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

November 9, 1978

MEMORANDUM FOR: Technical Activities Steering Committee Members

FROM: M. B. Aycock, Secretary
Technical Activities Steering Committee

SUBJECT: "UNRESOLVED SAFETY ISSUES"

The Advisory Group's recommendations regarding which issues meet the "Unresolved Safety Issues" definition adopted for trial use by the Steering Committee at the November 2, 1978 meeting are enclosed.

A meeting of the Steering Committee has been scheduled in P-422 at 1:00pm on Monday, November 13, 1978 to discuss the Advisory Group's recommendations and to finalize the list of issues that will be reported to Congress as "Unresolved Safety Issues."

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M. B. Aycock, Secretary
Technical Activities Steering Committee

Enclosure:
As stated

cc: H. Denton
E. Case

8012110 600

ADVISORY GROUP RECOMMENDATIONS REGARDING

"UNRESOLVED SAFETY ISSUES"

The following working definition of an unresolved safety issue was chosen for use in identifying the "Unresolved Safety issues" that should be reported to Congress pursuant to Section 210 of the Energy Reorganization Act of 1974.

> "An Unresolved Safety Issue is a matter affecting several nuclear power plants for which it is likely that actions may be needed to (1) compensate for a possible major reduction in the degree of protection of the public health and safety, or (2) provide a potentially significant decrease in risk to the public health and safety, when compared to that expected of plants currently being licensed for construction or operation."

A review of the generic issues in the NRR program against the definition of an "Unresolved Safety Issue" indicates that the generic tasks listed below1/ qualify for reporting to Congress.

| TASK NO. | TITLE |
|---|---|
| A-6 | Mark I Containment - Short-Term Program (1) |
| A-7 | Mark I Containment - Long-Term Program (1) |
| A-8 | Mark II Containment Program (1) |
| A-9 | Anticipated Transients Without Scram (1 or 2) |
| A-10 | BWR Nozzle Cracking (1) |
| A-11 | Reactor Vessel Materials Toughness (1) |
| A-24 | Environmental Qualification of Class IE Safety-Related Electrical Equipment (1) |
| A-26 | Reactor Vessel Pressure Transient Protection ( |
| B-18 | Vortex Suppression Requirements (1) |
| C-3 | Insulation Usage Inside Containment (1) |

In addition, we belive that the issue of "Pipe Cracks in Boiling Water Reactors" also qualifies as an "Unresolved Safety Issue," despite the fact that currently there is no approved generic Task Action Plan related to this issue.

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1/ The paranthetical numbers to the right of each issue indicate which part of the definition we believed the issue met (requested by Mr. Stello).

In addition, the Advisory Group identified the additional issues listed below as being potential "Unresolved Safety Issues." We believe that the Steering Committee should specifically focus on these issues in deciding which issues qualify for reporting to Congress.

| TASK NO. | TITLE |
|---|---|
| B-22 | LWR Fuel Program |
| B-57 | Station Blackout |
| A-15 | Primary Coolant System Decontamination and Steam Generator Chemical Cleaning |
| B-34 | Occupational Radiation Exposure Reduction |
| B-64 | Decommissioning of Reactors |

The procedure used by the Advisory Group to identify the "Unresolved Safety Issues" is described below. Also provided below are discussions providing the Advisory Group's rationale for including or excluding certain issues.

The information used by the Advisory Group in carrying out its review included the following:

1. Task Action Plans for Category A Tasks (including ALAB-444 type writeups)
2. Problem Descriptions of Category B, C and D Tasks (NUREG-0471)
3. RES Staff Risk-Based Evaluation of NRR Generic Issues
4. NRR Staff Comments on RES Staff Risk-Based Evaluation (Advisory Group members in each Division have copies of those comments received).
5. NRR Groupings of Generic Issues into Eight Groups by Type of Activity (provided to Steering Committee in Aycock memo of October 31, 1978).
6. List of Abnormal Occurrences Related to Power Reactors Reported to Congress to Date

The Advisory Group followed the procedural steps below in conducting its evaluation. These procedural steps were approved by the Steering Committee at its November 2, 1978 meeting.

