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October 19, 1982

Mr. Darrell G. Eisenhut
Director, Division of Licensing
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Eisenhut:

Your September 9, 1982, letter to Mr. D.W. Edwards, Chairman of the Atomic Industrial Forum's Subcommittee on Backfit Requirements of the Committee on Reactor Licensing and Safety, requested our comments on the proposed topic definitions and associated cost estimate for the Systematic Evaluation Program (SEP), Phase III. We appreciate the opportunity to offer an integrated industry perspective on your proposals. We offer for your consideration both general and topic specific comments in Attachment A and B respectively. In summary, our comments are:

- o In most of the "Safety Significance" sections of the topic definitions there is no qualitative or quantitative discussion of the safety significance of the findings in Phase II, which supports inclusion of the topic in Phase III. It is difficult to comment on the relative merits of the topics' significance without first understanding the safety issues found in Phase II which elevated the topic to Phase III. The fundamental question which has not, to our knowledge, been addressed is, "What generic, significant safety issues were found in SEP Phase II, which have not or are not being addressed in other regulatory activities and, consequently, support the need for Phase III?";

- o The acceptance criteria are not clearly defined in the topic definition. By referencing the Standard Review Plans (SRP) in the Review Guidelines section of the topic definition, there is a strong implication that documentation of deviations from the SRP's by the licensees participating in SEP Phase III is necessary. On the surface, this appears to be a repeat of a previous attempt by the NRC to require all operating plants to participate in this type of exercise. We continue to believe that this approach has little, if any, safety merit;

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- Since there are many unknowns at this time regarding the total scope and implementation of the program, it is not possible to provide any reliable cost estimate for Phase III. However, the topics which required major manpower and financing in Phase II remain in the proposed Phase III. The seismic and high energy line break studies alone could potentially exceed the range of your total plant estimate, based on the Phase II experience;

- The "Safety Significance" section of the topic definitions should differentiate its findings between the five newer plants in Phase II (Palisades, Ginna, Dresden, Oyster Creek and Millstone) and the balance of the plants participating in Phase II. Since these five newer plants are more representative of the plants which would take part in the proposed Phase III program, this important distinction should be made by the NRC;

- Since the inception of SEP Phase II in November, 1977, there have been a considerable number of regulatory requirements imposed on the industry. They have addressed many of the same topics proposed for Phase III. Therefore, including these same issues in Phase III appears to constitute an unnecessary duplication of effort with little or no benefit to safety;

- Questions such as how many plants will be required to be a part of Phase III (initially and subsequently), the basis for plant selection and the criteria to be used to conclude the program need to be addressed prior to going forward with Phase III;

- As stated many times, due to the finite and limited manpower and resources available, it is crucial to compare carefully the safety effectiveness of any contemplated additional program with that of ongoing programs, since such additional activity can act to displace important ongoing activities.

In summary, based on our review of the topic definitions provided, the information provided by the SEP Owners' Group and the previous and current input provided by the industry, we continue to question the basis for initiating the proposed SEP Phase III program. We encourage a detailed NRC review of the need for this program in light of these comments and other ongoing regulatory programs and an analysis of the experience of SEP Phase II and the changes made through other regulatory initiatives since its inception in November, 1977.

Sincerely,

John P. Cagnetta

John P. Cagnetta
Chairman
Committee on Reactor
Licensing and Safety

JPC:tly
Enclosures

cc: William Dircks
Victor Stello
Harold Denton

ATTACHMENT A

GENERAL COMMENTS ON THE SYSTEMATIC EVALUATION PROGRAM PHASE III TOPIC DEFINITIONS

Introduction

The Systematic Evaluation Program, Phase II, was initiated in November, 1977. In the approximately five years since its initiation, major regulatory requirements and associated guidance documents have been imposed on the industry. Examples include, 10CFR50 Appendix E (Revised Emergency Planning Program), Appendix R (Fire Protection), 10CFR73 (Revised Physical Protection Program), TMI Lessons Learned (NUREG-0578, 0660, 0737, etc.), many I & E bulletins (Environmental Qualification of Electrical Equipment, Masonry Walls, etc.) and generic letters (Heavy Loads, etc.). Except for some isolated cases, there were no exemptions from these requirements for any plants, operating or under construction. In determining the need to go forward with SEP Phase III, these additional activities must be reviewed to determine if in fact there is a need for Phase III. Having briefly reviewed the suggested topic definitions, we conclude that the majority of these topics have been or are being addressed elsewhere. In the limited time available to us in preparing comments, we have attempted to highlight some of those activities related to each topic. This in no way implies that this is a comprehensive list of activities. However, it does highlight a need for careful scrutiny prior to going forward with Phase III.

Codes and Standards

A phrase used in more than one topic is, "Due to the evolutionary nature of structural codes and standards, it is possible that some aspects of the original plant design are significantly less conservative than that required by current codes." Each plant was designed to a specific set of codes and standards. The burden should not now be placed on each licensee to go through the latest codes and standards and make the determinations as outlined in Phase III.

Since Franklin Institute performed a comparative study of code changes for the NRC during Phase II, how this information will be used in Phase III should be included in the appropriate topic definitions.

While the NRC may require backfitting, including reanalysis, it may do so only when the NRC finds that such action will provide substantial, additional protection which is required for public health and safety (10CFR50.109). This needs to be addressed by the NRC.

Cost of SEP Phase III

There are many unknowns which need to be addressed before an estimate can be developed. For example:

- What is the role of the ACRS in Phase III?
- What is the role and relationship of IREP/NREP to the Phase III program? Will all plants in Phase III be required to do a PRA analysis?
- How many plants will be in Phase III?
- Will a systems interaction study be part of Phase III?
- What will the licensee be asked to provide using what acceptance criteria?
- What acceptance criteria from Phase II will be applicable in Phase III?
- What is the schedule for what has to be completed by when and when will it be finalized (cost escalation)?

As an example of the difference between Phase II and Phase III that must be considered when preparing the cost estimate for Phase III, the NRC contracted work themselves in Phase II to address specific topics. These contracts involved EG & G (piping analyses and electrical system reviews), Franklin Institute (structural and code reviews), Lawrence Livermore and TERA (seismic issues), Texas Tech University (Tornado and straight wind hazard probabilities) and the establishment of a Senior Seismic Review Team. If required in Phase III, the licensee would have to perform these analyses and establish any review teams.

Although it can not presently be quantified, several issues will have a direct effect on manpower and cost. For example, if a full PRA is required, this potentially represents well in excess of a million dollars. If additional seismic analyses are required to demonstrate compliance with the latest SRPs, this could represent several million dollars alone. Recognizing also that the additional analysis costs are a function of the acceptance criteria for Phase III, it is this detail that needs to be included in the topic definitions.

For example, in your cover letter you stated that, "The Phase III topic list would be further reduced for a Phase III plant on a plant-specific basis where the licensee can demonstrate ...." What type of demonstration is required by the licensee to remove a topic from the list? We encourage the use of "good engineering judgement" based on experience and existing data. This represents a much more efficient utilization of resources than being required to provide new analyses to demonstrate compliance with the latest regulatory criteria.

Schedule

There have been and continue to be discussions between the NRC and certain utilities regarding the concept of the "living schedule" for operating plants. The Phase III program contrasts markedly with this concept which is designed to levelize the work over a reasonable schedule to accommodate utility and NRC initiated programs. The Phase III program potentially represents an intensive effort requiring a considerable number of individuals with specific expertise, which may not be available due to other regulatory activities, for a short period of time. As has been learned since the TMI accident, this does not necessarily represent the most efficient and effective utilization of resources to gain the most benefit in overall plant safety.

Acceptance Criteria

In the Review Guidelines section of the topic definition, it is stated that "the review process is conducted in accordance with Standard Review Plan Sections 2.5.4", "the acceptance guidelines are stated in Appendix A to 10CFR Part 100 as implemented in Standard Review PLan Section 2.5.2", "the acceptance guidelines are described in NRC Standard Review Plan 3.5.1.4", etc. Although it is not explicitly stated, it appears that the NRC is proposing that licensees identify and justify design deviations from the acceptance criteria specified in the latest SRP's, regulatory guides etc. - acceptance criteria which did not exist at the time many of these plants were licensed - this is an approach which has little, if any safety merit, but could have large manpower resource impacts.

On February 3, 1982, the Committee to Review Generic Requirements (CRGR) discussed the final rule concerning documentation of differences from the current Standard Review Plan (SECY-81-648) for plants under construction. The following conclusions were reached by the CRGR:

- No clear safety benefits have been articulated for the proposed rule.
- The industry resources needed to comply with the rule would probably fall in the higher part of the 10-40 staff year per plant estimate range.
- Implementation of the rule would adversely impact other ongoing priority safety work and could thereby have an adverse impact on overall nuclear safety.

We continue to embrace the view that any program to document compliance with the SRPs does not represent the most effective expenditure of valuable NRC and manpower resources in terms of safety. Furthermore, it is our view that the NRC staff proposed program, which focuses on the highly detailed and constantly evolving staff interpretive guidance that potentially will consume hundreds of staff man-years and thousands of industry man-years, are, in fact, counterproductive to safety.

