Attachment 23

UNITED STATES NUCLEAR REGULATORY COMMISSION

OFFICE OF NUCLEAR REACTOR REGULATION

REPORT ON THE

SYSTEMATIC EVALUATION OF OPERATING FACILITIES

I. INTRODUCTION

The Office of Nuclear Reactor Regulation (NRR) has initiated a program for the systematic evaluation of operating nuclear power facilities. The program, called the Systematic Evaluation Program (SEP), has the following objectives:

1. Reassess the safety margins of the design and operation of selected older operating nuclear power plants.
2. Establish documentation which shows how each operating plant reviewed in the SEP compares with current criteria on significant safety considerations, and which provides a basis for acceptance of any departures from these criteria.
3. Provide the capability to make integrated and balanced decisions with respect to any required safety improvements.
4. Identify and resolve significant safety deficiencies early in the SEP, if such deficiencies exist.
5. Efficiently use available personnel and minimize NRC and licensee resource requirements to perform the SEP.

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Phase I of the program consisted of developing the process for systematic evaluation of certain older operating reactors and of establishing the resource needs and schedules. Phase II consists of reviewing, within a period of three years, eleven older operating reactors. These reactors consist of the eight oldest facilities (excluding Indian Point 1 and Humboldt Bay) and the remaining facilities having provisional operating licenses (POLs), which will be converted to full term licenses (FTLs) by this program. Another facility, Monticello Nuclear Generating Plant, has a provisional operating license but has been excluded from Phase II of the SEP because the staff evaluation and ACRS review of the Monticello POL-FTL conversion has been completed. Certain matters related to this licensing action are still pending before the hearing board; however, they are expected to be resolved in the near future, which will permit converting this POL to a FTL. The results of the SEP review will provide an adequate safety basis for making a decision regarding granting a FTL. Therefore, conversion of the remaining POLs will be an adjunct to the SEP. The eleven reactors to be reviewed during Phase II of SEP are listed in Attachment 1. Following completion of this three-year program, the NRC will evaluate the appropriateness of whether the program should be extended to other operating reactor facilities.

II. BACKGROUND

The need for a more routine systematic evaluation of operating facilities in the light of current knowledge and licensing practices has been recognized for some time. On May 3, 1976, the Director, Division of Operating Reactors, established a Task Force to develop a program plan for (1) evaluating licensed nuclear power plants against current criteria, and (2) developing a framework from which backfitting decisions can be evaluated considering all plant features relating to safety. The Task Force was composed of representatives with a broad spectrum of technical backgrounds. Representatives from other NRC Office and NRR Divisions participated in the Task Force effort and provided valuable input.

The Task Force concluded that a systematic evaluation program is needed and recommended a concept for reviewing operating reactors. They also recommended that the staff take steps to limit the potential number of facilities that need systematic review and eliminate the need for future systematic evaluations. To implement these recommendations, NRR management initiated measures to assure that deviations from licensing requirements and their basis for acceptance be documented in future operating license reviews. In addition, all new licensing requirements which are identified by the Regulatory Requirements Review Committee, (RRRC) to be applicable to operating facilities are being assessed for each of these facilities and the conclusions documented as new requirements are adopted. These two actions will insure that in the future operating plants will have a record of the results of the staff review of all safety issues and that the record will be continuously updated as new issues are identified by the staff.

In January 1977, the Commission approved the objectives and general approach of the proposed Systematic Evaluation Program. In approving Phase I of the program, the Commission requested the staff to provide additional information regarding the scope of review and resource requirements prior to initiating any plant-specific review efforts.

In March 1977 a Systematic Evaluation Program (SEP) Review Group was established in the Division of Operating Reactors to implement Phase I of the program as approved by the Commission[1/]. The Group reports to the Assistant Director for Operational Technology and consists of seven full-time NRR professionals, one I&E professional, a Group Leader and a Secretary. The following activities associated with Phase I of the SEP have been completed:

1. A comprehensive List of SEP Topics of safety significance (hereafter referred to as the "Topic List") has been developed for use in the systematic evaluation of operating reactor facilities. The development of the Topic List involved an evolutionary process, starting with all known topics of safety concern. Topics in various general categories which did not need to be considered in the SEP were deleted. The resulting final Topic List is provided as Attachment 2.

1/ Memorandum from Samuel Chilk to Lee V. Gossick (SECY 76-545), "Commission Guidance on NRR Systematic Evaluation Program for Operating Reactors" (Policy Session Item), January 27, 1977.

2. Definitions for each of the items on the Topic List, including a statement of the safety substance, and a discussion of the status of any ongoing work related to the topic have been prepared.
3. A process for conducting plant-specific reviews including general criteria for determining design acceptability has been developed.
4. Resource requirements and associated schedules for implementing SEP have been reassessed.
5. The impact of the SEP on other NRR programs and on other NRC Offices has been evaluated.

The results of the Phase I effort were presented to the Commission on November 9, 1977. The Commission approved the program as presented and directed that Phase II, the review of eleven operating reactors, be initiated immediately.

III. PROGRAM

The Systematic Evaluation Program is based on a listing of specific areas to be examined (Topic List) and on an integrated review of the overall ability of a plant to respond to certain design basis events (challenges), including normal operation, transients and postulated accidents. The review procedure will result in a reassessment of the overall safety margins at each facility and documentation of the reassessment on the basis of current criteria.

