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JUN 20 1984

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FLitton
PNorion

Docket Nos. 50-266
and 50-301

Mr. C. W. Fay
Vice President-Nuclear Power
Wisconsin Electric Power Company
231 West Michigan Street
Milwaukee, Wisconsin 53201

Dear Mr. Fay:

As discussed with Mr. Krause of your staff, we would like to arrange for a site visit on July 10 and 11, 1984 at the Point Beach Nuclear Plant for NRC staff members and our contractor, Sandia National Laboratory. A list of attendees is enclosed.

The purpose of this site visit is to conduct a plant tour and obtain information related to resolution of Unresolved Safety Issue (USI) A-45 on Decay Heat Removal Capability. Enclosed is some background information on USI A-45 and a listing of topics which we would like to discuss with knowledgeable members of the plant staff.

We propose to meet with members of your staff on July 10 for a brief introductory meeting. After the meeting we would request a plant tour with special emphasis on those accessible areas containing equipment related to decay heat removal capability. We envision much of our information needs as listed in the enclosure will be satisfied during this plant tour.

We then would propose to discuss the remaining topics with members of your plant staff knowledgeable in plant systems and operational philosophy/strategy.

Our questions will be dealing with realistic operational responses to relatively high likelihood accident scenarios such as small break LOCA with additional questions on selected unique emergency situations such as fire, flood, etc. Your staff's responses should be viewed as providing information only and not as needed to meet any current requirements as it is quite likely that the scenarios proposed will exceed current design basis accident scenarios.

Your staff's responses will be annotated and typed and a copy will be provided for your review and concurrence prior to use as data for our study.

We would appreciate your consideration and cooperation in this matter. If we can be of assistance or if you have any questions, please contact the NRC project manager for your facilities, Mr. T. G. Colburn (301) 492-4709.

Sincerely,

Original signed by:

James R. Miller, Chief
Operating Reactors Branch #3
Division of Licensing

Enclosures:
1. List of Attendees
2. Request for Specific Plant Information

ORB#3:DL	ORB#3:DL	GIB	ORB#3:DL
PKreutzer	TColburn:dd	PNorion	JRMiller
6/ /84	6/18/84	6/19/84	6/ /84

Wisconsin Electric Power Company

cc:
Mr. Bruce Churchill, Esquire
Shaw, Pittman, Potts and Trowbridge
1800 M Street, N.W.
Washington, DC 20036

Mr. James J. Zach, Manager
Nuclear Operations
Wisconsin Electric Power Company
Point Beach Nuclear Plant
6610 Nuclear Road
Two Rivers, Wisconsin 54241

Mr. Gordon Blaha
Town Chairman
Town of Two Creeks
Route 3
Two Rivers, Wisconsin 54241

Ms. Kathleen M. Falk
General Counsel
Wisconsin Environmental Decade
114 N. Carroll Street
Madison, Wisconsin 53703

U.S. Environmental Protection Agency
Federal Activities Branch
Region V Office
ATTN: Regional Radiation
Representative
230 S. Dearborn Street
Chicago, IL 60604

Chairman
Public Service Commission
of Wisconsin
Hills Farms State Office Building
Madison, Wisconsin 53702

Regional Administrator
Nuclear Regulatory Commission,
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Office of Executive Director
for Operations
799 Roosevelt Road
Glen Ellyn, Illinois 60137

U.S. NRC Resident Inspectors Office
6612 Nuclear Road
Two Rivers, Wisconsin 54241

Enclosure 1

LIST OF ATTENDEES

Sandia National Laboratories

Wallis R. Cramond
David M. Ericson, Jr.
William J. Galyean
Gary A. Sanders

The above are holders of "Q" security clearance at Sandia National Laboratories

Nuclear Regulatory Commission

Andrew R. Marchese
Timothy G. Colburn

The above are holders of NRC "L" security clearance

REQUEST FOR SPECIFIC PLANT INFORMATION

Background

Task Action Plan A-45 was established to assess the safety adequacy of decay heat removal (DHR) in existing light water reactors, and to evaluate the value and impact of proposed alternative measures for improving the reliability of DHR. The assessment of the current safety adequacy of DHR systems is being performed through the use of both quantitative and qualitative screening criteria being developed for that purpose in this program, coupled with engineering analyses.

