NUREG-75/087

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STANDARD REVIEW PLAN FOR THE REVIEW OF SAFETY ANALYSIS REPORTS FOR NUCLEAR POWER PLANTS

LWR EDITION

MAY 1980

OFFICE OF NUCLEAR REACTOR REGULATION
U. S. NUCLEAR REGULATORY COMMISSION

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INTRODUCTION

The Standard Review Plan (SRP) is prepared for the guidance of staff reviewers in the Office of Nuclear Reactor Regulation in performing safety reviews of applications to construct or operate nuclear power plants. The principal purpose of the SRP is to assure the quality and uniformity of staff reviews, and to present a well-defined base from which to evaluate proposed changes in the scope and requirements of reviews. It is also a purpose of the SRP to make information about regulatory matters widely available and to improve communication and understanding of the staff review process by interested members of the public and the nuclear power industry.

The safety review is primarily based on the information provided by an applicant in a Safety Analysis Report (SAR). Section 50.34 of 10 CFR Part 50 of the Commission's regulations requires that each application for a construction permit for a nuclear facility shall include a Preliminary Safety Analysis Report (PSAR), and that each application for a license to operate such a facility shall include a Final Safety Analysis Report (FSAR). The SAR must be sufficiently detailed to permit the staff to determine whether the plant can be built and operated without undue risk to the health and safety of the public. Prior to submission of an SAR, an applicant should have designed and analyzed the plant in sufficient detail to conclude that it can be built and operated safely. The SAR is the principal document in which the applicant provides the information needed to understand the basis upon which this conclusion has been reached.

Section 50.34 specifies, in general terms, the information to be supplied in a SAR. The specific information required by the staff for an evaluation of an application is identified in Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants - LWR Edition." The SRP sections are keyed to the Standard Format, and the SRP sections are numbered according to the section numbers in the Standard Format. Review plans have not been prepared for SAR sections that consist of background or design data which are included for information or for use in the review of other SAR sections.

The Standard Review Plan is written so as to cover a variety of site conditions and plant designs. Each section is written to provide the complete procedure and all acceptance criteria for all of the areas of review pertinent to that section. However, for any given application, the staff reviewers may select and emphasize particular aspects of each SRP section as is appropriate for the application. In some cases, the major portion of the review of a plant feature may be done on a generic basis with the designer of that feature rather than in the context of reviews of particular applications from utilities. In other cases a plant feature may be sufficiently similar to that of a previous plant so that a de novo review of the feature is not needed. For these and other similar reasons, the staff may not carry out in detail all of the review steps listed in each SRP section in the review of every application.

The individual SRP sections address, in detail, who performs the review, the matters that are reviewed, the basis for review, how the review is accomplished, and the conclusions that are sought. The safety review is performed by 18 branches. One of the objectives of the SRP is to assign the review responsibilities to the various branches and to define the sometimes complex interfaces between them. Each SRP section identifies the branch that has the primary review responsibility for that section. In some review areas the primary branch may require support and the branches that are assigned these secondary review responsibilities are also identified for each SRP section.

Each SRP is organized into four subsections as follows:

I. Areas of Review

This subsection describes the scope of review, i.e., what is being reviewed by the branch having primary review responsibility. This subsection contains a description of the systems, components, analyses, data, or other information that is reviewed as part of the particular Safety Analysis Report section in question. It also contains a discussion of the information needed or the review expected from other branches to permit the primary review branch to complete its review.

II. Acceptance Criteria

This subsection contains a statement of the purpose of the review and the technical basis for determining the acceptability of the design or the programs within the scope of the area of review of the SRP section. The technical bases consist of specific criteria such as NRC Regulatory Guides, General Design Criteria, Codes and Standards, Branch Technical Positions, and other criteria.

The technical bases for some sections of the SRP are provided in Branch Technical Positions or Appendices which are included in the SRP. These documents typically set forth the solutions and approaches determined to be acceptable in the past by the staff in dealing with a specific safety problem or safety-related design area. These solutions and approaches are codified in this form so that staff reviewers can take uniform and well-understood positions as the same safety problems arise in future cases. Some Branch Technical Positions and Appendices may be converted into Regulatory Guides if it appears that this step would aid the review process. Like Regulatory Guides, the Branch Technical Positions and Appendices represent solutions and approaches that are acceptable to the staff, but they are not required as the only possible solutions and approaches. However, applicants should recognize that, as in the case of Regulatory Guides, substantial time and effort on the part of the staff has gone into the development of the Branch Technical Positions and Appendices and that a corresponding amount of time and effort will probably be required to review and accept new or different solutions and approaches. Thus, applicants proposing other solutions and approaches to safety problems or safety-related design areas than those described in the Branch Technical Positions and Appendices must expect longer review times and more extensive questioning in these areas. The staff is willing to consider proposals for other solutions and approaches on a generic basis, apart from a specific license application, to avoid the impact of the additional review time on individual cases.

III. Review Procedures

This subsection discusses how the review is accomplished. The section is generally a step-by-step procedure that the reviewer goes through to provide reasonable verification that the applicable safety criteria have been met.

IV. Evaluation Findings

This subsection presents the type of conclusion that is sought for the particular review area. For each section, a conclusion of this type is included in the staff's Safety Evaluation Report in which the staff publishes the results of their review. The SER also contains a description of the review including such subjects as which aspects of the review were selected or emphasized; which matters were modified by the applicant, require additional information, will be resolved in the future, or remain unresolved; where the plant's design or the applicant's programs deviate from the criteria stated in the SRP; and the bases for any deviations from the SRP or exemptions from the regulations.

V. References

This subsection lists the references used in the review process.

The SRP and the Standard Format are directed toward water-cooled reactor power plants. Staff reviewers will adapt the SRP for use in the reviews of other reactor types where applicable.

The Standard Review Plans result from many years of experience by the staff in establishing and using regulatory requirements in evaluating the safety of nuclear power plants and in reviewing Safety Analysis Reports. A great deal of progress has been made in the methods of review and in the development of regulations, guides and standards since the early years of review. This Standard Review Plan may be considered a part of a continuing regulatory standards development activity that not only documents current methods of review, but provides the base of orderly modifications of the review process in the future.

The SRP will be revised and updated periodically as the need arises to clarify the content or correct errors and to incorporate modifications approved by the Director of the Office of Nuclear Reactor Regulation. As necessary, corresponding changes to the Standard Format will also be made. Comments and suggestions for improvement will be considered and should be sent to the Director, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, DC 20555. Notices of errors or omissions should also be sent to the same address.

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SECTION 1.8 INTERFACES FOR STANDARD DESIGNS

REVIEW RESPONSIBILITIES

Primary - All review branches (except Quality Assurance Branch (QAB))

Secondary - None

I. AREAS OF REVIEW

The areas of review include all the safety-related interfaces, important to the staff review for safety, that exist between the systems, components and structures within a standard design as they relate to matching systems, components and structures within the remaining unspecified portion of the plant design. Also included is the implementation of interface requirements in the design of matching safety-related systems, components and structures. The specific interfaces identified for the nuclear steam supply system (NSSS) and for the balance-of-plant (BOP) portions of a nuclear plant design are given in "Interfaces for Standard Designs," Appendix A to Regulatory Guide 1.70.

II. ACCEPTANCE CRITERIA

The acceptance criteria for interfaces, as appropriate, are contained in the sections of the SRP applicable to the particular area under review. While these acceptance criteria are not specifically identified as pertinent to interfaces, they nevertheless apply to the interfaces that are encompassed by the review area to which the criteria apply.

III. REVIEW PROCEDURES

The reviewer in each responsible review branch will select and emphasize material from this review plan, as may be appropriate for a particular case. The particular interfaces to be addressed in SSARs describing standard NSSS and BOP designs and in SARs describing the entire plant design in support of licenses are presented in Regulatory Guide 1.70 Appendix A.

1. Standard designs for the NSSS are reviewed to assure that the applicable safety-related interfaces to, and in several instances assumed for, the matching, but unspecified, BOP, site, and utility portions are identified and defined.

2. Standard designs for the BOP are reviewed to assure that (a) those applicable safety-related interfaces established for the NSSS are addressed, (b) those interfaces assumed by the NSSS are compatible with the BOP design, and (c) the applicable safety-related interfaces to the matching, or unspecified, site, and utility portion are identified and defined.

3. Utility applications are reviewed to assure that the applicable safety-related interfaces established for the referenced standard NSSS design alone, or the referenced standard NSSS and BOP designs combined, are addressed.

IV. EVALUATION FINDINGS

Each review branch verifies that sufficient interface information has been provided and properly implemented, as appropriate, and that the review is adequate to support conclusions of the following type, to be included in the staff's safety evaluation report:

For standard design reviews:

"Interfaces provide the means for assuring that the standard design and other unspecified portions of the plant design are compatible with regard to the performance of all safety-related systems, components, and structures under all modes of operation and shutdown. The scope of our review included an evaluation of the interfaces defined by the applicant in accordance with the guidelines given in Regulatory Guide 1.70, Appendix A, and the applicable criteria for acceptance given in the appropriate sections of the SRP. Based on our evaluation, we find the interface information provided in the standard design application acceptable."

For applications referencing standard designs:

"Interfaces established for the standard design are addressed in the referencing application to demonstrate the compatibility of the standard design with the remaning portions of the plant design regarding the performance of all safety-related systems, components and structures under all modes of operation and shutdown. The scope of our review included an evaluation of the matching safety-related systems, components and structures to assure compatibility in accordance with the applicable criteria for acceptance given in the appropriate sections of the SRP. Based on our evaluation, we find that the interfaces have been properly addressed and that compatibility of the standard design with the remaining portions of the plant design can be achieved."

V. REFERENCES

1. "Interfaces for Standard Designs," Regulatory Guide 1.70, Appendix A.

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SECTION 2.1.1. SITE LOCATION AND DESCRIPTION

REVIEW RESPONSIBILITIES

Primary - Accident Analysis Branch (AAB)

Secondary - None

I. AREAS OF REVIEW

Reactor location is reviewed (1) as identified by latitude and longitude and by the UTM* coordinate system; (2) with respect to political subdivisions; and (3) with respect to prominent natural and man-made features of the area to ascertain the accuracy of the applicant's safety analysis report (SAR) description and for use in independent reviews of the surrounding population (SRP Section 2.1.3) and nearby man-made hazards (SRP Section 2.2).

The site area which contains the reactors and associated principal plant structures is reviewed to determine the distance from the reactor to boundary lines of the exclusion area, including the direction and distance from the reactor to the nearest exclusion area boundary line. A scaled plot plan of the exclusion area is reviewed which permits distance measurements to the exclusion area boundary in each of the 22-1/2 degree segments centered on the 16 cardinal compass points. The location and orientation of plant structures within the exclusion area are reviewed to identify potential release points and their distances to exclusion area boundary lines. The location, distance, and orientation of plant structures with respect to highways, railways, and waterways which traverse or lie adjacent to the exclusion area are reviewed to assure that they are adequately described to permit analyses (SRP Section 2.2) of the possible effects on the plant of accidents on these transportation routes.

II. ACCEPTANCE CRITERIA

The plant exclusion area and the location of the plant within the area are described in sufficient detail to allow a determination (in SRP Section 2.1.2 and those in Section 15) that Part 100 is met.

Highways, railways, and waterways which traverse the exclusion area are sufficiently distant from plant structures so that routine use of these routes is not likely to interfere with normal plant operation (Ref. 2).

*Universal Transverse Mercator coordinate system as found on USGS topographical maps.

Information included in this SAR section should allow two types of safety analyses to be conducted. The first addresses the consequences in the unlikely event that a serious release of radioactive material should occur. The second addresses the effect that accidents on, or routine use of, routes on or near the site will have on the operation of the plant. Adequacy of the data for these purposes should be decided jointly with the AAB reviewers having primary responsibilities for the particular analyses involved.

III. REVIEW PROCEDURES

Selection and emphasis of various aspects of the areas covered by this SAR section will be made by the reviewer on each case. The judgment on the areas to be given attention during the review is to be based on an inspection of the material presented, the similarity of the material to that recently reviewed on other plants, and whether items of special safety significance are involved.

The information in this section of the SAR forms the basis for evaluations performed in various other sections. The purpose of this review is to establish the validity of the basic data, to check the UTM coordinates to assure that they include the zone number, and that the Northing and Easting are presented to within 100 meters. The latitude and longitude should be checked to assure that they are expressed to the nearest second.

Cross-check the exclusion area distances with the distances used in the Accident Analyses, SAR Section 15. Scale the map provided to check distances specified in the SAR and to determine the distance-direction relationships to exclusion area boundaries, roads, railways, waterways, and other significant features of the area. At the operating license stage, the location and orientation of plant structures and effluent release points with respect to the exclusion area and plant property boundaries, transportation routes and political subdivisions will be reviewed to identify any changes since the construction permit (CP) review. Where changes have occurred, new analyses may be required to ensure that the findings reached during the CP review are not affected by these changes.

If, in the reviewer's judgment, maps of larger scale are desirable, they may be obtained from the U.S. Geological Survey (USGS). The USGS map index should be consulted for the specific names of the 7-1/2 minute quadrangles that bracket the site area. If available, these maps provide topographic information in addition to details of prominent natural and man-made features in the site area. This information may be supplemented by updated information as available, e.g., aerial photographs or information obtained on the site visit. Check the plant layout to determine that the orientation of plant structures with respect to nearby roads, railways, and waterways is clearly shown. Check to see that there are no obvious ways in which transportation routes which traverse the exclusion area can interfere with normal plant operations.

Site Visit

A visit to the site under review permits a better understanding of the physical characteristics of the site and its relationship to the surrounding area. It permits the reviewer to gather information, independent of that supplied in the Safety Analysis Report, which is useful in confirming SAR data.

Site visits should be made after initial review of the site data in the SAR has been completed and the reviewer has become generally familiar with the site and surrounding areas. Since one of the purposes of the site visit is to discuss the preliminary review findings with the applicant, the reviewer should plan to be in the site area one or two days in advance of the scheduled meeting with the applicant. This will permit gathering information from visits to local offices of federal, state, and county governments, industries, military facilities, etc. Specific visits to these offices should be made on the basis of the particular site characteristics and is left to the judgment of the individual reviewer. The reviewer should note that some of the local offices may have been contacted by the environmental reviewer. Generally, information sought by the respective reviewers is similar in scope but will differ in emphasis. To avoid duplication of visits to local officials, the reviewer should contact the Environmental Project Manager and, where feasible, arrange for a joint visit to those local offices in which there is a common interest. Sources investigated should include such local and state agencies as those concerned with population and land use and land use controls (zoning boards). County engineers are sources of information on public roads and traffic volumes. Local Councils of Government may have information on population growth, proposed new industries or transportation routes. Information sought should encompass, whenever possible, data in support of the review procedures for SRP Sections 2.1.3, 2.2.1 and 2.2.2.

If information gathered indicates the need for clarification of data contained in the SAR, this should be discussed with the applicant in the subsequent meeting on preliminary review findings.

IV. EVALUATION FINDINGS

Summary descriptions of the site location, the site itself, and transportation routes on or near the site will be prepared for the staff safety evaluation report. Any deficiencies of site parameters with respect to the proposed plant will be noted.

V. REFERENCES

1. U.S. Geological Survey Topographical Map Indices (one for each state).

2. 10 CFR Part 100, "Reactor Site Criteria," Section 100.3(a).

3. AEC Manual Appendix 0621, "Damage Assessment Handbook," Part III, "Universal Transverse Mercator Coordinate System."

4. Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants."

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SECTION 2.1.2 EXCLUSION AREA AUTHORITY AND CONTROL

REVIEW RESPONSIBILITIES

Primary - Accident Analysis Branch (AAB)

Secondary - Office of the Executive Legal Director (OELD)
Hydrology-Meteorology Branch (HMB)

I. AREAS OF REVIEW

The applicant's legal authority to determine all activities within the designated exclusion area is reviewed. 10 CFR §100.3(a) requires that a reactor licensee have authority to determine all activities within the designated exclusion area, including the exclusion and removal of personnel and property.

In any case where the applicant does not own all the land, including mineral rights, within the designated exclusion area, assistance may be required of OELD in determining whether or not the designated exclusion area meets the requirements of 10 CFR Part 100. Also, in some cases public roads which lie within the proposed exclusion area may have to be abandoned or relocated to permit plant construction. OELD assistance may be required to assure that no legal impediments to such abandonment or relocation are likely to ensue. Part 100 permits the exclusion area to be traversed by a highway, railway, or waterway provided arrangements are made to control these areas in event of an emergency.

Activities that may be permitted within the designated exclusion area, and that will not be related to routine operation of the plant, are reviewed. Review should include the type of activity, its specific location within the exclusion area, the number and kinds of persons engaged in the activity, and the frequency and length of time the activities are to be permitted.

II. ACCEPTANCE CRITERIA

Prior to issuance of a construction permit or limited work authorization, the applicant must demonstrate that it has the authority within the exclusion area as required by Part 100.3(a), or must provide reasonable assurance that it will have such authority prior to start of construction. Absolute ownership of all lands within the exclusion area, including mineral rights, is considered to carry with it the required authority to determine all activities on this land and is acceptable.

Where the required authority is contingent upon future procurement of ownership (e.g., by eminent domain proceedings), or by lease, easement, contract, or other means, the exclusion area may be acceptable if OELD can determine that the information provided by the applicant provides reasonable assurance that the required authority will be obtained prior to start of construction. In cases where ownership and control is to be acquired or completed during a construction period, a special review by OELD will be required. Also, in cases of proposed public road abandonment or relocation, OELD should determine that there is sufficient authority or that sufficient arrangements have been made to accomplish the proposed relocation or abandonment. The nature and extent of those activities which will be conducted within the exclusion area, and which are or will not be related to plant operation, are such that appropriate measures to evacuate persons engaged in those activities can be taken in the event of an accident.

Where the designated exclusion area extends into bodies of water such as a lake, reservoir, or river which is routinely accessible to the public, the reviewer must determine that the applicant has made appropriate arrangements with the local, state, Federal, or other public agency having authority over the particular body of water and the arrangements made provide for the exclusion and ready removal in an emergency, by either the applicant or the public agency in authority, of any persons on those portions of the body of water which lie within the designated exclusion area.

Section V contains several references to decisions made by Atomic Safety and Licensing Boards (ASLB) and Atomic Safety and Licensing Appeal Boards (ASLAB) which deal with exclusion area determinations in contested cases.

III. REVIEW PROCEDURES

Selection and emphasis of various aspects of the areas covered by this review plan will be made by the reviewer on each case. The judgment on the areas to be given attention during the review is to be based on an inspection of the material presented, the similarity of the material to that recently reviewed on other plants, and whether items of special safety significance are involved.

The reviewer should determine the basis on which the applicant claims authority within the exclusion area. If absolute ownership of all lands, including mineral rights, within the area is demonstrated, the acceptance criteria are satisfied. If any other method is claimed as providing the required authority, a memorandum should be prepared for OELD containing all of the appropriate information in the SAR, including copies of applicable SAR pages and figures, and requesting a written response as to whether or not the applicant's claimed authority meets the requirements of 10 CFR §100.3(a). In any case where there are technical reasons which the reviewer believes make the applicant's proposed method unacceptable, these reasons should be described and discussed in the memorandum. If the exclusion area extends into a body of water such as a lake, reservoir, or river, the area of the body of water encompassed should be reviewed against the guidelines of Part 100 regarding control of access and activities unrelated to operation of the reactor. The extent of the exclusion area over a waterway must be reviewed on a case-by-case basis.

The memorandum should also include information in the PSAR which describes the applicant's plans, procedures, and schedule for obtaining any abandonment or relocation of public roads which may be required. At the operating stage, review will emphasize those areas where the applicant did not possess absolute authority at the construction permit review.

If the designated exclusion area is traversed by a highway, railway, waterway, or other transportation route accessible to the public, the reviewer should determine that the applicant's emergency plan includes adequate provisions for control of traffic on these routes in the event of an emergency. At the construction permit stage, a finding that such provisions are feasible is adequate.

If activities unrelated to plant operation are to be permitted within the exclusion area, it will be nesessary to determine that the potential radiation exposures to persons engaged in these activities resulting from the design basis accidents postulated and evaluated in SAR Section 15 do not exceed the guidelines of 10 CFR Part 100. While the same method and model is to be used as was used to calculate the 2-hour exclusion area boundary dose, the dose may be calculated based on the projected duration of exposure of affected personnel assuming concentration. Appropriate X/Q values for the distances and directions to the activities should be obtained from HMB.

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information has been provided, and that his evaluation is sufficiently complete and adequate to support conclusions of the following type, to be included in the staff's safety evaluation report:

> "The applicant has described the plant exclusion area, the authority under which all activities within the exclusion area can be controlled, and the methods by which access and occupancy of the exclusion area can be controlled during normal operation and in the event of an emergency situation. The applicant has the required authority to control activities within the designated exclusion area, including the exclusion and removal of persons and property, and has established acceptable methods for control of the designated exclusion area. It is concluded that the exclusion area meets the criteria of 10 CFR Part 100, and is acceptable."

V. REFERENCES

1. 10 CFR Part 100, "Reactor Site Criteria."

2. Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," Revision 2.

3. NUREG-0386, United States Nuclear Regulatory Commission Staff Practice and Procedure Digest (Section 6.3(1), Exclusion Area Controls).

4. The Cleveland Electric Illuminating Company, et al. (Perry Nuclear Power Plant, Units 1 and 2), "Supplemental Partial Initial Decision, Site Suitability and Environmental Matters," LBP-74-76, 8 AEC 701 (October 20, 1974).

5. Southern California Edison Company, et al. (San Onofre Nuclear Generating Station, Units 2 and 3), "Decision," ALAB-248, 8 AEC 951 (December 24, 1974).

6. Southern California Edison Company, et al. (San Onofre Nuclear Generating Station, Units 2 and 3), "Decision," ALAB-268 1-NRC 383 (April 25, 1975).

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SECTION 2.1.3 POPULATION DISTRIBUTION

REVIEW RESPONSIBILITIES

Primary - Accident Analysis Branch (AAB)

Secondary - None

I. AREAS OF REVIEW

Areas of the applicant's safety analysis report (SAR) relating to the population surrounding a nuclear facility reviewed are:

1. The present population (based on 1970 census data or more recent census data, if available), and a comparison of the applicant's projected population growth with independent projections made by other agencies such as the Census Bureau, Bureau of Economic Analyses, Environmental Protection Agency, and local and state agencies and Councils of Government.

2. Whether population density should be a significant consideration at the construction permit (CP) stage in alternate site evaluation. Present and projected transient populations, appropriately weighted by occupancy, should be included.

3. Within the specified low population zone (LPZ), the ability to take appropriate protective measures in behalf of the populace contained within the LPZ in the event of a serious accident.

4. The distance to the nearest boundary of the closest population center (as defined in Part 100), and its relationship to the low population zone outer boundary distance.

II. ACCEPTANCE CRITERIA

The data on present population in the region of the site are based on 1970 or more recent census data and population numbers check reasonably well against other independently-obtained population data, if available; e.g., Census Bureau Enumeration District tapes. The projected populations at the approximate year of plant startup and over the expected life of the plant are acceptable if they check reasonably well against independently-obtained population projections, if available; e.g., OBERS,[1/] BEA,[1/] or Water Resource Council.

1/OBERS is the descriptive title of a projection program conducted by the U.S. Department of Commerce former Office of Business Economics (OBE), now renamed the Bureau of Economic Analysis (BEA), and the Economic Research Service (ERS) of the U.S. Department of Agriculture.

Transient populations are included for those sites where a significant number of people (other than those just passing through the area) work, reside part-time, or engage in recreational activities, and are not permanent residents of the area.

The specified low population zone is acceptable if (a) it is determined that appropriate protective measures could be taken in behalf of the enclosed populace in the event of a serious accident; and (b) the nearest boundary of the closest population center (as defined in Part 100) is at least one and one third times the distance from the reactor to the outer boundary of the low population zone.

The population center distance is acceptable if there are no likely concentrations of greater than 25,000 people over the plant lifetime closer than the distance designated by the applicant as the population center distance.

III. REVIEW PROCEDURES

Selection and emphasis of various aspects of the areas covered by this SRP section will be made by the reviewer on each case. The judgment on the areas to be given attention during the review is to be based on an inspection of the material presented, the similarity of the material to that recently reviewed on other plants, and whether items of special safety significance are involved.

Determine that the population data contained in the SAR is in the detail and in the format described in Reference 3, Section 2.1.3.

Compare the SAR present population data against whatever independent population data is available (e.g., Census Bureau CED tapes, special census which may have been conducted, local and state agencies, Councils of Government, etc). Note any significant differences which require clarification.

Compare the SAR population projections against whatever independent population projections are available (e.g., local and state agencies and Councils of Government, Census Bureau projections, Bureau of Economic Analysis, etc). Note any significant underestimates in the SAR which require clarification.

At the construction permit stage, use the population and its distribution, including weighted transients, projected to the year of plant startup and projected over the lifetime of the plant, to determine the population density in persons per square mile as a function of distance from the plant out to 30 miles. Compare results to the SAR plot of population density vs distance (Reference 3, Section 2.1.3.6). If the population density, including weighted transient population, projected at the time of initial operation exceeds 500 persons per square mile averaged over any radial distance out to 30 miles (cumulative population at a distance divided by the area at that distance), or the projected population density over the lifetime of the facility exceeds 1,000 persons per square mile averaged over any radial distance out to 30 miles, a memorandum should be prepared advising

appropriate staff personnel that an evaluation of alternative sites having lower population densities will be required.

Determine that the SAR includes a map of the low population zone and a table of population distribution which includes transients (Reference 3, Section 2.1.3.4). Determine the method used by the applicant to establish the boundary of the nearest population center (Reference 3, Section 2.1.3.5). Evaluate communities which are closer to the plant than the designated population center to determine the likelihood that any of them can be projected to grow to 25,000 people within the plant lifetime. Compare the distance to the boundary of the population center to the distance to the outer boundary of the low population zone and establish that the population center distance is at least one and one-third times the low population zone distance as required by Part 100.

Population and population density data of specific towns and cities within the low population zone can be checked against population data as contained in the Department of Commerce publication, "1970 Census of Population - Characteristics of the Population," and other Census Bureau publications.

Site population is compared to previously licensed sites to determine the relative population characteristics of the proposed site. Curves showing current and projected population as a function of distance may be prepared for use in the staff's safety evaluation report (SER).

Determine that the current and projected population data for the LPZ includes transients (e.g., workers, occupants of schools, hospitals, etc., recreational facilities).

Determine the acceptability of the LPZ with respect to the necessary finding that there is reasonable assurance that appropriate protective measures could be taken in behalf of the people within the LPZ in the event of a serious accident. [Part 100, Section 100.3(b)].

Determine that the nearest boundary of the closest population center is at least one and one-third times the distance to the outer boundary of the low population zone. Evaluate the characteristics of the land area between the plant and the nearest population grouping which has, or is projected to have during plant lifetime, a population of about 25,000. Use whatever data is available on land use, land use controls such as zoning, potential for growth, or factors which are likely to limit growth between the population grouping and the plant to determine the potential growth in population density toward the site. The population center boundary should be established at that point nearest the plant where, in the reviewers judgment, the population density may grow to a value comparable to the density of the community itself. Population density is the controlling criteria, and in this regard, the corporate boundary of the community itself is not limiting. The detail to which this aspect of the site is reviewed will depend on the distance of the nearest probable population center relative to the distance to the outer boundary of the low population zone. (See References 4 and 5.) Where a very large city is involved, a greater distance than the one and one-third factor may be required, and appropriate additional compensating engineered safeguards may be required. These will be evaluated on a case-by-case basis.

Results of the review under this SRP section should be forwarded to the Division of Operating Reactors whenever the site contains a previously licensed and operating nuclear unit.

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information has been provided, and that his evaluation is sufficiently complete and adequate to support conclusions of the following type, to be included in the staff SER:

> "The present and projected populations surrounding the site, including transients, have been reviewed and comparison with independently obtained population data confirms the applicant's estimates.
>
> "On the basis of the specified low population zone and population center distance, and the calculated radiological consequences of design basis accidents at the outer boundary of the low population zone (Section 15), it is concluded that the low population zone and population center distance meet the guidelines of 10 CFR Part 100 and are acceptable."

V. REFERENCES

1. "1972 OBERS Projections," Vol. 1-5, U.S. Water Resources Council, Washington, D.C. (1972).
2. "1970 Census of Population, Characteristics of the Population," Vol. 1, Part A, Sections 1 and 2, Bureau of the Census, U.S. Department of Commerce (1972).
3. Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," Revision 2.
4. NUREG-0308 Safety Evaluation Report, Arkansas Nuclear One, Unit 2. November 1977 and supplements.
5. NUREG-75/054 Safety Evaluation Report, Pilgrim Nuclear Generating Station, Unit 2. June 1975 and supplements.

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SECTIONS 2.2.1 - 2.2.2 IDENTIFICATION OF POTENTIAL HAZARDS IN SITE VICINITY

REVIEW RESPONSIBILITIES

Primary - Accident Analysis Branch (AAB)

Secondary - None

I. AREAS OF REVIEW

Locations and separation distances from the site of industrial, military, and transportation facilities and routes in the vicinity of the site. Such facilities and routes include air, ground, and water traffic, pipelines, and fixed manufacturing, processing, and storage facilities. Potential external hazards or hazardous materials that are present or which may reasonably be expected to be present during the projected life time of the proposed plant. The purpose of this review is to establish the information concerning the presence of potential external hazards which is to be used in further review in Sections 2.2.3, 3.5.1.5, and 3.5.1.6.

II. ACCEPTANCE CRITERIA

1. Data in the SAR adequately describes the locations and distances of industrial, military, and transportation facilities in the vicinity of the plant, and is in agreement with data obtained from other sources, when available.

2. Descriptions of the nature and extent of activities conducted at nearby facilities, including the products and materials likely to be processed, stored, used, or transported, are adequate to permit evaluations of possible hazards in Part 3 review sections dealing with specific hazards.

3. Where potentially hazardous materials may be processed, stored, used, or transported in the vicinity of the plant, sufficient statistical data on such materials are provided to establish a basis for evaluating the potential hazard to the plant.

III. REVIEW PROCEDURES

Selection and emphasis of various aspects of the areas covered by this review plan will be made by the reviewer on each case. The judgment of the areas to be given attention during the review is to be based on an inspection of the material presented, the similarity of the material to that recently reviewed on other plants, and whether items of special safety significance are involved. The following procedures are followed:

1. The reviewer should be especially alert, in the construction permit(CP) stage review, for any potentially hazardous activities in close proximity of the plant, since the variety of activities having damage potential at ranges under about one kilometer can be very extensive. All identified facilities and activities within eight kilometers (5 miles) of the plant should be reviewed. Facilities and activities at greater distances should be considered if they otherwise have the potential for affecting plant safety-related features. At the operating license (OL) stage, most hazards will already have been identified. Emphasis should be placed on any new information. At the operating license stage, any analyses pertaining to potential accidents involving hazardous materials or activities in the vicinity of the plant will be reviewed to ensure that results are appropriate in light of any new data or experience which is then available. Facilities which are likely to either produce or consume hazardous materials should be investigated as possible sources of traffic of hazardous materials past the site.

2. Information should be obtained from sources other than the SAR wherever available, and should be used to check the accuracy and completeness of the information submitted in the SAR. This independent information may be obtained from sources such as U.S. Geological Survey (USGS) maps and aerial photos, published documents, contacts with state and federal agencies, and from other nuclear plant applications (especially if they are located in the same general area or on the same waterway.) Information should also be obtained during the site visit and subsequent discussions with local officials. (See Standard Review Plan 2.1.1 for further guidance with regard to site visits.) An attempt should be made to investigate future potential hazards over the proposed life of the plant.

3. The specific information relating to types of potentially hazardous material, including distance, quantity, and frequency of shipment, is reviewed to eliminate as many of the potential accident situations as possible by inspection, based on past review experience. At the operating license stage, nearby industrial, military and transportation facilities and transportation routes will be reviewed for any changes or additions which may affect the safe operation of the plant. If these changes alter the data or assumptions used in previous hazards evaluations or demonstrate the need for new ones, appropriate evaluations will be performed.

   For pipeline hazards, Reference 7 may be used as an example of an acceptable risk assessment. For cryogenic fuels, Reference 9 may be used, and for tank barge risks, Reference 8. For military aviation, Reference 10 may be used. Safe separation distances for explosives are identified in Reference 2, and for toxic chemicals, References 3 and 4 should be consulted.

   The distance from nearby railroad lines is checked to determine if the plant is within the range of a "rocketing" tank car which, from Reference 5, is taken to be 350 meters with the range for smaller pieces extending to 500 meters.

4. Potential accidents which cannot be eliminated from consideration as design basis events because the consequences of the accidents, if they should occur, could be serious enough to affect plant safety-related features, are identified. Potential accidents so identified are assessed in detail, using criteria in Standard Review Plan Sections 2.2.3, 3.5.1.5, or 3.5.1.6, as appropriate.

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information has been provided, and that his evaluation is sufficiently complete and adequate to support conclusions of the following type, to be used in the staff's safety evaluation report:

> "The nature and extent of activities involving potentially hazardous materials which are conducted at nearby industrial, military, and transportation facilities have been evaluated to identify any such activities which have the potential for adversely affecting plant safety-related structures. Based on evaluation of information contained in the SAR, as well as information independently obtained by the staff, it is concluded that all potentially hazardous activities in the vicinity of the plant have been identified. The hazards associated with these activities have been reviewed and are discussed in Sections _____ and _____ of this SER."

If the activities are identified as being potentially hazardous, the evaluations described in Standard Review Plans 2.2.3, 3.5.1.5 and 3.5.1.6 are performed with respect to the inherent capability of the plant or special plant design measures to prevent radiological releases in excess of the 10 CFR Part 100 guidelines.

V. REFERENCES

1. Department of the Army Technical Manual TM5-1300, "Structures to Resist the Effects of Accidental Explosions," June 1969.

2. Regulatory Guide 1.91, "Evaluation of Explosions Postulated to Occur on Transportation Routes Near Nuclear Power Plant Sites."

3. Regulatory Guide 1.78, "Assumptions for Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release."

4. Regulatory Guide 1.95, "Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release."

5. National Transportation Safety Board Railroad Accident Report, "Southern Railway Company, Train 154, Derailment with Fire and Explosion, Laurel, Mississippi, January 25, 1969," October 6, 1969.

6. Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," Revision 2.

7. NUREG-0014 Safety Evaluation Report, Hartsville Nuclear Plants A1, A2, B1, and B2, April 1976, Docket STN 50-518.

8. Safety Evaluation of the Beaver Valley Power Station, Unit No. 2 November 9, 1976 and supplements. Docket 50-412.

9. Safety Evaluation Report, Hope Creek Generating Station, Units 1 and 2, Supplement No. 5, March 1976, Docket 50-354 and 50-355.

10. Project 485, Aircraft Considerations, Preapplication Site Review, Boardman Nuclear Plant. October 1973.

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SECTION 2.2.3 EVALUATION OF POTENTIAL ACCIDENTS

REVIEW RESPONSIBILITIES

Primary - Accident Analysis Branch (AAB)

Secondary - Applied Statistics Branch (ASB/MPA)

I. AREAS OF REVIEW

The applicant's identification of potential accident situations in the vicinity of the plant is reviewed to determine the completeness of and the bases upon which these potential accidents were or were not accommodated in the design. (See Standard Review Plans 2.2.1 and 2.2.2.)

The applicant's probability analyses of potential accidents involving hazardous materials or activities in the vicinity of the plant, if such analyses have been performed, are also reviewed by ASB/MPA on request by AAB to determine that appropriate data and analytical models have been utilized.

The analyses of the consequences of accidents involving nearby industrial, military, and transportation facilities which have been identified as design basis events are reviewed.

II. ACCEPTANCE CRITERIA

The identification of design basis events resulting from the presence of hazardous materials or activities in the vicinity of the plant is acceptable if the design basis events include each postulated type of accident for which the expected rate of occurrence of potential exposures in excess of the 10 CFR Part 100 guidelines is estimated to exceed the NRC staff objective of approximately 10^{-7} per year. Because of the difficulty of assigning accurate numerical values to the expected rate of unprecedented potential hazards generally considered in this review plan, judgment must be used as to the acceptability of the overall risk presented.

The probability of occurrence of the initiating events leading to potential consequences in excess of 10 CFR Part 100 exposure guidelines should be estimated using assumptions that are as representative of the specific site as is practicable. In addition, because of the low probabilities of the events under consideration, data are often not available to permit accurate calculation of probabilities. Accordingly, the expected rate of occurrence of potential exposures in excess of the 10 CFR Part 100 guidelines of approximately 10^{-6} per year is acceptable if, when combined with reasonable qualitative arguments, the realistic probability can be shown to be lower.

The effects of design basis events have been adequately considered if analyses of the effects of those accidents on the safety-related features of the plant have been performed and measures (e.g., hardening, fire protection) to mitigate the consequences of such events have been taken.

III. REVIEW PROCEDURES

In some cases it may be necessary to consult with or obtain specific data from other branches, such as the Structural Engineering Branch (SEB) or Auxiliary Systems Branch (ASB), regarding possible effects of external events on plant structures or components.

The applicant's probability calculations are reviewed, and an independent probability analysis is performed by the staff if the potential hazard is considered significant enough to affect the licensability of the site or is important to the identification of design basis events.

All stochastic variables that affect the occurrence or severity of the postulated event are identified, and judged to be either independent or conditioned by other variables.

Probabilistic models should be tested, where possible, against all available information. If the model or any portion of it, by simple extension, can be used to predict an observable accident rate, this test should be performed.

The design parameters (e.g., overpressure) and physical phenomena (e.g., gas concentration) selected by the applicant for each design basis event are reviewed to ascertain that the values are comparable to the values used in previous analyses and found to be acceptable by the staff.

Each design basis event is reviewed to determine that the effects of the event on the safety features of the plant have been adequately accommodated in the design.

If accidents involving release of smoke, flammable or nonflammable gases, or chemical bearing clouds are considered to be design basis events, an evaluation of the effects of these accidents on control room habitability should be made in SAR Section 6.4 and on the operation of diesels and other safety-related equipment in SAR Chapter 9.

Special attention should be given to the review of standardized designs which propose criteria involving individual numerical probability criteria for individual classes of external man-made hazards. In such instances the reviewer should establish that the envelope also includes an overall criterion that limits the aggregate probability of exceeding design criteria associated with all of the identified external man-made hazards. Similarly, special attention should be given to the review of a site where several man-made hazards are identified, but none of which, individually, has a probability exceeding the acceptance criteria stated herein. The objective of this special review should be to assure that the aggregate probability of an outcome that may lead to unacceptable plant damage meets the acceptance criteria of Part II of this SRP Section. (A hypothetical example is a situation where the probability of shock wave overpressure greater than design

overpressure is about 10^{-7} per reactor year from accidents at a nearby industrial facility, and approximately equal probabilities of exceeding design pressure from railway accidents, highway accidents and from shipping accidents. Individually each may be judged acceptably low; the aggregate probability may be judged sufficiently great that additional features of design are warranted.)

IV. EVALUATION FINDINGS

If the reviewer verifies that sufficient information has been provided and that his evaluation is sufficiently complete and adequate to meet the acceptance criteria in Section II of this SRP, conclusions of the following type may be prepared for the staff's safety evaluation report:

> "The applicant has identified potential accidents which could occur in the vicinity of the plant, and from these has selected those which should be considered as design basis events and has provided analyses of the effects of these accidents on the safety-related features of the plant. The applicant has demonstrated that the plant is adequately protected and can be operated with an acceptable degree of safety with regard to potential accidents which may occur as the result of activities at nearby industrial, military, and transportation facilities."

V. REFERENCES

Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," Revision 2.

Affidavit of Jacques B. J. Read before the Atomic Safety and Licensing Board in the matter of Skagit Nuclear Power Project, Units 1 and 2, July 15, 1976. Docket Nos. STN 50-522, 523.

Atomic Safety and Licensing Board, Supplemental Initial Decision in the Matter of Hope Creek Generating Station, Units 1 and 2, March 28, 1977. Docket Nos. 50-354, 355.

Section 2, Supplement 2 to the Floating Nuclear Plant Safety Evaluation Report, Docket No. STN 50-437, September 1976.

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SECTION 2.3.1 REGIONAL CLIMATOLOGY

REVIEW RESPONSIBILITIES

Primary - Hydrology-Meteorology Branch (HMB)

Secondary - None

I. AREAS OF REVIEW

Information is presented by the applicant and reviewed by the staff concerning averages and extremes of climatic conditions and regional meteorological phenomena which affect the safe design and siting of the plant. The review covers the following specific areas:

1. A description of the general climate of the region with respect to types of air masses, synoptic features (high and low pressure systems and frontal systems), general air-flow patterns (wind direction and speed), temperature and humidity, precipitation (rain, snow, and sleet), and relationships between synoptic-scale atmospheric processes and local (site) meteorological conditions.

2. Seasonal and annual frequencies of severe weather phenomena including tornadoes, waterspouts, thunderstorms, lightning, hail (including probable maximum size), and high air pollution potential.

3. Meteorological conditions used as design and operating bases including:

   a. The maximum snow and ice load (water equivalent) that the roofs of safety-related structures must be capable of withstanding during plant operation.

   b. Ultimate heat sink meteorological conditions resulting in maximum evaporation and drift loss of water and minimum water cooling.

   c. Tornado parameters including translational speed, rotational speed, and the maximum pressure differential with the associated time interval.

   d. 100-year return period "fastest mile of wind" including vertical velocity distribution and gust factor.

   e. Probable maximum annual frequency of occurrence and time duration of freezing rain (ice storms) and, where applicable, dust (sand) storms.

f. Other meteorological and air quality conditions used for design and operating basis considerations.

II. ACCEPTANCE CRITERIA

The information in this section will be acceptable if the regional meteorological conditions and phenomena which affect the safe design and siting of the plant are presented and substantiated in accordance with acceptable practice and data as promulgated by the National Oceanic and Atmospheric Administration (NOAA), industry standards, and Commission guides, criteria, and regulations. More specifically the following criteria are used to determine acceptability.

The description of the general climate of the region should be based on standard climatic summaries compiled by NOAA. Consideration of the relationships between regional synoptic-scale atmospheric processes and local (site) meteorological conditions should be based on appropriate meteorological data.

Data on severe weather phenomena should be based on standard meteorological records from nearby representative National Weather Service (NWS), military or other stations recognized as standard installations which have long periods of record. The applicability of these data to represent site conditions during the expected period of reactor operation must be substantiated.

Design basis tornado parameters should be based on Regulatory Guide 1.76 (Ref. 2) or an adequately substantiated study must have been performed to demonstrate that lower values apply to the specific site. Operating basis wind velocity (fastest mile of wind) should be based on a standard such as that published by the American National Standards Institute (ANSI) with suitable corrections for local conditions. The ultimate heat sink meteorological data, as stated in Regulatory Guide 1.27 (Ref. 1) should be based on long-period regional records which represent site conditions. Freezing rain estimates are to be based on representative NWS station data. All other meteorological and air quality data used for safety-related plant design and operating bases should be documented and substantiated.

High air pollution potential information should be based on U.S. Environmental Protection Agency (EPA) studies.

III. REVIEW PROCEDURES

1. General Climate

The general climatic description of the region in which the site is located is reviewed for completeness and authenticity. Climatic parameters such as air masses, general airflow, pressure patterns, frontal systems, and temperature and humidity conditions reported by the applicant are checked against standard references (Ref. 3 and 4) for appropriateness with respect to location and period of record.

The applicant's description of the role of synoptic-scale atmospheric processes on local (site) meteorological conditions is checked against the descriptions provided in References 4 and 5 and the reviewer's knowledge of the area.

2. Regional Meteorological Averages and Extremes

Since meteorological averages and extremes can only be obtained from stations in the region of the site which have long periods of record, and the stations are not usually very close to the site, a determination of the representativeness of the data to site conditions is the primary concern in the review. A determination of the adequacy of the stations and their data is also made.

Recorded meteorological averages and extremes are checked against standard publications such as Reference 5. Snow and ice load adequacy is checked for reasonableness against ANSI A58.1-1972 (Ref. 7) and regional data available in References 4, 5, and 6. References 4 and 5 provide information on other averages and extremes. References 8 and 9 provide information on high air pollution potential for verification. Extreme winds and the specific vertical velocity distribution are checked against References 7 and 11. Gust factors are checked against Reference 7. The design basis tornado parameters are checked for agreement with Regulatory Guide 1.76 (Ref. 2) and tornado data are verified using the procedures and data in WASH-1300 (Ref. 10).

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information has been provided and that his evaluation supports concluding statements of the following type to be included in the staff's safety evaluation report:

> "The staff has reviewed available information relative to the regional meteorological conditions of importance to the safe design and siting of this plant and has concluded that the following meteorological design parameters are appropriate."

This statement will be followed by a resume of the general climate and the meteorological design paramenters used for the plant.

V. REFERENCES

1. Regulatory Guide 1.27, "Ultimate Heat Sink for Nuclear Power Plants."

2. Regulatory Guide 1.76, "Design Basis Tornado for Nuclear Power Plants."

3. U.S. Department of Commerce, "Climatic Atlas of the United States," Environmental Data Service, NOAA, June 1968.

4. U.S Department of Commerce, "Local Climatological Data-Annual Summary with Comparative Data," Environmental Data Service, NOAA, published annually for all first-order NWS stations.

5. U.S. Department of Commerce, "State Climatological Summary," Environmental Data Service, NOAA, published annually by state.

6. U.S. Department of Commerce, "Storm Data," Environmental Data Service, NOAA, published monthly.

7. ANSI A58.1 "Building Code Requirements for Minimum Design Loads in Buildings and other Structures," American National Standards Institute (1972).

8. G. C. Holzworth, "Mixing Heights, Wind Speeds, and Potential for Urban Air Pollution Throughout the Contiguous United States," AP-101, Office of Air Programs, USEPA, January 1972.

9. J. Korshover, "Climatology of Stagnating Anticyclones East of the Rocky Mountains, 1936-1970," Publication No. 99-AP-34, Public Health Service, October 1971.

10. E. H. Markee, Jr., "Technical Basis for Interim Regional Tornado Criteria," WASH-1300, USAEC, May 1974.

11. H. C. S. Thom, "New Distribution of Extreme Winds in the United States," Journal of the Structural Division, Proceedings of the American Society of Civil Engineers, pp. 1787-1801, July 1968.

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SECTION 2.3.2 LOCAL METEOROLOGY

REVIEW RESPONSIBILITIES

Primary - Hydrology-Meteorology Branch (HMB)

Secondary - None

I. AREAS OF REVIEW

Information is presented by the applicant and reviewed by the staff concerning the local (site) meteorological parameters, an assessment of the potential influence of the plant and its facilities on local meteorological conditions, and a topographical description of the site and its environs. The review covers the following specific areas.

1. A description of the local (site) meteorology in terms of airflow, temperature, atmospheric water vapor, precipitation, fog, and atmospheric stability.

2. An assessment of the influence of the plant and its facilities on the local meteorological parameters listed in (1), including the effects of plant structures, terrain modification, and heat and moisture sources due to plant operation.

3. A topographical description of the site and its environs, as modified by the plant structures, including the site boundary, exclusion zone, and low population zone.

II. ACCEPTANCE CRITERIA

The information in this section will be acceptable if the local meteorological and topographic descriptions of the site area applicable both before plant construction and during plant operation are adequately documented such that meteorological impacts on plant design and operation as well as the impact of the plant on local meteorological conditions can be reliably predicted. Specifically, the following information is needed. This information should be fully documented and substantiated as to its representativeness of conditions at and near the site.

1. Local summaries of meteorological data based on onsite data and National Weather Service station summaries or other standard installation summaries from appropriate nearby locations should be presented as specified in Regulatory Guide 1.70, Section 2.3.2 (Ref. 1).

2. A discussion and evaluation of the influence of the plant and its facilities on the local meteorological conditions should be provided. A discussion of potential changes in normal and extreme values presented in SAR Section 2.3.2.1 resulting from plant construction and operation should be made.

3. A complete topographical description of the site and environs out to a distance of 50 miles from the plant, as described in Regulatory Guide 1.70, Section 2.3.2.2 should be provided (Ref. 1).

III. REVIEW PROCEDURES

1. The summaries listed in Section 2.3.2.1 of Regulatory Guide 1.70 are reviewed for completeness and adequacy of basic data. The wind and atmospheric stability data should be based on onsite data if possible since airflow and vertical temperature structure can vary substantially from one location to another and are inputs to the assessment of atmospheric diffusion conditions at the site. The other summaries should be based on nearby representative stations with long periods of record since the locally measured extremes in intensity and frequency are compared to design basis values presented in Section 2.3.1 of the safety analysis report or are used by other branches to determine whether these meteorological conditions are limiting conditions for design and emergency procedures. When offsite data are used, a determination is made of how well the data represent site conditions and whether more representative data are available. National Oceanic and Atmospheric Administration (NOAA) state meteorological summaries (Ref. 2), local climatological data (Ref. 3), and various NOAA Environmental Data Service summaries are used by the reviewer to evaluate the representativeness of stations and periods of record. The reviewer visits all primary meteorological data collection locations.

2. The review procedure for evaluating the contents of Section 2.3.2.2 of the SAR is as follows:

   a. Determine the terrain modifications that will occur as a result of plant construction such as removal of trees, leveling of ground, and installation of lakes and ponds.

   b. Determine the location, size, and materials used for plant structures including buildings, switchyard gear, parking lots, and roads.

   c. Determine and quantify the heat and moisture sources that will result from plant operations.

   d. Relate the input information in items a, b, and c, above, to local meteorological modifications.

   e. Compare the reviewer's assessment with that of the applicant.

3. The reviewer assures that all topographic maps and topographic cross-sections presented by the applicant are legible and well-labeled so that the information needed during the review can be readily extracted. Reference points and the direction of true north should be checked carefully. Points of interest such as plant structures, site boundary, and exclusion zone should be marked on the maps and diagrams.

The reviewer compares the applicant's assessment of the effect of topography to standard assessments such as those presented in "Meteorology and Atomic Energy -1968" (Ref. 4) and decides whether the standard regulatory atmospheric diffusion models (discussed in SRP sections 2.3.4 and 2.3.5) are appropriate for this site.

Section 2.3.2.3 is reviewed for content based on the specifications outlined in Regulatory Guide 1.70.

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information has been provided and that his evaluation supports concluding statements of the following type, to be included in the staff's safety evaluation report:

> "The staff has reviewed available information relative to local meteorological and air quality conditions that are of importance to the safe design and siting of this plant and has concluded that the preceding meteorological parameters are appropriate."

This statement will be preceded by a resume of local meteorological and air quality parameters appropriate to the site.

V. REFERENCES

1. Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants,"

2. U.S. Department of Commerce, "State Climatological Summary," Environmental Data Service, NOAA, published annually by state.

3. U.S. Department of Commerce, "Local Climatological Data - Annual Summary with Comparative Data," Environmental Data Service, NOAA, published annually for all first-order NWS stations.

4. D. H. Slade (ed.), "Meteorology and Atomic Energy - 1968," TID-24190, Division of Technical Information, USAEC (1968).

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SECTION 2.3.3 ONSITE METEOROLOGICAL MEASUREMENTS PROGRAMS

REVIEW RESPONSIBILITIES

Primary - Hydrology-Meteorology Branch (HMB)

Secondary - None

I. AREAS OF REVIEW

Information is presented by the applicant and reviewed by the staff concerning the onsite meteorological measurement programs including instrumentation, data summaries, and, at the operating license (OL) stage, provisions of the technical specifications. The review covers the following specific areas:

1. The meteorological instrumentation review includes siting of sensors, sensor performance specifications, methods and equipment for recording sensor output, the quality assurance program for sensors and recorders, and data acquisition and reduction procedures.

2. The review of meteorological data summaries includes consideration of the period of record and amenability of the data for use in making atmospheric diffusion estimates.

3. The review of meteorological technical specifications includes consideration of instrument siting, instrument specifications, control room monitoring, and data reporting and storage.

II. ACCEPTANCE CRITERIA

1. Generally, the onsite meteorological programs must produce data which can be summarized to provide an adequate meteorological description of the site and its vicinity for the purpose of making atmospheric diffusion estimates for accidental and routine airborne releases of effluents. Guidance on an adequate program is given in Regulatory Guide 1.23. More specifically:

   a. The siting of meteorological sensors should satisfy the position stated in Regulatory Guide 1.23.

b. The meteorological sensors should meet the sensitivity specifications of Regulatory Guide 1.23 and be capable of withstanding the expected range of environmental conditions at the site such that adequate data recovery is anticipated. Any deviation from Regulatory Guide 1.23 must be justified.

c. The meteorological recording systems must be capable of providing accurate, reliable data.

d. The instrument surveillance and calibration procedures must provide reasonable assurance that adequate, accurate data will be obtained.

e. The data reduction procedures should provide average data which are within the accuracy guidelines of Regulatory Guide 1.23. Any deviation must be justified.

2. The following criteria are used to judge the acceptability of meteorological data summaries for atmospheric diffusion estimates.

a. For the Preliminary Safety Analysis Report (PSAR), at least one annual cycle of onsite meteorological data should be provided at docketing. If adequate meteorological data (one year) are not available at docketing, the best available (onsite and offsite) data to describe atmospheric dispersion characteristics should be provided. Adequate onsite meteorological data must be provided prior to or with the scheduled response to the first set of requests for additional information.

b. For the Final Safety Analysis Report (FSAR), at least two consecutive annual cycles (and preferably three or more whole years), including the most recent one-year period, should be provided at docketing.

Meteorological data must be presented in the form of joint frequency distributions of wind speed and wind direction by atmospheric stability class in the format described in Regulatory Guide 1.23. An hour-by-hour listing of hourly-averaged parameters may also be provided on magnetic tape in the format described in Appendix 2.3A. Additional sources of meteorological data for consideration in the description of airflow trajectories from the site to a distance of 80 km may include National Weather Service (NWS) stations, other meteorological programs that are well-maintained and well-exposed (e.g., other nuclear facilities, university and private meteorological programs), and additional satellite facilities established by the applicant (or others) to characterize relevant meteorological conditions at critical locations.

Evidence of how well these data represent long-term conditions at the site must be presented.

III. REVIEW PROCEDURES

1. Meteorological Instrumentation

The basic meteorological parameters measured by instrumentation at all sites should include wind direction and wind speed at two levels, ambient air temperature difference between two levels, temperature, and atmospheric moisture (at sites where water vapor is emitted, as from cooling towers or spray ponds).

a. Instrument Siting

Instrument types, heights, and locations are compared generally to the position stated in Regulatory Guide 1.23, Positions C.1 and C.2. Detailed review procedures follow.

(1) Local Exposure of Instruments

The local exposure of the wind and temperature sensors is reviewed to assure that the measurements will represent the general site area. A determination is made whether the tower which supports the sensors will influence the wind or temperature measurements. Professional experience and studies have shown that wind sensors should be mounted on booms such that the sensors are at least one tower width away from an open-latticed tower and at least two stack or tower widths away from a stack or closed tower. For temperature sensors, mounting booms need not be as long as those for wind sensors but must be unaffected by thermal radiation from the tower itself. No temperature sensors may be mounted directly on stacks or closed towers. Mounting booms for all sensors should be oriented normal to the prevailing wind at the site.

A determination is made whether the terrain at or near the base of the tower will unnaturally affect the wind or temperature measurements. Heat reflection characteristics of the surface underlying the meteorological tower (grass, soil, gravel, paving, etc.) are estimated to assure that localized influences on measurements are minimal. The position, size, and materials used in the construction of the recorder shack and nearby trees are also examined for potential localized influence on the measurements.

(2) General Exposure of Instruments

Since the objective of the instrumentation is to provide measurements which represent the overall site meteorology without plant structure interference, the tower position(s) must have been selected with this general objective in mind. Examination of topographical maps, which have been modified to show finished plant grade, and a site visit along with professional judgment on airflow patterns are used to determine and evaluate the representativeness of the location(s).

The plant structure layout including structure heights are examined for potential influence on meteorological measurements. In general, sensors should be located at least five building heights away from the buildings to minimize this influence.

b. Meteorological Sensors

The type and performance specifications of the sensors are evaluated. Manufacturers' specifications and analysis, and operating experience for these sensors are considered in evaluation of adequacy with respect to accuracy and the potential for acceptable data recovery. Standardized evaluations such as Reference 5 and operational experience reports contained in research papers are utilized.

The suitability of the specific type of sensor for use in the environmental conditions at the site is evaluated. To this end, the range of wind conditions and the ability of the sensors to withstand corrosion, blowing sand, salt, air pollutants, birds, and insects are considered.

If the sensors are new and unique, a meteorological instrumentation expert (e.g., NOAA, Idaho Falls) may be consulted.

c. Recording of Meteorological Sensor Output

The methods of recording (e.g., digital or analog, instantaneous or average, engineering units or raw voltages) and the recording equipment including performance specifications and location of this equipment are evaluated. Manufacturers' specifications and operating experience for the recorders are considered in evaluation of adequacy with respect to accuracy and the potential for acceptable data recovery.

The controlled environmental conditions in which the recorders are kept (instrument shack or control room) are reviewed for adequacy in accordance with the manufacturers' specifications. The ability to obtain a direct readout from the recorders in situ during routine inspection of systems is checked so that the inspector will be able to relate the recorder output directly to what the sensor should be seeing. Some specific criteria are contained in Regulatory Guide 1.23, Position C.3.

The reviewer determines that there are provisions for proper monitoring of wind direction, wind speed, and vertical temperature difference in the control room during plant operation.

d. Instrumentation Surveillance

The inspection, maintenance, and calibration procedures and their frequency are evaluated. These surveillance procedures and the frequency of attention

that the instrumentation systems receive are compared to operating experience at this site and other sites with similar instrumentation with the objective of determining that acceptable data recovery with acceptable accuracy will be obtained throughout the duration of the meteorological program. Guidelines for acceptable accuracy and acceptable data recovery are specified in Regulatory Guide 1.23, Positions C.4 and C.5. Any deviations from Regulatory Guide 1.23 must be justified.

e. Data Acquisition and Reduction

The procedures, including both hardware and software, for data acquisition and reduction are evaluated. Since there are many methods of acquiring data from meteorological measurement systems which are acceptable to the staff, the review procedure varies. The basic components of the program which are reviewed to ascertain the acceptability of data acquisition and reduction are:

(1) Accuracy of measuring in units of direct measurement and their precision.
(2) Accuracy in conversion of direct measurement units to meteorological units.
(3) Accuracies involved in frequency and mode (instantaneous or average) of sampling.
(4) Time over which system outputs are averaged for final disposition and accuracy of these data.

Since the instrument accuracy suggestions in Regulatory Guide 1.23 refer to overall system accuracy for instantaneous recorded values or time averaged values, the overall system accuracy is evaluated in addition to the component (sensor, recorder and reduction) accuracies. The evaluation consists primarily of using statistical procedures for compound errors based on sensor accuracy, recorder accuracy, conversion of units accuracy, and frequency and mode of sampling (Ref. 6).

2. Meteorological Data Summaries

Annual (representing the annual cycle) joint frequency distributions of wind direction and wind speed by atmospheric stability class are evaluated from the viewpoint of sufficiency of detail to permit the staff to make an independent determination of the atmospheric diffusion conditions, relative concentrations and relative depositions for accidental and routine atmospheric releases of radioactive effluents from the reactor and its facilities. The distributions are to be based wholly on onsite data, a combination of onsite and offsite data, or offsite data in accordance with the criteria of subsections II.2.a. and II.2.b.
The joint frequency distributions are compared to the example distribution given in Regulatory Guide 1.23. If hourly meteorological data are provided on magnetic tape, the format is compared to the example given in Appendix 2.3A.

"Calm" wind conditions (which should be defined as wind speeds less than the starting speed of the anemometer or vane, whichever is higher) are checked for appropriateness and appearance in the distributions as a separate speed class, without directional assignment, by atmospheric stability class.

Annual joint frequency distributions for each expected mode of release (i.e., ground level and elevated) are checked for appropriateness of heights of measurements of wind direction, wind speed, and atmospheric stability. Winds at the 10-meter level and temperature difference (ΔT) between the vent height and the 10-meter level are used for vent and penetration releases. Winds from near release height and ΔT between release height and the 10-meter level are used for stack releases. A stack is defined as a release point which is greater than twice the height of adjacent structures.

The climatic representativeness of the joint frequency distribution is checked by comparison with nearby stations which have collected reliable meteorological data over a long period of time (10-20 years). The distributions are compared with sites in similar geographical and topographical locations to assure that the data are reasonable.

3. Meteorological Technical Specifications

The applicant's technical specifications are reviewed at the OL stage to determine if the operational meteorological monitoring program meets the recommendations of Regulatory Guide 1.23 with respect to tower siting, instrumentation specifications, and control room monitoring, and if the reporting requirements meet the recommendations of Regulatory Guide 1.21. Deviations from the Regulatory Guides may be accepted if justified.

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information has been provided in accordance with the requirements of this SRP section and that his evaluation supports the following type of concluding statement, to be used in the staff's Safety Evaluation Report:

"The onsite meteorological measurements program is in conformance with the position stated in Regulatory Guide 1.23, except for (identify deviations). These deviations were found to have minimal impact on our evaluation . We conclude that the meteorological measurements program has produced data which in turn have been summarized to provide an adequate meteorological description of the site and its vicinity for the purpose of making atmospheric diffusion estimates for accidental and routine airborne releases of effluents from the nuclear facility."

For the CP review, if adequate meteorological data have not been acquired by the applicant and presented to the staff, a statement requiring the applicant to obtain adequate data in a timely manner will be added.

The input to the Safety Evaluation Report will also include a brief summary description of the onsite meteorological measurements program covering the following items:

1. Height and location of meteorological sensors by type.
2. Period of data record.
3. Data recovery.
4. Period of data record and meteorological parameters used for atmospheric diffusion estimates.

V. REFERENCES

1. Regulatory Guide 1.21, "Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants."

2. Regulatory Guide 1.23, "Onsite Meteorological Programs."

3. Regulatory Guide 4.2.1, "Additional Guidance - Environmental Data."

4. R. C. Hilfiker, "Exposure of Instruments," Chapter in Air Pollution Meteorology Manual, Training Course 411 conducted by USEPA Air Pollution Training Institute, Research Triangle Park, North Carolina, August 1973.

5. D. H. Slade (ed.), "Meteorology and Atomic Energy - 1968," TID-24190, Division of Technical Information, USAEC (1968).

6. C. E. P. Brooks and N Caruthers, "Handbook of Statistical Methods in Meteorology," M.O. 538, Her Majesty's Stationary Office, London (1953).

7. D. A. Mazzarella, "An Inventory of Specifications for Wind Measuring Instruments," Bull. Amer. Meteor. Soc. 53, 860 (1972).

APPENDIX A

Standard Review Plan Section 2.3.3

RECOMMENDED* FORMAT FOR HOURLY METEOROLOGICAL DATA TO BE PLACED ON MAGNETIC TAPE

Use: 9-track tape (7 will be acceptable)
Standard Label which would include:
- Record Length = 160
- Block Size (3200 - fixed block size)
- Density (1600 BPI - 800 will be accepted)

Do Not Use: Magnetic tapes with unformatted or spanned records

At the beginning of each tape, use the first five (5) records (which is the equivalent of ten cards) to give a tape description. Include plant name; location (latitude, longitude); dates of data; information explaining data contained in the "other" fields if they are used; height of measurements; and any additional information pertinent to identification of the tape. Make sure all five records are included, even if some are blank. Format for the first five records will be 160A1. Meteorological data format is (I6, I2, I3, I4, 25F5.1, F5.2, 3F5.1). Decimal points should not be included when copying data onto the tape.

All data should be given to the tenth of a unit except solar radiation, which should be given to a hundredth of a unit. This does not necessarily indicate the accuracy of the data (e.g., wind direction is usually given to the nearest degree, but record it with a zero in the tenth's place; i.e., 275 degrees would be 275.0 degrees and placed on the tape as 2750). All nines in any field indicate a lost record (99999). All sevens in a wind direction field indicate calm (77777). If there are only two levels of data, use the upper and lower levels. If there is only one level of data, use the upper level.

*Data on magnetic tape are acceptable in any reasonable format, if the format is completely described (see NUREG-0158, Part 1), and if a sample tape dump is provided.

MAGNETIC TAPE METEOROLOGICAL DATA

LOCATION:

DATE OF DATA RECORD:

I6	Identifier (can be anything)	
I2	Year	
I3	Julian Day	
I4	Hour (on 24-hour clock)	
		ACCURACY
F5.1	Upper Measurements: Level = _____ meters	
F5.1	Wind Direction (degrees)	_____
F5.1	Wind Speed (meter/sec)	_____
F5.1	Sigma Theta (degrees)	_____
F5.1	Ambient Temperature (°C)	_____
F5.1	Moisture: _____	_____
F5.1	Other: _____	_____
F5.1	Intermediate Measurements: Level = _____ meters	
F5.1	Wind Direction (degrees)	_____
F5.1	Wind Speed (meters/sec)	_____
F5.1	Sigma Theta (degrees)	_____
F5.1	Ambient Temperature (°C)	_____
F5.1	Moisture: _____	_____
F5.1	Other: _____	_____
F5.1	Lower Measurements: Level = _____ meters	
F5.1	Wind Direction (degrees)	_____
F5.1	Wind Speed (meters/sec)	_____
F5.1	Sigma Theta (degrees)	_____
F5.1	Ambient Temperature (°C)	_____
F5.1	Moisture: _____	_____
F5.1	Other: _____	_____
F5.1	Temp. Diff. (Upper-Lower) (°C/100 meters)	_____
F5.1	Temp. Diff. (Upper-Intermediate) (°C/100 meters)	_____
F5.1	Temp. Diff. (Intermediate-Lower) (°C/100 meters)	_____
F5.1	Precipitation (mm)	_____
F5.2	Solar Radiation (cal/cm^2/min)	_____
F5.1	Visibility (km)	_____
F5.1	Other: _____	_____
F5.1	Other: _____	_____

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SECTION 2.3.4 SHORT TERM DIFFUSION ESTIMATES

REVIEW RESPONSIBILITIES

Primary - Site Analysis Branch (SAB)

Secondary - Accident Analysis Branch (AAB)

I. AREAS OF REVIEW

Information is presented by the applicant and reviewed by the staff concerning atmospheric diffusion estimates for accidental releases of effluents to the atmosphere. The review covers the following specific areas:

1. Atmospheric diffusion models to calculate relative concentrations for accidental radioactive and toxic gas release modes as determined by Accident Analysis Branch.

2. Meteorological data summaries used as input to diffusion models.

3. Derivation of diffusion parameters from meteorological data.

4. Probability distributions of relative concentrations.

5. Relative concentrations used for assessment of consequences of radioactive releases for design basis (10 CFR Part 100) accidents, onsite and offsite toxic gas releases, and accidents that result in limited releases of radioactivity.

II. ACCEPTANCE CRITERIA

This section will be acceptable if the applicant has provided conservative estimates of atmospheric diffusion at appropriate distances from the source for postulated accidental releases of radioactive and toxic materials to the atmosphere considering the plant as both a source and a receptor. Guidelines for acceptability of models and conservatism appropriate to design basis calculations are Regulatory Guides 1.3, 1.4, 1.5, 1.23, 1.24, 1.25, and 1.77; National Oceanic and Atmospheric Administration (NOAA) Technical Memorandum ERL ARL-42; standard references such as "Meteorology and Atomic Energy - 1968" and Accident Analysis Branch and Site Analysis Branch positions. Since the staff makes

an independent evaluation of atmospheric diffusion estimates based on data from the onsite meteorological measurements program and other nearby meteorological data, it is not necessary for the applicant to duplicate the staff's final estimates. However, the applicant's diffusion estimates should reasonably reflect staff positions and state-of-the-art atmospheric diffusion knowledge. Specifically the following information is required:

1. The atmospheric diffusion models used by the applicant to calculate relative concentrations resulting from accidental airborne releases of radioactive and toxic gases must be documented in detail and substantiated so that the staff can evaluate their appropriateness to site and plant characteristics.

2. Meteorological data summaries to be used as input to the diffusion models must be presented in joint frequency distribution form. These summaries must have been generated from the best available annual periods of data on record and contain data acceptable to the staff which represent appropriate hourly values of wind direction, wind speed, and atmospheric stability for each mode of accidental release.

3. The atmospheric fiffusion parameters, such as lateral and vertical plume spread (σ_y and σ_z) as a function of distance and windspeed, must be related to measured meteorological parameters and substantiated as to their validity and degree of conservatism for use in estimating the consequences of accidents within the range of distances which are of interest for the plant.

4. Cumulative frequency distributions of relative concentrations (X/Q) based on mode of release over appropriate time periods and on the aforementioned atmospheric diffusion models, meteorological data summaries, and atmospheric diffusion parameters must be presented for appropriate distances such as the site boundary distance and the outer boundary of the low population zone as specified in Section 2.3.4.2 of the "Standard Format and Contents of Safety Analysis Reports for Nuclear Power Plants," Revision 2. The methods of generating these distributions must be adequately described.

5. Relative concentrations used for assessment of consequences of radioactive releases for design basis (10 CFR Part 100) accidents, for onsite and offsite releases of toxic gases, and for radioactive releases resulting from other accidents must be presented.

III. REVIEW PROCEDURES

1. Atmospheric Diffusion Models

The applicant's diffusion models are compared to the general Gaussian models which are contained in Regulatory Guides 1.3 and 1.4 for elevated releases and ground level releases with a wake correction (see also Reference 3). The suitability of the models for mode of release, plant configuration, and site topography are reviewed. Accident Analysis Branch defines the modes of release and accidents to be considered.

A determination is made as to whether the release should be considered as an elevated point source or a ground level point source with a volumetric correction for turbulent mixing in the wake of buildings. Generally the release is considered to be elevated if the release point is at least twice as high as nearby solid structures. Otherwise, a ground level volumetric release formulation is usually used. The volumetric correction is usually based on 1/2 the minimum cross-sectional area of the structure from which the effluent is released.

Most accidental releases are considered as continuous releases (i.e., >5 minutes duration). However, in some instances, usually with explosions resulting in the release of toxic chemicals, the releases may be considered as instantaneous (puffs). For puff releases, instantaneous point source Gaussian diffusion equations are used with a correction for initial source volume (Ref. 10).

If a site is located such that the horizontal (or vertical) plume spread via diffusion is restricted by topography (or unusual meteorological conditions), the models are examined for appropriate modification. Some of these conditions are narrow, deep valleys, "fumigation" from elevated sources, and low level subsidence inversions of temperature in the vertical direction.

2. Meteorological Data Summaries

The data summaries in joint frequency distribution form are reviewed for compatability of data with the models utilized in the section above. General criteria are stated in Regulatory Guide 1.23 and in III.2 of Standard Review Plan 2.3.3.

3. Atmospheric Diffusion Parameters

Horizontal and vertical plume spread parameters, σ_y and σ_z, as functions of distance and atmospheric stability are reviewed. The current procedure is to relate $\sigma_y(X)$ and $\sigma_z(X)$ to vertical temperature difference classes as stated in Table 1 of Regulatory Guide 1.23. Departures from this procedure are reviewed for adequate justification. Such departures may be appropriate in the case of unusual sites (e.g., valley or coastal). The curves of σ_y and σ_z with distance, which appear in "Meteorology and Atomic Energy - 1968" are usually acceptable with the addition of a G stability class.

In instances when a puff diffusion equation is used, $\sigma_x = \sigma_y$ is usually a good assumption.