A. Type of Topics

The Topic List can be considered to consist of three types of topics. The first type consists of design basis events, i.e., transients, accidents and natural phenomena, which the plant should be able to withstand without exceeding specified acceptance criteria. Examples of such design basis events are turbine trip, failures of high energy piping lines, fires, seismic events, and floods. The second type consists of identified potential failure mechanisms within safety-related systems (e.g., steam generator integrity). The third type is associated with activities to limit the likelihood of failure in safety-related systems (e.g., equipment to monitor the condition of D.C. power system supplies).

The Topic List (Attachment 2) was generated in several steps. First, a listing of all known and previously identified safety considerations was compiled. This included generic issues (e.g., water hammer, fire protection), and other known safety topics (e.g., overpressurization protection). Various NRC Offices and Divisions were requested to identify any additional safety considerations and to suggest other review topics for inclusion in the Topic List. (The Introduction to the Topic List in Attachment 2 identifies the source of the topics and discusses the evolution of the List). More than 800 topics were considered in the development of the original list, which was created by the simple addition of all listings. This process, of course, led us to identify many items that were duplicative in nature and, consequently, omitted.

As a second step in developing a "Topic List", topics not normally included in the review of light water reactors and topics either related to research and development programs or to the development of analytical evaluation models and methodology were categorically deleted. Next, both those topics which are being reviewed on a periodic basis in accordance with current criteria (e.g., fuel performance) and those technical subjects that previously have been reviewed and implemented on operating reactors (e.g., BWR Channel Box Integrity) were also categorically deleted. These groups of topics were deleted because current NRR practices already appropriately address such issues, and they have been appropriately considered for operating facilities.

The remaining topics were arranged in groups corresponding to the outline of the NRR Standard Review Plan (SRP). "Titles" from the SRP were used as headings in the Topic List for the purpose of organizing the topics into the format of the SRP; however, all titles themselves do not necessarily represent topics for consideration. Following this structuring of topics, a "definition" was prepared for each topic to ensure a common understanding. This "definition" included a statement of the safety objective and the current status of the review topic. The listing so generated was termed the "Topic List".

The final Topic List has been reviewed by all Divisions of NRR and by the NRC Offices of I&E, SD, RES, and ELD. The Topic List thus developed includes general topics identified by DOR staff members for a systematic evaluation, specific safety considerations not previously considered generic issues, and previously identified generic issues from ACRS or other NRR generic issue lists. These previously identified generic issues are either under investigation, but only resolved on an interim basis (e.g., pressure-suppression BWR containment integrity), or are defined as generic issues but presently not being actively investigated for operati facilities (e.g., control room habitability).

Some of the generic topics are the subject of ongoing programs and will be resolved and implemented independently of SEP activities, i.e., resolution of these issues will not be accomplished as part of the SEP effort. However, for the plants which are part of the SEP effort, the ultimate conclusions of facility reviews for ongoing generic concerns and of the SEP will be closely integrated, where practicable, into an overall NRC position to improve the effectiveness and efficiency of these efforts and to minimize any impact on licensees. In addition, the status and conclusions of all generic issues will be molded into the final safety assessment for each plant.

When the Topic List is applied to a specific plant, additional topics will be deleted because (a) certain topics apply only to a specific reactor type (e.g., BWR or PWR) or design feature (e.g., ice condenser containment for PWRs) and (b) some topics have been previously resolved for a specific plant (e.g., Overhead Handling Systems - Cranes).

B. Safety Significance of Topics

The original Task Force expected that a large number of topics could be deleted generically, for all reactors on the basis of "lesser safety significance". The SEP Group initially identified some topics of "lesser safety significance" for all operating plants; however, it became apparent that the safety significance of many topics is plant dependent and only a few topics could be categorically deleted from the Topic List on this basis.