<u>Topic Definition Reviews</u>

Enclosed as Attachment B are specific comments on the individual topic definitions. Some of the generic comments are:

- The Safety Significance section in some cases simply restates the purpose of the topic;
- In some cases, the Introduction and Safety Significance sections are incompatible. For example, Topic 1.3, "Dam Integrity" gives as the objective, "to assure ··· uncontrolled releases of retained water are prevented." This is not reflected in the Safety Significance section;
- There is no qualitative or quantitative statement, based on Phase II reviews, of the significant findings and their importance to overall plant safety;
- When referring to the findings of Phase II, there needs to be a differentiation between the findings for the newer plants (Palisades, Ginna, Oyster Creek, Dresden and Millstone) and the balance of plants. Since these plants are more representative of those which would be part of Phase III, this is a significant distinction;

- Very seldom are the interpretative guidelines and alternate acceptable positions developed in Phase II referenced in the Review Guidelines for Phase III. An example of where it is referred to is for Topic 7.1.1, "Pipe Break Definition Criteria."

  > "The guidelines contained in Enclosure 2 to the December 4, 1981 letter, D. Crutchfield (NRC) to D. Hoffman (CPCo) may be used where separation or physical restraints for protection against dynamic effects of high energy piping failures are not practical."

  This type of guidance should be included in other topic definitions to assure that the experience in Phase II is factored into Phase III;

- For the procedural and hardware changes referred to in the Safety Significance section, there is no discussion of the significance of these changes. Since this is the basis for the topic being in Phase III, additional detail is needed to better understand the safety significant issue which should be addressed.

Also enclosed as Attachment C is a listing of each topic in Phase III, a cross reference to the appropriate Phase II topic and a listing of the proposed requirements which go beyond those used in Phase II. Reviewing this list, there are a significant number of regulatory documents referenced for Phase III which were not referenced in the Phase II topic definitions. This has a direct impact on both cost and schedule of the program without any clear understanding of the basis for the additional review references.

Overall Program

There are several questions which remain to be addressed when developing the overall SEP Phase III program prior to going forward with the program. The most fundamental of these is what is to be gained in overall enhancement of safety as a result of Phase III? What is the purpose of the program? If it is to demonstrate compliance with the regulations, is this the most effective and efficient use of the industry's resources to improve actual plant safety? Since Phase III is to be applied to the balance of the industry, how many plants will have to go through Phase III? How are they selected and on what basis is the program terminated?

There are two positive contributions which were utilized in Phase II which should be retained on a generic basis. The first is the concept of the integrated assessment of the issues to arrive at a solution which addressed several issues rather than the "piecemeal" approach of the past. The "living schedule" concept, if properly developed, could expand on the integrated assessment, levelize the regulatory work load without a compromise in safety and more efficiently and effectively utilize the NRC staff and industry resources. Therefore, the "living schedule" concept may be the concept mechanism to continue with the integrated assessment philosophy.

The second is the demonstrated need for strong project management. As stated in Carl Walske's letter of August 4, 1982 on this subject, it is our understanding that through the recent efforts of the SEP Branch in exercising project management control over the review branches in meeting schedules, addressing the issues unique to that facility and encouraging alternate solutions to meeting the regulatory requirements, the adversary climate was diminshed without compromising safety. This experience should be fostered generically. It should be noted that this is one of the key components in making the "living schedule" concept work.

ATTACHMENT B

SPECIFIC COMMENTS
ON THE
SYSTEMATIC EVALUATION PROGRAM
PHASE III TOPIC DEFINITIONS

Topic 1.1, Settlement of Foundations and Buried Equipment

The review guidelines reference regulatory guidance used by the NRC in reviewing a new plant for a construction permit or an operating license. As reflected in the Safety Significance portion of the topic, there has, in fact, been settlement problems experienced during plant construction. However, all the experience to date shows that this condition, if it exists will manifest itself early during construction or early in the operating life of the plant. It requires no more than a walk down to determine if the problem exists at an operating facility. The details of what was found in Phase II in order to justify going forward with this topic in Phase III are not found in the discussion of safety significance. It does not explicitly state the findings in Phase II and the significant safety concern to be addressed in Phase III. The statement that something "could" happen is insufficient.

Regarding a seismically induced settlement which may adversely affect the safety of the plant, it is our understanding that only one of the five newer plants in Phase II has a potential finding. It is presently not at all clear that there is a safety significance finding in this matter.

As part of the Review Guidelines section of the topic definition, it is stated that, the referenced documents, "describe a basis acceptable to the staff that may be used to implement the requirements of the criteria described in Section II above." Clarification as to just what the acceptance criteria is and what the licensee would have to demonstrate should be discussed in this section.

If this topic remains in Phase III, it is a candidate for deletion during the initial screening based on plant specific information.

Topic 1.2, Stability of Slopes

For the five newer plants, it is our understanding that no significant safety findings were found in the Phase II reviews of this topic. To our knowledge, this has not been a problem to date for any operating plants.

Again, the Review Guidelines section makes the "acceptable to staff" statement. This needs further clarification.

There is no discussion in the section on safety significance detailing what was found in Phase II which had sufficient significance to apply the topic to additional plants in Phase III. In fact, the Safety Significance Section is just a restatement of the Introduction section.

If this topic remains in Phase III, it is a candidate for deletion during the initial screening based on plant specific characteristics.

Topic 1.3, Dam Integrity

If the issue is as stated in the Introduction, it should be reflected in the Safety Significance section of the topic definition. Specifically, the objective of the topic is to "assure that adequate margins of safety are available under all loading conditions and uncontrolled releases of retained water are prevented." The Safety Significance section addresses flooding protection and cooling water supplies; it does not appear to address the topic of dam integrity specifically. This inconsistency should be corrected.

In the Safety Significance section of the topic, again there is no discussion of the significant issues found in Phase II and the basis, from a safety standpoint, of including this topic in Phase III. The general statement provided does not sufficiently identify the concern to warrant re-evaluating operating plants against the suggested regulatory guides.

Regarding dam models, it should be noted that the status of B-31, Dam Failure Models, is discussed in Stephen Hanauer's memorandum of February 18, 1982 in which is enclosed the Generic Issues Tracking System (GITS) Report. It states that, "The time failure models are unverified and the instantaneous failures model is possibly unnecessarily conservative. The issue is that significant over conservatism may require otherwise unwarranted flood protection or execute certain sites." Since the dam failure mechanisms are crucial to this concern, Phase III plants should not be required to perform analyses since the NRC recognizes the "significant over conservatism" of existing models.

If this topic remains in Phase III, it is a candidate for deletion during the initial screening based on plant specific characteristics.

Topic 1.4, Ground Motion

The Safety Significance section states that, "In order to ensure that a facility could withstand a very large earthquake required .··· (emphasis added)." This vague description of the earthquake to be accommodated by the licensee in Phase III is inadequate. As discussed in this section, "the free field seismic ground motion was conservatively specified in some cases." It is our understanding that extensive re-evaluation by the NRC, Lawrence Livermore and their consultants demonstrated that the original design basis response spectra for the five newer plants in Phase II were generally acceptable.

In the NRC's 1981 Annual Report it was stated that Generic Task A-40 "is intended to support re-evaluation of the seismic design of operating reactors and to develop requirements for licensing new plants." It also states that a NUREG report "Guidelines for Seismic Analysis and Review of Nuclear Power Plants" will be issued in 1982, presenting staff conclusions and recommendations. This information would be very helpful in determining whether this topic should be handled in Phase III or generically.

Based on our understanding of the Phase II findings in this area for the newer plants evaluated, and considering the fact that this topic is being addressed in other NRC programs, it should be deleted from Phase III.

Topic 1.5.1, Site Hydrologic Characteristics and Capability to Withstand Flooding

Although there were changes made to the Phase II plants in addressing this topic, the probability of the external event one has to design or, in this case, backfit a plant to, is the issue. On page 10 of the "Report on the Systematic Evaluation of Operating Facilities" which was issued to the Phase II plants in December, 1977, the threshold of "lesser safety significance" was discussed. It stated that, "Topics related to events which have an expected likelihood of occurrence on the order of 10^{-6} to 10^{-7} per year or, given the event occurs, results in consequences of only a small fraction of 10CFR Part 100 guidelines are considered to be of lesser safety significance." The need to design or backfit a plant to external events which have a 10^{-7} probability is, at best, questionable. If the acceptance criteria has changed, there needs to be a demonstration addressing the cost to the operating plants versus the benefit to the health and safety of the public of evaluating and potentially backfitting plants to meet specific regulatory guides or the latest Standard Review Plans review guidelines. For example, SRP 3.4.1 was extensively revised for Rev 2- July 1981. In the acceptance criteria portion of this SRP, "Acceptance is based on the design meeting the guidelines of Regulatory Guide 1.59 with regard to the methods utilized for establishing the probable maximum flood (PMF), probable maximum precipitation (PMP), seiche and other pertinent hydrologic considerations; and the guidelines of Regulatory Guide 1.102 regarding the means utilized for protection of SSC (structures, systems and components) important to safety from the effects of the PMF and PMP." The basis for requiring additional operating plants to go through this maze of addressing these low probability events is not discussed in the topic definition, and should be, before going forward with Phase III.

Topic 1.5.2, Site Severe Weather Characteristics

The comments on Topic 1.5.1 are applicable to this topic.

