NRC PDR

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

AUG 17 1978

Docket Nos. 50-10
50-237

50-10P

Commonwealth Edison Company
ATTN: Mr. Cordell Reed
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Gentlemen:

At our meeting with you on May 31, 1978, we indicated that our review of several SEP topics was essentially complete. We also stated that completed topic assessments would be sent to you for information and review and would be placed in the Public Document Rooms.

Our initial evaluation of eight of these essentially complete topics is enclosed. You are requested to carefully examine the facts upon which the staff has based its evaluation and respond either by confirming that the facts defining your plants are correct, or by identifying any errors. If in error, please supply corrected information for the docket. We encourage you to supply any other material for the docket related to these topics that you believe to be helpful.

At the May 31 meeting, the SEP Owners Group requested clarification of SEP documentation procedures and made several suggestions in that regard. Enclosure 1 is our response to the request and suggestions. It contains the documentation procedures to be used throughout the SEP program and discusses the bases for these procedures. Our documentation of the eight essentially complete topics in Attachment 1 illustrates the documentation procedure to be used.

We would appreciate any comments you may have to improve documentation of topic assessments.

Sincerely,

for Darrell G. Eisenhut, Assistant Director
for Systems & Projects
Division of Operating Reactors

Enclosure:
Response to the
SEP Owners Group
Suggestions

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NRC RESPONSE TO THE SEP OWNERS GROUP SUGGESTIONS FOR DOCUMENTATION PROCEDURES FOR THE NRC SYSTEMATIC EVALUATION PROGRAM

The suggestions for documentation procedures presented by the SEP Owners Group at the May 31, 1978 meeting closely parallels the staff views on the subject. We further believe that because of the nature of the program, the magnitude, and duration, it is imperative that all parties clearly understand the method by which the reviews will be performed and the process by which results will be documented.

It is just as important to realize that the underlying principle of our regulations holds the licensee ultimately responsible for the safety of its facility. However, the SEP is unlike normal staff safety assessments in that the staff will be initiating unilateral and in some cases de nova reviews which are not requested by the licensee. This approach places further emphasis on the importance for establishing, with common understanding, sound rules by which such assessments and reviews will be documented.

Regarding the evolution of the SEP topics, at the onset of the Systematic Evaluation Program, a list totaling more than eight hundred safety topics was compiled from several lists of outstanding concerns. Members of the staff were asked to submit any safety issues that they thought were germane and should be covered in the program. The Systematic Evaluation Program group studied the list and reduced it to only those safety-related appropriate topics. Many topics were deleted because of duplication, some because of non-applicability to light water reactors and others on the basis of being research and development. Topics relating to work

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the staff is involved in for the purpose of refining its techniques were also deleted. Areas that the NRC is considering but not yet implementing on new facilities were culled from the list. Additional deletions were items periodically reviewed or which have already been reviewed and implemented. Topics which were similar or related were combined in the preparation of topic definitions. Topics culled for "lesser safety significance" were defined and a justification written. At the completion of the culling process the remaining topics were divided into two groups, generic and non-generic; the generic topics being denoted by an asterisk in the final list. The scheduling and review of generic topics will be performed essentially independent of the SEP. However, implementation of resolved generic issues will be integrated into the program to ensure appropriate consideration of the impact of any changes or modifications. Throughout the entire culling phase, the staff maintained a tracable record of the disposition of every topic and refers to this whenever items surface that may impact the program, to ensure that the status of any previously considered topic has not changed. As stated throughout the program new issues of safety significance may be added to the program as they are identified or they may be resolved on an expeditious basis apart from any SEP schedule. An item of this nature and its resolution would require management approval prior to its inclusion in the SEP.

The staff has completed the review of several topics, the assessments for which are enclosed as Attachment 1. It is important to note that the staff's definition of "complete" for this phase does not necessarily mean that the topic has been closed out, but only that the staff has come to a decision point in that either the topic is satisfactorily resolved or cannot be resolved until consideration in the Design Basis Event review or other related topics. Once significant deviations from criteria are identified, the topics will be evaluated to determine their impact on related design basis events. It is necessary to follow this approach since some topic definitions are written such that extensive review would be required to complete the evaluation independent of specific DBE evaluations. By reviewing each topic only to the extent necessary to evaluate its effect on the DBE, for a given plant the complete review and evaluation can occur as part of the DBE evaluations in a more efficient manner with more balanced decisions.