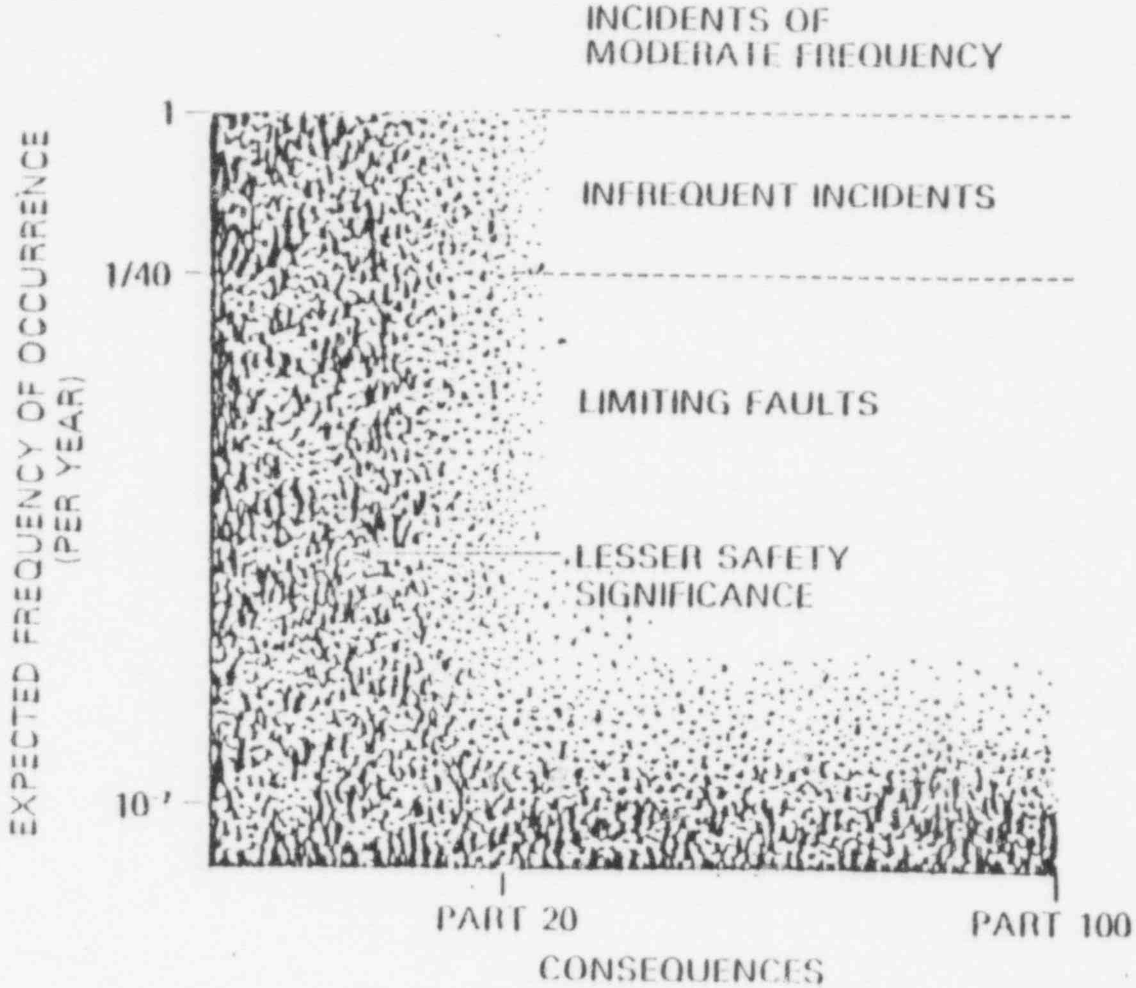
Early in the plant specific reviews of Phase II, the topics will be evaluated for safety significance to that plant. During this phase it is expected that additional topics will be deleted on the basis of "lesser safety significance". After a few plant reviews have been initiated, it may also be found that more topics can be categorically deleted from the generic Topic List on the basis of "lesser safety significance".

The "lesser safety significance" will be evaluated on the basis of the probability of occurrence of a given event and the magnitude of the radiological consequences of such an event should it occur. When the probability or consequence of an event related to a particular topic is judged to be sufficiently small, the topic will not be reviewed in detail and the basis for acceptability will be documented. For example, a minimum review effort will be required to determine the likelihood of an aircraft crash at a remote site away from airports and flight paths. Figure 1 shows graphically the relationship between the probability of occurrence and the resulting radiological consequences with commonly accepted thresholds for both. 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 result in consequences of only a small fraction of 10 CFR Part 100 guidelines are considered to be of "lesser safety significance".

Examples of topics which were generically deleted on the basis of "lesser safety significance" for all operating plants follow:

(1) Containment External Design Pressure with Respect to Containment Inadvertent Spray Operation

The NRC normally reviews the containment functional design (SRP 6.2.1) for CP and OL applications, including consideration of external design pressure. A conservative structural design is required to assure that the containment structure is capable of withstanding the maximum external pressure; or interlocks in the



GENERAL APPROACH FOR DETERMINATION OF "LESSER SAFETY SIGNIFICANCE"

FIGURE 1

plant protection system and administrative controls are required to preclude inadvertent operation of containment heat removal systems; or for steel containment vessels, vacuum relief devices are provided in accordance with applicable code requirements. Inadvertent containment spray operation results in cooling of the containment atmosphere and lowering its pressure if (a) the incoming containment spray liquid is colder than the building atmospheric temperature and (b) the building is a closed vessel (no vacuum breakers and no ventilation system in operation). Using conservative assumptions, the calculated equilibrium containment pressure would be about 3 psi below the outside air pressure. However, using realistic initial operating conditions, the differential pressure resulting from inadvertent spray operation would be no more than about 1.5 psi. Operator intervention and/or vacuum breaker actuation would reduce the pressure differential even more.

Because of the inherent containment strength resulting from requirements to meet its internal design pressure, it should withstand an external pressure of at least 3 psi. Thus, it is not expected to fail in the event of inadvertant spray operation even under conservative postulated conditions.

Hence, logical technical arguments coupled with the very remote possibility that a containment would be required to perform its safety function simultaneously with inadvertent spray operation, justifies deletion of this topic for operating facilities on the basis of lesser safety significance.

Inadvertent spray operation would necessitate a reactor shutdown and could require inspection, testing, and repairs of components and structures, if not adequately designed to withstand the external pressure, to assure their acceptability for continued operation. This would have an unnecessary, undesirable impact on facility operation. Therefore, it is prudent to continue to review this topic for new facility applications on a conservative basis to assure that reasonable precautions have been taken to accommodate inadvertent spray operation.