4. Cumulative Frequency Distributions of X/Q

A check is made of the cumulative frequency distributions for inclusion of pertinent modes and time periods of release, and adequacy of input data in accordance with the guidelines set forth in Section 2.3.4.2 of the Standard Format. The methods used to generate these distributions are reviewed for adequacy and conservatism.

5. Relative Concentrations Used for Accidents

The X/Q values used for assessment of consequences of radioactive releases for design basis accidents, for onsite and offsite releases of toxic gases, and for radioactive releases resulting from other accidents are reviewed for appropriateness and completeness of information.

An independent calculation of the probability distributions of X/Q is made for pertinent distances (usually the exclusion area boundary and the low population zone outer boundary, LPZ) using the computer program CHI/Q (Ref. 11) and the joint frequency distribution data for input. The most restrictive annual average X/Q values are also computed at the site boundary and the LPZ using the same program and input data. For assessment of the consequences of design basis accident releases, the value of X/Q at the "5% level" (value which is exceeded 5% of the time) is evaluated at the exclusion boundary and the LPZ. These values are assumed to represent conditions for a two-hour period. X/Q values for time periods greater than two hours are estimated for the LPZ distance by assuming a logarithmic relationship between the "two-hour" value and the annual average value.

Conservative (5%) values of X/Q from appropriate models for appropriate time intervals and distances are transmitted to AAB for dose assessment of design basis accidents.

For assessment of other accidents, the median (50%) values of X/Q for appropriate time intervals and distances (usually the site boundary) are evaluated and transmitted to AAB.

X/Q values based on site-specific meteorological data are calculated, as needed, for control room dose calculations and onsite and offsite releases of toxic gases. These estimates are made on a case-by-case basis since the mode of release and, therefore, the diffusion models vary.

IV. EVALUATION FINDINGS

The reviewer verifies that adequately conservative atmospheric diffusion models, with adequate onsite meteorological data as input to the models, have been used to calculate relative concentrations at appropriate distances and directions from postulated release points during accidental airborne releases of potentially hazardous materials. If adequate onsite meteorological data are not available for the construction permit review, the reviewer must assure that adequate conservatism has been applied to the calculated relative concentrations for accidental airborne effluent releases based on available data.

The reviewer's evaluation must support the following type of concluding statement, to be used in the staff's safety evaluation report:

"Conservative assessments of post-accident atmospheric diffusion conditions have been made by the staff from the applicant's meteorological data and appropriate diffusion models."

The input to the safety evaluation report will also include a brief summary of the relative concentrations (X/Q) calculated by the staff, reference to diffusion models used, and a comparison between the values computed by the staff and the applicant.

V. REFERENCES

1. Regulatory Guide 1.3, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Boiling Water Reactors."

2. Regulatory Guide 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Pressurized Water Reactors."

3. Regulatory Guide 1.5, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Steam Line Break Accident for Boiling Water Reactors."

4. Regulatory Guide 1.23, "Onsite Meteorological Programs."

5. Regulatory Guide 1.24, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Pressurized Water Reactor Gas Storage Tank Failure."

6. Regulatory Guide 1.25, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors."

7. Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," Revision 2.

8. Regulatory Guide 1.77, "Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors."

9. D. H. Slade (ed.), "Meteorology and Atomic Energy - 1968," TID-24190, Division of Technical Information, USAEC (1968).

10. G. R. Yanskey, E. H. Markee, and A. R. Richter, "Climatography of the National Reactor Testing Station," IDO-12048, Idaho Operations Office, USAEC (1966).

11. J. F. Sagendorf, "A Program for Evaluating Atmospheric Dispersion From A Nuclear Power Station," Technical Memorandum ERL ARL-42, National Oceanic and Atmospheric Administration (1974).

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SECTION 2.3.5 LONG-TERM DIFFUSION ESTIMATES

REVIEW RESPONSIBILITIES

Primary - Hydrology-Meteorology Branch (HMB)

Secondary - Effluent Treatment Systems Branch (ETSB)
Radiological Assessment Branch (RAB)

I. AREAS OF REVIEW

Information is presented by the applicant and reviewed by the staff concerning atmospheric diffusion estimates for routine releases of effluents to the atmosphere. The review covers the following specific areas:

1. Atmospheric diffusion models to calculate relative concentrations at specified receptor locations (identified by RAB) for routine radioactive gas releases (with the release point characteristics determined by ETSB).

2. Meteorological data summaries used as input to diffusion models (Regulatory Guide 1.23).

3. Derivation of diffusion parameters from meteorological data.

4. Relative concentration (X/Q) and relative deposition (D/Q) values used for assessment of consequences of routine airborne radioactive releases.

II. ACCEPTANCE CRITERIA

This section will be acceptable if the applicant has provided realistic estimates of atmospheric diffusion at appropriate distances from the source for routine releases of radioactive materials to the atmosphere. Guidelines for acceptability of models are presented in Regulatory Guide 1.111 and NUREG-0324 (Refs. 2 and 3); National Oceanic and Atmospheric Administration (NOAA) Technical Memorandum ERL ARL-42 (Ref. 4); standard references such as "Meteorology and Atomic Energy - 1968" (Ref. 5); and Effluent Treatment Systems Branch and Radiological Assessment Branch guides (Refs. 6 and 7). The staff makes an independent evaluation of atmospheric diffusion estimates based on data from the onsite meteorological measurements program and other nearby meteorological data. It is not necessary for the applicant to duplicate the staff's estimates. However, the applicant's diffusion estimates should reasonably reflect

staff positions and general atmospheric diffusion knowledge. Specifically, the following information is required:

1. The atmospheric diffusion models used by the applicant to calculate concentrations resulting from routine airborne releases of radioactive gases must be documented in detail and substantiated so that the staff can evaluate their appropriateness to site and plant characteristics.

2. Meteorological data summaries to be used as input to the diffusion models may be presented in joint frequency distribution form or hour-by-hour listings. These summaries (or listings) must have been generated from the best available annual periods of data on record and contain data acceptable to the staff which represent appropriate hourly values of wind direction, wind speed, and atmospheric stability for each mode of routine release.

3. The atmospheric diffusion parameters, such as vertical plume spread (σ_z) as a function of distance and wind speed, must be related to measured meteorological parameters and be substantiated as to their validity for use in estimating the consequences of routine releases from the site boundary to a radius of 50 miles from the plant.

4. Relative concentration (X/Q) and relative deposition (D/Q) values used for assessment of consequences of routine radioactive gas releases must be presented as described in Section 2.3.5.2 of the "Standard Format and Contents of Safety Analysis Reports for Nuclear Power Plants " (Reference 8).

III. REVIEW PROCEDURES

1. Atmospheric Diffusion Models

The applicant's diffusion models are compared to the general Gaussian models which are contained in Regulatory Guide 1.111 (Ref. 2) for elevated releases and ground level releases with a wake correction (see also Ref. 4). The suitability of the models for mode of release, plant configuration, and site topography are reviewed. ETSB defines the modes of release to be considered.

A determination is made as to whether the release should be considered as an elevated point source, a partially-elevated release, or a ground level point source with a volumetric correction for turbulent mixing in the wake of buildings using the criteria presented in Regulatory Guide 1.111.

If a site is located such that the effluent trajectories (or vertical plume spread via diffusion) are restricted by topography (or unusual meteorological conditions), the models are examined for appropriate modification. Some of these conditions are narrow, deep valleys, "fumigation" from elevated sources, low level subsidence inversions of temperature in the vertical direction, and land-sea (lake) breeze regimes.

2. Meteorological Data Summaries

The data summaries in joint frequency distribution form or hourly listings are reviewed for compatibility of data with the models utilized in the section above. General criteria are stated in Regulatory Guide 1.23 and III.2 of SRP Section 2.3.3.

3. Atmospheric Diffusion Parameters

The vertical plume spread parameter, σ_z as a function of distance and atmospheric stability, is reviewed. The current procedure is to relate σ_z (X) to vertical temperature difference classes as stated in Table 1 of Regulatory Guide 1.23 (Ref. 2). Departures from this procedure are reviewed for adequate reasons for the departures, such as in the case of unusual sites (e.g., valley or coastal). The curves of σ_z with distance are presented in Regulatory Guide 1.111.

4. Relative Concentrations Used for Routine Releases

The X/Q and D/Q values used for assessment of the consequences of routine radioactive releases are reviewed for appropriateness to site conditions and completeness of information.

An independent calculation of annual average X/Q and D/Q values is made for 16 radial sectors from the site boundary to a distance of 50 miles from the plant, as well as for specific receptor locations, using appropriate meteorological data in joint frequency distribution form and the computer program XOQDOQ (Ref. 3). RAB provides the locations of specific receptors (e.g., site boundary, residence, garden, cow). Adjustments of the X/Q and D/Q output may be made through use of other offsite meteorological data when unusual topographic conditions surround the site or when the onsite meteorological data are found to be inadequate.

IV. EVALUATION FINDINGS

The reviewer verifies that adequate atmospheric diffusion models, with adequate onsite meteorological data as input to the models, have been used to calculate relative concentration and relative deposition at appropriate distances and directions from postulated release points during routine airborne releases of radioactive gases. If adequate onsite meteorological data are not available for the construction permit review, the reviewer must assure that adequate conservatism has been applied to the calculated relative concentrations for routine airborne effluent releases based on available data. The reviewer's evaluation must support the following type of concluding statement, to be included in the staff's Safety Evaluation Report:

"Based on the meteorological data provided by the applicant and an atmospheric dispersion model that is appropriate for the characteristics of the site and release points, the staff has concluded that representative atmospheric diffusion conditions have been calculated at the potential receptor points."

The input to the Safety Evaluation Report will also include a brief summary of the relative concentration (X/Q) and relative deposition (D/Q) calculated by the staff, reference to diffusion models used, and a comparison between the values computed by the staff and the applicant.

V. REFERENCES

1. Regulatory Guide 1.23, "Onsite Meteorological Programs."

2. Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents In Routine Releases From Light-Water-Cooled Reactors."

3. NUREG-0324, "XOQDOQ Program for the Meteorological Evaluation of Routine Effluent Releases at Nuclear Power Stations" (DRAFT), September 1977.

4. J. F. Sagendorf, "A Program for Evaluating Atmospheric Dispersion From a Nuclear Power Station," Technical Memorandum ERL ARL-42, National Oceanic and Atmospheric Administration (1974).

5. D. H. Slade (ed.), "Meteorology and Atomic Energy - 1968," TID-24190, Division of Technical Information, USAEC (1968).

6. Regulatory Guide 1.112, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents From Light-Water-Cooled Power Reactors."

7. Regulatory Guide 1.109, "Calculation of Annual Doses to Man From Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I."

8. Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants."

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SECTION 2.4.1 HYDROLOGIC DESCRIPTION

REVIEW RESPONSIBILITIES

Primary - Hydrology-Meteorology Branch (HMB)

Secondary - None

I. AREAS OF REVIEW

The areas of review under this SRP section are:

1. Identification of the interface of the plant with the hydrosphere.

2. Identification of hydrologic causal mechanisms that may require special plant design bases or operating limitations with regard to floods and water supply requirements.

3. Identification of surface and groundwater uses that may be affected by plant operation.

The review of Section 2.4.1.1 (Site and Facilities) of safety analysis reports (SAR) consists of comparing the independently verified or derived hydrologic design bases (see subsequent sections of 2.4) with the critical elevations of safety-related structures and facilities. The review of SAR Section 2.4.1.2 (Hydrosphere) requires identification of the hydrologic characteristics of streams, lakes (e.g., location, size, shape, drainage area, etc.), shore regions, the regional and local groundwater environments, and existing or proposed water control structures (upstream and downstream) influencing the type of flooding mechanisms which may adversely effect safety aspects of plant siting and operation.

II. ACCEPTANCE CRITERIA

The description and elevations of safety-related structures, facilities, and accesses thereto should be sufficiently complete to allow evaluation of the impact of flood design bases. Site topographic maps must be of good quality and of sufficient scale to allow independent analysis of pre- and post-construction drainage patterns. All external plant structures and components should be identified on site maps. Data on surface water users, location with respect to the site, type of use, and quantity of surface water used are required.

The information presented in SAR Section 2.4.1.2 forms the basis for subsequent hydrologic engineering analysis. Therefore, completeness and clarity are of paramount importance. Maps must be legible and adequate in coverage to substantiate applicable data. Inventories of surface water users must be consistent with regional hydrologic inventories reported by applicable state and federal agencies. The description of the hydrologic characteristics of streams, lakes, and shore regions must correspond to those of the United States Geologic Survey (USGS), National Oceanic and Atmospheric Administration (NOAA), Soil Conservation Service (SCS), Corps of Engineers, or appropriate state and river basin agencies. Descriptions of all existing or proposed reservoirs and dams (both upstream and downstream) that could influence conditions at the site must be provided. Descriptions may be obtained from reports of the USGS, United States Bureau of Reclamation (USBR), Corps of Engineers, and others. Generally, reservoir descriptions of a quality similar to those contained in pertinent data sheets of a standard Corps of Engineers Hydrology Design Memorandum are adequate. Tabulations of drainage areas, types of structures, appurtenances, ownership, seismic and spillway design criteria, elevation-storage relationships, and short- and long-term storage allocations must be provided.

III. REVIEW PROCEDURES

The information presented in SAR Section 2.4.1.1 is generally amenable to independent verification through cross-checks with other SAR sections and chapters, available publications relating to hydrologic characteristics of the site region, and by site visits. The review procedure consists of evaluating the completeness of the information and data by sequential comparison with information available from references. Based on the description of the hydrosphere (e.g., geographic location and regional hydrologic features) potential site flood mechanisms are identified. Subsequent SAR sections addressing the mechanisms are cross-checked to assure that data and information required therein for review and substantiation are available.

An important facet of the review procedure for this and other SRP sections in hydrologic areas is the site visit. The site visit provides the principal technical reviewer with independent confirmation of hydrologic characteristics of the site and adjacent environs. The site visit is discussed in Appendix A to this SRP section.

IV. EVALUATION FINDINGS

For construction permit (CP) reviews, findings will consist of a brief general description of the site with respect to the general hydrosphere, and the off-site uses of surface water. For operating license (OL) reviews, findings will consist of the same material, updated as required for new information available since preparation of the CP findings. A sample description for a CP review follows:

> "The proposed site for the ABC Nuclear Plant is located about 26 miles SSE of Augusta, Maine, on the southwest bank of the DEF River at about river mile 152. Plant grade will be at about elevation 220 feet above mean sea level (MSL).

Significant hydrologically related plant features include the river intake structure, the natural draft cooling towers, mechanical draft nuclear service cooling towers (these are redundant towers and serve as the ultimate heat sink), and various groundwater wells."

V. REFERENCES

Because of the geographic diversity of plant sites and the large number of hydrologic references, no specific tabulation is given here. In general, maps and charts by the USGS, NOAA, Army Map Service (AMS), and Federal Aviation Administration (FAA); water-supply papers of the USGS; River Basin Reports of the Corps of Engineers; and other publications of state, federal and other regulatory bodies, describing hydrologic characteristics and water utilization in the plant vicinity and region, are referred to on an "as available" basis. Other SRP sections in the hydrology area (2.4.2 through 2.4.14) contain references that are to be used in evaluating the hydrologic description of the site.

1. Appendix A, SRP section 2.4.1, "Hydrologic Engineering Site Visits," attached.

2. Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants."

APPENDIX A

STANDARD REVIEW PLAN SECTION 2.4.1
HYDROLOGIC ENGINEERING SITE VISITS

I. PURPOSES

The purposes of hydrologic engineering site visits are as follows:

1. Acquaint the reviewer with general site and regional hydrologic characteristics and topography.

2. Confirm the applicant's general appraisal of the site/plant hydrologic interfaces.

3. Review specific hydrologic engineering problem areas with the applicant, his engineers, and his consultants.

The site visit objectives will have been achieved if, in addition to viewing pertinent hydrologic features, the reviewer has had the opportunity to discuss specific questions and concerns with the applicant's hydrologic engineers, and is assured that the questions and concerns are understood. In addition, generally acceptable techniques and procedures necessary to respond to staff concerns should be discussed.

II. PROCEDURES

Questions or items of staff concern are to be developed by the Hydrologic Engineering Section reviewer and discussed in detail with the Section Leader 7-14 days before the scheduled site visit. For any unscheduled site visit (which may be necessary to resolve issues or prepare for hearings), similar questions or items of staff concern should be prepared at least 3 days prior to such site visit and also discussed in detail with the Section Leader.

Areas of overlap or interfaces with reviewers in other areas (such as geology, foundation engineering, auxiliary and power conversion systems, mechanical engineering, effluent treatment systems and structural engineering) should be coordinated before questions or items of staff concern are finalized.

The Section Leader will discuss any unusual or potentially controversial areas of concern with the Chief, HMB, prior to transmittal of the questions or items of staff concern to the Project Manager. Transmittal will be forwarded by memo route slip through the Section Leader.

Site visits are generally to consist of a detailed reconnaissance of site areas and environs with the applicant and technical counterparts, discussions of questions (or items of staff concern), discussions of acceptable methods of analysis, and a general summarization of the areas discussed and conclusions reached.

Normally, a small group composed of the staff reviewer and licensing project manager (LPM) should meet with an applicant representative responsible for responding to staff questions and the applicant's technical advisor. For verbal summarization during the site visit, the recommended method is to have the applicant or his technical advisor summarize the discussions to assure understanding.

III. TRIP REPORT

A trip report on a site visit should be prepared within 5 days of the reviewer's return. The report is to be as brief as possible and should summarize the trip and the areas of discussion and should list the participants in technical discussions.

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SECTION 2.4.2 FLOODS

REVIEW RESPONSIBILITIES

Primary - Hydrology-Meteorology Branch (HMB)

Secondary - None

I. AREAS OF REVIEW

This section of the safety analysis report (SAR) identifies historical flooding (defined as occurrences of abnormally high water stage, or overflow from a stream, floodway, lake, or coastal area) at the proposed site or in the region of the site. It summarizes and identifies the individual types of flood-producing phenomena, and combinations of flood-producing phenomena, considered in establishing the flood design bases for safety-related plant features. It also covers the potential effects of local intense precipitation. Although topical information may appear in SAR sections 2.4.3 through 2.4.7, the types of events considered and the controlling event are reviewed in this section.

The flood history and the potential for flooding are reviewed for the following sources and events. Factors affecting potential runoff (such as urbanization, forest fire, or change in agricultural use), erosion, and sediment deposition are considered in the review.

1. Stream flooding;
   a. Probable maximum flood (PMF) with coincident wind-induced waves, considering dam failure potential due to inadequate capacity, inadequate flood-discharge capability or existing physical condition.
   b. Ice jams, both independently and coincident with a winter probable maximum storm.
   c. Tributary drainage area PMF potential.
   d. Combinations of less severe river floods, coincident with surges and seiches.

2. Surges;
   a. Probable maximum hurricane (PMH) at coastal sites.
   b. PMH wind translated inland and resulting wave action coincident with runoff-induced flood levels.
   c. Probable maximum wind-induced (non-hurricane) storm surges and waves.
   d. Combinations of less severe surges, coincident with runoff floods.

3. Seiches;
   a. Meteorologically-induced in inland lakes (e.g. Great Lakes and harbors) and at coastal harbors and embayments.
   b. Seismically-induced in inland lakes.
   c. Seismically-induced by tsunamis (seismic sea waves) on coastal embayments.
   d. Combinations of less severe surges and seiches, coincident with runoff floods.

4. Tsunamis;
   a. Near field, or local, excitation.
   b. Far field, or distant, excitation.

5. Seismically-induced dam failures (or breaches), and maximum water level at site from:
   a. Failure of dam (or dams) during safe shutdown earthquake (SSE) coincident with 25-year flood.
   b. Failure during operating basis earthquake (OBE) coincident with standard project flood (SPF).
   c. Failure during other earthquakes, coincident with runoff, surge, or seiche floods where the coincidence is at least as likely as for 5.a. and 5.b. above.

6. Flooding caused by landslides;
   a. Flood waves.
   b. Backwater effects due to stream blockage.

7. Ice loadings from water bodies.

II. ACCEPTANCE CRITERIA

For SAR Section 2.4.2.1 (Flood History): The potential flood sources and flood response characteristics identified by the staff's review (described in Review Procedures) are compared to those of the applicant. If similar, the applicant's conclusions are accepted. If, in the staff's opinion, significant discrepancies exist, the applicant will be requested to provide additional data, reestimate the effects on the plant, or revise the applicable flood design bases, as appropriate.

For SAR Section 2.4.2.2 (Flood Design Considerations): The applicant's estimate of controlling flood levels is acceptable if it is no more than 5% less conservative than the staff's independently determined (or verified) estimate. If the applicant's SAR estimate is more than 5 percent less conservative, the applicant should fully document and justify its estimate of the controlling level. Alternately, the applicant may accept the staff's estimate and redesign applicable flood protection.

For SAR Section 2.4.2.3 (Effect of Intense Precipitation): The applicant's estimates of local probable maximum precipitation (PMP) and the capacity of site drainage facilities (including drainage from the roofs of buildings and site ponding) are acceptable if the estimates are no more than 5% less conservative than the corresponding staff's assessment. Similarly, conclusions relating to the potential for any adverse effects of blockage of

site drainage facilities by debris, ice, or snow should be based upon conservative assumptions of storm and vegetation conditions likely to exist during storm periods. If a potential hazard does exist (e.g., the elevation of ponding exceeds the elevation of plant access openings), the applicant should document and justify his local PMP basis and analysis and redesign any affected facilities.

Appropriate sections of the following documents are used by the staff to determine the acceptability of the applicant's data and analyses. Regulatory Guide 1.59 provides guidance for estimating the design basis for flooding considering the worst single phenomena and combinations of less severe phenomena. Regulatory Guide 1.135 describes methods for determining normal water levels. (All estimated water levels should be referenced to mean or normal water levels.) Regulatory Guide 1.29 identifies the safety-related structures, systems, and components, and Regulatory Guide 1.102 describes acceptable flood protection to prevent the safety-related facilities from being adversely affected. Publications of the U.S. Geological Survey (USGS), National Oceanic and Atmospheric Administration (NOAA), Soil Conservation Service (SCS), Corps of Engineers, applicable state and river basin authorities, and other similar agencies are used to verify the applicant's data relating to hydrologic characteristics and extreme events in the region. SRP sections 2.4.3 through 2.4.7 discuss methods of analysis to determine the individual flood producing phenomena.

III. REVIEW PROCEDURES

Construction permit (CP) stage reviews are carried out under this SRP section to evaluate the significance of the controlling flood level with regard to the plant design basis for flood protection. At the operating license (OL) stage, a brief review is carried out to determine if new information has become available since the CP review and to evaluate the significance of the new information with regard to the plant design basis for flood protection. New information might arise, for instance, from the occurrence of a new maximum flood of record in the site region, from identification of a source of major flooding not previously considered, from construction of new dams, from flood plain encroachments, or from advances in predictive models and analytical techniques. If the CP-stage evaluation of flooding potential has been carefully done, all sources of major flooding should have been considered and any new floods of record should fall well within the design basis. Improvements in calculational methods may occur, but generally will be concerned with increased accuracy in stream flow and water level predictions rather than with substantive changes in the flows and levels predicted. Where the OL review reveals that the controlling flood level differs more than 5% less conservatively from the CP evaluation, any supplemental provisions needed in the flood protection design basis should be directed toward early warning measures and procedures for assuring safe shutdown of the plant or toward minor structural modification to accommodate the design flood level.

For SAR Section 2.4.2.1 (Flood History): The staff will review publications of the U.S. Geological Survey (USGS), National Oceanic and Atmospheric Administration (NOAA), Soil Conservation Service (SCS), Corps of Engineers, applicable state and river basin agencies, and others to ensure that historical maximum events and the flood response characteristics

of the region and site have been identified. Similar material, in addition to applicant-supplied information, will be reviewed to identify independently the potential sources of site flooding.

For SAR Section 2.4.2.2 (Flood Design Considerations): The potential flood levels from consideration of the worst single phenomenon and combinations of less severe phenomena, are identified SRP Sections 2.4.3 through 2.4.7 and the controlling flood level is selected. The controlling flood level is compared with the proposed protection levels to assure that the safety-related facilities will not be adversely affected. If appropriate, additional provisions for flood protection will be imposed to assure adequate protection of the safety-related facilities.

For SAR Section 2.4.2.3 (Effect of Local Intense Precipitation): The staff's estimates of flooding potential, excluding flooding potential from thunderstorms, are based on 24-hour PMP estimates (from Hydrometeorological Report 33 and similar NOAA publications for western sites) with time distributions from the Corps of Engineers EM 1110-2-1411. Staff estimates for local intense precipitation caused by thunderstorms are based on PMP estimates from reports such as Reference 11. The staff's estimates are compared with the applicant's estimates to determine conformity to Acceptance Criteria in Subsection II. Runoff models, such as the unit hydrograph if applicable, or other runoff discharge estimates presented in standard texts, are used to estimate discharge on the site drainage system. Where generalized runoff models are used, coefficients used for the site and region are compared to information available at documented locations to evaluate hydrologic conditions used in determining the probable maximum flood for the site-drainage system. Potential ponding on the site is also determined.

The above reviews are performed only when applicable to the site or site region. Some items of review may be done on a generic basis.

IV. EVALUATION FINDINGS

For construction permit reviews, the findings will consist of a statement indicating the completeness of the identification of site flood characteristics and flood design bases. For OL reviews, the flood history will be updated if necessary, with special attention to any new flood of record. Sample statements for CP reviews follow:

> "The maximum flood known to have occurred on the A River was in 1796. The peak discharge at B City, Montana, was estimated to be 360,000 cubic feet per second (cfs). The applicant estimated that a comparable flood would produce water surface elevation at the site of 116 feet MSL. The maximum flood during the period since records were maintained (1883) at B City was 350,000 cfs and occurred on October 3, 1929. These floods occurred prior to construction of several upstream dams. Flood flows are now regulated by C and D Reservoirs as well as by upstream hydropower plants.

"The applicant has estimated potential flooding from rainfall over the E River basin upstream from the site. The probable maximum flood (PMF), the upper level of flooding the staff considers to be reasonably possible, was estimated to produce a flow of 5,000,000 cfs near the City of F. This estimate was made by using 165% of the Corps of Engineers project design flood (PDF) estimate of 3,030,000 cfs at the same location, as modified by upstream flood control reservoirs. The 3,030,000 cfs project design flood flow is estimated to be partially diverted to the leveed G and H Floodways upstream of the site, with 1,500,000 cfs continuing downstream within the levee system past the plant site. The applicant concluded that the PMF could result in overtopping of levees and flooding of the river valley well upstream from the site, thereby causing generally low level flooding in the plant area. The upstream levee overtopping and resulting valley flow during such an event would reduce the flow in the main levee channel adjacent to the site to levels equal to or less than those that would exist during a project design flood. We conclude that the combination of a runoff-type flood less severe than a PMF, but more severe than a PDF, and a coincident levee break in the vicinity of the site could occur before water approaches levee grade upstream. A failure or levee breach, when the levee is full to design capacity (3 feet below the top of the levee adjacent to the site plus the effects of any coincident wind-generated wave activity), would result in a higher water surface at the plant than a PMF spread over the valley as a result of levee failures upstream. At our request the applicant evaluated various modes of levee failure in the vicinity of the plant. One of the conditions postulated is that of a flood, approaching the severity of a PMF, causing a massive failure of the upstream left bank levee along the G Floodway, resulting in flooding around the plant, coincident with a failure of the levee adjacent to the plant site. The applicant estimated the resulting water level at the plant would reach elevation 22.5 feet MSL for this case. The case of an instantaneous levee failure adjacent to the plant, with no upstream levee failure, resulted in an estimated water level of 24.6 feet MSL. The staff concludes that the applicant should design for the conditions associated with the 24.6 feet MSL water level."

V. REFERENCES[1/]

1. "Surface Water Supply of the United States,"[2/] U.S. Geological Survey.

2. "Tide Tables," National Oceanic and Atmospheric Administration (similar situation as identified in footnote 2).

3. Reports of Great Lakes levels by Lake Survey Denver, National Oceanic and Atmospheric Administration.

4. Corps of Engineers records maintained in District and Division Offices, Coastal Engineering Research Center, and Waterways Experiment Station.

5. Regulatory Guide 1.29, "Seismic Design Classification."

6. Regulatory Guide 1.59, "Design Basis Floods for Nuclear Power Plants."

7. "Seasonal Variation of the Probable Maximum Precipitation East of the 105th Meridian for Areas from 10 to 1,000 Square Miles and Durations of 6, 12, 24 and 48 hours," Hydrometeorological Report No. 33, U.S. Weather Bureau (1956).

8. "Standard Project Flood Determinations," Engineering Manual 1110-2-1411, Corps of Engineers, 26 March 1952 (rev. March 1965).

9. Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants."

10. ANSI N170, "Standards for Determining Design Basis Flooding at Power Reactor Sites"

11. "Generalized Estimates of Probable Maximum Precipitation for the United States West of the 105th Meridian for Areas to 400 Square Miles and Durations to 24 Hours," Technical Paper No. 38, U.S. Weather Service, NOAA (1960).

12. Regulatory Guide 1.102, "Flood Protection for Nuclear Power Plants."

13. Regulatory Guide 1.135, "Normal Water Level and Discharge at Nuclear Power Plants."

1/ References for PMP estimates, time distribution, etc., are in SRP section 2.4.3.

2/ "Surface Water Supply" is a continuing series of water discharge measurements by the USGS and others. It is not practical to list all the volumes (called "Water-Supply Papers") that are available. Numerous state and local authorities maintain river discharge, lake level, and tide data.

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SECTION 2.4.3 PROBABLE MAXIMUM FLOOD (PMF) ON STREAMS AND RIVERS

REVIEW RESPONSIBILITIES

Primary - Hydrology & Meteorology Branch (HMB)

Secondary - None

I. AREAS OF REVIEW

In this section of the safety analysis report (SAR) the hydrometeorological design basis is developed to determine the extent of any flood protection required for those structures, systems, and components necessary to assure the capability to shut down the reactor and maintain it in a safe shutdown condition. The areas of review include the probable maximum precipitation (PMP) potential and precipitation losses over the applicable drainage area, the runoff response characteristics of the watershed, the accumulation of flood runoff through river channels and reservoirs, the estimate of the discharge rate trace (hydrograph) of the PMF at the plant site, the determination of PMF water level conditions at the site, and the evaluation of coincident wind-generated wave conditions that could occur with the PMF. Included is a review of the details of design bases for site drainage (which is summarized in SAR Section 2.4.2); a review of the runoff for site drainage and drainage areas adjacent to the plant site, including the roofs of safety-related structures, resulting from potential PMP; and a review of the potential effects from erosion and sedimentation. The analyses involve modeling of physical rainfall and runoff processes to estimate the upper level of possible flood conditions adjacent to and on site.

Regulatory Guide 1.59 describes two positions with respect to flood protection for which a PMF estimate is required to determine the controlling design basis conditions. If Position 1 is chosen, all safety-related systems, structures, and components must be capable of withstanding the effects from the controlling flood design basis. Position 2 limits the review to specific safety-related structures, systems, and components necessary for cold shutdown and maintenance thereof.

II. ACCEPTANCE CRITERIA

The PMF as defined in Regulatory Guide 1.59 has been adopted as one of the conditions to be evaluated in establishing the applicable stream and river flooding design basis referred to in General Design Criterion 2, Appendix A, 10 CFR Part 50. PMF estimates are required

for all adjacent streams or rivers and site drainage (including the consideration of PMP on the roofs of safety-related structures). The criteria for accepting the applicant's PMF-related design basis depend on one of the following three conditions.

1. The elevation attained by the PMF (with coincident wind waves) establishes a required protection level to be used in the design of the facility.

2. The elevation attained by the PMF (with coincident wind waves) is not controlling; the design basis flood protection level is established by another flood phenomena (e.g., the probable maximum hurricane).

3. The site is "dry", that is, the site is well above the elevation attained by a PMF (with coincident wind waves).

When condition 1 is applicable the staff will assess the flood level (described in Subsection III). The assessment may be made independently from basic data, by detailed review and checking of the applicant's analyses, or by comparison with estimates made by others which have been reviewed in detail. The applicant's estimates of the PMF level and the coincident wave action are acceptable if the estimates are no more than 5% less conservative than the staff's estimates. If the applicant's estimates of discharge are more than 5% less conservative than the staff's, the applicant should fully document and justify its estimates or accept the staff's estimates and redesign applicable flood protection.

When conditions 2 or 3 apply, the staff analyses may be less rigorous (described in Subsection III). For condition 2, acceptance is based on the protection level estimated for another flood-producing phenomenon exceeding the staff estimate of PMF water levels. For condition 3, the site grade must be well above the staff assessment of PMF water levels. The evaluation of the adequacy of the margin (difference in flood and site elevations) is generally a matter of engineering judgment. The judgment is based on the confidence in the flood level estimate and the degree of conservatism in each parameter used in the estimate.

Appropriate sections of the following documents are used by the staff to determine the acceptability of the applicant's data and analyses. Regulatory Guide 1.59 provides guidance for estimating the PMF design basis. Regulatory Guide 1.135 describes methods for determining normal water levels. (All estimated water levels should be referenced to mean or normal water levels.) Regulatory Guide 1.29 identifies the safety-related structures, systems, and components, and Regulatory Guide 1.102 describes acceptable flood protection to prevent the safety-related facilities from being adversely affected. Publications of the National Oceanic and Atmospheric Administration (NOAA) and the Corps of Engineers may be used to estimate PMF discharge and water level condition at the site and coincident wind-generated wave activity.

III. REVIEW PROCEDURES

For conditions 1 and 2 (described in Subsection II), the methods used for evaluating flooding potential are separated into two parts - PMF on adjacent streams and local PMF. The review procedure is outlined in the attached Figures 2.4.3-1 (for PMF on adjacent streams) and 2.4.3-2 (for local PMF). (The procedure for evaluating the adequacy of site drainage facilities based on a local PMF is outlined in SRP section 2.4.2.) Corps of Engineers PMF assessments for specific locations or generalized PMF assessments for a geographical area approved by the Chief of Engineers and contained in published or unpublished reports of that agency may be used in lieu of staff-developed analyses. In the absence of such assessments, both large and small basin PMP estimates by the National Oceanic and Atmospheric Administration (NOAA); published techniques of the World Meteorological Organization; and runoff, impoundment, and river routing models of the Corps of Engineers are used by the staff to estimate PMF discharge and water level at the site. A comprehensive review of the applicant's analyses will be performed and a simplified analysis using calculational procedures or models with demonstrably conservative coefficients and assumptions is performed. If the applicant's PMF estimates are within acceptable margins (described in subsection II), the staff positions will indicate concurrence with the applicant's PMF estimates and the SER input will be written accordingly. If the simplified analysis indicates a potential problem with the applicant's estimates, a detailed analysis using more realistic techniques will be performed. The staff will develop a position based on the detailed analysis; resolve, if possible, differences between the applicant's and staff's estimates of PMF design basis; and prepare the SER input accordingly.

Wind-generated wave action will be independently estimated using Corps of Engineers criteria such as the "Shore Protection Manual." When sufficient water depth is available, the significant wave height and runup are used for structural design purposes, and the one percent wave height and runup are used for flood level estimates. Where depth limits wave height, the breaking or broken wave height and runup is used for both purposes.

For condition 3 (i.e., a "dry site"; one not subject to steam flooding by virtue of local topographic considerations), the following procedures apply:

1. Use Corps of Engineers PMF estimates for other sites in the region to develop "regional drainage area vs. PMF discharge (cubic feet per second/square mile)" data, for extrapolation to the site.
2. Envelope the above data points to obtain an estimate of the PMF applicable to the site.
3. Increase the estimate based on a judgment as to the applicability of the basic estimates. An increase in the range of 10 to 50 percent is generally appropriate.
4. If warranted by relative elevation differences between the site and adjacent stream, estimate the flood level at the site using slope-area techniques or water surface profile computations.
5. Estimate wind (2-yr. extreme windspeed) wave runup based on breaking or one percent wave heights. Criteria for estimating windspeed are discussed in ANSI N170 and Reference 16.

6. Compare resultant water level with proposed plant grade and lowest safety-related facility that can be affected.

The above items of review are performed only when applicable to the site or site region. Some items of review may be done on a generic basis.

IV. EVALUATION FINDINGS

For construction permit (CP) reviews the findings will summarize the applicant's and staff's estimates of the peak PMF runoff rate and water level (including allowance for coincident wind-generated wave activity) at the site. If the applicant's estimates are within the criteria (described in Subsection II), staff concurrence will be stated. If the staff's estimates are 5% more conservative than the applicant's estimates, if the flood conditions may adversely affect the proposed plant, and if the applicant has been unable to support his estimates, a statement requiring use of the staff bases will be made. If the flood conditions do not constitute a design basis, the findings will so indicate.

For operating license (OL) reviews which have received detailed PMF reviews during the CP review, the CP conclusions will be referenced. Any flood potential not identified during the CP review will be noted.

If Regulatory Guide 1.59, Position 2, is elected by the applicant, a statement describing lesser design bases will be included in the findings with a staff conclusion of adequacy.

A sample statement for a CP review follows:

> "The Probable Maximum Flood (PMF) resulting from the Probable Maximum Precipitation (PMP) on the ABC River drainage basin yielded an estimated maximum stillwater level at the intake structure on the D & E Canal of about 5.0 feet MSL.
>
> "The PMF resulting from a local PMP storm on the drainage basins for the small streams near the site yielded an estimated maximum stillwater level of about 60 feet MSL, which is about 20 feet below plant grade.
>
> "The local PMF resulting from the estimated local PMP was found not to cause flooding of safety-related facilities, since the site drainage system will be capable of functioning adequately during such a storm. Catch basins will be provided as part of the storm drainage system and will be located throughout the plant site to drain local areas. The plant yard will be graded with gentle slopes away from high points at the plant buildings, and storm water will drain away from the buildings into the local streams at lower elevations."

V. REFERENCES

In addition to the following specific references, Design Memoranda, Civil Works Investigations and research and development reports of the Corps of Engineers and reports of other federal and state agencies relevant to flood estimates at a specific site will be used on an "as available" basis.

1. Reports of the Corps of Engineers, Department of the Army:
EM 1110-2-1411, "Standard Project Flood Determinations," 26 March 1952 (rev. March 1965).

EC 1110-2-27, "Policies and Procedures Pertaining to Determination of Spillway Capacities and Freeboard Allowances for Dams," 19 February 1968.

EM 1110-2-1405, "Flood Hydrograph Analysis and Computations," 31 August 1959.

EM 1110-2-1408, "Routing of Floods Through River Channels," 1 March 1960.

EM 1110-2-1406, "Runoff from Snowmelt," 5 January 1960.

EM 1110-2-1603, "Hydraulic Design of Spillways," 31 March 1965.

EM 1110-2-1409, "Backwater Curves in River Channels," 7 December 1959.

Technical Bulletin No. 8, Sacramento District, "Generalized Snowmelt Runoff Frequencies," September 1962.

EM 1110-2-1601, "Hydraulic Design of Flood Control Channels," 1 July 1970.

EM 1110-2-1607, "Tidal Hydraulics," 2 August 1965.

CE 1308, "Stone Protection," January 1948.

EM 1110-2-1410, "Interior Drainage of Leveed Urban Areas: Hydrology," 3 May 1965.

Technical Report No. 4, Coastal Engineering Research Center, "Shore Protection, Planning and Design" (1966) and "Shore Protection Manual" (1973).

Waterways Experiment Station, "Hydraulic Design Criteria," continuously updated.

TSP37, "Riprap Stability on Earth Embankments Tested in Large- and Small-Scale Wave Tanks," June 1972.

ETL 1110-2-120, "Additional Guidance for Riprap Channel Protection," May 1971.

2. Hydrometeorological Reports of the U.S. Weather Bureau (now U.S. Weather Service, NOAA) Hydrometeorological Branch:
No. 1., "Maximum Possible Precipitation Over the Ompompanoosuc Basin above Union Village, Vt." (1943).

No. 2., "Maximum Possible Precipitation over the Ohio River Basin above Pittsburgh, Pa." (1942).

No. 3., "Maximum Possible Precipitation over the Sacramento Basin of California" (1943).

No. 4., "Maximum Possible Precipitation over the Panama Canal Basin" (1943).

No. 5., "Thunderstorm Rainfall" (1947).

No. 6., "A Preliminary Report on the Probable Occurrence of Excessive Precipitation over Fort Supply Basin, Okla." (1938).

No. 7., "Worst Probable Meteorological Condition on Mill Creek, Butler and Hamilton Counties, Ohio" (1937), unpublished. Supplement (1938).

No. 8., "A Hydrometeorological Analysis of Possible Maximum Precipitation over St. Francis River Basin above Wappapello, Mo." (1938).

No. 9., "A report on the Possible Occurrence of Maximum Precipitation over White River Basin above Mud Mountain Dam Site, Wash." (1939).

No. 10., "Maximum Possible Rainfall over the Arkansas River Basin above Caddoa, Colo." (1939). Supplement (1939).

No. 11., "A Preliminary Report on the Maximum Possible Precipitation over the Dorena, Cottage Grove, and Fern Ridge Basins in the Willamette Basin, Oreg." (1939).

No. 12., "Maximum Possible Precipitation over the Red River Basin above Denison, Tex." (1939).

No. 13., "A Report on the Maximum Possible Precipitation over Cherry Creek Basin in Colorado" (1940).

No. 14., "The Frequency of Flood-Producing Rainfall over the Pajaro River Basin in California" (1940).

No. 15., "A Report on Depth-Frequency Relations of Thunderstorm Rainfall on the Sevier Basin, Utah" (1941).

No. 16., "A Preliminary Report on the Maximum Possible Precipitation over the Potomac and Rappahannock River Basins" (1943).

No. 17., "Maximum Possible Precipitation over the Pecos Basin of New Mexico" (1944), unpublished.

No. 18., "Tentative Estimates of Maximum Possible Flood-Producing Meteorological Conditions in the Columbia River Basin" (1945).

No. 19., "Preliminary Report on Depth-Duration-Frequency Characteristics of Precipitation over the Muskingum Basin for 1- to 9-week Periods" (1945).

No. 20., "An Estimate of Maximum Possible Flood-Producing Meteorological Conditions in the Missouri River Basin above Garrison Dam Site" (1945).

No. 21., "A Hydrometeorological Study of the Los Angeles Area" (1939).

No. 21A., "Preliminary Report on Maximum Possible Precipitation, Los Angeles Area, California" (1944).

No. 21B., "Revised Report on Maximum Possible Precipitation, Los Angeles Area, California" (1945).

No. 22., "An Estimate of Maximum Possible Flood-Producing Meteorological Conditions in the Missouri River Basin Between Garrison and Fort Randall" (1946).

No. 23., "Generalized Estimates of Maximum Possible Precipitation over the United States East of the 105th Meridian, for Areas of 10, 200, and 500 Square Miles" (1947).

No. 24., "Maximum Possible Precipitation over the San Joaquin Basin, Calif." (1947).

No. 25., "Representative 12-hour Dewpoints in Major United States Storms East of the Continental Divide" (1947).

No. 25A., "Representative 12-hour Dewpoints in Major United States Storms East of the Continental Divide," 2d edition (1949).

No. 26., "Analysis of Winds over Lake Okeechobee during Tropical Storm of August 26-27, 1949" (1951).

No. 27., "Estimate of Maximum Possible Precipitation, Rio Grande Basin, Fort Quitman to Zapata" (1951).

No. 28., "Generalized Estimate of Maximum Possible Precipitation over New England and New York" (1952).

No. 29., "Seasonal Variation of the Standard Project Storm for Areas of 200 and 1,000 Square Miles East of the 105th Meridian" (1953).

No. 30., "Meteorology of Floods at St. Louis" (1953), unpublished.

No. 31., "Analysis and Synthesis of Hurricane Wind Patterns over Lake Okeechobee, Florida" (1954).

No. 32., "Characteristics of United States Hurricanes Pertinent to Levee Design for Lake Okeechobee, Florida" (1954).

No. 33., "Seasonal Variation of the Probable Maximum Precipitation East of the 105th Meridian for Areas from 10 to 1,000 Square Miles and Durations of 6, 12, 24, and 48 Hours" (1956).

Draft Report, "All-Season Probable Maximum Precipitation, United States East of the 105th Meridian for Areas From 1,000 to 20,000 Square Miles and Durations From 6 to 72 Hours" (1972).

No. 34., "Meteorology of Flood-Producing Storms in the Mississippi River Basin" (1956).

No. 35., "Meteorology of Hypothetical Flood Sequences in the Mississippi River Basin" (1959).

No. 36., "Interim Report, Probable Maximum Precipitation in California" (1961), revised (1969).

No. 37., "Meteorology of Hydrologically Critical Storms in California" (1962).

No. 38., "Meteorology of Flood-Producing Storms in the Ohio River Basin" (1961).

No. 39., "Probable Maximum Precipitation in the Hawaiian Islands" (1963).

No. 40., "Probable Maximum Precipitation, Susquehanna River Drainage above Harrisburg, Pa." (1965).

No. 41., "Probable Maximum and TVA Precipitation over the Tennessee River Basin above Chattanooga" (1965).

No. 42., "Meteorological Conditions for the Probable Maximum Flood on the Yukon River above Rampart, Alaska" (1966).

No. 43., "Probable Maximum Precipitation, Northwest States" (1966).

No. 44., "Probable Maximum Precipitation over South Platte River, Colorado, and Minnesota River, Minnesota" (1969).

No. 45., "Probable Maximum and TVA Precipitation for Tennessee River Basin up to 3,000 Square Miles in Area and Durations to 72 Hours" (1969).

No. 46., "Probable Maximum Precipitation, Mekong River Basin" (1970).

No. 47., "Meteorological Criteria for Extreme Floods For Four Basins in the Tennessee and Cumberland River Basins" (1973).

No. 48., "Probable Maximum Precipitation and Snowmelt Criteria for Red River of the North Above Pembinz, and Souris River Above Minot, North Dakota" (1973).

3. Technical Papers of the U.S. Weather Bureau (Now U.S. Weather Service, NOAA): No. 2, "Maximum Recorded United States Point Rainfall for 5 Minutes to 24 Hours at 207 First Order Stations," Rev. (1963).

No. 5., "Highest Persisting Dewpoints in the Western United States" (1948).

No. 10., "Mean Precipitable Water in the United States" (1949).

No. 13., "Mean Monthly and Annual Evaporation Data from Free Water Surface for the United States, Alaska, Hawaii, and the West Indies" (1950).

No. 14., "Tables of Precipitable Water and Other Factors for a Saturated Pseudo-Adiabatic Atmosphere" (1951).

No. 15., "Maximum Station Precipitation for 1, 2, 3, 6, 12, and 24 Hours:" Part I: Utah (1951); Part II: Idaho (1951); Part III: Florida (1952); Part IV: Maryland, Delaware, and District of Columbia (1953); Part V: New Jersey (1953); Part VI: New England (1953); Part VII: South Carolina (1953); Part VIII: Virginia (1954); Part IX: Georgia (1954); Part X: New York (1954); Part XI: North Carolina (1955); Part XII: Oregon (1955); Part XIII: Kentucky (1955); Part XIV: Louisiana (1955); Part XV: Alabama (1955); Part XVI: Pennsylvania (1956); Part XVII: Mississippi (1956); Part XVIII: West Virginia (1956); Part XIX: Tennessee (1956); Part XX: Indiana (1956); Part XXI: Illinois (1958); Part XXII: Ohio (1958); Part XXIII: California (1959); Part XXIV: Texas (1959); Part XXV: Arkansas (1960); Part XXVI: Oklahoma (1961).

No. 16., "Maximum 24-Hour Precipitation in the United States" (1952).

No. 25., "Rainfall Intensity-Duration-Frequency Curves for Selected Stations in the United States, Alaska, Hawaiian Islands, and Puerto Rico" (1955).

No. 28., "Rainfall Intensities for Local Drainage Design in Western United States for Durations of 20 Minutes to 24 Hours and 1- to 100-Year Return Periods" (1956).

No. 37., "Evaporation Maps for the United States" (1959).

No. 38., "Generalized Estimates of Probable Maximum Precipitation for the United States West of the 105th Meridian for Areas to 400 Square Miles and Durations to 24 Hours" (1960)

No. 40., "Rainfall Frequency Atlas of the United States for Durations from 30 Minutes to 24 Hours and Return Periods from 1 to 100 Years" (1961).

No. 42., "Generalized Estimates of Probable Maximum Precipitation and Rainfall-Frequency Data for Puerto Rico and Virgin Islands" (1961).

No. 43., "Rainfall-Frequency Atlas of the Hawaiian Islands for Areas to 200 Square Miles, Durations to 24 Hours, and Return Periods from 1 to 100 years" (1962).

No. 47., "Probable Maximum Precipitation and Rainfall-Frequency Data for Alaska for Areas to 400 Square Miles, Durations to 24 Hours, and Return Periods from 1 to 100 Years" (1963).

No. 48., "Characteristics of the Hurricane Storm Surge" (1963).

4. Unpublished Hydrometeorological Reports of the U.S. Weather Bureau (now U.S. Weather Service, NOAA):
"Rappahannock River above Salem Church Dam Site, Va." (11/28/50).

"Potomac River, Va., Md., W. Va. (12 sub-basins)" (6/29/56).

"Delaware River above Trenton, Chestnut Hill, and Belvidere Dam Sites" (11/19/56).

"Delaware River above Tock's Island Dam Site" (12/16/65).

"St. John River above Dickey Dam Site, and Between Dicky and Lincoln School Dam Sites, Maine" (12/20/66).

"Coosa River above Howell Mill Shoals Dam Site, Ala." (3/3/50).

"Cape Fear River above Smiley Falls Dam Site, N.C." (11/16/50).

"Savannah River above Hartwell Dam Site, N.C." (1/5/51).

"Alabama and Appalachicola Rivers, Ala. and Fla." (3/19/52).

"Black Warrior River above Holt Lock Dam Site, Ala." (12/10/59).

"South Fork of Holston River above Boone Dam Site, Tenn." (8/14/50).

"Allegheny River above Allegheny River Reservoir, Pa." (9/28/56).

"Kentucky River, Ky. (2 basins)" (3/12/58).

"New River above Moores Ferry Dam Site, Va." (5/13/63).

"Licking River, Ky, and White River, Ind." (11/9/64).

"Iowa River above Coralville Dam Site, Iowa" (11/20/47).

"Des Moines River above Saylorville, Iowa and Howell Dam Site, Iowa" (3/19/48).

"Salt River, Mo." (1/21/55).

"James River above Jamestown Dam Site, N. Dak." (9/16/48).

"Big Blue River above Tuttle Creek Dam Site, Kans." (10/23/51).

"Republican River at (a) above proposed Milford Dam Site, Kan.; and (b) between Harlan Co. Dam and proposed Milford Dam Site, Kans." (11/24/58).

"Meramec River Basin, Missouri" (12/21/61).

"Republican River above Harlan Co. Res., Neb." (3/7/69).

"Canadian River above Eufaula Dam Site, Okla." (12/19/47).

"White River above Table Rock Dam Site, Mo." (3/19/48).

"Eleven Point River above Water Valley Dam Site, Ark." (3/19/48).

"Kiamichi River above Hugo Dam Site, Okla." (4/9/48).

"Boggy Creek above Boswell Dam Site, Okla." (4/9/48).

"North Canadian River above Optima (Hardesty) Dam Site, Okla." (12/22/49).

"Lower Canadian River, Okla." (6/10/48).

"Gaines Creek Dam Site, Okla." (5/13/48).

"Onapa-Canadian (combined) Dam Site, Okla." (5/13/48).

"Verdigris River above Oologah Dam Site, Okla." (5/4/50).

"Little Red River above Green Ferry, Ark." (7/24/50).

"Grand (Neosho) River above Strawn Dam Site, Kans." (11/14/51).

"Pinon Canyon above Trinidad, Colo." (4/10/52).

"Beaver Reservoir, White River, Ark." (12/1/55).

"Kisatchie Dam Site on Kisatchie Bayou, La." (3/1/56).

"Cypress Creek above Mooringsport, La." (8/27/56).

"Little River above at (a) Millwood Dam Site, Ark.; and (b) Broken Bow, Okla." (5/14/59).

"White River Drainage above Wolf Bayou, Ark." (3/31/66).

"Upper Arkansas River, Colorado (sub-basins)" (2/13/67).

"Arkansas River Drainage Between John Martin Dam, Colo., and Great Bend, Kans." (9/23/69).

"Leon River above Belton Dam Site, Tex." (12/9/47).

"Jemez Creek, N. Mex." (12/9/49).

"Chama River above Chamita Dam Site, N. Mex." (1/18/50).

"Rio Hondo above Two Rivers Reservoir, N. Mex." (12/19/56).

"Richland Creek, Tex." (4/6/56).

"Basque River above Waco Reservoir, Tex." (4/6/56).

"Leon River above Proctor Reservoir Project near Hasse, Tex." (12/5/56).

"Pecos River above Alamogardo Reservoir, N. Mex." (7/24/57).

"Pecos River above Los Esteros, N. Mex." (7/24/57).

"Intervening Drainage between Los Esteros and Alamogardo, N. Mex." (7/24/57).

"Rio Grande between Cerro and Cochiti Dam Site, N. Mex." (2/26/58).

"Combined Drainage of Santa Fe Creek and Rio Galisto above Galisto Dam Site, N. Mex." (2/26/58).

"Lamposas River above proposed Lamposas Dam Site, Tex." (4/17/58).

"Navasota River, Tex. (7 sub-basins)" (11/2/59).

"Colorado River above Fox Crossing, Tex." (11/12/63).

"Lower Rio Grande, United States and Mexico (between Falcon and Anzalduas Dams)" (7/68).

"Gila River above Coolidge Dam Site, Ariz." (9/14/53).

"Queens Creek, Gila River Basin, Ariz." (4/26/55).

"Bill Williams River above proposed Alamo Dam Site, Ariz." (1/14/58).

"Santa Rosa Wash Basin, Ariz." (8/2/68).

"Black Creek, Ariz." (6/20/69).

"Preliminary Estimate for Drainages North of Phoenix, Ariz." (9/29/72).

"Humboldt River, Devils Gate Dam Site, Nev." (11/20/51).

"Mathews Canyon Dam Site (Virgin River), Nev. and Pine Canyon Dam Site (Virgin River), Nev." (8/9/54).

"Dell Canyon Reservoir, Utah" (8/26/57).

"Las Vegas Wash, Nev." (11/22/60).

"Henderson Wash, Nev." (11/22/60).

"West Fork (Mojave River), Calif." (11/22/60).

"Tahchevah Creek, Calif." (11/22/60).

"San Gorgonio River above Cabazon Dam Site, Calif." (4/13/62).

"Whitewater River above Garnet Dam Site, Calif." (4/13/62).

"Martis Creek, Calif." (3/18/64).

"Merced River, Calif." (6/4/62).

"American River above Folsom Dam, Calif." (8/1/68).

"North and Middle Forks of American River above Auburn Dam Site, Calif." (8/1/68).

"Intervening Drainage between Auburn Dam Site and Folsom Dam" (8/1/68).

"Yuba River above Marysville, Calif." (11/29/68).

"Los Angeles District, Calif. (18 basins in Calif., Nev. and Ariz.)" (12/2/68).

"San Diego River Watershed, Calif. (13 sub-basins)" (3/16/73).

"Skagway River, Alaska" (7/8/47).

"Bradley Lake Basin, Alaska" (5/19/61).

"Chena River, Alaska" (8/1/62).

"Long Lake portion of the Snettisham Project" (4/19/65).

"Takatz Creek, Baranof Island, Alaska" (2/21/67).

"Tanana River Basin for (a) Chena River above Chena Dam Site; (b) Little Chena River above Little Chena Dam; and (c) Tana River between Tanacross and Nenana, Alaska" (6/5/69).

"Preliminary Estimates, Vicinity of Junea: Mendenhall River, Lemon Creek, and Montana Creek" (11/7/69).

"Preliminary Estimates, Vicinity of Ketchikan: Whipple Creek near Wards Cove, Carlanna Creek near Ketchikan, Hoadley Creek near Ketchikan, and Ketchikan Creek near Ketchikan" (1/7/74).

"Eastern Panama and Northwest Colombia" (9/65).

"Hypothetical Rainstorms over Rio Atrato Basin, Colombia, South America" (7/67).

"Probable Maximum Thunderstorm Precipitation Estimates Southwest States" (3/30/73).

5. J. R. Weggel, "Maximum Breaker Height," Jour. Waterways, Harbors and Coastal Engineering Division, Proc. Am. Soc. of Civil Engineers, Vol. 98, No. WW4, pp. 529-548 (1972).

6. Technical Note 98, "Estimation of Maximum Floods," WMO-No. 233, World Meteorological Organization (1969).

7. C. O. Clark, "Storage and the Unit Hydrograph," Trans. Am. Soc. Civil Engineers, Vol. 110, No. 2261, pp. 1419-1488 (1945).

8. U.S. Department of Commerce, "Snow Hydrology," PB-151660, undated.

9. Bureau of Reclamation, "Effect of Snow Compaction from Rain on Snow," Engineering Monograph No. 35, U.S. Department of the Interior (1966).

10. Bureau of Reclamation, "Design of Small Dams," Second Edition, U.S. Department of the Interior (1973).

11. Regulatory Guide 1.59, "Design Basis Floods for Nuclear Power Plants."

12. Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants."

13. Regulatory Guide 1.102, "Flood Protection for Nuclear Power Plants."

14. Regulatory Guide 1.135, "Normal Water Level and Discharge at Nuclear Power Plants."

15. Regulatory Guide 1.29, "Seismic Design Classification."

16. H. C. S. Thom, "New Distribution of Extreme Winds in the United States," Journal of the Structural Division, American Society of Civil Engineers, ST7, July 1968.

17. ANSI N170, "Standards for Determining Design Basis Flooding at Power Reactor Sites".

FIGURE 2.4.3-1

STANDARD REVIEW PLAN SECTION 2.4.3
FLOODS ON STREAMS & RIVERS

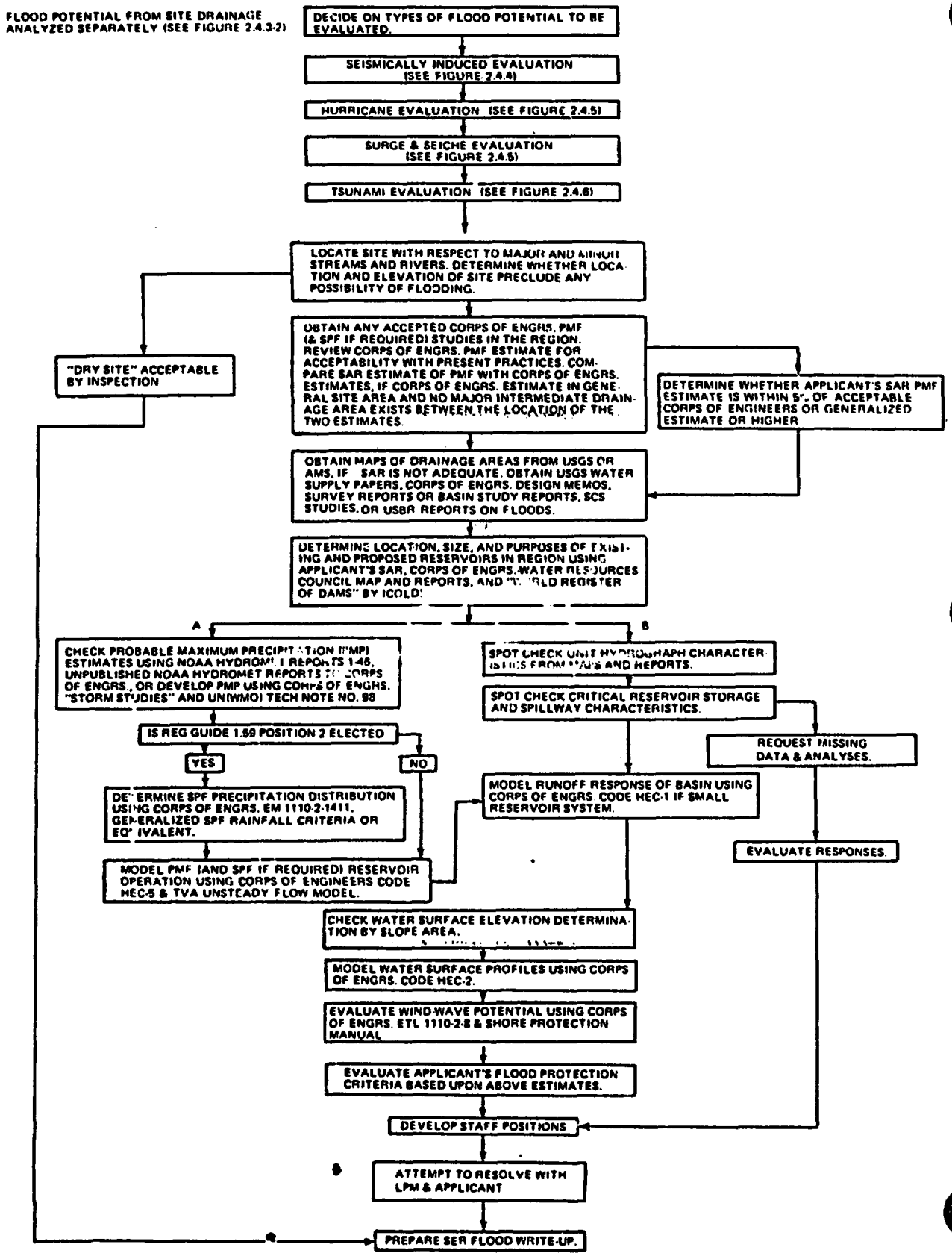
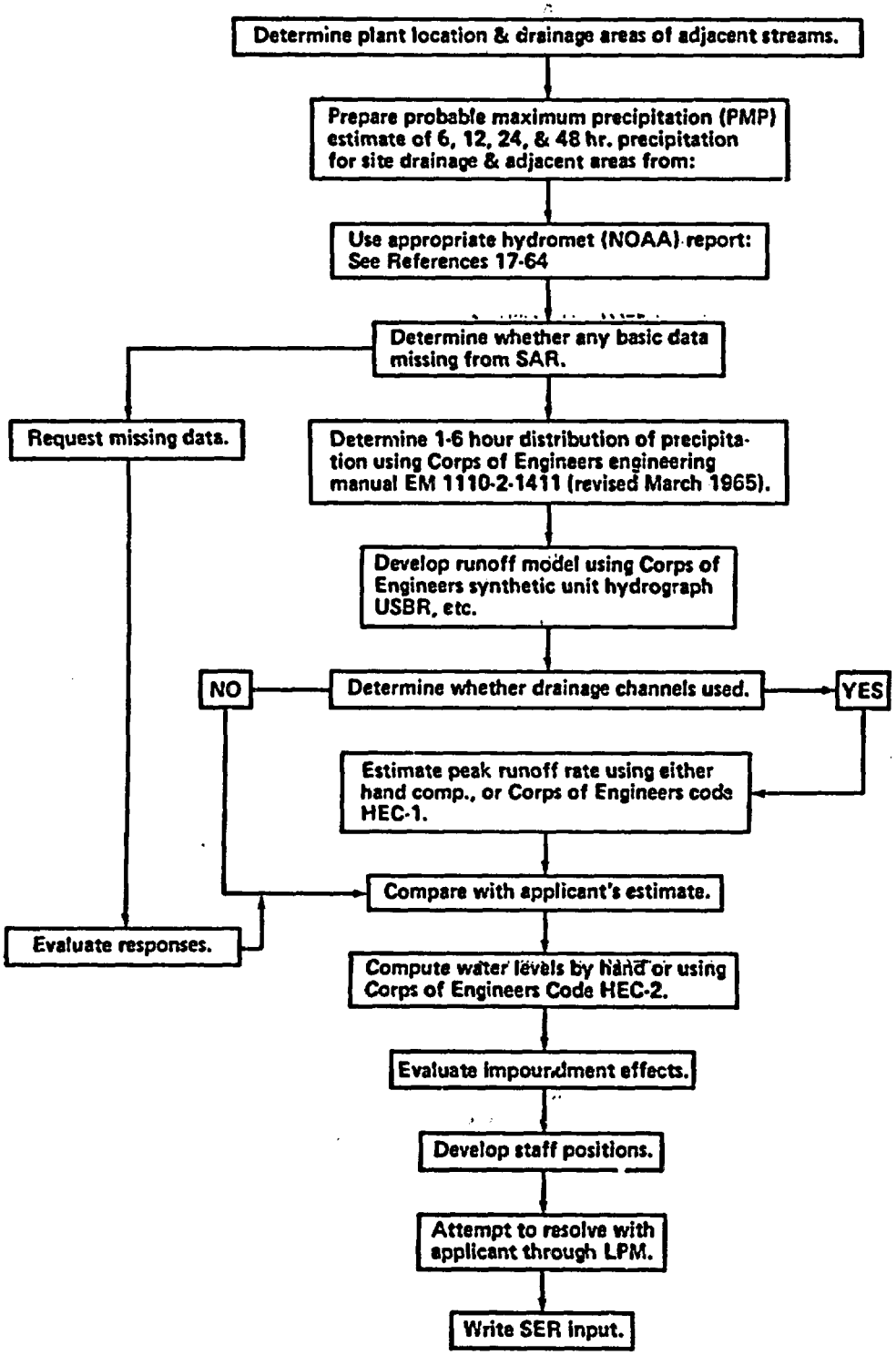


Figure 2. 4. 3-2

Standard Review Plan Section 2.4.3

Site Drainage and Adjacent Drainage



U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

SECTION 2.4.4 POTENTIAL DAM FAILURES (SEISMICALLY-INDUCED)

REVIEW RESPONSIBILITIES

Primary - Hydrology and Meteorology Branch (HMB)

Secondary - None

I. AREAS OF REVIEW

In this section of the safety analysis report (SAR) the hydrogeologic design basis is developed to assure consideration in plant design of any potential hazard to the safety-related facilities due to the failure of upstream and downstream water control structures from seismic causes. The areas of review include consideration of flood waves (bores) from severe breaching of upstream dams and the potential loss of water supply due to failure of a downstream dam, domino-type failures of dams, landslides, and effects of sediment deposition and erosion.

When data are provided to show that seismic events will not cause failures of upstream dams that could produce the governing flood at the plant, this section may contain additional data and other information to support a contention that the dams are equivalent to seismic Category I structures and will survive a local equivalent of the safe shutdown earthquake (SSE) or will survive the operating basis earthquake (OBE). In such cases the Geosciences Branch (GB) and Structural Engineering Branch (SEB), as necessary, will evaluate the data necessary to justify such a classification. GB and SEB review procedures are outlined in the appropriate geosciences and structural SRP sections. The balance of this SRP section applies to the hydrologic analyses of non-seismic Catagory I dams and to Category I dams that could be affected by floodwaves caused by upstream failures of the non-seismic Category I dams.

Where analyses are provided in support of either a conclusion that a probable maximum flood (PMF) should be the design basis flood for a stream, or that a postulated or arbitrarily assumed seismically-induced flood is the design basis flood for a stream, the areas of review consist of the following:

1. Conservatism of modes of assumed dam failure and deposition of debris downstream.

2. Consideration of full flood control reservoirs.

3. Conservatism of downstream flow rates and levels depending on whether failure is postulated with an equivalent SSE coincident with a 25-year flood, or an OBE coincident with a standard project flood (SPF). An SPF is considered to be about forty percent of a PMF.

4. Flood wave attenuation to downstream dams, or to the site, whichever would be encountered first.

5. Potential for mutiple dam failures; flood wave effects and potential for failure of downstream dams.

6. Hydraulic failure of downstream dams induced by upstream failures.

7. Dynamic effects on exposed plant facilities of possible bores.

8. Conservatism (see item 3 above) of ambient flow conditions for downstream dam failures that can influence safety-related water supplies.

9. Applicability and conservatism of models used to predict the effects of dam failure floods.

II. ACCEPTANCE CRITERIA

The staff will review the applicant's analyses and independently assess the coincident river flows at the site and at the dams being analyzed. The acceptable "worst conditions" to be postulated for analysis of upstream failures are: (1) dam able to withstand SSE (equivalent to seismic Category I structures)--assume no failures; (2) dam failure caused by SSE--assume dam failure coincident with 25-year flood and 2-year extreme windspeed at the site; and (3) dam failure caused by OBE--assume dam failure coincident with SPF and 2-year extreme windspeed at the site. The applicant's estimates (which may include landslide-induced failures) of the flood discharge resulting from the "worst conditions" should be no more than 5 percent less conservative than the staff's estimates to be acceptable. If the applicant's estimates differ by more than 5 percent, the applicant should fully document and justify its estimates or accept the staff's estimates and redesign applicable flood protection.

For SAR Section 2.4.4.1 (Dam Failure Permutations): The location of dams and potentially "likely" or severe modes of failure must be identified. The potential for multiple, seismically-induced dam failures and the domino failure of a series of dams must be discussed. Approved models of the Corps of Engineers and the Tennessee Valley Authority are used to predict the downstream water levels resulting from a dam breach (Refs. 4, 8, 13, 14 and 15). First-time use of other models will require complete model description

and documentation. Acceptance of the model (and subsequent analyses) is based on the staff review of model theory, available verification, and application. A determination of the peak flow rate and water level at the site for the worst possible combination of dam failures and a summary analysis (that substantiates the condition as the critical permutation) must be presented, along with a description (and the bases) of all coefficients and methods used. Also, the effects of other concurrent events on plant safety, such as blockage of the river and waterborne missiles, must be considered.

For SAR Sections 2.4.4.2 (Unsteady Flow Analysis of Potential Dam Failures) and 2.4.4.3 (Water Level at Plant Site): The effects of coincident and antecedent flood flows (or low flows for downstream structures) on initial pool levels must be considered. Use of the methods given in References 1 or 3 is acceptable for determination of initial pool levels. Depending upon estimated failure modes and the elevation difference between plant grade and normal river levels, it may be acceptable to use conservative simplified procedures to estimate flood levels at the site. Where calculated flood levels using simplified methods are at or above plant grade and using assumptions which cannot be demonstrated as conservative, it will be necessary to use unsteady flow methods to develop flood levels at the site. References 8 and 9 are acceptable methods; however, other programs would be acceptable with proper documentation and justification. Computations, coefficients, and methods used to establish the water level at the site for the most critical dam failures must be summarized. Coincident wind-generated wave activity should be considered in a manner similar to that discussed in SRP Section 2.4.3.

Appropriate sections of the guides described below are used by the staff to determine the acceptability of the applicant's data and analyses. Regulatory Guide 1.59 provides guidance for estimating the design basis for flooding considering the worst single phenomena and combination of less severe phenomena. Regulatory Guide 1.135 describes methods for determining normal water levels. (All estimated water levels should be referenced to normal or mean water levels.) Regulatory Guide 1.29 identifies the safety-related structures, systems, and components, and Regulatory Guide 1.102 describes acceptable flood protection to prevent the safety-related facilities from being adversely affected.

III. REVIEW PROCEDURES

The conservatism of the applicant's estimates of flood potential and low water levels from seismically-induced structure failures is judged against the criteria indicated in subsection II above. An analysis is performed using simplified, conservative procedures (such an instantaneous failure, coincident SPF flows, minimal flood wave attenuation, and extrapolated site discharge-rating curves). Techniques for such analyses are identified in standard hydraulic design references and text books, such as those listed in the reference section. If no potential flood problem exists, the staff safety evaluation report (SER) input is written accordingly. If the simplified analysis indicates a potential flooding problem, the analysis is repeated using a more refined technique which may include time rate of failure and hydrometeorologically compatible storm

centerings. Detailed failure models, such as those of the Corps of Engineers and the Tennessee Valley Authority, are utilized to identify the cutflows from various failure modes. Models of the Corps of Engineers or the Tennessee Valley Authority are used to identify the outflow characteristics and resultant water level at the site (Refs. 4, 8, 13, 14 and 15). The staff will develop a position based on the analyses performed; resolve, if possible, differences between the applicant's and staff's estimates; and write the SER input accordingly.

