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Docket No. 50-313

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Mr. John M. Griffin, Senior Vice President
of Energy Supply
Arkansas Power and Light Company
P. O. Box 551
Little Rock, Arkansas 72203

Dear Mr. Griffin:

As discussed with Mr. Larry Parscale of your staff, we would like to arrange for a three day site visit on November 13, 14 and 15, 1984 at Arkansas Nuclear One, Unit No. 1, for NRC staff members and our contractor, Sandia National Laboratory. A list of attendees is attached.

The purpose of this site visit is to obtain information related to the resolution of Unresolved Safety Issue (USI) A-45 on Shutdown Decay Heat Removal Requirements. The primary objectives of the USI A-45 program are to evaluate the safety adequacy of decay heat removal (DHR) systems in existing light water reactor (LWR) power plants and to assess the value and impact (or cost-benefit) of alternative measures for improving the overall reliability of the DHR function. The USI A-45 program is conducting probabilistic risk assessments and deterministic evaluations of those DHR systems and support systems required to achieve hot shutdown and cold shutdown conditions in both pressurized and boiling water reactors. Integrated systems analysis techniques are being used to asses the vulnerability of DHR systems to various internal and external events, including transients, small-break loss of coolant accidents, and special emergency challenges, such as fires, floods, earthquakes, and sabotage. State-of-the-art cost-benefit analysis techniques are being utilized to assess the net safety benefit of alternative measures to improve the overall DHR system reliability.

We propose to meet the first day of our visit to take your Health Physics Training to enable our party to enter areas which will be accessible to us. The second day we will meet with members of your staff to provide an overview of the A-45 program, including the scope, preliminary findings and the analytical model used in our analyses of the DHR and supporting systems at your plant. We also plan to describe the fault tree and event tree methodology being utilized in the USI A-45 program, including our plans for analyzing special emergency events. We want to verify specific plant system features, success criteria, operating procedures, and recovery actions. The third day of our visit will consist of a plant walk-through 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 the discussion and plant walk-through.

Our questions will deal with realistic operational and engineering responses to transients and small break LOCA accident scenarios, with additional questions on selected special emergency situations. Your responses should be viewed as providing information only and not as needed to meet any current requirements. It is quite likely that some of the scenarios proposed will exceed the current design basis of your plant. Your responses will be annotated and typed, and a copy of the responses will be provided for your review prior to use as information or data for our study.

We wish to emphasize that only several people need be involved from your organization. Based on our experience with other A-45 program plants visits, we believe that a couple of people from your engineering and design organization and a couple of people from your organizations staff would be sufficient. Your engineering people that are involved should be familiar with the mechanical, electrical and I&C functions, and the capabilities and performance of those systems required for decay heat removal in the event of transients and small break LOCAs. Also, it is important that they have knowledge of the physical layout of plant equipment. Your operations people that are involved should be familiar with emergency operating procedures in terms of responding to multiple failure events. Most of our questions will be answered either during the discussions with your staff or during the plant walk-through. We are not expecting a lot of preparatory or follow-up of work or extensive involvement on the part of your organization. There is no need for written responses on your part to the items in the enclosure. The list of information items in the enclosure will be used to guide and focus the discussions. Finally, we wish to emphasize that this is a fact finding effort on our part and is not associated with any licensing action.

We would appreciate your consideration and cooperation in this matter. If we can be of assistance or if you have any questions, please contact, Mr. Guy S. Vissing, the NRC Project Manager for your facility.

Sincerely,

"ORIGINAL SIGNED BY
JOHN F. STOLZ"

John F. Stolz, Chief
Operating Reactors Branch #4
Division of Licensing

Enclosures:
1. Attendees
2. Requirement for Plant-Specific Information

cc w/enclosures:
See next page

| | |
|---|---|
| ORB#4:DL | ORB#4:DL |
| GVissing;cf | JStolz |
| 10/11/84 | 10/1 /84 |

cc w/enclosure(s):