(2) Radioactive Wastes Systems

Based on past operating experience, including consideration of system and component failures, releases of gaseous, liquid, or solid radioactive waste have not exceeded 10 CFR Part 20 limits. In addition, if the worst possible failure of any of these systems were postulated to occur, it has been calculated that the resultant consequences would not exceed 10 CFR Part 100 guidelines. On this basis this topic has been considered of "lesser safety significance" and was deleted from the Topic List. This determination, however, does not obviate the need to evaluate the design of these systems in new facilities.

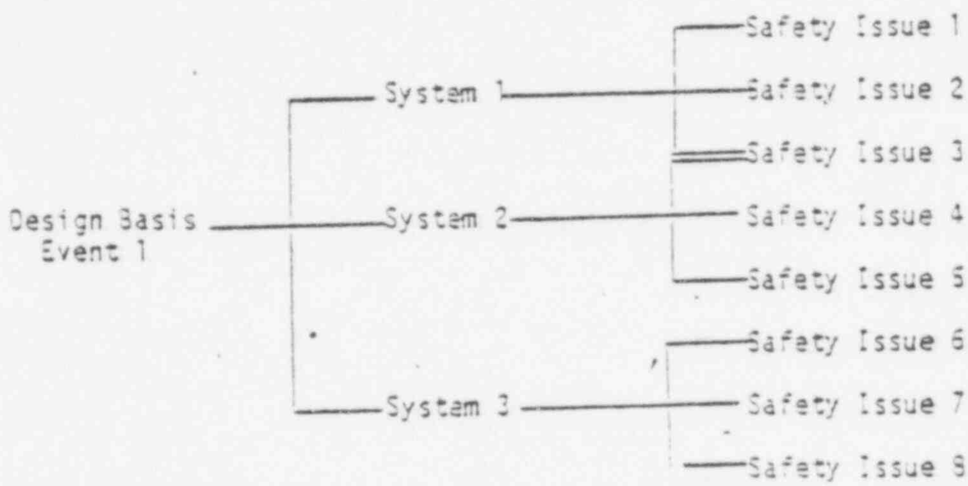
If a topic is associated with an event expected to occur more frequently than about 10^{-6} to 10^{-7} per year and which may potentially result in consequences greater than a small fraction of 10 CFR Part 100 guidelines, it is considered to have potential safety significance. As indicated in Figure 1, these guidelines are not defined by well delineated boundaries, but instead by "grey areas", where the safety significance of a topic has to be determined on a case-by-case basis. The determination of whether a topic in this "grey area" is safety significant involves consideration of other factors, such as the degree of conservatism in the safety analysis. For example, if an event is found to have a frequency of occurrence somewhat higher than about 10^{-6} per year by a very conservative method and/or if the consequences associated with the event have been determined to be small by a more realistic approach, then the topic may be judged by the SEP to be of "lesser safety significance".

C. Review Procedure

The capability of a plant to respond to selected design basis events will be the basis for assessing the safety adequacy of the operating facilities reevaluated. The selected design basis events are included in the Topic List and consist of identified single events which can potentially result in greater than routine releases of radioactive material from the site. This set of design basis events include: (1) natural phenomena such as earthquakes, fires and floods and (2) events resulting from postulated plant transients, accidents and failures such as turbine missiles and pipe breaks. An operating facility will be considered to be adequately safe if it can be demonstrated that the facility will reliably respond to all design basis events with 10 CFR Part 100 guidelines values not being exceeded.

Safety topics remaining on the Topic List after the removal of the design basis events are of two types, those that affect the likelihood of occurrence of an event and those that affect the response of the plant to the event (some topics affect both). Both the likelihood of the event and the plant response to the event will be considered in making safety assessments. A specific plant evaluation will be performed in two steps: First, each design basis event will be evaluated separately and secondly, an overall plant safety assessment will be made considering the plant response to all design basis events. A discussion of the two steps follows.

Step 1 Plant systems and their functions which respond to mitigate the consequences of each design basis event will be identified. Next the safety issues from the Topic List which affect each system or system function will be identified. The relationship between a design basis event and the systems and safety issues which affect it are shown below.



In this diagram, Systems 1, 2 and 3 are representative of plant systems designed to respond to Design Basis Event 1 (i.e., protect the plant or the public or both). Safety Issues 1, 2 and 3 are safety issues from the Topic List which potentially affect the capability of System 1 to perform its safety function. Likewise Safety Issues 3, 4 and 5 affect System 2 and Safety Issues 6, 7 and 8 affect System 3. It should be noted that some systems are designed for protection from more than one Design Basis Event and that some safety issues can affect more than one system.

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Consideration of both plant systems and design basis events is considered necessary to provide a meaningful context for resolving the safety issues, and to provide a basis for assessing the effect of the safety issues on overall plant safety. All plant systems which are important to protecting the plant from the design basis events will also be identified. Those systems will be compared with similar systems required for current plants to determine the extent of differences.

SRP criteria will be used as a gauge to determine the extent of conformance with current licensing criteria. If the results of the review of a given topic is that the plant meets current requirements, then it will be deemed to be satisfactory and will be so documented. If the results do not meet current requirements (which we expect to often be the case), the degree of deviation from current criteria, as well as any viable corrective measures, will be documented for later considera-tion when the overall plant safety assessment (as outlined in Step 2) is made.

When deviations from current licensing approaches are identified, the following alternatives (or combinations of alternatives) will be considered as a basis for establishing acceptability:

1. The deviation can be justified as not significantly decreasing the level of safety, i.e., the probability and consequences of events are sufficiently low.