Complete modeling and quantitative value assessment on all existing plants would be difficult to accomplish in a time frame consistent with the TAP A-45 objectives, and within reasonable resources. Therefore, a method had to be developed to focus the investigation on the most significant problem areas. The method selected was a screening process in which each plant would be examined using a set of qualitative screening criteria developed specifically for that purpose and applied in a consistent fashion to all the plants. The only purpose of this screening was to identify, insofar as practicable, potential problems or inadequacies which could then be addressed further in the program to better assess their importance and effect upon decay heat removal. It should be emphasized that this screening was not intended to be a pass-fail evaluation for decay heat removal capabilities, but it is a tool to provide initial insights into the potential problems in a relative sense. As noted above, it is a technique for guiding research and the criteria should not be used for any other purpose. Those plants for which the initial screening suggested there may be problems are being subjected to further analysis to confirm or reject the initial fundings. This analysis includes probabilistic modeling where feasible and appropriate deterministic or qualitative engineering analysis where necessary. In those instances where decay heat removal problems are identified, fixes to existing systems or alternative measures will be proposed and evaluated using similar analysis techniques including a value/impact assessment.

Initial Screening Criteria

The screening criteria referred to as "probabilistic" are based on an extensive review of completed U.S. and foreign probabilistic risk assessments (PRAs), systems analyses (such as the auxiliary feedwater analyses and the station blackout studies), and current regulations to determine those system characteristics which most often contribute to the unavailability of DHR systems. This effort used the results of completed quantitative probabilistic analyses in an attempt to identify, in a qualitative fashion, potential DHR system vulnerabilities. In addition, licensee event reports, precursor to core melt studies and "lessons learned" reports were typical sources of information used to develop criteria for failure modes (such as random, operator, or common-mode failures) which could be quantified in a probabilistic model. Neither operating procedures nor test and maintenance procedures were included in this criteria development effort.

A key point which must be kept in mind regarding these criteria is that they are only a subset of all the design criteria standards and codes which should be satisfied for safe nuclear power plants. However, as noted above, these criteria reflect issues, problems, or deficiencies which have been shown from a variety of studies to be significant contributors to decay heat removal unavailability. Certainly some plants (especially the newer designs) may satisfy many of these criteria. However, for purposes of guiding or focusing the TAP A-45 program effort, it is important that all plants being considered be reviewed in a consistent manner against the same set of standards. These criteria provided a vehicle for that purpose.

In addition to the probabilistically based criteria discussed above, there is concern with the potential for nuclear reactor damage from external events such as meteorological phenomena, airplane crashes, dam failures, etc., which could result in a core melt. In addition to challenges from outside the plant, there are a number of potential internal threats which include, among others, sabotage, fire, internal missiles, and flooding. Most of these special emergencies have not been included in probabilistic risk assessments to date because it is difficult to quantify the likelihood of the event and/or the probability of such an event damaging a plant. Nonetheless, it is generally agreed that nuclear reactors may be vulnerable to these special emergencies depending on their geographic location and design configuration.

The literature review to identify potential DHR vulnerabilities to special emergencies included such sources as the various sabotage, fire protection, equipment qualification, seismic, and accident precursor studies sponsored by NRC as well as the SEP reviews, the Standard Review Plan, Appendix R reviews and other related documentation.

The key point is that literally hundreds of documents were reviewed to establish criteria by which the plants could be qualitatively evaluated or screened. However, to conduct such a screening, knowledge of the plant systems is required.

Plant Characterization

It was quickly established that direct contact with all the existing plants in order to obtain a broad range of specific information was not feasible. Therefore, only such publically available information as the Final Safety Analysis Report, NRC sponsored generic assessments, etc., were used. The plant characterization was systemized and standardized by using a set of questionnaires developed specifically for that purpose. Information was sought on front line and support systems required for decay heat removal. For example, auxiliary feedwater, high and low pressure coolant injection, residual heat removal, component cooling water, and emergency AC and DC power systems are among those examined. The questions asked pertained to capacities, redundancies, arrangements, control, etc.

In all, information was collected on 56 reactor sites. Several of the plants included in the SEP program were not included and some future plants that are very similar to existing units were likewise excluded. Where twin units by the same vendor are located at the same site, one unit was examined and shared components and differences were identified. The document sources used for this study have been issued since March 1979 and are reasonably current. However, in some cases, deficiencies identified during the qualitative screening using this plant data may have been corrected as a result of post-TMI directives.

Qualitative Screening

A qualitative screening was conducted using the criteria developed from reviews of a wide range of requirements and analyses and the publically available plant data. A short summary paper was prepared for each reactor examined. This paper summarized the compliances, non-compliances and information inadequacies for each of the criterion. This information was then used to generate a relative ranking of the plants. This ranking accounts for the relative potential contribution to risk of the identified non-compliances (in terms of high, medium, and low, based upon PRA experience) and accounts for unanswered questions or information inadequacies. A group of approximately 20 plants were identified which appeared to warrant further study; of these eight were selected as examples for the program.