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Little Rock, Arkansas 72201

Enclosure 1

ATTENDEES FOR VISIT AT
ARKANSAS NUCLEAR ONE, UNIT NO. 1
NOVEMBER 13, 14, AND 15, 1984
CONCERNING INFORMATION RELATED TO RESOLUTION
OF USI A-45, SHUTDOWN DECAY HEAT REMOVAL REQUIREMENTS

| Name | Organization | Clearance |
|---|---|---|
| Wallis Cremond | Sandia | Q |
| Michael Bohn | Sandia | Q |
| Steve Hatch | Sandia | Q |
| Mark Jacobs | Sandia | Q |
| John Reed | JRB | None |
| Martin McCause | JRB | None |
| Andrew Marchese | NRC | L |
| Guy S. Vissing | NRC | L |
| Domenic Di Ianni | NRC | L |

ENCLOSURE 2

# INFORMATION REQUEST FOR PWR PLANTS

## 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 and questions 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 questions 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 vulnerabilities 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 screening should not be used for any other purpose. Those plants for which the initial screening suggested there <u>may be</u> vulnerabilities are being subjected to further analysis to confirm or reject the initial findings. This analysis includes probabilistic modeling where feasible and appropriate deterministic or qualitative engineering analysis where necessary. In those instances where decay heat removal vulnerabilities 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 Questions

The screening questions 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 screening questions 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 screening development effort.

A key point which must be kept in mind regarding these screening questions is that they are only based on a subset of all the design criteria standards and codes which should be satisfied for safe nuclear power plants. However, as noted above, these questions 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 concerns. 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 screening questions provided a vehicle for that purpose.

In addition to the probabilistically based questions 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 questions 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 generally have been issued since March 1979 and are reasonably current. However, it must be noted that detailed records were not kept of the data sources to indicate publication and/or revision dates. In some instances, modifications made post-TMI may not have been noted in the data sources, or the reviewers may not have had the most recent edition of the FSAR or similar material. Therefore, potential vulnerabilities identified during the qualitative screening discussed below may or may not exist. Furthermore, even if the identified vulnerabilities do exist, their potential contribution to actual core melt frequency has not been quantified. That step comes later in the analysis.

Qualitative Screening

A qualitative screening was conducted using the questions 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 questions. 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. Again, it must be noted that the qualitative screening is based upon information available to the reviewers at that time. The accuracy of the data cannot be absolutely guaranteed (as noted above), but a best effort was made to be as accurate as possible.

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 vulnerabilities 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 vulnerabilities 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. Prior to the plant visit the analysts will have examined a wide range of information, FSARs, prior PRAs, regulations, Tech Specs, and generic studies, but experience shows that questions will remain. Occasionally, questions may be raised to which the answers or the location of the answer may be obvious to someone very knowledgeable about the plant and its documentation. Unfortunately, it may not be that "obvious" to the analysts, therefore a question remains. Also, experience with related studies shows that it is frequently much more efficient to ask a question, than to spend hours searching for it in the plant docket files. At this point, external events analysts have not completed their reviews or may not yet have received all of the needed literature on a plant (e.g., Appendix R Reviews). However, many of these questions will be answered before the plant visit. Nonetheless, they are included here to make clear the type of information required and the knowledgeable people that we wish to talk to on a plant site visit.

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.

We would propose that the plant visit have the following agenda:

- Introduction of Sandia and utility attendees.

- Overview of the TAP A-45 program, scope, and preliminary findings - Sandia.

- Description of fault tree and event tree methodology being utilized - SAI.

- Verification of plant specific event trees, success criteria, and system schematics - SAI and utility.

- Discussion of specific plant DHR features, operating procedures, and recovery actions - all. (See questions following.)

- Presentation of Special Emergency Analysis Plan and Informational Requirements.

  Seismic - Sandia

  Fire/Internal Flood - Sandia

  Hurricane, Tornado, Lightning, External Flood - Sandia

- Tour of plant facilities - all.

- Concluding questions and wrap-up - all.

<u>Typical Plant Specific Questions</u>

I. Questions and Issues Related to Fault Tree Modeling

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 need to establish system alignments for many systems and need to obtain a copy of the applicable drawings. Also, often on 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 this plant? 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. Is the DC bus which is used for DG control also used by AC bus control power? If the DC control power bus is lost, will the AC bus also be lost?

7. During test or maintenance (T or M) procedures are written, check-off procedures used by maintenance personnel? Are components checked by an independent party to be in their correct position?

