PDR

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

9/17/82

Mr. O. D. Kingsley, Chairman
Westinghouse Owner's Group
Alabama Power Company
600 North 18th Street
Box 2641
Birmingham, Alabama 35291

Dear Mr. Kingsley:

In response to your letter dated November 30, 1981, the staff has reviewed your proposed Emergency Response Guideline Program. This review has progressed to the point where we believe the Program provides a basis for a significant improvement over current emergency operating procedures. Its implementation at operating reactors should provide a greater assurance of operational safety than presently exists. Although this letter is not an approval of the guidelines, our confidence in your approach warrants proceeding towards plant-specific implementation. We suggest that this implementation proceed in three steps:

(1) Preparation of plant specific Procedures which in general conform to the Emergency Response Guidelines referenced above and implementation of these Procedures as outlined in SECY-82-111.

(2) Preparation of supplements to the Guidelines which cover changes, new equipment, or new knowledge and incorporation of these supplements into the Procedures.

(3) Completion and improvement of the Guidelines to meet our long term requirements, followed by incorporation of improvements into plant specific Procedures.

Step 1 refers to the Guidelines you have already transmitted to us and any other information which we receive by approximately October of this year. We anticipate completing an SER by January 1983. Step 2 refers to Guidelines updates which will be generated as a matter of routine after the plant specific Procedures have been put in place. This is essentially a maintenance function. Step 3 refers to those aspects of the Guidelines and Procedures where we feel additional work is needed to complete our long term requirements. Step 3 can be completed at a mutually agreeable future time. The immediate implementation of Step 1 will allow the benefits of the significant improvements you have achieved to be realized quickly, and the remaining work needed to meet our requirements for the Guidelines and Procedures can follow at later dates.

During our review, we have identified a number of issues which require additional information or development effort. A list of our concerns and questions is enclosed. We ask that you address these issues concurrent with your development of the remaining program.

Our conclusion that steps to implement the Guidelines into plant-specific Procedures are warranted is based on the judgement that no further major conceptual changes to the proposed Guidelines are expected. Also, this preliminary acceptance is contingent upon your response to a concern regarding future supplements and changes to the Guidelines, both with operating plants and new plants coming on line. We request that you describe your program for future changes or supplements to the Guidelines. Such efforts include:

(1) Generic items such as ATWS rulemaking, SPDS designs, RCS vent installations, IC instrumentation, and ongoing Unresolved Safety Issues.

(2) Completion of the Guidelines to include adequate consideration of items such as core damage, effects of severe natural phenomena, improvement for steam generator tube breaks, hydrogen control, and resolution of questions as a result of our continuing review.

Moreover, we request that you either submit a set of Guideline alternatives or explain how significant differences in plant design (e.g., UHI, UPI) and equipment are addressed to make the Guidelines applicable for all Westinghouse designed plants. As an example, it is expected that there will be a variety of systems for reactor vessel water inventory measurement and, in the interim, some plants will have no systems at all.

Sincerely,

Darrell G. Eisenhut, Director
Division of Licensing

Enclosure:
Questions of the Westinghouse Owner's Group Emergency Response Guideline Program

cc: Westinghouse Licensees
Westinghouse Applicants
Westinghouse

QUESTIONS OF THE WESTINGHOUSE OWNER'S GROUP-

EMERGENCY RESPONSE GUIDELINE PROGRAM

GENERAL QUESTIONS

1. You consider the Emergency Response Guideline Program to provide a comprehensive and integrated set of guidelines which address the events occurring during power operation and which are entered following a reactor trip. Our position is that the scope of procedure guidelines for transients and accidents, as required in the TMI Action Plan, should be larger. In particular, each plant should have procedures developed from guidelines which:

   a) Are entered prior to reactor trip where appropriate;
   b) Address the emergencies which may occur during startup, shutdown or low power conditions; for example, an inadvertent boron dilution during cold shutdown, or a pipe break while on shutdown cooling;
   c) Are utilized under non-emergency conditions when appropriate, such as may be the case for a substantial leak which does not meet the technical definition of a LOCA, but may best be treated under the Procedures developed as part of this program.