Step 1

A number of issues can be eliminated from consideration because they are not directly related to safety. Others can be eliminated because they have only marginal safety significance. In recognition of this, the Advisory Group initially limited its consideration to those issues that appeared in NRR Group 1, 2 or 3 (i.e., are related to safety) and were identified by the RES staff as having risk significance (i.e., issues in either RES staff Category I - Potential High Risk Items or Category II - Potential Low Risk Items).

The issues identified for consideration in this step were compared to the definition of an "Unresolved Safety Issue." All items were assumed to qualify as "Unresolved Safety Issues" unless they could be eliminated based on other arguments. The generic issues considered in this step were the following:

| TASK NO. | TITLE |
|---|---|
| A-9 | Anticipated Transients Without Scram |
| A-6 | Mark I Short-Term Program |
| A-7 | Mark I Long-Term Program |
| A-8 | Mark II Program |
| A-39 | SRV Pool Dynamic Loads |
| B-55 | Reliability of Target Rock Relief Valves |
| A-17 | Systems Interactions |
| A-29 | Design Features to Control Sabotage |
| A-10 | BWR Nozzle Cracking |
| B-57 | Station Blackout |
| B-34 | Occupational Radiation Exposure Reduction |
| B-63 | Isolation of Low Pressure System from the Reactor Coolant Pressure Boundary |

| TASK NO. | TITLE |
|---|---|
| C-3 | Insulation Usage in Containment |
| A-3,A-4, A-5 | Westinghouse, CE and B&W Steam Generator Tube Integrity |
| A-1 | Water Hammer |
| A-12 | Fracture Toughness of Steam Generator and Reactor Coolant Pump Supports |
| A-2 | Asymmetric Blowdown Loads |
| A-30 | Adequacy of Safety Related DC Power Supplies |

Issues A-9, A-6, A-7, A-8, A-39, A-10, A-3, A-4, and A-5 could not be eliminated and accordingly, we recommend that they be adopted as "Unresolved Safety Issues."

Issue C-3, "Insulation Usage Inside Containment," (specifically the potential for sump blockage following a LOCA) was determined not to be risk significant for current plants, i.e., current requirements (Regulatory Guide 1.82 on sump design) reduce the risk significance considerably. However, the Advisory Group believes that the potential for sump blockage from post-LOCA debris could be a significant risk contributor for some operating plants and for this reason could qualify as an "Unresolved Safety Issue." We recommend that this issue be reported to the Congress as an "Unresolved Safety Issue" unless some rationale or additional information exists that we were not aware of that indicates that the potential risk significance is small for operating plants.

Issue B-34 is discussed under Step 3.

Brief discussions of why we determined that the remaining issues considered in Step 1 did not qualify as "Unresolved Safety Issues" are provided below:

## A-1 -- WATER HAMMER

This item was determined by the RES staff to have risk significance and was assigned to Group 1 by NRR.

Failure of a feedwater nozzle at Indian Point Unit 2 due to water hammer was reported as an Abnormal Occurrence in 1975. This water hammer issue, i.e., draining of the feedwater sparger, is now resolved and the new requirements are being implemented at operating reactors and in licensing reviews. In addition, the RES staff evaluation did not conclude that this type of water hammer was risk significant.

The risk significance of water hammer events identified by the RES staff was due to possible damage to the auxiliary feedwater pump steam turbines as a result of water hammer in the steam supply lines to the turbines.