2. Use of non-safety systems to perform safety functions.
3. Administrative or procedural changes to enhance system reliability
4. Augmented surveillance programs
5. Selected backfitting to enhance system reliability

An evaluation of relevant operating experience will also be included in each step of this process, as appropriate. This evaluation will include a review of licensee operating and event reports for indications of needed changes to reduce the likelihood of system or component failures. Favorable operating experience will also be considered in assessing deviations from current criteria where appropriate. The elements of the periodic (ten year) review recommended by the ACRS will be included for the plants reviewed during Phase II, but will be performed and documented in a manner to support the SEP review process.

Each design basis event will be evaluated as described above and any potential corrective measures will be documented for consideration when the overall plant safety assessment is made.

Step 2 After the preliminary assessment of each individual design basis event as outlined in Step 1 is completed, an overall plant safety assessment will be made. The effect of corrective measures identified during Step 1 for each individual event evaluated will be examined to determine the integrated effect of the preliminary assessment from all other design basis events. This will provide assurance that measures taken to assure an acceptable response to one event will not adversely affect the plant's response to some other event, that backfitting decisions can be balanced decisions, and that the most efficient approach to any necessary plant upgrading is taken. After potential conflicts in system functions have been identified and resolved, the alternatives listed in Step 1 will be considered and overall integrated decisions on the plant as a whole can be made on the appropriate corrective action.

The above procedure will provide a basis for making balanced backfit decisions because of the perspective gained by considering all identified plant deficiencies and alternate capabilities.

Deviations from current criteria will be deemed acceptable if the staff evaluation shows that the plant will respond satisfactorily to all design basis events; i.e. that the probability of design basis events are not significantly higher than for a facility licensed in accordance with current criteria and the consequences of their occurrence are within the guideline values of 10 CFR Part 100.

The SEP review will be somewhat more limited in scope than a complete construction permit (CP)/operating license (OL) review based on the complete set of licensing requirements. (For example, safety related features which are known to have been designed to criteria comparable to those currently used will not be re-evaluated. Also some review will be done utilizing on-site inspections, review of operating experience and comparisons of similar systems and components in lieu of detailed descriptions and analyses required for a CP/OL review.). Only criteria related to the safety issues on the Topic List plus those that specify system functions and plant response to the design basis events will be examined. Although the review will be considerably smaller in scope than the current CP and OL review, it will be sufficient to ascertain whether the facility satisfies the General Design Criteria and other applicable regulations.

IV. RESOURCE AND SCHEDULE CONSIDERATIONS

NRR considers the Systematic Evaluation Program to be a significant, important task and as such will staff it accordingly. Based on a consideration of the scope of the program and on any potential interference with other ongoing programs, the eleven reviews of Phase II are scheduled for completion within three years. This schedule will enable the assignment of sufficient manpower in each engineering discipline without significantly impacting other high priority efforts, including the ongoing generic technical activities reviews.

Licensee resource requirements will be minimized and will be limited by a staff screening and evaluation of available docketed information prior to requesting additional information from the licensee. To assure that information requests are limited to that necessary for completing an SEP review, information requests will be tailored for each facility.

V. SUMMARY

The NRC has determined that the Systematic Evaluation Program is a high priority task that should proceed expeditiously for the eleven identified older operating facilities. This program will provide:

a. A reassessment of the overall safety margins at each facility.
b. A rational basis for comparison with current licensing criteria, and a basis for accepting deviations from current criteria.
c. A means of identifying systems interactions in a manner compatible with the generic evaluation of system interaction being performed in the NRR Technical Activities Program.
d. An integrated and balanced approach to backfit considerations as necessary.
e. Comprehensive documentation which addresses the identified safety issues, without performing a complete CP/OL type review of each plant.

ATTACHMENT 1

PLANTS PROPOSED FOR PHASE II OF SEP

PLANT NAME*	DATE OF POL	DATE OF FTOL
Dresden 1	09/28/59	10/14/60
Yankee Rowe	07/09/60	06/23/61
Big Rock Point	08/30/62	05/01/64
San Onofre 1	03/27/67	
Conn Yankee	06/30/67	12/27/74
La Crosse	07/03/67	
Oyster Creek	04/09/69	
R. E. Ginna	09/19/69	
Dresden 2	12/22/69	
Millstone 1	10/07/70	
Palisades	03/21/71	

*The first six plants including two with provisional operating licenses, POL, were originally proposed for review during Phase II. The last five plants are facilities operating under a POL, which are recommended additions to Phase II.

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ATTACHMENT 2

TOPIC LIST FOR SEP REVIEW

(10/11/77)

I Introduction

This comprehensive Topic List provides the basis for the systematic evaluation of operating reactors. The Topic List was derived from the following listings of items:

(1) TSAR-264: Technical Safety Activities Report, DSS update, 264 items, December 1975

(2) Encl B-324: Memo Eisenhut to Stello (2/15/77) - Prioritization of NRR Technical Generic Activities, 324 items

(3) TFL-118: Task Force Report on the Systematic Evaluation of Operating Nuclear Power Plants, Appendix B, November 1976, 118 items

(4) DOT-102: List of 102 items prepared by DOR in March 1977

The above lists were compiled and organized in the general format of the Standard Review Plan (SRP) with respect to chapter identification.