Provide the necessary guidelines for writing the procedures mentioned above.

2. Provide information as to how natural phenomena were considered as initating events.

3. The submitted set of guidelines does not appear to address certain multiple failure events identified below. We require that you establish that the Guidelines are adequate for these cases or make necessary improvements in the existing guidelines and/or write new contingencies to address the following concerns:

   a) If there is a large steam line break upstream of the isolation valves and a consequent tube rupture in the same SG, the operator does not get any indication of the tube rupture in the control room (radiation monitors are beyond the isolation valves and level is lost in the faulted SG as a result of the blowdown). Diagnostic steps to make sure that a tube rupture does not go unobserved due to a coincident steam line break appear to be needed.

   b) The present guideline structure does not appear to address possibility of simultaneous SG tube rupture and loss of reactor coolant. Step 3 of E-1 tells the operator to go to E-3 if the narrow range water level increases in an unexplained manner in one SG. On the other hand, containment indications seen after step 10 in E-3 refer him back to E-1. Further, a loss of secondary pressure control in coincidence with a SG tube rupture and a RCP seal failure would cause the operator to return from ECA-3, step 20, to E-1.

4. The background information of the guidelines presents results of various analyses. The results are presented in terms which may not be sufficiently clear to the operator. For example, presenting SG

level and pressurizer level instead of (or in addition to) the water volume or the total mass, and subcooling margin instead of saturation temperature, might give an operator a more direct relationship to the parameters he expects to observe during an accident. In particular, re-examine the format of the figures in the background information to determine whether improvements should be made.

5. The background information is related to specific guidelines. However, there are phenomena which may occur unexpectedly during several different events and which may confuse the operator. Most recent examples are the reactor vessel upper head voiding and the pressurizer level increasing offscale high during the Ginna tube rupture event. Other phenomena might be oscillating or reversed flow in an idle loop during loss of forced reactor coolant flow. As a minimum such phenomena should be discussed in a generic background information document which is used for operator training.

6. We do not consider the guidelines to be applicable for the San Onofre-1 plant due to the lack of main steam isolation valves, one electric AFW pump, FW pumps which act as ECCS pumps, etc. Further justification will be needed for San Onofre-1 to refer to the generic guidelines.

<u>QUESTIONS RELATED TO PARTICULAR GUIDELINES</u>

Guideline E-1

1. Step 12 provides instruction to implement ES-1.2. The operator is expected to continue in guideline E-1 at the same time he implements ES-1.2. This should be clearly identified in step 12.

Guideline ES-1.1

1. The analysis presented as background information for E-1 (case D) shows that if the break size is in a certain range and the operator follows the SI termination and reinitiation criteria, he could create an oscillatory reactor coolant pressure. The instruction should be modified to provide the operator proper guidance on how to avoid multiple HPI termination and reinitiation.

Guideline E-2

1. The earlier Westinghouse procedures recommended an increased subcooling margin (+15°F) to be used in the SI termination criteria if there was a need for SI restart. Why have you deleted that requirement? (or add this guidance to the current version).
2. Step 7 instructs the operator to isolate AFW to the faulted SG. If this cannot be done immediately, e.g., because of failed open isolation valves, we assume that the operator should proceed in the guideline. Provide appropriate guidance in step 7, "response not obtained"-column.

Guideline E-3

1. A caution before step 1 instructs the operator to consider the SG with the lowest level as non-ruptured if all SGs ruptured. In such a case, there is a continuous loss of coolant outside the containment and a risk of loss of RWST water. For some leak combinations, the radioactive release to the environment and the

total loss of primary coolant outside the containment might be lower if alternative actions were taken.

Provide the rationale that your proposed method is the most desirable of the options available considering the above concerns, or provide the necessary guidance for alternative means of cooldown.