The RES staff concluded that water hammer induced damage to the auxiliary feedwater system (AFW) could contribute roughly 16% to the core melt probability. We disagree with the RES staff bounding assumption that water hammer events in the steam supply lines would result in complete loss of function of the pumps, since this did not occur in the two events to date. Furthermore the experience with water hammer in AFW systems has improved and the expected frequency would be less than the $10^{-2}$/RY assumed by the RES staff. Further, this particular type of water hammer problem is not likely to result in any action by the staff other than to assure that appropriate administrative procedures are in place in operating plants for proper draining of the AFW steam supply lines. Accordingly, A-1, "Water Hammer" does not qualify as an "Unresolved Safety Issue."

## A-2 -- ASYMMETRIC BLOWDOWN LOADS

This item was determined by the RES staff to have risk significance and was assigned to Group 1 by NRR.

This item has been determined not to qualify as an "Unresolved Safety Issue" because its resolution would not compensate for a possible major reduction in the degree of protection of the public health and safety and would not provide a significant decrease in risk.

We believe the staff's approach to asymmetric blowdown loads is very conservative. This is based on the fact that there is a problem only if a very large essentially instantaneous break occurs in a very limited portion of the primary system piping. The RES staff evaluation indicated that this was a low risk item (5% of core melt probability and 3% of overall risk). The RES staff, however, utilized extremely conservative assumptions, e.g., given the probability of a large pipe break in the right location the resultant asymmetric loads cause ECCS failure (probability-unity). Because of the factors discussed above, i.e., the requirement for large break areas and essentially instantaneous break opening times, we believe the probability, and therefore the risk, is much lower. Accordingly, we do not believe this issue qualifies as an "Unresolved Safety Issue." This issue was not reported to Congress as an Abnormal Occurrence.

## A-12 -- FRACTURE TOUGHNESS OF STEAM GENERATOR AND REACTOR COOLANT PUMP SUPPORTS

This item was determined by the RES staff to have risk significance and was assigned to Group 1 by NRR.

This item has been determined not to qualify as an "Unresolved Safety Issue" because it does not represent a major reduction in the degree of protection of the public health and safety and would not provide a significant decrease in risk. The NRR staff concluded in the ALAB-444 write-up for this task that the likelihood of an initiating event was low and that, based upon a preliminary survey of operating reactors in May 1976, the support materials are expected to have adequate toughness.

The NRR staff disagrees with the RES staff's assumed 0.1 probability of steam generator or reactor coolant pump displacement causing a LOCA and a complete failure of ECC. More realistic assumptions would result in a much lower calculated risk. However, even with these conservative assumptions, the RES staff determined that the risk associated with the task was low. Accordingly, this issue does not qualify as an "Unresolved Safety Issue."

## A-17 -- SYSTEMS INTERACTIONS IN NUCLEAR POWER PLANTS

This item was determined by the RES staff to have risk significance and was assigned to Group 3 (confirmatory) by NRR.

This item has been determined not to qualify as an "Unresolved Safety Issue" because it is not likely to result in action by the staff to change current staff requirements and does not represent a possible major reduction in the degree of protection of the public health and safety. We agree with the RES staff conclusion that "not performing this task would not likely alter the overall risk predicted in the RSS" since systems interactions were considered in the plants studied in the RSS and other plants are not likely to be substantially different in the overall effect of systems interactions. We believe the likely interactions that have significant consequences are being addressed by both the designers and the staff and that the proposed study will confirm this judgment. Accordingly, A-17, Systems Interactions does not qualify as an "Unresolved Safety Issue."

## A-29 -- DESIGN FEATURES TO CONTROL SABOTAGE

This item was determined to be risk significant by the RES staff and was assigned to Group 2 by NRR.

The RES staff designated this issue as potentially risk significant on the basis that the probabilities of successful sabotage attempts cannot be accurately quantified and such probabilities could be significant in terms of risk. The implementation of 10 CFR Part 73.55 provides high assurance of protection of the health and safety of the public. Although Task A-29 may identify design concepts that could provide alternative or more effective means of achieving protection against industrial sabotage, the implementation of such design improvements is not necessary to provide adequate protection of nuclear power plants.