Detailed Quantitative/Qualitative Analyses

The investigation is now at the point where more detailed analyses of eight individual plants are underway. These analyses will identify DHR deficiencies and potential fixes for the example plants, which then will be extrapolated to more generic statements of capabilities, requirements, and/or fixes. As noted above, the deficiencies identified in the qualitative screening may or may not exist depending upon the accuracy of our information, or if they do exist, they may or may not contribute to public risk. These are questions which can only be addressed by the detailed analyses, which to be accurate, requires input from the individual plants. At this point the analysts have examined a wide range of information, prior PRAs, regulations, Tech Specs, and generic studies, but questions remain.

Interaction with Utility Personnel

It should be understood that it is not the intent of the A-45 study to seek written responses from the utility personnel. Quite the contrary, we prefer to sit down with them and explore ideas and understandings in a very informal collegial atmosphere. Experience with the Interim Reliability Evaluation Program, the RSS Methodology Applications Program and the Risk Methods Integration and Evaluation Program has shown this to be a highly effective and non-threatening approach. This experience has also shown that most personnel are familiar enough with their plant and its characteristics that they can answer the questions of interest for us without significant study or research. In this approach we are not and will not ask them to certify their responses but to give us their best judgment. It is recognized that this is the only viable approach because many of our questions do go beyond design bases issues. They go beyond the existing requirements because that is the A-45 charter, and because we are attempting in this analysis to take maximum credit for existing plant capabilities even on non safety equipment.

I. Questions and Issues Related to Fault Tree Modeling

These are questions which arise in the detailed modeling activities and for which we have been unable to find suitable answers. In other instances the issues reflect judgments we have made and for which we seek utility comment as to reasonableness, accuracy, etc. Again, we are not seeking nor do we expect written responses only discussion.

1. Success Criteria - System level success criteria have been developed based upon Tech Specs and FSARs. These cover systems such as AFW, HPI, CCW, etc., but they are too extensive to completely writeout. We simply want to discuss them with the plant staff.

2. Emergency Procedures - We need to discuss system level procedures which lead to recovery of selected systems. We are in the process of identifying the specific events, but they will not be available prior to the visit.

3. P and IDs - We have been unable to establish system alignments for high pressure recirculation and need to obtain a copy of the applicable drawing. Also, on several P and IDs, there are notations "locked hand wheel" on motor operated valves (MOVs). How does this affect MOVs, are they still remotely operable? In other instances manual valves have a "locked open" notation; how often is the actual valve position checked?

4. Is "feed and bleed" a possible mode of operation at Point Beach? Is credit taken for it? Do procedures exist?

5. Is the CVCS required to prevent core melt under emergency conditions? Is CSIS required for response to transients and/or small LOCAs? This is an example of a question in which we seek knowledgeable comment, not detailed analyses.

6. Are boric acid tanks still considered part of ECCS? (Some plants are dropping/reducing BA concentrations.)

II. Questions and Issues Related to Special Emergencies

A. In doing an analysis for internal flooding it is convenient to define critical areas, which are areas in which redundant safety related equipment is susceptible to a common mode failure, as a result of a flood, which is identified as a potential precursor to core melt. Typical critical areas and associated equipment are listed:

Critical Area	Critical Components
Turbine building basement	Service water pumps and valves
	Main feedwater pumps
	Auxiliary feedwater pumps
	Air compressors
	Heater drain pumps
	Diesel generator control cabinet
	Boric acid pumps
	Condenser circulating water system
	Core spray pumps
PWR auxiliary building	High pressure coolant injection

Critical Area	Critical Components
Crib house	Service water pumps
	Circulating water pumps
	Fire water pumps
Control room	PORV control circuits
Containment building	Reactor coolant pumps
RHR spray equipment vault	RHR pumps and heat exchangers
	Safety injection pumps
	Containment spray pumps and HX
Primary auxiliary building	Charging pumps
	Component cooling pumps and HX
	Diesel generator heat exchangers

The following items relating to the above critical areas will be of interest during the plant visit. Again we do not expect prior written answers. Many of these can be addressed by a simple walking tour of the applicable areas. The reason for asking questions in this format is that based upon what is observed, experience and related analyses will then allow us to postulate reasonable/ potential scenarios that could be of concern.

1. Watertight Doors (WT) - Which rooms have WT doors? Are WT doors always closed? Are there WT doors between redundant areas?

2. Drains - Which rooms have drains? How large are they? Do they have covers (grills)? Are there interconnections? Check valves?

3. Dikes - Which rooms/equipments have them? How high are they?

4. Water Tanks - What are the capacities? Elevation within the building? Potential spill rate?

5. Room Penetrations (penetration here means a non-sealed opening) - Are there manholes? Size, number, administrative controls, destination?