2. As seen during the Ginna event of January 25, 1982, it is possible that a steam bubble will be created in the reactor vessel upper head in guideline step 12 (or 14). You should add the approriate cautions and consider the action possibly needed to control and/or eliminate the steam bubble.

3. After the RCS achieves 400 psi and 350°F, the RCS cooldown is continued using RHR. The guideline does not instruct the operator how to control the RCS pressure or the ruptured SG pressure during this stage. The situation is the same also in guidelines ES-3.2B and ECA-3 (steps 17, 18). You should add the steps providing guidance on pressure control, or a note if the operator is expected to let the pressure decay by ambient heat losses.

4. It is our position that the alternate cooldown methods to minimize the radiological consequences, as presented in the guidelines ES-3.2, are preferred to steps 20 through 23 (steaming of the ruptured SG), even if the condenser is available. Step 18 should be modified taking this position into account.

Guideline ES03.1

1. The earlier Westinghouse procedures instruct the operator to stop all but one RCP before starting cooldown. This instruction is not provided in the current guidelines at step 16. Please explain.

Guideline ECA-3

1. Steps 21 through 29 do not instruct the operator clearly in which order he is to take his actions. This section of the guideline should be clarified.

Guideline FR-C.1

1. The criteria for entering guideline FR-C.1 are based on small break LOCA calculations where no HPI flow is assumed and the secondary side is kept at full pressure. The calculations show, for example, that for a one-inch break, the time for developing ICC conditions is more than two hours. During this time , the operator would be able to smoothly depressurize the RCS well below the accumulator pressure, if he starts secondary steaming at 30 to 60 minutes after the reactor trip. Moreover, staff analyses of the Zion plant show that depressurizing the secondary system for a small break LOCA with no HPI is an effective means of providing alternative ECC water sources (Accumulator, LPI systems). Justify not instructing the operator to start secondary steaming (especially if he knows there is no HPI flow) well before he meets the ICC conditions and is forced to take undesirable actions like step 16 of the guidelines.

2. The applicability of the guidelines is limited to situations where reactor coolant is discharged to the containment. An ICC situation might develop also, for example, as a result of SG tube rupture in coincidence with a loss of secondary side pressure control. During such an event, the operator would be told to exit the guidelines at step 7. Revise the guideline in such a way that its applicability is not unnecessarily limited.

3. In step 16(or 26), the SGs are rapidly depressurized. This might result in uncovering of the SG tubes. During most of his training, the operator has learned that uncovering of the tubes is not permitted and he may have difficulties in deciding which is more important: keeping the tubes covered or depressurizing as fast as possible. Such guidance should be provided in steps 16 and 26.

4. The second note before step 25 instructs the operator to return to step 13 if the capability for dumping steam is restored while performing steps 25 through 34. Clarify the relationship between step 26 (which also requires capability to dump steam) and steps 13 through 24.

5. The background information for step 16 instructs the operator not to depressurize leaking SGs. INEL calculations prepared for the NRC and reported in EGG-CAAD-5428, April 1981 (C.D. Fletcher: Accident mitigation following a small break with coincident failure of charging and high pressure injection for the Westinghouse Zion I PWR), suggest that all SGs should be depressurized because any

hot loop could flash and keep the RCS above the accumulator pressure too long. Please provide justification to support your recommendations of keeping one or more leaking SGs hot.

Guideline FR-H.1

1. The "feed and bleed" cooling recommended by the guideline might cause a severe pressurized thermal shock because the reactor vessel downcomer would be essentially filled with cold ECC water. Demonstrate that your analyses of pressurized thermal shocks cover the proposed cooling and/or present the possible means to minimize the shock (e.g., maximum RCS pressure, maximum SI flow, cooling water temperature control during sump recirculation).

Status Tree

1. The pressure-temperature diagram included in the "Reactor Coolant Systems Integrity" status tree use the cold leg temperature. If forced coolant flow is lost and ECC water is injected into the RCS, the cold leg temperature reading may deviate significantly from the actual downcomer temperature (which is the critical temperature for system integrity). Provide guidance which is generally valid for determining the downcomer temperature.