The above reviews are performed only when applicable to the site or site region. Some items of review may be done on a generic basis.

IV. EVALUATION FINDINGS

For construction permit (CP) reviews, the findings will summarize the applicant and staff evaluations of the design basis maximum and minimum water levels caused by seismically-induced dam failures. If the applicant's estimates are within acceptable margins (described in subsection II), staff concurrence in the applicant's estimates will be stated. If the applicant's estimates are not within acceptable margins, and if the plant may be adversely affected, a position requiring use of the staff bases will be stated. If no seismically-induced dam failure review was undertaken at the construction permit stage (of the scope described), this fact will be indicated.

For operating license (OL) reviews of cases for which detailed seismically-induced dam failure analyses were made during the CP review, the CP-stage conclusions will be referenced. In addition, any further review done to reaffirm the maximum or minimum water levels based on any new information will be described and the results and conclusions stated.

Sample statements for CP reviews follow:

> "The distance (more than 300 miles) to upstream reservoirs of appreciable size is such that the staff assessment leads to the conclusion that their arbitrarily assumed failure, under NRC criteria of reasonably postulated combinations of floods and earthquakes, would not constitute a threat to the plant worse than that due to a severe runoff-type flood or to hurricane-induced surge.
>
> "Dam failure-caused 'worst case' floods were evaluated by the applicant based upon failures with consideration of only the location and sizes of upstream impoundments, and not on inherent capability of such structures to resist earthquakes, volcanic activity and severe landslide-induced floods. The most severe flood of this kind was estimated based upon an assumed catastrophic failure of Dam A some 420 miles upstream. The peak flow at the site from such a flood was estimated to be 3,000,000 cfs. This flow is estimated to occur about two days after the dam failure and reach elevation 41 feet MSL. Smaller dams on the river between Dam A and the site were also evaluated for such a flood and, it was concluded, would probably also fail.

centerings. Detailed failure models, such as those of the Corps of Engineers and the Tennessee Valley Authority, are utilized to identify the outflows from various failure modes. Models of the Corps of Engineers or the Tennessee Valley Authority are used to identify the outflow characteristics and resultant water level at the site (Refs. 4, 8, 13, 14 and 15). The staff will develop a position based on the analyses performed; resolve, if possible, differences between the applicant's and staff's estimates; and write the SER input accordingly.

The above reviews are performed only when applicable to the site or site region. Some items of review may be done on a generic basis.

IV. EVALUATION FINDINGS

For construction permit (CP) reviews, the findings will summarize the applicant and staff evaluations of the design basis maximum and minimum water levels caused by seismically-induced dam failures. If the applicant's estimates are within acceptable margins (described in subsection II), staff concurrence in the applicant's estimates will be stated. If the applicant's estimates are not within acceptable margins, and if the plant may be adversely affected, a position requiring use of the staff bases will be stated. If no seismically-induced dam failure review was undertaken at the construction permit stage (of the scope described), this fact will be indicated.

For operating license (OL) reviews of cases for which detailed seismically-induced dam failure analyses were made during the CP review, the CP-stage conclusions will be referenced. In addition, any further review done to reaffirm the maximum or minimum water levels based on any new information will be described and the results and conclusions stated.

Sample statements for CP reviews follow:

> "The distance (more than 300 miles) to upstream reservoirs of appreciable size is such that the staff assessment leads to the conclusion that their arbitrarily assumed failure, under NRC criteria of reasonably postulated combinations of floods and earthquakes, would not constitute a threat to the plant worse than that due to a severe runoff-type flood or to hurricane-induced surge.
>
> "Dam failure-caused 'worst case' floods were evaluated by the applicant based upon failures with consideration of only the location and sizes of upstream impoundments, and not on inherent capability of such structures to resist earthquakes, volcanic activity and severe landslide-induced floods. The most severe flood of this kind was estimated based upon an assumed catastrophic failure of Dam A some 420 miles upstream. The peak flow at the site from such a flood was estimated to be 3,000,000 cfs. This flow is estimated to occur about two days after the dam failure and reach elevation 41 feet MSL. Smaller dams on the river between Dam A and the site were also evaluated for such a flood and, it was concluded, would probably also fail.

13. W. A. Thomas, "A Method for Analyzing Effects of Dam Failures in Design Studies," Corps of Engineers Hydrologic Engineering Center, Davis California (for presentation at the ASCE Hydraulics Division Specialty Conference, Cornell University, August 1972).

14. "Flow Through a Breached Dam," Military Hydrology Bulletin No. 9, Corps of Engineers (1957).

15. "Floods Resulting From Suddenly Breached Dams, Conditions of High Resistance," Misc. Paper No. 2-374, Report 2, Corps of Engineers (1961).

16. Bureau of Reclamation, "Flood Routing," Chapter 6/0 in "Flood Hydrology," Part 6 in "Water Studies," Volume IV, U.S. Department of the Interior (1947).

17. Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants."

18. Regulatory Guide 1.59, "Flood Design Basis for Nuclear Power Plants."

19. ANSI N170, "Standards for Determining Design Basis Flooding at Power Reactor Sites" (1976).

20. Regulatory Guide 1.29, "Seismic Design Classification."

21. Regulatory Guide 1.102, "Flood Protection for Nuclear Power Plants."

22. Regulatory Guide 1.135, "Normal Water Levels and Discharge at Nuclear Power Plants."

23. 10 CFR Part 100, Appendix A, "Seismic and Geologic Siting Criteria for Nuclear Power Plants."

Figure 2.4.4-1

STANDARD REVIEW PLAN SECTION 2.4.4
SEISMICALLY - INDUCED FLOODS

Review location and sizes of upstream and immediately downstream dams using SAR & other available dam descriptions.

Determine which structures (1 or more) are hydrologically controlling based on size and location.

Perform quicky "pull the plug" analyses assuming half PMP, minimal flood wave attenuation, and extrapolated site rating curve.

Compare "quicky" estimate with applicant's analysis & location and elevations of safety-related facilities.

NO — Decide on whether potential flood problem exists. — YES

Request missing data.

Perform refined "quicky" analysis.

Evaluate answers & undertake detailed analysis of controlling cases using conceptual models of failures, hydraulic analyses of outflow characteristics and computerized models (corps of engrs.) of unsteady flow, HEC-1, HEC-2, and others.

Develop staff positions.

Attempt to resolve differences with applicant thru LPM.

Write SER input.

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SECTION 2.4.5 PROBABLE MAXIMUM SURGE AND SEICHE FLOODING

REVIEW RESPONSIBILITIES

Primary - Hydrology-Meteorology Branch (HMB)

Secondary - None

I. AREAS OF REVIEW

In this section of the safety analysis report (SAR) the hydrometeorological design basis is developed to determine the extent of flood protection required for safety-related plant systems. The areas of review include the characteristics of the assumed Probable Maximum Hurricane or other probable maximum wind storms and the techniques, methodologies, and parameters used in the determination of the design surge and/or seiche. Antecedent water levels, storm tracks, methods of analysis, coincident wind-generated wave action and wave runup on safety-related structures, potential for wave oscillation at the natural periodicity, and the resultant design bases for surge and seiche flooding are also reviewed.

II. ACCEPTANCE CRITERIA

If it has been determined that surge and seiche flooding estimates are necessary to identify flood design bases, the applicant's analysis will be considered complete and acceptable if the following areas are addressed and can be independently and comparably evaluated from the applicant's submission.

1. All reasonable combinations of Probable Maximum Hurricane, moving squall line, or other cyclonic wind storm parameters are investigated, and the most critical combination is selected for use in estimating a water level.

2. Models used in the evaluation are verified or have been previously approved by the staff.

3. Detailed descriptions of bottom profiles are provided (or are readily obtainable) to enable an independent staff estimate of surge levels.

4. Detailed descriptions of shoreline protection and safety-related facilities are provided to enable an independent staff estimate of wind-generated waves, runup, and potential erosion and sedimentation.

5. Ambient water levels, including tides and sea level anomalies, are estimated using NOAA and Corps of Engineers publications as described below.

6. Combinations of surge levels and waves that may be critical to plant design are considered, and adequate information is supplied to allow a determination that no adverse combinations have been omitted.

7. If Regulatory Guide 1.59, Position 2, is elected by the applicant, the design basis for flood protection of all safety-related facilities identified in Regulatory Guide 1.29 must be shown to be adequate in terms of time required for implementation of any emergency procedures. The applicant must also demonstrate that all potential flood situations that could negate the time and capability to initiate flood emergency procedures are provided for in the less severe design basis selected.

This section of the SAR may also state with justification that surge and seiche flooding estimates are not necessary to identify the flood design basis (e.g., the site is not near a large body of water).

Hydrometeorological estimates and criteria for development of Probable Maximum Hurricanes for east and Gulf coast sites, squall lines for the Great Lakes, and severe cyclonic wind storms for all lake sites by the Corps of Engineers, National Oceanic and Atmospheric Administration (NOAA), and the staff are used for evaluating the conservatism of the applicant's estimates of severe windstorm conditions, as discussed in Regulatory Guide 1.59. The Corps of Engineers and NOAA criteria require variation of the basic meteorological parameters within given limits to determine the most severe combination that could result. The applicant's hydrometeorological analysis should be based on the most critical combination of these parameters.

Data from publications of NOAA, the Corps of Engineers, and other sources (such as tide tables, tide records, and historical lake level records) are used to substantiate antecedent water levels. These antecedent water levels must be as high as the "10 percent exceedence" monthly spring high tide plus a sea level anomaly based on the maximum difference between recorded and predicted average water levels for durations of two weeks or longer for coastal locations or the 100-yr. recurrence interval high water for the Great Lakes. In a similar manner, the storm track, wind fields, effective fetch lengths, direction of approach, timing and frictional surface and bottom effects are evaluated by independent staff analysis to assure that the most critical values have been selected. Models used to estimate surge hyrographs that have not previously been reviewed and approved by the staff are verified by reproducing historical events, with any discrepancies in the model being on the conservative (i.e., high) side.

Criteria and methods of the Corps of Engineers as generally summarized in Reference 30 are used as a standard to evaluate the applicant's estimate of coincident wind-generated wave action and runup.

Criteria and methods of the Corps of Engineers and other standard techniques are used to evaluate the potential for oscillation of waves at natural periodicity.

Criteria and methods of the Corps of Engineers (Ref. 30) are used to evaluate the adequacy of protection from flooding, including the static and dynamic effects of broken, breaking, and nonbreaking waves.

Regulatory Guide 1.135 is used to determine normal water levels at plant. All water levels must be referenced to normal or mean water levels.

III. REVIEW PROCEDURES

The staff will evaluate the applicant's analysis, including all of the assumptions, techniques, and models used. If satisfied with their technical soundness and applicability to the problem, the staff's evaluation will be focused on the conservatism of parameters used by the applicant.

If not satisfied with the applicant's techniques, the staff will perform a simplified analysis of the controlling surge and seiche flooding level (coincident with wind-generated wave activity) and the resulting effects (static and dyanmic) to the safety-related facilities using simplified calculational procedures or models with demonstrably conservative coefficients and assumptions. If the applicant's estimates of critical water level are no more than 5% less conservative than the staff's estimates,* staff concurrence will be stated. If the applicant's estimates are more than 5% less conservative, the analysis is repeated using more realistic techniques. The staff will develop a position based on the analysis; resolve, if possible, differences between the applicant's and staff's surge and seiche flooding design basis; and write the SER input accordingly. The specific review procedures are described below and outlined in Figure 2.4.5.

In general, the conservatism of the applicant's estimates of flood potential from surges and seiches is judged against the criteria indicated in Subsection II above and as discussed in Regulatory Guide 1.59. If the site is not near a large body of water the staff findings may be prepared a priori. Methods of the Corps of Engineers and National Oceanic and Atmospheric Administration (NOAA) (HUR 7-97 and amendments) are used to develop the critical Probable Maximum Hurricane (PMH) parameters for the site. The Corps of Engineers model SURGE (or other verified models) may be used to estimate the maximum surge stillwater elevations at coastal sites. Coincident wind-generated waves and runup are estimated from publications by the Corps of Engineers (Ref. 30). Reports of NOAA and the Corps of Engineers are used to estimate probable maximum wind fields over the Great Lakes. Models such as Platzmann's, or other verified models, may be used to estimate the maximum surge or seiche stillwater elevation for Great Lakes sites; coincident wind-generated waves and runup are estimated as above.

Two-dimensional models (References 15, 26, and 42) include seiching effects. Seiching potential is evaluated using one-dimensional models by comparing the natural period of

*Based on the difference between normal water levels and the flood event.

oscillation (resonance) of the water body with the estimated meteorologically-induced wave periods. Resonance of a water body may be calculated by the methods presented in Ref. 30 or standard texts. Generally, a demonstration that the water body cannot generate or sustain waves of the required period for resonance is satisfactory to discuss the possibility of damaging seiching. Similarly, seismically induced seiching is precluded if the natural period of oscillation of the water body is dissimilar from the period of precluded seismic excitation. Coordination with GSB to determine the controlling seismic parameters may be required. If resonance is possible, the maximum seiche must be considered in the selection of the critical flood design bases.

The above reviews are performed only when applicable to the site or site region. Some items of review may be done on a generic basis.

IV. EVALUATION FINDINGS

For construction permit (CP) reviews the findings will summarize the applicant's and staff's estimates of critical water level (including wind-generated wave levels) at the site. If the estimates meet the criteria (described in Subsection II), staff concurrence will be stated. If the applicant's estimates do not meet the criteria in Subsection II, and the proposed plant may be adversely affected, a statement requiring use of the staff's estimates for the design basis will be made. If the flood conditions do not constitute a design basis, the statement will so indicate.

For operating license (OL) reviews of plants which have received detailed reviews during the CP review, the CP conclusions will be referenced. However, a review will be made to assure that protection against the design-basis water level conditions established in the CP review has been properly implemented. In addition, a review of surge and seiche history since the CP review will be made. Any new information or improvements in predictive models will be noted. If no detailed CP review was undertaken, this fact will be indicated in the OL findings.

If Regulatory Guide 1.59, Position 2, is elected by the applicant for protection, a statement describing lesser design bases will be included in the findings with the staff conclusion of adequacy.

A sample statement for an OL review follows:

> "The design basis hurricane-induced high and low stillwater levels were established during the CP review at elevations 22.0 feet MSL and -7.5 feet MSL, respectively. These levels are based upon the estimated water levels, exclusive of wave action, that would occur during passages of a Probable Maximum Hurricane (PMH) to the south and north, respectively, of the plant. At the request of the staff, the applicant analyzed the wave conditions on safety-related facilities that could accompany the 22 foot MSL surge level. The results of these analyses indicate the most severe wave action would be restricted to the canal, and that high ground levels would limit wave heights in the vicinity of exposed safety-related buildings, except the service water

intake, to 1.6 feet. For the intake, the applicant has estimated waves 3 feet high. The resulting wave runup levels were estimated to reach a maximum elevation of 28.3 feet MSL on the intake, and 25.6 feet MSL on other exposed buildings."

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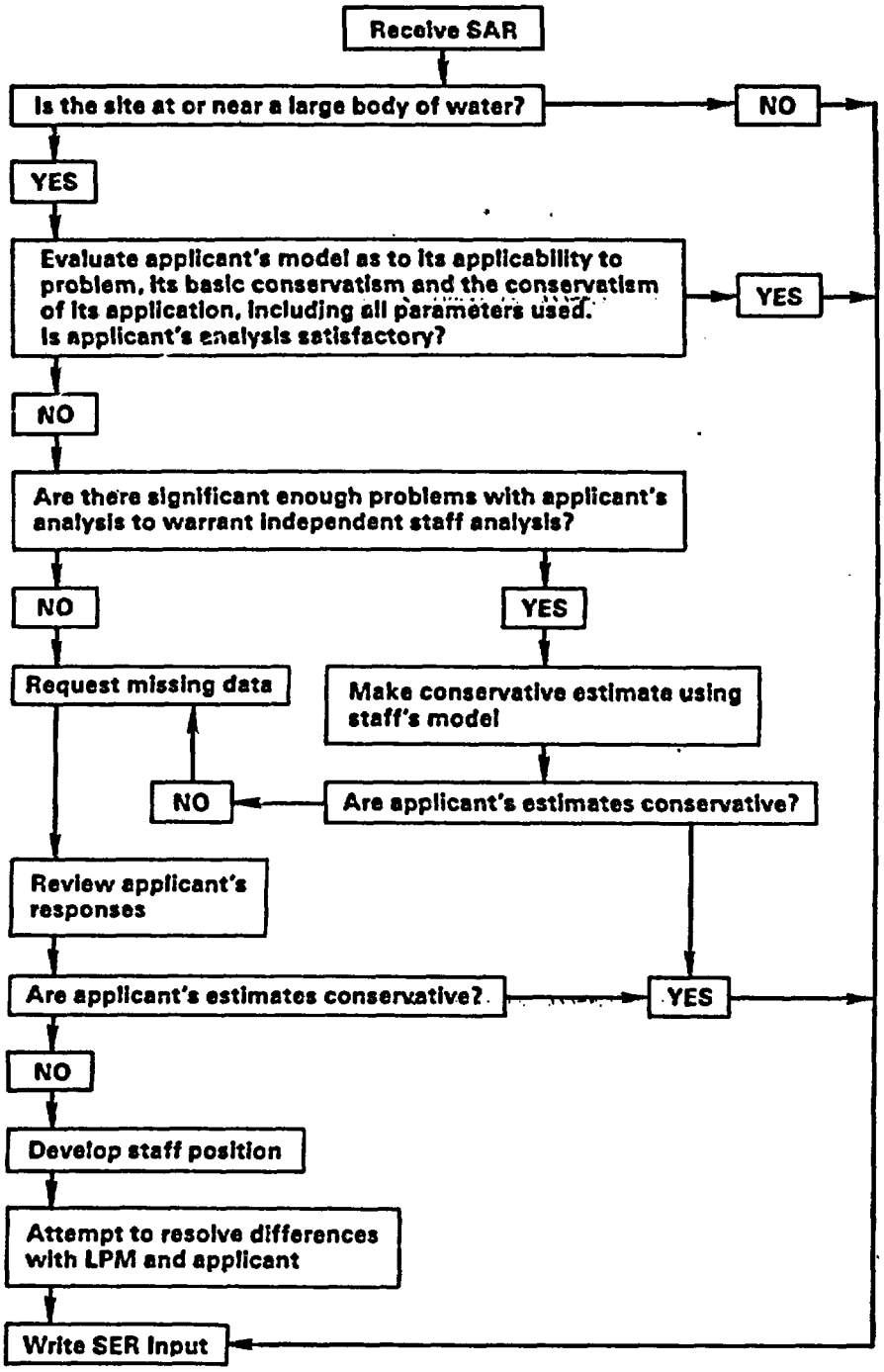
Figure 2.4.5-1
STANDARD REVIEW PLAN SECTION 2.4.5



U.S. NUCLEAR REGULATORY COMMISSION
STANDARD REVIEW PLAN
OFFICE OF NUCLEAR REACTOR REGULATION

SECTION 2.4.6 PROBABLE MAXIMUM TSUNAMI FLOODING

REVIEW RESPONSIBILITIES

Primary - Hydrology-Meteorology Branch (HMB)

Secondary - Geosciences Branch (GB)

I. AREAS OF REVIEW

The geohydrological design basis of the plant (discussed in Regulatory Guide 1.59) is developed in this section of the Safety Analysis Report (SAR) to determine the extent of plant protection required for tsunami flooding and drawdown (outlined in Regulatory Guide 1.102). The areas of review include the hydrologic characteristics of the maximum locally and distantly generated tsunami and the techniques, methodologies and parameters, including the geoseismic parameters of the generators, used in the determination of the design basis tsunami.

Geologic and seismic characteristics reviewed include earthquake magnitude, focal depth, source dimensions, fault orientation and vertical displacement. Hydrologic analysis techniques, including tsunami formation, propagation and shoaling models, and coincident water levels, including astronomical tide, storm surges and waves, are reviewed.

II. ACCEPTANCE CRITERIA

If it has been determined that tsunami estimates are necessary to identify flood or low water design bases, the analysis will be considered complete if the following areas are addressed and can be independently and comparably evaluated from the applicant's submission:

1. All potential distant and local tsunami generators, including volcanos and areas of potential landslides, are investigated and the most critical ones are selected.

2. Conservative values of seismic characteristics (source dimensions, fault orientation and vertical displacement) for the tsunami generators selected are used in the analysis.

3. All models used in the analysis are verified or have been previously approved by the staff.

4. Bathymetric data are provided (or are readily obtainable).

5. Detailed descriptions of shoreline protection and safety-related facilities are provided for wave runup and drawdown estimates.

6. Ambient water levels, including tides, sea level anomalies, and wind waves, are estimated using NOAA and Corps of Engineers publications as described below.

7. If Regulatory Guide 1.59, Position 2, is adopted by the applicant, the design basis for tsunami protection of all safety-related facilities identified in Regulatory Guide 1.29 must be shown to be adequate in terms of the time required for implementation of any emergency procedures.

The applicant's estimates of tsunami runup and drawdown levels are acceptable if the estimates are no more than 5% less conservative than the staff's estimates. If the applicant's estimates are more than 5% less conservative (based on the difference between normal water levels and the maximum runup or drawdown levels) than the staff's, the applicant should fully document and justify its estimates or accept the staff's estimates.

This section of the SAR will also be acceptable if it states the criteria used to determine that tsunami flooding estimates are not necessary to identify the flood design basis (e.g., the site is not near a large body of water).

III. REVIEW PROCEDURES

The review procedures are outlined in Figure 2.4.6. The references used are general geophysical, seismological, and hydrodynamic publications, such as published data by the National Oceanic and Atmospheric Administration (NOAA), and wave propagation models such as those developed by NOAA, WES and Tetra Tech.

Section 2.4.6 of the applicant's SAR is reviewed to identify any missing data, information or analyses necessary for the staff's evaluation of potential tsunami flooding. This section is evaluated when the applicant has responded to all of the additional information requested. If the site is not near a large body of water with potential tsunami generators, the staff findings may be prepared *a priori*.

The staff (with input from GB) will review the potential tsunami sources analyzed by the applicant to assure that all locations capable of generating a tsunami of significant magnitude at the site have been considered. The GB staff will evaluate the geoseismic parameters of the tsunami generators, including fault location and orientation, and amplitude and areal extent of vertical displacement, to assure that conservative values have been chosen.

An independent staff analysis, using one of the models listed in the references, may be performed. Staff estimates of tsunami levels are compared with the applicant's. The

applicant must justify, to the staff's satisfaction, tsunami levels more than 5% less conservative than the staff's.

As an alternative, the staff may perform an independent evaluation of the applicant's model and its utilization. The model's theoretical basis, its inherent conservatism and applicability to the problem, will be evaluated (this can be done on a generic basis). The conservatism of the models' use, including the conservatism of all input parameters, will be evaluated.

Coincident ambient tide and wave conditions will be evaluated to assure that they are of at least annual severity. Data from publications of NOAA, the Corps of Engineers, and other sources are used to substantiate these conditions chosen.

Criteria and methods of the Corps of Engineers as generally summarized in Reference 12 are used as a standard to evaluate the applicant's estimate of coincident wind-generated wave action and runup.

Criteria and methods of the Corps of Engineers and other standard techniques are used to evaluate the potential for oscillation of waves at natural periodicity.

Criteria and methods of the Corps of Engineers (Ref. 12) are used to evaluate the adequacy of protection from flooding, including the static and dynamic effects of broken, breaking and nonbreaking coincident waves.

IV. EVALUATION FINDINGS

For construction permit (CP) reviews the findings will consist of a statement summarizing estimates of the maximum and minimum tsunami water levels, static and dynamic effects of wave action, and a statement of acceptability of the tsunami-induced design basis. If the tsunami conditions do not constitute a design basis, the findings will so indicate. For operating license (OL) reviews, the findings will consist of the evaluation of any new information on tsunami potential, improvements in predictive models acceptability of specific design bases, and the acceptability of design provisions.

A sample statement for an CP review follows:

> "Analyses of tsunamic effects from local and distant generators were performed by the applicant at the staff's direction. The design tsunami results from a magnitude 8.7 earthquake in the Aleutian Trench. A finite difference numerical model was used to analyze tsunami generation and propagation to the continental shelf. Results of this computation were used in a nearshore model to calculate tsunami runup and drawdown. Including the effects of high and low tides of annual occurrence, the maximum tsunami runup and drawdown are estimated as +24.5 feet MLLW and -13.4 feet MLLW, respectively. Wind waves of annual severity were assumed coincident with the tsunami. The maximum wave runup, at the intake pumphouse, was

estimated as +31.2 feet MLLW. The maximum drawdown, at the location of the offshore intake, was estimated as -21.3 feet MLLW."

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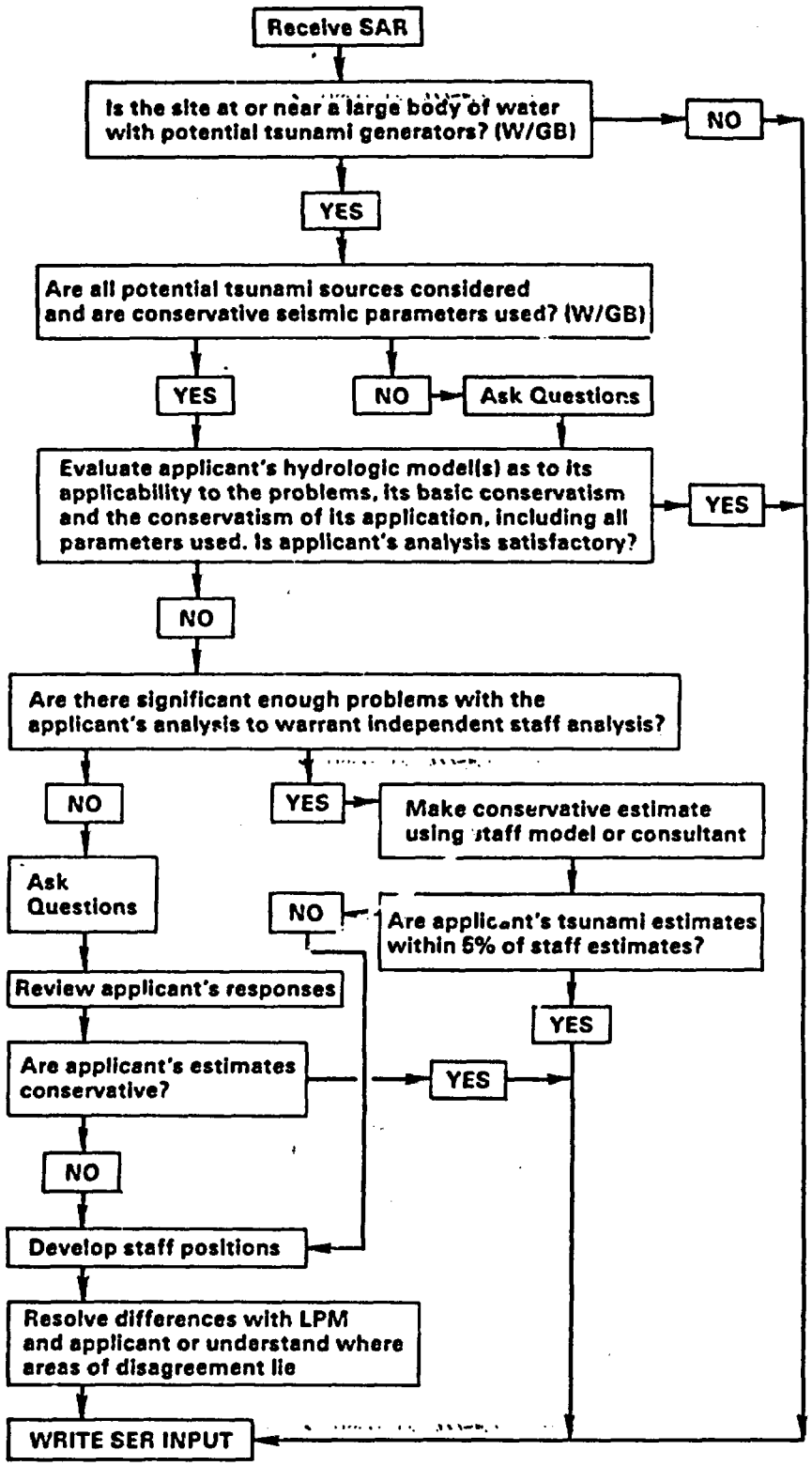
Figure 2.4.6
REVIEW PROCEDURES



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STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

SECTION 2.4.7 ICE EFFECTS

Primary - Hydrology & Meteorology Branch (HMB)

Secondary - None

I. AREAS OF REVIEW

The hydrometeorologic design basis is developed in this section of the safety analysis report (SAR) to assure that safety-related facilities and water supply are not affected by ice flooding or blockage. The areas of review include:

1. The regional history and types of historical ice accumulations (i.e., ice jams, wind-driven ice ridges, floes, etc.).
2. The potential for ice-produced forces on, or blockage of, safety-related facilities.
3. The potential effects of ice-induced high or low flow levels on safety-related facilities and water supplies.

II. ACCEPTANCE CRITERIA

Publications of the National Oceanic and Atmospheric Administration (NOAA), the United States Geologic Survey (USGS), the Corps of Engineers, and other sources are used to identify the history and potential for ice formation in the region. Historical maximum depths of icing should be noted, as well as mass and velocity of any large floating ice bodies. The phrase "historical low water ice affected," or similar phrases in streamflow records (USGS and state publications) will alert the reviewer to the potential for ice effects. The following items must be considered and evaluated, if found necessary, in the design of protection of safety-related facilities and water supplies.

1. The regional ice and ice jam formation history must be described to enable an independent determination of the need for including ice effects in the design basis.
2. If icing has not been severe, based on regional icing history, design considerations must be presented (e.g., return of a portion of low-grade heat to the intake) to assure that icing or ice blockage of intake screens and pumps will not adversely affect safety-related facilities and water supplies.
3. If the potential for icing is severe, based on regional icing history, it must be shown that water supplies capable of meeting safety-related requirements are available from under the ice formations postulated and that safety-related equipment is protected from icing as in 2., above. If not, it must be demonstrated that alternate sources of water are available, that they are protected from freezing, and that the alternate source is capable of meeting safety-related requirements in such situations. Ice loading must have been included in the structural design basis, if severe icing is possible.

4. If floating ice is prevalent, based on regional icing history, consideration of impact forces on the safety-related intakes must be a consideration in the design basis. The dynamic loading caused by floating ice must be included in the structural design basis.
5. If ice blockage of the river or estuary is possible, it must be demonstrated that the resulting wacer level in the vicinity of the site has been considered in establishing the flood and water supply design bases. If this water level would adversely affect the intake structure, or other safety-related facilities, it must be demonstrated that an alternate safety-related water supply will not also be adversely affected.

The applicant's estimates of potential ice flooding or low flows are acceptable if the estimates are no more than 5% less conservative than the staff's estimates. If the applicant's estimates are more than 5% less conservative than the staff's,* the applicant should fully document and justify its estimates or accept the staff's estimates and redesign applicable flood protection. The suggested criteria of Regulatory Guide 1.27 applies when the water supply comprises part of the ultimate heat sink.

Appropriate sections of the following documents are used by the staff to determine the acceptability of the applicant's data and analyses. Regulatory Guide 1.59 provides guidance for developing the hydrometeorological design basis. Regulatory Guide 1.135 describes acceptable methods for determining normal water levels. (Ali estimated water levels should be referenced to mean or normal water levels.) Regulatory Guide 1.29 identifies the safety-related structures, systems, and components, and Regulatory Guide 1.102 describes acceptable flood protection to prevent the safety-related facilities from being adversely affected. Regulatory Guide 1.27 describes the ultimate heat sink capabilities which apply.

III. REVIEW PROCEDURES

Applicable literature describing historical occurrences of icing in the region is reviewed to determine if icing protection should be considered in the design of safety-related facilities. If considered necessary, the most likely types of icing conditions (floating ice, river blockage by ice buildup, frazil, etc.) are listed, and the potential impact on plant design of each type is identified. Criteria of the Corps of Engineers and others provide a means of assessing icing impact and methods of mitigating adverse effects. For each type of icing condition, preliminary independent estimates of the "worst case" will be made by either conservative statistical or deterministic techniques.

If the applicant's estimates of ice effects are comparable to the staff's preliminary analysis, the staff will concur with the applicant's estimates. If the preliminary analysis indicates the applicant's estimates of ice effects are not comparable to the

*Based on the difference between normal water levels and the flood event or low water.

staff's estimates, the staff's analysis will be repeated using more realistic techniques. If there is evidence of potential structural effects, the Structural Engineering Branch (SEB) will be requested by HMB to ascertain whether these effects are properly considered in the structural design basis for the plant; similarly, if there is evidence of potential mechanical effects, the Mechanical Engineering Branch (MEB) and the Auxiliary Systems Branch (ASB) will be requested by HMB to ascertain whether these effects are properly considered in the mechanical design basis for the plant. The staff will develop a position based on the analysis; resolve, if possible, differences between the applicant's and staff's estimates of ice effects; and write the SER input accordingly.

The above reviews are performed only when applicable to the site or site regions. Some items of review may be done on a generic basis.

IV. EVALUATION FINDINGS

For construction permit (CP) reviews, the findings will summarize the applicant's and staff's estimates of the potential for ice flooding, and if applicable, the minimum low water levels (from upstream ice blockage). If the applicant's estimates are within acceptable margins (described in Acceptance Criteria), staff concurrence with the applicant's estimate will be stated. If the applicant's estimates are not within acceptable margins or, if the staff predicts potential blockage of the intake, or if the proposed plant may be adversely affected, a statement of the staff bases will be made. If the icing conditions do not constitute a design basis, the findings will so indicate.

For operating license (OL) reviews of plants for which detailed icing reviews were made at the CP stage, the CP conclusions will be referenced. However, a review will be made to assure that the design basis established in the CP review has been implemented properly. In addition, a review of icing records since the CP review will be made. If no CP review was undertaken (of the scope indicated), this fact will be noted in the OL findings.

A sample CP statement follows:

"Ice Flooding, which is common on the A River at the makeup intake structure, could only affect the river intake structure which would not result in any adverse effects to the plant's safety-related facilities. The applicant states that ice flooding may possibly raise the water surface near the A River intake to a maximum elevation of about 555 feet MSL. The applicant further states that ice and ice flooding on the A River tributaries outside the cooling lake will not affect the plant facilities. The major tributary nearest the plant is the B Creek with the closest point located about one mile to the southeast of the site. The applicant concludes that, because of the distance from the proposed site and the wide floodplain of the river, there will be no adverse effects at the plant site due to ice in the river and consequent flooding. We concur with this conclusion.

"The safety-related pumps from the cooling lake are to be protected from ice blockage by means of traveling screens, stop logs, and trash racks located at the front of the lake screenhouse. In addition, the applicant proposes a warm-up line from the circulating water discharge which will keep the inlet water temperature 40°F during winter operation.

An essential cooling water screen bypass pipe is also available. We concur with the applicant that icing or ice flooding should not adversely affect the plant's safety-related facilities."

V. REFERENCES

1. E. Brown and G. C. Clark, "Ice Thrust in Connection with Hydro-Electric Design," Engineering Journal, pp. 18-25, 1932.

2. V. T. Chow (ed.), "Handbook of Applied Hydrology," McGraw-Hill Book Company, New York (1964).

3. O. Devik, "Freezing Water and Supercooling," Jour. of Glaciology, Vol. 1, No. 6, pp. 307-309 (1949).

4. N. E. Dorsey, "Properties of Ordinary Water Substances," Reinhold Publishing Company, New York (1940).

5. H. T. Mautis (ed), "Review of Properties of Snow and Ice," Report 4, Corps of Engineers, Snow, Ice and Permafrost Research Establishment (1951).

6. E. Rose, "Thrust Exerted by Expanding Ice Sheet," Trans. Am. Soc. Civil Engineers, Vol. 112, pp. 871-900 (1947).

7. J. T. Wilson, "Coupling Between Moving Loads and Flexural Waves in Floating Ice Sheets," Report No. 34, Corps of Engineers, Snow, Ice, and Permafrost Research Establishment (1955).

8. J. T. Wilson, J. H. Zumberge, and E. W. Marshall, "A Study of Ice on an Inland Lake," Report No. 5, Corps of Engineers, Snow, Ice and Permafrost Research Establishment (1954).

9. "River Ice Jams - A Literature Review," Engineer Technical Letter No. 1110-2-58, Corps of Engineers (1969).

10. "Design of Small Dams," Bureau of Reclamation, U. S. Department of the Interior (1973).

11. J. H. Zumberge and J. T. Wilson, "Quantitative Studies of Thermal Expansion and Contraction of Lake Ice," Jour. of Geophysical Research, Vol. 61, pp. 374-383 (1953).

12. "Surface Water Supply of the United States," U.S. Geological Survey, surface water supply papers as applicable to the plant region.

13. Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants."

14. Regulatory Guide 1.59, "Design Basis Floods for Nuclear Power Plants."

15. ANSI N170, "Standards for Determining Design Basis Flooding at Power Reactor Sites."

16. Regulatory Guide 1.29, "Sesimic Design Classification."

17. Regulatory Guide 1.27, "Ultimate Heat Sink for Nuclear Power Plants."

18. Regulatory Guide 1.102, "Flood Protection for Nuclear Power Plants."

19. Regulatory Guide 1.135, "Normal Water Level and Discharge at Nuclear Power Plants."

20. G. D. Ashton, et al., "Icebreaking by Tow on the Mississippi River," SR 192, CRREL, Hanover, New Hampshire, August 1973.

21. Roscoe E. Perham, "Forces Generated in Ice Boom Structures," SR 200, CRREL, Hanover, New Hampshire, January 1974.

22. George D. Ashton, "Air Bubbler Systems to Suppress Ice," SR 210, CRREL, Hanover, New Hampshire, September 1974.

23. Darryl J. Calkins & George D. Ashton, "Arching of Fragmented Ice Covers," SR 222, CRREL, Hanover New Hampshire, April 1975.

24. W. H. Brierley, et al., "Lock Wall Deicing with Water Jets: Field Tests at Ship Locks in Montreal, Canada, and Sault Sainte Marie, Michigan," SR 239, CRREL, Hanover, New Hampshire, December 1975.

25. Bernard Michel, "Ice Pressure on Engineering Structures," CRREL, Hanover, New Hampshire, June 1970.

26. F. D. Haynes, et al., "Ice Force Measurements on the Pembina River, Alberta, Canada," SR 269, CRREL, Hanover, New Hampshire, October 1975.

27. K. L. Carey, et al., "Ice Engineering for Civil Works, Baseline Study," CRREL, Hanover, New Hampshire, August 1973.

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SECTION 2.4.8 COOLING WATER CANALS AND RESERVOIRS

REVIEW RESPONSIBILITY

Primary - Hydrology-Meteorology Branch (HMB)

Secondary None

I. AREAS OF REVIEW

This section of the applicant's safety analysis report (SAR) presents the basis for the hydraulic design of canals and reservoirs used to transport and impound plant cooling water. In addition, the hydraulic design basis for protection of structures (e.g., riprap) is reviewed. For canals, the areas of review include the design basis for capacity, protection against wind waves, erosion, sedimentation buildup, and freeboard, and (where applicable) the ability to withstand a Probable Maximum Flood (PMF), surges, etc. For reservoirs, the areas of review include the design basis for capacity, Probable Maximum Flood design basis, wind wave and runup protection, discharge facilities (low level outlet, spillway, etc.), outlet protection, freeboard, and erosion and sedimentation processes.

II. ACCEPTANCE CRITERIA

The acceptance criteria for the protection of cooling water canals from wind waves, PMF, surges, etc., are the same as those outlined in SRP Sections 2.4.3, 2.4.4, 2.4.5, and 2.4.7. The criterion for canal capacity is that the canal must be capable of transmitting to the plant sufficient water to meet all safety requirements during postulated extreme hydrologic events (i.e., both floods and droughts). Where canals comprise a part of the ultimate heat sink, Regulatory Guide 1.27 is used as a basis for the adequacy of design criteria and provisions. The design basis for canal capacity is analyzed, in any case, to assure that safety-related water requirements can be supplied under all postulated extreme hydrologic events, or that alternative conveyance systems are designed to be available during the postulated conditions.

The acceptance criteria for the hydraulic design of reservoirs are as follows:

1. For protection of structures against wind waves, input from SAR Sections 2.4.3, 2.4.4, 2.4.5, 2.4.6, and 2.4.7 for PMF, Probable Maximum Hurricane (PMH), surge, seiche, or tsunami levels and coincident waves and runup must be considered to establish the maximum and minimum water level and wave conditions. Also, normal pool level and coincident probable maximum wind-wave activity must be considered.

Criteria and methods as reported in Corps of Engineers publications are generally acceptable for design of embankment protection (riprap, grass, soil cement, tetrapods, dolosse, etc.) and freeboard.

2. For emergency storage evacuation, the spillways are acceptable if they can safely pass the PMF, or controlling design basis flood, without endangering safety-related facilities or increasing the hazard to downstream residents. In addition, a low level outlet may be provided to evacuate the storage in an emergency.

3. For reservoir routings, the maximum still water level is acceptable if the spillway design flood has been routed through the spillway (and outlet works, if applicable) using standard methods as suggested by the Corps of Engineers, USBR, and others, and a minimum of three feet of freeboard (including waves) is available. However, the antecedent reservoir level to be used with the flood routing must be at least as high as that suggested by Regulatory Guide 1.59, "Design Basis Floods for Nuclear Power Plants."

   The probable minimum low water level is acceptable if the flow during the design basis drought (from SAR Section 2.4.11) has been routed through the reservoir[1/] using standard methods as suggested by the Corps of Engineers, USBR, and others. The antecedent reservoir level for this routing, if reservoir storage is the sole water supply source, must be the lowest reasonably possible, considering regional conditions at the beginning of the drought and water demands, including plant requirements. In no case should the antecedent reservoir level be greater than the established normal operating level.

4. Where not covered above, the hydraulic design for the low level outlets, conduits, spillways (gated and ungated, regulating and emergency), and embankment protection is required where the failure of such items could constitute a threat to essential plant facilities or to safety-related water supplies. The design is acceptable if standard techniques have been used as suggested by the Corps of Engineers, USBR, and others such that the minimum design water level for safety-related pumps would not be violated.

5. If reservoirs comprise a part of the ultimate heat sink, Regulatory Guide 1.27 is used as a basis for judging the adequacy of the design criteria and provisions.

Applicable portions of the following documents are to be used to determine the acceptability of the applicant's data and analyses. Regulatory Guide 1.59 discusses the design basis for flooding. Regulatory Guide 1.135 describes the methods used to determine normal water levels. (All estimated water levels should be referenced to mean or

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1/ For those plants proposing multiple reservoirs for water supply, analyses must be provided to assure that storage allocated for safety-related water supply in alternate reservoirs will be available during postulated drought conditions. Additionally, evidence of the right to use the water consumptively must be documented.

normal water levels.) Regulatory Guide 1.29 identifies the safety-related structures, systems, and components and Regulatory Guide 1.102 describes acceptable flood protection to prevent the safety-related facilities from being adversely affected. Regulatory Guide 1.27 describes design criteria and provisions which the ultimate heat sink must meet. Publications of the Corps of Engineers and USBR provide guidance for canal and reservoir design criteria. SRP Sections 2.4.3 through 2.4.7 provide basic data for analyzing the hydraulic design of canals and reservoirs during high flow levels.

III. REVIEW PROCEDURES

The conservatism of the applicant's design basis is judged against the criteria indicated above. SAR Sections 2.4.3, 2.4.4, 2.4.5, 2.4.6, and 2.4.7 should provide the basic data for analyzing the high flow hydraulic design basis of the facility. The applicant's hydraulic design basis is judged against standard design practices discussed in Corps of Engineers (Waterway Experiment Station) or USBR publications. Low flow input data are taken from SAR Section 2.4.11. The review procedures consists of independently "designing" (hydrologically and hydraulically) the applicant's facilities (e.g., dams, canals, spillways) using the above methods and comparing the resultant "design" with the applicant's. Wave and runup protection is evaluated using the methods of References 20 and 21. Subsequently, the staff will develop a position based on the analyses; resolve, if possible, differences between the applicant's and staff's design bases; and prepare the SER input accordingly.

The above reviews are performed only when applicable to the site or site region. Some items of review may be done on a generic basis.

IV. EVALUATION FINDINGS

For construction permit (CP) reviews the findings will consist of a statement of the applicant and staff estimates of the type and adequacy of required structure protection and the hydraulic design basis of canals and reservoirs. Because of the advanced design required for the CP and where the design has received a detailed review at the CP stage, the operating license (OL) findings will only be an acknowledgement of any changes and a statement of acceptability. If a design or flooding potential was not reviewed in detail at the CP stage, it will be done at the OL stage.

Sample statements from CP reviews follow:

"Although postulated flood waters are not expected to reach plant grade, protection of the essential auxiliary and main dams against their respective probable maximum floods is to be provided by riprap protection of exposed embankment surfaces (including areas in the plant site vicinity along the auxiliary reservoir intake channel) and concrete overflow spillways. At our request, the applicant provided design bases for riprap protection and the hydraulic design criteria for the two spillways. The applicant at our request, in Amendment No. 31 to the PSAR, provided criteria for the windwave riprap protection based upon an empirical relationship for the median size stone to be placed in a blanket approximately two feet thick

and indicated its specifications for stone gradation. A filter blanket approximately one foot thick is to be placed under the riprap to prevent piping (removal of smaller material) through the larger armor riprap cover layer. Criteria were provided for the filter gradation, angularity, durability of the riprap, and placement which provides assurance that erosive failure of safety-related embankments should not occur. An armor protection layer also is provided. We find these riprap design bases and spillway hydraulic design criteria to be acceptable.

"Storage in the three reservoir system, runoff from the contributing drainage area, and diversion of A River flows to the main reservoir during periods of low runoff and high reservoir evaporation will constitute the water supply for the four-unit once-through cooling systems.

"The applicant has provided analyses of the capability of the main and auxiliary reservoirs to supply water during emergency conditions requiring emergency shutdown and cooldown of one unit and the simultaneous normal shutdown and cooldown of the remaining three units as suggested in Regulatory Guide 1.27 - Ultimate Heat Sink. In addition, the applicant has provided analyses of the operation of the plant and the main reservoir under historical and a synthesized 100-year drought condition. For the shutdown conditions the applicant has demonstrated that the two reservoir - A River diversion system constituting the ultimate heat sink would have a water supply available in excess of thirty days in the auxiliary reservoir if water were not available from the main reservoir - auxiliary reservoir - A River diversion facilities. The operation of the sink as a whole will require that the auxiliary reservoir be kept at its normal operating level of elevation 250 feet MSL at all times by pumping water from the main reservoir to make up for water lost to normal evaporation.

"For the analyses of evaporation under normal plant operation during periods of assumed reoccurrence of historical droughts, the applicant has used historical flow records for the A River and synthesized flow data for the drainage area contiguous to the reservoir system. For the analysis of evaporation during a more extreme drought than has occurred historically, the applicant has synthesized flows from both the A River and the contiguous drainage areas for what is called a 100-year frequency drought. The staff, in consonance with our consultant (the U.S. Geological Survey), independently developed and analyzed synthesized flows from both drainage areas. We concluded that it is likely that flows from both areas could be substantially less than estimated by the applicant. The applicant is installing a streamflow gage near the plant to determine runoff characteristics from the contiguous drainage which should allow more accurate analysis of the operating capability of the reservoir system prior to plant operation. Inaccuracies in estimation of runoff are considered to be only indirectly safety related since an adequate shutdown and cooldown water supply will be available in the auxiliary reservoir should evaporation and the lack of runoff prevent replenishment of main reservoir storage above the minimum operating level of elevation 244 feet MSL."

V. REFERENCES

1. Am. Soc. Civil Engineers, "Hydraulic Models," Manual of Engineering Practice No. 25 (1963).

2. Leo R. Beard, "Flood Control Operation of Reservoirs," Jour. Hydraulic Division, Proc. Am. Soc. Civil Engineers, Vol. 88, No. HY1, pp. 1-25 (1963).

3. Leo R. Beard, "Methods for Determination of Safe Yield and Compensation Water from Storage," Seventh International Water Supply Conference, Barcelona, Spain (1966).

4. E. F. Brater and H. W. King, "Handbook of Hydraulics for the Solution of Hydrostatic and Fluid-Flow Problems," McGraw-Hill Book Company, New York (1963).

5. V. T. Crow (ed), "Handbook of Applied Hydrology," McGraw-Hill Book Company, New York (1964).

6. V. T. Chow (ed), "Open Channel Hydraulics," McGraw-Hill Book Company, New York (1959).

7. C. V. Davis (ed), "Handbook of Applied Hydraulics," McGraw-Hill Book Company, New York (1964).

8. G. W. Fair, J. C. Geyer, and D. A. Okien, "Water Supply and Waste Water Removal," John Wiley & Son, Inc., New York (1966).

9. G. A. Hathaway, "Determination of Spillway Requirements for High Dams," Proc. Fourth International Conference on Large Dams, New Delhi, Vol. 2, pp. 301-347 (1951).

10. H. W. King and E. F. Brater, "Handbook of Hydraulics," McGraw-Hill Book Company, New York (1963).

11. R. K. Linsley and J. B. Franzini, "Water-Resources Engineering," McGraw-Hill Book Company, New York (1964).

12. H. Rouse (ed), "Engineering Hydraulics," John Wiley & Son, Inc., New York (1951).

13. "Hydraulic Design Criteria," prepared by the Corps of Engineers Waterways Experiment Station, loose-leaf by serials.

14. "Hydraulic Design of Flood Control Channels," Engineer Manual 1110-2-1601, Corps of Engineers, July 1970.

15. "Hydraulic Design of Spillways," Engineer Manual 1110-2-1603, Corps of Engineers, March 1965.

16. "Hydraulic Tables," Corps of Engineers (1944).

17. "Hydrologic Engineering Methods for Water Resources Development," Volumes 1 through 12, Corps of Engineers Hydrologic Engineering Center, Davis, California (1971).

18. "Reservoir Regulation," Engineer Manual 1110-2-3600, Corps of Engineers, May 1959.

19. "Reservoir Storage-Yield Procedures," Corps of Engineers Hydrologic Engineering Center (1967).

20. "Shore Protection Manual," Technical Report No. 4, Third Edition, Corps of Engineers Coastal Engineering Research Center (1966).

21. "Shore Protection Manual," Corps of Engineers Coastal Engineering Research Center (1973).

22. Hydraulic Model Studies of the Corps of Engineers Waterways Experiment Station.[2/]

23. "Design of Small Dams," Second Edition, Bureau of Reclamation, U.S. Dept. of the Interior (1973).

24. "Design Standards No. 3, Canals and Related Structures," Chapter 2 of "General Design Information for Structures," Bureau of Reclamation, U.S. Dept. of the Interior, April 1962.

25. "Hydraulic Model Studies"[2/] of the Bureau of Reclamation, U.S. Dept. of the Interior.

26. "Hydraulic Model Studies"[2/] of the Dept. of Water Resources, State of California.

27. Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants."

28. Regulatory Guide 1.27, "Ultimate Heat Sink for Nuclear Power Plants."

29. Regulatory Guide 1.59, "Flood Design Basis for Nuclear Power Plants."

30. ANSI N170, "Standards for Determining Design Basis Flooding at Power Reactor Sites"

31. Regulatory Guide 1.29, "Seismic Design Classification."

[2/] A series of such studies exists in the literature too numerous to mention here. In addition to the three specifically cited series, studies by others will be utilized on an "as available" basis.

32. Regulatory Guide 1.102, "Flood Protection for Nuclear Power Plants."

33. Regulatory Guide 1.135, "Normal Water Level and Discharge at Nuclear Power Plants."

34. ETL 1110-2-221, "Wave Runup and Wind Setup on Reservoir Embankments," Department of the Army, Corps of Engineers, November 29, 1976.

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SECTION 2.4.9 CHANNEL DIVERSIONS

REVIEW RESPONSIBILITIES

Primary - Hydrology-Meteorology Branch (HMB)

Secondary - None

I. AREAS OF REVIEW

In this section of the applicant's safety analysis report (SAR) the geohydrologic design basis is developed to assure that the plant and essential water supplies will not be adversely affected by natural stream channel diversion, or that in such an event, alternate water supplies are available to safety-related equipment.

The review includes:

1. Historical channel diversions, including cutoffs and subsidence.
2. Regional topographic evidence which suggests that future channel diversion may or may not occur (used in conjunction with evidence of historical diversions).
3. Alternate water sources and operating procedures (coordinate review with that of SAR Section 2.4.11.6).

II. ACCEPTANCE CRITERIA

The analyses will be considered acceptable if at least the following are addressed:

1. A description of the applicability (potential adverse effects) of stream channel diversions.
2. Historical diversions and realignments.
3. The topography and geology of the basin and its applicability to natural stream channel diversions.
4. If applicable, the safety consequences of diversion and the potential for high or low water levels caused by upstream or downstream diversion adversely to affect safety-related facilities, water supply or ultimate heat sink.

III. REVIEW PROCEDURES

Site-specific publications and maps are reviewed to identify historical channel diversions and evaluate (by independent conservative calculations and professional judgment) the

potential for future diversions. Where an alternate safety-related cooling water supply is provided, the criteria for SAR Section 2.4.11.6 apply and are checked for consistency.

The above reviews are performed only when applicable to the site or site region. Some items of review may be done on a generic basis.

IV. EVALUATION FINDINGS

For construction permit (CP) reviews and when applicable, findings will consist of a brief general description of historical channel diversions. If the staff concurs with the applicant that channel diversion is unlikely or that the plant is protected and alternate essential water supplies meet the criteria of Regulatory Guide 1.27, the findings will so indicate. If the staff evaluation does not support the applicant's contention of channel stability, an alternate source of water may be required.

For operating license reviews, findings will consist of the same material, updated as required to reflect new information available since preparation of the CP findings.

A sample CP-stage statement follows:

> "Diversions of the A River are well-documented in historical and topographic data. Oxbow lakes, low-lying swamps, and bars, and chutes provide eloquent evidence of historical diversion. Others are planning further bank protection measures, additional to the existing levee system, in the vicinity of the plant intake structure. However, the diversion of the main channel by degradation/aggradation within the confines of the levee system, or by breaching the west levee during major floods, cannot be discounted. The ultimate heat sink (as discussed in Section 2.4.11) is not directly dependent on the river intake. We conclude that channel diversions present no safety-related hazard to the plant."

V. REFERENCES

No specific publications can be cited for general use; however, site-specific publications and maps can be obtained from the United States Geologic Survey, Soil Conservation Service, National Oceanic and Atmospheric Administration, Corps of Engineers, and state and other agencies and organizations, to identify historical and potential future channel diversions.

1. Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants."

2. Regulatory Guide 1.27, "Ultimate Heat Sink for Nuclear Power Plants."

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SECTION 2.4.10 FLOODING PROTECTION REQUIREMENTS

REVIEW RESPONSIBILITY

Primary - Hydrology-Meteorology Branch (HMB)

Secondary - None

I. AREAS OF REVIEW

The locations and elevations of safety-related facilities and of structures and components required for protection of safety-related facilities are compared with the estimated static and dynamic effects of design basis flood conditions identified in safety analysis report (SAR) Section 2.4.2.2, to determine whether flood effects need be considered in plant design or emergency procedures.

If flood protection is required, the type of flood protection ("hardened facilities", sandbags, flood doors, bulkheads, etc.) is reviewed. Any emergency procedures required to implement flood protection and warning times available for implementation thereof are reviewed, based on the flood conditions identified in other sections.

If there is evidence of potential structural effects, the Structural Engineering Branch (SEB) will be requested by HMB to ascertain whether these effects are properly considered in the structural design bases for the plant; similarly, ASB will be requested by HMB to ascertain whether these effects are properly considered in the systems design bases for the plant. Guidance for determining whether these potential effects are considered properly is outlined in the appropriate SEB and ASB SRP sections.

II. ACCEPTANCE CRITERIA

The flood design basis for each facility must be comparable with the positions in Regulatory Guide 1.59. For construction permit (CP) reviews, the types of flood protection (discussed in Regulation Guide 1.102) proposed must be capable of protecting those safety-related structures, systems, and components identified in Regulatory Guides 1.59 and 1.29.

For operating license (OL) reviews the specific designs of flood protection measures are reviewed to assure the protection levels are adequate (including static and dynamic effects) for the controlling flood conditions and that any necessary technical specifications are considered.

Standard engineering practice in positive flood control and shore protection, such as that developed by the Corps of Engineers, provides the basis for acceptance of methods to be employed for protection. Where sites are "hardened," that is, where emergency action is required, the time available to implement emergency procedures must be estimated by analysis of the hydrologic design event. The environmental conditions likely to prevail during all potential flooding events up to and including events of the severity of the controlling event are compared with the requirements for implementing flood emergency procedures. If the environmental conditions likely are such that the procedures can be carried out, they will be considered acceptable. An appropriate item in the plant Technical Specifications will be required in cases where emergency procedures are required to assure adequate flood protection.

"Hardened" flood protection (as discussed in Regulatory Guide 1.59, for facilities identified in Regulatory Guide 1.29) will be interpreted to mean "almost always in place."

III. REVIEW PROCEDURES

The estimated design basis flood level is compared with the locations and elevations of safety-related components. The staff will independently determine from analyses of postulated individual hydrologic events whether flood protection is required, and if so, what protective levels (including static and dynamic effects) are applicable. These data are transmitted to Structural Engineering Branch (SEB) for determination of structural design criteria adequacy and to Auxiliary Systems Branch (ASB) and Instrumentation and Control Systems Branch (ICSB) for determination of safety system adequacy. For flood protection requiring emergency action, the design basis flood conditions, and other less severe events, are reviewed to establish the minimum time available for implementation of emergency procedures. Physical parameters such as rate-of-rise (of river or lake levels), as well as evaluation (based on experience and engineering judgment) of flood warning networks, provide the staff with an independent estimate of available time. These data are provided to ASB and ICSB for their independent evaluation of the time required to implement shutdown and flood protective measures.

For OL reviews, the design of flood protection measures is reviewed to assure compatibility with the original design basis. For those plants for which shutdown (if required under Regulatory Guide 1.59, Position 2) and installation of protective measures is required in the event of a major flood, the procedures for carrying out these measures are reviewed for compatibility of available and required times as established above. The Technical Specifications must reference an emergency plan which allows for the orderly installation of required flood protection.

The above reviews are performed only when applicable to the site or site region. Some items of review may be done on a generic basis.

IV. EVALUATION FINDINGS

For CP reviews, the findings will consist of statements of flood design bases for safety-related facilities. If emergency procedures are required, the findings will indicate staff conclusions that time for implementation and methods of providing flood protection provide the necessary protection.

For OL reviews the findings will indicate the flood protection measures provided for safety-related facilities, and will indicate the type of technical specifications required to assure that the protection will be in place.

If Regulatory Guide 1.59, Position 2, is elected by the applicant, a statement describing lesser design bases will be included in the findings with the staff's conclusion of adequacy.

A sample CP-stage statement follows:

> "The applicant states, and we concur, that the station is above the flood level of a Probable Maximum Flood (PMF), either on the A River or the two intermittent streams crossing the site.
>
> "Further, the applicant has stated that the roofs of safety-related buildings will be constructed to safely dispose of, or store, local precipitation as severe as the Probable Maximum Precipitation (PMP). Further, we conclude that the bases for plant grading and drainage will be sufficient to prevent a threat to safety-related facilities by a localized PMP."

V. REFERENCES

Other SRP sections in the 2.4 series provide hydrologic design basis flood levels and environmental condition descriptions. Reports of the Corps of Engineers, United States Geologic Survey, Bureau of Reclamation, National Oceanic and Atmospheric Administration, and others will be used on an "as available" basis to evaluate flood warning systems, if applicable. The references for acceptability of protection will be completed projects of the Corps of Engineers and other Federal, state, and local agencies, and similar types of protection previously reviewed and found acceptable for other nuclear plants.

1. Regulatory Guide 1.70, "Standard Format and Contents of Safety Analysis Reports for Nuclear Power Plants."

2. Regulatory Guide 1.59, "Design Basis Flood for Nuclear Power Plants."

3. Regulatory Guide 1.29, "Seismic Design Classification."

4. Regulatory Guide 1.102, "Flood Protection for Nuclear Power Plants."

5. ANSI N170, "Standards for Determining Design Basis Flooding at Power Reactor Sites".

6. Regulatory Guide 1.125, "Physical Models for Design and Operation of Hydraulic Structures and Systems for Nuclear Power Plants."

7. Regulatory Guide 1.127, " Inspection of Water-Control Structures Associated with Nuclear Power Plants."

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SECTION 2.4.11 LOW WATER CONSIDERATIONS

REVIEW RESPONSIBILITIES

Primary - Hydrology-Meteorology Branch (HMB)

Secondary - None

I. AREAS OF REVIEW

The purpose of this section of the applicant's safety analysis report (SAR) is to identify natural events that may reduce or limit the available cooling water supply, and to assure that an adequate water supply will exist to operate or shut down the plant, as required.

Depending on the site, the areas of review include:

1. The worst drought considered reasonably possible in the region.
2. Low water (setdown) resulting from surges, seiches, or tsunamis.
3. Low water resulting from icing.
4. The effect of existing and proposed water control structures (dams, diversions, dam failures, etc.).
5. The intake structure and pump design basis in relation to the events described in SAR Sections 2.4.11.1, 2.4.11.2, 2.4.11.3 and 2.4.11.4.
6. The use limitations imposed or under discussion by federal, state, or local agencies authorizing the use of the water.
7. The range of water supply required by the plant, including minimum operating and shutdown flows, compared to availability.
8. The effects of potential blockage of intakes by sediment and littoral drift.

II. ACCEPTANCE CRITERIA

Acceptance is based principally on the adequacy of the intake design basis for safe shutdown, cooldown (first 30 days), and long-term cooldown (periods in excess of 30 days) during adverse natural phenomena or other events which require shutdown. Where the specific design bases preclude plant operation during severe hydrologically-related events, sufficient warning time must be demonstrated so that the plant may be shut down during or in advance of adverse events without causing potential damage to safety-related facilities. In cases where sufficient warning time to permit advance shutdown is considered necessary to protect safety-related components, an item in the plant Technical Specifications will be required.

SAR Section 2.4.11.1 (Low Flow in Rivers and Streams): For essential water supplies the low flow/low level design for the primary water supply source must be based on the probable minimum low flow and level resulting from the most severe drought that can reasonably be considered possible for the region. The low flow and level design bases for operation (if different than the design bases for essential water requirements) should be such that shutdowns caused by inadequate water supply will not cause frequent use of emergency systems. In cases where a common source of cooling water for operation and safety is provided, and where operation can affect minimum levels required for safety, the system will be acceptable if technical specifications are provided for shutdown before the ultimate heat sink can be adversely affected.

SAR Section 2.4.11.2 (Low Water Resulting from Surges, Seiches or Tsunamis): If the site is susceptible to such phenomena, minimum water levels resulting from setdown (sometimes called runout or rundown) from hurricane surges, seiches, and tsunamis must be higher than the intake design basis for essential water supplies. For coastal sites, the appropriate probable maximum hurricane (PMH) wind fields must be postulated to give maximum winds blowing offshore, thus creating a probable minimum surge level. Low water levels on inland ponds, lakes, and rivers due to surges must be estimated from probable maximum winds oriented away from the plant site. The same general analysis methods discussed in SRP Sections 2.4.3, 2.4.5 and 2.4.6 are applicable to low water estimates due to the various phenomena discussed.

SAR Section 2.4.11.3 (Historical Low Water): If historical flows and levels are used to estimate design values by inference from frequency distribution plots, the data used must be presented so that an independent determination can be made. The data and methods of the National Oceanic and Atmospheric Administration, United States Geologic Survey, Soil Conservation Service, Bureau of Reclamation, and the Corps of Engineers are acceptable.

SAR Section 2.4.11.4 (Future Controls): This section is acceptable if water use and discharge limitations (both physical and legal), already in effect or under discussion by responsible federal, regional, state, or local authorities, that may affect water supply at the plant have been considered and are substantiated by reference to reports of the appropriate agencies. The most adverse possible effects of these controls must be shown and taken into account in the design basis to assure that essential water supplies are not likely to be affected adversely in the future.

SAR Section 2.4.11.5 (Plant Requirements): Acceptance is based on the following required information:

1. Minimum essential cooling water flow rates and levels must be presented (or cross-referenced) and shown to be less than the probable minimum low flows and levels from the applicable sources of supply.
2. Maximum water requirements for normal operation must be presented and (if applicable) shown to be less than the water available under all likely conditions from the sources of supply.

SAR Section 2.4.11.6 (Heat Sink Dependability Requirements): The required data and information are those necessary to determine that the facility meets the criteria of Regulatory Guide 1.27. The analyses will be considered complete and acceptable if the following are adequately addressed:

1. The initial water inventory must be sufficient for shutdown and cooldown of the plant.
2. Water losses (such as seepage, drift, and evaporation) must be conservatively estimated, as suggested in Regulatory Guide 1.27.
3. The design basis hydrometeorology (temperature, dewpoint, etc.) must be as conservative as the criteria of the guide (see SRP section 2.3).
4. The limit on the heat sink return water temperature must be less than the maximum allowable cooling water inlet design temperature.
5. The heat sink intakes are located such that no potential exists for blockage by littoral drift and/or sediment that would decrease water supply below minimum required levels.

III. REVIEW PROCEDURES

Minimum plant requirements (water level and flow) that are identified in SAR Sections 2.4.11.5 or 9.2.5 are compared to the estimated minimum water levels and flows given in Section 2.4.11.1. If normal operation is not assured at the minimum water supply conditions, and loss of normal operation capability can adversely affect safety-related components, estimates of warning time are reviewed to assure that shutdown or conversion to alternate water sources can be accomplished prior to the trip. For such cases, emergency operating procedures are required, and are reviewed to assure that they are consistent with the postulated conditions. The analysis of the dependability of the ultimate heat sink is reviewed and the conclusions are provided to the Auxiliary Systems Branch (ASB) and Power Systems Branch (PSB). Determination of the dependability of the ultimate heat sink is accomplished by using Regulatory Guide 1.27 as a standard of comparison.

Each source of water for normal or emergency shutdown and cooldown, and the natural phenomena and site-related accident design criteria for each should be identified. A systems analysis is first undertaken of all water supply sources to determine the likelihood that at least one source would survive (1) the most severe of each of the natural phenomena; (2) site-related accident phenomena; and (3) reasonable combinations of less severe natural and accident phenomena. Second, arbitrarily assumed mechanistic failures of water supply structures and conveyance systems are postulated and the systems analysis repeated, to assure that the failure of one component will not cause failure of the entire system. These analyses are coordinated with the ASB and PSB review of the ultimate heat sink and related cooling systems, to avoid duplication. Operating rules for each portion of the system are ascertained to determine the amount of water that can be assumed available in the event of normal or accidental shutdown. If there is evidence of potential structural or mechanical effects, the Structural Engineering Branch (SEB) or Mechanical Engineering Branch (MEB) will be requested by HMB to ascertain whether the effects are properly considered in the structural or mechanical design bases for the plant. Consultations with the Meteorology Section of HMB, the Geosciences Branch (GB), the Accident Analysis Branch, the Structural Engineering Branch, ASB and PSB are undertaken where design criteria are not firmly established.

Estimates of water loss due to drift, evaporation, and blowdown are evaluated based on observed severe hydrometeorological measurements at similar locations (coordinated with the Meteorology Section of HMB). If independent analyses are deemed necessary, computer programs such as HEC-2 (Water Surface Profiles), HEC-3 (Reservoir System Analysis), and HEC-4 (Monthly Streamflow Simulation) are utilized.

The potential for surges in intake sumps (i.e., seiching in intake structures and surges in intake pipes) that could cause adverse effects are reviewed to assure that the effects have been properly incorporated for the intake design. The potential for adverse hydrodynamic effects of a trip of the intake pumps is evaluated based on potential surges in intake sumps.

For multiple purpose (normal operation, normal shutdown, and emergency shutdown) water supply systems, the primary portion of the system is first reviewed to determine that the water supply will be maintained at minimum volume requirements at all times. The secondary portion of the system is then reviewed to determine whether an adequate emergency water supply can be expected to be available during operating conditions such as the regional drought of record (flows must be adjusted for historical and potential future effects). If not, the applicant is requested to provide a technical specification requiring plant shutdown at the point where an adequate shutdown water supply is still assured.

Institutional restraints on water use, such as limitations in water use and discharge permits, are reviewed to assure the plant will have an adequate supply and not exceed limitations imposed upon operation. If a conflict is foreseen, the applicant is requested to either obtain a variance or make a design change to accommodate the limitation.

The potential for blockage of the intakes by littoral drift and sediment is reviewed to assure that the intakes are located and sized to prevent blockage which would preclude use of the safety-related water supply. Applicable literature describing historic sediment accumulations in the site region is reviewed to determine if mitigative measures are required to protect safety-related facilities. Independent estimates of "worst case" buildups will be made using statistical or deterministic techniques.

For plants using rivers, minimum design service water levels are compared with asymtotic extrapolations of low flow frequency curves which have been corrected for historical and potential future effects. For ocean or estuary plants, design low water levels are compared with probable maximum hurricane and tsunami-induced low water levels. For Great Lakes plants, design low water levels are compared with minimum historical levels coincident with probable maximum surge or seiche-induced low water levels.

If the ultimate heat sink system is not capable of continued long-term water supply under the criteria in Regulatory Guide 1.27, or the above considerations, the system will be reviewed in two parts; short-term capability and long-term capability. For short-term capability, the ASB, PSB, and the Licensing Project Manager (LPM) will be informed if the independently-estimated supply appears to be less than 30 days. The applicant will be asked to determine whether sufficient personnel and equipment can safely be made available

to switch water supply sources in the event of an accident. If emergency procedures are required to obtain the use of alternate water supplies, the applicant's water supply sources and procedures will be reviewed with ASB, PSB, and the LPM to determine that there is continuity of water supply. The time period for which a highly dependable water supply would be available is compared with the time required to obtain water from an alternative supply, and the natural or accident environmental conditions which could prevail.

For long-term water supply capability, different sources and means of obtaining water may be required because of the limited capability of a "short-term" supply. In those cases where different sources are necessary to assure the long-term plant heat removal capability, the alternative sources and the means of supplying water from the sources to the plant should be identified. Any plant design provisions necessary for such situations should also be described or a reference provided to other SAR sections for the descriptions.

Emergency means for obtaining long-term water supplies will be judged on the basis of the time required to obtain such supplies, natural or accident phenomena likely to prevail or to have caused the need for such supplies, and the dependability of the supply itself.

IV. EVALUATION FINDINGS

For construction permit (CP) reviews the findings will summarize the applicant's and staff's estimates of the design basis minimum water flows and levels. If the applicant's estimates are no more than 5% less conservative than the staff's estimates, staff concurrence in the applicant's estimates will be stated. If the applicant's estimates are more than 5% less conservative and if the proposed plant may be adversely affected, a statement of the staff's position (bases) will be made. A similar finding on the design bases for the ultimate heat sink will be made. If technical specification requirements are needed to assure an adequate supply, they will be indicated in the CP statement and required for operation.