Following the compilation and organization of the above listed items, a culling procedure was initiated. Many topics were categorically removed from the list based on the following criteria:

1. Not related to operating LWR's and therefore not within the scope of SEP review (e.g., topics related to HGTR, LMFBR, off-shore plants, Research and Test Reactors and unlicensed facilities).

2. Research and Development programs in progress (e.g. evaluation of acoustic emission and other advanced NDE techniques)

3. Refinement of evaluation techniques (includes methodology and data collection to enhance staff capabilities but does not directly affect reactor performance or design).

4. Suggested topics which were too general (e.g. review geology) and non-safety related issues.

5. Duplications

6. Topics which the NRC is considering but is not yet implementing on new facilities.

7. Topics which have been reviewed and appropriately implemented.

8. Topics which are reviewed on a periodic basis in accordance with current criteria.

9. Topics of "Lesser Safety Significance" (Bases for deletion of all topics in this category have been written.)

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10. Combination of similar topics in the preparation of topic definitions.

This Topic List includes generic issues presently under review, those for which there is an interim NRC position, and generic issues not being actively investigated. These topics have been identified by an * in the left margin. They are retained on the list for SEP followup and appropriate evaluation and integration into plant specific reviews but SEP will not be responsible for development of positions on generic issues. Generic issues which involve long term research and development programs have not been included on the Topic List.

Figure 1 graphically displays the evolution of the Comprehensive Topic List.

ATTACHMENT 2

EVOLUTION OF SEP TOPIC LIST

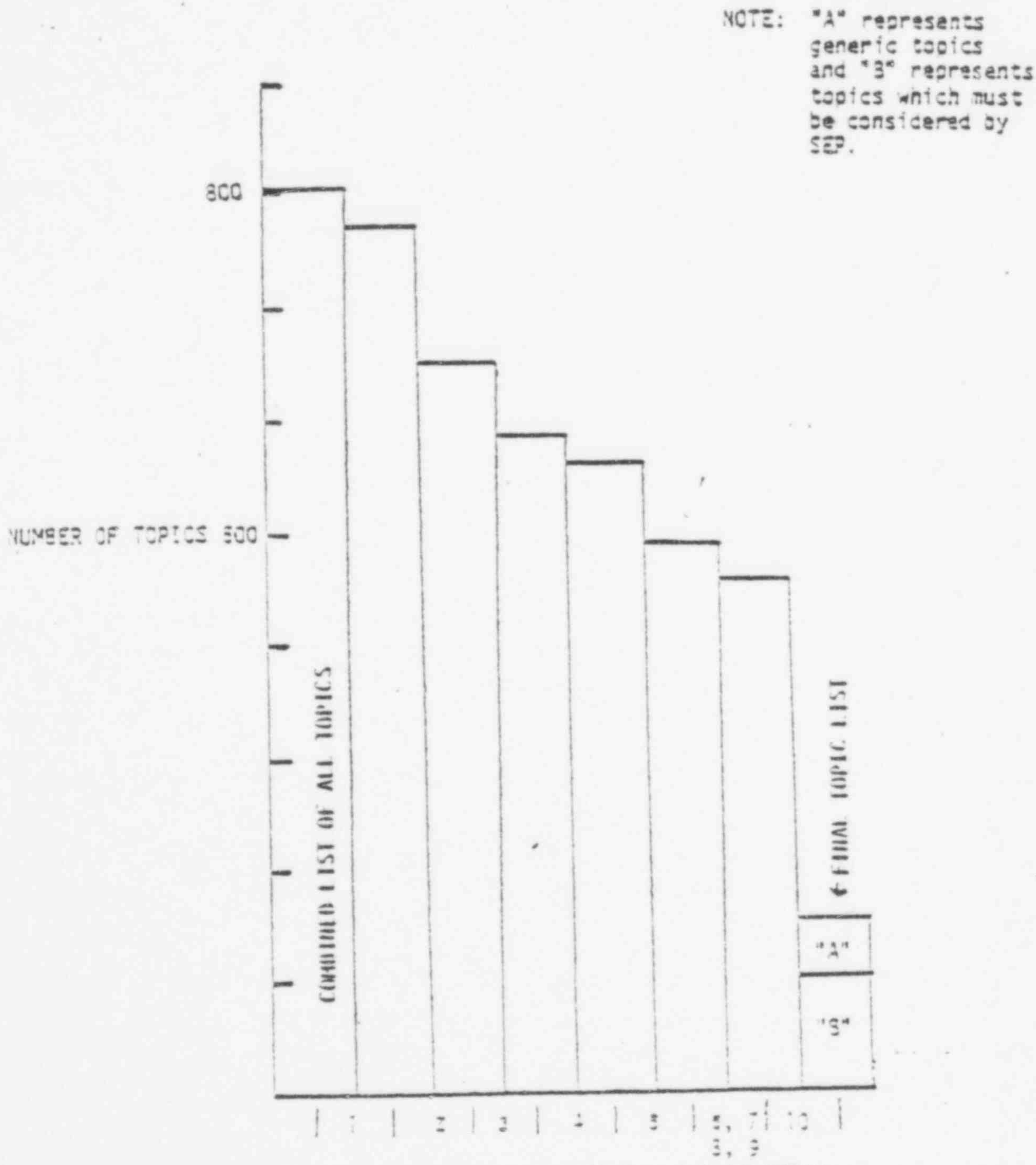

FIGURE 1

II SITE CHARACTERISTICS

II-1 Site

A. Exclusion Area Authority and Control

B. Population Distribution

C. Potential Hazards or Changes in Potential Hazards Due to Transportation, Institutional, Industrial, and Military Facilities

II-2 Meteorology

A. Severe Weather Phenomena

*B. Onsite Meteorological Measurements Program

C. Atmospheric Transport and Diffusion Characteristics for Accidents Analysis

D. Availability of Meteorological Data in the Control Room

II-3 Hydrology

A. Hydrologic Description

B. Flooding Potential and Protection Requirements

1. Capability of Operating Plant to Cope with Design Basis Flooding Conditions

C. Safety-related Water Supply (Ultimate Heat Sink (UHS))

*Generic issues under review.