8. How often are safety related standby components checked to be in their correct position? e.g., manual valves in the HPI system, circuit breakers for the AFW pumps, etc.

9. How often are DC bus batteries tested?

10. What success criteria exist for systems shared by both units?

11. If the control room is uninhabitable, what remote shutdown panels must be manned to reach hot shutdown? How does one control the reactor coolant pumps or boration, or monitor primary system temperature or pressure?

12. What is the logic for transfer of AFW suction to secondary sources and how is it accomplished?

13. Are remote shutdown stations designed for single failure and do they operate on electrically separate channels? Are redundant and separate vital power supplies and cabling available?

14. Discuss the means of pump protection provided to prevent damage to pumps due to overheating, cavitation or loss of adequate pump suction fluid.

15. Provide discussions on the design of the atmospheric steam dump valves, including the motive power, air supply, control system, hand wheels and their accessibility.

16. Provide a discussion of the PORVs design, including the motive power, control system and supporting systems.

17. Can the steam generator safety valves be manually opened? Are there other steam release pathways that could be made available other than the ADVs, in the event of a sustained loss of offsite power?

18. Discuss capability of the systems used for DHR to be operated from the control room with either only onsite or only offsite power available assuming a single failure. Is operator action outside of the control room needed for plant cooldown?

19. Discuss tests or analyses performed for cooldown using natural circulation to confirm adequate mixing and cooldown.

20. Once main feedwater has been tripped, describe the procedure to restart it.

21. On loss of instrument air, will the MSIVs, steam generator ADVs, or AFW pump minimum flow recirculation valves fail closed?

22. Discuss the arrangement of which pump trains are actuated by which actuation trains.

23. Does initiation of the CSR require manual action?

II. Questions and Issues Related to Special Emergencies (See earlier comment relative to the fact that added information is being made available to analysts and that some of these questions may be answered before the plant visit.)

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

Safety Injection and Containment Spray Pumps

Component Cooling Water Pumps

Auxiliary Feedwater Pumps

Switchgear Room

Cable Spreading Room

Control Room

Service Water Pumps

During the plant visit, the following issues will be of specific interest in those areas, most of which 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.

B. 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 walls or ceilings near critical equipment, e.g., batteries, diesel generators, 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. Layout of AFWS pipe feeding steam generators inside containment, including anchor points, and dimensions.

6. Battery racks and batteries including bolted supports.

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

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

9. Cable trays penetrations into walls. Could cables shear if trays shift?

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

C. In regard to internal flooding, many of the areas of concern are the same as those considered during a fire analysis. During a plant tour we would wish to gain insights regarding the following design characteristics:

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?

10. Sump Pumps - Where are they located? How are they actuated?

D. Many plants are in locations where hurricanes, tornadoes, and lightning are of high probability. As such, information is needed for further analysis pertaining to:

1. Protection of water tanks and external piping.

2. Protection of the electric power source.

3. Protection of pumps and other components located in the plant yard.

4. Building design and protection from wind, rain, and missiles.

5. Lightning protection by ring conductors, down conductors, and radial coursing conductors.

6. Building design with grounded air terminals and metal structures.

7. Earth shield wires.

8. Lightning surge arrestors for main and startup station transformers.

9. Plant ground grid for all safety system electrical apparatus.

<u>Summary</u>

A-45 is doing an extensive analysis using existing plants as examples. A significant volume of information has already been examined but questions will remain which, in our view, can best 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 i not to recommend specific modifications or requirements for that plant, but to form a source of information from which a decision as to whether or not generic requirements should be developed to supplement or replace existing NRC requirements or regulations can be made. The overall goal is to develop a more cost effective approach to DHR.

ENCLOSURE 3

## INFORMATION REQUEST FOR BWR PLANTS

### 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 and questions 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 questions 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 vulnerabilities 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 screening should not be used for any other purpose. Those plants for which the initial screening suggested there <u>may be</u> vulnerabilities are being subjected to further analysis to confirm or reject the initial findings. This analysis includes probabilistic modeling where feasible and appropriate deterministic or qualitative engineering analysis where necessary. In those instances where decay heat removal vulnerabilities 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 Questions

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In addition to the probabilistically based questions 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.