SEP staff members are assigned areas of review responsibility. Docketed and related background material will be carefully reviewed to obtain as much information about a specific area as possible with minimum impact on a licensee. It is expected that the review of a large number of topics on the NRC final topic list for each plant can be completed based on the presently available information on the docket. For this first category of topics, no information will be required from licensees. The NRC will send to the licensees its interim evaluation of each such topic as they are completed in order that the licensee can carefully examine the facts upon which the staff based its evaluation. The licensee should respond either that the bases (facts defining the plant) are correct, or are in error. If in error, correcting information should be supplied to the docket. The licensee is not required to agree with the NRC evaluation and is also encouraged to supply any other material for the docket on these topics as he may choose. Topics in Attachment 1 fall in this group.

For the second category of topics, the NRC may require information regarding the plants from the licensee. To expedite the review process, information required from the licensee will be obtained at working meetings or by conference calls. Information obtained by this process must, however, at some later date, be placed on the official docket or formally supplied to the NRC in accordance with standard practice.

Analyses performed by the staff utilizing input from the licensee need only be reviewed and verified by the licensee for those facts describing the plant design unless for some reason the licensee chooses to use the staff's calculations for subsequent licensing justification. In that case, the licensee must be prepared to completely support the correctness of the staff information, bases, assumptions and calculations.

The third category of topics will be those that require licensee analyses. These will be treated in the same manner as those in routine licensing actions; licensee analyses will be placed on the official docket in accordance with standard practice. The staff will make every attempt to identify such topics to the licensees as early in the program as possible.

The content of staff assessments documenting completion of individual topics or areas of concern will be informally discussed with the licensees to ensure that the information used is factual and current and accurately protrays the facility. Initial assessments of individual safety topics or design basis events will be placed in the Public Document Room and forwarded to the licensee for comment. The initial assessments will be supplemented as needed to include correction or additional comments. At the completion of the program all initial assessments will be consolidated and a final assessment will be issued. NRC meeting minutes will typically be forwarded to the licensees for their review. Comments received will be placed in the Public Document Room.

ATTACHMENT 1

ASSESSMENTS OF ESSENTIALLY COMPLETE TOPICS

TOPIC III - 10C - Surveillance Requirements on BWR Recirculation Pumps and Discharge Valves

SEP Plants Affected - Millstone 1, Dresden 2

DBEs Affected - Loss-of-Coolant Accident

Discussion

This topic applies to the Low Pressure Coolant Injection System (LPCIS) at Boiling Water Reactors and specifically only to those systems which have undergone the LPCIS modification to remove the LPCIS loop selection logic. This logic network, which is still installed on two of the three applicable SEP Boiling Water Reactors (Millstone Unit No. 1 and Dresden Unit No. 2), is designed to direct LPCIS flow to the intact recirculation loop in the event of a Loss-of-Coolant Accident (LOCA). Oyster Creek has no LPCIS.

The logic network also was designed to close the suction and discharge valves of the intact loop to prevent LPCIS flow from bypassing the core and flowing out the break in the event of a LOCA. This modification was performed on all BWR-3 units (including Millstone Unit No. 1 and Dresden Unit No. 2) to allow closure of only the discharge valve. This is because in the unlikely event of a LOCA occurring between the suction and discharge valves of a recirculation loop concurrent failure of the loop selection logic, rapid break isolation prior to sufficient reactor depressurization which would allow influx of low pressure, high volume cooling water could result in increased peak clad temperatures.

On BWR-4 facilities the loop selection logic has been disabled and LPCIS flow is now directed to both recirculation loops, with discharge valves on both loops directed to shut automatically. This topic is directed toward these facilities and concerns surveillance requirements for the discharge valves and recirculation pumps bypass valves.