On this basis, this issue does not qualify as an "Unresolved Safety Issue." Further, this issue should probably be reassigned to NRR Group 6.

A-30 -- ADEQUACY OF SAFETY-RELATED DC POWER SUPPLIES

This item was determined by the RES staff to have risk significance and was assigned to Group 3 (confirmatory) by NRR.

We agree with the RES staff conclusion that this item contributes to less than 1% of the core melt probability. Nevertheless, the RES staff "conservatively grouped" this item in its Category II - Potential Low Risk Significance. Based on the minimal risk significance attributed to this issue by the RES staff and the arguments in NUREG-0305, "Technical Report on DC Power Supplies in Nuclear Power Plants," we believe this item is of negligible risk potential. Furthermore, we believe it is unlikely that the staff will take any action as a result of this item. Accordingly, A-30, "Adequacy of Safety Related DC Power Supplies" does not qualify as an "Unresolved Safety Issue."

## B-55 -- IMPROVED RELIABILITY OF TARGET-ROCK SAFETY-RELIEF VALVES

This item was determined by the RES staff to have risk significance and was assigned to Group 1 by NRR.

This item has been determined not to qualify as an "Unresolved Safety Issue" because it is not likely to result in action by the staff to change current safety requirements and does not represent a major reduction in the degree of protection of the public health and safety. In July 1978, NRR issued NUREG-0462, "Technical Report on Operating Experience with BWR Pressure Relief Systems," which concluded inadvertent operation of relief valves, even at a more frequent rate than actually experienced, would not have any significant effects on the reactor vessel or its internals and would not result in significant offsite radiological consequences. With respect to Mark I pressure suppression containments, it was concluded that no additional short-term action was required due to the substantial margins available in affected structures. Further, licensees, manufacturers and GE are working to improve valve reliability. Because valve reliability is improving, NRR action, at this time, is limited to an assessment of continuing operating experience with respect to relief valve failures.

We disagree with the RES staff development of the event tree for the spurious relief valve operation which could lead to core meltdown. The RES staff used 53 "failure to reseat" events to calculate a 0.25 per reactor year event frequency for stuck open valves. Actually, only 11 events related to high flow through the valves (i.e., plants exceeded technical specification cooldown rates) and no events resulted in high suppression pool temperature such that excessive containment loads would occur. Additionally, the event tree does not consider all of the methods available to the operator to control suppression pool temperature, i.e., the RES staff assumed 0.5 probability of high suppression pool temperature. Accordingly, the risk significance of this issue is at least an order of magnitude lower than the RES staff evaluation implies and is probably in the negligible risk category. Accordingly, we do not believe that this issue qualifies as an "Unresolved Safety Issue."

## B-57 -- STATION BLACKOUT

This item was determined by the RES staff to have risk significance and was assigned to Group 2 by NRR.

This task is concerned with whether the capability of mitigating the consequences of a total loss of AC power should be made a design basis requirement. At the present time, it is not.

The RES staff assigned this item as a potentially high risk item in consideration of those PWR plants "which either do not have steam-driven auxiliary feedwater capability or which require the principal AC power sources in order to effectively operate or initiate operation of such a system." Those plants which do have steam-driven auxiliary feedwater pumps do have the capability of accommodating a loss of all AC power for some period of time.

The RES staff study states that, "For plants already designed for station blackout, the risk and core melt significance of additional station blackout requirements is small." We agree with this assessment, although some benefits could accrue by specifying a period of time during which a plant should be designed to operate without AC power. Further, it is not clear that all PWRs now have the diverse feedwater capability necessary to assure that the contribution to risk of a station blackout is small.

The Advisory Group could not reach a consensus as to whether this item qualifies as an "Unresolved Safety Issue." We do feel that it should be specifically discussed by the Steering Committee to determine whether or not to include it.

## B-63 -- ISOLATION OF LOW PRESSURE SYSTEMS CONNECTED TO THE REACTOR COOLANT PRESSURE BOUNDARY

This item was determined to be risk significant by the RES staff and was assigned to Group 2 by NRR.