Topic 1.6.1, Industrial Hazards

The Safety Significance section of the topic is unnecessarily broad. It states that industrial hazards could affect the plant. It does not detail the industrial hazards identified in Phase II nor the potential need for protection from these hazards for the five newer plants. This section simply restates the introduction to the topic definition.

Also, under NUREG-0737 all operating plants addressed this topic at least partially, as part of the control room habitability review.

If this topic remains in Phase III, it is a candidate for deletion during the initial screening based on plant specific characteristics.

Topic 1.6.2, Tornado Missiles

Dr. S. Hanauer's memorandum of March 26, 1982 which forwarded a preliminary ranking of NRR generic safety issues, placed a low priority on Generic Issue A-38, "Tornado Missiles". In the discussion section of the document, pages 2-42 thru 2-44, the following points were made.

- "The probability of energetic tornado generated missiles would be less than $5x10^{-6}$/reactor year;"

- "The likelihood of this causing a core melt accident or any other significant radioactive release would be less than $5x10^{-9}$;"

- "No improvements in safety will result".

In the Introduction to the topic, it states that "plants reviewed prior to 1972 may not be adequately protected, <u>in particular those reviewed before 1968</u> when AEC criteria <u>on tornado protection were developed</u> (emphasis added)". Since the Phase II plants represent the majority of the plants in this category, this topic has been adequately addressed in Phase II and need not be considered further.

The section is not specific in detailing the safety significance of the findings and their implications regarding the overall safety of the plant. It may be a fair statement to say that, "a number of systems necessary for safe shutdown were identified in several plants which had not protection from tornado missiles", but the safety significance of the findings (inability to shutdown due to loss of both trains, etc.) needs to be explicitly addressed.

<u>Topic 1.6.3, Internally Generated Missiles</u>

No justification for including this topic is made other than the statement, "a number of systems necessary to safe shutdown were identified which had no protection from internally generated missiles." The safety significance of this finding needs additional detail.

This topic is also related to generic issues A-32 "Missile Effects", A-37 "Turbine Missiles", and B-68 "Pump Overspeed during LOCA". In each case the NRC has ranked these issues as low priority items. (Stephen H. Hanauer to Harold R. Denton "Preliminary Ranking of NRR Generic Safety Issues", dated March 26, 1982)

Based on the other regulatory activites and our understanding that the Phase II reviews did not identify any open issues of major safety significance, this topic should be deleted from Phase III.

Topic 1.6.4, Turbine Missiles

This topic is being addressed by ongoing industry activities. As a result of actual plant experience, the NRC issued I & E Notice 79-37 on turbine disc cracking and subsequently sent out generic letters to operating plants in May, 1980. An inspection program for detection of disc cracking is ongoing in the industry. Also, technical discussions between the NRC and licensees on turbine blade cracking are ongoing.

In the preliminary ranking of NRR Generic Safety Issues, page 2-41, the issue was determined to be of low priority. It was also noted that plant safety systems are redundant and should be capable of compensating for one being damaged by a turbine missile.

Based on the increased surveillance required of the operating plants, the results of the Phase II work and the fact that this topic is being addressed generically, this topic should be deleted from Phase III.

Topic 2.1, Classification of Structures

The safety significance section, of this topic states that "Due to the evolutionary nature of structural codes and standards, it is possible that some aspects of the original plant design are significally less conservative than that required by current codes (emphasis added)." This says there may be a situation where the design is significantly less conservative but does not state the significance based on actual findings. In fact, is the change in the code a significant change? What is its impact on overall plant safety? Is the change generic or specific to a class of plants?

The burden should be on the NRC to complete a review of the changes to the applicable codes, identify the significant changes and determine their significance to the overall safety of the plant. To require each operating plant to go through this exercise is unreasonable. It is our understanding that Franklin Institute performed this comparison for the NRC during Phase II. This effort is not referenced in the review guidelines for this topic. The use of this information in Phase III should be addressed.

This topic should be either deleted or made explicit as to the significant changes in the structural codes and standards to be addressed.

Topic 2.2, Severe Weather Effects on Structures

As discussed in our comments on Topic 1.5.2, the probability of the external event one has to analyze or, in this case, backfit a plant to, is the issue. For example, using data developed for the NRC in Phase II, if the acceptable hazard probability to be used in Phase III for this topic is 10^{-5}, the expected wind speed for a specific site is 195 mph; but if the probability is 10^{-7}, the expected wind speed is 337 mph. For operating plants, the basis for requiring a plant backfit due to a more stringent standard should be fully justified by the NRC before any program goes forward. This topic potentially represents major costs to the industry based on a low probability event.

A description of the generic issues and the safety significance of those issues to be addressed in Phase III needs to be discussed in the Safety Significance section of the topic definition.

Topic 2.3, Design Codes, Criteria and Load Combinations for Structures

The points made for Topic 2.1 are applicable to this topic. Although codes and standards change, that is not a sufficient basis for having a licensee go through this process. The experience of Phase II should be reflected in the discussion of this topic relative to the significant findings which should go forward in Phase III. If Franklin Institute's work performed for the NRC during Phase II is applicable, it should be referenced in the topic definitions.

Topic 2.4, Containment Design and Inspection

The technical specifications for the individual operating plants eliminate the need for this topic. I & E has a standard inspection module on this topic. The Regional inspectors presently review the inspection programs, test procedures and witness the tests if they so desire.

This topic should be deleted from Phase III.

Topic 2.5, Seismic Design of Structures, Systems and Components

Regarding the points made in the safety significance section, there are several other NRC activities which address this topic. They include:

- IE Bulletin 74-3, "Failure of Structural or Seismic Support Bolts on Class 1 Components"
- IE Bulletin 79-02, "Pipe Support Base Plate Designs Using Concrete Expansion Anchor Bolts"
- IE Bulletin 79-07 "Seismic Stress Analysis of Safety - Related Piping"
- IE Bulletin 79-14, "Seismic Analysis for As-Built Safety Related Piping Systems"
- IE Bulletin 80-11, "Masonry Wall Design"
- Information Notice 80-21, "Anchorage and Support of Safety Related Electrical Equipment"
- Generic Issue A-41, "Long Term Seismic Program"

These other activities should be reviewed prior to including this topic in Phase III.

Also, based on the experience of Phase II, the scope of this topic should be limited to those areas found to require additional review. This experience is not reflected in the topic definition.

Topic 3.1, RCPB Leakage Detection

This topic has been addressed in :

-NUREG-0313, Rev 1 (See comments on Topic 6.2)

-NUREG-0737, Item II.F.1, Attachment 5, "Containment Water Level Monitor"

-IE Bulletin 80-24, "Prevention of Damage Due to Water Leakage Inside Containment"

If the intention of the NRC is to have all operating plants be able to measure a one gpm leak within one hour, and have three separate systems-one being seismically qualified as reflected in a regulatory guide, the basis for requiring this backfit should be developed by the staff.

Topic 3.2, Reactor Core Isolation Cooling System

The Safety Significance section for this topic provides no basis for its inclusion in Phase III. It basically says that since there are no plants in Phase II with a RCIC and since it is relied upon for emergency core cooling and decay heat removal, the NRC will keep looking for a potential safety problem by having the industry do this.

In NUREG-0737, Item II.K.3.13, an analysis of the separation of High-Pressure Coolant Injection and Reactor Core Isolation Cooling System Initiation levels was submitted to the NRC by affected BWR licensees. This partially addresses the issue. Finally, if the RCIC were to be relied upon for emergency core cooling, its use would be evaluated in fuel reload reviews.

Therefore, this topic should be deleted from Phase III.

Topic 4.1, Classification and Design of Systems and Components

This is addressed in 10CFR50.55a, Codes and Standards, and the individual plant In-Service Inspection programs. The comments provided on Topic 2.1 are applicable to this topic.

The Safety Significance section is a restatement of the Introduction with no basis for its retention other than, "it is possible" that something has changed.

Based on existing programs (In-Service Inspection), existing regulatory requirements (10CFR50.55a) and the results of the SEP Phase II reviews which generally showed that code changes have not been significant due to other factors such as Section XI testing, this topic should be deleted from Phase III.

Topic 4.2.1, Shutdown Systems

This topic has or is being addressed in several regulatory requests and requirements.

These include:

- 10CFR50 Appendix R - The fire hazards analysis evaluates the ability to safely shutdown the plant and addresses alternative or dedicated shutdown capabilities.
- IE Bulletin 79-01B, "Environmental Qualification of Class IE Equipment"
- IE Bulletin 80-12, "Decay Heat Removal System Operability"
- NUREG-0737, Item I.C.1-Short Term Accident & Procedures Review. The rewriting of emergency procedures using the "symptom oriented" concept is ongoing. They address the use of non-safety related as well as safety related systems.
- Unresolved Safety Issue-A-45, "Shutdown Decay Heat Removal Requirements"
- Unresolved Safety Issue-A-31, "RHR Shutdown Requirements"

As discussed in the Safety Significance section of the topic, although no single shutdown method or set of systems met <u>all</u> of the review criteria, there is no discussion of the safety significance of this finding.

It should be noted that although several plants were involved in the Phase II program, they had to additionally respond to the referenced bulletins and Appendix R.

<u>Topic 4.2.2, Shutdown Electrical Instrumentation and Controls</u>

The comments on Topic 4.2.1 are applicable to this topic.