For operating license (OL) reviews of plants for which detailed low water reviews were done at the CP stage, the CP conclusions will be referenced. In addition, the results of a review to reaffirm the low water design bases will be noted. If no changes have been made to the ultimate heat sink design since the CP review, the conclusions of the CP will be referenced. However, for both the low water considerations and the ultimate heat sink, an evaluation will be made during the OL review to assure that the design bases have been properly implemented. The availability of long-term water supply will be noted. If no low water and ultimate heat sink review was undertaken at the CP stage (of the scope described), this fact will be noted also.

A sample CP-stage statement follows:

> "The applicant proposes two sources of water supply; groundwater and the adjacent A River.

"Groundwater would be used for make-up to the essential service water cooling towers, for potable water supply, and for demineralizer water. The applicant estimates the demineralizer would require about 825 gallons per minute (gpm) for the first several months and an average rate of 425 gpm thereafter. Potable water requirements are estimated at about 10 gpm.

"The A River is to provide the principal source of cooling water. The applicant estimates the maximum water requirement for the plant will be 107 cfs. Of this, 61 cfs would be consumptively used and 46 cfs would be returned to the A River. The historical recorded low flow in the A River in the site region was about 500 cfs at the B gage on September 14, 1958 and about 440 cfs at the C gage on August 20, 1934. The applicant estimates the comparable low flow at the site to be 400 cfs. Assuming breaching of D Dam five miles downstream, the low flow would result in an estimated water surface elevation of 664 ft MSL.

"Emergency cooling sources and associated principal facilities comprise the A River, groundwater, the river screenhouse, the essential service cooling towers, groundwater well(s) and attendant distribution systems. The river screenhouse is to be a seismic Category I facility and was initially proposed to be protected from flooding up to the Standard Project Flood (SPF). Groundwater wells, located at the plant site, are above estimated PMF water levels. The applicant proposes to use groundwater for make-up to the essential service towers whenever the A River, screenhouse, or piping is unavailable. Estimated groundwater use would be 1600 gpm. At the staff's request the applicant reconsidered the flood design basis for the river screenhouse for relatively long periods of time when the A River could be higher than a SPF and an earthquake could prevent water from being available from wells. The applicant subsequently upgraded the flood design basis for the screenhouse to a Probable Maximum Flood, and concludes the proposed facilities meet the suggested criteria of Regulatory Guide 1.27 - Ultimate Heat Sink. We concur.

V. REFERENCES

1. L. R. Beard, "Methods for Determination of Safe Yield and Compensation Water from Storage," Seventh International Water Supply Congress, Barcelona, Spain (1966).

2. L. R. Beard, "Statistical Methods in Hydrology," Corps of Engineers (1962).

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4. D. K. Brady, et al., "Surface Heat Exchange at Power Plant Cooling Lakes," EEI Publication 69901, Edison Electric Institute, New York, Nov. 1969.

5. V. T. Chow (ed), "Handbook of Applied Hydrology," McGraw-Hill Book Company, New York (1964).

6. J. E. Edinger and J. C. Geyer, "Heat Exchange in the Environment," EEI Publication 69-902, Edison Electric Institute, New York, June 1965.

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8. "Scientific Hydrology," Ad Hoc Panel on Hydrology, Federal Council for Science and Technology, Washington, D.C., June 1962.

9. M. B. Fiering, and M. M. Hufschmidt, "Simulation Techniques for Design of Water-Resource Systems," Harvard University Press, Cambridge, Mass. (1966).

10. R. K. Linsley, et al., "Hydrology for Engineers," McGraw-Hill Book Company, New York (1958).

11. R. K. Linsley and J. B. Franzini, "Water-Resources Engineering," McGraw-Hill Book Company, New York (1964).

12. A. Maas, et al., "Design of Water-Resources Systems," Harvard University Press, Cambridge, Mass. (1962).

13. G. W. Platzman, "The Dynamical Prediction of Wind Tides on Lake Erie," Technical Report No. 7, Department of Geophysical Sciences, University of Chicago (1962).

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15. "Hydrologic Engineering Methods for Water Resources Development," Vol. 1-12, Corps of Engineers Hydrologic Engineering Center, Davis, California (1971).

16. "Reservoir Storage-Yield Procedures", Corps of Engineers Hydrologic Engineering Center, Davis, California (1967).

17. "Shore Protection Planning and Design," Technical Report No. 4, Third Edition, Corps of Engineers Coastal Engineering Research Center (1966); and "Shore Protection Manual," 1973.

18. Regulatory Guide 1.27, "Ultimate Heat Sink."

19. "Design of Small Dams," Second Edition, Bureau of Reclamation, U.S. Department of Interior (1973).

20. "Interim Report - Meteorological Characteristics of the Probable Maximum Hurricane, Atlantic and Gulf Coasts of the United States," Report HUR 7-97 (see also HUR 7-97A), U.S. Weather Bureau (now NOAA) (1968).

21. "Water Surface Profiles," HEC-2, Corps of Engineers Hydrologic Engineering Center (continuously updated).

22. "Reservoir System Analysis," HEC-3, Corps of Engineers Hydrologic Engineering Center (updated).

23. "Monthly Streamflow Simulation," HEC-4, Corps of Engineers Hydrologic Engineering Center (updated).

24. Regulatory Guide 4.4, "Reporting Procedure for Mathematical Models Selected to Predict Heated Effluent Dispersion in Natural Water Bodies."

U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

SECTION 2.4.12 DISPERSION, DILUTION, AND TRAVEL TIMES OF ACCIDENTAL RELEASES OF LIQUID EFFLUENTS IN SURFACE WATERS

REVIEW RESPONSIBILITIES

Primary - Hydrology-Meteorology Branch (HMB)

Secondary - Effluent Treatment Systems Branch (ETSB)

I. AREAS OF REVIEW

The ability of the surface water environment to disperse, dilute, or concentrate accidental radioactive liquid effluent releases is reviewed with emphasis on relating the effects of such releases to existing and known future uses of surface water resources. (Note that effects of normal releases and of the more likely accidents are discussed in the applicant's environmental report.)

II. ACCEPTANCE CRITERIA

Transport characteristics of the surface water environment with respect to existing and known future users must be described for conditions which reflect worst case release mechanisms and source terms so as to postulate the most pessimistic contamination from accidentally released liquid effluents. Estimates of physical parameters necessary to calculate the transport of liquid effluent from the points of release to the site of existing or known future users must be described. Potential pathways of contamination to surface water users must be identified. Sources of information and data must be described and referenced. Acceptance is based on the staff's evaluation of the applicant's computational methods and the apparent completeness of the set of parameters necessary to perform the analysis.

One- or two-dimensional mathematical models are acceptable to analyze the flow field and convective dispersion of contaminants in surface waters, providing that the models have been verified by field data and that conservative site-specific hydrologic parameters are used. Furthermore, conservatism must be the guide in selecting the proper model to represent a specific physical situation. Radioactive decay and sediment adsorption may be considered, if applicable, providing that the adsorption factors are conservative and site-specific.

III. REVIEW PROCEDURES

Section 2.4.12 of the applicant's SAR is reviewed to identify any missing data, information or analysis necessary for the staff's evaluation. Applicant responses to the requested

information will be evaluated using the methods outlined below and staff positions will be developed. Resolution, if possible, of differences between staff's and the applicant's estimation of liquid effluent dispersion will be coordinated through the LPM; and the SER will be written accordingly.

Independent calculations will be made of liquid effluent transport for the contamination pathways identified. For preliminary analysis, the staff will employ simplified calculational procedures or models, such as those contained in References 2 and 8. The analysis will be performed using demonstrably conservative coefficients and assumptions, and the physical conditions (such as lowest recorded river flow) likely to give the most adverse dispersion of the liquid effluent. The applicant's model assumption and results will be compared with the staff's results to assure that the results are comparably conservative. The estimation of liquid effluent dispersion will reflect potent'al future changes that might result from variations in use by known future surface and groundwater users.

Concentrations of radionuclides in the body of water under consideration will be calculated based on the staff's dispersion computations and with initial concentrations provided by the Effluent Treatment Systems Branch (ETSB) for the most critical event. Acceptability of the resultant concentrations of radioactive effluent at the points of interest will be determined by consultation with ETSB. If the concentrations of the diluted liquid effluents computed by the staff are within acceptable limits of Appendix B, Table II, Column 2, of 10 CFR Part 20, no further computation effort is indicated. If the concentrations computed by conservative simplified methods exceed the limits of Part 20, more precise and less conservative models, such as those used for hydrothermal prediction (Reference 9), and coefficients will be employed by the staff.

IV. EVALUATION FINDINGS

For construction permit (CP) reviews, the findings will summarize the applicant's and staff's estimates of dilution factors, dispersion coefficients, flow velocities, travel times, and potential contamination pathways between the site and the nearest water user. If the estimates are comparable, or if no potential problem exists, staff concurrence with the applicant's estimates will be stated. If the staff predicts substantially more conservative conditions, a statement of the staff basis will be made.

For operating license (OL) reviews of plant designs that have had detailed reviews of severe accidental effluent releases at the CP stage, the CP conclusions will be referenced. If no CP review of effluent releases was undertaken of the scope indicated herein, this will be indicated. Any new potential pathways or changes in water usage that can be identified in the OL review will also be analyzed and reported.

Sample statements for CP reviews follows:

> "At the staff's request, the applicant provided analyses of the effects (travel times, dispersion coefficients, dilution factors, etc.) of an accidental spill of liquid radioactive wastes into the surface water. A postulated failure of the

condensate storage tank, releasing 500,000 gallons of water containing low-level activity was evaluated. The applicant assumed that this volume of water would travel overland to the adjacent stream before any dilution would occur. The applicant concluded, and the staff concurs, that adequate dilution would occur in the surface water prior to reaching any potential users. The applicant also investigated the possibility of the spill being recirculated through the plant circulating water system. This analysis showed that it was extremely unlikely that recirculation could occur since the condensate storage tank is located downstream of the circulating water intake structure. The staff concurs in this evaluation. Accidental spills that could enter the ground water and reach potential users before or after discharging into surface waters are discussed in Sections 2.4.13 and Section 15 of this report.

"No accidental release of sufficient volume of liquids containing radioactivity directly into surface waters is considered reasonable at the site because storage facilities are located inside of safety-related buildings and the manner in which liquids are to be handled at the site precludes this possibility. Accidental spills of liquids into the groundwater, which could eventually reach surface waters, are discussed in Sections 2.4.13 and 15 of this report."

V. REFERENCES

In addition to the following references describing methods and techniques of evaluation, published data by federal, state, and other agencies and organizations will be used as available.

1. N. H. Brooks, "Diffusion of Sewage Effluent in an Ocean Current," in "Waste Disposal in the Marine Environment," Pergamon Press, New York (1960).

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5. S. N. Davis and R. J. M. DeWiest, "Hydrogeology," John Wiley & Sons, Inc., New York (1966).

6. Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants."

7. "NRC Dispersion Workbook," (in preparation).

8. Regulatory Guide 1.113, "Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I."

9. G. H. Jirka, G. Abraham, D. R. F. Harleman, "An Assessment of Techniques for Hydrothermal Prediction," USNRC, NUREG 0044, 1976.

10. Regulatory Guide 4.4, "Reporting Procedure for Mathematical Models Selected to Predict Heated Effluent Dispersion in Natural Bodies of Water."

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U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

SECTION 2.4.13 GROUNDWATER

REVIEW RESPONSIBILITIES

Primary - Hydrology-Meteorology Branch (HMB)

Secondary - Effluent Treatment Systems Branch (ETSB)

I. AREAS OF REVIEW

Data presented in the applicant's safety analysis report (SAR) on local and regional groundwater reservoirs are reviewed to establish the effects of groundwater on plant foundations. Other areas reviewed under this plan include identification of the aquifers and the type of onsite groundwater use, the sources of recharge, present and future withdrawals, an evaluation of accident effects, monitoring and protection requirements, and design bases for groundwater levels and hydrodynamic effects of groundwater on safety-related structures and components. Flow rates, travel time, gradients, other properties pertaining to the movement of accidental contamination, and groundwater levels beneath the site are reviewed, as are seasonal and climatic fluctuations, or those caused by man, that have the potential for long-term changes in the local groundwater regime. ETSB will provide accident scenarios for HES staff use in evaluating accidental spills.

II. ACCEPTANCE CRITERIA

For SAR Section 2.4.13.1: A full, documented description of regional and local groundwater aquifers, sources, and sinks is required. In addition, the type of groundwater use, wells, pump and storage facilities, and the flow requirements of the plant must be described. If groundwater is to be used as an essential source of water for safety-related equipment, the design basis for protection from natural and accident phenomena must compare with Regulatory Guide 1.27 guidelines. Bases and sources of data must be adequately described.

For SAR 2.4.13.2: A description of present and projected local and regional groundwater use must be provided. Existing uses, including amounts, water levels, location, drawdown, and source aquifers must be discussed and should be tabulated. Flow directions, gradients, velocities, water levels, and effects of potential future use on these parameters, including any possibility for reversing the direction of groundwater flow, must be indicated. Any potential groundwater recharge area within the influence of the plant and effects of construction, including dewatering, must be identified. The influence of existing and potential future wells with respect to groundwater beneath the site must also be discussed. Bases and sources of data must be described and referenced.

For SAR Section 2.4.13.3: Radionuclide transport characteristics of the groundwater environment with respect to existing and future users must be described for both operating and accident conditions. Estimates and bases for coefficients of dispersion, adsorption, groundwater velocities, travel times, gradients, permeabilities, porosities, and groundwater or piezometric levels between the site and existing or future surface and groundwater users must be described and be consistent with site characteristics. Potential pathways of contamination to groundwater users must also be identified. Sources of data must be described and referenced. One- or two-dimensional mathematical models are acceptable to analyze the flow field and convective dispersion of contaminants in surface waters, providing that the models have been verified by field data and that conservative site-specific hydrologic parameters are used. Furthermore, conservatism must be the guide in selecting the proper model to represent a specific physical situation. Radioactive decay and sediment adsorption may be considered, if applicable, providing that the adsorption factors are conservative and site-specific.

For SAR Section 2.4.13.4: The need for and extent of procedures and measures to protect present and projected groundwater users, including monitoring programs, must be discussed. These items are site-specific and will vary with each application.

For SAR Section 2.4.13.5: The design bases (and development thereof) for groundwater-induced loadings on subsurface portions of safety-related structures, systems, and components must be described. If a permanent dewatering system is employed to lower design basis groundwater levels, the bases for the design of the system and determination of the design basis for groundwater levels must be provided. Information must be provided regarding (a) all structures, components, and features of the system, (b) the reliability of the system as related to available performance data for similar systems used at other locations, (c) the various soil parameters (such as permeability, porosity, and specific yield) used in the design of the system, (d) the bases for determination of groundwater flow rates and areas of influence to be expected, (e) the bases for determination of time available to mitigate the consequences of system failure where system failure could cause design bases to be exceeded, (f) the effects of malfunctions or failures (such as a single failure of a critical active component or failure of circulating water system piping) on system capacity and subsequent groundwater levels, and (g) a description of the proposed groundwater level monitoring program and outlet flow monitoring program. Specific criteria relating to the design of permanent dewatering systems are presented in the attached Branch Technical Position HMB/GSB 1 "Safety-Related Permanent Dewatering Systems". In addition, if wells are proposed for safety-related purposes, the hydrodynamic design bases (and development thereof) for protection against seismically-induced pressure waves must be described and be consistent with site characteristics.

III. REVIEW PROCEDURES

Section 2.4.13 of the applicant's SAR is reviewed to identify any missing data, information, or analyses necessary for the staff's evaluation. Applicant responses to the requested information will be evaluated using the methods outlined below; and staff positions will be developed based on the results of the analysis. Resolution, if possible, of potential groundwater problems or of differences between applicant's and staff's design bases, will

be coordinated through the LPM, and the SER will be written accordingly. The review sequence is shown in Figure 2.4.13.

Local and regional groundwater conditions are reviewed by comparing the applicant's description with reports by the U. S. Geological Survey (USGS), other agencies, and professional organizations. Other NRC organizational elements with related review responsibilities will be notified of any applicable groundwater data and analyses. If onsite groundwater use and facilities are safety-related, the criteria of Regulatory Guide 1.27 are applied.

The staff will compare the applicant's description of present and projected local and regional groundwater use, existing users, including ambient use, water levels, location, and drawdown with information and data from references. Drawdown effects of projected future groundwater use, including the possibility for reversing the groundwater flow, will be evaluated and may be checked by independent calculations. Construction effects, including dewatering, on potential recharge areas may also be evaluated.

The staff will make independent calculations of the transport capabilities and potential contamination pathways of the groundwater environment under accidental conditions with respect to existing and future users. Special attention should be directed to proposed facilities with permanent dewatering systems to assure that pathways created by those systems have been identified. The staff will, in consultation with the Effluent Treatment Systems Branch (ETSB), choose the accident scenarios leading to the most adverse contamination of the groundwater or to surface water via the groundwater pathways. Analysis of the contamination will commence with the simplest models, such as those presented in References 22 and 23, using demonstratably conservative assumptions and coefficients. Dilutions and travel times (or alternatively, concentrations directly) resulting from the preliminary analyses will then be checked by ETSB to determine acceptability. If the indicated concentrations of radionuclides, identified by ETSB, are less than 10 CFR Part 20, Part B, no further computational efforts will be warranted. Further analyses using progressively more realistic and less conservative modeling techniques, such as those of References 9 and 26, will be undertaken if the preliminary results are not acceptable. The needs and plans for procedures, measures, and monitoring programs will be reviewed based upon site-specific groundwater features. Design bases for groundwater-induced loadings on subsurface portions of safety-related structures are reviewed. Independent calculations are performed to determine the adequacy of the design criteria and the capability to reflect any potential future changes which can be induced by variations in precipitation, construction of future wells and reservoirs, accidents, pipe failures, or other natural events. For dewatering systems, calculations are performed to determine phreatic surfaces, normal flow rates, flow rates into the system as a result of pipe breaks (circulating and service water system pipes), groundwater rebound times assuming total failure of the system, and system capacity.

The above reviews are performed only when applicable to the site or site region. Some items of review may be done on a generic basis.

IV. EVALUATION FINDINGS

For construction permit (CP) reviews, the findings will summarize the applicant's and staff's estimates of groundwater levels associated with safety-related structures, and where applicable, groundwater flow directions, gradients, velocities, effects of potential future use on these parameters, applicability and reliability of dewatering systems, and the effects of an accidental release of radioactive liquid effluent on existing and future users. If the design bases estimates are comparable, staff concurrence in the applicant's estimates will be stated. If the staff predicts substantially more conservative groundwater conditions and the proposed plant may be adversely affected, a statement of the staff bases will be made. If groundwater conditions do not constitute design bases, the findings will so indicate.

For operating license (OL) reviews of plants that have had detailed groundwater reviews at the CP stage, the CP conclusions will be referenced. In addition, a review of groundwater history since the CP review will be indicated and note of any changes in groundwater conditions or usage will be made. For permanent dewatering systems, any additional information regarding soil properties and groundwater conditions gathered during construction will be evaluated to determine the applicability of the assumed CP design basis. If no CP groundwater review was undertaken, of the scope indicated above, this fact will be noted in the OL findings in addition to the results of the current review.

A sample CP statement follows:

"Groundwater is available at the site in low to moderate yields from the following four aquifers listed by increasing depth below the surface: (1) the unconfined watertable aquifer consisting of the A and B formations, (2) the confined C-Upper D aquifer, (3) the confined upper D aquifer, and (4) the confined middle D aquifer. Groundwater in the A-B town aquifer generally moves toward the local streams; whereas, in the deeper confined aquifers, groundwater generally moves toward centers of pumping. At the present, saltwater intrusion into the aquifers at the site is not evident as a result of brackish water movement from the E Bay, the F Canal, or G Bay.

"The applicant plans to use groundwater during plant operation at a continuous rate of 140 gpm, of which 100 gpm will be used for demineralized water requirements, and 40 gpm will be service water for drinking, washing, and filling the fire protection storage tanks. The source of this supply will probably be the A-B aquifer, for which the applicant has conducted pumping tests at two locations. The applicant has indicated he may utilize another deeper aquifer for this supply, and has agreed to supply additional pumping test data to the staff for evaluation if another aquifer is chosen. This is acceptable to the staff.

"Precipitation is the source for groundwater recharge to the A-B aquifer. The recharge area for this aquifer lies to the southwest of the plant site and extends beyond the City of H. No major recharge areas for the lower confined aquifers are believed to exist in the vicinity of the site.

"A water-table design level of 65 feet MSL (15 feet below plant grade) was selected by the applicant to determine hydrostatic loadings on safety-related structures. The staff concurs that this level is conservative since the highest measured water table elevation at the plant site following an extremely rainy season was 63.4 feet MSL.

"The staff postulated an instantaneous rupture of the boron recycle tank with no containment by plant structures as being the design basis event for contamination of groundwater and surface water. The travel time to the nearest groundwater user was conservatively estimated to be a minimum of 29 years with a dilution of at least 6000. The resultant concentrations were found to be less than 10 CFR Part 20 B."

V. REFERENCES

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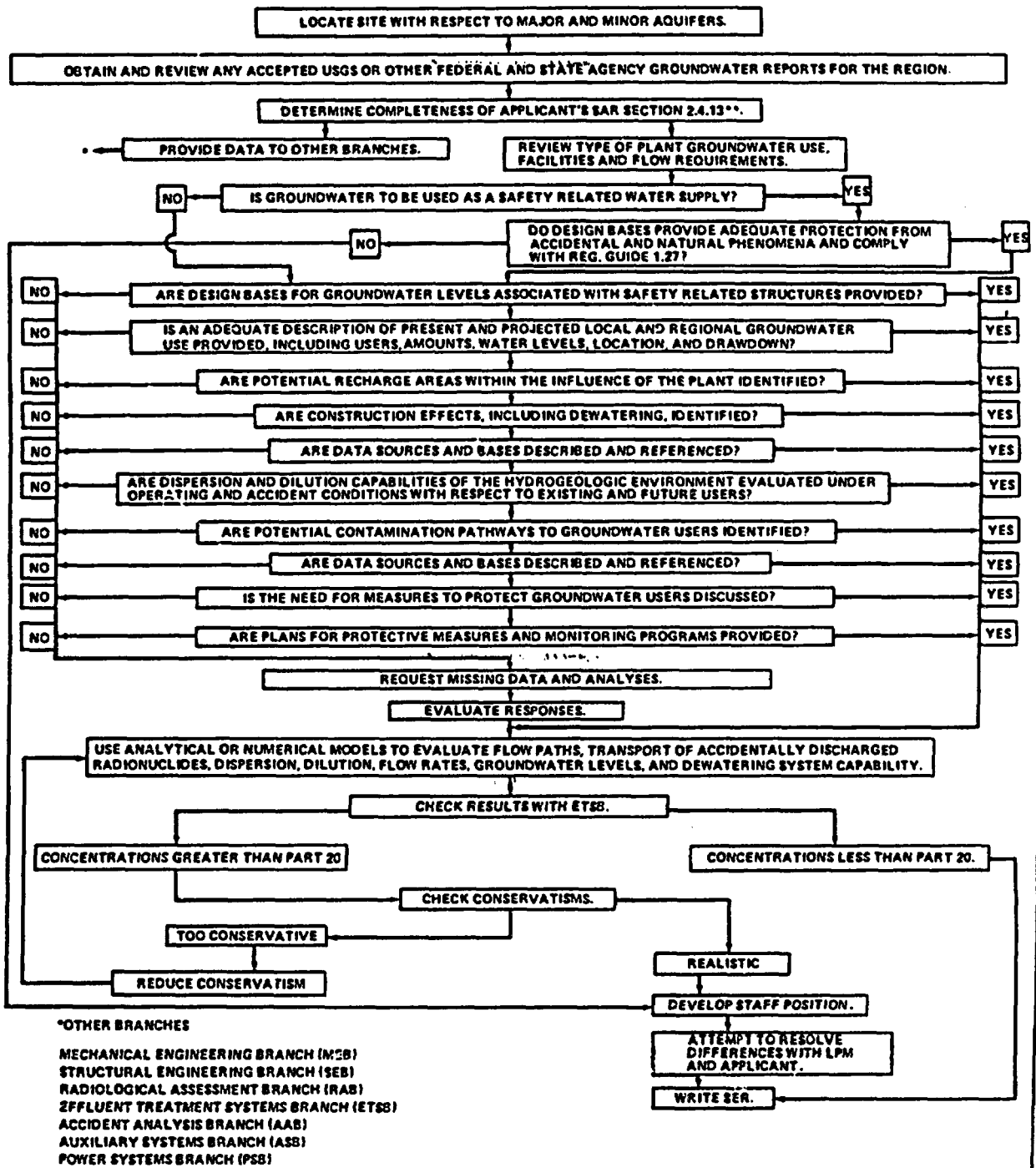
34. Branch Technical Position HMB/GSB 1, "Safety-Related Permanent Dewatering Systems", attached to this section.

FIGURE 2.4.13

STANDARD REVIEW PLAN SECTION 2.4.13
GROUNDWATER



*OTHER BRANCHES

MECHANICAL ENGINEERING BRANCH (MEB)
STRUCTURAL ENGINEERING BRANCH (SEB)
RADIOLOGICAL ASSESSMENT BRANCH (RAB)
EFFLUENT TREATMENT SYSTEMS BRANCH (ETSB)
ACCIDENT ANALYSIS BRANCH (AAB)
AUXILIARY SYSTEMS BRANCH (ASB)
POWER SYSTEMS BRANCH (PSB)

**APPLICANT PROVIDED INFORMATION IN SAR SECTION 2.4.13 IS COMPARED TO GUIDANCE GIVEN IN SF&C DOCUMENT (R.G.1.70) TO DETERMINE COMPLETENESS.

BRANCH TECHNICAL POSITIONS HMB/GSB 1
SAFETY-RELATED PERMANENT DEWATERING SYSTEMS

I. Summary

This position has been formulated to minimize review problems common to permanent dewatering systems that are depended upon to serve safety-related purposes by describing acceptable geotechnical and hydrologic engineering design bases and criteria. A safety-related designation for permanent dewatering systems is provided since they protect other safety-related structures, systems and components from the effects of natural and man caused events such as groundwater. In addition, the level of documentation of data and studies which are considered necessary to support safety-related functions is defined. This position applies to both active (e.g., uses pumps) and passive (e.g., uses gravity drains) dewatering systems. This position does not reflect structural, mechanical and electrical criteria.

II. Background

The staff has reviewed a number of permanent dewatering systems, including McGuire 1 & 2, Cherokee 1 & 2, Perkins 1 & 2, Perry 1 & 2, WPPSS 3 & 5, Douglas Point 1 & 2, and Catawba 1 & 2. Perry, beginning in 1975, was the first plant reviewed with such systems, and was reviewed very late in the CP process. Only WPPSS 3 & 5 and Douglas Point use a passive system (no pumps).

Permanent dewatering systems lower groundwater levels to reduce subsurface water loads on plant structures. In addition, they can increase plant operational dependability and reduce costs. These effects are accomplished by providing added means of keeping seepage water out of lower building levels during the later stages of plant life when normal waterproofing provisions may have deteriorated, and reducing radwaste system operating costs by minimizing the amount of drain water that must be treated. Benefits are, therefore, of two types, tangible (dollars) and intangible ("insurance"). We understand the construction costs of underdrains can vary widely depending on the design. Construction costs of between $125K to $1000K per unit have been suggested. The costs of coping with significant amounts of groundwater inleakage in safety-related building areas, which underdrains are expected to minimize, is estimated to be in the range of $100K to $200K per year per reactor. The construction costs of alternatives to underdrains for structural purposes alone (exclusive of inleakage treatment) is estimated to range upward from $300K per unit and is highly dependent on site conditions. Structural alternatives to permanent underdrains include additional concrete and steel in the lower portions of buildings, and the use of anchor systems to resist floatation.

Dewatering systems are generally composed of three components; the collector system, the drain system, and the discharge system. Water is first collected in collector drains

adjacent to buildings or excavations. Interceptor drains or piping are then used to convey this water to a final discharge system. The discharge system can be either gravity flow or a pumping system. Most underdrain structures, systems and components are buried alongside and under structures, although some systems employ pumping systems within larger structures (such as reactor or auxiliary buildings) to discharge collected water. Finally, permanent dewatering systems are not a required feature at any plant, but may be proposed as a cost effective feature.

Many permanent dewatering systems at nonnuclear facilities, such as dams and large buildings, have functioned over the years. However, the likelihood of a portion of such a system becoming ineffective and, therefore, not performing its intended function may well be considerably greater than the probability of occurrence of a nuclear power plant design basis event such as a Probable Maximum Hurricane, Probable Maximum Flood, or Safe Shutdown Earthquake. Losses of function in the past have generally been attributable to piping of fines, inadequate capacity, or clogging. We have concluded that safety analyses of such systems should consider reliability and failures of features of the system itself, as well as potentially adverse effects of failures of nearby nonsafety-related features. Such systems need not be designed for design earthquakes if they are not intended to perform as underdrains fully during or immediately following a severe earthquake, or if the system can be expected to perform an underdrain function in a degraded condition. Certain portions of such systems, however, may be required to regularly perform other safety functions (e.g., porous concrete base mats) and should be designed for severe earthquakes. Failure of a dewatering system could cause groundwater levels to rise above design levels, resulting in overloading concrete walls and mats not designed to withstand the resulting hydrostatic pressures. In addition to causing potential structural and equipment damage, groundwater could enter safety-related buildings and flood components necessary for plant safety.

The basis for staff concerns over the use of such systems is whether they can be expected to perform their function, and prevent structural failures and interior flooding of safety-related structures. The degree of concern is directly related to the corresponding degree to which the safety of the structures and systems rely on the integrity of the dewatering system, particularly with a dewatering system in a degraded situation. For example, if structures can accommodate hydrostatic loads that would result with a total failure of a dewatering system, our concerns have been primarily limited to the capability of such systems to perform their functions under relatively infrequent earthquake situations. If, however, such systems must remain functional (e.g., keep water levels down), whether in a degraded situation or not to prevent structural failures and internal flooding under potentially frequent conditions, we have been very concerned with system reliability.

Many applicants have indicated that their plants can withstand, or have been designed against, full hydrostatic loadings that would occur in the absence of the underdrain systems, but not if an earthquake were to occur. If the plant can withstand full hydrostatic loading, assuming degradation of the underdrain system, many of the staff's concerns may be eliminated from further consideration because of the time available for remedial action after detection of system degradation.

III. Situations Identified During Previous Reviews

Four general categories of situations have been identified during case reviews as follows:

(a) Estimating and Confirming Permeability Values

It is necessary to estimate the amount of water that will be collected so that system components such as strip drains, blanket drains, collector pipes, and pumps are adequately designed and sized. One of the most important and most difficult parameters to evaluate is the permeability of the soil and rock existing at a site. A permeability value could be affected significantly by conditions of concentrated flow along joints in fractured and weathered rocks, or within other aquifers affected by foundation excavation. In addition, geological and foundation conditions that were not detected in site explorations may affect flow conditions and cause the estimated permeability values and flow regimes to be substantially different from those assumed at the CP preliminary design stage. These conditions are often first detected during construction dewatering. Therefore, we have required a commitment to consider construction excavation and dewatering data in the final design of underdrain systems. (See situation (d) below.)

(b) Operational Monitoring Requirements

To guard against system malfunctions and to assure sufficient time is available for implementation of remedial measures before groundwater could rise to an unacceptable level, provisions must be made for early detection of system failures, and contingency measures for these failures must be well defined prior to plant operation. Since drain systems are usually buried and concealed and there may be no direct way of inspecting them, reliance must be placed on piezometers, observation wells, manholes, and monitoring of collected water to detect problems or malfunctioning of the system. The details of an operational monitoring program are necessary prior to construction of the underdrain to assure that each of the following will be provided: (a) an early detection alarm system during normal operating conditions; (b) regularly scheduled inspection and monitoring; and (c) competent evaluation of observations during both construction and operation. In addition, the bases for acceptable contingency measures suitable for coping with various possible hazards must be established at the CP stage.

(c) Pipe Breaks

A dewatering system might be overloaded by such conditions as leaks or breaks in either the circulating or service water systems. A leak through a pipe break may be a very small percentage of the total flow of the cooling water system, but large enough to exceed the hydraulic capacity of drains, pipes and pumps in the dewatering system. For example, a complete failure of circulating water system piping has been required in the design of the dewatering systems reviewed to date. This requirement was made to assure that such abnormal occurrences do not adversely affect the integrity of safety-related structures, systems, and components.

(d) Sequence of Review

Underdrain systems are usually one of the first items constructed and, after backfilling and construction of subsurface facilities, are then no longer visible for

regular inspection. In most cases, these systems are initially designed based on rather limited information from preconstruction field activities, and are tailored specifically for the site and facilities. By necessity then, final review and approval by the staff of the design must rely in some part on information gathered during construction. Therefore, the review and approval can be accomplished in two ways: (1) design details of the permanent underdrain system, the operational monitoring program and plans for construction dewatering can be submitted in the PSAR, with only confirmation of the details required prior to actual construction; or (2) conceptual designs of the permanent underdrain system and the operational monitoring program and details of construction dewatering can be submitted in the PSAR with the more complete review and approval based on construction dewatering requiring review and approval prior to actual construction. Review and approval of unique designs as post-CP matters is based upon 10 CFR Part 50, Subsections 35(b) and 55(e)(1)(iii). To prevent extending the review schedule, the first procedure would be the most desirable, but the staff recognizes that the detail required may not always be available at the time the PSAR is submitted.

IV. Proposed Staff Position

We have reviewed and approved the design of a limited number of permanent dewatering systems. However, because of the importance of these systems to plant safety, we have always required that they be designed and used in a conservative manner. The following is a list of required design provisions which are consistent with requirements in recent CP reviews:

(a) if the dewatering system is relied upon for any safety-related function, the system must meet the appropriate criteria of Appendix A and Appendix B to 10 CFR Part 50. In addition, guidance for structural, mechanical and electrical design criteria is provided in related sections of the Standard Review Plan for Category I structures, systems and components. However, all portions of the system need not be designed to accommodate all design basis events, such as earthquakes and tornados, provided that such events cannot either influence the system, or that the consequences of failure from such events is not important to safety; nevertheless, a clear demonstration of the effectiveness of a backup system and the timeliness of its implementation must be provided;

(b) the potential for localized pressures developing in areas which are not in contact with the drainage system, or in areas where pipes enter or exit the structural walls or mat foundations, must be considered.

(c) uncertainty in detecting operational problems and providing a suitable monitoring system must be considered;

(d) the potential for piping fines and clogging of filter and drainage layers must be considered;

(e) assurance must be provided that the system as proposed can be expected to reliably perform its function during the lifetime of the plant; and

(f) where the system is safety-related, is not totally redundant or is not designed for all design basis events, provide the bases for a technical specification to assure that in the event of system failure, necessary remedial action can be implemented before design basis conditions are exceeded.

V. SAR's (Std. Format & Content Information, Sections 2.4 & 2.b) for each of the plants with permanent dewatering systems should include the following information:

(a) Provide a description of the proposed dewatering system, including drawings showing the proposed locations of affected structures, components and features of the system. Provide information related to the geotechnical and hydrologic design of all system components such as interceptors, drainage blankets, and pervious fills with descriptions of material source, gradation limits, material properties, special construction features, and placement and quality control measures. (Note structural, mechanical and electrical information needs described elsewhere.) Where the dewatering system is important to safety, provide a discussion of its expected functional reliability. The discussion of the bases for reliability should include comparisons of proposed systems and components with the performance of existing and comparable systems and components for applications under site conditions similar to those proposed. Where such information is unavailable or unfavorable, or the application (design and/or site) is unique, the unusual features of the design should be supported by additional tests and analyses to demonstrate the conservative nature of the design. In such cases the staff will meet with the applicant, on request, to establish the bases for such additional tests and analyses.

(b) Provide estimates, and their bases, for soil and rock permeabilities, total porosity, effective porosity (specific yield), storage coefficient and other related parameters used in the design of the dewatering system. In general, these site parameters should be determined utilizing field and, if necessary, laboratory tests of materials representative of the entire area of influence of the expected drawdown of the system. Unless it can be substantiated that aquifer materials are essentially homogeneous, or that obviously conservative estimates have been used as design bases, provide preconstruction pumping tests and other in-situ tests performed to estimate the pertinent hydrologic parameters of the aquifer. Monitoring of pumping rates and flow patterns during dewatering for the construction excavation is also necessary to verify assumed design bases relating to such factors as permeability and aquifer continuity. In addition, the final design of the system should be based on construction dewatering data and related observations to assure that the values estimated from site exploration data are conservative. Lastly, the final design of the dewatering system and its hydrologic and geotechnical operational monitoring program should be confirmed by construction excavation and dewatering information.

If such information fails to support the conservatism of design information previously reviewed by the staff, the changed information should be reviewed under 10 CFR Part 50, Subsections 35(b) and 55(e)(1)(iii).

(c) Provide analyses and their bases for estimates of groundwater flow rates in the various parts of the permanent dewatering system, the area of influence of drawdown, and the shapes of phreatic surfaces to be expected during operation of the system. The extent of influence of the drawdown may be especially important if a natural or man-made water body affects, or is affected by, the dewatering systems.

(d) Provide analyses, including their bases, to establish conservative estimates of the time available to mitigate the consequences of system degradation* that could cause groundwater levels to exceed design bases. Document the measures that will be taken to either repair the system, or provide an alternate dewatering system that would become operational before the design basis groundwater level is exceeded.

(e) Provide both the design basis and normal operation groundwater levels for safety-related structures, systems and components. The design basis groundwater level is defined as the maximum groundwater level used in the design analysis for dynamic or static loading conditions (whichever is being considered), and may be in excess of the elevation for which the underdrain system is designed for normal operation. This level should consider abnormal and rare events (such as an occurrence of the Safe Shutdown Earthquake (SSE), a failure of a circulating water system pipe, or a single failure within the system), which can cause failure or overloading of the permanent dewatering system.

(f) A single failure of a critical active feature or component must be postulated during any design basis event. Unless it can be documented that the potential consequences of the failure will not result in Regulatory Guides 1.26 and 1.29 dose guidelines being exceeded, either (1) document by pertinent analyses that groundwater level recovery times are sufficient to allow other forms of dewatering to be implemented before the design basis groundwater level is exceeded, discuss the measures to be implemented and equipment needed, and identify the amount of time required to accomplish each measure, or (2) design for all system components for all severe natural phenomena and events. For example, if the design basis groundwater level can be exceeded only as a result of a single nonseismically induced failure of any component or feature of the system, the staff may allow the design basis level of the dewatering system to be exceeded for a short period of time (say 2 or 3 days), provided that (1) effective alternate dewatering means can be implemented within this time period, or that (2) it can be shown that Regulatory Guides 1.26 and 1.29 guidelines will not be exceeded by groundwater induced impairments of safety-related structures, systems, or components.

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*See (f) for considerations of differing system types.

(g) Where appropriate, document the bases which assure the ability of the system to withstand various natural and accidental phenomena such as earthquakes, tornadoes, surges, floods, and a single failure of a component feature of the system (such as a failure of any cooling water pipes penetrating, or in close proximity to, the outside walls of safety-related buildings where the groundwater level is controlled by the system). An analysis of the consequences of pipe ruptures on the proposed underdrain system must be provided, and should include considerations of postulated breaks in the circulating system pipes at, in, or near the dewatering system building either independently of, or as a result of the SSE. Unless it can be documented that the potential consequences will not be serious enough to affect the safety of the plant to the extent that Regulatory Guides 1.26 and 1.29 guidelines could be exceeded, provide analyses to document that (1) water released from the pipe break cannot physically enter the dewatering system, or (2) if water enters the dewatering system, the system will not be overloaded by the increased flow such that the design basis groundwater level is subsequently exceeded.

(h) State the maximum groundwater level the plant structures can tolerate under various significant loading conditions in the absence of the underdrain system.

(i) Provide a description of the proposed groundwater level monitoring programs for dewatering during plant construction and for permanent dewatering during plant operation. Monitoring information requested includes (1) the general arrangement in plan and profile with approximate elevation of piezometers and observation wells to be installed, (2) intended zone(s) of placement, (3) type(s) of piezometer (closed or open system), (4) screens and filter gradation descriptions, (5) drawings showing typical installations showing limits of filter and seals, (6) observation schedules (initial and time intervals for subsequent readings), (7) plans for evaluation of recorded data, and (8) plans for alarm devices to assure sufficient time for initiation of corrective action. Provide a commitment to base the final design of the operational monitoring program on data gathered during the construction monitoring program (if construction experience shows the assumed operational program bases to be nonconservative or impractical). Changes to the operational program are to be documented in the FSAR.

(k) Provide information regarding the outlet flow monitoring program. The information required includes (1) the general location and type of flow measurement device(s), and (2) the observation plan and alarm procedure to identify unanticipated high or low flow in the system and the condition of the effluent.

(l) For OL reviews, but only if not previously reviewed by the staff, provide (1) substantiation of assumed design bases using information gathered during dewatering for construction excavation, and (2) all other details of the dewatering system design that implement design bases established during the CP review.

(m) For OL reviews, provide a Technical Specification for periods when the dewatering system may be exposed to sources of water not considered in the design. An example of such a situation would be the excavation of surface seal material for repair of

piping such that the underdrain would be exposed to direct surface runoff. In addition, where the permanent dewatering system is safety related, is not completely redundant, or is not designed for all design basis events, provide the bases for a technical specification with action levels, the remedial work required and the estimated time that it will take to accomplish the work, the sources, types of equipment and manpower required and the availability of the above under potentially adverse conditions. [See Section V(f)].

U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

SECTION 2.4.14 TECHNICAL SPECIFICATIONS AND EMERGENCY OPERATION REQUIREMENTS

REVIEW RESPONSIBILITIES

Primary - Hydrology-Meteorology Branch (HMB)

Secondary - None

I. AREAS OF REVIEW

The purpose of this section of the applicant's safety analysis report (SAR) is to identify the technical specifications and emergency procedures required to implement flood protection for safety-related facilities and to assure an adequate water supply for shutdown and cooldown purposes.

If there is evidence of potential structural effects, the Structural Engineering Branch (SEB) will be requested by HMB to ascertain whether these effects are properly considered in the structural design bases for the plant; similarly, ASB will be requested by HMB to ascertain whether these effects are properly considered in the systems design bases for the plant. Guidance for determining whether these potential effects are considered properly is outlined in the appropriate SEB and ASB SRP sections.

II. ACCEPTANCE CRITERIA

If the hydrologic design bases (discussed in Regulatory Guide 1.59) developed in preceding sections do not necessitate technical specifications or emergency procedures to ensure safety-related plant functions, this section should so state. It technical specifications or emergency procedures are necessary this section will be acceptable if the following are identified:

1. The controlling hydrologic events, as developed in the preceding sections of SAR Chapter 2.

2. The actions to be taken, and the effect of such actions on the protection of safety-related facilities and water supplies.

3. The appropriate water levels and conditions at which action is to be initiated.

4. The appropriate emergency procedures, and the amount of time required to implement each procedure.

III. REVIEW PROCEDURES

The review procedures consist of comparing the proposed specifications and procedures with the flood protection and water supply design bases derived in the preceding sections, or considered necessary by the staff. Data in, or derived from, the preceding sections are used to estimate the time available to complete any required emergency action (e.g., sandbagging, shutdown, installing flood gates and stop logs). This information will also serve to substantiate the water levels and other conditions used to initiate the action. Specific questions on the structural adequacy of protective measures are referred to Structural Engineering Branch, and the general experience of the Corps of Engineers in such situations, as reflected in reports and manuals, is the principal basis for comparison. Issues involving shutdown water supplies should be coordinated with Auxiliary Systems Branch.

IV. EVALUATION FINDINGS

For both construction permit and operating license reviews the findings will consist of a brief statement of technical specifications and emergency procedures and time required to implement flood protection of safety-related facilities and assure an adequate water supply for safety-related equipment. The flood or water levels and other conditions at which action is to be initiated will also be stated. If none are required, the findings will so state.

A sample Operating License statement follows:

> "The staff has taken a position that it would be prudent to shut the plant down before water could reach plant grade during severe hurricanes. The applicant has maintained that design of the safety-related facilities includes provision for protection. The staff believes the implementation of emergency procedures, required in the event of severe hurricanes to assure the watertightness of exterior doors and to minimize the possible equipment failure which could occur during such an event (should the applicant's single water barrier design provisions not be adequate), would be extremely difficult from a practical standpoint. The staff, therefore, will require a provision in the plant's Technical Specifications requiring a flood alert, referring to emergency procedures, when water levels exceed elevation 15 feet MSL. In the case of PMH, this would allow a minimum of about four hours before water would cross plant grade (some six hours before maximum water levels would be reached) to implement emergency action. Examples of required action are: assuring all exterior accesses are closed and sealed, adequate diesel fuel oil supplies are protected, sandbagging of vulnerable areas may be undertaken, and any necessary emergency equipment is available and operational. The weather conditions during such a situation would be severe (high winds, rain, the likelihood of tornadoes in the area, etc.), but implementation of outdoor emergency procedures is considered reasonable if accomplished before maximum storm conditions occur.

"The applicant has installed a control room water level alarm that is activated when the water level in the intake canal reaches elevation 17.5 feet MSL. The staff will require the same technical specification to necessitate an orderly plant shutdown upon activation of the alarm. The requirement is prudent in view of the single line of defense inherent in the water barriers installed by the applicant. Failure of such barriers with the reactor at or near operating levels would allow a very limited time, during extreme weather conditions, for plant operating personnel to prevent a major accident. No other technical specification provisions are considered necessary for hydrologically-related events."

V. REFERENCES

Data and information presented in, or derived from, previous SRP sections in the 2.4 series provide the basic reference material for this section.

1. Regulatory Guide 1.59, "Design Basis Floods for Nuclear Power Plants."

2. ANSI N170, "Standards for Determining Design Basis Flooding at Power Reactor Sites" (1976).

3. Regulatory Guide 1.127, "Inspection of Water-Control Structures Associated with Nuclear Power Plants."

4. Regulatory Guide 1.102, "Flood Protection for Nuclear Power Plants."

U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

SECTION 2.5.1 BASIC GEOLOGIC AND SEISMIC INFORMATION

REVIEW RESPONSIBILITIES

Primary - Geosciences Branch (GB)

Secondary - None

I. AREAS OF REVIEW

GB reviews the geologic and seismic information submitted in the applicant's safety analysis report (SAR) in accordance with Appendix A to 10 CFR Part 100, "Seismic and Geologic Siting Criteria for Nuclear Power Plants." GB judges the adequacy of the geologic and seismic information cited in support of the applicant's conclusions concerning the suitability of the plant site. The geologic and seismic information which must be provided in order for the site review to proceed is divided into the following three categories:

1. Geologic features: mass-wasting, differential subsidence, faulting, soil and foundation instability, chemical weathering, cavernous or karst terrains, evidence of preconsolidation via erosional processes and volcanism.

2. Seismic features: ground failure under dynamic loading, liquefaction, vibratory ground motion, tsunami, and residual stresses.

3. Man-made conditions: changes in groundwater conditions, subsidence or collapse caused by withdrawal of fluids or mineral extraction, induced seismicity and fault movement caused by fluid injection or withdrawal.

Information relating to the above conditions as presented in SAR Sections 2.5.1.1 (Regional Geology) and 2.5.1.2 (Site Geology), should be reviewed in terms of the regional and site physiography, geomorphology, stratigraphy, lithology, and tectonics. In addition, with specific reference to site geology, the following subjects should be reviewed as they relate to the above-mentioned conditions: topography, slope stability, fluid injection or withdrawal, mineral extraction, faulting, shearing, jointing, seismicity and fracturing.

The above information should be documented by appropriate references to all relevant published and unpublished materials. Illustration should include but should not be

limited to physiographic, topographic, geologic, tectonic, gravity, and magnetic maps, structure and stratigraphic sections, boring logs, and aerial photographs. Certain sites will require illustrations of specialized character such as maps of subsidence, irregular weathering conditions, landslide potential, hydrocarbon extraction (oil or gas wells), faults, joints, and karst features. Some site characteristics must be documented by reference to seismic reflection or refraction profiles or to maps produced by various remote sensing techniques.

As appropriate, maps should include a superimposed plot plan of the plant facilities. Other documentation should show the relationship of all seismic Category I facilities (clearly identified) to subsurface geology. Core boring logs, logs and maps of trenches, aerial and Environmental Resources Technology Satellite (ERTS) photographs, and geophysical data should be presented for evaluation. In addition, a plot plan showing the locations of all structures, borings, trenches, profiles, etc. should be included.

The review can be brought to an earlier conclusion if the following suggestions are followed by the applicant. The SAR should contain sufficient data to allow the reviewer to make an independent assessment of the applicant's conclusions. That is, the reviewer should be led in a logical manner from the data and premises given to the conclusions that are drawn without having to make an extensive independent literature search. Controversial information should not be ignored so as to enhance a particular position. The geologic terminology used should conform to standard reference works (Refs. 3, 6). Finally, the objective of Section 2.5 of the SAR is to describe geologic and seismic features as they affect the site under review, and all data, information, discussions, interpretations, and conclusions should be directed to this objective. Aimless presentation of data, although it may appear to satisfy the investigative requirements, will result in a disjointed SAR and cause needless delays in completing the safety review.

II. ACCEPTANCE CRITERIA

The "Seismic and Geologic Criteria for Nuclear Power Plants" (Ref. 1) and the Standard Format (Ref. 2) are the basis for the staff review of all cases. The information presented in the SAR must be complete and thoroughly documented, and must be consistent with the requirements of References 1 and 2. United States Geological Survey (USGS) and other Federal or State agency published and open file papers, maps, aerial photographs, geophysical data, etc., covering the region in which the site is located, are used to establish the staff's conclusions as to the completeness and acceptability of the SAR.

Subsection 2.5.1.1, "Regional Geology," will be considered acceptable if a complete and documented discussion is presented of all geologic, seismic, and man-made features. This section should contain a review of the regional physiography, geomorphology, stratigraphy, structure, and geologic history to provide a framework within which the geologic, seismic and man-made features of safety significance to the site can be evaluated.

Subsection 2.5.1.2, "Site Geology," will be judged acceptable if it contains a description and evaluation of site-related geologic features, seismic conditions, and man-made conditions which are a potential hazard to the site. This section should also contain the following general site information:

1. The site stratigraphy, including relationship to and correlation with the regional stratigraphy.

2. The structural geology of the site and the relationship of site structure to regional tectonics.

3. The geologic history of the site as it relates to the regional geologic history.

4. The engineering significance of geologic features underlying the site as they relate to:

   a. Dynamic behavior during prior earthquakes.

   b. Zones of alteration, irregular weathering, or zones of structural weakness.

   c. Unrelieved residual stresses in bedrock.

   d. Materials that could be unstable because of their mineralogy or unstable physical properties.

   e. Effects of man's activities in the area.

5. The site groundwater conditions.

III. REVIEW PROCEDURES

The staff review is conducted in three phases. The first phase is the acceptance review, a brief review of the SAR to evaluate its completeness and to identify obvious safety issues that could result in delays at subsequent stages of the review. After an SAR is docketed, the staff conducts a thorough review of the material. In this second phase of the review an effort is made to identify all safety issues. The reviewer should carefully examine the SAR to see that all interpretations are founded on sound geological and seismological practice and do not exceed the limits of validity of the applicant's data or of other data published in the literature. The questions and comments transmitted to the applicant will identify issues that have not been addressed, areas where staff interpretations differ from those given in the SAR, and issues that have not been sufficiently documented to permit the staff to concur in the conclusions reached by the applicant. When possible, the staff should take positions on safety-related issues at this point. The third review phase is the staff evaluation of the applicant's responses to questions raised in the second phase. At the end of the third phase, the staff takes positions on all safety-related issues, either concurring with

the applicant's positions or taking more conservative positions as may be necessary in the staff's view to assure the required degree of safety.

Pertinent references, such as published geological reports, professional papers, open file material, university theses, physiographic and geological maps, and aeromagnetic and gravity maps, are ordered from the appropriate sources and reviewed. The general references used extensively by the staff are References 3 and 4. The GEO-Reference File (Ref. 5) is used to identify specific references.

The judgments on acceptance or rejection of the SAR are governed by two criteria: (1) adherence to the Standard Format in identifying and describing the geologic, seismic and man-made features that affect safety of the site; and (2) provision of adequate information and documentation to allow for an independent review of the conclusions made therein.

During the acceptance review the staff decides to what extent consultants such as the USGS, the Corps of Engineers, State geological survey organizations, or other specialists should be involved. The necessary information is then made available to these consultants. Consultants are asked to handle such varied tasks as reviewing the geotechnical engineering aspects of plants located at sites with complex foundation conditions, verifying an applicant's mineral identifications, or evaluating the suitability of foundations with respect to bearing capacity and settlement and evaluation of slope stability conditions for safety-related slopes, dams and dikes.

After docketing, a detailed review of the SAR and relevant references is conducted by the staff and its advisors. Questions and comments are developed from items that have not been adequately addressed by the applicant, those which become apparent during the detailed review, or those which develop from the additional information provided as a result of the acceptance review. These questions (Q-1) usually require the applicant to conduct additional investigations or to supply clarifying information. Many questions result from the reviewer's discovery of references not cited by the applicant that contain conclusions which are in conflict with those made by the applicant. When the applicant provides insufficient data to support his interpretations and conclusions, and there are reasonable and more conservative alternative interpretations in the literature, the staff will request additional investigations. This phase of the review will usually involve meetings with the applicant to clarify questions and allow him to present new data. In addition, during the Q-1 phase, the staff visits the site.

The applicant's responses to Q-1 are reviewed and any remaining issues are settled either by additional questions or by staff positions. A staff position is usually in the form of a requirement to design for a specific condition in a way which the staff considers to be sufficiently conservative and consistent with the requisites of Reference 1. When all safety issues have been resolved, the staff provides its input to the safety evaluation report (SER).

IV. EVALUATION FINDINGS

The staff's findings for construction permit (CP) reviews will consist of a report summarizing the geology at the site and the pertinent design aspects of the plant. All geologic features that may potentially affect the safety of the plant will be identified, described, and measures taken to deal with them will be given. The seismic design basis will be described.

Operating license (OL) applications are reviewed for any new information developed subsequent to the CP. The review will also determine whether the CP recommendations have been implemented.

A typical CP-stage finding for this section of the SER follows:

> "Based on our review of the PSAR materials and our independent review of the relevant published literature, we have concluded that the site is located in the Piedmont tectonic province. The last recognizable regional tectonic event occurred here in Triassic to Jurassic time (225 - 136 mybp). No Holocene faulting of tectonic origin is known in the province and no capable faults within the meaning of Appendix A to 10 CFR Part 100 have been recognized."

V. REFERENCES

1. 10 CFR Part 100, Appendix A, "Seismic and Geologic Siting Criteria for Nuclear Power Plants."

2. Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power PLants," Revision 2.

3. M. Gray, R. McAfee, Jr., and C. L. Wolf, eds., "Glossary of Geology," American Geological Institute, Washington (1972).

4. G. V. Cohee (chairman) et al., "Tectonic Map of the United States," U. S. Geological Survey and American Association of Petroleum Geologists (1962).

5. "Geo-Reference: Computerized File of Earth Science Titles," American Geological Institute, Washington.

6. M. W. Higgins, "Cataclastic Rocks," Professional Paper 687, U. S. Geological Survey (1971). (Includes extensive discussion of the terminology of cataclastic rocks.)

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SECTION 2.5.2 VIBRATORY GROUND MOTION

REVIEW RESPONSIBILITIES

Primary - Site Analysis Branch (SAB)

Secondary - None

I. AREAS OF REVIEW

The SAB review covers the seismological and geological investigations carried out to establish the acceleration for seismic design of the plant, the procedures and analyses used by the applicant to determine the safe shutdown earthquake (SSE) and the operating basis earthquake (OBE) for the site, and the seismic design bases for foundations.

Specific areas of review include; seismicity, relationship of earthquake occurrence to geologic or tectonic characteristics of the region, determination of the earthquake-generating potential of the geologic structures and tectonic provinces in the region, characteristics of seismic wave transmission at the site, and determination of the level and properties of the vibratory ground motion at the site resulting from potential earthquakes in the region.

II. ACCEPTANCE CRITERIA

1. The required investigations are described in 10 CFR Part 100, Section IV(a) of Appendix A. The acceptable procedures for determining the seismic design bases are given in Section V(a) of the same appendix. The seismic design bases are predicated on a reasonable, conservative determination of the safe shutdown earthquake and the operating basis earthquake. As defined in Section III of 10 CFR Part 100, Appendix A, the SSE and OBE are based on consideration of the regional and local geology and seismology and on the characteristics of the subsurface materials at the site and are described in terms of the vibratory ground motion which they would produce at the site. No comprehensive definitive rules can be promulgated regarding the investigations needed to establish the seismic design bases; the requirements vary from site to site.

2. Subsection 2.5.2.1 (Seismicity): The applicant's presentation is accepted when the complete historical record of earthquakes in the region is listed and when all available parameters are given for each earthquake in the historical record. The listing should include all earthquakes MM intensity greater than IV or magnitude greater than 3 which have been reported in all tectonic provinces any parts of which are within 200 miles of the site. A regional-scale map should be presented showing all listed earthquake

epicenters and, in areas of high seismicity, should be supplemented by a larger-scale map showing earthquake epicenters within 50 miles of the site. The following information concerning each earthquake is required whenever it is available: epicenter coordinates, depth of focus, origin time, highest intensity, magnitude, seismic moment, source mechanism, source dimensions, source rise time, rupture velocity, total dislocation, fractional stress drop, and any strong-motion recordings; references from which the specified information was obtained should be identified. In addition, any reported earthquake-induced geologic failure, such as liquefaction, landsliding, landspreading, and lurching should be described completely, including the level of strong motion which induced failure and the material properties of the materials. The completeness of the earthquake history of the region is determined by comparison to the historical earthquake data (HED) file (Ref. 4) and other published sources of information (e.g., Refs. 5, 6, 7). When conflicting descriptions of individual earthquakes are found in the published references, a reasonable description which results in the more conservative interpretation of the seismicity is accepted.

3. Subsection 2.5.2.2 (Geologic and Tectonic Characteristics of Site and Region): The applicant's presentation is accepted when all regional geologic structures and tectonic activity which are significant in determining the earthquake potential of the region are identified. Information presented in Section 2.5.1 of the applicant's safety analysis report (SAR) and information from other literature sources (e.g., Refs. 8, 9, 10, 11, 12) dealing with regional tectonics should be developed into a coherent, well-documented discussion to be used as the basis for determining tectonic provinces and the earthquake-generating potential of the identified geologic structures. Specifically, each tectonic province, any part of which is within 200 miles of the site, must be identified. Those characteristics of geologic structure, tectonic history, present and past stress regimes, and seismicity which distinguish the various tectonic provinces and the particular areas within those provinces where historical earthquakes have occurred should be described. Alternative regional tectonic models from available literature sources should be discussed. When several of the alternative models conform equally well with the observed phenomena, the model which results in the more conservative assessment of the earthquake potential at the site is accepted. In addition, in those areas where there are capable faults, the results of the additional investigative requirements described in 10 CFR Part 100, Appendix A, Section IV(a)(8), must be presented. The discussion should be augmented by a regional-scale map showing the tectonic provinces, earthquake epicenters, locations of geologic structures and other features which characterize the provinces, and the locations of any capable faults.

4. Subsection 2.5.2.3 (Correlation of Earthquake Activity with Geologic Structure or Tectonic Provinces): Acceptance is based on the development of the relationship between the relatively short history of earthquake activity and the geologic structures or tectonic provinces of a region. The applicant's presentation is accepted when the earthquakes discussed in Subsection 2.5.2.1 of the SAR are shown to be associated with either geologic structure or a tectonic province. Whenever an earthquake epicenter or concentration of earthquake epicenters can be reasonably correlated with geologic

structure, the rationale for the association should be developed considering the properties of the geologic structure and the regional tectonic model. The discussion should include identification of the methods used to locate the earthquake epicenters, an estimate of their accuracy, and a detailed account which compares and contrasts the geologic structure involved in the earthquake activity with other areas within the tectonic province. Particular attention should be given to determining the capability of faults with which instrumentally-located earthquake epicenters are associated.

The applicant may choose to define tectonic provinces to correspond to subdivisions generally accepted in the literature. A subdivision of a tectonic province is accepted if it can be corroborated on the basis of detailed seismicity studies, tectonic flux measurements, contrasting structural fabric, different geologic history, differences in stress regime, etc. If detailed investigations reveal no significant differences between areas within a tectonic province, the areas should be considered to compose a single tectonic province. The presentation should be augmented by a regional-scale map showing the tectonic provinces, the earthquake epicenters, and the locations of geologic structures and measurements used to define provinces. Acceptance of the proposed tectonic provinces is based on the staff's independent review of the seismicity, tectonic flux (Ref. 31), geologic structure, and stress regime in the region of the site.

5. Subsection 2.5.2.4 (Maximum Earthquake Potential): The applicant's presentation is accepted when the vibratory ground motion due to the maximum credible earthquake associated with each geologic structure or the maximum historic earthquake associated with each tectonic province has been assessed and when the earthquake which would produce the maximum vibratory ground motion at the site has been determined. Earthquakes associated with each geologic structure or tectonic province must be identified. Where an earthquake is associated with geologic structure, the maximum earthquake which could occur on that structure should be evaluated, taking into account such factors as the type of the faulting, fault length, fault displacement, and earthquake history, (e.g., Refs. 14, 15).

In order to determine the maximum earthquake that could occur on those faults which are shown or assumed to be capable, the staff accepts conservative values based on historic experience in the region and specific considerations of the earthquake history, sense of movement, and geologic history of movement on the faults. Where the earthquakes are associated with a tectonic province, the largest historical earthquake within the province should be identified and, whenever possible, the return period for the earthquake should be estimated. Isoseismal maps should also be presented for the most significant earthquakes. The ground motion at the site should be evaluated assuming seismic energy transmission effects are constant over the region of the site and assuming that the largest earthquake associated with each geologic structure or with each tectonic province occurs at the point of closest approach of that structure or province to the site.

The set of conditions describing the occurrence of the earthquake which would produce the largest vibratory ground motion at the site should be defined. If different potential earthquakes would produce the maximum ground motion in different frequency bands, the conditions describing all such earthquakes should be specified. The description of the potential earthquake occurrence is to include the maximum intensity or magnitude and the distance from the assumed location of the potential earthquake to the site. The staff independently evaluates the effects on site ground motion of the largest earthquake associated with each geologic structure or tectonic province. Acceptance of the description of the potential earthquake which would produce the largest ground motion at the site is based on the staff's independent analysis.

6. Subsection 2.5.2.5 (Seismic Wave Transmission Characteristics of the Site):
The applicant's presentation is accepted when the seismic wave transmission characteristics (amplification or deamplification) of the materials overlying bedrock at the site are described as a function of the significant frequencies. The following material properties should be determined for each stratum under the site: seismic compressional and shear velocities, bulk densities, soil properties and classification, shear modulus and its variation with strain level, and water table elevation and its variation. In each case, methods used to determine the properties should be described or a cross-reference should be given indicating where in the SAR the description is provided. For each set of conditions describing the occurrence of the maximum potential earthquake, determined in Subsection 2.5.2.4, the type of seismic waves producing the maximum ground motion and the significant frequencies must be determined. For each set of conditions an analysis should be performed to determine the effects of transmission in the site material for the identified seismic wave types in the significant frequency bands.

Where horizontal shear waves produce the maximum ground motion, an analysis similar to that of Schnabel, et al. (Ref. 16) is appropriate. Where compressional or surface waves produce the maximum ground motion, other methods of analysis (Refs. 17, 18) may be more appropriate. However, since the latter techniques are still in the developmental stages and no generally agreed-on procedures can be promulgated at this time, the staff accepts the shear wave model as representative of site amplification. The site amplification determined in this way should be compared with characteristics of site amplification in the epicentral area of the historical earthquake used as the basis for each maximum potential earthquake. If detailed soils investigations have been made in the epicentral area, the amplification analysis should be based on these. Because detailed geologic investigations are generally not available for the epicentral areas of historical earthquakes, several factors should be considered in assessing amplification effects there, including: regional geology and soil conditions, earthquake isoseismal maps, and descriptions of earthquake effects.

7. Subsection 2.5.2.6 (Safe Shutdown Earthquake): The applicant's presentation is accepted when the vibratory ground motion specified for the safe shutdown earthquake is described in terms of the level of acceleration for seismic design and its time history and is as conservative as that which would result at the site from the maximum potential earthquake (determined in Subsection 2.5.2.4) and considering the variations in site transmission effects (determined in Subsection 2.5.2.5). If several different maximum potential earthquakes produce the largest ground motions in different frequency bands (as noted in Subsection 2.5.2.4), the vibratory ground motion specified for the SSE must be as conservative in each frequency band as that for each earthquake, including site transmission effects (as noted in Subsection 2.5.2.5).

The amplitude of acceleration at the ground surface, the effective frequency range, and the duration corresponding to each maximum potential earthquake must be identified. The acceleration is to be expressed as a fraction of the acceleration of gravity (g). Where the earthquake has been associated with a specific geologic structure, the acceleration should be determined using a relation between acceleration, magnitude or fault length and distance from the fault (cf. Refs. 13, 15). Where the earthquake has been associated with a tectonic province, the acceleration should be determined using appropriate relations between acceleration, intensity, epicentral intensity, and distance (e.g., Refs. 19, 20, 21, 24).

Numerous correlations between intensity and acceleration are given in the literature (Refs. 19, 20, 21, 22, 23); several of them are considered acceptable by the staff. The correlation used is accepted if it is conservative when compared to the actual observational data. Acceptance is based on an analysis of the site's seismic energy transmission properties (Ref. 16, or equivalent). Conservatism should be assessed based on consideration of the amplification analysis and in comparison with the actual published data. The staff will generally accept an acceleration for seismic design as being conservative if, when applied at the ground surface, it results in a value at the foundation free field level as large as would be obtained from the empirical relation of the mean of the intensity acceleration values in Reference 23.

Available ground motion time histories for earthquakes of comparable values of magnitude, epicentral distance, and acceleration level should be presented. The spectral content for each potential maximum earthquake should be described; it should be based on consideration of the available ground motion time histories and regional characteristics of seismic wave transmission. The dominant frequency associated with the peak acceleration should be determined either from analysis of ground motion time histories or by inference from descriptions of earthquake phenomenology, damage reports, and regional characteristics of seismic wave transmission.

In some cases, the peak acceleration may not be as significant for engineering design purposes as a sustained acceleration at a lower level. One situation where the sustained acceleration level may differ from the peak acceleration is in proximity to the causative fault of the earthquake. It is appropriate in such cases to define the

"reference acceleration for seismic design" as representative of the level of sustained acceleration. The "reference acceleration for seismic design" determined in this section of the applicant's SAR is taken to be the high frequency asymptote to the design response spectrum defined in Reference 2. At this time, the staff is not aware of any published relations between earthquake intensity or magnitude and sustained acceleration. Such relations could be developed from analyses of the response spectra of accelerograph time histories in those areas where magnitude and intensity measurements are also available. In lieu of such studies, the peak accelerations are considered to represent conservative reference accelerations for seismic design. Lower levels of reference acceleration may be justified on a site-specific basis.

The staff's review of proposed reference accelerations for seismic design considers: the proximity of the site to the geologic structure or province with which the potential earthquake is associated, characteristics of acceleration time histories at epicentral distances similar to that of the potential SSE, results of time-dependent spectral analyses of such time histories (cf. Refs. 25, 26), the level and dominant frequency of the peak acceleration, and seismic wave amplitude attenuation as a result of transmission from the source to the site and in the material underlying the site.

The design response spectrum is reviewed under Standard Review Plan (SRP) 3.7.1; however, as noted above there are certain seismological conditions which may require special modifications of the response spectrum. In general, the design response spectrum is acceptable if it is as conservative as the response spectrum from each of the potential earthquakes as described above.

The time duration of strong ground motion is required for analysis of site foundation liquefaction potential and for design of many plant components. The adequacy of the time history for structural analysis is reviewed under SRP 3.7.1. The time history is reviewed in this standard review plan to confirm that it is compatible with the seismological and geological conditions in the site vicinity and with the accepted SSE model. At present, there is no truly adequate model for deterministically computing the time history of strong ground motion from a given source-site configuration. It is, therefore, acceptable to generate the time history record from the design response spectrum for the SSE using the method of Tsai (Ref. 27) or an equivalent method. Total duration of the motion is acceptable when (1) it is as conservative as values determined using the procedure described by Bolt (Ref. 28) for hard rock sites or for analyses where nonstationarity of strong motion time functions is unimportant* and (2) the spectrum of the derived accelerogram is found acceptable in the review under SRP 3.7.1.

8. Subsection 2.5.2.7 (Operating Basis Earthquake): The vibratory ground motion for the OBE should be described with the SSE and the acceleration level at the site specified. The minimum value of the acceleration level for the OBE is currently one-half the reference acceleration for seismic design corresponding to the SSE. For sites in highly seismic regions, mainly in the western United States, the complete description of the OBE, as given in 10 CFR Part 100, Appendix A, Section III(d),

*For sites on sediments or for analyses where nonstationarity is important, more conservative values may be required. See, e.g., Refs. 24 and 30.

is required. In some cases, probability calculations, like those described by Algermissen (Ref. 29), would be helpful in estimating the acceleration level reasonably expected to affect the plant site during the operating life of the plant. Acceptable source regions that can be used as input to these calculations are those geologic structures or tectonic provinces with which historical earthquake activity has been associated. Such descriptions should include the acceleration level of the OBE and a determination of the probability of exceeding that level during the 40-year operating life of the plant.

III. REVIEW PROCEDURES

1. Upon receiving the applicant's SAR, an acceptance review is conducted to determine: compliance with the investigative requirements of 10 CFR Part 100, Appendix A, and conformance with the Standard Format (Regulatory Guide 1.70). The reviewer also identifies any site-specific problems, the resolution of which could result in extended delays in completing the review.

2. After SAR acceptance and docketing, those areas are identified where additional information is required to determine the earthquake hazard and to establish the design acceleration. These are transmitted to the applicant in requests for additional information (Q-1).

3. A site visit is conducted during which the reviewer inspects the foundation conditions, local faulting, and other geologic conditions. During the site visit the reviewer also discusses and clarifies the Q-1 questions with the applicant and his consultants so that it is clearly understood what additional information is required by the staff to continue the review.

4. Following the site visit a revised set of requests for additional information (Q-2), including any additional questions which may have been developed during the site visit, is formally transmitted to the applicant. At the Q-2 stage the review procedure consists mainly of an evaluation of the applicant's response to the Q-1 questions. The reviewer prepares requests for additional clarifying information and formulates positions which may agree or disagree with those of the applicant. These are formally transmitted to the applicant.

5. The safety analysis report and supplements responding to the requests for additional information (Q-1, Q-2) are reviewed to determine that the information presented by the applicant is acceptable according to the critieria described in Section II above. Based on information supplied by the applicant, obtained from site visits, or from staff consultants or literature sources, the reviewer independently identifies the relevant seismotectonic provinces, evaluates the capability of faults in the region, and determines the earthquake potential for each province and each capable fault using procedures noted in Section II above. The reviewer evaluates the vibratory ground motion which the potential earthquakes could produce at the site and defines the safe shutdown earthquake and operating basis earthquake.