Conclusion

This topic does not apply to Phase II SEP facilities.

TOPIC IV-1A - Operation with less than all loops in service

SEP Plants Affected - PWR's and BWR's

DBEs Affected - Loss-of-Coolant Accident

Discussion

The majority of the presently operating BWRs and PWRs are designed to operate with less than full reactor coolant flow. If a PWR reactor coolant pump or a BWR recirculation pump becomes inoperative, the flow provided by the remaining loops is sufficient for steady state operation at a power level less than full power.

Plants authorized for long term operation with one reactor coolant pump out of service have submitted, and the staff has approved, the necessary ECCS, steady state, and transient calculations. The remaining PWR and BWR licensees have Technical Specifications which require a reactor shutdown within a fairly short time if one of the operating loops becomes inoperable (with the exception of two which are discussed below).

SEP APPLICABILITY

The docketed material for the 11 systematic evaluation program plants has been reviewed with respect to operation with less than all loops in service. One licensee (Dresden 2) has requested authorization to operate with less than all loops in service, the staff is reviewing the analyses submitted with the request and approval will be granted when the staff approves the analysis. Five facilities (Yankee Rowe, Millstone 1, Ginna, Palisades, and San Onofre) are not authorized to operate with less than all loops in service, Technical Specifications restrict this mode to a period of 24 hours at which time the facility must have the idle loop restored to service or shutdown. Three facilities (Connecticut Yankee, Oyster Creek, and Dresden 1) have had an analysis reviewed and approved by the staff which authorizes N-1 loop operation. Two facilities LACBWR and Big Rock Point) have had authorization to operate in the N-1 loop mode since they were licensed, however there is no supporting ECCS analysis to justify operation.

Conclusion

This topic is complete for all the SEP facilities with the exception of LACBWR and Big Rock Point, for the latter two if continued authorization is to be permitted an analysis will have to be submitted which describes the thermal-hydraulic conditions of N-1 loop operation during ECCS, steady state, and transient conditions. Until such an analysis is performed and approved. Operation with less than all loops in service should be restricted to a 24 hour period at which time the plant should be shutdown unless the idle loop has been made operable.

TOPIC IV-3 - BWR Jet Pump Operating Indications

SEP Plants Affected - Millstone 1, Dresden 2

DBEs Affected - Loss-of-Coolant Accident

Discussion

The capability to reflood the core may be precluded in the event of a LOCA if all jet pumps are not operable. A jet pump instrument sensing line failure could result in inaccurate core flow measurements or the inability to detect a jet pump failure.

This topic applies only to Dresden Unit 2 and Millstone Unit 1; therefore, it should be removed from the review list for the nine remaining SEP plants.

The review of BWR Jet Pump operating indications has not begun for the two applicable facilities. The SEP staff cannot proceed any further until additional information is obtained from the licensee.[1] I&E and NRR are working closely to determine the adequacy of present jet pump operability technical specifications. If resolution cannot be made prior to the start of the Design Basis Events (DBE's) assessments the topic will be reviewed considering the potential effects on related DBEs.

[1] Dresden 2 has not replied to request for information.

TOPIC V-9 - Reactor Core Isolation Cooling System

SEP Plants Affected - None

DBEs Affected - None

Discussion

This topic applies to the RCIC system, a BWR system consisting of a steam-driven turbine/pump combination, piping, valves, and controls. RCIC was designed to inject water into the vessel in the case of vessel isolation upon loss of both on-site and off-site A-C power. In the General Electric Standard Safety Analysis Report (GESSAR), GE took credit for RCIC as a backup for the High Pressure Coolant Injection System in Loss-of-Coolant Accident (LOCA) analyses for certain small breaks. The NRC concern is that the RCIC system may not have been classified as a safety system, although credit was assumed in the safety analyses.

Conclusion

This topic does not apply to the SEP BWRs (Oyster Creek, Millsone Unit No. 1, Dresden Unit Nos. 1 and 2, La Crosse and Big Rock Point) since none of these facilities has an RCIC system.