<u>Topic 4.3, Service and Cooling Water Systems</u>

The comments on Topic 4.2.1 are applicable to this topic. The two major areas of review are 10CFR50 Appendix R and NUREG-0737.

<u>Topic 4.4, Ventilation Systems</u>

In the preliminary ranking of the NRR generic safety issues, Generic Issue #1, "Failure in Air-Monitoring, Air-Cleaning, and Ventilation Systems", was given a low priority. Also referenced was NUREG-0572.

This topic has been addressed in 10CFR50 Appendix R, the equipment qualification program and NUREG-0737, Control Room Habitability.

Although deviations from the acceptance criteria were found in Phase II which resulted in the need for hardware modifications, the safety significance of these findings is not addressed in the Safety Significance section. Since the Phase II plants were not licensed to the most recent regulatory guidance, finding deviations from the latest regulatory guidance is not surprising.

Unless a significant safety issue can be identified which will not be covered by these other regulatory activities, this topic should be deleted from Phase III.

Topic 4.5, Spent Fuel Storage

This topic will be handled in the individual review and licensing proceedings required if a utility plans to expand their spent fuel pool. Staff review in the course of this proceeding would necessarily address loss of cooling events, reactivity shutdown margin, structural capability and shielding of the suggested spent fuel expansion. Therefore, this topic should be deleted from Phase III.

Topic 4.6, Isolation of High and Low Pressure Systems

This topic was addressed when the utilities had to address "Event V". Licensees were requested to describe the valve configuration at their plants and indicate if an "Event V" isolation valve configuration existed within the Class I boundary of the high pressure piping connecting Primary Coolant System piping to low pressure system piping.

The safety significance based on the findings in Phase II is not adequately discussed in the Safety Significance section.

Topic 4.7.1, Automatic ECCS Switchover

Certain aspects of this topic were addressed as part of NUREG-0737, I.C.1, "Guidance for the Evaluation and Development of Procedures for Transients and Accidents" and the associated emergency procedures which were written. Control room design reviews also address operator action. This topic is part of these other activities.

The Safety Significance section gives no technical basis for including this topic based on the findings of Phase II.

Topic 4.7.2, Recirculation Loop Isolation (Non-Jet Pump BWR)

The statement in the Safety Significance section of the topic definition that, "however, future SEP plants may be affected by this issue" is not a basis at all. It is conjecture without any technical basis to justify the need to pursue it.In fact, there are no BWR, non-jet pump plants, with Low Pressure Coolant Injection systems.

This topic should be deleted from Phase III.

Topic 4.8.1, Emergency AC Power Systems

There have been several regulatory activities in this area. These include:

- o IE Bulletins 75-04, 04A, 04B, "Cable Fire at Browns Ferry"
- o IE Bulletin 79-01B, "Environmental Qualification of Class IE Equipment"
- o IE Bulletin 79-23, "Potential Failure of Emergency Diesel Generator Field Exciter Transformer"
- o IE Bulletin 79-27, "Loss of Non-Class 1E Instrumentation and Control Power System Bus During Operation"
- o IE Bulletin 80-15, "Possible Loss of Emergency Notification Systems (ENS) with Loss of Offsite Power"
- o 10CFR 50 Appendix R, "Fire Protection"
- o NUREG-0737, reevaluation of emergency procedures.

In addition, in the preliminary ranking of NRR generic safety issues, generic issues #17 and #26 and Unresolved Safety Issue A-25 and B-56 were discussed. The conclusions were:

- o Generic Issue #17, "Loss of Offsite Power Subsequent to a LOCA" - "This issue was determined to be of low priority after preliminary screening of all generic issues was conducted."
- o Generic Issue #26, "Diesel Generator Loading Problems Related to SIS Reset on Loss of Offsite Power" - Conclusion - "This issue is deemed to have a very low priority because of the low probability of the specific failure sequence LOCA, erroneous early reset, loss of offsite power, and delayed restoration of SI function. This determination is documented in NUREG-0138."
- o Unresolved Safety Issues A-25, "Nonsafety Loads on Class IE Power Sources" - "This issue was determined to be of medium priority after preliminary screening of all generic issues was conducted."
- o Unresolved Safety Issue B-56, "Diesel Reliability" - "This issue was determined to be of high priority after preliminary screening of all generic issues was conducted."

Furthermore, in addressing Unresolved Safety Issue A-44, a request for diesel generator reliability data was sent to all licensees in July, 1981. This data has been received and is presently being reviewed by the NRC and their contractor. There have also been generic letters sent to the licensees regarding station blackout and degraded grid voltage review.

The statement in the Safety Significance section that, "several plant electrical distribution systems and onsite generator systems failed to satisfy the acceptance criteria" is a meaningless statement without a discussion of the safety significance of the finding.

Based on the regulatory activities on this subject, this topic should be deleted from Phase III.

Topic 4.8.2, Emergency DC Power Systems

This topic has been addressed in 10CFR50 Appendix R (associated circuits, shutdown capability, etc.) and environmental qualification of electrical equipment. The following regulatory activities are also related to this topic:

- IE Bulletin 79-27, "Loss of Non-Class IE Instrumentation and Control Power System Bus During Operation"
- IE Bulletin 80-12, "Decay Heat Removal System Operability"
- Unresolved Safety Issue A-30, "Adequacy of Safety-Related DC Power Supplies"

The significance of the findings from Phase II are not discussed in the topic definition.

Based on the regulatory activities on this subject this topic should be deleted from Phase III.

Topic 4.8.3, Swing Bus Design (BWR-4)

Since no BWR-4's were reviewed in Phase II, this topic should have been addressed in the review of that license application. It appears that including this topic gives the NRC staff a mechanism to re-review BWR-4 plants with operating licenses in case they missed something the first time around. It is also potentially covered in Topics 4.8.1 and 4.8.2.

Therefore, since there is no stated safety significant finding, this topic should be deleted.

Topic 4.9, Shared Systems

The Safety Significance section of the topic definition does not address the significant issues found in Phase II.

Also in the Review Guidelines section it states that the "acceptance guidelines ··· are presented in many Standard Review Plan Sections ··· (emphasis added)." This is a very broad definition of the acceptance guidelines and needs to be more explicit.

Topic 5.1, Reactor Protection System and Engineered Safety Feature Systems Isolation

On page 5 of enclosure 2 of the September 9, 1982, letter from Darrell Eisenhut to Don Edwards, it states that, "In previous risk assessments it has shown that the failure to generate initiating signals has not contributed to risk. Additionally in evaluation of systems where isolation was not present for certain signals (Oyster Creek SEP) it was shown that due to the presence of backup signals the lack of isolation did not contribute to the failure rate of initiation systems. The reason for this is that multiple isolation failures would be required to render the initiation system totally inoperable. However, this issue would be of high importance if a common fault could disable the protection system function (emphasis added)."

Based on previous risk assessments performed, the topic should be deleted from Phase III.

Topic 5.2, RPS and ESF Testing

As stated in one of the draft Draft Integrated Plant Safety Assessment sections of a Phase II plant, "The staff performed a limited PRA of the issue ··· to estimate the improvements in overall safety if additional response-time testing was required. The results of this PRA indicated that additional response-time testing has low safety significance. This occurs because response-time testing is concerned with events on the order of seconds. The IREP studies (Millstone Unit 1, Browns Ferry (NUREG/CR-2802), Arkansas Nuclear One, Calvert Cliffs, and Crystal River Unit 3) have shown that response times of 20 to 40 minutes are sufficient for emergency core cooling system actuation for both BWR and pressurized water reactors, ···. Therefore, it is the staff's judgement that response-time testing of instrumentation, other than that already required by ··· Technical Specifications, should not be required."

Also, component and system test requirements for the Reactor Protection Systems and Engineered Safety Features, including the frequency and scope of the periodic testing, is explicitly identified in the Technical Specifications for operating plants.

Based on this and the status of the five newer plants in SEP Phase II, this topic should be deleted from Phase III.

Topic 6.1, Organic Materials

There is no discussion on the safety significance of the findings in Phase II for the newer plants.

Regarding Unresolved Safety Issue A-43, "Containment Emergency Sump Performance, " there have been several activities on this subject including sump hydraulic tests, plant insulation surveys, and pump air and particulate ingestion effects. Draft NUREG-0897 has also been prepared.

Based on this regulatory activity and since the Phase II (newer plants) met the acceptance criteria for this topic, this topic should be deleted from Phase III.

Topic 6.2, RCS Water Purity (BWR)

The subject of intergranular stress corrosion cracking of austenitic stainless steels in BWRs has been addressed in Unresolved Safety Issue A-42, "Pipe Cracks in Boiling Water Reactors." The 1978 Study Group completed its evaluation and published NUREG-0531, February 1979. The findings of this investigation reaffirmed the conclusions and recommendations reached in NUREG-75/067 and the implementation document, NUREG-0313, is still valid. There has been and continues to be extensive activity in this area. In February, 1981, NUREG-0313, Rev. 1, was issued to all BWR operating licensees and plants under construction. By July 1, 1981 the licensees were to provide their program for replacement of certain lines and welds, the licensees program for augmented inservice inspection and the program for improving the water chemistry environment and incorporation of adequate leak detection capability. Therefore, based on this effort, this topic should be deleted from Phase III.