IV. EVALUATION FINDINGS

For construction permit (CP) reviews, the findings are included in the staff's safety evaluation report and consist of statements (including or referencing diagrams, maps, etc.) describing the applicant's and the staff's (1) definitions of seismotectonic provinces; (2) evaluations of the capability of geologic structures in the region; (3) determinations of the SSE acceleration at ground surface, reference acceleration for seismic design, time duration of strong ground motion, and any alterations in the design response spectrum based on evaluation of the potential earthquakes; and (4) determinations of the OBE acceleration at ground surface. If the staff's findings are consistent with those of the applicant, staff concurrence is stated; otherwise, a statement requiring use of the staff's findings is made.

For operating license (OL) reviews, the staff's positions from the CP review are referenced and a detailed review of any new data which might affect the seismic design bases is presented.

V. REFERENCES

1. 10 CFR Part 100, Appendix A, "Seismic and Geologic Siting Criteria for Nuclear Power Plants."

2. Regulatory Guide 1.60, "Design Response Spectra for Seismic Design of Nuclear Power Plants," Revision 1.

3. Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," Revision 2.

4. "Historical Earthquake Data File," National Geophysical and Solar-Terrestrial Data Center, National Oceanic and Atmospheric Administration.

5. "Earthquake History of the United States," Publication 41-1, National Oceanic and Atmospheric Administration, U. S. Department of Commerce (1973).

6. S. D. Townley and M. W. Allen, "Description Catalog of Earthquakes of the Pacific Coast of the United States, 1769 to 1928," Bulletin Seismological Society of America, Vol. 29 (1939).

7. W. E. T. Smith, "Earthquakes of Eastern Canada and Adjacent Areas," Publications of the Dominion Observatory (1962).

8. P. B. King, "The Tectonics of North America - A Discussion to Accompany the Tectonic Map of North America Scale 1:5,000,000," Professional Paper 628, U. S. Geological Survey (1969).

9. A. J. Eardley, "Tectonic Divisions of North America," Bulletin American Association of Petroleum Geologist, Vol. 35 (1951).

10. J. B. Hadley and J. F. Devine, "Seismotectonic Map of the Eastern United States," Publication MF-620, U. S. Geological Survey.

11. M. L. Sbar and L. R. Sykes, "Contemporary Compressive Stress and Seismicity in Eastern North America: An Example of Intra-Plate Tectonics," Bulletin Geological Society of America, Vol. 84 (1973).

12. R. B. Smith and M. L. Sbar, "Contemporary Tectonics and Seismicity of the Western United States with Emphasis on the Intermountain Seismic Belt," Bulletin Geological Society of America, Vol. 85 (1974).

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SECTION 2.5.3 SURFACE FAULTING

REVIEW RESPONSIBILITIES

Primary - Geosciences Branch (GB)

Secondary - None

I. AREAS OF REVIEW

GB reviews information in the applicant's safety analysis report (SAR) related to the existence of a potential for surface faulting affecting the site. The information presented in this section results largely from detailed surface and subsurface geological and geophysical investigations performed in the site and vicinity. The following specific subjects are addressed: the structural and stratigraphic conditions of the site and vicinity (Subsection 2.5.3.1), any evidence of fault offset or evidence demonstrating the absence of faulting (Subsection 2.5.3.2), earthquakes associated with faults (Subsection 2.5.3.3), determination of age of most recent movement on faults (Subsection 2.5.3.4), determination of structural relationships of site area faults to regional faults (Subsection 2.5.3.5), identification and description of capable faults (Subsection 2.5.3.6), and zones requiring detailed fault investigations (Subsection 2.5.3.7).

II. ACCEPTANCE CRITERIA

The data and analyses presented in the SAR are acceptable if, as a minimum, they describe and document the information required by References 1 and 2, and other data that are necessary, depending on the complexity of the site. The GEO-Reference File (Ref. 3) is used by the staff as the principal reference guide to judge whether or not all of the pertinent references have been consulted. References 4 through 9 are also used by the staff.

Subsection 2.5.3.1 is considered acceptable if the discussions of the stratigraphy, methods of fault dating, structural geology, and geologic history of the site are complete, compare well with studies conducted by others in the same area, and are supported by detailed investigations performed by the applicant. Site and regional geologic maps and profiles constructed at scales adequate to illustrate clearly the surficial and bedrock geology, structural geology topography, and the relationship of the safety-related foundations of the nuclear power plant to these features should be included in the SAR.

Subsection 2.5.3.2 is acceptable if sufficient surface and subsurface information is provided and supported by detailed investigations, either to confirm the absence of

faulting or, if faulting is present, to demonstrate its age. If faulting is present in the site vicinity, it must be defined as to fault geometry, amount and sense of movement, and age of latest movement. In addition to geologic evidence which may indicate faulting, linears interpreted from topographic maps, low-altitude aerial photographs and Environmental Resource Technology Satellite imagery should be documented and investigated. Evidence for absence of faulting is obtained by conducting site surface and subsurface investigations in such detail and areal extent to ensure that undetected offsets are not likely to exist. These investigations will vary in detail according to the geological complexity of the specific site.

Subsection 2.5.3.3 is acceptable if all historically reported earthquakes within five miles of the site or near faults which trend within five miles of the site, as discussed in Section 2.5.2, are evaluated with respect to hypocenter accuracy and source origin. In conjunction with these discussions, a plot of the earthquake epicenters superimposed on a map showing the local tectonic structures as defined in Section 2.5.1 should be provided. Estimated error regions of the earthquake epicenters should be shown.

Subsection 2.5.3.4 is acceptable when every fault, any part of which is within five miles of the site, is investigated in sufficient detail using geological and geophysical techniques of sufficient sensitivity to demonstrate the age of most recent movement. An evaluation of the sensitivity and resolution of the exploratory techniques used should be given.

Subsection 2.5.3.5 is acceptable when a discussion is given of the structural and genetic relationship between site area faulting and the regional tectonic framework. In regions of active tectonism it may be necessary to conduct detailed geological and geophysical investigations to demonstrate the structural relationships of site area faults to regional faults known to be seismically active. Both a theoretical and an observational basis for the conclusions reached should be given.

Subsection 2.5.3.6 is acceptable when it has been demonstrated that the investigative techniques used have sufficient sensitivity to identify all faults greater than 1000 feet in length within five miles of the site and when the geometry, sense of movement, and amount of offset is given for each.

Subsection 2.5.3.7 is judged acceptable if the zone designated by the applicant as requiring detailed faulting investigation is consistent with the description of such a zone in Reference 1.

Subsection 2.5.3.8 must be presented by the applicant if the aforementioned investigations reveal that surface displacement must be taken into account. No nuclear plant has ever been constructed on a capable fault and it is an open question as to whether it is possible to design for surface or near-surface displacement with confidence that the integrity of the safety-related features of the plant would remain intact should displacement occur. It is, therefore, staff policy to recommend relocation of plant sites found

to be located on capable faults as determined by the detailed faulting investigation. If in the future it becomes possible to design for surface faulting, it will be necessary to present the design basis for surface faulting and supporting data in considerable detail.

III. REVIEW PROCEDURES

The staff review procedure involves an evaluation to determine that the applicant has followed the investigations outlined in Reference 1. The U.S. Geological Survey (USGS) acts as staff advisor in reviewing this section of the SAR, on a case-by-case basis. On request, the USGS provides expertise in numerous earth science disciplines and often is able to provide first-hand knowledge of the site. A literature search is conducted concerning the regional and local geology. The staff also contacts State geological surveys and universities to obtain additional data.

Generally, the steps that applicants must follow in determining the presence and extent of faulting, and whether near-surface faulting (if present) represents a hazard or not, is outlined in the seismic and geologic siting criteria (Ref. 1). Specific investigative techniques are not given in the criteria, however. The site area must be investigated by a combination of exploratory methods which may include borings, trenching, seismic profiling, geologic mapping, and geophysical investigations. The results of these explorations are cross-compared and evaluated by the staff.

It has been the policy of the staff to encourage applicants to avoid areas where there is a possibility for surface faulting. As the question of whether or not a surface faulting condition exists is so critical in determining whether a particular site is suitable, this consideration is usually addressed very early in the review. Exceptions are those cases in which a fault, the existence of which was previously unknown, is revealed in excavations during construction or is discovered during the course of other investigations in the area.

When faults are identified in the site vicinity, it must be demonstrated that the faults are not capable. This is accomplished by determining the ages of the faults by absolute age dating (radiometric), associating the faulting with regional tectonic activity of known age, stratigraphic or geomorphic evidence, etc. In such cases the staff will carry out limited site observations and investigations of its own such as examinations of excavations, and selecting and dating samples taken from shear zones. Applicants are usually required to trench in the areas where major facilities are to be located.

Subsection 2.5.3.1 is evaluated by conducting an independent literature search and cross-comparing the results with the information submitted in the SAR. The comparison should show that the conclusions presented by the applicant are based on sound data, are consistent with the published reports of experts who have worked in the area, and are consistent with the conclusions of the staff and its advisors. If the applicant's conclusions and assumptions conflict with the literature, substantive investigative results to support those conclusions must be submitted to the staff for review.

Subsection 2.5.3.2 is evaluated by first determining through a literature search that all known evidences of fault offset have been considered in the investigation. The results of the applicant's site investigations are studied and cross-compared in detail to see if there is evidence of existing or potential displacements. If such evidence is found, additional investigations such as field mapping, geophysical investigations, borings, trenching, etc., must be carried out to demonstrate that there is no offset or to define the characteristics of the fault if it does exist.

Subsection 2.5.3.3 is reviewed in conjunction with the consideration of Section 2.5.2. Historic earthquake data derived from the review of Section 2.5.2 are compared with known local tectonic features and a determination is made as to whether any of these earthquakes can reasonably be associated with the local structures. This determination includes an evaluation of the error regions of the earthquake locations. When available, the earthquake source mechanisms should be evaluated with respect to fault geometry.

Subsection 2.5.3.4 is evaluated to determine if the age dating methodology used by the applicant is based on accepted geological procedures. In some cases unusual age dating techniques may be used. When such methods are employed, the staff will require extensive documentation of the technique and may treat it as a generic review item. The resolution of all age dating techniques should be carefully documented.

Subsection 2.5.3.5 is evaluated by determining through a literature search that the applicant's evaluation of the regional tectonic framework is consistent and recognized by experts whose reports appear in the published literature. The conclusions reached by the applicant should be based on sound geologic principles and should explain the available geological and geophysical data. When special investigations are made to determine the structural relationship between faults which pass within five miles of the site and regional faults, the resolution of the investigative techniques should be given.

Subsection 2.5.3.6 is evaluated to determine if a sufficiently detailed investigation has been made by the applicant to define the specific characteristics of all capable faults located within 5 miles of the site. The fault characteristics requiring definition include: length, orientation, relationship of the fault to regional structures; the nature, amount, and geologic history of displacements along the fault; and the outer limits of the fault zone established by mapping fault traces 10 miles along trends in both directions from the point of nearest approach to the site. The staff must be satisfied that the investigation covers a large enough area in sufficient detail to demonstrate that there is little likelihood of near-surface displacement hazards associated with capable faults existing undetected near the site.

Subsection 2.5.3.7 Criteria for determining the zone requiring detailed faulting investigation are clearly outlined in Reference 1. The staff reviews the results of the applicant's faulting investigation together with the published literature. The investigative techniques employed by the applicant are evaluated to ascertain that they are consistent with the state of the art. As part of this phase, experts in specific

disciplines are asked to review certain aspects of the investigative program. The results of the investigation are analyzed to determine whether the outer limits of the zone requiring faulting investigation are appropriately conservative. If there is insufficient data to substantiate the outer boundaries, more conservative assumptions are required.

Subsection 2.5.3.8 If the detailed faulting investigations reveal that there is a potential for surface displacement at the site, the staff recommends that the site be moved to an alternate location. In the future, when it may be possible to design a nuclear power plant for displacements, substantial information will be required to support the design basis for surface faulting.

IV. EVALUATION FINDINGS

After completing the review, the staff summarizes its conclusions regarding surface faulting in the SER. If after the staff completes a detailed review of the applicant's investigations and conclusions and it has been effectively demonstrated that near-surface displacement cannot occur at the site, the entire section of the SER can be summarized by a statement such as: "The staff concludes that there are no surface or near-surface displacement potentialities existing at the site." If it is determined that surface displacement cannot be precluded, the staff notifies the applicant of its conclusions well in advance of publication of the SER.

V. REFERENCES

1. 10 CFR Part 100, Appendix A, "Seismic and Geologic Siting Criteria for Nuclear Power Plants."

2. Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," Revision 2.

3. "GEO-Reference: Computerized File of Earth Science Titles," American Geological Institute, Washington.

4. M. R. Grey, R. McAfee, Jr., and C. L. Wolf, eds., "Glossary of Geology," American Geological Institute, Washington (1972).

5. G. V. Cohee (chairman) et. al., "Tectonic Map of the United States," U. S. Geological Survey and American Association of Petroleum Geologists (1962).

6. State geological maps and accompanying texts.

7. U. S. Geological Survey 7.5- and 15-minute topographic and geologic quadrangle maps.

8. U. S. Department of Agriculture and U. S. Geological Survey aerial photographs.

9. Environmental Resources Technology Satellite photographs.

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SECTION 2.5.4 STABILITY OF SUBSURFACE MATERIALS AND FOUNDATIONS

REVIEW RESPONSIBILITIES

Primary - Geosciences Branch (GB)

Secondary - Structural Engineering Branch (SEB)

I. AREAS OF REVIEW

Information must be presented by the applicant concerning the properties and stability of all soils and rock which may affect the nuclear power plant facilities, under both static and dynamic conditions including the vibratory ground motions associated with the safe shutdown earthquake. Stability of these materials, as they influence the safety of seismic Category I facilities, must be demonstrated. In addition an assessment of the properties and stability of these materials should be consistent with SRP Sections 3.7 and 3.8. Much of the information discussed in this section may be presented in other sections, in which case it may be cross-referenced rather than repeated here. The results of the stability evaluations of subsurface materials and foundations are reviewed by SEB to assure that the loads and deflections including any reduction in support capability of subsurface materials can safely be accommodated by structural components.

The staff review covers the following specific areas:

1. Geologic features (Subsection 2.5.4.1) in the vicinity of the site:

   a. Areas of actual or potential surface or subsurface subsidence, solution activity, uplift, or collapse.

   b. Zones of alteration or irregular weathering profiles, and zones of structural weakness.

   c. Unrelieved stresses in bedrock and their potential for creep and rebound effects.

   d. Rocks or soils that might be unstable because of their mineralogy, lack of consolidation, water content, or potentially undesirable response to seismic or other events.

e. History of deposition and erosion, including glacial and other preloading influence on soil deposits.

f. Estimates of consolidation and preconsolidation pressures and methods used to estimate these values.

2. The static and dynamic engineering properties of soil and rock strata underlying the site (Subsection 2.5.4.2) as supported by representative field and laboratory data provided by the applicant.

3. The relationship of the foundations for safety-related facilities and the engineering properties of underlying materials as illustrated on plot plans and profiles (Subsection 2.5.4.3) provided by the applicant.

4. The results of seismic refraction and reflection surveys, including in-hole and cross-hole explorations, as presented in the safety analysis report (SAR) by discussions, plot plans, boring logs, tables, and profiles to support the assumed dynamic soil or rock characteristics (Subsection 2.5.4.4) and stratigraphy.

5. Safety-related excavation and backfill plans and engineered earthwork analyses and criteria (Subsection 2.5.4.5) as illustrated on plot plans and profiles, discussed in the text, and supported by explorations for borrow material, test fills and adequate representative laboratory test records.

6. Groundwater conditions and piezometric pressure in all critical strata (Subsection 2.5.4.6) as they affect the loading and stability of foundation materials. This part of the staff review also includes an evaluation of the applicant's plans for dewatering during construction as well as groundwater control throughout the life of the plant.

7. The responses of site soils or rocks to dynamic loading (Subsection 2.5.4.7), including appropriate laboratory and field test records in sufficient number and detail adequate to support conclusions derived from the analyses. Soil-structure interaction analyses are reviewed to evaluate stability and to provide input to SEB regarding the validity of the soil profile model used in the analyses.

8. The liquefaction potential (Subsection 2.5.4.8) and consequences of liquefaction of all subsurface soils, including the settlement of foundations. These analyses are based on soil properties obtained by state-of-the-art laboratory and field tests.

9. The earthquake design bases (Subsection 2.5.4.9) are evaluated in detail in Section 2.5.2. These are summarized and cross-referenced in this subsection. The safe shutdown earthquake (SSE) and the operating basis earthquake (OBE) are evaluated in this subsection in combination with other hazards (floods, etc.) to assess the adequacy of the site materials under dynamic conditions.

10. The results of investigations and analyses conducted to determine foundation material stability, deformation and settlement under static conditions (Subsection 2.5.4.10).

11. Criteria, references, and design methods (Subsection 2.5.4.11) used in static and seismic analyses of foundation materials, including an explanation of computer programs used in the analyses and soil loads on subsurface facilities.

12. Techniques and specifications to improve subsurface conditions (Subsection 2.5.4.12), which are to be used at the site to provide suitable foundation conditions.

13. Additional information on foundations is covered in SRP Sections 3.8.5 and should be cross-referenced to this section.

II. ACCEPTANCE CRITERIA

A thorough evaluation of the geotechnical engineering aspects of the nuclear plant site as described in the following subsections must be presented along with the basic data supporting all conclusions. Sufficient information must be provided to allow the staff and its advisors to conduct independent analyses. The site investigations must be adequate in scope and in technique to provide the necessary data. Guidelines for site investigations are given in Regulatory Guide 1.132.

Subsection 2.5.4.1. The section defining geologic features is acceptable if the discussions, maps, and profiles of the site stratigraphy, lithology, structural geology, geologic history, and engineering geology are complete and are supported by site investigations sufficiently detailed to obtain an unambiguous representation of the geology. The information must be presented in this subsection or cross-referenced to the appropriate subsection in Section 2.5.1.

Subsection 2.5.4.2. The description of properties of underlying materials is considered acceptable if state-of-the-art methods are used to determine the static and dynamic engineering properties of all foundation soils and rocks in the site area. These methods are described, for example, in geotechnical journals published by the American Society of Civil Engineers (Ref. 3), applicable standards published by the American Society for Testing and Materials (Ref. 4), publications of the Institution of Civil Engineers (Ref. 5), and various research reports prepared by universities (Ref. 6). The properties of foundation material must be supported by field (Ref. 10 and 11) and laboratory (Ref. 12) test records.

Normally, a complete field investigation and sampling program must be performed to define the occurrence and properties of underlying materials at a given site (Ref. 7). Summary tables must be provided which catalog the important test results; test results should be plotted when appropriate. Also, a detailed discussion of laboratory sample preparation must be given when applicable. For critical laboratory tests, full details must be given, e.g., how saturation of the sample was determined and maintained during testing, how the pore pressures changed.

The applicant should provide a detailed and quantitative discussion of the criteria used to determine that the samples were properly taken and tested in sufficient number to define all the critical soil parameters for the site. For sites that are underlain by saturated soils and sensitive clays, it should be shown that all zones which could become unstable due to liquefaction or strain-softening phenomena have been adequately sampled and tested. The relative density of the soils at the site should be determined. The applicant must also show that he has adequately defined the consolidation behavior of the soils as well as their static and dynamic strength. The discussion should explain how the developed data is used in the safety analyses, how the test data is enveloped for design, and why the design envelope is conservative.

Subsection 2.5.4.3. The discussion of the relationship of foundations and underlying materials is acceptable if it includes:

1. A plot plan or plans showing the locations of all site explorations, such as borings, trenches, seismic lines, piezometers, geologic profiles, and excavations with the locations of the safety-related facilities superimposed thereon.

2. Profiles illustrating the detailed relationship of the foundations of all seismic Category I and other safety-related facilities to the subsurface materials.

3. Logs of core borings and test pits.

4. Logs and maps of exploratory trenches in the preliminary safety analysis report (PSAR), and geologic maps and photographs of the excavations for the facilities of the nuclear power plant in the final safety analysis report (FSAR).

Subsection 2.5.4.4. The presentation of the dynamic characteristics of soil or rock is acceptable if geophysical investigations have been performed at the site and the results obtained therefrom are presented in detail. Completeness of the presentation is judged by whether or not the exploratory techniques used by the applicant yield unambiguous and useful information, whether they represent state-of-the-art exploration methods (Refs. 3, 4, 7, 9), and whether the applicant's interpretations are supported by adequate field records in the SAR. See also Subsection 2.5.2.3.

Subsection 2.5.4.5. The presentation of the data concerning excavation, backfill, and earthwork analyses is acceptable if:

1. The sources and quantities of backfill and borrow are identified and are shown to have been adequately investigated by borings, pits, and laboratory property and strength testing (dynamic and static) and these data are included, interpreted, and summarized.

2. The extent (horizontally and vertically) of all Category I excavations, fills, and slopes are clearly shown on plot plans and profiles.

3. Compaction specifications and embankment and foundation designs are justified by field and laboratory tests and analyses to assure stability and reliable performance.

4. Quality control methods are discussed and the quality assurance program described and referenced.

5. Control of groundwater during excavation to preclude degradation of foundation materials is described and referenced.

Subsection 2.5.4.6. The analysis of groundwater conditions is acceptable if the following are included in this subsection or cross-referenced to the appropriate subsections in Section 2.4:

1. Discussion of critical cases of groundwater conditions relative to the foundation stability of the safety-related facilities of the nuclear power plant.

2. Plans for dewatering during construction.

3. Analysis and interpretation of seepage and potential piping conditions during construction.

4. Records of field and laboratory permeability tests.

5. History of groundwater fluctuations as determined by periodic monitoring of local wells and piezometers. Flood conditions should also be considered.

Subsection 2.5.4.7. Descriptions of the response of soil and rock to dynamic loading are acceptable if:

1. An investigation has been conducted and discussed to determine the effects of prior earthquakes on the soils and rocks in the vicinity of the site. Evidence of liquefaction and sand cone formation should be included.

2. Field seismic surveys (surface refraction and reflection and in-hole and cross-hole seismic explorations) have been accomplished and the data presented and interpreted to develop P and S wave velocity profiles.

3. Dynamic tests have been performed in the laboratory on samples of the foundation soil and rock and the results included. The section should be cross-referenced with Subsection 2.5.2.5.

The soil-structure interaction analysis should be described in SRP Sections 3.7.1 and 3.7.2 and cross-referenced to this subsection. In the soil-structure interaction analysis, the following parameters are reviewed:

1. The static and dynamic properties of the soil supporting the structure are properly determined and compatible with the characteristics of the analytical model used to evaluate soil-structure interaction effects.

2. The soil profile has been properly modeled when a two-dimensional finite-element analysis is used, or if a half-space analysis method is used, when foundation moduli and damping are consistent with soil properties and soil profiles at the site.

3. The static and dynamic loads, and the stresses and strains induced in the soil surrounding and underlying the structure are adequately and realistically evaluated.

4. The consequences of the induced soil stresses and strains, as they influence the surrounding and underlying the structure, have been conservatively assessed.

Subsection 2.5.4.8. If the foundation materials at the site adjacent to and under Category I structures and facilities are saturated soils and the water table is above bedrock, then an analysis of the liquefaction potential at the site is required. The need for a detailed analysis is determined by a study on a case by case basis of the site stratigraphy, critical soil parameters, and the location of safety-related foundations. Undisturbed samples obtained at the site and appropriate laboratory tests are required to show if the soils are likely to liquefy.

When the need for an in-depth analysis is indicated, it may be based on cyclic triaxial test data obtained from undisturbed soil samples taken from the critical zones in the site area. The shear stresses induced in the soil by the postulated earthquake should be determined in a manner that is consistent with Standard Review Plan (SRP) Section 2.5.2. The criterion that should be used to determine when the soil samples tested "liquefied" should be taken as the onset of liquefaction (defined as the cycle when the pore pressure first equals the confining pressure). Test data showing the rate of pore pressure increase with number of load cycles should be presented. If the behavior of the pore pressure is such that peak to peak axial strains greater than a few percent occur before liquefaction, then the applicant must include the effects of these strains in his assessment of the potential hazards that complete or partial liquefaction could have on the stability and settlement of any Category I structures.

Nonseismic liquefaction (such as that induced by erosion, floods, wind loads on structures and wave action) should be analyzed using state-of-the-art soil mechanics principles.

Subsection 2.5.4.9. The earthquake design basis analysis is acceptable if a brief summary of the derivation of the safe shutdown and operating basis earthquakes (SSE and OBE) is presented and references are included to Subsections 2.5.2.6 and 2.5.2.7.

Subsection 2.5.4.10. The discussions of static analyses are acceptable if the stability of all safety-related facilities has been analyzed from a static stability standpoint including bearing capacity, rebound, settlement, and differential settlements under deadloads of fills and plant facilities, and lateral loading conditions. Field and laboratory test procedures and results must be included to document soil and rock properties used in the analyses. The applicant must show that the methods of analysis used are appropriate for the local soil conditions and the function of the facility.

Subsection 2.5.4.11. The discussion of criteria and design methods is acceptable if the criteria used for the design, the design methods employed, and the factors of safety obtained in the design analyses are described and a list of references presented. An explanation and verification of the computer analyses used and source references should be included.

Subsection 2.5.4.12. The discussion of techniques to improve subsurface conditions is acceptable if plans, summaries of specifications, and methods of quality control are described for all techniques to be used to improve foundation conditions (such as grouting, vibraflotation, dental work, rock bolting, or anchors).

III. REVIEW PROCEDURES

The review process is conducted in a similar manner and concurrent with that described in SRP 2.5.1. The services of the Corps of Engineers are used on selected sites to aid the staff in evaluating the geotechnical engineering aspects of particular sites.

After acceptance of the SAR, the results of site investigations (such as borings, geologic maps, logs of trenches and pits, permeability test records, results of seismic investigations, laboratory test results, profiles, and plot plans) are studied and cross-checked in considerable detail to determine whether or not the assumptions used in the evaluation are conservative. The design criteria are reviewed to ascertain that they are within the present state-of-the-art. Staff comments and questions at this phase of the review, concerning the information in the SAR, are sent to the applicant as first-round questions (Q-1). For those facilities that have complex subsurface conditions, where marginal safety has been achieved, or where the applicant proposes to construct a seismic Category I earth or rockfill dam, an independent analysis of the design is performed by the staff or its advisors, the Corps of Engineers. The evaluations conducted by the staff and its advisors may identify additional unresolved items, or reveal that the applicant's investigations and analyses are not complete or sufficiently conservative. Additional information is then requested in a second round of questions (Q-2), or a staff position is taken requiring adoption of a more conservative approach.

The data needed to satisfy the requirements of this section are not usually complete in the early stages. Detailed design investigations are usually still in progress and final conclusions have often not been made. Because of this, the question and answer exchange may not be complete at the Q-2 stage. Most of the open items of Section 2.5

remaining at the time that the safety evaluation report (SER) input is required are in the geotechnical engineering area because actual site conditions may not be fully revealed until excavations are opened and construction has begun. Thus, a site visit, in addition to that noted in Section 2.5.1, "Basic Geologic and Seismic Information," is necessary during the post-CP period to examine the foundation materials exposed in excavations during construction. Information and final designs, including confirming tests and revised analyses, are to be submitted in the FSAR.

Generally, the staff is guided by the Seismic and Geologic Siting Criteria (Ref. 1) and the Standard Format (Ref. 2) in reviewing Section 2.5.4.

Following is a brief description of the review procedures conducted by the staff in evaluating the geotechnical engineering aspects of nuclear power plant sites.

Subsection 2.5.4.1. Geologic features are evaluated by conducting an independent literature search and comparing these results with the information included in the applicant's SAR. References used in reviewing this subsection include published or unpublished reports, maps, geophysical data, construction records, etc., by the USGS, other Federal agencies, State agencies, and private companies (such as oil corporations and architect engineering firms). In conjunction with the literature search, the staff and its USGS advisors review the geological investigations conducted by the applicant. Using the references listed at the end of this section and other sources, the following questions are considered in detail:

1. Are the exploratory techniques used by the site investigator representative of the present state-of-the-art? Do the samples represent the in situ soil conditions?

2. Do the applicant's investigations provide adequate coverage of the site area and in sufficient detail to define the specific subsurface conditions with a high degree of confidence?

3. Have all areas or zones of actual or potential surface or subsurface subsidence, uplift or collapse, deformation, alternation, solution cavities or structural weakness, unrelieved stresses in bedrock, or rocks or soils that might be unstable because of their physical or chemical properties been identified and adequately evaluated?

Subsection 2.5.4.2. Properties of underlying materials are evaluated to determine whether or not the investigations performed (including laboratory and field testing) were sufficient to justify the soil and rock properties used in the foundation analyses.

To determine whether sufficient investigations were performed, the staff carefully reviews the criteria developed and used by the applicant in laying out the boring, sampling and testing program and evaluates the effectiveness of the program in defining the specific foundation conditions at the site to assure that all critical conditions

have been adequately sampled and tested. If suitable criteria have not been developed and used by the applicant, the staff develops appropriate criteria, using Regulatory Guide 1.132 and the data given in the SAR, and determines if sufficient investigation and testing have been carried out. If criteria are given, the staff reviews them to determine if they are appropriate and have been implemented.

If it is the staff's judgment that the applicant's investigations or testing are inappropriate or insufficient, additional investigations will be required. The final conclusion is based on professional judgment, considering the complexity of the site subsurface conditions. As part of the review, the staff must ascertain, often with the help of the Corps of Engineers, that state-of-the-art laboratory and field techniques and equipment are employed in determining the material properties.

Subsection 2.5.4.3. Plot plans and profiles are reviewed by comparing the subsurface materials with the proposed locations (horizontal and vertical) of foundations and walls of all seismic Category I facilities. The profiles and plot plans are cross-checked in detail with the results of all subsurface investigations conducted at the site to ascertain that sufficient exploration has been carried out and to determine whether or not the interpretations made by the investigators are valid and the foundation design assumptions contain adequate margins of safety.

Subsection 2.5.4.4. Staff evaluation consists of a detailed review of all geophysical explorations conducted at the site, including seismic refraction, reflection, and in-hole surveys and magnetic and gravity surveys. Expertise within the USGS regarding specific techniques is drawn upon in this review. Logs of core borings, trenches, and test pits are reviewed and compared with data from the seismic surveys and other geophysical explorations. Results must be consistent or additional investigations are required, or the applicant must use the most conservative values. Following the PSAR review and during the FSAR review the staff compares conditions as mapped in the open excavations with interpretations and assumptions derived during the investigation program.

Subsection 2.5.4.5. Excavations, backfill, and earthwork are evaluated by the staff as follows:

1. The investigations for borrow material, including boring and test pit logs, and compaction test data are reviewed and judged as to their adequacy.

2. Laboratory dynamic and static records of tests performed on samples compacted to the design specifications are reviewed to ascertain that state-of-the-art criteria are met.

3. Analyses and interpretations are reviewed to assure that static and dynamic stability requirements are met.

4. Excavation and compaction specifications and quality control procedures are reviewed to ascertain conformance to state-of-the-art conservative standards.

Subsection 2.5.4.6. Groundwater conditions as they affect foundation stability are evaluated by studying the applicant's records of the historic fluctuations of groundwater at the site as obtained by monitoring local wells and springs and by analysis of piezometer and permeability data from tests conducted at the site. The applicant's dewatering plans during and following construction are also reviewed. Adequacy of these plans is evaluated by comparing with the results of the groundwater investigations and by professional judgment of groundwater and soil conditions at the site.

Subsection 2.5.4.7. Response of soil and rock to dynamic loading and soil-structure interaction is evaluated by a detailed study of the results of the investigations and analyses performed. Specifically, the effects of past earthquakes on site soils or rocks (a requirement in SRP 2.5.2) are determined. The data from core borings, from geophysical investigations, and from dynamic laboratory tests such as sonic and cyclic triaxial tests on undisturbed samples are evaluated. The object of the staff review is to ascertain that reasonably conservative dynamic soil and rock characteristics are used in the design and analyses and that all the significant soil and rock strata have been considered in the analyses. In some cases, independent analyses and interpretations are carried out as outlined in SRP 2.5.2, or as required to verify the liquefaction analysis discussed in Subsection 2.5.4.8.

Subsection 2.5.4.8. Liquefaction potential is reviewed by a study of the results of geotechnical investigations including boring logs, laboratory classification test data and soil profiles to determine if any of the site soils could be susceptible to liquefaction. The results of in-situ tests such as the standard penetration tests and the density and strength data obtained from undisturbed samples obtained in exploration borings are examined and, when appropriate, related to the liquefaction potential of in situ soils.

If it is determined that there may be liquefaction-susceptible soils beneath the site, the applicant's site exploration methods, laboratory test program, and analyses are reviewed for adequacy and reasonableness. The analysis submitted by the applicant is reviewed in detail and compared to an independent study performed by the staff. As a minimum, the staff study consists of:

1. A review of appropriate Standard Penetration test results, other in-situ test data and groundwater conditions to assess li' 'action potential.

2. A careful review of conventional laboratory and cyclic triaxial test data to insure that appropriate samples were obtained and tested from critical, liquefiable zones.

3. Confirmation that an adequate number of samples were properly tested and that the test results account for the natural variation in different samples as well as define the cyclic resistance to liquefaction of the soils.

4. An assessment of the liquefaction potential using a conservative envelope of the test data submitted.

5. A calculation of the stress induced by the earthquake that has been arrived at by an envelope of critical conditions calculated for the site based on variations in the properties of the soil strata.

6. Assurance that conservative ranges of relative density of the soils are estimated. The applicant's estimates of the "safety factor" obtained from his analysis is compared to the safety margins estimated by the staff. (The applicant's plans to "eliminate" the liquefaction condition, usually by excavation and backfill, vibroflotation, or chemical grouting is evaluated as discussed in Subsections 2.5.4.5 and 2.5.4.12.)

7. An assessment of post-earthquake stability and settlements due to partial liquefaction using state-of-the-art techniques.

8. An assessment of nonseismic liquefaction based on state-of-the-art techniques.

Subsection 2.5.4.9. The in-depth staff evaluation of the safe shutdown and operating basis earthquakes is contained in SRP 2.5.2. The staff's evaluation of the amplification characteristics of specific soils and rocks beneath the site as determined by procedures discussed in that section and in Subsections 2.5.4.2, 2.5.4.4, and 2.5.4.7 are summarized and cross-referenced herein.

The review of Subsection 2.5.4.9 concentrates on determining its consistency or inconsistency with other subsections. Cross-referencing with other sections is expected.

Subsection 2.5.4.10. Static analyses of the bearing capacity and settlement of the supporting soils under the loads of fills, embankments, and foundations are evaluated by conventional, state-of-the-art methods (Ref. 8). In general, the evaluation procedure includes:

1. Determining whether or not the soil and rock properties used in the analyses represent the actual site conditions beneath the plant facilities. The site investigation, sampling, and laboratory test programs must be adequate for this evaluation.

2. Determining whether or not the methods of analysis are appropriate for the earthworks, foundations, and soil conditions at the site.

3. Coordinate with SEB to determine whether or not the bearing capacity, settlement, differential settlement, and tilt estimates indicate conservative and tolerable behavior of the plant foundations when these values are compared to design criteria and quality assurance specifications.

4. Evaluation of particularly complex cases on the basis of accepted principles and techniques as supplemented by case histories and confirmatory measurement and analysis programs (Ref. 8).

Subsection 2.5.4.11. Criteria and design methods, including construction control and monitoring systems, are evaluated on the basis of conservative accepted practice for similar facilities. Site exploration, sampling, testing, and interpretation are judged with respect to completeness, care and technique, meaningful documentation, performance records for similar projects, published guidelines, and state-of-the-art practice. However, unconventional or research-oriented tests and interpretations are encouraged whenever such work aids or supplements conventional practices. Design criteria and methods are compared to similar standards published or utilized by public agencies such as the U. S. Navy Department, U. S. Army Engineers, and U. S. Department of the Interior. Design safety features, the applicant's proposed confirmatory tests and measurements, and monitoring of performance for safety-related foundations and earthworks are reviewed and evaluated on a case-by-case basis.

Subsection 2.5.4.12. Techniques to improve subsurface conditions are evaluated by reviewing the applicant's specifications and techniques for performance and quality control for such activities as grouting, excavation and backfill, vibroflotation, rock bolting, and anchoring. Confirmatory data should be contained in the FSAR.

IV. EVALUATION FINDINGS

If the evaluation by the staff, on completion of the review of geotechnical engineering aspects of the plant site, confirms that of the applicant, the conclusion in the SER states that the investigations performed at the site are adequate to justify the soil and rock characteristics used in the design, and that the design analyses contain adequate margins of safety for construction and operation of the subject nuclear power plant. Staff reservations about any portion of the applicant's analyses are stated, in sufficient detail to make clear the precise nature of the staff concern.

A typical staff SER finding follows:

> "The site is located in the Piedmont at an average elevation of +395 feet mean sea level (msl). Exploratory borings have been made and refraction and reflection seismic surveys conducted to establish the stratigraphy of the site. Additionally, undisturbed samples of representative soils and core borings have been obtained to evaluate the characteristics of the foundation materials; close-centered cross-hole seismic tests have been conducted to determine the elastic properties of these materials. Groundwater at the site varies from +375 to +380 feet msl.

"The area has been exposed to subaerial weathering and erosion since middle Mesozoic time, and a deep weathering profile has developed. The depth of weathering depends on the location and degree of jointing, orientation of schistosity, and composition of the parent rock.

"The applicant has categorized the foundation material into three zones according to the degree of weathering:

(a) Zone 1 contains residual soil derived from severely weathered slate. The soil is a sandy, silty clay containing slate and quartz fragments. Decomposed to severely weathered slate is also present. The slate still retains the original rock structure, although it is soft and partly friable. Quartz veins within the slate are extremely fractured. Seismic compression (P) and shear (S) wave velocities exceed 4000 ft/sec and 1800 ft/sec, respectively. Zone 1 ranges in thickness from less than 20 feet to more than 50 feet.

(b) Zone 2 consists of moderately weathered slate and varies from 15 to 60 feet thick. P and S wave velocities generally exceed 6500 ft/sec and 2500 ft/sec, respectively.

(c) Zone 3 contains slightly weathered to unweathered slate and is encountered at depths of 60 to 90 ft below ground surface.

"The site area will be leveled to about elevation +390 feet msl, and containments will be founded on a 10 foot thick, reinforced concrete mat on slightly weathered slate or fill concrete over slightly weathered slate. The reactor service building between the reactors and the control building will be on mats at elevation +385 feet msl on compacted structural fill resting on slightly to moderately weathered rock. The turbine generators will be founded on compacted structural fill over moderately weathered rock at elevation +380 feet msl. The diesel generator building, reactor plant component air-cooled heat exchanger enclosures, and the CACS air-cooled heat exchanger will be founded on either individual concrete footings or continuous footings (grade beams) at +385 feet msl, on compacted structural fill over moderately weathered slate. Allowable bearing capacities from laboratory tests and field plate tests for Zone 1, Zone 2, and Zone 3 materials are 4, 10, and 25 tons per square foot, respectively. All piping will be entrenched and bedded in moderately to severely weathered slate.

"Settlement and differential settlement of safety-related facilities has been estimated to be less than one inch.

"The applicant states that severely weathered or soft zones of rock will be excavated and replaced with lean concrete. This procedure will also be followed wherever severe weathering extends along joints, schistosity, etc., below the base of the foundations; this material will be excavated to a depth 1-1/2 times the width of the zone and backfilled with concrete.

Category I structural backfill under structures will either be concrete or compacted granular backfill. If granular backfill is used, it will be compacted to at least 85 percent relative density or to 95 percent of the maximum density determined by the Modified Proctor test. These backfill criteria are acceptable criteria for soil pressures on foundations and buried pipes and are suitable and conservative for both static and dynamic conditions.

"Suitable borrow material for dikes, dams and impervious linings are available for the ultimate heat sink ponds. The applicant's tests on these materials and the construction criteria to be followed ensure that leakage, piping and cracking hazards of these vital earthworks are minimal. Filters, blanket drains, relief wells, piezometers and settlement monuments will assure the reliable performance of the ultimate heat sink water-retention facilities.

"The applicant has shown that the appropriate acceleration level on sound rock is 0.12g for the safe shutdown earthquake (SSE). The operating bases earthquake (OBE) value is taken as 0.06g. The applicant has performed a site-dependent analysis to estimate the site amplification effects and found that the weathered rock or structural backfill would amplify the rock motion. An acceleration level of 0.17g for the SSE will be used for those structures founded on weathered rock or structural backfill over weathered rock. The time history used for seismic design of Category I earth dams and for liquefaction assessment envelopes the response spectra for the site and has a conservative duration.

"Based on the results of the applicant's investigations, laboratory and field tests, analyses, and criteria for design and construction, we conclude that the site and the plant foundations will be adequate to safely support the planned nuclear power plant and that safety-related earthworks will perform their functions reliably."

V. REFERENCES

1. 10 CFR Part 100, Appendix A, "Seismic and Geologic Siting Criteria for Nuclear Power Plants."

2. Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," Revision 2.

3. Journal of the Geotechnical Engineering Division, Proceedings of the American Society of Civil Engineers.

4. Book of ASTM Standards and Special Technical Publications, American Society for Testing and Materials.

5. Geotechnique, The Institution of Civil Engineers, London.

6. Earthquake Engineering Research Center, University of California, Berkeley.

7. M. Juul Hvorslev, "Subsurface Exploration and Sampling of Soils for Civil Engineering Purposes," Waterways Experiment Station, U. S. Army Corps of Engineers, November 1949.

8. GEODEX INTERNATIONAL, Soil Mechanics Information Service, Sonoma, California.

9. Regulatory Guide 1.132, "Site Investigations for Foundations of Nuclear Power Plants."

10. Engineering Manual EM1110-2-1907, "Soil Sampling," U.S. Army Corps of Engineers, March 1972.

11. Engineering Manual EM110-2-1908, "Instrumentation of Earth and Rock Fill Dams," U.S. Army Corps of Engineers, August 1971.

12. Engineering Manual EM1110-2-1906, "Laboratory Soil Testing," U.S. Army Corps of Engineers, November 1970.

U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

SECTION 2.5.5 STABILITY OF SLOPES

REVIEW RESPONSIBILITIES

Primary - Geosciences Branch (GB)

Secondary - Structural Engineering Branch (SEB)

I. AREAS OF REVIEW

Information, including analyses and substantiation, must be presented in the applicant's safety analysis report (SAR) and reviewed by the staff concerning the stability of all earth and rock slopes both natural and man-made (cuts, fills, embankments, dams, etc.) whose failure, under any of the conditions to which they could be exposed during the life of the plant, could adversely affect the safety of the plant. The following subjects must be evaluated using the applicant's data in the SAR and information available from other sources: slope characteristics (Subsection 2.5.5.1); design criteria and design analyses (Subsection 2.5.5.2); results of the investigations including borings, shafts, pits, trenches, and laboratory tests (Subsection 2.5.5.3); properties of borrow material, compaction and excavation specifications (Subsection 2.5.5.4). The results of the stability of slopes evaluations are reviewed by SEB to assure that displacements or failure of site slopes as indicated in the SAR do not have an adverse impact on structural components.

II. ACCEPTANCE CRITERIA

The information in the SAR must be in compliance with the Standard Format (Ref. 2) and the Seismic and Geologic Siting Criteria (Ref. 1). This section of the SAR is judged acceptable if the information presented is sufficient to demonstrate the dynamic and static stability of all slopes whose failure could adversely affect, directly or indirectly, safety-related structures of the nuclear plant or pose a hazard to the public. The emergency cooling water source is of particular interest with regard to slope stability. The secondary source of emergency cooling water should survive the operating basis earthquake (OBE) and design basis flood. Completeness is determined by the ability to make an independent evaluation on the basis of information provided by the applicant.

Subsection 2.5.5.1. The discussion of slope characteristics is acceptable if the subsection includes:

a. Cross sections and profiles of the slope in sufficient quantity and detail to represent the slope and foundation conditions.

b. A summary and description of static and dynamic properties of the soil and rock comprising seismic Category I embankment dams and their foundations, natural and cut slopes, and all soil or rock slopes whose stability would directly or indirectly affect safety-related and Category I facilities. The text should include a complete discussion of procedures used to estimate, from the available field and laboratory data, conservative soil properties and profiles to be used in the analysis.

c. A summary and description of groundwater, seepage, and high and low groundwater conditions.

Subsection 2.5.5.2. The discussion of design criteria and analyses is acceptable if the criteria for the stability and design of all seismic Category I slopes are described and valid static and dynamic analyses have been presented to demonstrate that there is an adequate margin of safety. A number of different methods of analysis are available in the literature. Computer analyses should be verified by manual methods.

To be acceptable, the static analyses should include calculations with different assumptions and methods of analysis to assess the following factors:

1. The uncertainties with regard to the shape of the slope, boundaries of the several types of soil within the slope and their properties, the forces acting on the slope, and pore pressures acting within the slope.

2. Failure surfaces corresponding to the lowest factor of safety.

3. The effect of the assumptions inherent in the method of analysis used.

4. Adverse conditions such as high water levels due to the probable maximum flood (PMF), sudden drawdown, or steady seepage at various levels. In general, safety factors related to the slope hazard are needed; however, actual values depend somewhat on the method of analysis, on the assumptions concerning the soil properties, on construction techniques, and on the range of material parameters.

To be acceptable, the dynamic analyses must account for the effect of cyclic motion of the earthquake on soil strength properties. Actual test data are needed for both the in situ soils as well as for any materials used in the construction of dams or embankments. As discussed above, the various parameters, such as geometry, soil strength, modeling method (location and number of elements (mesh) if a finite-element analysis is used), and hydrodynamic and pore pressure forces, should be varied to show that there is an adequate margin of safety (Refs. 16 and 17). Where liquefaction is possible, major dam foundation slopes and embankments should be analyzed by state-of-the-art finite-element or finite-difference methods of analysis. Where there are liquefiable

soils, changes in pore pressure due to cyclic loading must be considered in the analysis to assess not only the potential for liquefaction but also the effect of pore pressure increase on the stress-strain characteristic of the soil and the post-earthquake stability of the slopes.

Subsection 2.5.5.3. In discussing the soil investigations, the applicant should describe the borings and soil testing that was carried out for slope stability studies and dam and dike analyses. The test data, which must meet the criteria set forth in Sections 2.5.1 and 2.5.4, could be presented in those sections and referenced in this subsection. Because dams, dikes, and natural or cut slopes are often remote from the main plant area, additional exploration, tests, and analyses for these areas should be presented in this subsection.

Subsection 2.5.5.4. Compaction specifications should be discussed in this section. The applicant should describe the excavation, backfill, and borrow material planned for any dams, dikes, and embankment slopes. Planned construction procedures and control of earthworks should be described. To be acceptable, the information must be given as discussed in Subsection 2.5.4.5. Some of this information could be presented in Subsection 2.5.4.5. Because dams, dikes, and other earthworks are often remote from the main seismic Category I structures, it is necessary to complete this information in this subsection. Quality control techniques and requirements during and following construction must also be discussed and referenced to quality assurance sections of the SAR.

III. REVIEW PROCEDURES

The review process is conducted in a similar manner and concurrent with that described in Standard Review Plans (SRP) 2.5.1, 2.5.2, and 2.5.4. The Corps of Engineers is the principal advisor to the staff regarding foundation engineering and slope stability analyses, particularly in the evaluation of safety-related and seismic Category I earthworks, earth and rock-fill dams, dikes, and reservoirs. Standard references used by the staff are listed in Section V of this SRP.

An acceptance review is conducted to determine if the Standard Format (Ref. 2) has been adhered to and to judge whether or not the information presented is sufficient to permit an independent in-depth review and analysis of the safety of the proposed facility. After acceptance of the SAR, the results of site investigations such as borings, maps, logs of trenches, permeability test records, results of seismic investigations, laboratory test results, profiles, plot plans, and stability analyses are studied and cross-checked in considerable detail to determine whether or not the assumptions and analyses used in the design are conservative. The degree of conservatism required depends upon the type of analysis used, the reliability of parameters considered in the slope stability analysis, the number of borings, the sampling program, the extent of the laboratory test program, and the resultant safety factor. In general, the applicable soil strength data should be conservatively selected for the various possible soil profiles and slope conditions. For lower safety factors, several soil profiles should be analyzed

to insure that reasonable ranges of soil properties have been considered. Other factors such as flood conditions, pore pressure effects, possible erosion of soils, and possible seismic amplification effects should be conservatively assessed.

The design criteria and analyses are reviewed to ascertain that the techniques employed are appropriate and represent the present state-of-the-art. Staff comments and questions at this phase of the review, concerning the information in the SAR, are sent to the applicant as first-round questions (Q-1). An independent analysis of the design of safety-related earth or rock-fill embankments is performed by the staff's advisors, the Corps of Engineers, or by the staff as deemed necessary. The Corps also evaluates natural or cut slopes, as required, on a case-by-case basis. The evaluations conducted by the staff and its advisors may identify additional unresolved items or reveal that the applicant's analyses are not conservative. Additional information is then requested in a second round of questions (Q-2), or a staff position is taken requiring conformance to a more conservative approach.

After completing the review, if the staff's conclusions are consistent with those reached by the applicant, these conclusions are summarized in the safety evaluation report (SER) or in a supplement to the SER. In the event that the applicant's investigation and design are not judged to be sufficiently conservative, a staff position is stated and the applicant is asked to further substantiate his position by additional investigations or monitoring to demonstrate that a failure of the slopes in question will not harm the safety functions of the plant, or to concur in the staff position.

The data needed to satisfy the requirements of this section are often incomplete in the early stages. However, sufficient field and laboratory data should be presented and conservatively interpreted to allow a realistic assessment of the safety of proposed slopes and supporting foundations. Detailed design investigations are usually still in progress and final design conclusions have often not been made. Because of this, the question and answer exchange is not generally complete at the Q-2 stage. Most of the open items of Section 2.5 remaining at the time that the safety evaluation report (SER) input is required are in the foundation engineering and slope stability areas because actual conditions may not be revealed until excavations are opened; site visits conducted after construction permit (CP) issuance are therefore necessary.

All natural safety-related slopes are examined during at least one of the two site visits required of the staff. Because excavated slopes or embankments are not usually constructed until after a construction permit has been granted, detailed as-built documentation of these slopes and embankments, as well as complete stability and safety analyses, are necessary in the FSAR.

Following is a brief description of the review procedures conducted by the staff in evaluating the slope stability aspects of nuclear power plant sites.

Subsection 2.5.5.1. Plot plans, cross sections, and profiles of all safety-related slopes in relation to the topography and physical properties of the underlying materials are reviewed and compared with exploratory records to ascertain that the most critical conditions have been addressed and that the characteristics of all slopes have been defined. The soil and rock test data are reviewed to insure that there is sufficient relevant test data to verify the soil strength characteristics assumed for the slopes, dikes, and dams under analysis. The evaluation is to some extent a matter of engineering judgment; however, if the safety factors resulting from the analysis are not appropriate to the hazards posed by a slope failure and other than clearly conservative soil properties and profiles were used, the applicant is required to obtain additional data to verify his assumptions, or to show that, even if the worst possible conditions are assumed, there is an adequate margin of safety. With respect to seismic analysis, this subsection and Subsection 2.5.5.2 are reviewed concurrently because different methods of analysis may involve different approximations, assumptions, and soil properties.

In addition to generic state-of-the-art literature, other potential sources of information are those containing design, construction, and performance records of natural slopes, excavation slopes, and dams that may have been constructed in the general vicinity of the nuclear power plant. Examples of such documents are design memoranda and construction reports regarding nearby projects of public agencies such as the Corps of Engineers, the Tennessee Valley Authority, the Bureau of Reclamation, and private construction contractors or architect-engineers.

Subsection 2.5.5.2. The criteria, design techniques, and analyses are evaluated by the staff to ascertain that:

1. Appropriate state-of-the-art methods have been employed.

2. Conservative assumptions regarding soil and rock properties have been used in the design and analysis of slopes and embankments as discussed above in Subsection 2.5.5.1.

3. Appropriately conservative margins of safety have been incorporated in the design.

The criteria and design methods used by the applicant are reviewed to ascertain that state-of-the-art techniques are being employed. The design analyses are reviewed to be sure that the most conservative failure approach has been used and that all adverse conditions to which the slope might be subjected have been considered. Such conditions include ground motions from the safe shutdown earthquake, settlement, cracking, flood or low-water steadystate seepage, sudden drawdown of an adjacent reservoir, or a reasonable assumption of the possible simultaneous occurrence of two natural events such as an earthquake and flood. The review is also concerned with determining whether or not the soil and rock characteristics derived from the investigations described in Subsection 2.5.5.3 have been completely and conservatively incorporated into the design. When marginal factors of safety are indicated by the independent analyses performed by

the staff and its consultants, additional substantiation and refinement is required or the applicant must use more conservative assumptions.

No single method of analysis is entirely acceptable for all stability assessments; thus, no single method of analysis can be recommended. Relevant manuals issued by public agencies (such as the U.S. Navy Department, U.S. Army Corps of Engineers, and U.S. Bureau of Reclamation) are often used in reviews to ascertain whether the analyses performed by the applicant are reasonable. Many of the important interaction effects cannot be included in current analyses and must be treated in some approximate fashion. Engineering judgment is an important factor in the staff's review of the analyses and in assessing the adequacy of the resulting safety factors.

If the staff review indicates that questionable assumptions have been made by the applicant or some non-standard or inappropriate method of analysis has been used, then the staff or its consultant may model the dam or slope in a manner which it feels is more consistent with the data and perform an independent analysis.

During the operating license review, all open items requiring resolution, including construction data and as-built analyses, settlement records, piezometer records, and absence of seepage, that support the adequacy and safety of the design, are reviewed by the staff.

Subsection 2.5.5.3. A comprehensive program of site investigations including borings, sampling, geophysical surveys, test pits, trenches, and laboratory and field testing must be carried out by the applicant to define the physical characteristics of all soil and rock beneath safety-related and seismic Category I slopes, and borrow material that is to be used to construct safety-related dams, fills, and embankments. The staff reviews these investigations to ascertain that the program has been adequate to define the in situ and earthwork soil and rock characteristics. The decision as to the adequacy of the investigation program is based on the methods discussed in Section 2.5.4.

Subsection 2.5.5.4. The preliminary specifications and quality control techniques to be used during construction are reviewed by the staff to ascertain that all design conditions are likely to be met. During this part of the review the following are among those subjects reviewed for adequacy:

1. Proposed construction dewatering plan to ensure that it will not result in damage either to the natural or engineered foundation materials or to the structural foundation.

2. The excavation plan to remove all unsuitable materials from beneath the foundations and the quality control procedures which establish suitable materials.

3. The techniques and equipment to be used in compacting foundation and embankment materials.

4. The quality control and testing program to provide a high level of assurance that:

   a. The selected borrow material is as good and as relatively homogeneous as anticipated from the investigation program.

   b. The compacted foundation soil meets design specifications.

5. The techniques for improving the stability of natural slopes such as drainage, grouting, rock bolting, and applying gunite.

6. The plans for monitoring during and after construction to detect occurrences that could detrimentally affect the facility. Such monitoring includes periodic examination of slopes, survey of settlement monuments, and measurements of local wells and piezometers.

IV. EVALUATION FINDINGS

The staff's conclusions regarding the stability of slopes are summarized in the safety evaluation report (SER) or in a supplement to the SER. The following is an example:

"Both natural and man-made slopes exist at the site. At the plant site, which is located several hundred feet from the Green Valley and about 280 feet above the level of Jones Pond, the slope is relatively gentle for about 250 feet west of the westernmost Category I structures, then steepens, attaining an angle of more than 45° near the bottom of the valley wall. Major structural trends, schistosity, and one of the predominant joint trends are nearly perpendicular to the slope. A second predominant joint set is nearly parallel to the river and dips to the southwest, but no slope movements have apparently affected the valley walls in the vicinity of the site. Seven other joint trends were detected by the applicant. These joint sets are reported to be moderately spaced and discontinuous. The applicant has drilled several exploratory holes and cored others to assess the natural slope characteristics and groundwater regime. Even though the natural slopes are some distance from safety-related plant facilities and slope failures are not obvious safety hazards, the applicant has performed stability analyses of these slopes under safe shutdown earthquake (SSE) conditions. The minimum computed safety factor was 1.6 using conservative slope and material parameters.

"Man-made earth slopes related to the safety of the plant include excavation cuts for the ultimate heat sink canal and dams and dikes for the ultimate heat sink storage pond. An extensive investigation and test program has determined all the significant characteristics and properties of cut slopes and fill embankments. Earthwork compaction criteria, construction control, and select fill materials are consistent with high-quality water-retention facilities. Conservative stability analyses of these slopes under SSE conditions indicated minimum safety factors of 1.5.

"Based on the results of the applicant's investigations, laboratory and field tests, analyses, and criteria for design and construction, we and our consultants conclude that natural and man-made slopes will remain stable under SSE conditions and that safety-related earthworks will function reliably."

V. REFERENCES

1. 10 CFR Part 100, Appendix A, "Seismic and Geologic Siting Criteria for Nuclear Power Plants."

2. Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," Revision 2.

3. "ASCE Soil Mechanics and Foundation Division Conference on Stability and Performance of Slopes and Embankments, August 22-26, 1966." Published in J. Soil Mech. and Found., ASCE, Vol. 93 (1967).

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U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

SECTION 3.2.1 SEISMIC CLASSIFICATION

REVIEW RESPONSIBILITIES

Primary - Reactor Systems Branch (RSB)

Secondary - Containment Systems Branch (CSB)
Auxiliary and Power Conversion Systems Branch (APCSB)
Effluent Treatment Systems Branch (ETSB)

I. AREAS OF REVIEW

Nuclear power plant structures, systems, and components important to safety should be designed to withstand the effects of earthquakes without loss of capability to perform necessary safety functions. Information presented by the applicant identifying those structures, systems, and components (including their foundations and supports) which are important to safety and are designed to withstand, without loss of function, the effect of a safe shutdown earthquake (SSE) is reviewed. The SSE is based upon an evaluation of the maximum earthquake potential and is that earthquake which produces the maximum vibratory ground motion for which structures, systems, and components important to safety are designed to remain functional. Those structures, systems, and components that are designed to remain functional if the SSE occurs are designated seismic Category 1.

The RSB reviews the classification of those plant features (excluding electrical features) specified as seismic Category I by the applicant in his safety analysis report (SAR). Where required, specific information or assistance may be obtained from the EICSB to review classification of electrical and instrumentation systems. This review is done for both construction permit (CP) and operating license (OL) applications.

The applicant's proposed classifications may be presented in the form of a table which identifies structures and fluid systems that are seismic Category I. Where portions of structures and fluid systems are seismic Category I they also must be clearly identified. For fluid systems important to safety, the classification tables in the application should identify system components such as pressure vessels, heat exchangers, storage tanks, pumps, piping, and valves, have suitable footnotes defining interfaces, and be in sufficient detail so that there is a clear understanding of the extent of those portions of the system that are classified as seismic Category I.

Alternately, such information may be presented on suitable piping and instrumentation diagrams, or may be combined with the information presented in SAR Section 3.2.2, in which case it may be cross-referenced rather than repeated here.

The CSB reviews, in SAR Section 6.2.5, the detailed system design of seismic Category I fluid systems that are provided for the control of combustible gas concentrations in containment following a loss-of-coolant accident.

The APCSB reviews, in SAR Sections 9 and 10, the detailed system design of auxiliary fluid systems important to safety that are designated seismic Category I.

The ETSB reviews, in SAR Sections 11.2 and 11.3, the detailed system design of seismic Category I liquid, gaseous, and solid radioactive waste systems that are provided to reduce the radioactivity to levels which will not be in excess of the appropriate limits.

In the event a branch that has secondary review responsibility identifies other plant features important to safety that have not been previously identified by the RSB, this information should be transmitted to the RSB.

II. ACCEPTANCE CRITERIA

1. 10 CFR Part 50, Appendix A, General Design Criterion 2. This criterion 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 necessary safety functions.

2. Regulatory Guide 1.29, "Seismic Design Classification." This Regulatory Guide describes an acceptable method of identifying and classifying those plant features that should be designed to withstand the effects of the SSE.

III. REVIEW PROCEDURES

Selection and emphasis of various aspects of the areas covered by this review plan will be made by the reviewer on each case. The judgement on the areas to be given attention during the review is to be based on an inspection of the material presented, the similarity of the material to that recently reviewed on other plants, and whether items of special safety significance are involved.

Regulatory Guide 1.29, which identifies structures, systems, and components of light-water-cooled reactors on a functional basis, is the principal document used for identifying those plant features important to safety which, as a minimum, should be designed to seismic Category I requirements.

The staff review should establish whether the applicant has indicated compliance with Regulatory Guide 1.29 in the SAR. Where there are differences with respect to the Guide, these differences should be identified.

The information supplied by the applicant identifying seismic Category I structures, systems, and components is reviewed for completeness and to assure there is sufficient detail to permit identification of specific equipment. Where portions of a system are

classified seismic Category I, the boundary limits of that portion of the system designed to Category I requirements should be identified on the piping and instrumentation diagrams. In addition, where portions of a structure are classified seismic Category I, those portions of the building foundations and supports designed to Category I requirements should be identified on the plant arrangement drawings. The interfaces between components and associated support structures designed to seismic Category I requirements are then checked to assure compatability.

For systems which are partially seismic Category I, the Category I portion of the system should extend to the first seismic restraint beyond the isolation valves which isolate that part which is Category I from the non-seismic portion of the system.

In the event an applicant intends to take exception to Regulatory Guide 1.29 and has not provided an adequate justification for his proposed seismic classification, questions are prepared by the staff which may require additional documentation or analysis to establish an acceptable basis for his proposed seismic classification. Staff comments may also be prepared requesting clarification in order to assure a clear understanding of the seismic classification assigned to a system by the applicant.

If the staff's questions are not resolved in a satisfactory manner, a staff position is taken requiring conformance to Regulatory Guide 1.29.

IV. EVALUATION FINDINGS

The staff's review should verify that adequate and sufficient information is contained in the SAR and amendments to arrive at conclusions of the following type, which are to be included in the staff's safety evaluation report:

> "Structures, systems, and components important to safety that are required to withstand the effects of a safe shutdown earthquake and remain functional have been properly classified as seismic Category I items. These plant features are those necessary to assure (1) the integrity of the reactor coolant pressure boundary, (2) the capability to shut down the reactor and maintain it in a safe shutdown condition, and (3) the capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the guideline exposures of 10 CFR Part 100.
>
> "All other structures, systems, and components that may be required for operation of the facility are designed to other than seismic Category I requirements. Included in this classification are those portions of Category I systems which are not required to perform a safety function. Structures, systems, and components important to safety that are designed to withstand the effects of a safe shutdown earthquake and remain functional have been identified in an acceptable manner in Tables 3.X.X and 3.X.X, and on system piping and instrumentations diagrams.
>
> "The basis for acceptance in the staff's review has been conformance of the applicant's designs, design criteria and design bases for structures, systems and components important to safety with the Commission's regulations as set forth in General Design Criterion 2, and to Regulatory Guide 1.29, staff technical positions, and industry standards."

V. REFERENCES

1. 10 CFR Part 50, Appendix A, General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena."

2. 10 CFR Part 100, Appendix A, "Seismic and Geologic Siting Criteria for Nuclear Power Plants."

3. Regulatory Guide 1.29, "Seismic Design Classification."

4. ANSI N18.2a-1975, Revision and Addenda to ANSI N18.2, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," American National Standards Institute (1973).

5. ANS N212, "Nuclear Safety Criteria for the Design of Stationary Boiling Water Reactor Plants," Draft No. 4, Rev. 2, April 1974, ANS Standard Issued for Trial Use and Comment, American Nuclear Society (1974).

6. ANS N213, "Nuclear Safety Criteria for the Design of Stationary Gas Cooled Reactor Plants," Draft No. 9, Rev. 2, January 1974, ANS Standard Issued for Comment, American Nuclear Society (1974).

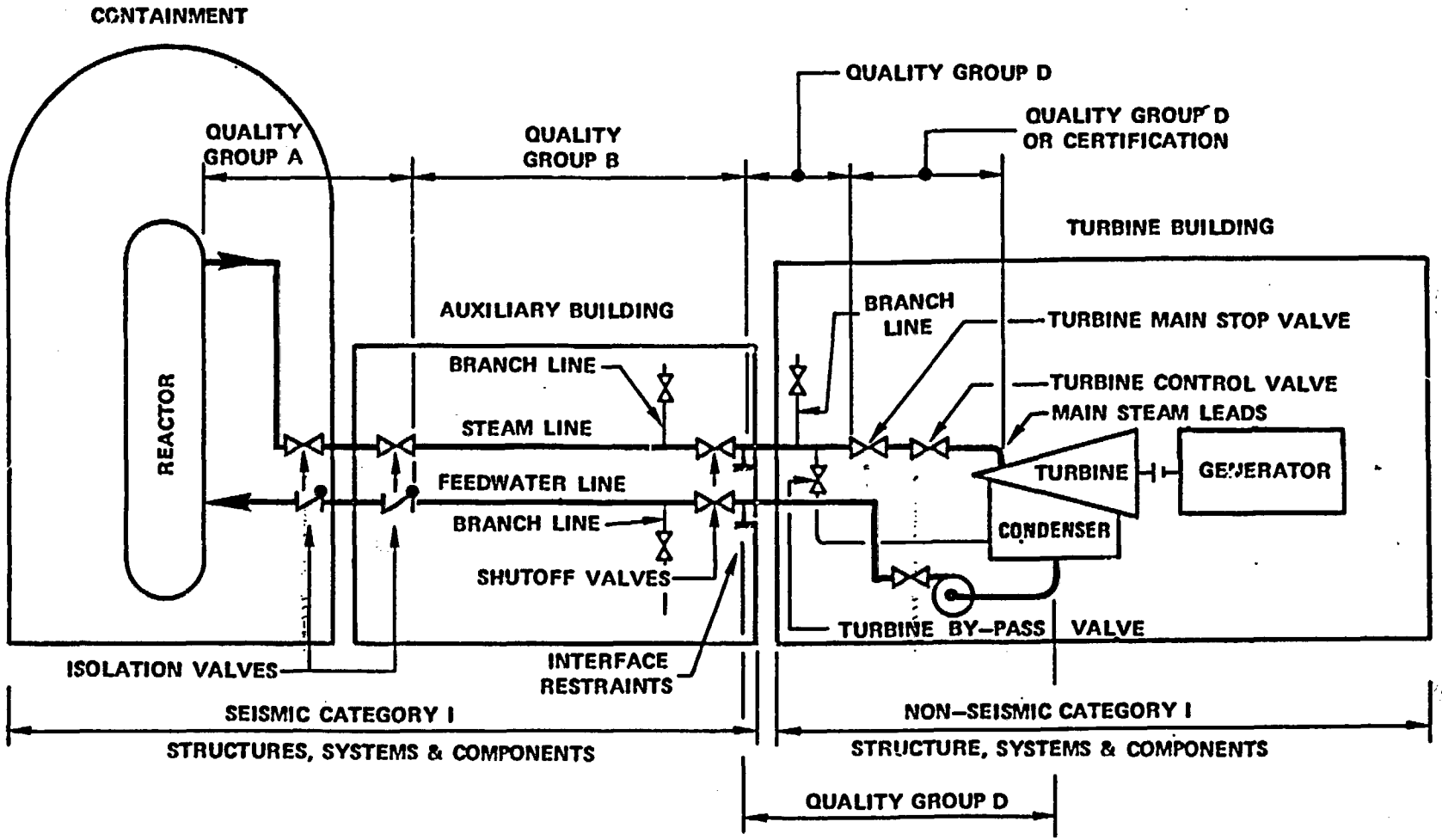

Figure 3-2.1 AEC Quality Group and Seismic Category Classifications Applicable to Power Conversion System Components in BWR/6 Plants.

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U.S. NUCLEAR REGULATORY COMMISSION
STANDARD REVIEW PLAN
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SECTION 3.2.2 SYSTEM QUALITY GROUP CLASSIFICATION

REVIEW RESPONSIBILITIES

Primary - Reactor Systems Branch (RSB)

Secondary - Containment Systems Branch (CSB)
Auxiliary and Power Conversion Systems Branch (APCSB)
Effluent Treatment Systems Branch (ETSB)

I. AREAS OF REVIEW

Nuclear power plant systems and components important to safety should be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function to be performed.

The RSB reviews the applicant's classification system for pressure-retaining components such as pressure vessels, heat exchangers, storage tanks, pumps, piping, and valves in fluid systems important to safety, and the assignment by the applicant of quality groups to those sections of systems required to perform safety functions. Where required, specific information or assistance may be required from the EICSB to review electrical and instrumentation systems needed for functioning of plant features important to safety. This review is done for both construction permit (CP) and operating license (OL) applications. Excluded from this review are: structures; parts such as pump motors, shafts, seals, impellers, packing, and gaskets; containment; fuel and reactor core internals; mechanical, electrical, and instrumentation systems and valve actuation devices; vessel and piping supports and snubbing devices.

The applicant presents data in his safety analysis report (SAR) in the form of a table which identifies the fluid systems important to safety; the system components such as pressure vessels, heat exchangers, storage tanks, pumps, piping, and valves; the associated quality group classification, ASME Code and code class; and the quality assurance requirements. In addition, the applicant presents on suitable piping and instrumentation diagrams the system quality group classifications.

The CSB reviews, in SAR Section 6.2.5, the detailed system design of fluid systems designated AEC Quality Group B, which are provided for the control of combustible gas concentrations in containment following a loss of coolant accident.

The APCSB reviews, in SAR Sections 9 and 10, the detailed system design of auxiliary fluid systems important to safety that are designated AEC Quality Groups B and C.

The ETSB reviews, in SAR Sections 11.2 and 11.3, the detailed design of liquid, gaseous, and solid radioactive waste systems designated AEC Quality Groups C and D.

The RSB will review the detailed system design of engineered safeguards systems that are designated AEC Quality Group B.

The branches that have secondary review responsibility will confirm that the quality group classifications of systems and components within their review scopes are acceptable. If there are systems or components other than those identified by the RSB that are deemed to be important to safety this information should be transmitted to the RSB.

II. ACCEPTANCE CRITERIA

1. 10 CFR Part 50, Appendix A, General Design Criterion 1, "Quality Standards and Records." This criterion requires that structures, systems, and components important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed.

2. 10 CFR Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants." This appendix establishes quality assurance requirements for the design, construction, and operation of those structures, systems, and components of nuclear power plants important to safety.

3. Regulatory Guide 1.26, "Quality Group Classification and Standards." This Regulatory Guide describes an acceptable method for determining quality standards for Quality Group B, C, and D water- and steam-containing components important to safety of water-cooled nuclear power plants. The applicant may use the AEC Group Classification system identified in the Regulatory Guide 1.26 or, alternately, the corresponding ANS classification system of Safety Classes which can be cross-referenced with the classification groups in Regulatory Guide 1.26. Clarification of Regulatory Guide 1.26 provisions with respect to boiling water reactor plant main steam and feedwater systems, and acceptable alternate provisions for these systems are given in branch technical positions attached to this plan.

III. REVIEW PROCEDURES

Selection and emphasis of various aspects of the areas covered by this review plan will be made by the reviewer on each case. The judgement on the areas to be given attention during the review is to be based on an inspection of the material presented, the similarity of the material to that recently reviewed on other plants, and whether items of special safety significance are involved.

Section 50.55a of 10 CFR Part 50 identifies those ASME Section III, Code Class 1 components of light-water-cooled reactors important to safety which are part of the reactor coolant pressure boundary. These components are designated in Regulatory Guide 1.26 as Quality Group A. In addition, Regulatory Guide 1.26 identifies, on a functional basis, water- and steam-containing components of those systems important to safety which are

Quality Groups B and C. Quality Group D applies to water- and steam-containing components of systems that are less important to safety.