ATTACHMENT 2

II-4 Geology and Seismology

*A. Tectonic Province

B. Proximity of Capable Tectonic Structures in Plant Vicinity

C. Historical Seismicity within 200 Miles of Plant

D. Stability of Slopes

E. Dam Integrity

F. Settlement of Foundations and Buried Equipment

*Generic issues under review.

ATTACHMENT 2

III DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

III-1 Classification of Structures, Components and Systems (Seismic and Quality)

III-2 Wind and Tornado Loadings

III-3 Hydrodynamic Loads

A. Effects of High Water Level on Structures

B. Structural and Other Consequences (e.g. Flooding of Safety-Related Equipment in Basements) of Failure of Underdrain Systems

C. Inservice Inspection of Water Control Structures

III-4 Missile Generation and Protection

A. Tornado Missiles

B. Turbine Missiles

C. Internally Generated Missiles

D. Site Proximity Missiles (Including Aircraft)

III-5 Evaluation of Pipe Breaks

*A. Effects of Pipe Break on Structures, Systems and Components Inside Containment

*B. Pipe Break Outside Containment

III-6 Seismic Design Considerations

*Generic issues under review.

ATTACHMENT 2

III-7 Category I Structures Integrity

*A. Inservice Inspection, Including Prestressed Concrete Containments with Either Grouted or Ungrouted Tendons

*B. Design Codes, Design Criteria, Load Combinations, and Reactor Cavity Design Criteria

C. Delamination of Prestressed Concrete Containment Structures

D. Containment Structural Integrity Tests

III-8 Reactor Vessel Internals Integrity

*A. Loose Parts Monitoring and Core Barrel Vibration Monitoring

*B. Control Rod Drive Mechanism Integrity

C. Irradiation Damage, Use of Sensitized Stainless Steel and Fatigue Resistance

*D. Core Supports and Fuel Integrity

*III-9 Support Integrity

III-10 Pumps and Valves Integrity

A. Thermal-Overload Protection for Motors of Motor-operated Valves

B. Pump Flywheel Integrity

*C. Surveillance Requirements on BWR Recirculation Pumps and Discharge Valves

III-11 Component Integrity

III-12 Environmental Qualification of Safety Related Equipment

*Generic issues under review.

ATTACHMENT 2

IV REACTOR

IV-1 Thermal Hydraulic Design and Performance

*A. Operation With Less Than All Loops In Service

IV-2 Reactivity Control Systems Including Functional Design and Protection Against Single Failures

IV-3 BWR Jet Pumps Operating Indications

*Generic issues under review.

ATTACHMENT 2

V REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

*V-1 Compliance with Codes and Standards (10 CFR 50.55a)

*V-2 Applicable Code Cases

*V-3 Overpressurization Protection

*V-4 Piping and Safe End Integrity

V-5 Reactor Coolant Pressure Boundary (RCPB) Leakage. Detection

*V-6 Reactor Vessel Integrity

*V-7 Reactor Coolant Pump Overspeed

*V-8 Steam Generator (SG) Integrity

V-9 Reactor Core Isolation Cooling System (BWR)

V-10 Residual Heat Removal (RHR) System

A. RHR Heat Exchanger Tube Failures

*B. RHR Reliability

V-11 High Pressure/Low Pressure Interface

*A. Requirements for Isolation of High and Low Pressure Systems

B. RHR Interlock Requirements

V-12 Reactor Water Cleanup System (BWR)

A. Water Purity of Boiling Water Reactor Primary Coolant

*V-13 Water Hammer

*Generic issues under review.

ATTACHMENT 2

VI ENGINEERED SAFETY FEATURES

VI-1 Organic Materials and Post Accident Chemistry

VI-2 Containment Functional Design

*A. Pressure-Suppression Type BWR Containments

*B. Subcompartment Analysis

*C. Ice Condenser Containment

*D. Mass and Energy Release for Possible Pipe Break Inside Containment

*VI-3 Containment Pressure and Heat Removal Capability

VI-4 Containment Isolation System

*VI-5 Combustible Gas Control

*VI-6 Containment Leak Testing

VI-7 Emergency Core Cooling System

A. Emergency Core Cooling System Performance

1. ECCS Re-evaluation to Account for Increased Vessel Head Temperature

*Generic issues under review.