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

<u>Plant Characterization</u>

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 publicly 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, 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 generally have been issued since March 1979 and are reasonably current. However, it must be noted that detailed records were not kept of the data sources to indicate publication and/or revision dates. In some instances, modifications made post-TMI may not have been noted in the data sources, or the reviewers may not have had the most recent edition of the FSAR or similar material. Therefore, potential vulnerabilities identified during the qualitative screening discussed below may or may not exist. Furthermore, even if the identified vulnerabilities do exist, their potential contribution to actual core melt frequency has not been quantified. That step comes later in the analysis.

## Qualitative Screening

A qualitative screening was conducted using the questions developed from reviews of a wide range of requirements and analyses and the publicly 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 questions. 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. Again, it must be noted that the qualitative screening is based upon information available to the reviewers at that time. The accuracy of the data cannot be absolutely guaranteed (as noted above), but a best effort was made to be as accurate as possible.

## 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 vulnerabilities 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 vulnerabilities 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. Prior to the plant visit the analysts will have examined a wide range of information, FSARs, prior PRAs, regulations, Technical Specifications, and generic studies, but experience shows that questions will remain. Occasionally, questions may be raised to which the answers or the location of the answer may be obvious to someone very knowledgeable about the plant and its documentation. Unfortunately, it may not be that "obvious" to the analysts, therefore a question remains. Also, experience with related studies shows that it is frequently much more efficient to ask a question, than to spend hours searching for it in the plant docket files. At this point, external events analysts have not completed their reviews or may not yet have received all of the needed literature on a plant (e.g., Appendix R Reviews). However, many of these questions will be answered before the plant visit. Nonetheless, they are included here to make clear the type of information required and the knowledgeable people that we wish to talk to on a plant site visit.

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.

We would propose that the plant visit have the following agenda:

- Introduction of Sandia and utility attendees.

- Overview of the TAP A-45 program, scope, and preliminary findings - Sandia.

- Description of fault tree and event tree methodology being utilized - SAI.

- Verification of plant specific event trees, success criteria, and system schematics - SAI and utility.

- Discussion of specific plant DHR features, operating procedures, and recovery actions - all. (See questions following.)

- Presentation of Special Emergency Analysis Plan and Informational Requirements.

  Seismic - Sandia

  Fire/Internal Flood - Sandia

  Hurricane, Tornado, Lightning, External Flood - Sandia

- Tour of plant facilities - all.

- Concluding questions and wrap-up - all.

Typical Plant Specific Questions

I. Questions and Issues Related to Fault Tree Modeling

1. Success Criteria - System level success criteria following small LOCAs and transients have been developed based on the FSAR and cover systems such as HPCI, LPCI, ADS, and core spray. It is our intent to make these criteria as realistic as possible. Therefore, we would like to discuss them with the plant staff. For example, what role might the RCIC system play in responding to an accident? Also, what credit can be given for cross-connecting systems between units?

2. Emergency Procedures - We would like to discuss the emergency procedures which may lead to recovery of unavailable systems during accidents.

3. System Configurations - We are using the FSAR and a limited set of P and IDs to construct our system fault tree models. In some cases it is not clear what the normal alignment for a system (or component) is. In order to model the plant correctly, we would like to verify some of our assumptions.

4. We are especially interested in finding out if any system and/or component modifications have been made to the plant which are not described in the FSAR. For example:

   a) Have you modified your safety/relief valves or their settings from what is described in the FSAR? If so, how?

   b) Have you modified your ADS logic from what is described in the FSAR? If so, how?

   Are there any other major system or component additions or modifications which are not described in the FSAR? ATWS-related modifications need not be included as ATWS is not being evaluated in this study.

5. Are there any heat balance, accident phenomenology, containment response, hazard and/or thermal-hydraulic calculations done for their plant which are not described in the FSAR. Failure mode effects analyses for the power conversion system and other systems outside of what is given in the FSAR would also be useful. Some examples of calculations which would be of interest are:

   a) Pump (RCIC, HPCI, etc) room heatup given no HVAC,

   b) Suppression pool heatup given a plant shutdown and no pool cooling,

   c) Suppression pool heatup given a plant shutdown, a stuck open relief valve and no pool cooling,

   d) Drywell heatup given drywell cooling fails,

   e) Pump (RCIC, HPCI, etc) seal degradation given no seal cooling,

   f) Effect of the Reactor Water Cleanup System on residual heat removal rates,

   g) Effect of the Control Rod Drive System on post-accident coolant injection requirements, and

   h) Plant fire, flood, lightning, wind, and/or seismic analyses.