The RES staff evaluation indicates that because the failure of two check valves in series in the Low Pressure Injection System was a dominant contributor to risks associated with the PWR analyzed in the Reactor Safety Study, improved procedures for reducing the likelihood that such accidents will be dominant in other types of PWRs. On this basis the RES staff concluded that this issue was potentially risk significant. Current staff safety reviews of license applications for CPs and OLs are based on guidelines set forth in the Standard Review Plan, specifically Standard Review Plan 5.4.7. The staff positions in Standard Review Plan 5.4.7 acceptably resolve this concern for plants in the CP or OL review stage. However, since these guidelines were not available during the reviews of plants which are currently operating, Task B-63 was undertaken to review representative operating plants to assess the isolation capabilities of low pressure systems. This task has, in fact, been completed with a conclusion that there is adequate high pressure-low pressure isolation protection in operating reactors.

The NRR staff did not expect to identify any significant problems from this study as evidenced by Mr. Rusche's statement in a January 28, 1977 letter to ACRS prior to initiating the study: "We do not believe that this issue represents a significant safety problem for operating reactors at this time." For this reason, we do not believe that it was "likely that actions may be needed." Further, the study confirmed this initial judgment. Accordingly, B-63, "Isolation of Low Pressure Systems Connected to the Reactor Coolant Pressure Boundary" does not qualify as an "Unresolved Safety Issue.

STEP 2

In Step 2, the Advisory Group reviewed those issues in Groups 1, 2 and 3 that were not considered in Step 1, against the "Unresolved Safety Issue" definition to assure that utilizing the RES evaluation to determine risk significance had not caused us to inadvertently omit an issue that would qualify as an "Unresolved Safety Issue." In this Step, we identified three additional issues that qualify as "Unresolved Safety Issues." They are:

A-11 Reactor Vessel Materials Toughness

A-26 Reactor Vessel Pressure Transients

B-18 Vortex Suppression Requirements for Containment Sumps

In addition, we identified another issue (B-22, "LWR Fuel Program) that could meet the definition of an "Unresolved Safety Issue." Since the Advisory Group could not reach consensus on whether this issue should be included, we recommend that the Steering Committee consider it specifically. Discussions of each of these issues are provided below:

## A-11 REACTOR VESSEL TOUGHNESS

This item was determined by the RES staff to have negligible risk significance and was assigned to Group 1 by NRR.

This item has been determined to qualify as an "Unresolved Safety Issue: because reduction in the fracture toughness properties of the reactor vessel involves a major reduction in the degree of protection of the public health and safeth and it is likely that action will be needed to assure adequate fracture toughness in about 20 of the older vessels. We agree with the RES results that "this issue has negligible importance over about the next decade." However, they support our conclusion that this qualifies as an "Unresolved Safety Issue" in that they stated that "this could potentially be a long term concern having a significant risk impact". On this basis, we believe that this issue qualifies as an "Unresolved Safety Issue".

## A-26 REACTOR VESSEL PRESSURE TRANSIENT PROTECTION

This item was determined by the RES staff to have negligible risk significance and was assigned to Group 1 by NRR.

When the task was initially identified, problems had been observed in operating plants. The task was designed to identify the possible causes and to develop guidelines for design and operation to avoid future problems. At that time the staff believed that transient overpressure events represented possible major reduction in the degree of protection of the health and safety of the public. In addition, action was likely (action is being taken on operating plants and in CP and OL reviews). On this basis, we believe that this item qualifies as an "Unresolved Safety Issue." Further, since it was resolved during the last year, with issuance of NUREG-0224, "Reactor Vessel Pressure Transient Protection for Pressurized Water Reactors," it should be reported to Congress as a resolved issue. This issue was reported to Congress in an Abnormal Occurrence Report as another "Event of Interest."