TOPIC VI-7.A.2 - Upper Plenum Injection

SEP Plants Affected - Ginna

DBEs Affected - Loss-of-Coolant Accident

Discussion

On May 1, 1978, NRC issued Amendment No. 19 to operating license No. DPR-18 The staff Safety Evaluation Report which supported the license amendment addressed the upper plenum injection topic.

Ginna submitted ECCS performance analyses for the Westinghouse and new Exxon Nuclear Company (ENC) fuels. The Westinghuse analysis was performed for Cycle 7 fuel which the staff believes is a conservative evaluation for the Westinghouse fuel during Cycle 8. The ENC analysis was performed for Cycle 8 using the ENC WREM-II ECCS evaluation model. The ENC evaluation model has been reviewed and approved conditionally by the NRC.

The staff has recently considered whether the Westinghouse generic evaluation adequately represented the flow characteristics of Westinghouse two loop units. The generic evaluation model assumes that all safety injection water is introduced directly into the lower plenum. For the two loop units, the safety injection water is injected into the upper plenum. Thus, the staff was concerned that the Westinghosue model did not consider interaction between UPI water and steam flow. After plant specific submittals by licensees operating two loop plants were reviewed, the staff concluded that the calculations provided by the licensees (with certain modifications to the staff's model) are acceptable on an interim basis for continued safe operation of Westinghouse two loop plants, while long term efforts continue for developing a model specifically treating UPI. For the Ginna plant the calculations which specifically considered UPI using the modified version of the staff model, resulted in a change of only 15°F from those using the generic model in which the UPI-core interaction was not specifically considered. In the interim, before these models are developed, Ginna has provided a modification to the current Westinghouse model which accounts for UPI-core interaction. It was demonstrated that the modification resulted in the increase of peak clad temperature by 15°F. Since for the Ginna plant both ENC WREM-II and Westinghouse models predict similar PCT's (1922°F for ENC WREM-II and 1957°F for Westinghouse) it can be expected that the UPI modification, when applied to the ENC WREM-II model, would allow about the same increase in PCT. The licensee has drawn a similar conclusion.

Conclusion

The staff has concluded that although the Westinghouse and Exxon two-loop generic-evaluation models should be changed to consider upper plenum injection (unless the plant is modified), analyses at the specific operating conditions applicable to the Ginna plant demonstrate that the effect of disregarding upper plenum injection interaction on refill and reflood conditions will not be significant (less than 20°F PCT). Therefore, the staff believes that, for the limited range to which the models do not deviate from the requirements of 10 CFR 50 Appendix K item I.D.3, and the calculations are acceptable.

TOPIC VI-7D - Long Term Cooling Pressure Failures

SEP Plants Affected - All PWRs

DBEs Affected - Loss-of-Coolant Accidents

Discussion

This issue was raised by Mr. Ronald M. Fluegge in an October 24, 1976 letter to then Chairman Rowden. It was later defined in the Office of Nuclear Reactor Regulation as follows:

> "The General Design Criteria require that the Emergency Core Cooling Systems (ECCS) shall be capable of providing adequate core cooling following a Loss of Coolant Accident, assuming a single failure in Emergency Core Cooling Systems. The staff assumes the single failure to be either an active failure during the injection phase, or an active or passive failure during the long-term recirculation phase. The physical layouts of engineered safety feature pumps and components on some pressurized water reactors makes them vulnerable to flooding that might result from large passive failures in system piping, although they are protected for more likely events, such as sudden seal failure. Large pipe ruptures are not required to be protected against because of their low probability during the ECCS recirculation mode."

As stated in the "NRR Reports on Allegations Made by Mr. Ronald M. Fluegge" (11/76):

"The General Design Criteria (Appendix A to 10 CFR 50) include the following footnote regarding single failures:"

'single failures of passive components in electrical systems should be assumed in designing against a single failure. The conditions under which a single failure of a passive component in a fluid system should be considered in designing the system against a single failure are under development.'