Topic 7.1.1, Pipe Break Definition Criteria

This topic has been or is being addressed in several areas. For example:

- IE Bulletin 76-04, "Cracks in Cold Worked Piping at BWR's"
- IE Bulletin 79-02, "Pipe Support Base Plate Designs Using Concrete Expansion Anchor Bolts"
- IE Bulletin 79-07, "Seismic Stress Analysis of Safety Related Piping"
- IE Bulletin 79-13, "Cracking in Feedwater System Piping"
- IE Bulletin 79-14, "Seismic Analysis for As-Built Safety-Related Piping Systems"
- IE Bulletin 79-17, "Pipe Cracks in Stagnant Borated Water Systems at PWR Plants"
- IE Bulletin 80-11, "Masonry Wall Design"

- IE Bulletin 80-13, "Cracking in Core Spray Spargers"
- IE Bulletin 81-01, "Surveillance of Mechanical Snubbers"
- USI A-14, "Flaw Detection"
- USI A-18, "Pipe Rupture Design Criteria"

INACTIVE

- USI B-6, "Loads, Load Combinations, Stress Limits"

The studies on high energy lines/systems interactions performed for all operating plants have also addressed this issue.

Based on the other regulatory activities performed or on-going in this area, this topic should be deleted from Phase III.

Topic 7.1.2, Pipe Break Effects on Systems and Components

See comments on Topic 7.1.1. The most significant result of Phase II on this issue was the acceptance by the NRC of the "leak before break" concept as on acceptable position to take.

Since this concept reduces the number of pipe breaks which must be assumed and the issue of pipe breaks is being or has been exhaustively considered (see comments on Topic 7.1.1), this topic should be deleted from Phase III.

Topic 7.2, Containment Isolation

In NUREG-0578, Item 2.1.4, "Containment Isolation Provisions for PWRs and BWRs", the recommendation was;

> "Provide containment isolation on diverse signals in conformance with Section 6.2.4 of the Standard Review Plan, review isolation provisions for non-essential systems and revise as necessary, and modify containment isolation designs as necessary to eliminate the potential for inadvertent reopening upon reset of the isolation signal."

During the review of the NUREG-0578 and NUREG-0737 requirements, teams went to operating plants to review in detail the containment isolation capability of the facilities. Also, the extensive leak reduction program required by NUREG-0737 addressed this area.

The significant deviations referred to in the topic definition section needs to be expanded giving the impact on the overall safety of the plant.

Topic 7.3.1, RCS Specific Activity Limits

This topic is covered in 10CFR50 Appendix I, and associated Technical Specifications, the Steam Generator Program and the Technical Specifications for the individual operating plants.

Regarding the statement in the introduction that, "The scope of the topic will be to examine the plant technical specifications to determine if they comply with the appropriate STS", the Committee to Review Generic Requirements (CRGR) addressed this topic at their meeting on Wednesday, February 3, 1982. As reflected in the minutes of CRGR meeting #7, "The Committee suggested that NRR develop and implement office procedures to control the retrofit of current STS on operating reactors licensed to earlier Tech Specs."

The plant specific atmospheric dispersion factors referred to are covered in response to 10CFR50 Appendix I and associated technical specifications.

Based on these activities, this topic should be deleted from Phase III.

Topic 7.3.2, MSIV Leakage

There has been and continues to be extensive work in this area. A BWR Owners Group is actively pursuing this topic. There are technical specification limits on acceptable leakage of these valves. As noted in the Safety Significance section of the topic definitions, significant operating experience is available and it is recognized as a generic issue for BWRs.

Since there are ongoing activities in this area on a generic basis and technical specification limits are in place for operating plants, this topic should be deleted from Phase III.

ATTACHMENT C

COMPARATIVE EVALUATION OF PHASE II/PHASE III TOPIC DEFINITION REFERENCES

ATTACHMENT C

COMPARATIVE EVALUATION
PHASE II/III REFERENCES

Introduction

A comparative evaluation of the original topic definitions' references listed for SEP Phase II against those proposed in the Phase III topic definitions was performed. Where the references listed in the Phase II topic definitions coincide with those in Phase III, the references were deleted; the references listed under each topic in this attachment are additional references listed in the Phase III topic definitions. This potentially represents addition criteria not used in the original Phase II program.

Due to limited time, this evaluation has not reviewed the actual references listed in the Draft Integrated Plant Safety Assessment documents for Phase II plants. For example, in Phase II, Topic III-4.A, "Tornado Missiles" refers to GDC-2 and Regulatory Guide 1.117 as the review requirements. In the Phase III topic definition, Topic 1.6.2, SRP 3.5.1.4 is referenced in the Review Guidelines section. This same SRP may be referenced in the Reg Guide used in Phase II, but this is not clear.

The value of going through this review prior to going forward with Phase III is to determine if additional regulatory guidance must be addressed in Phase III and if so, the basis for the addition.

SEP Phase III Topic No. 1.1 "Settlement of Foundations and Buried Equipment" (cross-reference SEP Phase II Topic No. II-4.F)

- SEP Phase III criteria imposed above and beyond that of SEP Phase II:

  Review Criteria

  GDC 2, "Design Bases for Protection Against Natural Phenomena," requires that structures, systems and components important to safety shall be designed to withstand the effects of natural phenomena, such as earthquakes, without loss of capability to perform their safety function.

  Review Guidelines

  Regulatory Guides 1.132, "Site Investigations for Foundations of Nuclear Power Plants," and 1.138, "Laboratory Investigation of Soil for Engineering Analysis and Design of Nuclear Power Plants," provide information, recommendations, and guidance and describe a basis acceptable to the staff that may be used to implement the requirements of the criteria described in Section II above.

SEP Phase III Topic No. 1.2 "Stability of Slopes" (cross-reference SEP Phase II Topic No. II.-4.D)

- SEP Phase III criteria imposed above and beyond that of SEP Phase II:

  Review Criteria

  GDC 2, "Design Bases for Protection Against Natural Phenomena," requires that structures, systems and components important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornados, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety function.

  Review Guidelines

  Regulatory Guides 1.132, "Site Investigations for Foundation of Nuclear Power Plants," and 1.138, "Laboratory Investigation of Soil for Engineering Analysis and Design of Nuclear Power Plants," provide information, recommendations, and guidance and describe a basis acceptable to the staff that may be used to implement the requirements of the criteria described in Section II above.

SEP Phase III Topic No. 1.3 "Dam Integrity" (cross-reference SEP Phase II Topic No. II-4.E)

- SEP Phase III criteria imposed above and beyond that of SEP Phase II:

  Review Criteria

  GDC 2, "Design Bases for Protection Against Natural Phenomena," requires that structures, systems and components important to safety shall be designed to withstand effects such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety function.

  Review Guidelines

  The review process is conducted in accordance with the Standard Review Plan Sections 2.5.4, "Suitability of Subsurface Materials and Foundations," and 2.5.5, "Stability of Slopes". Additional information and guidance are presented in Regulatory Guides 1.27, "Ultimate Heat Sink for Nuclear Power Plants, " 1.132, "Site Investigations for Foundations of Nuclear Power Plants," and 1.138, "Laboratory Investigations of Soils for Engineering Analysis and Design of Nuclear Power Plants."

SEP Phase III Topic No. 1.4 "Ground Motion" (cross-reference SEP Phase II Topic No. II-4.A)

- SEP Phase III criteria imposed above and beyond that of SEP Phase II:

  Review Criteria

  The acceptance criteria for the design bases for protection against natural phenomena are contained in Appendix A to 10 CFR 50, General Design Criterion 2. It states, in part, "The design bases for these structures, systems and components shall reflect (1) appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area, with sufficient margin for the limited accuracy, quantity and period of time in which the historical data have been accumulated..."

  Review Guidelines

  The acceptance guidelines are stated in Appendix A to 10 CFR Part 100 as implemented thru:

  A. Standard Review Plan Section 2.5.2, "Vibratory Ground Motion", and/or

  B. "Seismic Hazard Analysis", NUREG/CR-1582, August 1980.

SEP Phase III Topic No. 1.5.1 "Site Hydrologic Characteristics and Capability to Withstand Flooding" (cross-reference SEP Phase II Topic No. II-3.A)

- SEP Phase III criteria imposed above and beyond that of SEP Phase II:

  Review Criteria

  The acceptance criteria for this topic are stated in the Appendix A to 10 CFR Part 100. GDC 2, "Design Bases for Protection Against Natural Phenomena," requires that structures, systems and components important to safety be designed to withstand the effects of natural phenomena, such as flooding. 10 CFR Part 100, Appendix A, "Seismic and Geologic Siting Criteria for Nuclear Power Plants", as it relates to establishig the design basis flood.

  Review Guidelines

  The current guidelines used to determine if plant design meets the topic acceptance criteria are those provided in Standard Review Plan Sections 2.4, the site hydrology review plans, 3.4.1, Flood Protection", and 9.2.5, "Ultimate Heat Sink". Additional information and guidance are provided in Regulatory Guides 1.27, "Ultimate Heat Sink for Nuclear Power Plants," 1.59, "Design Basis Floods for Nuclear Power Plants," 1.102,, "Flood Protection for Nuclear Power Plants," and 1.127, "Inspection of Water Control Structures Associated with Nuclear Power Plants".