## B-18 VORTEX SUPPRESSION REQUIREMENTS FOR CONTAINMENT SUMPS

This item was determined by the RES staff to have negligible risk significance and was assigned to Group 1 by NRR.

The RES staff concluded that since this item involved only large loss-of-coolant accidents, and vortex formation would not likely have a significant effect on the ability to cool the core, this task is of negligible risk potential. Since recirculation through the sump is required for all but the smallest LOCA, we believe the significance of this issue is greater than determined by the RES staff. Furthermore, the RES staff conclusion that, at worst, vortex formation would only result in self-limiting flow oscillations and no damage to the pumps, is too speculative to support their conclusions. Therefore, we believe vortex formation is potentially risk significant. It is not, however, a concern for plants currently under CP or OL review, in that applicants are required to show that vortices will not form in containment sumps prior to operation. It could, however, represent a significant risk contributor for some operating plants since vortex formation was not necessarily a consideration in the sump designs of older plants and sump testing has not been performed. Accordingly, we recommend that this issue be reported to Congress as an "Unresolved Safety Issue" unless some rationale or additional information exists, that we are not currently aware of, that indicates that the risk significance is small for operating plants.

## B-22 LWR FUEL PROGRAM

This item was determined by the RES staff to be "Not Directly Relevant to Risk" (Category IV) and was assigned to Group 2 by NRR. Although the RES staff considers this task to be a "rouine procedural effort," they acknowledged that work in this area "may indirectly have a unquantifiable impact on reducing occupational exposures." The purpose of this task is to improve fuel performance. Although the potential reduction in "risk" is small, the improvement in the integrity of the fuel cladding, which is the first fission product barrier, could be viewed as compensating for a major degradation in essential safety related equipment. Fuel failures in the LaCrosse reactor have been reported to Congress as an Abnormal Occurrence, which indicates that fuel failures have been determined to meet this "major degradation" standard before.

Step 3

In Step 3, the Advisory Group reviewed any issues identified as risk significant by the RES evaluation that are in NRR Groups 4-8 to assure that our initial consideration of issues in Groups 1, 2 and 3 only did not cause us to omit an issue that would qualify as an "Unresolved Safety Issue." The following issues were considered in this step.

| TASK NO. | TITLE |
|---|---|
| A-40 | Seismic Design Criteria |
| A-24 | Qualification of Class IE Safety-Related Equipment |
| A-15 | Decontamination |
| B-64 | Decommissioning of Reactors (or parts of NSSS) |
| B-30 | Design Basis Floods and Probability |

In this step we identified one additional issue that qualifies as an "Unresolved Safety Issue." The issue is A-24, "Environmental Qualification of Safety-Related Electrical Equipment."

In addition, there were two issues (A-15 and B-64) considered in this step and one issue (B-34) considered under Step 1 that involve the potential risk significance of occupational radiation exposures. The Advisory Group could not reach concensus on whether or not these issues should be reported as "Unresolved Safety Issues." Accordingly, we recommend that the Steering Committee consider the occupational radiation exposure issue specifically.

Each of these six issues is discussed below.

# A-40 -- SEISMIC DESIGN CRITERIA

This item was determined by the RES staff to be risk significant and assigned to Group 4 by NRR.

This task was determined to be risk significant by the RES staff because recent analyses (Cornell and Newmark and PGE for Diablo Canyon) predict seismically-induced accident sequences of significant probability. They further stated that although these analyses are still preliminary, it is apparent that seismically-induced accidents could be significant contributors to predicted LWR risks.

A long-term research program directly related to this concern has been initiated at LLL to evaluate more completely the seismically-induced core meltdown accidents. Task A-40 will not provide information directly related to this concern, i.e., it does not include plans to predict probabilities of core meltdown accidents.

We do believe that the preliminary studies indicate that a "possible major reduction in the degree of protection" exists. However, we believe it is premature to conclude on the basis of these preliminary studies, that it is "likely that actions may be needed." Accordingly, we do not believe that this issue qualifies as an "Unresolved Safety Issue" for reporting to Congress.