There are also systems of light-water-cooled reactors important to safety that are not identified in Regulatory Guide 1.26 and which the staff considers should be classified Quality Group C. Examples of these systems are: diesel fuel oil system; diesel generator cooling, lubricating oil, and air startup systems; instrument and service air systems required to perform a safety function; and certain ventilation systems. Gas treatment systems which are considered as engineered safeguards systems should be classified Quality Group B.

The information supplied in the application identifying fluid systems important to safety is reviewed for completeness, and the quality group classification, ASME Code and code class, and quality assurance requirements of each individual major component are checked for compliance with the above criteria. The various modes of system operation are checked to assure that the assigned AEC quality groups are acceptable.

The piping and instrumentation diagrams are reviewed to assure that the applicant has delineated in detail the system quality group classification boundaries for systems important to safety. Each individual line on a diagram is checked to assure the accuracy of the assigned quality group classification, including branch lines such as vents, drains, and sample lines. Changes in quality group classification are permitted normally only at valve locations, with the valve assigned the higher classification. A change in quality group classification with no valve present is permitted only when it can be demonstrated that the safety function of the system is not impaired by a failure on the lower-classification side of the boundary.

The following fluid systems important to safety for pressurized water reactor (PWR) and boiling water reactor (BWR) plants are reviewed by the RSB with regard to quality group classification.

FLUID SYSTEMS IMPORTANT TO SAFETY FOR PWR PLANTS

Reactor Coolant System
Emergency Core Cooling System
Containment Spray System
Chemical and Volume Control System
Boron Thermal Regeneration System
Boron Recycle System
Residual Heat Removal System
Component Cooling Water System[2/]
Spent Fuel Pool Cooling and Cleanup System[2/]
Sampling System[3/]
Service Water System[2/]
Compressed Air System[2/]
Diesel Fuel Oil System

Diesel Generator Auxiliary Systems
Main Steam System[3/]
Feedwater System[3/]
Auxiliary Feedwater System
Liquid Waste Processing System[1/]
Gaseous Waste Processing System[1/]
Containment Cooling System
Containment Purge System
Ventilation Systems for Areas such as Control Room and Engineered Safety Features Rooms
Fire Protection System[1/2/]
Combustible Gas Control System
Condensate Storage System[1/]

FLUID SYSTEMS IMPORTANT TO SAFETY FOR BWR PLANTS

Reactor Recirculation System
Main Steam System (up to but not including the turbine)
Feedwater System[3/]
Relief Valve Discharge Piping
Control Rod Drive Hydraulic System
Standby Liquid Control System
Reactor Water Cleanup System
Liquid Radwaste System[1/]
Gaseous Radwaste System (Off-gas)[1/]
Fuel Pool Cooling and Cleanup System[2/]
Sampling System[3/]
Residual Heat Removal System
High Pressure Core Spray System
Low Pressure Core Spray System
Reactor Core Isolation Cooling System
RHR Service Water System
Emergency Equipment Service Water System
Compressed Air System[2/]
Diesel Generator Auxiliary Systems
Standby Gas Treatment System
Combustible Gas Control System
Containment Cooling System
Main Steam Line Isolation Valve Sealing System
Condensate and Refueling Water Storage System[2/]
Ventilation Systems for Areas such as Control Room and Engineered Safety Features Rooms
Fire Protection System[1/2/]

[1/] On some plants this system may be non-safety-related, providing it complies with the requirements of Regulatory Guide 1.26.

[2/] Portions of the system that perform a safety-related function.

[3/] Portions of the system to outermost containment isolation valve.

Provisions applicable to BWR main steam and feedwater system quality group and seismic classifications, for those portions of the system on the turbine side of the containment isolation valves, are given in Branch Technical Positions RSB No. 3-1 and 3-2, attached to this plan.

In the event an applicant intends to take exception to Regulatory Guide 1.26 and has not provided adequate justification for his proposed quality group classification, questions are prepared by the staff which may require additional documentation or an analysis to establish an acceptable basis for his proposed quality group classification. Staff comments may also be prepared requesting clarification, in order to assure a clear understanding of the quality group classifications assigned to a system by the applicant.

Exceptions and alternatives to the specified quality group classifications of Regulatory Guide 1.26 are unacceptable unless "equivalent quality level" is justified. In such cases, justification can be demonstrated if: the component is classified to meet the requirements of a higher group classification than specified in Regulatory Guide 1.26 or alternative design rules are based on the use of a more conservative design; the extent of component nondestructive examination is equal to or greater than required by the specified code; and the quality assurance requirements of Appendix B, 10 CFR Part 50 are met.

If the staff's questions are not resolved in a satisfactory manner, a staff position is taken requiring conformance to Regulatory Guide 1.26.

IV. EVALUATION FINDINGS

The staff's review should verify that adequate and sufficient information is contained in the SAR and amendments to arrive at a conclusion of the following type, which is to be included in the staff's Safety Evaluation report:

> "Fluid system pressure-retaining components important to safety will be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function to be performed. Water-and steam-containing components which are part of the reactor coolant pressure boundary and other fluid systems important to safety, where reliance is placed on these systems (1) to prevent or mitigate the consequences of accidents and malfunctions originating within the reactor coolant pressure boundary, (2) to permit shutdown of the reactor and maintenance in the safe shutdown condition, and (3) to contain radioactive material, have been classified in an acceptable manner in Tables 3.X.X and 3.X.X and on system piping and instrumentation diagrams.
>
> "The basis for acceptance in the staff's review has been conformance of the applicant's designs, design criteria, and design bases for pressure-retaining components such as pressure vessels, heat exchangers, storage tanks, pumps, piping and valves in fluid systems important to safety with the Commission's regulations

as set forth in General Design Criterion 1, with the requirements of the Codes specified in Section 50.55a of 10 CFR Part 50, with Regulatory Guide 1.26, and with staff technical positions and industry standards."

V. REFERENCES

1. 10 CFR Part 50, Appendix A, General Design CRiterion 1, "Quality Standards and Records."

2. 10 CFR Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants."

3. Regulatory Guide 1.26, "Quality Group Classifications and Standards."

4. ANSI N18.2a-1975, Revision and Addenda to ANSI N18.2, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," American National Standards Institute (1973).

5. ANS N212, "Nuclear Safety Criteria for the Design of Stationary Boiling Water Reactor Plants," Draft No. 4, Rev. 2, April 1974, ANS Standard Issued for Trial Use and Comment, American Nuclear Society (1974).

6. ANS N213, "Nuclear Safety Criteria for the Design of Stationary Gas Cooled Reactor Plants," Draft No. 9, Rev. 2, January 1974, ANS Standard Issued for Comment, American Nuclear Society (1974).

7. ASME Boiler and Pressure Vessel Code, 1974 Edition, Section III, "Nuclear Power Plant Components," American Society of Mechanical Engineers (1974).

8. ASME Boiler and Pressure Vessel Code, 1974 Edition, Section VIII, Division 1, "Pressure Vessels," American Society of Mechanical Engineers (1974).

9. ANSI B31.1-1973, "power Piping," American National Standards Institute (1973).

10. API Standard 620, Fifth Edition, "Recommended Rules for Design and Construction of Large, Welded, Low-Pressure Storage Tanks," Americal Petroleum Institute (1973).

11. API Standard 650, Fifth Edition, "Welded Steel Tanks for Oil Storage," American Petroleum Institute (1973).

12. AWWA D100-73, "AWWA Standard for Steel Tanks-Standpipes, Reservoirs, and Elevated Tanks for Water Storage," American Water Works Association (1973).

13. ANSI B96.1-1973, "Specification for Welded Aluminum-Alloy Field-Erected Storage Tanks," American National Standards Institute (1973).

14. Branch Technical Position - RSB No. 3-1, "Classification of Main Steam Components Other Than the Reactor Coolant Pressure Boundary for BWR Plants."

15. Branch Technical Position - RSB No. 3-2, "Classification of BWR/6 Main Steam and Feedwater Components Other Than the Reactor Coolant Pressure Boundary."

BRANCH TECHNICAL POSITION - RSB NO. 3-1

CLASSIFICATION OF MAIN STEAM COMPONENTS OTHER THAN THE REACTOR COOLANT PRESSURE BOUNDARY FOR BWR PLANTS

A. BACKGROUND

A pipe classification of "D + QA" for main steam line components of BWR plants was proposed by the General Electric Company in 1971 as an alternative to Quality Group B and has been accepted by the staff in a number of licensing case reviews.

However, we have recently identified a number of potential problems which are applicable to main steam lines of BWR plants. These problems relate to postulated breaks in high-energy fluid-containing lines outside the containment. The criteria pertaining to protection required for structures, systems, and components outside containment from the effects of postulated pipe breaks, as contained in the Director of Licensing's letter to utilities dated July 12, 1973, reference ASME Section III, Class 2, which corresponds to AEC Quality Group B.

The recent ASME Code Section XI revision contains in-service inspection requirements for Class 2 components. Steam lines classified as "D + QA" could be intrepreted to be exempt from these inspection requirements. Such interpretations would be contrary to the intent of the code and inconsistent with requirements of the AEC Codes and Standards rule, Section 50.55a of 10 CFR Part 50.

Furthermore, the applicability of the following AEC Regulatory Guides and Regulations, as they relate to ASME Class 2 components is not always clearly identified or implemented in case applications wherever "D + QA" classification is adopted:

1. Regulatory Guide 1.51, "In-service Inspection of ASME Code Class 2 and 3 Nuclear Power Plant Components."

2. Regulatory Guide 1.48, "Design Limits and Loading Combinations for Seismic Category I Fluid System Components."

3. Regulatory Guide 1.26, "Quality Group Classifications and Standards."

4. 10 CFR § 50.55a, "Codes and Standards for Nuclear Power Plants."

5. 10 CFR Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants."

In view of the foregoing, we find it necessary to clarify the quality group classification criteria for main steam components for BWR plants.

B. BRANCH TECHNICAL POSITION

The main steam line components of BWR plants should conform to the criteria listed in the attached Table 3-1.1.

C. REFERENCES

1. Letter of March 22, 1973, J.A. Hinds to J.M. Hendrie.

2. Letters of August 13, 1973 and November 26, 1973, J.M. Hendrie to J.A. Hinds.

Table 3-1.1

CLASSIFICATION REQUIREMENTS FOR MAIN STEAM COMPONENTS OTHER THAN THE REACTOR COOLANT PRESSURE BOUNDARY

Item	System or Component	Classification Quality Group	
1.	Main Steam Line from 2nd Isolation Valve to Turbine Stop Valve.	B	
2.	Main Steam Line Branch Lines to First Valve.	B	
3.	Main Turbine Bypass Line to Bypass Valve.	B	
4.	First Valve in Branch Lines Connected to Either Main Steam Lines or Turbine Bypass Lines.	B	
5.	a. Turbine Stop Valves, Turbine Control Valves, and Turbine Bypass Valves.	D + QA or Certification	1/ 2/
	b. Main Steam Leads from Turbine Control Valves to Turbine Casing.	D + QA or Certification	1/3/ 2/

1/ The following requirements shall be met in addition to the Quality Group D requirements:

1. All cast pressure-retaining parts of a size and configuration for which volumetric examination methods are effective shall be examined by radiographic methods by qualified personnel. Ultrasonic examination to equivalent standards may be used as an alternate to radiographic methods.

2. Examination procedures and acceptance standards shall be at least equivalent to those specified as supplementary types of examination in ANSI B31.1-1973, Par. 136.4.

2/ The following qualification shall be met with respect to the certification requirements:

1. The manufacturer of the turbine stop valves, turbine control valves, turbine bypass valves, and main steam leads from turbine control valves to the turbine casing shall utilize quality control procedures equivalent to those defined in General Electric Publication GEZ-4982A, "General Electric Large Steam Turbine - Generator Quality Control Program."

2. A certification shall be obtained from the manufacturer of these valves and steam leads that the quality control program so defined has been accomplished.

3/ The following requirements shall be met in addition to the Quality Group D requirements:

1. All longitudinal and circumferential butt weld joint shall be radiographed (or ultrasonically tested to equivalent standards). Where size or configuration does not permit effective volumetric examination, magnetic particle or liquid penetrant examination may

Table 3-1.1 (cont'd)

be substituted. Examination procedures and acceptance standards shall be at least equivalent to those specified as supplementary types of examinations, Paragraph 136.4 in ANSI B31.1-1973.

2. All fillet and socket welds shall be examined by either magnetic particle or liquid penetrant methods. All structural attachment welds to pressure retaining materials shall be examined by either magnetic particle or liquid penetrant methods. Examination procedures and acceptance standards shall be at least equivalent to those specified as supplementary types of examinations, Paragraph 136.4 in ANSI B31.1-1973.

3. All inspection records shall be maintained for the life of the plant. These records shall include data pertaining to qualification of inspection personnel, examination procedures, and examination results.

BRANCH TECHNICAL POSITION - RSB NO. 3-2

CLASSIFICATION OF BWR/6 MAIN STEAM AND FEEDWATER COMPONENTS
OTHER THAN THE REACTOR COOLANT PRESSURE BOUNDARY

A. BACKGROUND

At various times the AEC staff has discussed with the General Electric Company the subject of appropriate classification requirements in boiling water reactor (BWR) plants for main steam system components. These discussions have included consideration of components that are (a) not classified as safety-related items but are located downstream of the isolation valves, (b) not specifically designed to seismic Category I standards, and (c) not housed in seismic Category I structures.

To date, BWR plant reviews have resulted in various approaches for different individual applications. While these different approaches have resulted in acceptable levels of safety in each case, they have required time-consuming case-by-case reviews. The GESSAR BWR/6 application, under review as part of our standardization program, includes this portion of the BWR plant.

In the course of the GESSAR review, we have identified a systematic basis for classification of such components that will result in an acceptable and uniform design basis for the main steam lines (MSL) and main feedwater lines (MFL) in BWR/6 plants.

B. BRANCH TECHNICAL POSITION

The main steam and feedwater system components of BWR/6 plants should be classified in accordance with BTP-RSB No. 3-1, or alternately, in accordance with the attached Table 3-2.1. The classifications indicated are acceptable alternates to the guidelines currently specified in Regulatory Guide 1.26 and Regulatory Guide 1.29.

As an additional requirement, a suitable interface restraint should be provided at the point of departure from the Class I structure where the interface exists between the safety and nonsafety-related portions of the MSL and MFL.

A sketch is attached (Figure 3-2.1) to clarify the specified alternate classification system.

C. REFERENCES

1. Letter of April 19, 1974, J.M. Hendrie to J.A. Hinds.

Table 3-2.1

CLASSIFICATION REQUIREMENTS FOR BWR/6 MAIN STEAM AND FEEDWATER SYSTEM COMPONENTS OTHER THAN THE REACTOR COOLANT PRESSURE BOUNDARY

ITEM	SYSTEM OR COMPONENT	QUALITY GROUP CLASSIFICATION
1.	Main Steam Line (MSL) from second isolation valve to and including shutoff valve.	B
2.	Branch lines of MSL between the second isolation valve and the MSL shutoff valve, from branch point at MSL to and including the first valve in the branch line.	B
3.	Main feedwater line (MFL) from second isolation valve and including shutoff valve.	B
4.	Branch lines of MFL between the second isolation valve and the MFL shutoff valve, from the branch point at MFL to and including the first valve in the branch line.	B
5.	Main steam line piping between the MSL shutoff valve and the turbine main stop valve.	D (1)
6.	Turbine bypass piping..	D
7.	Branch lines of the MSL between the MSL shutoff valve and the turbine main stop valve.	D
8.	Turbine valves, turbine control valves, turbine bypass valves, and main steam leads from the turbine control valves to the turbine casing.	D (1,2) or Certification (3)
9.	Feedwater system components beyond the MFL shutoff valve.	D

(1) All inspection records shall be maintained for the life of the plant. These records shall include data pertaining to qualification of inspection personnel, examination procedures, and examination results.

(2) All cast pressure-retaining parts of a size and configuration for which volumetric methods are effective shall be examined by radiographic methods by qualified personnel. Ultrasonic examination to equivalent standards may be used as an alternate to radiographic methods. Examination procedures and acceptance standards shall be at least equivalent to those defined in Paragraph 136.4, "Examination Methods of Welds - Non-Boiler External Piping," ANSI B31.1-1973.

(3) The following qualifications shall be met with respect to the certification requirements:

1. The manufacturer of the turbine stop valves, turbine control valves, turbine bypass valves, and main steam leads from turbine control valves to the turbine casing shall utilize quality control procedures equivalent to those defined in General Electric Publication GEZ-4982A, "General Electric Large Steam Turbine-Generator Quality Control Program."

2. A certification shall be obtained from the manufacturer of these valves and steam leads that the quality control program so defined has been accomplished.

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SECTION 3.3.1 WIND LOADINGS

REVIEW RESPONSIBILITIES

Primary - Structural Engineering Branch (SEB)

Secondary - Hydrology-Meteorology Branch (HMB)

I. AREAS OF REVIEW

The following areas relating to the design of structures that have to withstand the effects of the design wind* specified for the plant are reviewed to assure conformance with the requirements of General Design Criterion 2.

1. The design wind velocity and its recurrence interval, the velocity variation with height, and the applicable gust factors are reviewed from the standpoint of use in defining the input parameters for the structural design criteria appropriate to account for wind loadings. The bases for the selection and the values of these parameters are within the review responsibility of the Hydrology-Meteorology Branch (HMB) as stated in SRP Sections 2.3.1 and 2.3.2.

2. The procedures that are utilized to transform the design wind velocity into an effective pressure applied to structures are reviewed taking into consideration the geometrical configuration and physical characteristics of the structures and the distribution of wind pressure on the structures.

II. ACCEPTANCE CRITERIA

The acceptance criteria for the areas of review are as follows:

1. General Design Criterion 2 with regard to the design bases for protection of structures important to safety against wind loadings.

2. The acceptance criteria for the design wind velocity and its recurrence interval, the velocity variation with height, the applicable gust factors, and the bases for determining these site-related parameters, are established by the Hydrology-Meteorology Branch (HMB) and are contained in SRP Sections 2.3.1 and 2.3.2. The approved values of these parameters should serve as basic input to the review and evaluation of the structural design procedures.

*Referred to as 100-year return period "fastest mile of wind" in SRP Section 2.3.1.

3. The procedures utilized to transform the wind velocity into an effective pressure to be applied to structures and parts and portions of structures, as delineated in ANSI A58.1, "Building Code Requirements for Minimum Design Loads in Buildings and Other Structures" (Ref. 3), are acceptable. In particular, the procedures utilized are acceptable if found in accordance with the following:

For a design wind velocity of V_{30} mph specified at a height of 30 feet above the ground, the velocity pressure, q_{30}, is given by:

$$q_{30} = 0.00256\ V_{30}^2 \text{ psf}$$

The effective pressure for structures, q_F, and for portions thereof, q_p, at various heights above the ground should be in accordance with Table 5 and Table 6 of ANSI A58.1, respectively. Since most nuclear power plants are located in relatively open country, Exposure C, as defined in ANSI A58.1, should be selected for both tables.

Depending upon the structure geometry and physical configuration, pressure coefficients may be selected in accordance with Section 6.4 of ANSI A58.1. Geometrical shapes that are not covered in this document are reviewed on a case-by-case basis. ASCE Paper No. 3269, "Wind Forces on Structures" (Ref. 2), may be used to obtain the effective wind pressures for cases which ANSI A58.1 does not cover.

III. REVIEW PROCEDURES

The reviewer selects and emphasizes material from the review procedures described below as may be appropriate for a particular case.

1. The site-related parameters described in subsection I.1 are reviewed by the Hydrology-Meteorology Branch (HMB) under SRP Sections 2.3.1 and 2.3.2. The structural reviewer examines the approved values of these parameters to assure himself that the procedures utilized in designing the structures to withstand the specified wind loadings are appropriate and applicable.

2. After the applicability of the site-related parameters is established, the reviewer proceeds with the review of the structural aspects of wind design. The procedures utilized by the applicant to transform wind velocities into effective pressures are reviewed and compared with those procedures delineated in the ANSI A58.1 standard.

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information has been provided to satisfy the requirements of this SRP section, and concludes that his evaluation is sufficiently complete and adequate to support the following type of conclusive statement to be included in the staff's safety evaluation report:

"The procedures utilized to determine the loadings on structures induced by the design wind specified for the plant are acceptable since these procedures provide a conservative basis for engineering design to assure that the structures will withstand such environmental forces.

"The use of these procedures provides reasonable assurance that in the event of design basis winds, the structural integrity of the plant structures that have to be designed for the design wind will not be impaired and, in consequence, safety-related systems and components located within these structures are adequately protected and will perform their intended safety functions if needed. Conformance with these procedures is an acceptable basis for satisfying, in part, the requirements of General Design Criterion 2."

V. REFERENCES

1. 10 CFR Part 50, Appendix A, General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena."

2. ASCE Paper No. 3269, "Wind Forces on Structures," Transactions of the American Society of Civil Engineers, Vol. 126, Part II (1961).

3. ANSI A58.1, "Building Code Requirements for Minimum Design Loads in Buildings and Other Structures," Committee A58.1, American National Standards Institute.

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SECTION 3.3.2 TORNADO LOADINGS

REVIEW RESPONSIBILITIES

Primary - Structural Engineering Branch (SEB)

Secondary - Hydrology-Meteorology Branch (HMB)
Accident Analysis Branch (AAB)
Auxiliary Systems Branch (ASB)

I. AREAS OF REVIEW

The following areas relating to the design of structures that have to withstand the effects of the design basis tornado specified for the plant are reviewed to assure conformance with the requirements of General Design Criterion 2.

1. The design parameters applicable to the tornado, including the tornado wind translational and tangential velocities, the tornado-generated pressure differential and its associated time interval, and the spectrum of tornado-generated missiles including their characteristics, are reviewed from the standpoint of use in defining the input parameters for the structural design criteria appropriate to account for tornado loadings. The bases for the selection and the values of these parameters are within the review responsibility of the Hydrology-Meteorology Branch (HMB) and the Accident Analysis Branch (AAB), as stated in SRP Sections 2.3.1, 2.3.2 and 3.5.1.4.

2. The procedures that are utilized to transform the tornado parameters into effective loads on structures are reviewed, including the following:
   a. The transformation of the tornado wind into an effective pressure applied to structures, taking into consideration the geometrical configuration and physical characteristics of the structures and the distribution of wind pressure on the structures.
   b. If venting of a structure is utilized, the procedures for transforming the tornado-generated differential pressure into an effective reduced pressure are reviewed, upon request, by the Auxiliary Systems Branch (ASB).
   c. The transformation of tornado-generated missile loadings, which are considered impactive dynamic loads, into effective loads.
   d. The combination of the above individual loadings in a manner that will produce the most adverse total tornado effect on structures.

3. The information provided to demonstrate that failure of any structure or component not designed for tornado loads will not affect the capability of other structures or components to perform necessary safety functions.

II. ACCEPTANCE CRITERIA

The acceptance criteria for the areas of review are as follows:

1. General Design Criterion 2 with regard to the design bases for protection of structures important to safety against tornado wind loadings.

2. The acceptance criteria for the tornado wind velocity, the differential pressure and its associated time interval, the spectrum of tornado-generated missiles and their characteristics, and the bases for determining these parameters, are established by the Hydrology-Meteorology Branch (HMB) and the Accident Analysis Branch (AAB) as described in SRP Sections 2.3.1, 2.3.2 and 3.5.1.4. The approved values of these paramenters should serve as basic input to the review and evaluation of the structural design procedures.

3. The acceptance criteria for the procedures utilized to transform the tornado parameters into effective loadings on structures are as follows:
   a. For transforming the tornado wind velocity into an effective pressure applied to structures, the criteria delineated in either the American Society of Civil Engineers (ASCE) Paper No. 3269, "Wind Forces on Structures" (Ref. 1), or in ANSI A58.1, "Building Code Requirements for Minimum Design Loads in Buildings and Other Structures" (Ref. 2), are, in general, acceptable. In particular, the following shall apply:
      (i) The maximum velocity pressure, p, should be based upon the maximum tornado velocity, V, using the following formula:
      $$p = 0.00256\ V^2 \text{ psf, in which V is in mph.}$$
      (ii) The velocity pressure should be assumed constant with height.
      (iii) The maximum velocity pressure, p, applies at the radius of the tornado funnel at which the maximum velocity occurs. The tangential velocity varies with the radial distance from the center of the tornado core. The variation may be considered in accordance with that described in the paper, "Tornado Resistant Design of Nuclear Power Plants" (Ref. 4).
      (iv) For calculating velocity pressures on external surfaces of structures, on external portions thereof, and on internal surfaces, where there are openings in the structure, appropriate shape coefficients shall be used in accordance with ASCE Paper No. 3269 (Ref. 1). Gust factors may be taken as unity.
   b. If venting of a structure is adopted as a design measure to permit transforming the tornado-generated differential pressure into an effective reduced pressure, the acceptance criteria are established on a case-by-case basis, upon request, by the Auxiliary Systems Branch (ASB).

c. The acceptance criteria for transforming the tornado-generated missile impact into an effective or equivalent static load on structures are delineated in subsection II of SRP Section 3.5.3.

d. Having established the effective loads for each of the above three individual tornado-generated effects, the combination thereof should then be determined in a conservative manner for each particular structure, as applicable. An acceptable method of combining these effects is as follows:

(i) $W_t = W_w$

(ii) $W_t = W_p$

(iii) $W_t = W_m$

(iv) $W_t = W_w + .5\ W_p$

(v) $W_t = W_w + W_m$

(vi) $W_t = W_w + .5\ W_p + W_m$

where: W_t ..... total tornado load,
W_w ..... tornado wind load,
W_p ..... tornado differential pressure load, and
W_m ..... tornado missile load.

For each particular structure or portion thereof, the most adverse of the above combinations should be used, as appropriate.

These combined effects constitute the total tornado load which should then be combined with other loads as specified in SRP Sections 3.8.1, 3.8.4 and 3.8.5.

4. The information provided to demonstrate that failure of any structure or component not designed for tornado loads will not affect the capability of other structures or components to perform necessary safety functions, is acceptable if found in accordance with either of the following:

a. The postulated collapse or structural failure of structures and components not designed for tornado loads, including missiles, can be shown not to result in any structural or other damage to safety-related structures or components.

b. Safety-related structures are designed to resist the effects of the postulated structural failure, collapse, or generation of missiles from structures and components not designed for tornado loads.

III. REVIEW PROCEDURES

The reviewer selects and emphasizes material from the review procedures described below, as may be appropriate for a particular case.

1. The site-related parameters described in subsection I.1. are reviewed by the Hydrology-Meteorology Branch (HMB) and the Accident Analysis Branch (AAB) in

accordance with SRP Sections 2.3.1, 2.3.2 and 3.5.1.4. The structural reviewer examines the approved values of these parameters to assure himself that the design basis and procedures utilized in designing structures to withstand tornado loads are appropriate and applicable.

2. After the applicability of the site-related parameters is established, the reviewer proceeds with his review of the structural aspects of tornado design in the following manner:
   a. The procedures utilized by the applicant to transform tornado wind velocities into effective pressures are reviewed and compared with those procedures delineated in either ASCE Paper No. 3269 or in ANSI A58.1, whichever is selected, and, in particular, with the acceptance criteria delineated in subsection II.3.a.
   b. Where venting is utilized, procedures for transforming the tornado-generated differential pressure into an effective reduced pressure are reviewed, upon request, by the Auxiliary Systems Branch (ASB).
   c. The treatment of tornado-generated missiles is covered in SRP Section 3.5.1.4 and the review procedures for design of missile barriers are described in SRP Section 3.5.3.
   d. After procedures for determining the individual tornado effects are reviewed, the manner in which these effects are then combined to arrive at the most adverse total tornado effect is reviewed and compared with the acceptance criteria delineated in subsection II.3.d. Other proposed methods which may depend upon the geometry and configuration of a particular structure are reviewed on a case-by-case basis.

3. The information provided to demonstrate that failure of any structure or component not designed for tornado loads will not affect the capability of other structures or components to perform necessary safety functions is reviewed to assure that one of the acceptance criteria of subsection II.4 is satisfied.

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information has been provided to satisfy the requirements of this SRP section, and concludes that his evaluation is sufficiently complete and adequate to support the following type of statement to be included in the staff's safety evaluation report:

"The procedures utilized to determine the loadings on structures induced by the design basis tornado specified for the plant are acceptable since these procedures provide a conservative basis for engineering design to assure that the structures withstand such environmental forces.

"The use of these procedures provides reasonable assurance that in the event of a design basis tornado, the structural integrity of the plant structures that have to be designed for tornadoes will not be impaired and, in consequence, safety-related systems and components located within these structures will be adequately protected and may be expected to perform necessary safety functions as required. Conformance with these procedures is an acceptable basis for satisfying, in part, the requirements of General Design Criterion 2."

V. REFERENCES

1. ASCE Paper No. 3269, "Wind Forces on Structures," Transactions of the American Society of Civil Engineers, Vol. 126, Part II (1961).

2. ANSI A58.1, "Building Code Requirements for Minimum Design Loads in Buildings and Other Structures," Committee A58.1, American National Standards Institute.

3. 10 CFR Part 50, Appendix A, General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena."

4. J. R. McDonald, K. C. Mehta and J. E. Minor, "Tornado-Resistant Design of Nuclear Power Plant Structures," Nuclear Safety, Vol. 15, No. 4, July-August 1974.

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SECTION 3.4.1 FLOOD PROTECTION

REVIEW RESPONSIBILITIES

Primary - Auxiliary Systems Branch (ASB)

Secondary - Hydrology and Meteorology Branch (HMB)
Instrumentation and Control Systems Branch (ICSB)
Structural Engineering Branch (SEB)
Power Systems Branch (PSB)

I. AREAS OF REVIEW

The ASB review of the plant flood protection includes all systems and components whose failure could prevent safe shutdown of the plant or result in uncontrolled release of significant radioactivity to assure conformance with the requirements of General Design Criterion 2. The facility design and equipment arrangements presented in the applicant's safety analysis report (SAR) are reviewed with respect to the following considerations: to identify the safety-related systems and components that must be protected against flooding; to determine the capabilities of structures housing safety-related systems or equipment to withstand flood conditions, i.e., the relationship between structure elevation and flood elevation including waves and wind effects as determined in the review described in SRP Section 2.4; to determine the adequacy of the isolation of redundant safety-related systems or equipment subject to flooding; to identify possible inleakage sources, such as cracks in structures not designed to withstand seismic events and exterior or access openings or penetrations in structures located at a lower elevation than the flood level and associated wave activity. The applicant's proposed technical specifications are reviewed for operating license applications, as they relate to areas covered in this SRP section.

The HMB reviews the underground drainage system in accordance with SRP Section 2.4.13. ASB uses the information provided by the HMB to assure that the integrated design of the underground drainage system is capable of performing its safety function.

The review of flood protection involves secondary evaluations performed by other branches. The conclusions of their evaluations will be used by the ASB to complete the overall evaluation of the subject area. The HMB verifies the elevations and coincident conditions determined for the various conditions of site flooding, including the adequacy of the type of flood protection utilized (SRP Section 2.4). The SEB determines the acceptability of the design analyses, procedures, and criteria used for structures that must withstand the effects of the design basis flood. The Instrumentation and Control Systems Branch and the Power Systems Branch will, upon request, verify the adequacy of instrumentation

needed for flood protection, including adequacy of detectors and alarms necessary to detect rising water levels within structures, and will evaluate the consequences of flooding on other safety-related instrumentation and electrical equipment in affected areas (SRP Section 7.6).

II. ACCEPTANCE CRITERIA

Acceptability of the flood protection measures described in the SAR, including related portions of Chapter 3 of the SAR, is based on specific general design criteria and regulatory guides and on the reviewer's independent evaluation and calculations with respect to area or component flooding. Listed below are specific criteria as they relate to flooding:

1. General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena," as related to system and components capable of withstanding flood conditions.

2. Regulatory Guide 1.102, "Flood Protection for Nuclear Power Plants," as related to the protection of structures, systems, and components important to safety from the effects of floods.

3. Branch Technical Position ASB 3-1 and MEB 3-1, as related to structures, systems and components capable of withstanding the effects of flooding from failures in fluid piping systems.

4. If safety-related structures are protected from below-grade groundwater seepage by means of a permanent dewatering system, then the system should be designed as a safety-r 'ated system and meet the single failure criterion requirements.

An additional basis for determining the acceptability of the facility will be the degree of similarity to previously approved plants with respect to means of providing flood protection.

For those areas of review identified in subsection I of this SRP section as being the responsibility of other branches, the acceptance criteria and their methods of application are contained in the SRP sections corresponding to those branches.

III. REVIEW PROCEDURE

The review procedures below are used during the construction permit (CP) review to determine that the design criteria and bases and the preliminary design as set forth in the preliminary safety analysis report meet the acceptance criteria given in subsection II. For the review of operating license (OL) applications the procedures are utilized to verify that the initial design criteria and bases have been appropriately implemented in the final design as set forth in the final safety analysis report. The reviewer will select and emphasize material from the paragraphs below as may be appropriate for a particular case.

Upon request from the primary reviewer, the secondary review branches will provide input for the areas of review stated in subsection I. The primary reviewer obtains and uses such input as required to assure that this review procedure is complete.

The general review procedures for OLs include a determination that the content and intent of the technical specifications prepared by the applicant are in agreement with the requirements developed as a result of the staff's review. Where necessary, the review will include requirements for system testing, minimum performance, and surveillance.

The review procedure consists of:

1. A determination from the SAR as to which systems and components are safety-related and should be protected against floods or flooded conditions.

2. An evaluation using the plant arrangement and layout drawings as to the various means to prevent flooding of safety-related systems or components, such as pumping systems, stoplogs, and watertight doors. The measures utilized are reviewed and coordinated with HMB to determine their ability to cope with the design basis flood conditions, as established in SRP Section 2.4.

3. An assessment of leakage, a determination if liquid-carrying systems could produce flooding, and an evaluation of the measures taken to protect safety-related equipment. A failure modes and effects analysis may be performed to determine that the flooding consequences resulting from failures of such liquid-carrying systems close to essential equipment will not preclude required functions of safety systems.

4. A review of the SAR to ascertain if safety-related systems or components are capable of normal function while completely or partially flooded.

5. A review of plant arrangement and layout drawings to determine if any safety-related equipment or components are located within individual compartments or cubicles which act as positive barriers against possible means of flooding.

6. Review plant structure design drawings to determine if any safety-related structures have been provided with a safety-related permanent dewatering system for control of ground water seepage. The dewatering system should be designed to safety grade requirements. In addition, see SRP Section 2.4.13.

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information has been provided and that his evaluation supports conclusions of the following type, to be included in the staff's safety evaluation report:

"The flood protection review included all systems and components whose failure could prevent safe shutdown of the plant and maintenance thereof or result in significant uncontrolled release of radioactivity. Based on the review of the applicant's proposed design criteria, design bases, and safety classification for safety-related systems, structures and components necessary for a safe plant shutdown during and following the flood condition, the staff concludes that the design of the facility for flood protection conforms to the Commission's regulations as set forth in General Design Criterion 2, "Design Bases for the Protection Against Natural Phenomena" and meets the guidelines recommended in Regulatory Guide 1.102, "Flood Protection for Nuclear Power Plants," and "Branch Technical Positions ASB 3-1 and MEB 3-1," and is acceptable.

V. REFERENCES

1. 10 CFR Part 50, Appendix A, General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena."

2. Regulatory Guide 1.102, "Flood Protection for Nuclear Power Plants."

3. Branch Technical Positions ASB 3-1, "Protection Against Postulated Piping Failures in Fluid Systems Outside Containment," attached to SRP Section 3.6.1, and MEB 3-1, "Postulated Break and Leakage Locations in Fluid System Piping Outside Containment," attached to SRP Section 3.6.2.

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SECTION 3.4.2 ANALYSIS PROCEDURES

REVIEW RESPONSIBILITIES

Primary - Structural Engineering Branch (SEB)

Secondary - Site Analysis Branch (SAB)

I. AREAS OF REVIEW

The following areas relating to the design of seismic Category I structures to withstand the effects of the flood or highest ground water specified for the plant are reviewed.

1. The design parameters of the flood or highest groundwater are reviewed from the standpoint of use in defining the input parameters for the structural design criteria appropriate to account for flood and groundwater loadings. Further, for plants where the flood level is higher than the proposed grade around the plant structures, the dynamic phenomena associated with such a flooding such as currents, wind waves, and their hydrodynamic effects, are similarly reviewed. The bases for these parameters are within the review responsibility of the Site Analysis Branch (SAB) as stated in Standard Review Plan 2.4.2.

2. The procedures that are utilized to transform the static and dynamic effects of the flood and highest ground water into effective loads applied to seismic Category I structures are reviewed.

II. ACCEPTANCE CRITERIA

The acceptance criteria for the areas of review are as follows:

1. The acceptance criteria for the flood or highest groundwater level, for establishing the dynamic effects of the flood where it is above the plant grade, and for the bases for determining these site-related and hydrodynamic parameters, are established by the Site Analysis Branch (SAB) as stated in Standard Review Plan 2.4.2.

2. In most situations, the flood level is below the proposed plant grade and only its hydrostatic effects need be considered. Unless the hydrostatic head associated with the flood or with the highest groundwater level is relieved by utilizing a drainage

and pumping system around the foundations of structures, it has to be considered as a structural load on the basement walls and foundation slab of the building. Another consideration in such a situation is to prevent any uplift or floating of the structure. The total bouyancy force may be based on the flood or highest groundwater head excluding wave action, if applicable. However, the lateral, overturning, and upward hydrostatic pressures acting on the side walls and on the foundation slab, respectively, which should be considered in the structural design of these elements, should be based on the total head including wave action, if any.

Where the flood level is above the proposed plant grade, the dynamic loads of wave action should be considered. Procedures for determining such dynamic loads are acceptable if they are in accordance with or similar to those delineated in the U.S. Army Coastal Engineering Research Center, Technical Report No. 4 (Ref. 1), as applicable. Other methods are reviewed on a case-by-case basis.

III. REVIEW PROCEDURES

The reviewer selects and emphasizes material from the review procedures described below as may be appropriate for a particular case:

1. The site-related and hydrodynamic parameters described in Section II.1 of this plan are reviewed by the Site Analysis Branch (SAB) and are covered in Standard Review Plan 2.4.2. The structural reviewer examines the approved values of these parameters to assure that the procedures utilized in designing the structures to withstand the specified flood loadings are appropriate and applicable.

2. After the applicability of the site-related and hydrodynamic parameters is established, the reviewer proceeds with his review of the structural aspects of the design for flood or groundwater. The procedures utilized by the applicant to determine effective flood loads are reviewed and compared with those procedures delineated in Section II.2 of this plan.

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information has been provided to satisfy the requirements of this review plan, and concludes that his evaluation is sufficiently complete and adequate to support the following type of conclusive statement to be included in the staff's safety evaluation report:

"The procedures utilized to determine the loadings on seismic Category I structures induced by the design flood or highest groundwater level specified for the plant are acceptable since these procedures provide a conservative basis for engineering design to assure that the structures will withstand such environmental forces.

"The use of these procedures provides reasonable assurance that in the event of floods or high groundwater, the structural integrity of the plant seismic Category I structures will not be impaired and, in consequence, seismic Category I systems and components located within these structures will be adequately protected and may be

expected to perform necessary safety functions, as required. Conformance with these design procedures is an acceptable basis for satisfying, in part, the requirements of General Design Criterion 2."

V. REFERENCES

1. U.S. Army Coastal Engineering Research Center, Technical Report No. 4, "Shore Protection, Planning and Design," 3rd Edition, 1966.

2. 10 CFR Part 50, Appendix A, General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena."

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SECTION 3.5.1.1 INTERNALLY GENERATED MISSILES (OUTSIDE CONTAINMENT)

REVIEW RESPONSIBILITIES

Primary - Auxiliary Systems Branch (ASB)

Secondary - Structural Engineering Branch (SEB)

I. AREAS OF REVIEW

The ASB reviews all structures, systems and components (SSC) provided to support the reactor facility that require protection from internally generated missiles (outside containment) to assure conformance with the requirements of General Design Criterion 4. The review includes missile sources and internally generated missiles associated with component overspeed failures and missiles that could originate from high-pressure system ruptures.

The ASB reviews the functional operations and performance requirements for all structures, systems, and components outside containment and identifies the SSC that are necessary for the safe shutdown of the reactor facility and the SSC whose failure could result in a significant release of radioactivity. All SSC will be reviewed to assure adequate protection from internally generated missiles if the SSC are necessary to perform functions required for attaining and maintaining a safe shutdown condition or if the SSC are necessary to mitigate the consequences of an accident.

The review of internally generated missile protection includes the following: structures, systems or portion of systems, and components that require protection from internally generated missiles are identified; pressurized components and systems are reviewed to determine their potential for generating missiles such as valve bonnets and hardware retaining bolts, relief valve parts, and instrument wells; high speed rotating machinery are reviewed to determine their potential for generating missiles from component overspeed or failure, such as failure of the pump itself (resulting from seizure), pump or component parts, and rotating segments (e.g., impellers and fan blades).

If safety-related systems or components are located in areas containing nonsafety-related SSC, then the nonsafety-related SSC are reviewed with respect to internal missile effects if the failure could preclude the intended safety function of the safety-related SSC.

The Structural Engineering Branch determines the acceptability of the analysis and criteria used for the design of structures or barriers that protect essential systems and components from internally generated missiles (SRP Section 3.5.3). Their results are used by the ASB to complete the overall evaluation of protection against internally generated missiles.

II. ACCEPTANCE CRITERIA

Acceptability of the design information on protection of essential systems and components from internally generated missiles persented in the applicant's safety analysis report (SAR) is based on meeting specific general design criteria and regulatory guides.

The design of structures, systems, and components is acceptable if the integrated design affords missile protection in accordance with the following criteria:

1. General Design Criterion 4, with respect to protecting structures, systems, and components against the effects of internally generated missiles to maintain their essential safety functions.

2. Regulatory Guide 1.13, as related 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.

3. Regulatory Guide 1.27, as related to the ultimate heat sink and connecting conduits being capable of withstanding the effects of internally generated missiles.

4. Regulatory Guide 1.115, as related to the protection of structures, systems, and components important to safety from the effects of turbine missiles.

A commitment in the SAR that essential structures, systems, and components will be protected from internally generated missiles (outside containment) by locating the systems or components in individual missile-proof structures, or providing special localized protective shields or barriers, is acceptable for the construction permit stage.

For those areas of review identified in subsection I of this SRP section as being the responsibility of other branches, the acceptance criteria and their methods of application are contained in the SRP sections corresponding to those branches.

III. REVIEW PROCEDURES

The review procedures set forth below are used during the construction permit (CP) application review to determine that the design criteria and bases and the preliminary design in applicant's preliminary safety analysis report meet the acceptance criteria given in subsection II. For the review of the operating license (OL) application, the review procedures and acceptance criteria are used to verify that the initial design criteria and bases have been appropriately implemented in the final design as set forth in the final safety analysis report. The reviewer selects and emphasizes areas within the scope of this SRP section as may be appropriate in a particular case.

Upon request from the primary reviewer, the secondary review branches will provide input for the areas of review stated in subsection I. The primary reviewer obtains and uses such input as required to assure that this review procedure is complete.

The objective in the review of the reactor facility, structures, systems and components, with regard to protection requirements for internally generated missiles, is to identify the SSC that are needed to perform a safety function. Some structures and systems are designed as safety-related in their entirety, others have portions that are safety-related, and others are classified as not needed for safety. In order to determine their safety category, the ASB evaluates the SSC with regard to their function in achieving and maintaining a safe reactor shutdown condition or in preventing accidents or mitigating the consequences of such accidents. The single failure criterion is used in the analysis. The safety functions to be performed by the SSC in the various plant designs are essentially the same. However, the location and arrangement of the SSC and the methods used vary from plant to plant depending upon the individual design. The review identifies variations in plant designs that must be evaluated on an individual case basis. Structures, systems, or components that perform a safety function, or by virtue of their failure could have an adverse effect on a safety function should be protected from the effects of internally generated missiles.

The information provided in the SAR pertaining to SSC design bases and criteria, system descriptions and safety evaluations, piping and instrumentation diagrams, station layout drawings, and system and component characteristic and classification tables are reviewed to identify potential sources of missiles and to determine the SSC that require protection in order to maintain their safety-related functions. The reviewer may use failure mode and effect analyses and the results of reviews by other branches in evaluating specific SSC and the origin of possible missiles, in identifying the structures, systems, and components that require protection from internally generated missiles and the adequacy of the protection provided. Components within one train need not be protected from missiles originating from the same train.

The reviewer determines that nonsafety-related structures, systems, or components are protected from internally generated missiles if their failure by a missile impact could prevent the intended safety function of the SSC.

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information has been provided to satisfy the requirements of this SRP section and that his evaluation is complete and adequate to support conclusions of the following type, to be included in the staff's safety evaluation report:

> "The review of possible effects of internally generated missiles (outside containment) included structures, systems, and components whose failure could prevent safe shutdown of the plant or result in significant uncontrolled release of radioactivity. Based on the review of the applicant's proposed design criteria, design bases, and safety classifications for essential structures, systems, and components necessary to maintain a safe plant shutdown, the staff concludes that the structures, systems

and components to be protected from internally generated missiles (outside containment) conform to the Commission's Regulations as set forth in General Design Criterion 4, "Environmental and Missile Design Basis," and meet the guidelines in Regulatory Guide 1.13, "Fuel Storage Facility Design Basis," Regulatory Guide 1.27, "Ultimate Heat Sink for Nuclear Power Plants," and Regulatory Guide 1.115, "Protection Against Low-Trajectory Turbine Missiles," and, therefore, are acceptable.

V. REFERENCES

1. 10 CFR Part 50, Appendix A, General Design Criterion 4, "Environmental and Missile Design Bases."

2. Regulatory Guide 1.13, "Fuel Storage Facility Design Basis."

3. Regulatory Guide 1.27, "Ultimate Heat Sink for Nuclear Power Plants."

4. Regulatory Guide 1.115, "Protection Against Low-Trajectory Turbine Missiles."

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SECTION 3.5.1.2 INTERNALLY GENERATED MISSILES (INSIDE CONTAINMENT)

REVIEW RESPONSIBILITIES

Primary - Reactor Systems Branch (RSB)

Secondary - Containment Systems Branch (CSB)
Mechanical Engineering Branch (MEB)

I. AREAS OF REVIEW

The RSB review of the structures, systems, and components (SSC) to be protected from internally generated missiles (inside containment) to assure conformance with the requirements of General Design Criterion 4 includes all SSC within the containment and the containment itself. The review includes internally generated missiles associated with component overspeed failures, missiles that could originate from high energy fluid system failures, and missiles due to gravitational effects.

The RSB with the assistance of the CSB reviews the functional operations and performance requirements for structures, systems, and components inside containment and identifies which of the operations are necessary for the safe shutdown of the reactor facility in the event of an accident or other circumstances that might result in an internally generated missile, or for the mitigation of the effects of loss-of-coolant or other accidents. Safety-related SSC are reviewed with respect to their capability to perform functions required for attaining and maintaining a safe shutdown condition during such accident conditions.

The review of internally generated missile protection includes the following:

1. Structures, systems or portion of systems, and components requiring protection from internally generated missiles and the methods of protection provided against such missiles are described.

2. Credible primary missiles are identified including, valve hardware, retaining bolts, relief valves parts, instrument wells and reactor vessel seal rings (PWR).

3. Credible secondary missiles generated as a result of impact with primary missiles.

The SEB, in SRP Section 3.5.3, determines the acceptability of the analytical procedures and criteria used for structures or barriers that protect the containment structure and liner, essential systems, and safety-related components from internally generated missiles. Their results are used by the RSB and CSB to complete the overall evaluation of protection against internally generated missiles.

The RSB identifies those systems designed to withstand the effects of postulated high energy piping failures in accordance with the criteria stated in SRP Section 3.6.2. These systems provide substantial protection from potential missiles and are reviewed by MEB for missile consequences only in those situations for which the protection provided for piping failures is not considered completely adequate by RSB or CSB.

II. ACCEPTANCE CRITERIA

Acceptability of the design information on protection of structures and essential systems and components from internally generated missiles, as presented in the applicant's safety analysis report (SAR), is based on specific general design criteria and regulatory guides. An additional basis for determining acceptability is the degree of similarity of the design to that of previously approved plants.

The design of structures, systems, and components is acceptable if the integrated design affords missile protection in accordance with the following criteria:

1. General Design Criterion 4 (Ref. 1), as it relates to structures housing essential systems and to the systems being capable of withstanding the effects of internally generated missiles.
2. ASME Code Section III (Ref. 2), as it relates to the design of steel or concrete containment, whichever is appropriate.

A statement in the SAR that essential structures, systems, and components will be afforded protection by locating the systems or components in individual missile-proof structures, physically separating redundant systems or components of the system, or providing special localized protective shields or barriers, is an acceptable design basis at the construction permit stage for providing protection from internally generated missiles (inside containment).

For those areas of review identified in Subsection I of this SRP section as being the responsibility of other branches, the acceptance criteria and their methods of application are contained in the SRP sections corresponding to those branches.

III. REVIEW PROCEDURES

The review procedures below are used during the construction permit (CP) review to determine that the design criteria and bases and the preliminary design as set forth in the preliminary safety analysis report meet the acceptance criteria given in Subsection II. For the review of operating license (OL) applications, the review procedures and acceptance criteria are used to verify that the initial design criteria and bases have been appropriately implemented in the final design as set forth in the final safety analysis report. The reviewer selects and emphasizes areas within the scope of this SRP section as may be appropriate in a particular case.

The first objective in the review of the structures, systems, and components, with regard to protection requirements for internally generated missiles, is to determine whether the equipment is needed to perform a safety function. Some structures and systems are designed as safety-related in their entirety, others have portions that are safety-related, and

others are classified as not needed for safety. In order to determine the safety category of the SSC, the RSB and CSB evaluate the SSC with regard to their function in achieving safe reactor shutdown conditions or in preventing accidents or mitigating the consequences of accidents. Upon request from the primary reviewer, other secondary review branches will provide input for the areas of review stated in Subsection I. The primary reviewer obtains such input as required to assure that this review procedure is complete. Structures, systems, or components that perform a safety function, or by virtue of their failure could have an adverse effect on a safety function should be protected from the effects of internally generated missiles.

A review is conducted of the information provided in the SAR pertaining to SSC design bases and criteria, the listing of credible primary missiles, secondary missiles, damage or failures to safety-related SSC as a result of missile impingement and missile protection provided. The reviewer may use failure mode and effect analyses and the results of other portions of the facility review in evaluating specific SSC and the origin of possible missiles, and in determining which structures, systems, and components require protection from internally generated missiles and whether the degree of protection provided is adequate.

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information has been provided and that his evaluation supports conclusions of the following type, to be included in the staff's safety evaluation report:

> "The review of possible effects of internally generated missiles (inside containment) included structures, systems, and components whose failure could prevent safe shutdown of the plant or result in significant uncontrolled release of radioactivity. The scope of review in this area for the _____ plant included the review of credible missile sources, missile protection provided, and descriptive information for structures, systems, and components essential to the safe operation and shutdown of the plant. [The review has included the applicant's proposed design criteria and bases for essential structures, systems, and components, the adequacy of those criteria and bases, and the equipment necessary to maintain the capability for a safe plant shutdown in the event of an internally generated missile (inside containment) (CP).] [The review has included the applicant's analysis of the manner in which the design of essential structures, systems, and components conforms to the previously approved design criteria and bases and demonstrates the ability to perform a safe plant shutdown after any internally generated missile accident (inside containment) (OL).]
>
> "The staff concludes that the facility design with regard to protection from internally generated missiles (inside containment) conforms to the Commission's regulations and to applicable regulatory guides, staff technical positions, and industry standards, and is acceptable."

V. REFERENCES

1. 10 CFR Part 50, Appendix A, General Design Criterion 4, "Environmental and Missile Design Bases."

2. ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NE, "Class MC Components," and Division 2 (ACI-359), "Standard Code for Concrete Reactor Vessels and Containments," American Society of Mechanical Engineers.

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Section 3.5.1.3 TURBINE MISSILES

REVIEW RESPONSIBILITIES

Primary - Accident Analysis Branch (AAB)

Secondary - Auxiliary Systems Branch (ASB)
Power Systems Branch (PSB)
Materials Engineering Branch (MTEB)
Structural Engineering Branch (SEB)
Mechanical Engineering Branch (MEB)

I. AREAS OF REVIEW

Plant designs are reviewed with the objective of establishing whether safety-related plant structures, systems, and components have adequate protection against the effects of potential turbine missiles. The primary review area is the evaluation of turbine missile generation, strike, and damage probabilities with respect to the safety-related missile targets. The following items are identified as supporting review areas which may be necessary in the overall evaluation of the turbine missile protection requirements:

1. Turbine missile impact effects on steel and concrete barriers (e.g., penetration depth, scabbing, and structural response) - SEB.
2. Turbine disc failure analysis, fracture toughness properties, and turbine startup procedures. - MTEB.
3. Turbine overspeed protection.
   a. Overspeed sensing and tripping - PSB.
   b. Steam valves - MEB.
4. Target identification, redundancy, and independence - ASB.
5. Inservice inspection - MTEB, MEB, PSB.

II. ACCEPTANCE CRITERIA

Plant design and layout must satisfy General Design Criterion 4 (Ref. 1), which states that structures, systems, and components important to safety should be protected against the effects of missiles that might result from equipment failures.

Consideration of turbine missile protection is relevant for essential systems, i.e., those structures, systems, and components necessary to ensure:

- The integrity of the reactor coolant pressure boundary.
- The capability to prevent accidents that could result in potential offsite exposures that are comparable to the guideline exposures of 10 CFR Part 100, "Reactor Site Criteria."

Specifically, in the areas reviewed by the AAB, acceptability will be based on the following considerations:

1. Plant designs with a favorable turbine generator placement and orientation, and adhering to the guidelines presented in Regulatory Guide 1.115 (Ref. 2) will be considered to be adequately protected against turbine missile hazards. Exclusions of safety-related structures, systems, or compounds from low trajectory turbine missile strike zones constitutes adequate protection against low trajectory turbine missiles. In those cases where exclusion of safety-related targets from the low trajectory missile strike zones is impractical (e.g., location dictated by site characteristics, such as a water intake structure for the ultimate heat sink) target size, shielding, or redundancy may be considered with respect to missile protection. The acceptance criterion is that the combined strike and damage probability for these targets be less than 10^{-3} per turbine failure.

2. Plant designs with unfavorable turbine-generator placement and orientation, such that safety-related structures, systems, or components are within the low trajectory turbine missile strike zones and are susceptible to potential missile damage, should have sufficient missile protection in terms of one or more of the following:
   a. Missile barriers.
   b. Target redundancy.
   c. Turbine disc integrity.
   d. Overspeed protection.

   The SRP Section 2.2.3 risk acceptance guidelines used for potential accident situations in the vicinity of the plant will also be used in determining the sufficiency of protection against turbine missiles.
3. Protection against turbine missiles by means of structural barriers should conform to the recommendations and acceptance criteria provided in SRP Section 3.5.3. The SEB is responsible for evaluating the design adequacy of structural barriers.
4. Turbine overspeed protection system should conform to the recommendations outlined in SRP Section 10.2.
5. Low pressure turbine disc materials, manufacturing processes, and operating conditions should conform to the recommendations outlined in SRP Section 10.2.3.
6. The safety-related structures, systems, and components to be protected against turbine missiles should include those described in Regulatory Guide 1.117 (Ref. 5) for tornado missiles.
7. The following criteria apply specifically to Construction Permit applications docketed prior to 11/15/76 and to all Operating License reviews:
   a. When the estimated turbine missile risks exceed the guidelines of SRP Section 2.2.3, the following requirements should be met:
      i. The design and on-line testing of the overspeed sensing and tripping system, including the main steam stop and control valves, and reheat stop and intercept valves, should be in accordance with SRP Section 10.2. For Operating License reviews a determination should be made of whether increased valve testing should be required, based on cost-benefit considerations.
      ii. The applicant should submit a detailed strike and damage analysis with respect to all vulnerable targets (with the aim of assessing the margin

available) and/or provide local shielding (if the above analyses indicate that SRP Section 2.2.3 guidelines are still exceeded). The procedures used for describing missile interactions with structural barriers and barrier damage analysis should conform to those of SRP Section 3.5.3. The SEB will review the interaction aspects of turbine missiles with respect to structural barriers and their damage analysis. The AAB reviewer will perform an overall risk assessment of turbine missile hazard based on an independent evaluation of the detailed strike and damage analyses.

For those areas of review identified in subsection I of this SRP section as being the responsibility of other branches, the acceptance criteria and their methods of application are contained in the SRP sections corresponding to those branches.

III. REVIEW PROCEDURES

The reviewer selects and emphasizes aspects of the areas covered by this SRP section as may be appropriate for a particular case. The judgment on areas to be given attention and emphasis in the review is based on an inspection of the material presented to see whether it is similar to that recently reviewed on other plants and whether items of special safety significance are involved.

Upon request from the primary reviewer, the secondary review branches will provide input for the areas of review stated in subsection I. The primary reviewer obtains and uses such input as required to assure that this review procedure is complete.

The review procedure involves the following:

1. The plant layout drawings are reviewed to determine the relative placement of safety-related structures, systems, and components with respect to the turbine-generator unit(s). This review is focused on determining if the plant layout conforms to the turbine placement and orientation recommendations outlined in Reguatory Guide 1.115. If the orientation is such that all safety-related targets are excluded from the low trajectory turbine missiles and further review in this regard is not necessary. This procedure also encompasses the possibility of having some safety-related targets within the strike zones when their placement is unavoidable. However, these systems must be protected against the effects of destructive overspeed turbine missiles. As indicated in the Regulatory Guide 1.115, this condition is met if the size, placement, and/or shielding by barriers is such that the total strike and damage probability for all such targets within the strike zones is less than 10^{-3} per turbine failure. Adequate protection will also be identified with targets which are redundant and sufficiently independent (e.g., by separation distance, barriers) such that a turbine failure could not compromise two or more members of a redundant train.

   The following specific information is necessary in order to perform the above review:

   a. Dimensioned plant layout drawings (plan and elevation views).
   b. Barriers (e.g., structural wall material strength properties, thicknesses).

c. Identification of safety-related structures, systems, and components in terms of location, redundancy, and independence.

d. Identification of all turbine-generator units (present and future) in the vicinity of the plant being reviewed.

e. A quantitative description of the turbine-generator in terms of rotor shaft, wheels, steam valve characteristics, rotational speed and turbine internals pertinent to turbine missiles analyses. Postulated missiles should be identified in terms of missile size, mass, shape, and exit speed for design overspeed and destructive overspeed turbine failures. A description should be provided of the analysis used in estimating the missile exit speeds. The sense of rotation should be identified with respect to each turbine-generator under consideration.

Most of this information can be obtained from the applicant's SAR. The relevant Standard Format Sections are 1.2, 3.5, 3.8, and 10.2.

2. Plants which do not conform to the recommendations of Regulatory Guide 1.115 should be reviewed on a case-by-case basis for each safety-related target. The review centers around the evaluation of the individual probability components in the relation

$$P = \frac{1}{N} \sum_{i=1}^{2} P_{1i} \sum_{j=1}^{N} P_{2ij} P_{3ij}$$

where

P = Total probability for incurring damage which exceeds the criteria described in subsection II, per turbine year.

N = Total number of distinct turbine missile sources per turbine-generator unit, usually identified with the number of low pressure wheels.

P_{1i} = Probability for turbine failure leading to the ejection of missiles due to i^{th} type of turbine failure.

P_{11} = 6×10^{-5} per turbine year for design speed failures.

P_{12} = 4×10^{-5} per turbine year for destructive overspeed failures (Ref. 6).

P_{2ij} = The strike probability with respect to a barrier between the turbine and the target. In case of multiple barriers, it is equivalent to the probability for striking the final barrier between the turbine and the target. The j - index refers to the jth wheel on the turbine rotor.

P_{3ij} = The probability for damaging the target. This can be either due to primary missile penetration of a barrier or due to the generation of secondary missiles (e.g., scabbing in concrete), or both.

It should be noted that in the case of multiple barriers the value of P_{2ij} will be determined by a combination of geometric considerations, missile deflections, and intermediate barrier penetration estimates. The usual procedure is to estimate the portion of the total solid angle associated with each ejected missile that is subtended by the target in question. If there are no intermediate barriers, or if all barriers up to the final barrier are penetrated independently of missile state (i.e., energy, impact orientation) then P_{2ij} can be approximated by

$$P_{ij} = \left(\frac{\Delta\theta_j}{\Delta\theta_{j,max}}\right) \left(\frac{\Delta\phi}{\Delta\phi max}\right)$$

where

$\Delta\phi_j$ = Plan view angle subtended by the target with the respect to the jth wheel.

$\Delta\phi_{j,}$ = Maximum plan view angular displacement of missiles ejected from the jth wheel.

= 10° for inner wheels

= 25° for end wheels.

$\Delta\phi$ = Elevation angle subtended by the target.

$\Delta\phi_{max}$ = Maximum range of elevation angle for a missile (e.g., for a symmetrical four piece break, $\Delta\phi_{max}$ = 90°).

An additional factor f may be used to multiply the above relation if penetration of intermediate barriers is conditional on missile state. This can be done by considering the ratio of all missile states that penetrate the barrier to the total number of missile states If there are M barriers, this may be expressed as

$$f_{ij} = \prod_{m=1}^{M} \frac{(\text{All missile states that penetrate } m^{th} \text{ barrier})ij}{(\text{Total number of possible missile states})ij}$$

where the i and j indices refer to the turbine failure mode and failed wheel, respectively.

Estimates of the potential for concrete penetration and/or scabbing are based on the missile penetration criteria described in SRP Section 3.5.3.

The evaluation of the overall probability P is performed by considering conservative as well as realistic estimates of all the individual parameters that are used in the analysis. The conservative and realistic estimates of P are used in conjunction with the risk acceptance guidelines described in SRP Section 2.2.3 in determining the acceptability of the plant design with respect to turbine missiles risk.

3. The reviewer may request technical assistance on an as needed basis in the following areas in order to complete the turbine missile evaluation:
   a. Where the design basis protection against turbine missiles is primarily by use of barriers, the adequacy of structural turbine barrier procedures are verified by the SEB in accordance with the criteria of SRP Section 3.5.3.
   b. The effect of fracture toughness properties on the failure probability of the low pressure turbine wheels is reviewed by the MTEB.
   c. The turbine overspeed protection system and its testing are evaluated by the MEB (turbine steam valve reliability) and the PSB (tripping and overspeed sensing systems).
   d. The identification of plant essential systems to be protected against turbine missiles is reviewed by the ASB.

4. For Construction Permit applications docketed prior to 11/15/76 and to all Operating License reviews, a summary should be prepared of the following items:
   a. Identification of all safety-related targets vulnerable to turbine missiles.
   b. MTEB findings regarding turbine disc and rotor integrity and inservice inspection program.
   c. When appropriate, SEB evaluation of credit for missile barriers.
   d. PSB findings regarding turbine overspeed protection system.
   e. A general value impact assessment of localized missile shielding (CP's and OL's) and/or system relocation (CP's only).
   f. Identification of additional plant requirements, if any.

5. High trajectory turbine missiles are characterized by their nearly vertical trajectories. Missiles ejected more than a few degrees from the vertical, either have sufficient speed such that they land offsite, or their speeds are low enough so that their impact on most plant structures is not a significant hazard. The probability of a high trajectory turbine missile landing within a few hundred feet from the turbine is on the order of 10^{-7} per square foot of horizontal target area. Consequenty the risk from high trajectory turbine missiles is insignificant unless the vulnerable target area is on the order of 10^4 square feet or more.

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information has been provided and that the review and calculations support conclusions of the following types:

1. The turbine missile risk for the proposed plant design are acceptably low, so that the safety-related plant structures, systems and components are protected adequately against potential turbine missile damage.
2. The turbine missile risks for the proposed plant design are too high. Additional protection against turbine missiles is required in order to reduce the overall risk to an acceptably low level (turbine reorientation, vulnerable system relocation, missile barriers, overspeed protection, turbine disc integrity and inservice inspection, or other appropriate measures may be recommended).

V. REFERENCES

1. 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants."
2. Regulatory Guide 1.115, "Protection Against Low-Trajectory Turbine Missiles," Rev. 1, July 1977.
3. Regulatory Guide 1.117, "Tornado Design Classification," June 1976.
4. S. H. Bush, "Probability of Damage to Nuclear Components," Nuclear Safety, Vol. 14, No. 3, May-June 1973.
5. "Fundamentals of Protective Design," TM-5-855-1, Department of the Army, July 1965.

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SECTION 3.5.1.4 MISSILES GENERATED BY NATURAL PHENOMENA REV. 1

REVIEW RESPONSIBILITIES

Primary - Accident Analysis Branch (AAB)

Secondary - Structural Engineering Branch (SEB)
Auxiliary System' Branch (ASB)

I. AREAS OF REVIEW

The applicant's assessment of possible hazards due to missiles generated by the design basis tornado, flood, and any other natural phenomena identified in Section 3.5 of the safety analysis report (SAR) is reviewed. The purpose of the review is to assure that hazards due to these missiles are acceptably small so that they need not be included in the plant design basis, and that appropriate design basis missiles have been chosen and properly characterized. Currently, only missiles from the design basis tornado (Ref. 1) are consistently considered in the plant design bases. Missiles from other phenomena are considered on a case-by-case basis when they are identified.

The ASB, under Standard Review Plan (SRP) 3.5.2, identifies those structures, systems, and components that should be protected against missile impact and the SEB, under SRP 3.5.3, assures that adequate protection is provided by structures and missile barriers.

II. ACCEPTANCE CRITERIA

1. The identification of appropriate design basis missiles generated by natural phenomena is considered acceptable if the methodology is consistent with the acceptance criteria defined for the evaluation of potential accidents from external sources in SRP 2.2.3 (Ref. 2).

2. The staff's position regarding the systems to be protected against missiles generated by natural phenomena is covered in Regulatory Guide 1.117, "Tornado Design Classification" (Ref. 3). Acceptable spectra of tornado missiles and impact velocities are listed in item 4 under Review Procedures (Section III, below).

III. REVIEW PROCEDURES

The reviewer selects and emphasizes aspects of the area covered by this review plan as may be appropriate for a particular case. The judgment on areas to be given attention and

emphasis in the review is to be based on an inspection of the material presented to see whether it is similar to that recently reviewed on other plants and whether items of special safety significance are involved.

1. The reviewer obtains from SAR Section 3.5 the identification of the design basis natural phenomena which could possibly generate missiles. Any postulated missiles are reviewed to determine if they have been properly characterized.

2. The probability per year of damage to the total of all important structures, systems, and components (as covered in Regulatory Guide 1.117, Ref. 3) due to a specific design basis natural phenomena capable of generating missiles is estimated.

3. If this probability is greater than about 10^{-7} per reactor-year, then specific design provisions must be provided to reduce the estimate of unacceptable damage probability to about 10^{-7} per reactor-year.

4. All plants are required to be designed to protect safety-related equipment against damage from missiles which might be generated by the design basis tornado for that plant. The reviewer should confirm that the applicant has postulated missiles that include at least three objects; a massive high kinetic energy missile which deforms on impact, a rigid missile to test penetration resistance, and a small rigid missile of a size sufficient to just pass through any openings in protective barriers. Until more definitive guidelines, based upon long-term research currently under way, are developed, these missiles may be assumed to be an 1800 Kg automobile, a 125 Kg 8" armor piercing artillery shell, and a 1" solid steel sphere, all impacting at 35% of the maximum horizontal windspeed of the design basis tornado (Ref. 6). The first two missiles are assumed to impact at normal incidence, the last to impinge upon barrier openings in the most damaging directions. These missiles are identified as Spectrum I.

   Alternately, the missiles selected by the National Bureau of Standards as representative of construction site debris in report NBSIR 76-1050 (Ref. 4) may be used. These are identified as Spectrum II missiles. Tornado regions are defined in WASH-1300 (Ref. 5).

SPECTRUM II

MISSILE	Mass (Kg)	Dimensions (m)	Velocity (m/sec) Region I	Region II	Region III
A Wood Plank	52	.092 x .289 x 3.66	83	70	58
B 6" Sch 40 pipe	130	.168 D X 4.58	52	42	10
C 1" Steel rod	4	.0254D x .915	51	40	8
D Utility pole	510	.343D x 10.68	55	48	26
E 12" Sch 40 pipe	340	.32D x 4.58	47	28	7
F Automobile	1810	5 x 2 x 1.3	59	52	41

Vertical velocities of 70% of the postulated horizontal velocities are acceptable in both spectra except for the small missile, or missile C above, used to test barrier openings, which should be assumed at the same speed in all directions.

5. At the operating license stage, applicants who were not required at the construction permit stage to design to the missile spectrum of Rev. 0 to this SRP (shown below) and the corresponding velocity set, should show the capability of the existing structures and components to withstand at least missiles C and F of the Rev. 0 to this SRP. The adequacy of existing protection and any requirements for improvements will be determined on a case-by-case basis in conjunction with ASB and SEB. The AAB Branch Chief should be consulted in making such determinations. Applicants who were required at the construction permit stage to design to one of the missile spectra of Rev. 0 to this SRP (or a review modification such as a 24" vertical and 21" horizontal wall thickness commitment in Region I), shall have the option at the OL stage of showing conformance with either their original commitment or Rev. 1 to this SRP. Partial compliance with each is not acceptable.

SRP 3.5.1.4 REV. 0 MISSILE SPECTRUM

		Fraction of total tornado velocity
A.	Wood plank, 4 in. x 12 in. x 12 ft, weight 200 lb.	0.8
B.	Steel pipe, 3 in. diameter, schedule 40, 10 ft long weight 78 lb.	0.4
C.	Steel rod, 1 in. diameter x 3 ft long, weight 8 lb.	0.6
D.	Steel pipe, 6 in. diameter, schedule 40, 15 ft long, weight 285 lb.	0.4
E.	Steel pipe, 12 in. diameter, schedule 40, 15 ft long, weight 743 lb.	0.4
F.	Utility pole, 13-1/2 in. diameter, 35 ft long, weight 1490 lb.	0.4
G.	Automobile, frontal area 20 ft^2, weight 4000 lb.	0.2

These missiles are considered to be capable of striking in all directions with vertical velocities equal to 80% of the acceptable horizontal velocities. Missiles A, B, C, D, and E are to be considered at all elevations and missiles F and G at elevations up to 30 feet above all grade levels within 1/2 mile of the facility structures.

6. The capability of structures to withstand the postulated missile impacts are reviewed by the SEB and vital target areas are defined by the ASB.

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information has been provided and the review and calculations support conclusions of the following type, to be included in the staff's safety evaluation report:

"The applicant's assessment of hazardous missiles generated by natural phenomena at the site has been independently verified by the staff. Based on the likelihood and severity of these phenomena and the protection provided against tornado generated missiles, we have determined that the probability of an accident having radiological consequences

worse than the exposure guidelines of 10 CFR Part 100 is less than about 10^{-7} per year. We conclude, therefore, that the construction and operation of the _____ plant on the proposed site does not present an undue risk to the health and safety of the public from missiles generated by natural phenomena, including tornadoes."

V. REFERENCES

1. Regulatory Guide 1.76, "Design Basis Tornado for Nuclear Power Plants."

2. Standard Review Plan 2.2.3, "Evaluation of Potential Accidents."

3. Regulatory Guide 1.117, "Tornado Design Classification."

4. "Tornado-Borne Missile Speeds," NBSIR 76-1050, National Bureau of Standards (April 1976).