- Are there vents? Size, number, destination?
- Are there cable penetrations? Size, number, locations, destination?

6. Piping - Number, location, size pressure?

7. Floor Area/Room Volume (see also fire issues).

8. Wall Construction (see also seismic issues).

9. Critical Equipment/Instrumentation/Control Cabinets - Proximity of redundant components? Elevation in the building? Spray guards? Minimum water depth to damage?

B. From the results of previous PRAs and fire studies there are a number of plant areas of particular interest in the fire analysis. These include the:

Control Room

Cable Spreading Rooms

Auxiliary Electric Equipment Rooms

Switchgear Rooms

Electrical Tunnels

Inner and Outer Cable Penetration Areas

Cable Vault Areas

Rooms with Redundant Pumps in them or a Pump Cabling for a Redundant Train

During the plant visit the following issues will be of specific interest in these areas, most of which as noted above, can be addressed by a simple examination as analysts tour the plant with the staff.

1. Cable Trays

 a. Stacking Arrangement (number of trays stacked vertically).

 b. Types of Trays (e.g., ladder, solid bottom, solid top, fire retardant wrappings employed).

 c. Routing of Redundant Trains Cables in Cable Trays.

d. Distance Cable Trays are from Floor, or Conversely from Roof of Room.

e. Percent Cable Fill in Tray.

2. Cables

   a. Routing of Safety Related Cables Through Areas.

   b. Method of Routing: Cable Tray, Conduit.

   c. Types of Cables Used

      1. Unqualified

      2. Qualified IEEE - 383 Type

      3. Brand (i.e., PVC, EPR/Hypalon, etc.)

3. Ventilation

   a. Designed Inlet Temperature.

   b. Inlet Flow Rate.

   c. Location, Size, Number of Ventilation Openings.

4. Detection/Suppression

   a. Types of Suppression Systems Used (e.g., dry pipe, wet pipe, pipe-action, deluge, etc.) and location.

   b. Suppression System Designed Fire Coverage Area.

5. Physical Parameters of Rooms

   a. Room Dimensions.

   b. Major Obstructions in Ceiling (i.e., support beams that extend down 18"-24" into ceiling area of room, thus creating "small" bays in ceiling).

   c. Openings in the Room (number, location, size, e.g., doors, grills, openings).

   d. Operating Temperature of Room

C. There are a number of items of interest during the visit to support the seismic analysis. They do not require prior answers, but the analyst will note the conditions. As with other special emergencies, actual plant conditions will establish what scenarios are reasonable. For example, if there are no un-reinforced walls, then equipment cannot be damaged by falling walls.

1. Presence of un-reinforced masonary walls near critical equipment, e.g., battery room enclosures, in diesel generator rooms, near AFWS pumps.

2. Motor control centers not bolted to floor or not tied together so they would "hammer" each other during an earthquake.

3. Suspended ceilings in control room or near emergency shutdown panels.

4. Pipe runs between auxiliary building and reactor building. Estimate span length between nearest anchors in each building.

5. Sketch layout of AFWS pipe feeding steam generators inside containment, showing anchor points, and estimate dimensions.

6. Examine battery racks and batteries for proper bolted supports.

7. Look for important AOV's to see that sufficient slack exists in air lines and that air tanks are properly bolted down.

8. Examine important MOV's for support of motor operators. Do electrical cables have sufficient slack?

9. Watch for cable tray penetrations into walls. Could cables shear if trays shift?

10. Check lube oil pumps on AFWS pumps. Are they tied down? Is there slack in feed lines and electrical cables. Are oil tanks tied down?

11. Is condensate storage tank bolted to concrete pad. Are other (secondary) storage tanks (e.g., demineralized water tank, pre-treated water tank, etc.) bolted down? Is pipe from CST anchored so relative motion of CST could cause problems? Is this pipe underground? Could this pipe fail at the building penetration?

12. Are there cranes (e.g., polar crane) which could jump rails and damage safety systems?

13. Examine service water pumps for seismic vulnerability. How is overhead roof configured? Is crib house embedded on all four sides?

14. Are reactor coolant pumps or steam generators pipe supported or beam/skirt supported?

<u>Summary</u>

A-45 is doing our extensive analysis using existing plants as examples. A significant volume of information has already been examined but questions remain which, in our view, can only be answered by interaction with the utility. As stated earlier, it is our belief that most of these questions or issues can be answered by discussion or inspection and do not require a prepared response by the licensee. The purpose of the individual plant DHR analyses is not to recommend specific modifications or requirements for that plant, but to form a source from which generic requirements can be developed that supplement or replace existing NRC requirements or regulations. The overall goal being a more cost effective approach to DHR.