## B-34 -- OCCUPATIONAL RADIATION EXPOSURE REDUCTION

## A-15 -- DECONTAMINATION

## B-64 -- DECOMMISSIONING OF REACTOR

Tasks B-34, A-15 and B-64 were determined by the RES staff to have risk significance and were assigned to Groups 1, 6 and 6 respectively by NRR. The RES staff concluded that these issues had the potential for significant risk reduction because of their potential importance to reducing occupational radiation exposures. They noted that occupational exposures of station personnel was high compared to the expected value for the annual accident exposure to the public associated with the plants analyzed in the RSS.

In the case of Task A-15, the RES staff states that decontamination has a potential for reducing risk because of the high levels of personnel exposure currently being experienced. In the case of Task B-64 the logic used by the RES staff is somewhat different. They state that because of the high levels of exposure of station personnel that will result from decommissioning, any reduction in this exposure could constitute a significant reduction in risk.

We agree with the RES staff that relatively small improvements in the levels of exposure of station personnel could result in a significant reduction in the total number of man-rem received per reactor per year. However, we note that some means of reducing occupational exposures addressed by these generic tasks have the potential for reducing the current level of protection of the general public (e.g., chemical decontamination may cause piping degradation; waiving inspection requirements in order to avoid excessive occupational exposures may degrade component reliability). It is not possible at this time to make quantifiable trade-offs between decreased occupational exposures and increased potential exposures to the public resulting from accidents. (In fact one of the subtasks in Task B-34 that is currently being actively pursued is an attempt to develop criteria for making decisions regarding such trade-offs.)

Our regulatory approach to occupational exposures is to require that they be maintained as-low-as-reasonably achievable. We have certain requirements and a large amount of guidance to aid designers and operators in accomplishing this goal. These generic tasks will improve this guidance and will aid us in making regulatory decisions regarding the safety trade-offs that must be considered in our continuing responsibility to assure safe operation of plants.

## A-24 -- ENVIRONMENTAL QUALIFICATION OF CLASS IE SAFETY-RELATED ELECTRICAL EQUIPMENT

This item was determined by the RES staff to be risk significant and was assigned to Group 5 by NRR. The assignment to Group 5 was based on the fact that the Task Action Plan as currently written only involves receiving vendor and architect/engineer qualification programs on a generic basis so that the results could be used in case-by-case reviews.

The issue of environmental qualification has, in fact, expanded considerably beyond the initial scope of the Task Action Plan. The issue now includes such questions as (1) the degree to which electrical equipment used in older plants is capable of withstanding accident conditions, and (2) the adequacy of tests or analyses conducted for equipment in newer plants to qualify such equipment as capable of withstanding accident conditions.

The RES staff evaluation concluded that this issue was potentially risk significant because of the potential effect on the containment heat removal capability of plants relying, in part, on fan coolers for heat removal. Although we have not made a detailed study of other equipment, we believe there is other equipment inside containment whose failure, if not properly qualified, could be risk significant.

Further, we believe that with all of the staff activity on this issue, it is likely that additional action will be taken to compensate for reductions in the currently perceived degree of protection of the public. Accordingly, we believe this issue qualifies as an "Unresolved Safety Issue."

## B-30 -- DESIGN BASIS FLOODS AND PROBABILITY

This item was determined by the RES staff to be risk significant and was assigned to Group 6 by NRR.

The RES staff evaluation indicates that although application of existing methodology suggests that the probability of a flood-induced core melt accident at most sites is very low, detailed probabilistic analyses have not been performed to evaluate the likelihood of such an accident on a quantitative basis. Preliminary indications from flood-data analyses indicate that the risk associated with flooding may be significant for some sites.

