of the incidence of pipe cracking and made further recommendations, primarily related to the influence of the choice of material and to the potential for inservice inspection.

The ACRS believes that it is appropriate to extend the scope of the NRC Staff review beyond that examined in NUREG-0531. The relatively high frequency of BWR pipe cracking suggests that there may be a significant probability of a loss of coolant accident (LOCA), particularly a small LOCA, and that it may be relevant to examine, in greater depth than usual, a range of matters including the following:

1. The reliability of the safety features needed to cope with such an event.

2. The possibility of determining the location of a leak or break more rapidly and more directly than is now the practice.

3. The adequacy of operational procedures for such LOCAs, including combinations of circumstances that could arise in connection with such an event.

Furthermore, the seeming long-time existence of large, deep cracks in the recirculation pipes at Duane Arnold suggests that a range of possible accident initiators such as water hammer, earthquakes or other potential sources of

751321

large additional forces could lead to a previously unexplored accident such as concurrent multiple failures. If so, consideration may have to be given to further analysis of the course of such an event in order to ascertain what, if any, additional measures are needed to reduce the probability of the accident or to mitigate its consequences.

The presence of the large, multiple cracks at Duane Arnold in sections of the pipe in which no inservice inspection was required, points to a need for a comprehensive reexamination of all safety-related piping systems for similar or equivalent design, fabrication or construction flaws, as well as the adequacy of the NRC requirements for inservice inspection. Furthermore, high priority should be given by both the industry and the NRC to the early implementation of improved crack detection capability.

Some types of cracking in PWRs and BWRs can be retarded through the control of water purity. For example, most foreign and some domestic BWRs deaerate the primary coolant during reactor startup. The NRC Staff is considering a regulatory guide on this matter. A program should be initiated to develop optimum water specifications, particularly in the areas of BWR primary coolant and PWR secondary coolant.

Sincerely yours,

Max W Carbon

Max W. Carbon
Chairman

References

1. NUREG-75/067, "Technical Report -- Investigation and Evaluation of Cracking of Cracking in Austenitic Stainless Steel Piping in Boiling Water Reactor Plants," Pipe Crack Study Group, U.S. Nuclear Regulatory Commission, October 1975.
2. NUREG-0531, "Investigation and Evaluation of Stress-Corrosion Cracking in Piping of Light Water Reactor Plants," Pipe Crack Study Group, U.S. Nuclear Regulatory Commission, February 1979.
3. "Metallurgical Investigation of Cracking in a Reactor Vessel Nozzle, Safe-End, Final Report," Southwest Research Institute, dated December 26 1978.