5. "Technical Basis for Interim Regional Tornado Criteria," WASH-1300, U.S. Atomic Energy Commission (May 1974).

6. "An Assessment of the Bases for Selecting Criteria for Protection Against Tornado-Entrained Debris," U.S. Nuclear Regulatory Commission, NUREG-0121 (April 1977).

BRANCH TECHNICAL POSITION AAB 3-2

TORNADO DESIGN CLASSIFICATION

A. BACKGROUND

General Design Criterion 2 requires, in part, that structures, systems, and components important to safety be designed to withstand the effects of natural phenomena such as tornadoes without loss of capability to perform their safety functions. Criterion 2 also requires that the design bases for these structures, systems, and components reflect (1) appropriate combinations of the effects of normal and accident conditions with the effects of natural phenomena and (2) the importance of the safety functions to be performed.

General Design Criterion 4 requires, in part, that structures, systems, and components important to safety, be protected against the effects of missiles from events and conditions outside the plant.

Nuclear power plants should be designed so that the plants can be placed and maintained in a safe shutdown condition in the event of the most severe tornado that can reasonably be predicted to occur at a site as a result of severe meteorological conditions. Protection of structures, systems, and components necessary to place and maintain the plant in a cold shutdown condition may generally be accomplished by designing protective barriers to preclude missile strikes. For example, the primary containment, reactor building, auxiliary building, and control structures should be designed against collapse and should provide an adequate barrier against missiles. However, the primary containment need not necessarily maintain its leak-tight integrity under pressure loadings due to the pressure differentials developed by the tornado. If protective barriers are not installed, the structures and components themselves should be designed to withstand the effects of the tornado, including tornado missile impacts.

It is not necessary to maintain the functional capability of all seismic Category I structures, because the combined probability of a joint occurrence of low probability events (loss-of-coolant accident with design basis or smaller tornado, or earthquake and design basis or smaller tornado) is so small as to not warrant consideration in the plant design basis. However, a source of water should be available to provide long-term core cooling.

Similarly, it is not necessary to protect radioactive liquid waste holdup tanks since even in the event of gross failure, the spills would be limited to small amounts of waste and would be expected to be collected in the building foundations, which are designed for that purpose.

Structures, systems, and components important to safety which should be designed to withstand the effects of a design basis tornado are those necessary to ensure:

1. The integrity of the reactor coolant pressure boundary.

2. The capability to shut down the reactor and maintain it in a safe shutdown condition.

3. The capability to prevent accidents which could result in potential offsite exposures that are a significant fraction of the guideline values of 10 CFR Part 100. Designs which differ substantially from those now in use may require reevaluation with respect to this objective.

The physical separation of redundant or alternative structures or components required for the safe shutdown of the plant is generally not considered an acceptable method for protecting against tornado effects, including tornado-generated missiles.

This branch position describes a method acceptable to the staff for identifying those structures, systems, and components of light-water reactors which should be designed to withstand the effects of the design basis tornado (as defined by Regulatory Guide 1.76), including tornado missiles, and to remain functional.

B. BRANCH TECHNICAL POSITION

1. Those structures, systems, and components, including foundations and supports, which should be designed to withstand the effects of a design basis tornado (as defined in Regulatory Guide 1.76), including tornado missiles, without loss of capability to perform essential safety functions are listed below.

   a. The reactor coolant pressure boundary.[1/]

   b. Those portions of the main steam and main feedwater systems of pressurized water reactors (PWRs) up to and including the outermost isolation valves.

   c. The reactor core and reactor vessel internals.

   d. Systems[2/] or portions of systems, and those auxiliary systems necessary to support these systems (for example, service water, cooling water source, component cooling and auxiliary feedwater) that are required for (1) reactor shutdown, (2) residual heat removal, (3) cooling the spent fuel storage pool, or (4) makeup water for the primary system.

   e. The spent fuel storage facility to the extent necessary to preclude significant loss of watertight integrity of the storage pool and to prevent missiles from contacting fuel within the pool.

   f. The reactivity control systems, e.g., control rod drives and boron injection systems.

[1/] As defined in 10 CFR § 50.2

[2/] The system boundary includes those portions of the system required to accomplish the specified safety function and connecting piping up to and including the first valve (including a safety or relief valve) that is either normally closed or capable of automatic closure when the safety function is required.

g. The control room, including its associated vital equipment, cooling systems for the vital equipment and life support systems, and any structures or equipment inside or outside of the control room whose failure could result in an incapacitating injury to individuals occupying the control room.

h. Those portions of the gaseous radwaste treatment systems which by design are intended to store or delay gaseous radioactive waste and portions of structures housing these systems including isolation valves, equipment, interconnecting piping, and components located between the upstream and downstream valves used to isolate these components from the rest of the system (e.g., charcoal delay tanks in a boiling water reactor (BWR) plant and waste gas storage tanks in a PWR plant).

i. Systems or portions of systems that are required for (1) monitoring systems important to safety and (2) actuating and operating systems important to safety.

j. All electric and mechanical devices and circuits between the process sensors and the input terminals of the actuator systems involved in generating signals that initiate protective action.

k. Those portions of the long-term emergency core cooling system that would be required to maintain the plant in a safe condition for an extended time after a loss-of-coolant accident.

l. Primary reactor containment and other safety-related structures, such as the control room building and auxiliary building, should be protected against collapse. The primary containment need not necessarily maintain its leak-tight integrity under pressure loadings due to pressure differentials developed by the tornado, tornado-borne missiles which could jeopardize contained safety-releated systems and components.

m. Class 1E electric systems, including the auxiliary systems for the onsite electric power supplies that provide emergency electric power needed for functioning of plant features included in items a through k above.

2. Those portions of structures, systems, or components whose continued function is not required but whose failure could reduce to an unacceptable safety level the functional capability of any feature included in the items listed above should be designed and constructed so that the effects of the design basis tornado would not cause failure (for example, of the containment walls).

C. REFERENCES

1. 10 CFR Part 100, "Reactor Site Criteria."

2. Regulatory Guide 1.76, "Design Basis Tornado for Nuclear Power Plants."

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SECTION 3.5.1.5 SITE PROXIMITY MISSILES (EXCEPT AIRCRAFT)

REVIEW RESPONSIBILITIES

Primary - Accident Analysis Branch (AAB)

Secondary - Structural Engineering Branch (SEB)
Auxiliary and Power Conversion Systems Branch (APCSB)

I. AREAS OF REVIEW

The staff reviews the applicant's assessment of possible hazards due to missiles generated by the design basis explosions identified in Section 2.2 of the safety analysis report (SAR). The purpose of the review is to assure that hazards due to there missiles are acceptably small so that they need not be included in the plant design basis, or that appropriate design basis missiles have been chosen and properly characterized. The APCSB determines those systems and components that should be protected against missile impacts, and the SEB assures that adequate protection is provided.

II. ACCEPTANCE CRITERIA

The plant is considered adequately designed against site proximity missiles if the resulting probability of a missile affecting the safety-related features of the plant is within the guidelines established in Section II of Standard Review Plan 2.2.3.

III. REVIEW PROCEDURES

The reviewer selects and emphasizes aspects of the areas covered by this review plan as may be appropriate for a particular case. The judgment on areas to be given attention and emphasis in the review is based on an inspection of the material presented to see whether it is similar to that recently reviewed on other plants and whether items of special safety significance are involved.

1. The identification of accidents which could possibly generate missiles is obtained from Section 2.2 of the SAR.

2. The total probability of the missiles striking a critical area of the plant is estimated. The total probability per year (P_T) may be estimated by using the following expression:

$$P_T = P_E \times P_{MR} \times P_{SC} \times P_P \times N$$

where:

P_E = probability per year of design basis explosion calculated in Section 2.2,

P_{MR} = probability of missiles reaching the plant,

P_{SC} = probability of missiles striking a critical area of the plant,

P_P = probability of missiles exceeding the energies required to penetrate to vital areas (e.g., based on wall thickness provided for tornado missiles), and

N = number of missiles generated by the design basis explosion.

P_{MR}, P_{SC} and P_P are assumed to be equal to 1 unless the analyses in this section demonstrate lower values.

3. If P_T is greater than about 10^{-7} per year, the reviewer should verify that the proper design basis events have been chosen and the missiles properly characterized.

4. The capability of structures to withstand the postulated missile impacts will be reviewed by the SEB, and the vital target areas will be defined by the APCSB.

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information has been provided and the review and calculations support conclusions of the following type, to be included in the staff's safety evaluation report:

1. "The staff analysis shows that the probability of an accident having serious radiological consequences is extremely remote and is within the guidelines established for low probability events of site proximity missiles. We conclude, therefore, that the probability of a missile impact causing radiological consequences of the order of 10 CFR Part 100 guidelines is so small that such an event does not present an undue risk to the health and safety of the public."

or

2. "The staff analyses verify that a design basis missile impact has been properly chosen and characterized."

V. REFERENCES

1. Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," Revision 2.

2. Regulatory Guide 1.76, "Design Bases Tornado for Nuclear Power Plants."

3. Regulatory Guide 1.91, "Evaluation of Explosions Postulated to Occur on Transportation Routes Near Nuclear Power Plant Sites."

4. Standard Review Plan 2.2.3, "Evaluation of Potential Accidents."

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SECTION 3.5.1.6 AIRCRAFT HAZARDS

REVIEW RESPONSIBILITIES

Primary - Accident Analysis Branch (AAB)

Secondary - Structural Engineering Branch (SEB)
Auxiliary and Power Conversion Systems Branch (APCSB)

I. AREAS OF REVIEW

The staff reviews the applicant's assessment of aircraft hazards to the plant. The purpose of the review is to assure that either aircraft hazards are eliminated as a design basis concern or appropriate design basis aircraft have been chosen and properly characterized as to impact and fire hazards. The review also involves a determination of adequate protection against fire hazards for design basis events. Some information relating to this review is contained in Section 2.2 of the applicant's safety analysis report (SAR), e.g., facility locations, projected traffic, and accident statistics.

The APCSB determines which structures and components are to be protected, and the SEB assures that adequate protection has been provided.

II. ACCEPTANCE CRITERIA

1. The plant is considered adequately designed against aircraft hazards if the probability of aircraft accidents resulting in radiological consequences greater than 10 CFR Part 100 exposure guidelines is less than about 10^{-7} per year (see Standard Review Plan 2.2.3).

2. The probability is generally considered acceptable by inspection if the level of aircraft activity near the site falls below the criteria given in Section 2.2.3 of Regulatory Guide 1.70 (Ref. 2) for analysis of hazards due to commercial, experimental, and general aviation aircraft. For military airspace, a minimum distance of five miles from the reactor is adequate for low level training routes except those associated with usage greater than 1000 flights per year or activities (such as practice bombing) where an unusual stress situation exists.

3. Aircraft accidents which could lead to radiological consequences in excess of the exposure guidelines of 10 CFR Part 100 with a probability of occurrence greater than about 10^{-7} per year should be considered in the design of the plant.

4. The evaluation of fire hazards will be done on an individual case basis. Concrete structures are generally assumed to withstand fire, but protection must be provided to prevent fire, smoke, or flammable mixtures from entering safety-related ventilation intakes, such as those for the control room, areas housing shutdown equipment, and the diesel generators.

III. REVIEW PROCEDURES

The reviewer selects and emphasizes aspects of the areas covered by this review plan as may be appropriate for a particular case. The judgment on areas to be given attention and emphasis in the review is based on an inspection of the material presented to see whether it is similar to that recently reviewed on other plants and whether items of special safety significance are involved.

The staff's review of the aircraft hazard assessment consists of the following steps:

1. Data describing aviation uses in the airspace near the proposed site, including airports and their approach paths, federal airways, Federal Aviation Administration (FAA) restricted areas, and military uses is obtained from Section 2.2 of the SAR. For many cases, no detailed analysis need be made as the probability can be judged adequately low based on a comparison with analyses previously performed. In such cases the conclusion reached and a citation of the cases used for comparison should be transmitted by buck slip to the AAB site analyst for retention in the case workbook.

2. For situations where federal airways or aviation corridors pass through the vicinity of the site, the probability per year of an aircraft crashing into the plant (P_{FA}) should be estimated. This probability will depend on a number of factors such as the altitude and frequency of the flights, the width of the corridor, and the corresponding distribution of past accidents.

   One way of calculating P_{FA} is by using the following expression:

   $$P_{FA} = C \times N \times A / w$$

   where:

   C = inflight crash rate per mile for aircraft using airway,

   w = width of airway (plus twice the distance from the airway edge to the site when the site is outside the airway) in miles,

   N = number of flights per year along the airway, and

   A = effective area of plant in square miles.

   This gives a conservative upper bound on aircraft impact probability if care is taken in using values for the individual factors that are meaningful and conservative. For

commercial aircraft a value of $C = 3 \times 10^{-9}$ per aircraft mile has been used. For heavily traveled corridors (greater than 100 flights per day), a more detailed analysis may be required to obtain a proper value for this factor.

3. The probability of an aircraft crashing into the site should be estimated for cases where either of the following apply:

   a. An airport is located within five miles of the site.

   b. An airport with projected operations greater than 500 d^2 movements per year is located within ten miles of the site, or an airport with projected operations greater than 1000 d^2 movements per year is located beyond ten miles from the site, where "d" is the distance in miles from the site.

The probability per year of an aircraft crashing into the site for these cases (P_A) may be calculated by using the following expression:

$$P_A = \sum_{i=1}^{L} \sum_{j=1}^{M} C_j \; N_{ij} \; A_j$$

where:

M = number of different types of aircraft using the airport,

L = number of flight paths affecting the site,

C_j = probability per square mile of a crash per aircraft movement, for the jth aircraft,

N_{ij} = number (per year) of movements by the jth aircraft along the ith flight path, and

A_j = effective plant area (in square miles) for the jth aircraft.

As noted earlier, the choice of values for the parameters should be made judiciously in order to arrive at a meaningful result. The manner of interpreting the individual factors may vary on a case-by-case basis because of the specific conditions of each case or because of changes in aircraft accident statistics.

Values for C_j currently being used are taken from the data summarized in the following table:

Distance From End of Runway (miles)	Probability ($\times 10^8$) of a Fatal Crash per Square Mile for Aircraft Movements			
	U.S. Air Carrier[1]	General Aviation[2]	USN/USMC[1]	USAF[1]
0-1	16.7	84	8.3	5.7
1-2	4.0	15	1.1	2.3
2-3	0.96	6.2	0.33	1.1
3-4	0.68	3.8	0.31	0.42
4-5	0.27	1.2	0.20	0.40
5-6	0	NA[3]	NA	NA
6-7	0	NA	NA	NA
7-8	0	NA	NA	NA
8-9	0.14	NA	NA	NA
9-10	0.12	NA	NA	NA

[1] Reference 2.

[2] Reference 4.

[3] NA indicates that data was not available for this distance.

4. For military installations or any other airspace usages, a detailed quantitative modeling of all operations should be verified. The result of the model should be the total probability (C) of an aircraft crash per unit area and time in the vicinity of the proposed site.

   The probability per year of a potentially damaging crash at the site due to operations at the facility under consideration (P_M) is then given for this case by the following expression:

$$P_M = C \times A$$

   where:

   C = total probability of an aircraft crash per square mile per year in the vicinity of the site, and

   A = effective area of the plant in square miles.

5. The total aircraft hazard probability at the site equals the sum of the individual probabilities obtained in the preceding steps.

6. The effective plant areas used in the calculations should include the following:

   a. A shadow area of the plant elevation upon the horizontal plane based on the assumed crash angle for the different kinds of aircraft and failure modes.

b. A skid area around the plant as determined by the characteristics of the aircraft under consideration. Artificial berms or any other man-made and natural barriers should be taken into account in calculating this area.

c. Areas of the plant susceptible to structural damage as a result of aircraft impact.

d. Areas of the plant susceptible to fire hazards resulting from aircrift accidents on the site.

For those classes of aircraft hazard having a probability of occurrence of causing radiological consequences in excess of 10 CFR Part 100 guidelines greater than about 10^{-7} per year, the reviewer should verify that the proper design basis events have been chosen and the aircraft properly characterized in terms of impact and fire parameters.

The capability of structures to withstand the postulated aircraft impacts will be reviewed by the SEB, and the vital target areas will be defined by the APCSB. In the past, external fire effects have been evaluated by the AAB with assistance from consultants (Ref. 3), but the APCSB will review this area for future applications.

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information has been provided and drafts an introductory paragraph for the evaluation findings indicating those facilities described in SAR Section 2.2 for which an aircraft hazards analysis was performed. A brief description of the methods used in the analysis should be provided, together with references to any sources of statistical data utilized.

The reviewer also verifies that the review and calculations support conclusions of the following type, to be included in the staff's safety evaluation report:

1. "The applicant's assessment of aircraft hazards at the site has been independently verified by the staff and results in a probability less than about 10^{-7} per year of an accident having radiological consequences worse than the exposure guidelines of 10 CFR Part 100. We conclude, therefore, that operation of the _____ plant in the vicinity of _____ does not present an undue risk to the health and safety of the public."

2. "Plant sites reviewed in the past which had equivalent aircraft traffic in equal or closer proximity were, after careful examination, found to present no undue risk to the safe operation of those plants. Based upon this experience, in the staff's judgment, no undue risk is present from aircraft hazard at the plant site now under consideration."

3. "The applicant's assessment of aircraft hazards at the site has been independently verified by the staff and we corroborate that if the plant (or appropriate parts of

the plant) is designed to withstand the aircraft selected as the design basis aircraft, the probability of an aircraft strike causing radiological consequences in excess of the exposure guidelines of 10 CFR Part 100 is less than about 10^{-7} per year. We conclude, therefore, that the operation of the _____ plant in the vicinity of _____ does not present an undue risk to the health and safety of the public."

V. REFERENCES

1. 10 CFR Part 100, "Reactor Site Criteria."

2. D. G. Eisenhut, "Reactor Siting in the Vicinity of Airfields." Paper presented at the American Nuclear Society Annual Meeting, June 1973.

3. I. I. Pinkel, "Appraisal of Fire Effects from Aircraft Crash at Zion Power Reactor Facility," July 17, 1972 (Docket No. 50-295).

4. D. G. Eisenhut, "Testimony on Zion/Waukegan Airport Interaction" (Docket No. 50-295).

5. USAEC Regulatory Staff, "Safety Evaluation Report," Appendix A, "Probability of an Aircraft Crash at the Shoreham Site" (Docket No. 50-322).

6. "Addendum to the Safety Evaluation by the Division of Reactor Licensing, USAEC, in the Matter of Metropolitan Edison Company (Three Mile Island Nuclear Station Unit 1, Dauphin County, Pennsylvania)," April 26, 1968 (Docket No. 50-289).

7. Letter to Honorable J. R. Schlesinger from S. H. Bush, Chairman, Advisory Committee on Reactor Safeguards, "Report on Rome Point Nuclear Generating Station," November 18, 1971 (Project No. 455).

8. Letter to Mr. Joseph L. Williams, Portland General Electric Company, from R. C. DeYoung (in reference to Mr. Williams' letter of May 7, 1973), November 23, 1973 (Project No. 485).

9. "Aircraft Considerations-Preapplication Site Review by the Directorate of Licensing, USAEC, in the Matter of Portland General Electric Company, Boardman Nuclear Plant, Boardman, Oregon," October 12, 1973 (Project No. 485).

10. Letter to Mr. J. H. Campbell, Consumers Power Company, from Col. James M. Campbell, Dep. Chief, Strategic Division, Directorate of Operations, U. S. Air Force, May 19, 1971 (Docket No. 50-155).

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SECTION 3.5.2 STRUCTURES, SYSTEMS, AND COMPONENTS TO BE PROTECTED FROM EXTERNALLY GENERATED MISSILES

REVIEW RESPONSIBILITIES

Primary - Auxiliary Systems Branch (ASB)

Secondary - None

I. AREAS OF REVIEW

The ASB review of the structures, systems, and components (SSC) to be protected from externally generated missiles includes all safety-related SSC on the plant site that have been provided to support the reactor facility. These include such elements as essential service water intakes, buried components (e.g., essential service water piping, storage tanks), and access openings and penetrations in structures.

The ASB reviews the functional operations or performance requirements for structures, systems, and components to assure conformance with the requirements of General Design Criterion 4 and identifies the SSC that are necessary for the safe shutdown of the reactor facility and the SSC whose failure could result in a significant release of radioactivity. Safety-related SSC are reviewed with respect to their capability to perform functions required for attaining and maintaining a safe shutdown condition during normal or accident conditions, mitigating the consequences of an accident, or preventing the occurrence of an accident assuming impact from externally generated missiles.

Based on their relation to safety, structures or areas of structures, systems or portions of systems, and components are identified as requiring protection from externally generated missiles if a missile could prevent the intended safety function, or if as a result of missile impact on a nonsafety-related SSC, its failure could prevent the intended safety function of a safety-related SSC.

II. ACCEPTANCE CRITERIA

Acceptability of the list of structures, systems, and components to be protected against externally generated missiles, presented in the applicant's safety analysis report (SAR), is based on specific general design criteria and regulatory guides.

The identification of structures, systems, and components to be protected against externally generated missiles is acceptable if it is in accordance with the following criteria:

1. General Design Criterion 4, with respect to protection of structures, systems, and components against the effects of externally generated missiles to maintain their essential safety functions.

2. Regulatory Guide 1.13, as related to the spent fuel pool systems and structures being capable of withstanding the effects of externally generated missiles and preventing missiles from contacting stored fuel assemblies.

3. Regulatory Guide 1.27, as related to the ultimate heat sink and connecting conduits being capable of withstanding the effects of externally generated missiles.

4. Regulatory Guide 1.115, as related to the protection of structures, systems, and components important to safety from the effects of turbine missiles.

5. Regulatory Guide 1.117, as related to the protection of structures, systems, and components important to safety from the effects of tornado missiles.

III. REVIEW PROCEDURES

The procedures below are used during the construction permit (CP) review to determine that the applicant's list of SSC that require protection from externally generated missiles is complete and meets the acceptance criteria given in subsection II. For operating license (OL) applications, the procedures are used to verify that the CP-stage list continues to be applicable and complete, or has been supplemented as appropriate. The reviewer will select and emphasize material from the paragraphs below, as may be appropriate for a particular case.

The first step in the review is to determine the safety-related SSC. Some structures and systems are considered safety-related in their entirety, others have only portions that are safety-related, and others are classified as nonsafety-related. In order to determine the safety category of the SSC, the ASB evaluates the SSC of the facility with respect to their necessity for achieving and maintaining safe reactor shutdown, or for performing accident prevention or mitigation functions. The information provided in the SAR pertaining to SSC design bases, design criteria, descriptions and safety evaluations, together with the system and component characteristic tables and safety classification tables are reviewed to identify safety functions performed by the SSC. The safety functions to be performed by the SSC in various designs remain essentially the same. However, the location or arrangement of the SSC and the methods used vary from plant to plant depending upon the individual designer. The reviewer identifies variations in design and evaluates them on a case-by-case basis.

The second step in the review is to determine the SSC, or portions of SSC, that require protection against externally generated missiles. The reviewer uses engineering judgment and the results of failure modes and effects analyses in conjunction with the results of reviews under other SRP sections for specific SSC in determining the need for missile protection. Most safety-related systems are located within structures that are resistant

to external missiles by virtue of design for other purposes (e.g., primary containment), or because the structures are constructed specifically to withstand missiles. Systems and components located within such structures are considered adequately protected. The reviewer concentrates his attention on safety-related SSC located outside such structures and on penetrations and access openings in the structures. Essential service water piping and components, storage tanks, and ultimate heat sink components are examples of SSC typically located outside missile-resistant structures. Depending on the nature and source of the externally generated missiles, protection may be provided by missile barriers for individual components or by locating independent redundant systems in compartments located in a missile protected structure. Physical separation alone is not normally an acceptable method of missile protection for redundant safety-related systems and components located in non-missile protected independent structures. Specific missile sources and the protection needed are considered in other SRP sections in the 3.5.1 series.

The reviewer determines that nonsafety-related structures, system or components are identified as requiring protection from externally generated missiles if as a result of their failure by a missile, the consequences could prevent the intended safety function of a safety-related SSC.

The reviewer compares his listing of SSC needing protection against externally generated missiles with the applicant's list of such SSC.

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information has been provided and his review supports conclusions of the following type, to be included in the staff's safety evaluation report:

> "The review of the structures, systems, and components to be protected from externally generated missiles included all safety-related structures, systems, and components provided to support the reactor facility. Based on the review of the applicant's proposed design criteria, design bases, and safety classifications for systems, structures, and components necessary for safe reactor shutdown, the staff concludes that the structures, systems, and components to be protected from externally generated missiles are in conformance with the Commission's regulations as set forth in General Design Criterion 4, "Environmental and Missile Design Bases," and meets the guidelines in Regulatory Guide 1.13, "Fuel Storage Facility Design Basis," Regulatory Guide 1.27, "Ultimate Heat Sink," Regulatory Guide 1.115, "Protection Against Low-Trajectory Turbine Missiles," and Regulatory Guide 1.117, "Tornado Design Classification," and therefore are acceptable.

V. REFERENCES

1. 10 CFR Part 50, Appendix A, General Design Criterion 4, "Environmental and Missile Design Bases."

2. Regulatory Guide 1.13, "Fuel Storage Facility Design Basis."

3. Regulatory Guide 1.27, "Ultimate Heat Sink."

4. Regulatory Guide 1.115, "Protection Against Low-Trajectory Turbine Missiles."

5. Regulatory Guide 1.117, "Tornado Design Classification."

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SECTION 3.5.3 BARRIER DESIGN PROCEDURES

REVIEW RESPONSIBILITIES

Primary - Structural Engineering Branch (SEB)

Secondary - None

I. AREAS OF REVIEW

The following areas relating to procedures utilized in the design of seismic Category I structures, shields, and barriers to withstand the effects of missile impact are reviewed.

1. Procedures utilized for the prediction of local damage in the impacted area. This includes estimation of the depth of penetration and, in case of concrete barriers, the potential for generation of secondary missiles by spalling or scabbing effects.

2. Procedures utilized for the prediction of the overall response of the barrier or portions thereof due to the missile impact. This includes assumptions on acceptable ductility ratios where elasto-plastic behavior is relied upon, and procedures for estimation of forces, moments, and shears induced in the barrier by the impact force of the missile.

II. ACCEPTANCE CRITERIA

The acceptance criteria for the areas of review are as follows:

1. For Local Damage Prediction

   a. In Concrete

   Among the empirical equations available to estimate missile penetration into concrete barriers, the one most commonly used is the modified Petry equation, as given by A. Amirikian (Ref. 1). The use of this equation is acceptable. However, other equations may be used provided the results obtained are either comparable to those obtained from the modified Petry equation, or penetration tests are conducted to validate the equation used. Sufficient thickness of concrete should be provided to prevent perforation and to prevent spalling or scabbing, when protection from spalling or scabbing is required. To prevent perforation, a concrete thickness

of at least twice the penetration thickness determined for an infinitely thick slab is acceptable. When spalling or scabbing is critical, the procedures used to determine the required thickness are reviewed on a case-by-case basis.

b. In Steel

The results of tests conducted by the Stanford Research Institute on the penetration of missiles into steel plates are summarized by W. B. Cottrell and A. W. Savolainen in "U.S. Reactor Containment Technology" (Ref. 2). The equations presented in reference 2 are acceptable. Other equations may be used provided the results are either comparable to those referenced above, or are validated by penetration tests.

c. In Composite Sections

For composite or multi-element missile barriers, procedures for prediction of local damage are acceptable if the residual velocity of the missile perforating the first element is considered as the striking velocity for the next element. For determining this residual velocity, the equations presented by Recht and Ipson (Ref. 3) are acceptable when the first barrier of a multi-element missile barrier is steel. When the first barrier is concrete, procedures are reviewed on a case-by-case basis.

2. For Overall Damage Prediction

The response of a structure or barrier to missile impact depends largely on the location of impact (e.g., mid-span of a slab or near a support), on the dynamic properties of the target and missile, and on the kinetic energy of the missile. In general, the assumption of plastic collisions is acceptable, where all of the missile initial momentum is transferred to the target and only a portion of its kinetic energy is absorbed as strain energy within the target. However, where elastic impacts are expected, the additional momentum transferred to the target by missile rebound should be included.

After it has been demonstrated that the missile will not penetrate the barrier, an equivalent static load concentrated at the impact area should then be determined, from which the structural response, in conjunction with other design loads, can be evaluated using conventional design methods. An acceptable procedure for such an analysis, where the impact is assumed to be plastic, is presented in a paper by Williamson and Alvy (Ref. 4). Other procedures may be used provided the results obtained are comparable to those referenced above.

III. REVIEW PROCEDURES

The reviewer selects and emphasizes material from the review procedures described below as may be appropriate for a particular case.

1. For the prediction of local damage, the equations proposed by the applicant for estimation of missile penetration are reviewed in the following manner:

   a. For missile penetration in concrete, the reviewer verifies that the applicant has made a commitment to utilize the modified Petry formula. If other equations are selected, the applicability and validity of such equations are reviewed to determine that the results are at least as conservative as those obtained from the modified Petry formula. If sufficient justification for the use of alternate equations is not provided, additional information is requested from the applicant at the first

stage of the review. The reviewer also verifies that the applicant has made a commitment to provide sufficient barrier thickness to prevent perforation and to prevent spalling or scabbing when protection from spalling or scabbing is considered necessary.

b. For missile penetration in steel, the reviewer verifies that the applicant has made a commitment to utilize the Stanford equations. If other equations are selected, the applicability and validity of such equations are reviewed to assure that the results are at least as conservative as those obtained from the Stanford equations. If sufficient justification for the use of alternate equations is not provided, additional information is requested from the applicant at the first stage of the review.

c. For missile penetration in composite or multi-element barriers, the reviewer verifies that the applicant has made a commitment to utilize the criteria delineated in Section II.1.c of this plan. If other criteria are proposed, the justification provided is reviewed to assure that such equations give results which are at least as conservative as those referenced above.

2. For the prediction of overall damage and response of the barrier, the reviewer verifies that the applicant has made a commitment to utilize the criteria delineated in Section II.2 of this plan. If other criteria are selected, the applicant's justification is reviewed to assure that the results obtained are at least as conservative as those delineated in Section II.2. If sufficient justification is not provided, additional information is requested from the applicant at the first stage of the review.

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information has been provided to satisfy the requirements of this review plan, and concludes that his evaluation is sufficiently complete and adequate to support the following type of conclusive statement to be included in the staff's safety evaluation report:

"The procedures utilized to determine the effects and loadings on seismic Category I structures and missile shields and barriers induced by design basis missiles selected for the plant are acceptable since these procedures provide a conservative basis for engineering design to assure that the structures or barriers are adequately resistant to and will withstand the effects of such forces.

"The use of these procedures provides reasonable assurance that in the event of design basis missiles striking seismic Category I structures or other missile shields and barriers, the structural integrity of the structures, shields, and barriers will not be impaired or degraded to an extent that will result in a loss of required protection. Seismic Category I systems and components protected by these structures are, therefore, adequately protected against the effects of missiles and will perform their intended safety function, if needed. Conformance with these procedures is an acceptable basis for satisfying, in part, the requirements of General Design Criteria 2 and 4."

V. REFERENCES

1. A. Amirikian, "Design of Protective Structures," Bureau of Yards and Docks, Publication No. NAVDOCKS P-51, Department of the Navy, Washington, D.C. (August 1950).

2. W. B. Cottrell and A. W. Savolainen, "U.S. Reactor Containment Technology," ORNL-NSIC-5, Vol. 1, Chapter 6, Oak Ridge National Laboratory.

3. R. F. Recht and T. W. Ipson, "Ballistic Perforation Dynamics," Journal of Applied Mechanics, Transactions of the ASME, Vol. 30, Series E, No. 3, September 1963.

4. R. A. Williamson and R. R. Alvy, "Impact Effect of Fragments Striking Structural Elements," Holmes and Narver, Inc., Revised November 1973.

5. 10 CFR Part 50, Appendix A, General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena."

6. 10 CFR Part 50, Appendix A, General Design Criterion 4, "Environmental and Missile Design Bases."

NUREG-75/087

U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

SECTION 3.6.1 PLANT DESIGN FOR PROTECTION AGAINST POSTULATED PIPING FAILURES IN FLUID SYSTEMS OUTSIDE CONTAINMENT

REVIEW RESPONSIBILITIES

Primary - Auxiliary and Power Conversion Systems Branch (APCSB)

Secondary - Structural Engineering Branch (SEB)
Electrical, Instrumentation and Control Systems Branch (EICSB)
Mechanical Engineering Branch (MEB)
Materials Engineering Branch (MTEB)
Containment Systems Branch (CSB)
Reactor Systems Branch (RSB)

I. AREAS OF REVIEW

The plant design for protection against piping failures outside containment is reviewed to assure that such failures would not cause the loss of needed functions of safety-related systems and to assure that the plant could be safely shut down in the event of such failures. The review includes high energy and moderate energy fluid system piping located outside of containment. If such a system penetrates containment (except for the auxiliary feedwater system) the review starts with the first isolation valve outside of containment. The review boundary for auxiliary feedwater systems extends either to the steam generator or to the feedwater (or steam) line, as appropriate. The specific areas of review are as follows:

1. APCSB reviews the general layout of high and moderate energy piping systems with respect to the plant arrangement criteria of Section B.1 of Branch Technical Position (BTP) APCSB 3-1, which is attached to this plan. Three arrangement situations are covered by the criteria and all three may be encountered in a single plant. They are:

   a. Arrangements where protection of safety-related plant features is provided by separation of high and moderate energy systems from essential systems and components.

   b. Arrangements where protection of safety-related plant features is provided by enclosing either the high and moderate energy systems or the safety-related features in protective structures.

   c. Arrangements where neither separation nor protective enclosures are practical and special protective measures are taken to ensure the operability of safety-related features.

2. APCSB, in conjunction with the secondary review branches as detailed below, reviews design features recommended in Section B.2 of BTP APCSB 3-1 as follows:

a. APCSB confirms with the RSB the seismic design classifications of systems and components defined as essential safety-related features in Appendix A of BTP APCSB 3-1.

b. APCSB identifies protective structures, piping restraints, and other measures used for protection against pipe breaks outside containment. Review of the specific aspects of these elements recommended in B.2.b of BTP APCSB 3-1 is done by the SEB and MEB as follows:

(1) SEB reviews the design of protective structures in connection with the review of other Category I structures under Standard Review Plan (SRP) 3.8.4.

(2) MEB reviews the design of piping restraints and other protective measures in connection with the review of break locations and dynamic effects of piping failures under SRP 3.6.2.

c. APCSB identifies portions of high and moderate energy fluid system piping between containment isolation valves that are subject to the recommendations of B.2.c of BTP APCSB 3-1. MEB reviews the design of these portions of piping in connection with the review of break locations and dynamic effects of piping failures under SRP 3.6.2.

d. MTEB reviews inservice inspection aspects of piping within protective structures or guard pipes, between containment isolation valves, or subject to other protective measures, with regard to the recommendations of B.2.d of BTP APCSB 3-1. This review is done in connection with the review of inservice inspection of Class 2 and 3 components under SRP 6.6.

3. APCSB reviews analyses of postulated piping failures with respect to the guidelines of Section B.3 of BTP APCSB 3-1. The locations and types of failures to be considered and the dynamic effects associated with the failures are reviewed by the MEB under SRP 3.6.2.

a. APCSB reviews analyses of piping failures in high and moderate energy fluid systems postulated according to the guidelines of B.3.a of BTP APCSB 3-1.

b. APCSB reviews the assumptions made in the analyses with regard to:

(1) The availability of offsite power.

(2) The failure of a single active component in systems used to mitigate the consequences of the piping failure.

(3) The special provisions applicable to certain dual purpose systems.

(4) The use of available systems to mitigate the consequences of the piping failure.

c. APCSB reviews the effects of postulated failures on the habitability of the control room and access to areas important to safe control of post-accident operations.

d. APCSB reviews the effects of piping failures in systems not designed to seismic Category I standards on essential systems and components.

4. Other secondary review evaluations are performed as required. These include:

a. EICSB verifies, on request, the capability of power supplies, instrumentation, and controls to initiate, actuate, and complete needed safety actions, considering the effects of a nearby piping failure such as the release of steam, water, or gases.

b. CSB verifies, on request, the magnitudes of any differential pressures in structures in which piping failures may be postulated.

II. ACCEPTANCE CRITERIA

Acceptance of the plant design for protection against postulated piping breaks outside containment, as described in the applicant's safety analysis report (SAR), will be based on conformance to Branch Technical Position APCSB 3-1, attached to this plan.

III. REVIEW PROCEDURES

All the systems of concern in this section have been reviewed under other standard review plans with respect to design functions for normal operation and for the prevention or mitigation of accidents. The review under this plan does not deal with individual system design requirements necessary to assure that each system performs as intended, but rather considers the protection necessary to assure the operation of such systems in the event of nearby piping failures. The reviewer will select and emphasize material in the review, as may be appropriate for a particular case.

1. APCSB reviews the information presented in the SAR identifying all high and moderate energy fluid systems, and verifies by comparison with individual system temperatures and pressures that they have been correctly identified. The reviewer will then, by reviewing system descriptions of the high and moderate energy piping runs, and by reviewing the appropriate system arrangement and piping drawings, examine the plant arrangement measures that were taken to assure protection from the effects of postulated pipe breaks of high energy systems, or of leakage cracks for moderate energy systems. The reviewer will determine from SAR that the following means either by themselves or in combination have been used by the applicant to achieve this protection:

a. High and moderate energy fluid systems are separated from essential systems and components, as defined in Appendix A to BTP APCSB 3-1. The reviewer inspects plant arrangement drawings and other information to verify that this is the case.

b. High and moderate energy fluid systems, or portions thereof, are enclosed within structures or compartments designed to protect nearby essential systems or components. Or, the essential systems and components are enclosed in protective structures. The reviewer traces the routing of the systems identified in the SAR as high or moderate energy systems on appropriate plant arrangement drawings, locates the postulated break locations specified in the applicant's analyses, and determines all locations where the effects from the breaks or leaks interface with safety-related equipment. The reviewer then determines that at these locations, enclosures have been provided that protect the safety-related equipment. Where questions as to break locations arise, the reviewer consults the MEB for a determination on the proper locations.

c. For cases where neither physical separation nor protective enclosures are considered practical by the applicant, the APCSB will review the SAR information to verify the following:

(1) The reasons for which the applicant judged both physical separation and system enclosure to be impractical as means of protection are consistent with Subsection B.1.c of BTP APCSB 3-1.

(2) Redundant design features or additional protection have been provided in these situations and are such that failure modes and effects analyses for all failure situations show that the performance of safety features will be assured, assuming a single active failure in any required system. These analyses are done under the criteria and assumptions of Section B.3 of BTP APCSB 3-1. Special measures taken to provide additional protection are reviewed on an individual case basis, with assistance from the secondary review branches as needed.

2. APCSB reviews the information presented in the SAR that identifies the principal design features. The reviewer performs his evaluation by comparing the design basis information given in the SAR with that described in Section B.2 of BTP APCSB 3-1. By this comparison of individual design features, the reviewer verifies as follows that the necessary measures have been provided by the applicant in his design.

a. APCSB reviews the seismic design classification of plant systems and checks with RSB to verify that essential systems and components, as defined in Appendix A of BTP APCSB 3-1, have been designed to meet the seismic requirements of Section B.2.a of BTP APCSB 3-1.

b. APCSB, with assistance from SEB and MEB, reviews the design features provided for protective structures or compartments, fluid system piping restraints, and other protective measures as described in Section B.2.b of BTP APCSB 3-1. The reviewer compares the design features and bases given in the SAR with the stated section in BTP APCSB 3-1. The comparative review may include the use of plant arrangement and layout drawings as necessary to clarify the design intentions and implementation. In the majority of case reviews, SAR statements and drawings indicating

that the design meets the intent of the acceptance criteria are accepted. However, there may be cases where engineering judgment and independent staff analyses are needed to verify the capability of structures and components to withstand the dynamic pressure and mechanical effects of a pipe rupture.

c. APCSB reviews the SAR information, as supplemented by engineering sketches or drawings where necessary, to determine that fluid system piping between containment isolation valves conforms to Section B.2.c of BTP APCSB 3-1. This includes piping penetrations between single and dual barrier containments that may have enclosing protective structures. The review is mainly performed on a comparative basis by APCSB. MEB reviews these piping details to verify the design limits, break locations, and dynamic effects, in accordance with BTP MEB 3-1.

d. APCSB reviews the broad aspects of the applicant's inservice inspection program by comparison of the items included in the program with the provisions given in Section B.2.d of BTP APCSB 3-1. The review of the actual inservice inspection program is performed by MTEB.

3. APCSB reviews the results of the applicant's evaluation of the consequences of postulated piping failures of high and moderate energy fluid piping systems. The type and location of each postulated piping failure (i.e., longitudinal or circumferential) in either a high or moderate energy system will be reviewed by MEB on the basis of BTP MEB 3-1. The review by APCSB will be based upon the information provided by applicants in the SAR concerning the effects of postulated failures on essential equipment and the ability of the plant to be safely shut down, as described in Section B.3 of BTP APCSB 3-1.

The reviewer verifies that the applicant's evaluation has properly considered the following points, and in certain cases, as necessary, performs an independent evaluation especially with regard to single failure analyses.

a. APCSB reviews the applicant's plant arrangements and design features using layout drawings to assure that all potentially affected essential systems and components have been considered with respect to the effects of an assumed pipe break.

b. APCSB reviews the effects of postulated piping failures as determined from the information given in the SAR. The reviewer will confirm the results of the applicant's evaluations by performing a comparative, but abbreviated as appropriate, failure modes and effects analysis that includes the considerations given in Section B.3.d of BTP APCSB 3-1 for the following effects:

(1) The availability of offsite power.

(2) The effects of a single active component failure in systems necessary to mitigate consequences of the postulated piping break.

(3) Permissible exclusions to (2) above based upon the provision given in Section B.3.b(3) of BTP APCSB 3-1 for certain dual purpose moderate energy systems.

(4) The considerations involved in to the selection of available systems to mitigate the consequences of the piping failure.

c. The reviewer will verify from a review of arrangement drawings that control room habitability or access to necessary surrounding areas is not jeopardized as a consequence of the postulated piping failure.

d. APCSB evaluates the applicant's analysis of the postulated failure of non-seismic Category I piping systems by performing a failure modes and effects analysis using SAR information and engineering sketches as necessary.

4. Systems defined in Appendix A to BTP APCSB 3-1 as "essential systems" are those that are needed to shut down the reactor and mitigate the consequences of the pipe break for a given postulated piping break. However, depending upon the type and location of the postulated pipe break, certain safety equipment may not be classed as "essential" for that particular event (e.g., emergency power system or high and low pressure core spray systems). On the other hand, some safety equipment will be "essential" for almost all cases (e.g., service water to ultimate heat sink). Table 3.6.1-1 is a list of those essential systems generally in the latter category.

TABLE 3.6.1-1

SYSTEMS USUALLY REQUIRED FOR SAFE SHUTDOWN

PWR	BWR
Service Water System	Service Water System
Auxiliary Feedwater System	Reactor Coolant Injection System
Volume Control System	
Decay Heat Removal System	Residual Heat Removal System
Component Cooling Water System (if provided)	Component Cooling Water System (if provided)

Table 3.6.1-2 is a listing of systems typically classified as either high or moderate energy systems that are located outside the primary containment in pressurized water reactor (PWR) and boiling water reactor (BWR) plants.

TABLE 3.6.1-2

TYPICAL HIGH ENERGY SYSTEMS OUTSIDE CONTAINMENT

PWR	BWR
Main Steam Line System	Main Steam Line System
Main Feedwater Line System	Main Feedwater Line System
Auxiliary Feedwater System	High Pressure Core Spray System
Volume Control System	Process Sampling System
Process Sampling System	Condensate System
Condensate System	Reactor Cleanup System
Steam Generator Blowdown Line	Standby Liquid Control System

TYPICAL MODERATE ENERGY SYSTEMS OUTSIDE CONTAINMENT

PWR	BWR
Service Water System	Service Water System
Decay Heat Removal System (outside of reactor coolant pressure boundary)	Residual Heat Removal System (outside of reactor coolant pressure boundary)
Circulating Water System	Circulating Water System
Fire Protection System	Fire Protection System
Component Cooling Water System	Component Cooling Water System

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information has been provided and his review supports conclusions of the following type, to be included in the staff's safety evaluation report:

"The review of the plant design for protection against postulated piping failures outside containment included all high and moderate energy piping systems located outside containment. The review of these high and moderate energy systems for the _____ plant included layout drawings, piping and instrumentation diagrams, and descriptive information. [The review has included the applicant's proposed design criteria and design bases for the systems, structures, and components of interest, the adequacy of those criteria and bases, and the functions necessary to maintain the capability for a safe plant shutdown during any failure of high or moderate energy system piping. (CP)] [The review has included the applicant's analysis of the manner in which the design of all structures, systems, and components conforms to the design criteria and design bases and demonstrates the ability to perform a safe plant shutdown after any postulated piping failure of a high or moderate energy system. (OL)]

"The staff concludes that the facility design for protection against postulated piping failures outside containment conforms to the Commissions's regulations and to

applicable regulatory guides, staff technical positions, and industry standards, and is acceptable."

V. References

1. Branch Technical Positions APCSB 3-1, "Protection Against Postulated Piping Failures in Fluid Systems Outside Containment," attached to this plan, and MEB 3-1, "Postulated Break and Leakage Locations in Fluid System Piping Outside Containment," attached to Standard Review Plan 3.6.2.

PROTECTION AGAINST POSTULATED PIPING FAILURES IN FLUID SYSTEMS OUTSIDE CONTAINMENT

A. BACKGROUND

General Design Criterion 4, "Environmental and Missile Design Bases," of Appendix A to 10 CFR Part 50, "General Design Criteria for Nuclear Power Plants," requires that systems and components important to safety "...shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit." Guidance on acceptable design approaches to meet General Design Criterion 4 for existing plants and for plants for which applications for construction permits were then under review was provided in letters to applicants and licensees from A. Giambusso, Deputy Director of Licensing for Reactor Projects, most of which were dated in December 1972. The guidance document from these letters is attached as Appendix B to this position. Similar interim guidance for new plants was provided in a letter to applicants, prospective applicants, reactor vendors, and architect-engineers from J. F. O'Leary, Director of Licensing, dated July 12, 1973. This document is attached as Appendix C to this position.

Guidance is available for protection against pipe whipping and other effects of postulated fluid system piping failures (e.g., a break or rupture resulting in a loss-of-coolant accident) of systems and components important to safety located within primary reactor containment. As an example, this problem is addressed by Regulatory Guide 1.46, "Protection Against Pipe Whip Inside Containment."

Reviews of nuclear power plant designs have indicated that the functional or structural integrity of systems and components required for safe shutdown of the reactor and maintenance of cold shutdown conditions could be endangered by fluid system piping failures at locations outside containment. The staff has evolved an acceptable approach for the design, including the arrangement, of fluid systems located outside of containment to assure that the plant can be safely shut down in the event of piping failures outside containment. This approach is set forth in this position and in the companion Branch Technical Position (BTP) attached to Standard Review Plan 3.6.2, BTP MEB 3-1.

It is the intent of this design approach that postulated piping failures in fluid systems should not cause a loss of function of essential safety-related systems and that nuclear plants should be able to withstand postulated failures of any fluid system piping outside containment, taking into account the direct results of such failure and the further failure of any single active component, with acceptable offsite consequences.

The detailed provisions of the position below and of BTP MEB 3-1 are intended to implement this intent with due consideration of the special nature of certain dual purpose systems and the need to define and to limit to a finite number the types and locations of piping failures

to be analyzed. Although various measures for the protection of safety-related systems and components are outlined in this position, the preferred method of protection is based upon separation and isolation by plant arrangement.

B. BRANCH TECHNICAL POSITION

1. Plant Arrangement

Protection of essential systems and components[1/] against postulated piping failures in high or moderate energy fluid systems that operate during normal plant conditions and that are located outside of containment should be provided by one of the following plant arrangement considerations:

a. Plant arrangements should separate fluid system piping from essential systems and components. Separation should be achieved by plant physical layouts that provide sufficient distances between essential systems and components and fluid system piping such that the effects of any postulated piping failure therein (e.g., pipe whip, jet impingement, and the environmental conditions resulting from the escape of contained fluids as appropriate to high or moderate-energy fluid system piping) cannot impair the integrity or operability of essential systems and components.

b. Fluid system piping or portions thereof not satisfying the provisions of B.1.a should be enclosed within structures or compartments designed to protect nearby essential systems and components. Alternatively, essential systems and components may be enclosed within structures or compartments designed to withstand the effects of postulated piping failures in nearby fluid systems.

c. Plant arrangements or system features that do not satisfy the provisions of either B.1.a or B.1.b should be limited to those for which the above provisions are impractical because of the stage of design or construction of the plant; because the plant design is based upon that of an earlier plant accepted by the staff as a base plant under the Commission's standardization and replication policy; or for other substantive reasons such as particular design features of the fluid systems. Such cases may arise, for example, (1) at interconnections between fluid systems and essential systems and components, or (2) in fluid systems having dual functions (i.e., required to operate during normal plant conditions as well as to shut down the reactor). In these cases, redundant design features that are separated or otherwise protected from postulated piping failures, or additional protection, should be provided so that the effects of postulated piping failures are shown by the analyses and guidelines of B.3 to be acceptable. Additional protection may be provided by restraints and barriers or by designing or testing essential systems and components to withstand the effects associated with postulated piping failures.

2. Design Features

a. Essential systems and components should be designed to meet the seismic design requirements of Regulatory Guide 1.29.

[1/] See Appendix A for definitions of underlined phrases.

b. Protective structures or compartments, fluid system piping restraints, and other protective measures should be designed in accordance with the following:

(1) Protective structures or compartments needed to implement B.1 should be designed to seismic Category I requirements. The protective structures should be designed to withstand the effects of a postulated piping failure (i.e., pipe whip, jet impingement, pressurization of compartments, water spray, and flooding, as appropriate) in combination with loadings associated with the operating basis earthquake and safe shutdown earthquake within the respective design load limits for structures. Piping restraints, if used, may be taken into account to limit effects of the postulated piping failure.

(2) High-energy fluid system piping restraints and protective measures should be designed such that a postulated break in one pipe cannot, in turn, lead to rupture of other nearby pipes or components if the secondary rupture could result in consequences that would be considered unacceptable for the initial postulated break. An unrestrained whipping pipe should be considered capable of (a) rupturing impacted pipes of smaller nominal pipe sizes and (b) developing through-wall leakage cracks in larger nominal pipe sizes with thinner wall thicknesses, except where experimental or analytical data for the expected range of impact energies demonstrate the capability to withstand the impact without failure.

c. Fluid system piping in containment penetration areas should meet the following design provisions:

(1) Portions of fluid system piping between the required restraints located inside and outside containment beyond the isolation valves of single barrier containment structures (including any rigid connection to the containment penetration) that connect, on a continuous or intermittent basis, to the reactor coolant pressure boundary, or the steam and feedwater systems of PWR plants, should be designed to the stress limits specified in B.1.b or B.2.b of Branch Technical Position (BTP) MEB 3-1, attached to Standard Review Plan 3.6.2.

These portions of high-energy fluid system piping should be provided with pipe whip restraints that are capable of resisting bending and torsional moments produced by a postulated piping failure either upstream or downstream of the containment isolation valves. The restraints should be located reasonably close to the containment isolation valves and should be designed to withstand the loadings resulting from a postulated piping failure beyond these portions of piping so that neither isolation valve operability nor the leaktight integrity of the containment will be impaired.

(2) Portions of fluid system piping between the required restraints located inside and outside containment beyond the isolation valves of dual barrier

containment structures should also meet the design provisions of B.2.c.(1). In addition, those portions of piping that pass through the containment annulus, and whose postulated failure could affect the leaktight integrity of the containment structure or result in pressurization of the containment annulus beyond design limits should be provided with an enclosing protective structure.

For the purpose of establishing the design parameters (i.e., pressure, temperature) of the enclosing protective structure, a full flow area opening should be assumed in that portion of piping within the enclosing structure and vent areas should be taken into account, if provided, in the enclosing structure. Where guard pipes for individual process pipes are used as an enclosing protective structure, such guard pipes should be designed to meet the requirements specified in B.1.b(6) of BTP MEB 3-1.

(3) Terminal ends of the piping runs extending beyond these portions of high-energy fluid system piping should be considered to originate at a point adjacent to the required pipe whip restraints located inside and outside containment.

(4) Piping classification as required by Regulatory Guide 1.26 should be maintained without change until beyond the outboard restraint. If the restraint is located at the isolation valve, a classification change at the valve interface is acceptable.

d. Inservice examination and related design provisions should be in accordance with the following:

(1) The protective measures, structures, and guard pipes should not prevent the access required to conduct the inservice examinations specified in the ASME Boiler and Pressure Vessel Code, Section XI, Division 1, "Rules for Inspection and Testing of Components in Light-Water Cooled Plants."

(2) For those portions of fluid system piping identified in B.2.c, includes piping running from inboard to outboard restraints in containment penetration areas, the extent of inservice examinations completed during each inspection interval (IWA-2400, ASME Code, Section XI) should provide 100 percent volumetric examination of circumferential and longitudinal pipe welds within the boundary of these portions of piping.

(3) For those portions of fluid systems piping enclosed in guard pipes, inspection ports should be provided in guard pipes to permit the required examination of circumferential pipe welds. Inspection ports should not be located in that portion of the guard pipe passing through the annulus of dual barrier containment structures.

(4) The areas subject to examination should be defined in accordance with Examination Categories C-F and C-G for Class 2 piping welds in Tables IWC-2520.

3. Analyses and Effects of Postulated Piping Failures

a. To show that the plant arrangement and design features provide the necessary protection of essential systems and components, piping failures should be postulated in accordance with BTP MEB 3-1, attached to Standard Review Plan 3.6.2. In applying the provisions of BTP MEB 3-1, each longitudinal or circumferential break in high-energy fluid system piping or leakage crack in moderate-energy fluid system piping should be considered separately as a single postulated initial event occurring during normal plant conditions. An analysis should be made of the effects of each such event, taking into account the provisions of BTP MEB 3-1 and of the system and component operability considerations of B.3.b. below. The effects of each postulated piping failure should be shown to result in offsite consequences within the guidelines of 10 CFR Part 100 and to meet the provisions of B.3.c and d below.

b. In analyzing the effects of postulated piping failures, the following assumptions should be made with regard to the operability of systems and components:

(1) Offsite power should be assumed to be unavailable if a trip of the turbine-generator system or reactor protection system is a direct consequence of the postulated piping failure.

(2) A single active component failure should be assumed in systems used to mitigate consequences of the postulated piping failure and to shut down the reactor, except as noted in B.3.b.(3) below. The single active component failure is assumed to occur in addition to the postulated piping failure and any direct consequences of the piping failure, such as unit trip and loss of offsite power.

(3) Where the postulated piping failure is assumed to occur in one of two or more redundant trains of a dual-purpose moderate-energy essential system, i.e., one required to operate during normal plant conditions as well as to shut down the reactor and mitigate the consequences of the piping failure, single failures of components in the other train or trains of that system only need not be assumed provided the system is designed to seismic Category I standards, is powered from both offsite and onsite sources, and is constructed, operated, and inspected to quality assurance, testing, and inservice inspection standards appropriate for nuclear safety systems. Examples of systems that may, in some plant designs, qualify as dual-purpose essential systems are service water systems, component cooling systems, and residual heat removal systems.

(4) All available systems, including those actuated by operator actions, may be employed to mitigate the consequences of a postulated piping failure. In judging the availability of systems, account should be taken of the postulated failure and its direct consequences such as unit trip and loss of offsite power, and of the assumed single active component failure and its direct consequences. The feasibility of carrying out operator actions should be judged on the basis of ample time and adequate access to equipment being available for the proposed actions.

c. The effects of a postulated piping failure, including environmental conditions resulting from the escape of contained fluids, should not preclude habitability of the control room or access to surrounding areas important to the safe control of reactor operations needed to cope with the consequences of the piping failure.

d. A postulated failure of piping not designed to seismic Category I standards should not result in any loss of capability of essential systems and components to withstand the further effects of any single active component failure and still perform all functions required to shut down the reactor and mitigate the consequences of the postulated piping failure.

4. Implementation

a. Designs of plants for which construction permit applications are tendered after July 1, 1975 should conform to the provisions of this position.

b. Designs of plants for which construction permit applications are tendered after July 1, 1973 and before July 1, 1975 should conform to the provisions of either (a) the letter of July 12, 1973 from J. F. O'Leary, Appendix C to this position, or (b) this position, at the option of the applicants.

c. Designs of plants for which construction permit applications were tendered before July 1, 1973 and operating licenses are issued after July 1, 1975 should follow the guidance provided in the December 1972 letter from A. Giambusso, Appendix B to this position and provide analyses of moderate energy lines made in conformance with B.3 of this position, as part of the operating license application for these plants to demonstrate that acceptable protection against the effects of piping failures outside containment has been provided. Alternately, this position may be used in its entirety as an acceptable basis for this finding.

For plants in this category for which construction permits are not issued as of February 1, 1975, a commitment by the applicant to either (a) follow the guidance of Appendix B and submit B.3 analyses of moderate energy lines with the plant final safety analysis report (FSAR), or (b) conform the plant design to the provisions of this position, should provide an acceptable basis for issuance of the construction permit with regard to effects of piping failures outside containment.

d. Designs of plants for which operating licenses are issued before July 1, 1975 are considered acceptable with regard to effects of piping failures outside containment on the basis of the analyses made and measures taken by applicants and licensees in response to the December 1972 letter from A. Giambusso, and the staff review and acceptance of these analyses and measures.

For plants in this category for which the staff review and acceptance of protection against the effects of piping failures outside containment is not substantially complete as of February 1, 1975, a commitment by the applicant to carry out analyses according to B.3 of this position, to submit them for staff review, and to carry out any system modifications found necessary before extended operation of the plant at power levels above one-half the license power level, should provide an acceptable basis for issuance of the operating license.

C. REFERENCES

1. 10 CFR Part 50, Appendix A, General Design Criterion 4, "Environmental and Missile Design Bases."

2. Regulatory Guide 1.29, "Seismic Design Classification."

3. Regulatory Guide 1.46, "Protection Against Pipe Whip Inside Containment."

4. Letter from A. Giambusso, Deputy Director for Reactor Projects, Directorate of Licensing, to applicants and licensees, December 1972, and attachment entitled "General Information Required for Consideration of the Effects of a Piping System Break Outside Containment." The corrected attachment is Appendix B to this position.

5. Letter from J. F. O'Leary, Director of Licensing, to applicants, reactor venders, and architect-engineers, July 12, 1972, and attachment entitled "Criteria for Determination of Postulated Break and Leakage Locations in High and Moderate Energy Fluid Piping Systems Outside of Containment Structures." The letter and attachment is Appendix C to this position.

6. ASME Boiler and Pressure Vessel Code, Sections III and XI, American Society of Mechanical Engineers.

APPENDIX A
BRANCH TECHNICAL POSITION APCSB 3-1
DEFINITIONS

Essential Systems and Components. Systems and components required to shut down the reactor and mitigate the consequences of a postulated piping failure, without offsite power.

Fluid Systems. High and moderate energy fluid systems that are subject to the postulation of piping failures outside containment against which protection of essential systems and components is needed.

High-Energy Fluid Systems. Fluid systems that, during normal plant conditions, are either in operation or maintained pressurized under conditions where either or both of the following are met:

a. maximum operating temperature exceeds 200°F, or

b. maximum operating pressure exceeds 275 psig.

Moderate-Energy Fluid Systems. Fluid systems that, during normal plant conditions, are either in operation or maintained pressurized (above atmospheric pressure) under conditions where both of the following are met:

a. maximum operating temperature is 200°F or less, and

b. maximum operating pressure is 275 psig or less

Normal Plant Conditions. Plant operating conditions during reactor startup, operation at power, hot standby, or reactor cooldown to cold shutdown condition.

Upset Plant Conditions. Plant operating conditions during system transients that may occur with moderate frequency during plant service life and are anticipated operational occurrences, but not during system testing.

Postulated Piping Failures. Longitudinal and circumferential breaks in high-energy fluid system piping and through-wall leakage cracks in moderate-energy fluid system piping postulated according to the provisions of BTP MEB 3-1, attached to SRP 3.6.2.

S_h and S_A . Allowable stresses at maximum (hot) temperature and allowable stress range for thermal expansion, respectively, as defined in Article NC-3600 of the ASME Code, Section III.

S_m. Design stress intensity as defined in Article NB-3600 of the ASME Code, Section III.

Single Active Component Failure. Malfunction or loss of function of a component of electrical or fluid systems. The failure of an active component of a fluid system is considered to be a loss of component function as a result of mechanical, hydraulic, pneumatic, or electrical

malfunction, but not the loss of component structural integrity. The direct consequences of a single active component failure are considered to be part of the single failure.

Terminal Ends. Extremities of piping runs that connect to structures, components (e.g., vessels, pumps, valves), or pipe anchors that act as rigid constraints to piping thermal expansion. A branch connection to a main piping run is a terminal end of the branch run.

Intersections of runs of comparable size and fixity need not be considered terminal ends when so justified in the analysis. Terminal ends for the purpose of postulating breaks should be selected at points located immediately outside or beyond the required pipe whip restraints located inside and outside containment at penetration areas.

In piping runs which are maintained pressurized during normal plant conditions for only a portion of the run (i.e., up to the first normally closed valve) a terminal end of such runs is the piping connection to this closed valve.

APPENDIX B

BRANCH TECHNICAL POSITION APCSB 3-1

This appendix consists of the attachment to the letters sent by A. Giambusso, Deputy Director for Reactor Projects, Directorate of Licensing, in December 1972 to applicants and licensees on the subject of postulated piping failures outside containment. The attachment provided guidance on measures to be taken and on information to be submitted. An errata sheet for the attachment was sent in January 1973 to recipients of the original letters. The attachment as given here has been corrected for the errata.

General Information Required for Consideration of the Effects of a Piping System Break Outside Containment

The following is a general list of information required for AEC review of the effects of a piping system break outside containment, including the double ended rupture of the largest pipe in the main steam and feedwater systems, and for AEC review of any proposed design changes that may be found necessary. Since piping layouts are substantially different from plant to plant, applicants and licensees should determine on an individual plant basis the applicability of each of the following items for inclusion in their submittals.

1. The systems (or portions of systems) for which protection against pipe whip is required should be identified. Protection from pipe whip need not be provided if any of the following conditions will exist:
   (a) Both of the following piping system conditions are met:
      (1) the service temperature is less than 200°F; and
      (2) the design pressure is 275 psig or less; or
   (b) The piping is physically separated (or isolated) from structures, systems, or components important to safety by protective barriers, or restrained from whipping by plant design features, such as concrete encasement; or
   (c) Following a single break, the unrestrained pipe movement of either end of the ruptured pipe in any possible direction about a plastic hinge formed at the nearest pipe whip restraint cannot impact any structure, system, or component important to safety; or
   (d) The internal energy level[1/] associated with the whipping pipe can be demonstrated to be insufficient to impair the safety function of any structure, system or component to an unacceptable level.

[1/] Footnotes are collected at the end of this appendix.

2. Design basis break locations should be selected in accordance with the following pipe whip protection criteria: however, where pipes carrying high energy fluids are routed in the vicinity of structures and systems necessary for safe shutdown of the nuclear plant, supplemental protection of those structures and systems shall be provided to cope with the environmental effects (including the effects of jet impingement) of a single postulated open crack at the most adverse location(s) with regard to those essential structures and systems, the length of the crack being chosen not to exceed the critical crack size. The critical crack size is taken to be 1/2 the pipe diameter in length and 1/2 the wall thickness in width.

The criteria used to determine the design basis piping break locations in the piping systems should be equivalent to the following:

(a) ASME Section III Code Class I piping[2/] breaks should be postulated to occur at the following locations in each piping run[3/] or branch run:

(1) The terminal ends;

(2) Any intermediate locations between terminal ends where the primary plus secondary stress intensities S_n (circumferential or longitudinal) derived on an elastically calculated basis under the loadings associated with one-half safe shutdown earthquake and operational plant conditions[4/] exceeds 2.0 S_m[5/] for ferritic steel, and 2.4 S_m for austenitic steel;

(3) Any intermediate locations between terminal ends where the cumulative usage factor (U)[6/] derived from the piping fatigue analysis and based on all normal, upset, and testing plant conditions exceeds 0.1; and

(4) At intermediate locations in addition to those determined by (1) and (2) above, selected on a reasonable basis as necessary to provide protection. As a minimum, there should be two intermediate locations for each piping run or branch run.

(b) ASME Section III Code Class 2 and 3 piping breaks should be postulated to occur at the following locations in each piping run or branch run:

(1) The terminal ends;

(2) Any intermediate locations between terminal ends where either the circumferential or longitudinal stresses derived on an elastically calculated basis under the loadings associated with seismic events and operational plant conditions exceed 0.8 $(S_h + S_A)$[7/] or the expansion stresses exceed 0.8 S_A; and

(3) Intermediate locations in addition to these determined by (2) above, selected on a reasonable basis as necessary to provide protection. As a minimum, there should be two intermediate locations for each piping run or branch run.

3. The criteria used to determine the pipe break orientation at the break locations as specified under (2) above should be equivalent to the following:

(a) Longitudinal[8/] breaks in piping runs and branch runs, 4 inches nominal pipe size and larger, and/or

(b) Circumferential[9/] breaks in piping runs and branch runs exceeding 1 inch nominal pipe size.

4. A summary should be provided of the dynamic analyses applicable to the design of Category I piping and associated supports which determine the resulting loadings as a result of a postulated pipe break including:
   (a) The locations and number of design basis breaks on which the dynamic analyses are based.
   (b) The postulated rupture orientation, such as a circumferential and/or longitudinal break(s), for each postulated design basis break location.
   (c) A description of the forcing functions used for the pipe whip dynamic analyses including the direction, rise time, magnitude, duration, and initial conditions that adequately represent the jet stream dynamics and the system pressure difference.
   (d) Diagrams of mathematical models used for the dynamic analysis.
   (e) A summary of the analyses which demonstrates that unrestrained motion of ruptured lines will not damage to an unacceptable degree, structures, systems, or components important to safety, such as the control room.

5. A description should be provided of the measures, as applicable, to protect against pipe whip, blowdown jet, and reactive forces including:
   (a) Pipe restraint design to prevent whip impact;
   (b) Protective provisions for structures, systems, and components required for safety against pipe whip and blowdown jet and reactive forces;
   (c) Separation of redundant features;
   (d) Provisions to separate physically piping and other components of redundant features; and
   (e) A description of the typical pipe whip restraints and a summary of number and location of all restraints in each system.

6. The procedures that will be used to evaluate the structural adequacy of Category I structures and to design new seismic Category I structures should be provided including:
   (a) The method of evaluating stresses, e.g., the working stress method and/or the ultimate strength method that will be used;
   (b) The allowable design stresses and/or strains; and
   (c) The load factors and the load combinations.

7. The structural design loads, including the pressure and temperature transients, the dead, live and equipment loads, and the pipe and equipment static, thermal, and dynamic reactions should be provided.

8. Seismic Category I structural elements such as floors, interior walls, exterior walls, building penetrations, and the buildings as a whole should be analyzed for eventual reversal of loads due to the postulated accident.

9. If new openings are to be provided in existing structures, the capabilities of the modified structures to carry the design loads should be demonstrated.

10. Verification that failure of any structure, including non-seismic Category I structures,

caused by the accident, will not cause failure of any other structure in a manner to adverseley affect:

(a) Mitigation of the consequences of the accidents; and

(b) Capability to bring the unit(s) to a cold shutdown condition.

11. Verification that rupture of a pipe carrying high energy fluid will not directly or indirectly result in:

(a) Loss of required redundancy in any portion of the protection system (as defined in IEEE Std 279), Class IE electric system (as defined in IEEE Std 308), engineered safety feature equipment, cable penetrations, or their interconnecting cables required to mitigate the consequences of that accident and place the reactor(s) in a cold shutdown condition; or

(b) Environmentally induced failures caused by a leak or rupture of the pipe which would not of itself result in protective action but does disable protection functions. In this regard, a loss of redundancy is permitted; but a loss of function is not permitted. For such situations, plant shutdown is required.

12. Assurance should be provided that the control room will be habitable and its equipment functional after a steam line or feedwater line break or that the capability for shutdown and cooldown on the unit(s) will be available in another habitable area.

13. Environmental qualification should be demonstrated by test for that electrical equipment required to function in the steam-air environment resulting from a high energy line break. The information required for our review should include the following:

(a) Identification of all electrical equipment necessary to meet requirements of (11) above. The time after the accident in which they are required to operate should be given.

(b) The test conditions and the results of test data showing that the systems will perform their intended function in the environment resulting from the postulated accident and time interval of the accident. Environmental conditions used for the tests should be selected from a conservative evaluation of accident conditions.

(c) The results of a study of steam systems identifying locations where barriers will be required to prevent steam jet impingement from disabling a protection system. The design criteria for the barriers should be stated and the capability of the equipment to survive within the protected environment should be described.

(d) An evaluation of the capability for safety-related electrical equipment in the control room to function in the environment that may exist following a pipe break accident should be provided. Environmental conditions used for the evaluation should be selected from conservative calculations of accident conditions.

(e) An evaluation to assure that the onsite power distribution system and onsite sources (diesels and batteries) will remain operable throughout the event.

14. Design diagrams and drawings of the steam and feedwater lines including branch lines showing the routing from containment to the turbine building should be provided. The drawings should show elevations and include the location relative to the piping runs of safety-related equipment including ventilation equipment, intakes, and ducts.

15. A discussion should be provided of the potential for flooding of safety-related equipment in the event of failure of a feedwater line or any other line carrying high energy fluid.

16. A description should be provided of the quality control and inspection programs that will be required or have been utilized for piping systems outside containment.

17. If leak detection equipment is to be used in the proposed modifications, a discussion of its capabilities should be provided.

18. A summary should be provided of the emergency procedures that would be followed after a pipe break accident, including the automatic and manual operations required to place the reactor unit(s) in a cold shutdown condition. The estimated times following the accident for all equipment and personnel operational actions should be included in the procedure summary.

19. A description should be provided of the seismic and quality classification of the high energy fluid piping systems including the steam and feedwater piping that run near structures, systems, or components important to safety.

20. A description should be provided of the assumptions, methods, and results of analyses, including steam generator blowdown, used to calculate the pressure and temperature transients in compartments, pipe tunnels, intermediate buildings, and the turbine building following a pipe rupture in these areas. The equipment assumed to function in the analyses should be identified and the capability of systems required to function to meet a single active component failure should be described.

21. A description should be provided of the methods or analyses performed to demonstrate that there will be no adverse effects on the primary and/or secondary containment structures due to a pipe rupture outside these structures.

1/ The internal fluid energy level associated with the pipe break reaction may take into account any line restrictions (e.g., flow limiter) between the pressure source and break location, and the effects of either single-ended or double-ended flow conditions, as applicable. The energy level in a whipping pipe may be considered as insufficient to rupture an impacted pipe of equal or greater nominal pipe size and equal or heavier wall thickness.

2/ Piping is a pressure-retaining component consisting of straight or curved pipe and pipe fittings (e.g., elbows, tees, and reducers).

3/ A piping run interconnects components such as pressure vessels, pumps, and rigidly fixed valves that may act to restrain pipe movement beyond that required for design thermal displacement. A branch run differs from a piping run only in that it originates at a piping intersection, as a branch of the main pipe run.