ATTACHMENT 2

2. Upper Plenum Injection

3. ECCS Actuation System

*4. Core Spray Nozzle Effectiveness

B. ESF Switchover from Injection to Recirculation Mode (Automatic ECCS Realignment)

*C. ECCS Single Failure Criterion and Requirements for Locking Out Power to Valves Including Independence of Interlocks on ECCS Valves

1. Appendix K - Electrical Instrumentation and Control (EIC) Re-reviews

2. Failure Mode Analysis - ECCS

3. The Effect of PWR Loop Isolation Valve Closure During a LOCA on ECCS Performance

D. Long Term Cooling - Passive Failures (e.g., Flooding of Redundant Components)

*E. ECCS Sump Design and Test for Recirculation Mode Effectiveness

F. Accumulator Isolation Valves Power and Control System Design

VI-8 Control Room Habitability

*Generic issues under review.

VI-9 Main Steam Isolation

A. Main Steam Line Isolation Seal System - BWR

VI-10 Selected Engineered Safety Features (ESF) Aspects

A. Testing of Reactor Trip System and Engineered Safety Features Including Response Time Testing

B. Shared Engineered Safety Features, On-site Emergency Power, and Service Systems for Multiple Unit Facilities

ATTACHMENT 2

VII INSTRUMENTATION AND CONTROLS

VII-1 Reactor Trip Systems (IEEE-279)

A. Isolation of Reactor Protection System from Non-safety Systems, Including Qualifications of Isolation Devices

B. Trip Uncertainty and Setpoint Analysis Review of Operating Data Base

VII-2 Engineered Safety Features (ESF) System Control Logic and Design

VII-3 Systems Required for Safe Shutdown

*VII-4 Effects of Failure in Non-safety Related Systems on Selected Engineered Safety Features

*VII-5 Instruments for Monitoring Radiation and Process Variables During Accidents

*VII-6 Frequency Decay

VII-7 Acceptability of Swing Bus Design on BWR-4 Plants

*Generic issues under review.

ATTACHMENT 2

VIII ELECTRIC POWER

VIII-1 Offsite Power Systems

*A. Potential Equipment Failures Associated with a Degraded Grid Voltage

*VIII-2 Onsite Emergency Power Systems - Diesel Generator

VIII-3 Emergency DC Power Systems

*A. Station Battery Capacity Test Requirements

B. DC Power System Bus Voltage Monitoring and Annunciation

*VIII-4 Electrical Penetrations of Reactor Containment

IX AUXILIARY SYSTEMS

*IX-1 Fuel Storage

*IX-2 Overhead Handling Systems - Cranes

IX-3 Station Service and Cooling Water Systems

IX-4 Boron Addition System

IX-5 Ventilation Systems

*IX-6 Fire Protection

*Generic issues under review.

ATTACHMENT 2

X AUXILIARY FEEDWATER SYSTEM

XI RADIOACTIVE WASTE MANAGEMENT

XI-1 Appendix I

*XI-2 Radiological (Effluent and Process) Monitoring Systems

XII (Section on RADIATION PROTECTION Intentionally Left Blank)

XIII OPERATIONS

XIII-1 Conduct of Operations

*XIII-2 Safeguards/Industrial Security

XIV (Section on STARTUP TESTS AND CRITICALITY Intentionally Left Blank)

*Generic issues under review.

ATTACHMENT 2

XV ACCIDENTS AND TRANSIENTS

XV-1 Decrease in Feedwater Temperature, Increase in Feedwater Flow, Increase in Steam Flow and Inadvertent Opening of a Steam Generator Relief or Safety Valve

XV-2 Spectrum of Steam System Piping Failures Inside and Outside of Containment (PWR)

XV-3 Loss of External Load, Turbine Trip, Loss of Condenser Vacuum, Closure of Main Steam Isolation Valve (BWR), and Steam Pressure Regulator Failure (Closed)

XV-4 Loss of Non-Emergency A-C Power to the Station Auxiliaries

XV-5 Loss of Normal Feedwater Flow

XV-6 Feedwater System Pipe Breaks Inside and Outside Containment (PWR)

XV-7 Reactor Coolant Pump Rotor Seizure and Reactor Coolant Pump Shaft Break

XV-8 Control Rod Misoperation (System Malfunction or Operator Error)

XV-9 Startup of an Inactive Loop or Recirculation Loop at an Incorrect Temperature, and Flow Controller Malfunction Causing an Increase in BWR CORE FLOW RATE

XV-10 Chemical and Volume Control System Malfunction that Results in a Decrease in the Boron Concentration in the Reactor Coolant (PWR)

ATTACHMENT 2

*XV-11 Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position

XV-12 Spectrum of Rod Ejection Accidents (PWR)

XV-13 Spectrum of Rod Drop Accidents (BWR)

XV-14 Inadvertent Operation of ECCS and Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory

XV-15 Inadvertent Opening of a PWR Pressurizer Safety/Relief Valve or a BWR Safety/Relief Valve

XV-16 Radiological Consequences of Failure of Small Lines Carrying Primary Coolant Outside Containment

XV-17 Radiological Consequences of Steam Geneator Tube Failure (PWR)

XV-18 Radiological Consequences of Main Steam Line Failure Outside Containment (BWR)

XV-19 Loss-of-Coolant Accidents Resulting from Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary

XV-20 Radiological Consequences of Fuel Damaging Accidents (Inside and Outside Containment)

*XV-21 Spent Fuel Cask Drop Accidents

*XV-22 Anticipated Transients Without Scram

XV-23 Multiple Tube Failures in Steam Generators

*XV-24 Loss of All A-C Power

XVI TECHNICAL SPECIFICATIONS

*XVII OPERATIONAL QA PROGRAM

*Generic issues under review.

ATTACHMENT 2