SEP Phase III Topic No. 1.5.2 "Site Severe Weather Characteristics" (cross-reference SEP Phase II Topic No. II-2.A)

- SEP Phase III criteria imposed above and beyond that of SEP Phase II:

  Review Criteria

  None

  Review Guidelines

  None

SEP Phase III Topic No. 1.6.1 "Industrial Hazards" (cross-reference SEP Phase II Topic No. II-1.C)

- SEP Phase III criteria imposed above and beyond that of SEP Phase II:

  Review Criteria

  General Design Criterion 4, "Environmental and Missile Design Basis", of Appendix A to 10 CFR Part 50, requires that nuclear power plant structures, systems and components important to safety be appropriately protected against events and conditions that may occur outside the nuclear power plant.

  Review Guidelines

  The review will be conducted in accordance with the guidance given in Standard Review Plan Sections. 2.2.3, "Evaluation of Potential Accidents;" 3.5.1.5, "Site Proximity Missiles (except Aircraft)"; and 3.5.1.6 "Aircraft Hazards".

SEP Phase III Topic No. 1.6.2 "Tornado Missiles" (cross-reference SEP Phase II Topic No. III-4.A)

- SEP Phase III criteria imposed above and beyond that of SEP Phase II:

  Review Criteria

  The criteria governing the design for tornado missiles are given in the General Design Criteria of Appendix A, 10 CFR Part 50. GDC 2, "Design Bases for Protection Against Natural Phenomena," requires that structures, systems and components important to safety be designed to withstand the effects of natural phenomena such as tornados without loss of capability to perform their safety functions.

  Review Guidelines

  None

SEP Phase III Topic No. 1.6.3 "Internally Generated Missiles" (cross-reference SEP Phase II Topic No. III-4.C)

- SEP Phase III criteria imposed above and beyond that of SEP Phase II:

  Review Criteria

  The acceptability of the design of protection for facility structures, systems and components from internally generated missiles is based on meeting General Design Criterion 4, "Environmental and Missile Design Bases". This criterion requires that structures, systems and components important to safety be appropriately protected against the effects of missiles.

  Review Guidelines

  The scope of review is as outlined in Regulatory Guide 1.13, as it relates to the spent fuel pool systems and structures being capable of withstanding the effects of internally generated missiles, and preventing missiles from impacting stored fuel assemblies and Regulatory Guide 1.27, as it relates to the ultimate heat sink being capabile of withstanding the effects of internally generated missiles, provide additional review guidelines.

SEP Phase III Topic No. 1.6.4 "Turbine Missiles" (cross-reference SEP Phase II Topic No. III.4.B)

- SEP Phase III criteria imposed above and beyond that of SEP Phase II:

  Review Criteria

  The acceptance criteria governing the design for turbine missiles is given in the General Design Criteria of Appendix A, 10 CFR Part 50. GDC 4, Environmental and Missile Design Bases", requires that structures, systems and components important to safety be adequately protected against dynamic effects including missiles.

  Review Guidelines

  None.

SEP Phase III Topic No. 2.1 "Classification of Structures" (cross-reference SEP Phase II Topic No. III-1)

- SEP Phase III criteria imposed above and beyond that of SEP Phase II:

  Review Criteria

  The acceptance criteria for classification of structures are stated in the General Design Criteria of Appendix A to 10 CFR Part 50. GDC 1, "Quality Standards and Records," requires that structures be designed to generally recognized codes and standards acceptable to the NRC. GDC 2, "Design Bases for Protection Against Natural Phenomena," requires structures important to safety be designed to withstand the effects of earthquakes.

  Review Guidelines

  The review guidelines for this topic are contained in Standard Review Plan Section 3.8, the structural design sections.

SEP Phase III Topic No. 2.2 "Severe Weather Effects on Structures" (cross-reference SEP Phase II Topic No. II-2.A)

- SEP Phase III criteria imposed above and beyond that of SEP Phase II:

  Review Criteria

  The acceptance criteria governing the design for the above loads is given in the General Design Criteria of Appendix A, 10 CFR Part 50, GDC 2, "Design Bases for Protection Against Natural Phenomena," requires that structures, systems and components important to safety be designed to withstand the effects of natural phenomena such as tornados, hurricanes, floods, tsunamis, and seiches without loss of capability to perform their safety functions.

  Review Guidelines

  The acceptance guidelines are described in Standard Review Plan Sections 3.3.1, "Wind Loadings", 3.3.2, "Tornado Loadings", 3.8.1, "Concrete Containment", 3.8.2, "Steel Containment", 3.8.3, "Concrete and Steel Internal Structures of Steel or Concrete Containments", 3.8.4, "Other Seismic Category I Structures", 3.8.5, "Foundations", 9.2.5, "Ultimate Heat Sink", 2.4, Hydrologic topics, 3.4.1, "Flood Protection", and 3.4.2, "Analyses Procedures". Guideline recommendations are also described in Regulatory Guides 1.59, "Design Basis Floods for Nuclear Power Plants", 1.76, "Design Basis Tornado for Nuclear Power Plants", 1.102, "Flood Protection for Nuclear Power Plants", and 1.117, "Tornado Design Classification".

SEP Phase III Topic No. 2.3 "Design Codes, Criteria and Load Combinations for Structures" (cross-reference SEP Phase II Topic No. III-7.B)

- SEP Phase III criteria imposed above and beyond that of SEP Phase II:

  Review Criteria

  None.

  Review Guidelines

  The review guidelines are presented in Standard Review Plan Sections 3.8.1, "Concrete Containment," 3.8.2, "Steel Containment," 3.8.3, "Concrete and Steel Internal Structures of Concrete or Steel Containments," 3.8.4, "Other Seismic Category I Structures". and 3.8.5, "Foundations".

SEP Phase III Topic No. 2.4 "Containment Design and Inspection" (cross-reference SEP Phase II Topic No. III-7.A)

- SEP Phase III criteria imposed above and beyond that of SEP Phase II:

  Review Criteria

  10 CFR 50, Appendix J and General Design Criterion 50 discuss containment design and inspection. GDC 50, "Containment Design Basis", requires that the containment structure be designed to accommodate the calculated pressure and temperature conditions.

  Review Guidelines

  Regulatory Guides 1.40, "Inservice Inspection of Prestressed Concrete Containment Structures with Grouted Tendons," and Standard Review Plan Section 3.8.1, "Concrete Containment".

SEP Phase III Topic No. 2.5 "Seismic Design of Structures, Systems and Components" (cross-reference SEP Phase II Topic No. III-6)

- SEP Phase III criteria imposed above and beyond that of SEP Phase II:

  Review Criteria

  The acceptance criteria for the seismic design consideration are stated in the General Design Criteria of Appendix A, 10 CFR Part 50. Criterion 2, "Design Bases for Protection Against Natural Phenomena," requires that structures, systems and components important to safety shall be designed to withstand the effects of natural phenomena, such as earthquakes, without loss of capability to perform their safety functions.

  Review Guidelines

  The following review criteria and guidelines to be used for review are:

  1. NUREG/CR-0098, "Development of Criteria for Seismic Review of Selected Nuclear Power Plants," by N. M. Newmark and W. J. Hall, May 1978.

  2. "SEP Guidelines for Soil-Structures Interaction Review," by SEP Senior Seismic Review Team, December 8, 1980.

  For the cases that are not covered by the criteria stated above, the following Standard Review Plans and Regulatory Guides will be used:

  1. Regulatory Guides 1.29 and 1.100.

SEP Phase III Topic No. 3.1 "RCPB Leakage Detection"
(cross-reference SEP Phase II Topic No. V-5)

- SEP Phase III criteria imposed above and beyond that of SEP Phase II:

Review Criteria

The acceptance criteria for the detection of leakage from reactor coolant pressure boundary are stated in the General Design Criteria of Appendix A, 10 CFR Part 50. Criterion 30, "Quality of Reactor Coolant Pressure Boundary," requires that means shall be provided for detecting and to the extent practical, identifying the location of the sources of leakage in the reactor coolant pressure boundary.

Review Guidelines

None.

SEP Phase III Topic No. 3.2 "Reactor Core Isolation Cooling System (BWR)" (cross-reference SEP Phase II Topic No. V-9)

- SEP Phase III criteria imposed above and beyond that of SEP Phase II:

  Review Criteria

  General Design Criteria 35 of Appendix A to 10 CFR Part 50, "Emergency Core Cooling", states that a system to provide abundant emergency core cooling should be provided. The system should have suitable redundancy to assure that with just onsite or just offsite power, the system safety function can be accomplished assuming a single failure.

  Review Guidelines

  The acceptance criteria for emergency core cooling systems are described in Standard Review Plan Section 6.3, Emergency Core Cooling System".

  The criteria for the RCIC are described in SRP 5.4.6, "Reactor Core Isolation Cooling System (BWR)."

SEP Phase III Topic No. 4.1 "Classification and Design of Systems and Components" (cross-reference SEP Phase II Topic No. III-1)

- SEP Phase III criteria imposed above and beyond that of SEP Phase II:

  Review Criteria

  The acceptance criteria for classification of systems and components are stated in the General Design Criteria (GDC) of Appendix A of 10 CFR Part 50. GDC 1, "Quality Standards and Records," requires that systems and components be designed to generally recognized codes and standards acceptable to the NRC. GDC 2, "Design Basis for Protection Against Natural Phenomena," requires that systems and components important to safety be designed to withstand the effects of earthquakes.