The purpose of Task B-30 was to prepare a paper for presentation to the Advisory Committee on Reactor Safeguards (ACRS) detailing the bases for design basis flood events used by the NRR staff in case reviews. Additionally, the task was to address the possible use of probability estimates for the principal flood producing events. This task has been completed and a report to the ACRS was issued in July 1977. The report presents discussion and definitions of Probable Maximum Flood events which may be used as Design Basis Floods for review of nuclear power plants. It also presents arguments for continued use by the staff of a deterministic approach for identifying the Probable Maximum Flood events in preference to possible use of a probabilistic approach. As indicated in the report, the NRR staff does not feel that a probabilistic approach is appropriate at the present time because of the lack of confidence one has in estimates of extreme events using current techniques and statistics. Nonetheless, ongoing research efforts, being conducted by RES in response to a request by NRR, are aimed toward developing improved methodological techniques for the probabilistic analysis of flooding.

The deterministic approach for establishing design basis flood levels provides an adequately conservative basis for nuclear power plant design. Further, it is not likely that the staff will take action on specific plants based on the results of the study. For these reasons, this issue does not qualify as an "Unresolved Safety Issue."

Step 4

In Step 4, the Advisory Group reviewed those events that have been reported to Congress as Abnormal Occurrences (a listing is provided below) to determine if there were any additional generic issues that resulted from these reported events that would qualify as "Unresolved Safety Issues."

Pipe cracks at boiling water reactors have been reported as Abnormal Occurrences on two different occasions (in 1975 and in 1978). Based on this and the fact that the Pipe Crack Study Group has been reconstituted to study the latest information on this problem, we recommend that this issue be reported as an "Unresolved Safety Issue."

# ABNORMAL OCCURRENCE REPORTING
## NUCLEAR POWER PLANTS

| | Description | Plants |
|---|---|---|
| CY-75: | Steam Generator (S/G) Tube Failure | Point Beach 1 |
| | Fire in Electrical Cable Trays | Browns Ferry 1&2 |
| | Loss of Main Coolant Pump Seals | H.B. Robinson 2 |
| | Improper Control Rod Withdrawals | Dresden 2<br>Quad-Cities 1 |
| | Cracks in Pipes at Boiling Water Reactors | Various BWRs |
| | Fuel Channel Box Wear at Boiling Water Reactors | Various BWRs |
| | S/G Feedwater Flow Instability at Pressurized Water Reactors | Various BWRs |
| | Occupational Whole Body Overexposure | Zion 1 |
| | Occupational Whole Body Overexposure | Indian Point 2 |
| | Failure of Undervoltage Trip Logic and Consequent Loss of Safeguard AC Power | Millstone 2 |
| | Core Power Distribution Anomaly | St. Lucie 1 |
| | S/G Tube Integrity | Various PWRs |
| | Inadvertent Criticality | Millstone 1 |
| | Feedwater Nozzle Cracking in BWRs | Various BWRs |
| CY-77: | Breach of Physical Security System | Ft. St. Vrain |
| | Fuel Rod Failures | LaCrosse |
| | Management and Procedural Control Deficiencies | Zion 1&2 |
| | Generic Design Deficiency (Net Positive Suction Head) | North Anna 1&2<br>Surry 1&2<br>Beaver Valley 1 |
| | Environmental Qualifications Safety-Related Electrical Equipment Inside Containment | All Plants |
| CY-78: | Insulation Failures in Containment Electrical Penetrations | Millstone 2 |
| | Fuel Assembly Control Rod Guide Tube Integrity | Millstone 2 |
| | Overexposure of Two Radiation Protection Technicians | Trojan |
| | Degraded Primary Coolant Boundary in a BWR | Duane Arnold |

Events Reported as Other Events of Interest*

| | |
|---|---|
| Reactor Vessel Pressure Transients | Various PWRs |
| Unplanned Release of Radioactive Gaseous Material | Ft. St. Vrain |
| Burnable Poison Rod Assembly Failures | Crystal River 3 |
| Deviation from Seismic Design Criteria | Trojan |