"Thus, the General Design Criteria do not provide an explicit requirement for the treatment of failures of passive components. Appendix K to 10 CFR 50 pertains to ECCS performance requirements and also does not provide explicit guidelines on the treatment of failures of passive components after a loss-of-coolant accident (LOCA). Present plants are reviewed, however, to assure that the plant arrangement and design features provide the necessary protection of essential systems and components (such as shutdown cooling and pressurized portions of emergency core cooling systems) due to potential piping failures as an initiating event (not concurrent with or consecutive to a LOCA).

Piping failures outside containment are postulated in accordance with Branch Technical Positions MEB 3-1 and APCSB 3-1 in the USNRC Standard Review Plan Section 3.6. Longitudinal or circumferential breaks in high energy fluid system piping or leakage-cracks in a moderate energy* fluid system piping are considered separately as a single postulated event occurring during normal plant conditions. The crack size assumed for a moderate energy pipe is equivalent to a slot of dimensions (1/2 x pipe thickness) x (1/2 x diameter). The plant must be designed such that the effects of such a postulated piping failure, including the environmental conditions resulting from the escape of container fluids, do not affect function of equipment essential to safe shutdown of the reactor.

With regard to postulation of failures in emergency core cooling systems subsequent to a loss-of-coolant accident, the USNRC Standard Review Plan on Emergency Core Cooling System (Section 6.3) provides additional guidance with the statement that: 'The ECCS should retain its capability to

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*Subsequent to a LOCA, all pipes of relevance are moderate energy pipes defined as a piping system carrying fluid at a temperature below 200°F and at a pressure below 275 psig.

cool the core in the event of a failure of any single active or passive failure during the long-term recirculation cooling phase following an accident.' Based on this guidance, the staff assures the ECCS design and layout satisfies the requirement for redundancy in such systems. The implementation of the passive failure statement does not require significant ruptures of moderate-energy piping subsequent to LOCA, as this combined event would be extremely unlikely. The more credible passive failure is at pump or valve seals, or measurement devices. The staff review of the effects of such a postulated leak rate includes consideration of: (1) the flow paths of the radioactive fluid through floor drains, sump pump discharge piping, and the auxiliary building; (2) the operation of the auxiliary systems that would receive this radioactive fluid; (3) the ability of the leakage detection system to detect the passive failure; and (4) the ability of the operator to isolate the ECCS passive failure.

Therefore, the ECCS passive failure criterion being implemented by the staff requires the consideration of additional leakage but not pipe breaks beyond the initiating LOCA. The basis for this is the staff's judgment that the probability of serious multiple pipe failures is sufficiently low that they need not be considered a design basis event, since when operating in the long-term recirculation mode, the ECCS is subjected to temperatures and pressures much less than those for which the system is designed. In addition, after long-term cooling has been initiated, the need for recirculation diminishes due to the decrease in available core decay heat. For example, for a 3500 MWt reactor, the amount of core decay heat which is being produced at the beginning of a normal shutdown is 203 MWt; after one week it has decreased to 13 MWt; and after eight weeks it is only 5.7 MWt. This means that significantly less coolant recirculation would be necessary after several weeks. The needed cooling water to prevent core overheating can be provided by the RHR system even considering leakage in the suction or discharge side of the piping. In addition, should recirculation cooling be temporarily interrupted at the end of one week, the core would be adequately cooled by the heat transfer effected by vessel boiloff. To maintain vessel level, a makeup of only about 100 gpm would be necessary."

CONCLUSIONS

We consider this issue to be closed. The effect of ECCS leakage will be assessed on the SEP plants during the DBE evaluation of LOCAs.

TOPIC VII-1.B - Trip Uncertainty and Setpoint Analysis Review of Operating Data Base

SEP Plants Affected - All SEP Plants

DBEs Affected - All transients

Discussion

This issue was identified in September 1976 by the Electrical, Instrumentation, and Control System Branch of the Division of Systems Safety, Office of Nuclear Reactor Regulation. The issue was defined as follows:

> "Inclusion is needed in Technical Specifications of instrument errors in determining instrument trip setpoints in relation to allowable values of th- measured variable. Operating and under review LWRs are likely to have trip setpoints set at unsafe levels. The margin between trip setpoints and "allowable values" has not been reviewed. Standard Technical Specifications for BWRs for instrance do not even define "allowable values." Numerical values listed in the Standard Technical Specifications for trip setpoints and "allowable values" are identical."