  Review Guidelines

  None

SEP Phase III Topic No. 4.2.1 "Shutdown Systems" (cross-reference SEP Phase II Topic No. VII-3)

- SEP Phase III criteria imposed above and beyond that of SEP Phase II:

  Review Criteria

  The acceptance criteria for systems provided to remove decay heat are stated in the General Design Criteria (GDC) of Appendix A, 10 CFR Part 50. GDC 34, "Residual Heat Removal," requires that system function be accomplished with a single failure and just onsite or just offsite power available. GDC 2, "Design Bases for Protectin Against Natural Phenomena" and GDC 4, "Environmental and Missile Design Bases", require that systems and components important to safety be designed to withstand the effects to earthquakes, tornadoes, hurricanes, floods and missiles without loss of safety function. GDC 19, "Control Room," requires that equipment be located outside of the control room with a design capability for prompt hot shutdown and potential capability for cold shutdown through the use of suitable procedures.

  Review Guidelines

  Current licensing guidelines for the review of decay heat removal capability are contained in Standard Review Plan Section 5.4.7, "Residual Heat Removal (RHR) System," Branch Technical Position (BTP) RSB 5-1, "Design Requirements of the Residual Heat Removal System," and Regulatory Guide 1.139, "Guidance for Residual Heat Removal".

SEP Phase III Topic No. 4.2.2 "Shutdown Electrical Instrumentation and Controls" (cross-reference SEP Phase II Topic No. VII-3)

- SEP Phase III criteria imposed above and beyond that of SEP Phase II:

  Review Criteria

  10 CFR 50.55a(h) requires that new plants satisfy the requirements of IEEE Standard 279-1971.

  Review Guidelines

  The acceptance guidelines for the review of safe shutdown systems are presented in Standard Review Plan Sections 7.3, "Engineered Safety Features Systems," 7.4, "Safe Shutdown Systems," and 7.6, "Interlock Systems Important to Safety." IEEE Standard 279-1971 presents design criteria for assuring that systems required for safety will function even in the event of a single random failure. Required techniques include redundancy, independency and periodic testing. Required design reviews include design basis events, response times, instrument accuracy and setpoints.

SEP Phase III Topic No. 4.3 "Service and Cooling Water Systems" (cross-reference SEP Phase II Topic No. II-3.C)

- SEP Phase III criteria imposed above and beyond that of SEP Phase II:

  Review Criteria

  The acceptance criteria for the service and cooling water system are stated in the General Design Criteria (GDC) of Appendix A, 10 CFR Part 50, GDC 44, "Cooling Water," GDC 45, Inspection of Cooling Water System," and GDC 46, "Testing of Cooling Water System". These General Design Criteria require that a cooling water system be provided, inspected and tested and that the system be capable of transferring heat from structures, systems and components important to safety to the ultimate heat sink.

  Review Guidelines

  The acceptance criteria are described in Standard Review Plan Sections 9.2.1, "Station Service Water System", and 9.2.2, "Reactor Auxiliary Cooling Water Systems".

SEP Phase III Topic No. 4.4 "Ventilation Systems" (cross-reference SEP Phase II Topic No. IX-5)

- SEP Phase III criteria imposed above and beyond that of SEP Phase II:

  Review Criteria

  The acceptance criteria for ventilation systems are stated in the General Design Criteria (GDC) of Appendix A, 10 CFR Part 50. GDC 4, "Environmental and Missile Design Bases," requires that systems and components important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation and postulated accidents. Ventilation systems maintain those conditions.

  Review Guidelines

  None

SEP Phase III Topic No. 4.5 "Spent Fuel Storage" (cross-reference SEP Phase II Topic No. IX-1)

- SEP Phase III criteria imposed above and beyond that of SEP Phase II:

  Review Criteria

  The plant design will be reviewed with regard to Section VI, "Fuel and Radioactivity Control", of Appendix A to 10 CFR Part 50, "General Design Criteria for Nuclear Power Plants", which requires that the fuel storage systems shall be designed to assure adequate safety under normal and postulated accident conditions.

  Review Guidelines

  Current guidance for the review of spent fuel storage is provided in Standard Review Plan, Section 9.1.2, "Spent Fuel Storage," Section 9.1.3, "Spent Fuel Pool Cooling and Cleanup System", Section 9.1.4, "Fuel Handling System," and Regulatory Guides 1.29, "Seismic Design Classification," 1.13, "Fuel Storage Facility Design Basis," 1.26, "Quality Group Classification and Standards for Water-Steam and Radioactive Waste-Containing Components for Nuclear Power Plants," as well as the guidance contained in the April 14, 1978 generic letter - "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications".

SEP Phase III Topic No. 4.6 "Isolation of High and Low Pressure Systems" (cross-reference SEP Phase II Topic No. V-11.A)

- SEP Phase III criteria imposed above and beyond that of SEP Phase II:

  Review Criteria

  General Design Criterion (GDC) 14, "Reactor Coolant Pressure Boundary", of Appendix A to 10 CFR Part 50, requires that reactor systems be designed, fabricated, erected, and tested so as to have an extremely low probability of failure. GDC 34, "Residual Heat Removal", requires that the residual heat removal system operate to maintain core integrity. 10 CFR 50.55a(h) establishes IEEE Standard 279-1971 as the principal standard for the instrumentation and control of systems required for safety.

  Review Guidelines

  The acceptance guidelines for the review of systems that isolate low pressure systems from high pressure systems is presented in Standard Review Plan Section 5.4.7, "Residual Heat Removal (RHR) System," and 7.6, "Interlock Systems Important to Safety".

SEP Phase III Topic No. 4.7.1 "Automatic ECCS Switchover" (cross-reference SEP Phase II Topic No. VI-7.B)

- SEP Phase III criteria imposed above and beyond that of SEP Phase II:

  Review Criteria

  10 CFR 50.55a(h) requires that new plants satisfy the requirements of IEEE Standard 279-1971. General Design Criterion 35, Appendix A to 10 CFR Part 50, "Emergency Core Cooling", requires that a system to provide abundant core cooling should be provided and that the system be such that system safety function can be accomplished assuming a single failure.

  Review Guidelines

  The acceptance guidelines for the review of emergency core cooling systems is presented in Standard Review Plan Section 7.3, "Engineered Safety Features Systems".

  IEEE Standard 279-1971 presents design criteria for assuring that systems required for safety will function even in the event of a single random failure. Required techniques include redundancy, independancy and periodic testing. Required design reviews include design basis events, response times, instrument accuracy and setpoints.

  The review is conducted in accordance with Standard Review Plan Section 6.3, "Emergency Core Cooling System", Regulatory Guide 1.62, "Manual/Initiation of Protection Actions," and Branch Technical Position ICSB 20, "Design of Instrumentation and Controls Provided to Accomplish Changeover from Injection to Recirculation Mode.

SEP Phase III Topic No. 4.8.1 "Emergency AC Power Systems" (cross-reference SEP Phase II Topic No. VIII-2)

- SEP Phase III criteria imposed above and beyond that of SEP Phase II:

Review Criteria

General Design Criteria (GDC) 17, "Electrica Power Systems," of Appendix A to 10 CFR Part 50, requires that systems important to safety be powered from both onsite and offsite sources. The onsite and offsite sources are required to provide sufficient power, assuming the failure of the other source.

10 CFR 50.55a(h) requires that new plants satisfy the requirements of IEEE Standard 279-1971.

Review Guidelines

The acceptance guidelines for the review of ac system are presented in Standard Review Plan Section 8.3.1, "AC Power Systems (ONSITE)." IEEE Standard 279-1971 presents design criteria for assuring that systems required for safety will function even in the event of a single random failure. Required techniques include redundancy, independency and periodic testing. Required design reviews include design basis events, response times, instrument accuracy and setpoints.

SEP Phase III Topic No. 4.8.2 "Emergency DC Power Systems" (cross-reference SEP Phase II Topic No. VIII-3.B)

- SEP Phase III criteria imposed above and beyond that of SEP Phase II:

Review Criteria

General Design Criteria (GDC) 17, "Electric Power Systems", of Appendix A to 10 CFR Part 50, requires that systems important to safety be powered from both onsite and offsite sources. The onsite and offsite sources are required to provide sufficient power assuming the failure of the other source.

10 CFR 50.55a(h) requires that new plants satisfy the requirements of IEEE Standard 279-1971.

Review Guidelines

IEEE Standard 279-1971 presents design criteria for assuring that systems required for safety will function even in the event of a single random failure. Required techniques include redundancy, independency and periodc testing. Required design reviews include design basis events, response times, instrument accuracy and setpoints.

SEP Phase III Topic No. 4.8.3 "Swing Bus Design (BWR-4)" (cross-reference SEP Phase II Topic No. VII-7)

- SEP Phase III criteria imposed above and beyond that of SEP Phase II:

Review Criteria

Appendix A to 10 CFR Part 50, requires that systems important to safety be powered from both onsite and offsite sources. The onsite and offsite sources are required to be reviewed assuming the failure of the other source.

10 CFR 50.55a(h) requires that new plants satisfy the requirements of IEEE Standard 279-1971.

Review Guidelines

The acceptance guidelines for the review of electrical systems is presented in Standard Review Plan Sections 8.3.1, "AC Power Systems (Onsite)," and 8.3.2, "DC Power Systems (Onsite)." IEEE Standard 279-1971 presents design criteria for assuring that systems required for safety will function even in the event of a single random failure. Required techniques include redundancy, independancy and periodic testing. Required design reviews include design basis events, response times, instrument accuracy and setpoints.