6. Are there emergency procedures for venting the drywell or suppression pool during an accident?

7. Are there piping connections which would allow the fire protection system or service water systems to pump water directly into the core? If so, are there written procedures for performing these actions during an accident?

8. Can the recirculation pumps be isolated quickly following an unanticipated reactor shutdown?

9. Discuss the arrangement of which pump trains are actuated by which actuation trains. Is there a diverse initiation signal for RCIC (separate from HPCI)?

10. What alternate water makeup sources to the suppression pool which might be utilized during an accident such as the fuel pool, other unit's condensate storage tank, fire protection system, service water system, etc. Do procedures describe the required actions and when they would be performed?

11. Once feedwater has tripped, describe the procedure to restart it. Will any signals interlock the MSIVs closed? What support systems are needed to reopen the MSIVs?

12. How will the plant be affected by a loss of instrument air? In particular, what will the effect be on the safety/relief valves and MSIVs?

13. What is the general plant philosophy on component maintenance: maintenance on demand (no scheduled maintenance during power operation), preventive or scheduled, or other? During component test or maintenance, are written, checkoff procedures used and how are components verified or double checked afterwards?

14. If the control room is uninhabitable, what remote shutdown panels must be manned to reach hot shutdown? Are the remote shutdown panels designed for single failures and do they operate on electrically separate channels?

15. Discuss the means of protection provided to prevent damage to the HPCI, RCIC, LPCI, and core spray pumps from overheating, cavitation, or loss of adequate net positive suction head.

16. Can the failure of any vital AC or DC bus cause the plant to trip?

17. What is the station battery capacity (hours) given emergency loads following a loss of offsite power and all diesels (how long can RCIC and HPCI run on batteries alone?)? How often are the station batteries tested (load or voltage)?

II. Questions and Issues Related to Special Emergencies (See earlier comment relative to the fact that added information is being made available to analysts and that some of these questions may be answered before the plant visit.)

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

High Pressure, Low Pressure and Core Spray Pump Rooms

Residual Heat Removal Pump Room

Reactor Core Isolation Cooling Pump Room (if applicable)

Switchgear Room

Cable Spreading Room

Control Room

Service Water Pump Room

During the plant visit, the following issues will be of specific interest in those areas, most of which 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. Fire 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.

B. 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 walls or ceilings near critical equipment, e.g., batteries, diesel generators, RHR 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. Battery racks and batteries including bolted supports.

6. Important AOV's to see that sufficient slack exists in air lines and that air tanks are properly bolted down.

7. Important MOV's for support of motor operators. Do electrical cables have sufficient slack?

8. Cable trays penetrations into walls. Could cables shear if trays shift?

9. Lube oil pumps on critical pumps. Are they tied down? Is there slack in feed lines and electrical cables. Are oil tanks tied down?

10. 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?

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

C. In regard to internal flooding, many of the areas of concern are the same as those considered during a fire analysis. During a plant tour we would wish to gain insights regarding the following design characteristics:

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?

10. Sump Pumps - Where are they located? How are they actuated?

D. Many plants are in locations where hurricanes, tornadoes, and lightning are of high probability. As such, information is needed for further analysis pertaining to:

   1. Protection of water tanks and external piping.

   2. Protection of the electric power source.

   3. Protection of pumps and other components located in the plant yard.

   4. Building design and protection from wind, rain, and missiles.

   5. Lightning protection by ring conductors, down conductors, and radial coursing conductors.

   6. Building design with grounded air terminals and metal structures.

   7. Earth shield wires.

   8. Lightning surge arrestors for main and startup station transformers.

   9. Plant ground grid for all safety system electrical apparatus.

## Summary

A-45 is doing an extensive analysis using existing plants as examples. A significant volume of information has already been examined but questions will remain which, in our view, can best 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 of information from which a decision as to whether or not generic requirements should be developed to supplement or replace existing NRC requirements or regulations can be made. The overall goal is to develop a more cost effective approach to DHR.