Staff consideration of instrument errors in the evaluation and approval of trip setpoints for safety related instrumentation has been performed by either of two methods. Operating licenses issued on plants after the Spring of 1977 contain trip setpoints in their technical specifications whose values have been evaluated and approved based upon consideration of the individual factors used to assure an adequate margin of safety for each safety related channel. The information upon which our evaluations are made is contained in the detailed Regulatory Positions of Regulatory

Guide 1.105, Revision 1, "Instrument Setpoints," reissued in November 1976, and in the NRC Standard Review Plan.

Most operating licenses issued prior to this were evaluated in the more generalized manner. In this approach, the discrete components of each of the margins to safety in trip setpoint values are not evaluated on an individual basis but are included in an overall safety margin. Each set point value is based upon the most limiting transient or postulated accident condition associated with the bases for that set point. The magnitude of this safety margin and the resulting set points are established to ensure that there is a low probability of the margin being removed by an adverse combination of instrument calibration error, instrument error and instrument drift. The staff believes that this method is acceptable.

The staff has,however, changed from a generalized method of trip setpoint evaluation to a method that considers each of the discrete factors that make up the margins of safety for each safety related instrumentation channel. Either method contains conservatism; however, the newer method allows the safety margin in the trip setpoints to be quantified in a more detailed manner. In addition, consideration of instrument error is explicit in the newer method, whereas previously it was an implicit assumption presumed to be considered as part of the overall margin, As new operating license reviews are completed, additional information will be included in FSARs relating to instrument drift and error because of the guidance now provided int eh NRC's Standard Review Plan and in Regulatory Guide 1.105. Accordingly, all Technical Specifications that are issued with new operating licenses after the Spring of 1977 will have the instrument drift allowance factored into the trip setpoint specifications. The staff is reviewing this more detailed information on instrument errors and draft to evaluate its impact, if any, upon the safety margins of the trip setpoints being used in older plants. Independent of the SEP, appropriate action will be taken to assure that the setpoitns in use retain an adequate degree of conservatism in maintaining safety margins as a result of this staff effort.

Conclusions

Adequate safety margins have been provided by the trip setpoints now in use for SEP plants, and this Topic does not warrant additional review apart from that for Topic XVI, Technical Specifications.

TOPIC XVII - Operational QA Program

SEP Plants Affected - All

DBEs Affected - All

Discussion

Since 1973 new guidance for operational quality assurance programs have been issued in the form of Regulatory Guides and WASH documents describing methods to comply with criteria of 10 CFR 50 Appendix B. The objective of this guidance is to assure that operation, maintenance, modifications and test activities do not degrade the capability of safety-related equipment to perform their intended function.

This topic has been completed for all SEP plants. Attached is a listing of the dates and specific reports containing the basis for their acceptance. Ten of the facilities were reviewed by the Quality Assurance Branch; the last (LACBWR) was reviewed by the Plant Systems Branch of DOR.

ATTACHMENT

DOCKET NO.	SEP PLANT	DOCUMENT
50-155	Big Rock Point	Topical Report Evaluation, 4/21/76
50-213	Connecticut Yankee	Letter, Switzer to Purple, 2/28/75
50-10	Dresden 1	Topical Report Evaluation, 4/78
50-237	Dresden 2	Topical Report Evaluation, 4/78
50-244	Ginna	Safety Evaluation Report, 9/30/74
50-409	LaCrosse	Memorandum, Eisenhut to Stello, 2/2/78
50-245	Millstone 1	Amendment 35 to SAR, 7/16/76
50-219	Oyster Creek	Safety Evaluation Report, 11/22/76
50-255	Palisades	Topical Report Evaluation, 4/21/76
50-206	San Onofre	Safety Evaluation Report, 4/8/75
50-29	Yankee Rowe	Topical Report Evaluation, 4/4/77