SEP Phase III Topic No. 4.9 "Shared Systems" (cross-reference SEP Phase II Topic No. VI-10.B)

- SEP Phase III criteria imposed above and beyond that of SEP Phase II:

  Review Criteria

  General Design Criterion 5, "Sharing of Structures, Systems and Components," of Appendix A to 10 CFR Part 50, prohibits structures, systems and components important to safety from being shared among nuclear power units unless it can be shown that such sharing will not significantly impair their ability to perform their safety functions. These safety functions include the capability to perform an orderly shutdown and cooldown of the remaining units in the event of an accident in one unit.

  Review Guidelines

  The acceptance guidelines for systems that are required for the protection of public health and safety presented in many Standard Review Plan Sections. The plant design information presented in the safety analysis report, technical specifications, and drawings are reviewed to assure that: (1) the interconnection of ESF, onsite emergency power, and service systems between different units are such that a failure, maintenance or testing between different are such that a failure, maintenance or testing operation in one unit will not affect the accomplishment of the protective function of the system(s) in other units, (2) the required coordination between unit operators can cope with an incident in one unit and safe shutdown of the remaining unit(s), and (3) system overload conditions will not arise as a consequence of an accident on one unit coincident with a spurious accident signal or any other single failure in another unit.

SEP Phase III Topic No. 5.1 "Reactor Protection System and Engineered Safety Feature Systems Isolation" (cross-reference SEP Phase II Topic No. VII-1.A)

- SEP Phase III criteria imposed above and beyond that of SEP Phase II:

  Review Criteria

  10 CFR 50.55a(h) requires that new plants satisfy the requirements of IEEE Standard 279-1971. General Design Criterion 24 of Appendix A to 10 CFR Part 50, "Separation of Protection and Control Systems" also applies.

  Review Guidelines

  The acceptance guidelines for the review of isolation systems is presented in Standard Review Plan Section 7.3, "Engineered Safety Features Systems." IEEE Standard 279-1971 presented design criteria for assuring that systems requied for safety will function even in the event of a single random failure. Required techniques include redundancy, independancy and periodic testing. Required design reviews include basis events, response times, instrument accuracy and setpoints.

  The plant design information presented in the Safety Analysis Report, technical specifications and drawings are reviewed to verify that operating reactors have RPS and ESF designs which provide isolation of non-safety systems for safety systems to assure that safety systems will function as required.

SEP Phase III Topic No. 5.2 "RPS and ESF Testing" (cross-reference SEP Phase II Topic No. VI-7.A.3)

- SEP Phase III criteria imposed above and beyond that of SEP Phase II:

  Review Criteria

  10 CFR 50.55a(h) requires that new plants satisfy the requirements of IEEE Standard 279-1971. General Design Criteria (GDC) 21, "Protection System Reliability and Testability," and GDC 37, "Testing of Emergency Core Cooling System," also apply as related to periodic testing requirements of RPS and ESF.

  Review Guidelines

  The acceptance guidelines for the review of safety systems is presented in Standard Review Plan Section 7.2, "Reactor Trip System" and Section 7.3, "Engineered Safety Features Systems." IEEE Standard 279-1971 presents design criteria for assuring that systems required for safety will function even in the event of a single random failure. Required techniques include redundancy, independancy, periodic testing, and instrument accuracy and setpoints.

  The plant design information presented in the Safety Analysis Report, technical specifications and drawings are reviewed to assure that all ECCS components (e.g., valves and pumps) are included in the component and system test, that the frequency and scope of the periodic testing are adequate and meet the requirements of GDC 37. Also, the review should verify that the operability of the RPS and ESF (on a periodic basis), and that the test program demonstrates a high degree of availability of the systems.

SEP Phase III Topic No. 6.1 "Organic Materials" (cross-reference SEP Phase II Topic No. VI-1)

- SEP Phase III criteria imposed above and beyond that of SEP Phase II:

  Review Criteria

  The plant design will be reviewed with regard to General Design Criterion 1, "Quality Standards and Records," of Appendix A to 10 CFR Part 50, "General Design Criteria for Nuclear Power Plants," which requires that structures and systems important to safety be designed and tested to quality standards commensurate with the importance of the safety function to be performed. Also, Appendix B to 10 CFR 50, "Quality Assurance Criteria for Nuclear Power Plants and fuel Reprocessing Plants", describes an acceptable method of complying with the Commissions quality assurance requirements with regard to protective coatings.

  Review Guidelines

  Current guidance for the review of organic materials in containment is provided in Standard Review Plan Sections 6.1.1, "Engineered Safety Features Materials",

SEP Phase III Topic No. 6.2 "RCS Water Purity (BWR)"
(cross-reference SEP Phase II Topic No. V-12.A)

- SEP Phase III criteria imposed above and beyond that of SEP Phase II:

  Review Criteria

  General Design Criteria 14, "Reactor Coolant Pressure Boundary," of Appendix A to 10 CFR Part 50 require assurance that the reactor coolant pressure boundary will have minimal probability of gross rupture of rapidly propagating failure.

  General Design Criterion 15, "Reactor Coolant System Design," requires that the reactor coolant system and associated systems be designed with sufficient margin to ensure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences.

  General Design Criterion 13, "Instrumentation and Control" requires that instrumentaton be provided to monitor variables and systems that can affect the reactor coolant pressure boundary over their anticipated operational occurrences, and for accident conditions as appropriate to ensure adequate safety.

  Review Guidelines

  The review is conducted in Standard Review Plan Section 5.4.8, "Reactor Water Cleanup Systems", SEP Phase III Topic No. 7.1.1 "Pipe Break Definition Criteria" (cross-reference SEP Phase II Topic No. III-5.A, III-5.B)

- SEP Phase III criteria imposed above and beyond that of SEP Phase II:

  Review Criteria

  None.

  Review Guidelines

  The licensee's break location criteria and methods of analysis for evaluating postulated breaks in high energy piping systems inside and outside containment will be compared with the currently accepted review guidelines as described above. In addition, further guidance can be found in the July 20, 1978 letter from D. David to the SEP Phase II Owners Group (KMC, Inc.) and in a January 4, 1980 NRC letter to each of the SEP Phase II licensees.

  The guidelines contained in Enclosure 2 to the December 4, 1981 letter, D. Crutchfield (NRC) to D. Hoffman (CPCo.) may be used where separation or physical restraints for protection against dynamic effects of high energy piping failures are not practical.

SEP Phase III Topic No. 7.1.2 "Pipe Break Effects on Systems and Components" (cross-reference SEP Phase II Topic No. III-5.A)

- SEP Phase III criteria imposed above and beyond that of SEP Phase II:

  Review Criteria

  None

  Review Guidelines

  The current licensing guidelines for review of the effects of pipe break are contained in Standard Review Plan 3.6.1, "Plant Design of Protection Against Postulated Piping Failures in Fluid Systems Outside Containment".

  The guidelins contained in Enclosure 2 to the December 4, 1981 letter D. Crutchfield (NRC) to D. Hoffman (CPCo) may be used where separation or physical restraints for protection against dynamic effects of high energy piping failures are not practical.

  The review does not include consideration of component presssurizaton, pipe whip, jet impingement, and pipe reaction loads on structures, loading combinations and other design aspects of protective structures or compartments used to protect essential systems and components.

SEP Phase III Topic No. 7.2 "Containment Isolation" (cross-reference SEP Phase II Topic No. VI-4)

- SEP Phase III criteria imposed above and beyond that of SEP Phase II:

  Review Criteria

  None

  Review Guidelines

  None.

SEP Phase III Topic No. 7.3.1 "RCS Specific Activity Limits" (cross-reference SEP Phase II Topic No. XV-16, XV-18)

- SEP Phase III criteria imposed above and beyond that of SEP Phase II:

  Review Criteria

  Section 50.36, "Technical Specifications," of 10 CFR Part 50, requires that each license authorizing operation of a nuclear power reactor include technical specifications derived from the analysis and evaluation included in the safety analysis report. The technical specifications are to include limiting conditions for operation (LCO) and surveillance requirements. LCO's provide the lowest performance level required for safe operation of the plant.

  Review Guidelines

  The initial review will compare the existing plant technical specifications with those recommended in the STS associated with the plant design.

  Atmospheric dispersion factors will be calculated in accordance with SRP 2.3.4, "Short-term Diffusion Estimates for Accidental Atmospheric Release."

SEP Phase III Topic No. 7.3.2 "MSIV Leakage (BWR)" (cross-reference SEP Phase II Topic No. VI-9.A)

- SEP Phase III criteria imposed above and beyond that of SEP Phase II:

  Review Criteria

  The acceptance criteria for the main steam isolation valve leakage review is stated in the General Design Criteria of Appendix A, 10 CFR Part 50. GDC 54, "Piping Systems Penetrating Containment," requires leak detection capabilities for piping penetrating containment.

  Review Guidelines

  The review for this topic is conducted in accordance with Standard Review Plan Sections 15.6.5, "Radiological Consequences of a Design Basis Loss of Coolant Accident Including Containment Leakage Contribution".