4/ Operational plant conditions include normal reactor operation, upset conditions (e.g., anticipated operational occurrences) and testing conditions.

5/ S_m is the design stress intensity as specified in Section III of the ASME Boiler and Pressure Vessel Code, "Nuclear Plant Components."

6/ U is the cumulative usage factor as specified in Section III of the ASME Boiler and Pressure Vessel Code, "Nuclear Power Plant Components."

7/ S_h is the stress calculated by the rules of NC-3600 and ND-3600 for Class 2 and 3 components, respectively, of the ASME Code Section III Winter 1972 Addenda.

S_A is the allowable stress range for expansion stress calculated by the rules of NC-3600 of the ASME Code, Section III, or the USA Standard Code for Pressure Piping, ANSI B31.1.0-1967.

8/ Longitudinal breaks are parallel to the pipe axis and oriented at any point around the pipe circumference. The break area is equal to the effective cross-sectional flow area upstream of the break location. Dynamic forces resulting from such breaks are assumed to cause lateral pipe movements in the direction normal to the pipe axis.

9/ Circumferential breaks are perpendicular to the pipe axis, and the break area is equivalent to the internal cross-sectional area of the ruptured pipe. Dynamic forces resulting from such breaks are assumed to separate the piping axially, and cause whipping in any direction normal to the pipe axis.

APPENDIX C

BRANCH TECHNICAL POSITION APCSB 3-1

This appendix consists of the letter and attachment sent by J. F. O'Leary, Director of Licensing, to applicants, reactor vendors, and architect-engineers on the subject of postulated piping failures outside containment. The letter was dated July 12, 1973.

Late last year, the Atomic Energy Commission's Regulatory staff requested those utilities that operate nuclear power plants, have applied for operating licenses, or have plants whose construction permit review was essentially complete, to assess the effects and consequences of a postulated rupture of piping containing high-energy fluids and located outside of the containment structure. These requests were issued by Mr. A. Giambusso, Deputy Director for Reactor Projects, Directorate of Licensing, in letters, most of which were dated in December 1972.

Because these plants were either in operation or in advanced stages of engineering design and construction, the request included guidance for corrective modifications that could be implemented by in-situ measures. Such modifications included relocation or rerouting of piping, installation of impingement barriers and encapsulation sleeves around high stressed piping regions, provisions for venting of compartments subject to pressurization, addition of piping restraints, and strengthening of structural components of buildings.

From our review of responses submitted to the Regulatory staff, and from discussions with architect-engineering firms, we have learned that some of these organizations have inferred that the criteria contained in Mr. A. Giambusso's letter pertaining to corrective modifications for plants in advanced stages of construction and operation are applicable for the design of high-energy fluid systems outside the containment in new designs of nuclear power plants. It was not our intent that the criteria for corrective plant modifications be applied to new power plants that are in the initial design stages. We believe that a more direct approach, involving a rearrangement of the physical plant layout with a view to relocation of essential safety systems and components is appropriate for the new plants.

For the present, pending issuance of a planned AEC Regulatory Guide - "Protection Against Postulated Events and Accidents Outside Containment," an acceptable implementation of Criterion 4 of the Commission's General Design Criteria listed in Appendix A of 10 CFR Part 50, as applied to new plants with respect to the design of structures, systems and components important to safety and located outside of containment is as follows:

I. <u>PIPING SYSTEMS CONTAINING HIGH-ENERGY FLUIDS* DURING NORMAL REACTOR OPERATION</u>

(a) The piping systems are isolated by adequate physical separation and remotely located from safety systems and components that are required to shut down the reactor safely and maintain the plant in a cold shutdown condition.

*Refer to Appendix A for identification of high-energy fluid systems.

NUREG-75/087

U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

SECTION 3.7.2 SEISMIC SYSTEM ANALYSIS

REVIEW RESPONSIBILITIES

Primary - Structural Engineering Branch (SEB)

Secondary - None

I. AREAS OF REVIEW

The following areas related to the seismic system analysis described in the applicant's safety analysis report (SAR) are reviewed.

1. Seismic Analysis Methods

For all Category I structures, systems, and components, the applicable seismic analysis methods (response spectra, time history, equivalent static load) are reviewed. The manner in which the dynamic system analysis method is performed, including the modeling of foundation torsion, rocking and translation, is reviewed. The method chosen for selection of significant modes and an adequate number of masses or degrees of freedom is reviewed. The manner in which consideration is given in the seismic dynamic analysis to maximum relative displacements between supports is reviewed. In addition, other significant effects that are accounted for in the dynamic seismic analysis such as hydrodynamic effects and nonlinear response are reviewed. If tests or empirical methods are used in lieu of analysis for any Category I structure, the testing procedure, load levels, and acceptance basis are also reviewed.

2. Natural Frequencies and Response Loads

For the operating license review, significant natural frequencies and response loads for major Category I structures are reviewed. In addition, the response spectra at critical major Category I equipment elevations and points of support are reviewed.

3. Procedures Used for Analytical Modeling

The criteria and procedures used in modeling for the seismic system analyses are reviewed. The criteria and bases for determining whether a component or structure is analyzed as part of a system analysis or independently as a subsystem are also reviewed.

4. Soil-Structure Interaction

The design earthquake input is defined in the "free field," i.e., the effect of the presence of structures is not included. When plants are founded on soil deposits or soft media, the resulting motions of the base slab will differ from those defined at the same elevation in the free field, due to deformability of the foundation. This difference in the base slab motion and the free field motion is known as soil-structure interaction effect.

As applicable, the methods of soil-structure interaction analysis used in the seismic system analysis and their bases are reviewed. The factors to be considered in accepting the validity of a particular method are: (1) the extent of embedment, (2) the depth of soil over rock, and (3) the layering of the soil strata. If the finite element approach is used, the criteria for determining the location of the bottom boundary and side boundary are reviewed. The procedures by which strain-dependent soil properties (damping, shear modulus) are incorporated in the analysis are also reviewed.

If lumped spring methods are used, the parameters used in the analysis are reviewed. Also, the procedures by which strain-dependent soil properties (damping, shear modulus), layering, and variation of soil properties are incorporated in the analysis are reviewed. The applicability of a lumped spring method used for the particular site conditions is reviewed.

Any other methods used for soil-structure interaction analysis are also reviewed as is any basis for not using soil-structure interaction analysis. The procedures used to account for effects of adjacent structures on structural response in the soil-structure interaction analysis are reviewed.

5. Development of Floor Response Spectra

The procedures for developing floor response spectra considering the three components of earthquake motion are reviewed. If a modal response spectrum method of analysis is used to develop floor response spectra, the justification for its conservatism and equivalency to a time history method is evaluated.

6. Three Components of Earthquake Motion

The procedures by which the three components of earthquake motion are considered in determining the seismic response of structures, systems, and components are reviewed.

7. Combination of Modal Responses

When a response spectrum approach is used for calculating the seismic response of structures, systems, or components, the phase relationship between various modes is lost. Only the maximum response for each mode can be determined. The maximum responses for modes do not in general occur at the same time and these responses have to be combined according to some procedure selected to approximate or bound the response of the system. When a response spectra method is used, the description of the procedure for combining modal responses (shears, moments, stresses, deflections, and accelerations) is reviewed, including that for modes with closely spaced frequencies.

8. Interaction of Non-Category I Structures with Category I Structures

The design criteria to account for the seismic motion of non-Category I structures or portions thereof in the seismic design of Category I structures or portions thereof are reviewed. The procedures that are used to protect Category I structures from the structural failure of non-Category I structures, due to seismic effects, are reviewed.

9. Effects of Parameter Variations on Floor Responses

The procedures that are used to consider the effects of the expected variations of structural properties, dampings, soil properties, and soil-structure interaction on the floor response spectra and time histories are reviewed.

10. Use of Constant Vertical Static Factors

Where applicable, justification for the use of constant static factors as vertical response loads for designing Category I structures, systems, and components in lieu of the use of a vertical seismic system dynamic analysis is reviewed.

11. Methods Used to Account for Torsional Effects

The method employed to consider torsional effects in the seismic analysis of Category I structures is reviewed. The review includes the evaluation of the conservatism of any approximate methods to account for torsional accelerations in the seismic design of Category I structures.

12. Comparison of Responses

For the operating license review, where applicable, the comparison of seismic responses for major Category I structures using modal response spectrum and time history approaches is evaluated.

13. Methods for Seismic Analysis of Category I Dams

The analytical methods and procedures that will be used for seismic analysis of Category I dams are reviewed. The assumptions made, the boundary conditions used, and the procedures by which strain-dependent soil properties are incorporated in the analysis are reviewed.

14. Determination of Category I Structure Overturning Moments

The description of the dynamic methods and procedures used to determine design overturning moments for Category I structures are reviewed.

15. Analysis Procedure for Damping

The analysis procedure to account for the damping in different elements of the model of a coupled system is reviewed.

II. ACCEPTANCE CRITERIA

The acceptance criteria for the areas of review described in Section I of this plan are given below. Other approaches which can be justified to be equivalent to or more conservative than the stated acceptance criteria may be used.

1. Seismic Analysis Methods

The seismic analysis of all Category I structures, systems, and components should utilize either a suitable dynamic analysis method or an equivalent static load method, if justified.

a. Dynamic Analysis Method

A dynamic analysis (e.g., response spectrum method, time history method, etc.) should be used when the use of the equivalent static load method cannot be justified. To be acceptable such analyses should consider the following items:

(1) Use of either the time history method or the response spectrum method.

(2) Use of appropriate methods of analysis to account for effects of soil-structure interaction.

(3) Consideration of the torsional, rocking, and translational responses of the structures and their foundations.

(4) Use of an adequate number of masses or degrees of freedom in dynamic modeling to determine the response of all Category I and applicable non-Category I structures and plant equipment. The number is considered adequate when additional degrees of freedom do not result in more than a 10% increase in responses. Alternately, the number of degrees of freedom may be taken equal to twice the number of modes with frequencies less than 33 cps.

(5) Investigation of a sufficient number of modes to assure participation of all significant modes. The criterion for sufficiency is that the inclusion of additional modes does not result in more than a 10% increase in responses.

(6) Consideration of maximum relative displacements among supports of Category I structures, systems, and components.

(7) Inclusion of significant effects such as piping interactions, externally applied structural restraints, hydrodynamic (both mass and stiffness effects) loads, and nonlinear responses.

b. Equivalent Static Load Method

An equivalent static load method is acceptable if:

(1) Justification is provided that the system can be realistically represented by a simple model and the method produces conservative results in terms of responses. Typical examples or published results for similar structures may be submitted in support of the use of the simplified method.

(2) The design and associated simplified analysis account for the relative motion between all points of support.

(3) To obtain an equivalent static load of a structure, equipment, or component which can be represented by a simple model, a factor of 1.5 is applied to the peak acceleration of the applicable floor response spectrum. A factor of less than 1.5 may be used if adequate justification is provided.

In addition, for equipment which can be modeled adequately as a one-degree-of-freedom system, the use of a static load equivalent to the peak of the floor response spectra is acceptable. For piping supported at only two points, the use of a static load equivalent to the peak of the floor response spectra is also acceptable.

2. Natural Frequencies and Response Loads

To be acceptable for the operating license review, the following information should be provided.

a. A summary of natural frequencies, response loads, mode shapes, and modal responses for a representative number of major Category I structures, including the containment building.

b. A time history of acceleration (or equivalent parameters) or response spectrum at the major plant equipment elevations and points of support.

3. Procedures Used for Analytical Modeling

A nuclear power plant facility consists of very complex structural systems. To be acceptable, the stiffness, mass, and damping characteristics of the structural systems should be adequately incorporated into the analytical models. Specifically, the following items should be considered in analytical modeling:

a. Designation of Systems Versus Subsystems

Major Category I structures that are considered in conjunction with foundation media in forming a soil-structure interaction model are defined as "seismic systems." Other Category I structures, systems, and components that are not designated as "seismic systems" should be considered as "seismic subsystems."

b. Decoupling Criteria for Subsystems

It can be shown, in general, that the absolute frequencies of systems and subsystems have negligible effect on the error due to decoupling. It can be shown that the mass ratio, R_m, and the frequency ratio, R_f, govern the results where R_m and R_f are defined as:

$$R_m = \frac{\text{Total mass of the supported subsystem}}{\text{Mass that supports the subsystem}}$$

$$R_f = \frac{\text{Fundamental frequency of the supported subsystem}}{\text{Frequency of the dominant support motion}}$$

The following criteria are acceptable:

(1) If $R_m < 0.01$, decoupling can be done for any R_f.

(2) If $0.01 \leq R_m \leq 0.1$, decoupling can be done if $0.8 \geq R_f \geq 1.25$.

(3) If $R_m > 0.1$, an approximate model of the subsystem should be included in the primary system model.

If the subsystem is comparatively rigid and also is rigidly connected to the primary system, it is sufficient to include only the mass of the subsystem at the support point in the primary system model. On the other hand, in case of a subsystem supported by very flexible connections, e.g., pipe supported by hangers, the subsystem need not be included in the primary model. In most cases the equipment and components, which come under the definition of subsystems, are analyzed (or tested) as a decoupled system from the primary structure and the seismic input for the former is obtained by the analysis of the latter. One important exception to this procedure is the reactor coolant system, which is considered a subsystem but is usually analyzed using a coupled model of the reactor coolant system and primary structure.

c. Lumped Mass Considerations

The acceptance criteria given under II.1.a(4) are applicable.

d. Modeling for Three Component Input Motion

In general, three-dimensional models should be used for seismic analyses. However, simpler models can be used if justification can be provided that the coupling effects of those degrees of freedom that are omitted from the three-dimensional models are not significant.

4. Soil-Structure Interaction

Table 3.7.2-1 summarizes acceptable procedures for soil-structure interaction analyses. To be acceptable, a finite element technique (Ref. 1, 11) or equivalent is required as the analytical tool for soil-structure interaction analysis for all Category I structures where the foundations are deeply embedded in soil. This technique may also be used for all cases where soil-structure interaction is involved. For structures supported on rock, a fixed base approach is acceptable. For shallowly embedded structures on shallow soil overburden over rock, or layered soil with varying soil properties, either the finite element approach or multiple mass-spring (shear beam) approach (Ref. 2) may be used.

Table 3.7.2-1

Acceptable Methods for Soil-Structure Interaction Analysis

Method of Soil-Structure Interaction Analysis	Rock†† Foundation	Deeply Embedded Case†	Soil Foundation** Shallowly Embedded Case: Deep Soil Found. w/Uniform Properties	Soil Foundation** Shallowly Embedded Case: Deep Soil Found. w/Layered Properties	Soil Foundation** Shallowly Embedded Case: Shallow Soil Foundation
Single Lumped Mass-Spring Approach	x		x		
Multiple Lumped Mass-Spring Approach*	x		x	x	x
Finite Element Approach*	x	x	x	x	x

*Or equivalent.
†Deep embedment: actual embedded depth >15% of the least base width or other appropriate value to be justified.
††A medium for which the soil-structure interaction effect is negligible or alternatively, a medium with a shear wave velocity greater than or equal to 3500 fps.
**Soil foundation means the depth of soil between the bottom of the foundation slab and the base rock.

The lumped mass-spring method or "compliance function method" may be used for cases where the depth of embedment is shallow and the soil foundation is relatively uniform and of sufficient depth that it can be considered as a half-space.

The acceptability of the procedures used to account for the effects of adjacent structures on structural responses in soil-structure interaction analyses will be reviewed on a case-by-case basis.

Other techniques which give an equivalent degree of conservatism as the appropriate acceptable technique and which are justified are also acceptable. Since the finite element and the lumped mass spring approaches are the most commonly used in current practice, these two approaches are discussed below.

Finite Element Approach

The finite element approach may be used for all cases where soil-structure interaction is involved. The acceptance criteria for different aspects of the finite element technique are as follows:

a. Boundary Conditions

(1) Bottom Boundary

Wherever possible the base of the model is placed at the rock level. However, if the bedrock is too deep, the bottom boundary can also be placed at a

reasonable depth from the structure foundation such that the effect of soil-structure interaction below this depth is negligible. This should be justified. The nodes on the base of the finite element model are fixed and the earthquake input motion is applied there.

(2) Side Boundaries

The side boundaries should be kept at a distance away from the structures such that the motion at the boundary is not affected by the structural motion. It is acceptable if the distance of the boundaries from the edge of the structure is kept equal to or greater than three times the base slab dimension. Any other selection of the side boundaries should be justified.

b. Soil Properties

In a finite element model, different kinds of soils present should be adequately represented. Since the soil moduli and damping ratios are in general highly strain-dependent, the strain-compatible properties for each layer or element should be computed with the use of soil property curves which relate the moduli and damping ratios with strain for the soils present at the site.

Lumped Mass-Spring Approach

In the lumped mass-spring approach the compliance functions in common use are based upon the analytical solution of a rigid plate on an elastic half-space. Reference 8 gives an acceptable set of spring and damping constants for a plate supported on an elastic half-space. Compliance functions for layered sites have also been developed but their applicability for soil-structure interaction analysis for layered sites should be justified by the applicant.

As mentioned earlier, the lumped mass-spring method may be used for cases where the depth of embedment is shallow and the soil foundation is relatively uniform and of sufficient depth that it can be considered as a half-space. The justification for the sufficiency of the depth should be provided by the applicant and will be reviewed by the staff on a case-by-case basis. This approach should not be used for other cases for the following reasons:

(1) It is well known that the lumped soil spring parameters are frequency dependent. This dependence for a layered space is expected to be large. Thus, using a constant set of soil spring parameters for all situations may lead to incorrect results.

(2) The lumped parameters are usually derived without any regard to the actual embedment of the structure. Thus, for all structures, whether sitting on the ground surface or embedded to any extent, the answers obtained using lumped parameters will be the same. It is known that the earthquake motion is not the same at all elevations of the soil profile, hence the motion at the foundation and surface will, in general, be different. Effects of this

variation in the earthquake motion itself and also the inertia effort of the soil cannot be taken care of adequately by the lumped parameter method.

(3) In the lumped parameter technique the entire soil medium is assumed to be homogeneous and elastic. However, in any stratum of soil deposits there are generally many different types of soils present. The properties also are strain-dependent. These factors could have a significant effect on the structural response, which cannot be accounted for by a single set of stiffness and damping values.

(4) As stated earlier, if the site situation can be approximated as an elastic half-space, the design earthquake can be directly input in this approach. This is a relative advantage of the method. However, as Whitman (Ref. 8) points out, for a structure sitting on a stratum whose thickness is less than twice the width of the foundation, the effects of soil amplification and soil-structure interaction cannot be separated. So, he notes, the input motion at the spring support in such cases should be the rock motion, <u>not</u> the design motion specified at the foundation level. It is, therefore, obvious that in cases of shallow overburden, an uncertainty about the input motion exists.

Thus, the actual site conditions for a particular plant should be carefully reviewed before accepting the lumped spring approach.

5. <u>Development of Floor Response Spectra</u>

To be acceptable, the floor response spectra should be developed taking into consideration the three components of the earthquake motion. The individual floor response spectral values for each frequency are obtained for one vertical and two mutually perpendicular horizontal earthquake motions and are combined according to the "square root of the sum of the squares" method to predict the total floor response spectrum for that particular frequency.

In general, development of the floor response spectra is acceptable if a time history approach is used. If a modal response spectra method of analysis is used to develop the floor response spectra, the justification for its conservatism and equivalency to that of a time history method must be demonstrated by representative examples.

6. <u>Three Components of Earthquake Motion</u>

Depending upon what basic methods are used in the seismic analysis, i.e., response spectra or time history method, the following two approaches are considered acceptable for the combination of three-dimensional earthquake effects. (Ref. 3, 4, and 5.)

a. <u>Response Spectra Method</u>

When the response spectra method is adopted for seismic analysis, the maximum structural responses due to each of the three components of earthquake motion

should be combined by taking the square root of the sum of the squares of the maximum codirectional responses caused by each of the three components of earthquake motion at a particular point of the structure or of the mathematical model.

b. Time History Analysis Method

When the time history analysis method is employed for seismic analysis, two types of analysis are generally performed depending on the complexity of the problem. (1) To obtain maximum responses due to each of the three components of the earthquake motion: in this case the method for combining the three-dimensional effects is identical to that described in (a) except that the maximum responses are calculated using the time history method instead of the spectrum method. (2) To obtain time history responses from each of the three components of the earthquake motion and combine them at each time step algebraically: the maximum response in this case can be obtained from the combined time solution. When this method is used, to be acceptable, the earthquake motions specified in the three different directions should be statistically independent.

7. Combination of Modal Responses

When the response spectrum method of analysis is used to determine the dynamic response of damped linear systems, the most probable response is obtained as the square root of the sum of the squares of the responses from individual modes. Thus, the most probable system response, R, is given by

$$R = \left(\sum_{k=1}^{N} R_k^2\right)^{1/2} \qquad (1)$$

where R_k is the response for the kth mode and N is the number of significant modes considered in the modal response combination.

When modes with closely spaced modal frequencies exist, an acceptable method for obtaining the system response is to take the absolute sum of the responses of the closely spaced modes and combine this sum with other remaining modal responses using the square root of the sum of the squares rule. Two modes having frequencies within 10% of each other are considered as modes with closely spaced frequencies.

This approach is simple and straightforward in all those cases where the group of modes with closely spaced frequencies is tightly bundled, i.e., the lowest and the highest modes of the group are within 10% of each other. However, when the group of closely spaced modes is spaced widely over the frequency range of interest (while the frequencies of the adjacent modes are closely spaced), the absolute sum method of combining responses tends to yield over-conservative results. To obviate this problem, a general approach applicable to all modes is considered appropriate. The following equation is merely a mathematical representation of this approach.

The most probable system response, R, is given by

$$R = \left(\sum_{K=1}^{N} R_k^2 + 2 \sum |R_\ell R_m|\right)^{1/2} \qquad (2)$$

where the second summation is to be done on all ℓ and m modes whose frequencies are closely spaced to each other.

Other approaches which give an equivalent degree of conservatism to the above methods, and which are adequately justified are also acceptable.

8. Interaction of Non-Category I Structures with Category I Structures

To be acceptable, the interfaces between Category I and non-Category I structures and plant equipment must be designed for the dynamic loads and displacements produced by both the Category I and non-Category I structures and plant equipment. In addition, a statement indicating the fact that all non-Category I structures meet any one of the following requirements should be provided.

a. The collapse of any non-Category I structure will not cause the non-Category I structure to strike a seismic Category I structure or component.

b. The collapse of any non-Category I structure will not impair the integrity of seismic Category I structures or components.

c. The non-Category I structures will be analyzed and designed to prevent their failure under SSE conditions in a manner such that the margin of safety of these structures is equivalent to that of Category I structures.

9. Effects of Parameter Variations on Floor Response Spectra

To be acceptable, consideration should be given in the analyses to the effects on floor response spectra (e.g., peak width and period coordinates) of expected variations of structural properties, dampings, soil properties, and soil-structure interactions. An acceptable method for determining the amount of peak widening associated with the structural frequency is described below.

Let f_j be the jth mode structural frequency which is determined from the structure model. The variation in the structural frequency is determined by evaluating the individual frequency variation due to the variation in each parameter that has significant effect, such as the soil shear modulus, damping, and material density. The total frequency variation, Δf_j, is then determined by taking the square root of the sum of squares of a minimum variation of $0.05f_j$ and the individual frequency variation $(\Delta f_j)_n$, that is,

$$\Delta f_j = \pm \sqrt{(0.05f_j)^2 + \Sigma(\Delta f_j)_n^2}$$

A value of $0.10\ f_j$ is used if the actual computed value of Δf_j is less than $0.10f_j$.

If the above procedure is not used, the peak width should be increased by a minimum of $\pm$ 15% to be acceptable.

Time histories of floor motion may be used as excitations to the subsystems. To account for the effect of possible frequency variation of the structure, the same time history data can be used with at least three different time intervals: Δt and $(1 \pm \Delta f_j/f_j)\ \Delta t$, for the analysis of equipment, where f_j is the dominant structural frequency and Δf_j is a parameter defining the frequency variation due to uncertainties as given above. This variation of the time interval has a similar effect to widening the spectral peak when generating the smoothed response spectrum. If one of the equipment frequencies, f_e, is known to be within the range $f_j \pm \Delta f_j$, the time history can also be used with a time interval of $(1 - (f_e - f_j)/f_j)\ \Delta t$. This method of modifying the time history data is described in Reference 12. As in the case of the broadened response spectrum, the variation of time interval has little effect on those equipment response modes with frequencies outside the range of the broadened peak of the corresponding spectrum.

10. Use of Constant Vertical Static Factors

The use of constant vertical load factors as vertical response loads for the seismic design of all Category I structures, systems, and components in lieu of the use of a vertical seismic system dynamic analysis is acceptable only if it can be justified that the structure is rigid in the vertical direction. The criterion for rigidity is that the lowest frequency in the vertical direction is more than 33 cps.

11. Methods Used to Account for Torsional Effects

An acceptable method of treating the torsional effects in the seismic analysis of Category I structures is to carry out a dynamic analysis which incorporates the torsional degrees of freedom. An acceptable alternative, if properly justified, is the use of static factors to account for torsional accelerations in the seismic design of Category I structures in lieu of the use of a combined vertical, horizontal and torsional system dynamic analysis.

12. Comparison of Responses

The responses obtained from both modal analysis response spectrum and time history methods at selected points in typical Category I structures should be compared to demonstrate approximate equivalency between the two methods.

13. Methods for Seismic Analysis of Category I Dams

For the analysis of all Category I dams, a finite element approach which takes into consideration the time history of forces (due to both horizontal and vertical earthquake loadings), the behavior of the soil under simulated earthquake loadings, and an evaluation of deformations should be used. Appropriate nonlinear stress-strain relations for the soil are to be used in the finite element analysis. For earth-filled dams, procedures presented in References 6 and 7 are acceptable. For rock-filled dams, the analytical procedure used will be reviewed on a case-by-case basis.

14. Determination of Category I Structure Overturning Moments

To be acceptable, the determination of the design moment for overturning should incorporate the following items:

a. Three components of input motion.

b. Conservative consideration of vertical and lateral seismic forces.

15. Analysis Procedure for Damping

Either the composite modal damping approach or the modal synthesis technique can be used to account for element-associated damping.

For the composite modal damping approach, two techniques of determining an equivalent modal damping matrix or composite damping matrix are commonly used. They are based on the use of the mass or stiffness as a weighting function in generating the composite modal damping. The formulations lead to:

$$\bar{\beta}_j = \{\phi\}^T [\bar{M}] \{\phi\} \tag{3}$$

$$\bar{\beta}_j = \frac{\{\phi\}^T [\bar{K}] \{\phi\}}{\{\phi\}^T [K] \{\phi\}} \tag{4}$$

where

$[K]$ = assembled stiffness matrix,

$\bar{\beta}_j$ = equivalent modal damping ratio of the jth mode,

$[\bar{K}]$, $[\bar{M}]$ = the modified stiffness or mass matrix constructed from element matrices formed by the product of the damping ratio for the element and its stiffness or mass matrix, and

$\{\phi\}$ = j^{th} normalized modal vector.

For models that take the soil-structure interaction into account by the lumped soil spring approach, the method defined by equation (4) is acceptable. For fixed base models, either equation (3) or (4) may be used. Other techniques based on modal synthesis (Ref. 9) have been developed and are particularly useful when more detailed data on the damping characteristics of structural subsystems are available. The modal synthesis analysis procedure consists of (1) extraction of sufficient modes from the structure model, (2) extraction of sufficient modes from the finite element soil model, and (3) performance of a coupled analysis using the modal synthesis technique, which uses the data obtained in steps (1) and (2) with appropriate damping ratios for structure and soil subsystems. This method is based upon satisfaction of

displacement compatibility and force equilibrium at the system interfaces and utilizes subsystem eigenvectors as internal generalized coordinates. This method results in a nonproportional damping matrix for the composite structure and equations of motion have to be solved by direct integration or by uncoupling them by use of complex eigenvectors.

Another technique which is also considered acceptable for estimating the equivalent modal damping of a soil-structure interaction model is given by Tsai (Ref. 10).

II. REVIEW PROCEDURES

For each area of review, the following procedure is implemented. The reviewer will select and emphasize material from the procedures given below, as may be appropriate for a particular case.

1. Seismic Analysis Methods

   For all Category I structures, systems, and components, the applicable methods of seismic analysis (response spectra, time history, equivalent static load) are reviewed to ascertain that the techniques employed are in accordance with the acceptance criteria as given in Section 2.1 of this plan. If empirical methods or test are used in lieu of analysis for any Category I structure, these are evaluated to determine whether or not the assumptions are conservative, and whether the test procedure adequately models the seismic response.

2. Natural Frequencies and Response Loads

   For the operating license review, the summary of natural frequencies and response loads is reviewed for compliance with the acceptance criteria in Section II.2 of this plan.

3. Procedures Used for Analytical Modeling

   The procedures used for modeling for seismic system analyses are reviewed to determine whether the three-dimensional characteristics of structures are properly modeled in accordance with the acceptance criteria of Section II.3 and all significant degrees of freedom have been incorporated in the models. The criteria for decoupling of a structure, equipment, or component and analyzing it separately as a subsystem are reviewed for conformance with the acceptance criteria given in Section II.3.

4. Soil-Structure Interaction

   The methods of soil-structure interaction analysis used are examined to determine that the techniques employed are in accordance with the acceptance criteria as given in Section II.4. Typical mathematical models for soil-structure interaction analysis are reviewed to ensure the adequacy of the representation in accordance with Section II.4 of this plan. In addition, the methods used to assess the effects of adjacent structures on structural response in soil-structure interaction analysis are reviewed to establish their acceptability.

5. Development of Floor Response Spectra

   Procedures for developing the floor response spectra are reviewed to verify that they are in accordance with the acceptance criteria specified in Section II.5. If a modal

response spectrum method of analysis is used to develop the floor response spectra, its conservatism compared to that of a time history approach is reviewed. The applicant is requested to provide additional technical justification for any procedure considered not adequately justified.

6. Three Components of Earthquake Motion

The procedures by which the three components of earthquake motion are considered in determining the seismic response of structures, systems, and components are reviewed to determine compliance with the acceptance criteria of Section II.6. Any other procedures that are considered not adequately justified are so identified, and the applicant is asked to provide additional justification.

7. Combination of Modal Responses

The procedures for combining modal responses (shears, moments, stresses, deflections, and accelerations) for closely spaced modes are reviewed to determine compliance with the acceptance critieria of Section II.7, when a response spectrum modal analysis method is used.

8. Interaction of Non-Category I Structures with Category I Structures

The design and analysis criteria for interaction of non-Category I structures with Category I structures are reviewed to ensure compliance with the acceptance criteria of Section II.8.

9. Effects of Parameter Variations on Floor Response Spectra

The seismic system analysis is reviewed to determine whether the analysis considered the effects of expected variations of structural properties, dampings, soil properties, and soil-structure interaction on floor responses spectra (e.g., peak width and period coordinates) and to determine compliance with the acceptance criteria of Section II.9.

10. Use of Constant Vertical Static Factors

Use of constant static factors as response loads in the vertical direction for the seismic design of any Category I structure, system or component in lieu of a detailed dynamic method is reviewed to determine that constant vertical static factors are used only if the structure is rigid in the vertical direction.

11. Methods Used to Account for Torsional Effects

The methods of seismic analysis are reviewed to determine that the torsional effects of vibration are incorporated by including the torsional degrees of freedom in the dynamic model. Justification provided by the applicant for the use of any approximate method to account for torsional effects is judged to assure that it results in a conservative design. If such justification is deemed inadequate, it is identified and the applicant requested to provide additional justification.

12. Comparison of Responses

Where applicable, the responses obtained from both response spectrum and time history methods at selected points in major Category I structures are compared to judge the

accuracy of the analyses conducted. Large differences in the results obtained by use of the two methods are identified and the applicant is asked to discuss the reasons for the large differences.

13. Methods for Seismic Analysis of Category I Dams
Methods for the seismic analysis of Category I dams are reviewed to determine compliance with the acceptance criteria of Section II.13.

14. Determination of Category I Structure Overturning Moments
Methods to determine Category I structure overturning moments are reviewed to determine compliance with the acceptance criteria of Section II.14.

15. Analysis Procedure for Damping
The analysis procedure to account for damping in different elements of the model of a coupled system is reviewed to determine that it is in accordance with the acceptance criteria of Section II.15. It is verified that composite damping based on mass weighting is not used for sites where the lumped mass spring approach is used to model the soil-structure interaction.

Any matters identified during the review of the SAR where additional information or justification are needed are included in the "Additional Technical Information Request" prepared by SEB for transmittal to the Division of Reactor Licensing. Such requests not only identify any portions of the seismic system analysis considered unacceptable without further justification, but also specify the changes that should be made in the SAR to meet the acceptance criteria. Subsequent amendments of the SAR received in response to these SEB requests are reviewed for conformance with the staff positions.

IV. EVALUATION FINDINGS
(Combined for Sections 3.7.2 and 3.7.3)
The reviewer verifies that sufficient information has been provided and that his evaluation is sufficiently complete and adequate to support conclusions of the following type, to be included in the staff's safety evaluation report:

"The scope of review of the seismic system and subsystem analysis for the (_____) plant included the seismic analysis methods for all Category I structures, systems, and components. It included review of procedures for modeling, seismic soil-structure interaction, development of floor response spectra, inclusion of torsional effects, seismic analysis of Category I dams, evaluation of Category I structure overturning, and determination of composite damping. The review has included design criteria and procedures for evaluation of the interaction of non-Category I structures and piping with Category I structures and piping and the effects of parameter variations on floor response spectra. The review has also included criteria and seismic analysis procedures for reactor internals and Category I buried piping outside containment.

"The system and subsystem analyses are performed by the applicant on an elastic basis. Modal response spectrum multidegree of freedom and time history methods form the bases

for the analyses of all major Category I structures, systems, and components. When the modal response spectrum method is used, governing response parameters are combined by the square root of the sum of the squares rule. However, the absolute sum of the modal responses are used for modes with closely spaced frequencies. The square root of the sum of the squares of the maximum codirectional responses is used in accounting for three components of the earthquake motion for both the time history and response spectrum methods. Floor spectra inputs to be used for design and test verifications of structures, systems, and components are generated from the time history method, taking into account variation of parameters by peak widening. A vertical seismic system dynamic analysis will be employed for all structures, systems, and components where analyses show significant structural amplification in the vertical direction. Torsional effects and stability against overturning are considered.

"The finite element (or the lumped soil spring) approach is used to evaluate soil-structure interaction and structure-to-structure interaction effects upon seismic responses. For the finite element analysis, appropriate nonlinear stress-strain and damping relationships for the soil are considered in the analysis.

"For the analysis of Category I dams, a finite element approach which takes into consideration the time history of forces, the behavior and deformation of the dam due to the earthquake, and applicable stress-strain relations is used.

"We conclude that the seismic system and subsystem analysis procedures and criteria proposed by the applicant provide an acceptable basis for the seismic design."

V. REFERENCES

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U.S. NUCLEAR REGULATORY COMMISSION
STANDARD REVIEW PLAN
OFFICE OF NUCLEAR REACTOR REGULATION

SECTION 3.7.3 SEISMIC SUBSYSTEM ANALYSIS

REVIEW RESPONSIBILITIES

Primary - Structural Engineering Branch (SEB)

Secondary - None

I. AREAS OF REVIEW

The following areas related to the seismic subsystem analysis are reviewed:

1. Seismic Analysis Methods

   The information reviewed is similar to that described in Section I.1 of Standard Review Plan (SRP) 3.7.2, but as applied to seismic Category I subsystems.

2. Determination of Number of Earthquake Cycles

   Criteria or procedures used to establish the number of earthquake cycles during one seismic event and the maximum number of cycles for which applicable Category I subsystems and components are designed are reviewed.

3. Procedures Used for Analytical Modeling

   The criteria and procedures used for modeling for the seismic subsystem analysis are reviewed.

4. Basis for Selection of Frequencies

   As applicable, criteria or procedures used to separate fundamental frequencies of components and equipment from the forcing frequencies of the support structure are reviewed.

5. Use of Equivalent Static Load Method of Analysis

   The basis for the use of the equivalent static load method of analysis and the procedures used for determining the equivalent static loads are reviewed.

6. Three Components of Earthquake Motion

   The information reviewed is similar to that described in Section I.6 of SRP 3.7.2, but as applied to Category I subsystems.

7. Combination of Model Responses

   The information reviewed is similar to that described in Section I.7 of SRP 3.7.2, but as applied to Category I subsystems.

8. Analytical Procedures for Piping Systems

The analytical procedures applicable to seismic analysis of piping systems, including methods used to consider differential piping support movements at different support points located within a structure and between structures, are reviewed.

9. Multiply-Supported Equipment and Components with Distinct Inputs

The criteria and procedures for seismic analysis of equipment and components supported at different elevations within a building and between buildings with distinct inputs are reviewed.

10. Use of Constant Vertical Static Factors

The information reviewed is similar to that described in Section I.10 of SRP 3.7.2, but as applied to Category I subsystems.

11. Torsional Effects of Eccentric Masses

The criteria and procedures that are used to consider the torsional effects of eccentric masses (e.g., valve operators) in seismic subsystem analyses are reviewed.

12. Category I Buried Piping Systems and Tunnels

For Category I buried piping and tunnels, the seismic criteria and methods which consider the compliance of soil media, settlement due to earthquake, and differential movements at support points, penetrations, and entry points into other structures provided with anchors are reviewed.

13. Interaction of Other Piping With Category I Piping

The seismic analysis procedures to account for the seismic motion of non-Category I piping systems in the seismic design of Category I piping are reviewed.

14. Seismic Analyses for Reactor Internals

The seismic subsystem analyses that are utilized in establishing seismic design adequacy of the reactor internals including fuel elements, control rod assemblies, and control rod drive mechanisms are reviewed. The information reviewed includes the following:

a. Typical diagrams of mathematical dynamic modeling of reactor internal structures.

b. Damping values and their justification.

c. A description of the methods and procedures that will be used to compute seismic responses.

d. For the operating license review, a summary of the results of the dynamic seismic analysis.

15. Analysis Procedure for Damping

The information reviewed is similar to that described in Section I.15 of SRP 3.7.2, but as applied to Category I subsystems.

II. ACCEPTANCE CRITERIA

The acceptance criteria for the areas of review described in Section I of this plan are given below. Other approaches which can be justified to be equivalent to or more conservative than the stated acceptance criteria may be used.

1. Seismic Analysis Methods

The acceptance criteria provided in SRP 3.7.2, Section II.1, are applicable.

2. Determination of Number of Earthquake Cycles

During the plant life at least one safe shutdown earthquake (SSE) and five operating basis earthquakes (OBE) should be assumed. The number of cycles per earthquake should be obtained from the synthetic time history (with a minimum duration of 10 seconds) used for the system analysis, or a minimum of 10 maximum stress cycles per earthquake may be assumed.

3. Procedures Used for Analytical Modeling

The acceptance criteria provided in SRP 3.7.2, Section II.3, are applicable.

4. Basis for Selection of Frequencies

To avoid resonance, the fundamental frequencies of components and equipment should preferably be selected to be less than 1/2 or more than twice the dominant frequencies of the support structure. Use of equipment frequencies within this range is acceptable if the equipment is adequately designed for the applicable loads.

5. Use of Equivalent Static Load Method of Analysis

The acceptance criteria provided in SRP 3.7.2, Section II.1, are applicable.

6. Three Components of Earthquake Motion

The acceptance criteria provided in SRP 3.7.2, Section II.6, are applicable.

7. Combination of Modal Responses

The acceptance criteria provided in SRP 3.7.2, Section II.7, are applicable.

8. Analytical Procedures for Piping Systems

The seismic analysis of Category I piping may use either a dynamic analysis or an equivalent static load method. The acceptance criteria for the dynamic analysis or equivalent static load methods are as given in SRP 3.7.2, Section II.1.

9. Multiply-Supported Equipment and Components With Distinct Inputs

Equipment and components in some cases are supported at several points by either a single structure or two separate structures. The motions of the primary structure or structures at each of the support points may be quite different.

A conservative and acceptable approach for equipment items supported at two or more locations is to use an upper bound envelope of all the individual response spectra for these locations to calculate maximum inertial responses of multiply-supported items. In addition, the relative displacements at the support points should be considered. Conventional static analysis procedures are acceptable for this purpose. The maximum relative support displacements can be obtained from the structural response calculations or, as a conservative approximation, by using the floor response spectra. For the latter option, the maximum displacement of each support is predicted by $S_d = S_a g/\omega^2$, where S_a is the spectral acceleration in "g's" at the high frequency end of the spectrum curve (which, in turn, is equal to the maximum floor acceleration), g is the gravity constant, and ω is the fundamental frequency of the primary support structure in radians per second. The support displacements can then be imposed on the supported item in the most unfavorable combination. The responses due to the inertia effect and relative displacements should be combined by the absolute sum method.

In the case of multiple supports located in a single structure, an alternate acceptable method using the floor response spectra involves determination of dynamic responses due to the worst single floor response spectrum selected from a set of floor response spectra obtained at various floors and applied identically to all the floors, provided there is no significant shift in frequencies of the spectra peaks. In addition, the support displacements should be imposed on the supported item in the most unfavorable combination using static analysis procedures.

In lieu of the response spectrum approach, time histories of support motions may be used as excitations to the subsystems (Ref. 3). Because of the increased analytical effort compared to the response spectrum techniques, usually only a major equipment system would warrant a time history approach. The time history approach does, however, provide more realistic results in some cases as compared to the response spectrum envelope method for multiply-supported systems.

10. Use of Constant Vertical Static Factors

The acceptance criteria provided in SRP 3.7.2, Section II.10, are applicable.

11. Torsional Effects of Eccentric Masses

For seismic Category I subsystems, if the torsional effect of an eccentric mass such as a valve operator in a piping system is judged to be significant, the eccentric mass and its eccentricity should be included in the mathematical model. The criteria for significance will have to be determined on a case-by-case basis.

12. Category I Buried Piping Systems and Tunnels

For Category I buried piping systems and tunnels the following items should be considered in the analysis:

a. The inertial effects due to an earthquake upon buried piping systems and tunnels should be adequately accounted for in the analysis. Use of the procedures described in References 1 and 2 is acceptable.

b. The effects of static resistance of the surrounding soil on piping deformations or displacements, differential movements of piping anchors, bent geometry and curvature changes, etc., should be adequately considered. Use of the procedures described in Reference 4 is acceptable.

c. When applicable, the effects due to local soil settlements, soil arching, etc., should also be considered in the analysis.

13. Interaction of Other Piping with Category I Piping

To be acceptable, each non-Category I piping system should be designed to be isolated from any Category I piping system by either a constraint or barrier, or should be remotely located with regard to the seismic Category I piping system. If it is not feasible or practical to isolate the Category I piping system, adjacent non-Category I piping should be analyzed according to the same seismic criteria as applicable to the Category I piping system. For non-Category I piping systems attached to Category I piping systems, the dynamic effects of the non-Category I piping should be simulated in the modeling of the Category I piping. The attached non-Category I piping, up to the first anchor beyond the interface, should also be designed in such a manner that during an earthquake of SSE intensity it will not cause a failure of the Category I piping.

14. Seismic Analyses for Reactor Internals

To be acceptable, the seismic responses of the reactor pressure vessel and internals must be determined by a dynamic analysis. The analysis should comply with the applicable acceptance criteria provided in Sections II.1 and II.6 of SRP 3.7.2. In addition, the effects of piping-vessel interactions, externally applied structural restraints, hydrodynamic masses, etc., should be considered in the analysis.

15. Analysis Procedure for Damping

The acceptance criteria provided in SRP 3.7.2, Section II.15, are applicable.

III. REVIEW PROCEDURES

For each area of review, the following review procedure is followed. The reviewer will select and emphasize material from the procedures given below, as may be appropriate for a particular case.

1. Seismic Analysis Methods

The seismic analysis methods are reviewed to determine that these are in accordance with the acceptance criteria of SRP 3.7.2, Section II.1.

2. Determination of Number of Earthquake Cycles

Criteria or procedures used to establish the number of earthquake cycles are reviewed to determine that they are in accordance with the acceptance criteria as given in Section II.2. Justification for deviating from the acceptance criteria is requested from the applicant, as necessary.

3. Procedures Used for Analytical Modeling

The criteria and procedures used for modeling for the seismic subsystem analysis are reviewed to determine that these are in accordance with the acceptance criteria of SRP 3.7.2, Section II.3.

4. Basis for Selection of Frequencies

As applicable, criteria or procedures used to separate fundamental frequencies of components and equipment from the forcing frequencies of the support structure are reviewed to determine compliance with the acceptance criteria of Section II.4.

5. Use of Equivalent Static Load Method of Analysis

The criteria for the use of the equivalent static load method of analysis are reviewed to determine that these are in accordance with the acceptance criteria of Section II.5.

6. Three Components of Earthquake Motion

The procedures by which the three components of earthquake motion are considered in determining the seismic response of subsystems are reviewed to determine compliance with the acceptance criteria of SRP 3.7.2, Section II.6.

7. Combination of Modal Responses

The procedures for combining modal responses are reviewed to determine compliance with the acceptance criteria of SRP 3.7.2, Section II.7, when a response spectrum modal analysis method is used.

8. Analytical Procedures for Piping Systems

For all Category I piping and applicable non-Category I piping, the methods of seismic analysis (response spectra, time history, equivalent static load) are reviewed to determine that the techniques employed are in accordance with the acceptance criteria of Section II.8. Typical mathematical models are reviewed to judge whether all significant degrees of freedom have been included.

9. Multiply-Supported Equipment and Components With Distinct Inputs

The criteria for the seismic analysis of multiply-supported components and equipment with distinct inputs are reviewed to determine that the criteria are in accordance with the acceptance criteria of Section II.9.

10. Use of Constant Vertical Static Factors

Use of constant static factors as response loads in the vertical direction for the seismic design of any Category I subsystems in lieu of a detailed dynamic method is reviewed to determine that constant static factors are used only if the structure is rigid in the vertical direction.

11. Torsional Effects of Eccentric Masses

The procedures for seismic analysis of Category I piping subsystems are reviewed to determine compliance with the acceptance criteria of Section II.11.

12. Category I Buried Piping Systems and Tunnels

The analysis procedures for Category I buried piping and tunnels are reviewed to determine that they are in accordance with the acceptance criteria of Section II.12. This includes review of the procedures used to consider the inertial effects of soil media and the differential displacements at structural penetrations, etc. Any procedures that are not adequately justified are so identified and the applicant is requested to provide additional justification.

13. Interaction of Other Piping with Category I Piping

The criteria used to design the interfaces between Category I and non-Category I piping are reviewed to determine compliance with the acceptance criteria of Section II.13.

14. Seismic Analyses for Reactor Internals

The applicable methods of seismic analysis for reactor internals are reviewed to determine that the techniques employed are in accordance with the acceptance criteria of Section II.14. Typical mathematical models are reviewed to judge whether the characteristics of the reactor pressure vessel and internals are properly modeled and that all significant degrees of freedom have been incorporated, including any hydrodynamic effects. The number of modes used in the analysis are reviewed to assure that all significant modes have been included in the analysis.

15. Analysis Procedure for Damping

The analysis procedure to account for damping in different elements of the model of a coupled system is reviewed to determine that it is in accordance with the acceptance criteria of SRP 3.7.2, Section II.15.

IV. EVALUATION FINDINGS

Evaluation findings for SRP 3.7.3 have been combined with those of SRP 3.7.2 and are given under SRP 3.7.2, Section IV.

V. REFERENCES

1. N. M. Newmark, and E. Rosenblueth, "Fundamentals of Earthquake Engineering," Prentice Hall (1971).

2. N. M. Newmark, "Earthquake Response Analysis of Reactor Structures," Nuclear Engineering and Design, Vol. 20, pp. 303-322 (1972).

3. R. P. Kassawara, and D. A. Peck, "Dynamic Analysis of Structural Systems Excited at Multiple Support Locations," 2nd ASCE Specialty Conference on Structural Design of Nuclear Plant Facilities, Chicago, Dec. 17-18, 1973.

4. M. Hetenyi, "Beams on Elastic Foundation," The University of Michigan Press (1946).

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SECTION 3.7.4 SEISMIC INSTRUMENTATION

REVIEW RESPONSIBILITIES

Primary - Structural Engineering Branch (SEB)

Secondary - None

I. AREAS OF REVIEW

The following areas related to the seismic instrumentation program are reviewed:

1. Comparison with Regulatory Guide 1.12

   A comparison of the proposed seismic instrumentation with the seismic instrumentation guidelines of Regulatory Guide 1.12 (Ref. 2) is made. In addition, the bases for elements of the program that differ from Regulatory Guide 1.12 are reviewed.

2. Location and Description of Instrumentation

   The locations for the installation of seismic instrumentation such as triaxial peak accelerographs, triaxial time history accelerographs, and triaxial response spectrum recorders that will be installed in selected Category I structures and components are reviewed. The bases for selection of the instrumentation and the locations and a discussion of the extent to which the seismic instrumentation will be employed to verify the seismic analyses following an earthquake are reviewed.

3. Control Room Operator Notification

   The procedures that will be followed to inform the control room operator of the peak acceleration level and the input response spectra values shortly after occurrence of an earthquake are reviewed. Also reviewed are the bases for establishing predetermined values for activating the readout of the seismic instrumentation to the control room operator.

4. Comparison of Measured and Predicted Responses

   The criteria and procedures that will be used to compare measured responses of Category I structures and selected components in the event of an earthquake with the results of the seismic system and subsystem analyses are reviewed.

II. ACCEPTANCE CRITERIA

The acceptance criteria for the areas of review described in Section I of this plan are given below. Any other seismic instrumentation program which can be justified to be equivalent to the acceptance criteria may be used.

1. Comparison with Regulatory Guide 1.12

   The seismic instrumentation program is considered to be acceptable if it is in accordance with Regulatory Guide 1.12 (see also Table 3.7.4-1). This guide recommends provision of a triaxial time history accelerograph and a triaxial response spectrum recorder to measure the input time history and response spectra directly. Additional time history accelerographs, response spectrum recorders, peak accelerographs, and seismic switches are recommended to measure the responses of structures, equipment, and components at selected locations. The bases for elements of the proposed seismic instrumentation program that differ from Regulatory Guide 1.12 must be provided.

2. Location and Description of Instrumentation

   For the construction permit review there should be a commitment by the applicant to provide the following instruments at the given locations:

   a. A triaxial time history accelerograph in the free field or at the containment foundation, with readout in the control room.

   b. A seismic switch on the containment foundation, with readout in the control room.

   c. A triaxial response spectrum recorder on the containment foundation, with readout in the control room.

   In addition, a commitment to provide the recommended additional instrumentation at the various response locations should be made without providing details of actual locations.

   For the operating license review, a detailed seismic instrumentation plan including locations and descriptions of the remaining instrumentation should be provided. To be acceptable, the remaining instrumentation locations are related to the locations of the output vibratory motions used in the seismic design. Typical general locations are:

   a. Containment structure or reactor building.

   b. Reactor piping.

   c. Reactor equipment.

   d. Other category I structures, equipment, and piping.

Instrumentation should be provided depending upon the plant safe shutdown earthquake acceleration as given in Regulatory Guide 1.12. The specific locations are determined by the plant designer so as to obtain the most pertinent information (Ref. 3). A possible approach to the specification of the seismic instrumentation system is given in Regulatory Guide 1.12. Other desirable combinations of instruments which may prove to be as useful as the instrumentation plan outlined in the guide may be utilized.

The criteria for selection of Category I structures, components, and equipment to be instrumented and the location of instrumentation, as well as the extent to which this instrumentation is employed to verify the seismic analyses following an earthquake, should be specified. The criteria will be reviewed on a case-by-case basis.

3. Control Room Operator Notification

To be acceptable, the seismic switch located at the foundation of the containment should be connected to event indicators that are located in the control room, so that a signal is given when the preset threshold level (OBE acceleration level) resulting from the earthquake is exceeded. Also both audio and visual signals should be provided to the control room operators in the event of an earthquake.

In addition, the triaxial time history accelerograph located in the containment foundation or in the free field should be connected to the control room, so that peak acceleration level experienced in the basement of the reactor containment structure or in the free field is indicated to the control room operator. The response spectrum recorder in the reactor containment foundation or in the free field is also connected to the control room to indicate if the design response spectra values for discrete frequencies are exceeded during an earthquake.

4. Comparison of Measured and Predicted Responses

In the event of an earthquake, the control room operator should be immediately informed through the event indicators. If the instrumentation shows that the peak acceleration or the response spectra experienced at the foundation of the containment building or in the free field exceed the OBE acceleration level or response spectra, the plant should be shut down (Ref. 1) pending permission to resume operations. To help predict the capability of the plant for resuming operations, field inspection of safety-related items should be implemented and the measured responses from both the peak-recording and strong motion accelerographs should be compared with those assumed in the design.

The procedures for comparison of measured and predicted responses are acceptable if a commitment is made to provide detailed comparisons, as outlined below, between measured seismic responses of Category I structures and equipment with calculated responses determined from dynamic analysis. First, the time history records are digitized and corrected for time signal variations and baseline variations. The time history records from the triaxial sensors located in the free field or at the foundation of the containment building are used to calculate response spectra at appropriate critical damping

TABLE 3.7.4-1 SEISMIC INSTRUMENTATION REQUIREMENTS

Instrumentation		Triaxial Time-History Accelerograph		Triaxial Response Spectrum Recorder		Triaxial Peak Accelerograph		Seismic Switch	
Location	SSE	0.3 g or less	over 0.3 g	0.3 g or less	over 0.3 g	0.3 g or less	over 0.3 g	0.3 g or less	over 0.3 g
I. Free Field		1*#	1*#						
II. Inside Containment									
Basement		1*	1*	1*	1*			1*	1*
At Elevation		1	1						
Reactor Equip. Sup.				}1	}1				}1*
Reactor Piping Sup.									
Reactor Equipment						1	1		
Reactor Piping						1	1		
III. Outside Containment									
Cat. I Structure			1	1	1				
Cat. I Equip. Sup.					1				
Cat. I Piping Sup.				}1	1				
Cat. I Equipment							1		
Cat. I Piping						}1	1		

*Control room readout.
#May be omitted if soil-structure interaction is negligible.
}Denotes either of the two locations.

values. The response spectra thus obtained, or the response spectra from the response spectrum recorder, are compared with the design response spectra. In addition, the time history records from the free field triaxial sensor are used as input ground motion for the reactor building dynamic model, including soil where applicable. Amplified response spectra are then calculated at the locations of the other sensors in the reactor building for comparison and correlation with the response spectra directly measured. Structural responses and amplified response spectra are calculated using the free field time history records with the dynamic model for comparison with the original design and analysis parameters. This comparison permits evaluation of seismic effects on structures and equipment and forms the basis for remodeling, detailed analyses, and physical inspection.

III. REVIEW PROCEDURES

For each area of review, the following review procedure is followed. The reviewer will select and emphasize material from the procedures given below, as may be appropriate for a particular case.

1. Comparison with Regulatory Guide 1.12

   The seismic instrumentation program is checked to assure that the instrumentation is in accordance with the guidelines of Regulatory Guide 1.12. Any differences between the proposed and the guide seismic instrumentation, which have not been adequately justified, are identified and the applicant is informed of the need for additional technical justification.

2. Location and Description of Instrumentation

   At the operating license stage, the locations and descriptions of the seismic instrumentation are reviewed to determine that these are in accordance with the acceptance criteria of Section II.2. If the instrumentation provided is judged to be insufficient, the need for additional instrumentation is transmitted to the applicant.

3. Control Room Operator Notification

   The seismic instrumentation is checked to verify that the seismic switch located at the foundation of the containment structure or in the free field is connected to event indicators that are located in the control room, so that a signal is given when the preset threshold level is exceeded. If there is no provision for both audio and visual signals in the applicant's seismic instrumentation plan, the applicant is so informed with a request for compliance.

4. Comparison of Measured and Predicted Responses

   The criteria and procedures that will be used to compare measured responses of Category I structures and selected components in the event of an earthquake with the results of the seismic system and subsystem analyses are checked to verify that sufficient information as specified in Section II.4 is included. Any deficiency in the required information is identified and the applicant is requested to provide further information.

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information has been provided and his review supports conclusions of the following type, to be included in the staff's safety evaluation report:

> "The installation of the specified seismic instrumentation in the reactor containment structure and at other Category I structures, systems, and components constitutes an acceptable program to record data on seismic ground motion as well as data on the frequency and amplitude relationship of the seismic response of major structures and systems. A prompt readout of pertinent data at the control room can be expected to yield sufficient information to guide the operator on a timely basis for the purpose of evaluating the seismic response in the event of an earthquake. Data obtained from such installed seismic instrumentation will be sufficient to determine that the seismic analysis assumptions and the analytical model used for the design of the plant are adequate and that allowable stresses are not exceeded under conditions where continuity of operation is intended. Provision of such seismic instrumentation complies with Regulatory Guide 1.12."

V. REFERENCES

1. 10 CFR Part 100, Appendix A, "Seismic and Geologic Siting Criteria for Nuclear Power Plants."

2. Regulatory Guide 1.12, "Instrumentation for Earthquakes," Revision 1.

3. K. Kapur, "Seismic Instrumentation for Nuclear Power Plants," in "Proceedings of the Topical Meeting on Water Reactor Safety, Salt Lake City, March 1973," CONF-730304, U. S. Atomic Energy Commission Technical Information Center (1973).

(b) Where isolation by remote location is impracticable, systems containing high-energy fluids, or portions of the systems, are enclosed within the structures suitably designed to protect adjoining safety systems and components required to shut down the reactor safely and maintain the plant in a cold shutdown condition from postulated pipe failures within the enclosure.

(c) Where both isolation by remote location (as specified in I.a) and enclosure in protective structures (as specified in I.b) are impracticable, systems containing high-energy fluids, or portions of the systems, are provided with restraints and protective measures such that the operability and integrity of structures, safety systems and components that are required to shut down condition are not impaired.

(d) Protective enclosures for the piping systems containing high-energy fluids are designed as Seismic Category I structures to withstand the combined effects of a postulated pipe break, the dynamic effects of pipe whipping, the jet impingement forces, and the compartment pressurization as a consequence of discharging fluids in combination with the specified seismic event of the Safe Shutdown Earthquake and normal operating loads.

(e) Piping systems containing high-energy fluids are designed so that the effects of a single postulated pipe break cannot, in turn, cause failures of other pipes or components with unacceptable consequences.

In addition, any systems, or portions of systems, that are designed to mitigate the consequences of a postulated pipe failure, and to place the reactor in the cold shutdown condition, are provided with design features that will assure the performance of their safety function, assuming a single active component failure.

(f) For a postulated pipe failure, the escape of steam, water, and heat from structures enclosing the high-energy fluid containing piping does not preclude: 1) the accessibility to surrounding areas important to the safe control of reactor operations, 2) the habitability of the control room, 3) the ability of instrumentation, electric power supplies, and components and controls to initiate, actuate and complete a safety action. In this regard, a loss of redundancy is permissible but not the loss of function.

(g) The criteria for determination of postulated break locations are contained in the attached Appendix A, "Criteria for Determination of Postulated Pipe Break or Leakage Locations in Fluid Piping Systems Outside Containments."

II. PIPING SYSTEMS CONTAINING MODERATE-ENERGY FLUIDS* DURING REACTOR OPERATION

(a) Piping systems containing moderate-energy fluids are designed to comply with the criteria applied to high-energy fluid piping systems as listed under I., above, except that the piping is postulated to develop a limited-size through-wall leakage crack instead of a pipe break.

(b) For each postulated leakage, design measures are included that provide protection from the effects of the resulting water spray and flooding to the same extent required to satisfy criterion I(e).

(c) The criteria for determination of postulated leakage locations are contained in Appendix A.

The measures taken for the protection of structures, systems and components important to safety should not preclude the conduct of inservice examinations of ASME Class 2 and 3 pressure-retaining components as required by the rules of ASME Boiler and Pressure Vessel Code - Section XI, "Inservice Inspection of Nuclear Power Plant Components."

*Refer to Appendix A for identification of moderate-energy fluid systems.

Although compliance with the design criteria listed above should be accomplished by plant arrangement and layouts utilizing the separation concept to the extent practicable, special consideration will be necessary to provide adequate protection where interconnection is unavoidable between high-energy fluid containing piping and piping of systems important to safety.

We are prepared to discuss with you these guidelines for the design of new nuclear power plants with regard to protection required against postulated breaks of high and moderate energy piping outside of containment, particularly for those plants with construction permit applications currently under consideration.

Sincerely,

John F. O'Leary, Director
Directorate of Licensing

Enclosure:
Appendix A

APPENDIX A TO J.F. O'LEARY LETTER OF JULY 12, 1972

CRITERIA FOR DETERMINATION OF POSTULATED BREAK AND LEAKAGE LOCATIONS IN HIGH[1/] AND MODERATE[2/] ENERGY FLUID PIPING SYSTEMS OUTSIDE OF CONTAINMENT STRUCTURES[10/]

A. High-Energy Fluid Systems

1. For piping systems that by plant arrangement and layout are isolated by remote location for structures, systems, and components important to safety[3/], pipe breaks[4/] need not be postulated provided the requirements of A.4 are satisfied.

2. For piping systems that are enclosed in suitably designed concrete structures or compartments to protect structures, systems, and components important to safety, pipe break should be postulated at the following locations in each piping or branch run within the protective structure:

   a. the terminal ends[9/] of the piping or branch run (except as exempted by the provisions of A.4), if located within the protective structure or compartment, and
   b. each fitting (i.e., elbow, tee, cross, non-standard fitting), and
   c. a minimum of one break selected in each piping or branch run within the protective structure or compartment at a location that results in the maximum loading from the impact of the postulated ruptured pipe and jet discharge force on wall, floor, and roof of the structure or compartment, including internal pressurization, and taking into account any piping restraints provided to limit pipe motions.

3. For portions of piping systems that can neither be isolated as specified A.1, nor enclosed in protective structures as specified in A.2, pipe breaks should be postulated at the following locations in each piping or branch run within the confines of the structures or compartments that enclose or adjoin areas containing systems and components important to safety:

   a. the terminal ends[9/] of piping or branch run (except as exempted by A.4), if located within the boundary of the confining structure or each compartment within the structure; and
   b. any intermediate location within the boundary of the confining structure or each compartment within the structure where the stresses[5/] under the loadings associated with specified seismic events[6/] and operational plant conditions[7/] exceed 0.8 $(S_h + S_A)$[8/] or, in lieu of these calculated stress-related locations, at each fitting (i.e., elbow, tree, cross, non-standard fitting); and
   c. a minimum of two separated locations within the boundary of the confining structure or each compartment within the structure in piping or branch runs exceeding twenty pipe diameters in length; a minimum of one location in piping or branch runs twenty pipe-diameters or less in length except that no intermediate locations need to be postulated in branch runs that are three pipe-diameters or less in length. Intermediate break locations should be selected such that the maximum pipe whip and jet impingement will result, assuming for this purpose an unrestrained ruptured pipe.

4. For those portions of the piping passing through primary containment penetrations and extending to the first outside isolation valve, pipe breaks need not be postulated provided such piping is conservatively reinforced and restrained beyond the valve such that, in the event of a postulated pipe break outside containment, the transmitted pipe loads will neither impair the operability of the valve nor the integrity of the piping or the containment penetration. (A terminal end of such piping is considered to originate at this restraint location.)

B. Moderate-Energy Fluid Systems

1. For piping systems that by plant arrangement and layout are isolated and physically separated and remotely located from systems and components important to safety, through-wall leakage cracks need not be postulated.

2. For piping systems that are located in the same areas as high-energy fluid systems which, by the criteria of A.1 to A.3 have postulated pipe break locations, through-wall leakage cracks need not be postulated.

3. For piping systems that are located in areas containing systems and components important to safety, but where no high-energy fluid systems are present, through-wall leakage cracks should be postulated at the most adverse location to determine the protection needed to withstand the effects of the resulting water spray and flooding.

C. Size and Types of Pipe Breaks and Cracks

1. The following types of breaks should be postulated at the locations specified by the criteria listed under A. High-Energy Fluid Systems:

   a. longitudinal breaks in piping runs and branch runs with nominal pipe sizes of 4 inches and larger,

   b. circumferential breaks in piping runs and branch runs exceeding a nominal pipe size of 1 inch.

2. The following leakage cracks are postulated at the locations specified by the criteria listed under B. Moderate-Energy Fluid Systems:

   a. through-wall leakage cracks in piping and branch runs exceeding a nominal pipe size of 1 inch, where the crack opening is assumed as 1/2 the pipe diameter in length and 1/2 the pipe wall thickness in width.

FOOTNOTES

1/ High-energy systems include those systems where either of the following conditions are met:

a) the maximum operating temperature exceeds 200°F, and

b) the maximum operating pressure exceeds 275 psig.

2/ Moderate energy systems include those systems where both of the following conditions are met:

a) the maximum operating temperature is 200°F or less, and

b) the maximum operating pressure is 275 psig or less.

3/ Structures, systems, and components important to safety, as specified herein refer to those plant features required to shut down the reactor safely and maintain the plant in the cold shutdown condition.

4/ Break in piping means (a) a complete circumferential pipe severance and, (b) a longitudinal split opening an area equal to the pipe area, but without pipe severance. Such breaks are assumed to occur at each specified break location, but not concurrently.

5/ Either circumferential or longitudinal stresses derived on an elastically-calculated basis.

6/ Specified seismic events are earthquakes that produce at least 50 percent of the vibratory motion of the Safe Shutdown Earthquake (SSE).

7/ Operational plant conditions include normal reactor operation, upset conditions, (e.g., anticipated operational occurrences) and testing conditions.

8/ S_h is the allowable stress at maximum temperature, and S_A is the allowable stress range for expansion stresses for Class 2 and 3 piping as permitted by the rules of ASME Code Section III.

9/ Terminal ends of pipe runs originate at points of maximum constraint (e.g., Connections to vessels, pumps, valves, fittings that are rigidly anchored to structures) terminal ends of branch runs originate at pipe intersections and components that act as rigid constraints.

10/These criteria are intended for the purpose of designing piping restraints and do not preclude consideration of other aspects of the AEC General Design Criteria, such as single failure criteria and other additional protective measures required to provide protection against environmental conditions incident to postulated accidents.

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SECTION 3.6.2 DETERMINATION OF BREAK LOCATIONS AND DYNAMIC EFFECTS ASSOCIATED WITH THE POSTULATED RUPTURE OF PIPING

REVIEW RESPONSIBILITIES

Primary - Mechanical Engineering Branch (MEB)

Secondary - Auxiliary and Power Conversion Systems Branch (APCSB)
Structural Engineering Branch (SEB)
Electrical, Instrumentation and Control Systems Branch (EICSB)
Reactor Systems Branch (RSB)
Materials Engineering Branch (MTEB)

I. AREAS OF REVIEW

Information concerning break and crack location criteria and methods of analysis for evaluating the dynamic effects associated with postulated breaks and cracks in high and moderate energy fluid system piping, including "field run" piping, inside and outside of containment should be provided in the applicant's safety analysis report (SAR). This information is reviewed by the MEB in accordance with this plan, to confirm that requirements for the protection of structures, systems, and components relied upon for safe reactor shutdown or to mitigate the consequences of a postulated pipe break are met. At the construction permit (CP) stage, the staff review covers the following specific areas:

1. The criteria used to define break and crack locations and configurations.

2. The analytical methods used to define the forcing functions, including the jet thrust reaction at the postulated pipe break or crack location and jet impingement loadings on adjacent safety-related structures, systems, and components.

3. The dynamic analysis methods used to verify the integrity and operability of mechanical components, component supports, and piping systems, including restraints and other protective devices, under postulated pipe rupture loads.

At the operating license (OL) stage, the staff review covers the following specific areas:

1. The implementation of criteria for defining pipe break and crack locations and configurations.

2. The implementation of criteria dealing with special features, such as augmented inservice inspection programs or the use of special protective devices such as pipe whip restraints, including diagrams showing final configurations, locations, and orientations in relation to break locations in each piping system.

3. The acceptability of the analysis results, including the jet thrust and impingement forcing functions and pipe whip dynamic effects.

4. The design adequacy of systems, components, and component supports to assure that the intended design functions will not be impaired to an unacceptable level of integrity or operability as a result of pipe whip or jet impingement loadings.

Secondary reviews related to the areas of this plan are performed by other branches and the results are used by MEB to complete its evaluation. The secondary reviews are as follows:

1. The APCSB reviews plant arrangements where separation of high and moderate energy systems is the method of protection for essential systems and components, Sections B.1.a and B.2.a of BTP MEB 3-1. The APCSB identifies high and moderate energy systems outside containment and the essential systems and components that must be protected from postulated piping failures in these high and moderate energy systems.

2. The SEB reviews loading combinations and other design aspects of protective structures or compartments used to protect essential systems and components.

3. The MTEB reviews inservice inspection and related design provisions of high and moderate energy systems.

4. The RSB identifies high and moderate energy systems inside containment and the essential systems and components that must be protected from postulated piping failures in these high and moderate energy systems.

5. The EICSB reviews the environmental effects of pipe rupture, such as temperature, humidity, and spray-wetting with respect to the functional performance of essential electrical equipment and instrumentation.

II. ACCEPTANCE CRITERIA

The acceptance criteria for the areas of review described in Section I of this plan are as follows:

1. Postulated Pipe Break Locations Inside Containment

   Acceptable criteria to define postulated pipe break locations and configurations inside containment are specified in Regulatory Guide 1.46 (Ref. 2). If the criteria specified in Regulatory Guide 1.46 are inpractical to implement for a specific application, the criteria of Branch Technical Position (BTP) MEB 3-1 (Ref. 5) will be considered.

2. Postulated Pipe Break Locations Outside Containment

For protection against postulated pipe ruptures outside containment, References 3 and 4, and BTB MEB 3-1 provide acceptable criteria to define postulated rupture locations and plant layout considerations.

Reference 3 includes the area of concern in this plan and may be used for those plants for which construction permit applications were tendered before July 1, 1973, as specified in Section B.4 of BTP APCSB 3-1 (Ref. 6).

Reference 4 specifically emphasizes protection via plant arrangement and layouts utilizing the concept of physical separation to the extent practical, and may be used for those plants for which construction permit applications are tendered after July 1, 1973 and before July 1, 1975, as specified in Section B.4 of BTP APCSB 3-1.

BTP MEB 3-1 may be used for all applications, in lieu of References 3 and 4, at the option of applicants. After July 1, 1975, only BTP MEB 3-1 will be used by the staff in the review of all new construction permit applications.

3. Methods of Analysis

Detailed acceptance criteria covering pipe whip dynamic analysis, including determination of the forcing functions of jet thrust and jet impingement, are included in Section III, "Review Procedures," of this plan. The general bases and assumptions of the analysis are given in BTP MEB 3-1, Section B.3.

III. REVIEW PROCEDURES

The reviewer will select and emphasize material from this plan, as may be appropriate for a particular case.

1. The locations and configurations of breaks in high energy piping and leakage cracks in moderate energy piping are reviewed.

   a. At the CP stage, the applicant's criteria for determining break and crack locations are reviewed for conformance with the acceptance criteria referenced in Section II of this plan.

      Exceptions taken by the applicant to the referenced pipe break location and configuration criteria must be identified and the basis clearly justified so that evaluation is possible. Deviations from approved criteria and the justifications provided are reviewed to determine acceptability.

   b. At the OL stage, the following are reviewed to ensure that the pipe break criteria have been properly implemented:

      (1) Sketches showing the locations of the resulting postulated pipe ruptures, including indentification of longitudinal and circumferential breaks, structural barriers, if any, restraint locations, and the constrained directions in each restraint. APCSB reviews this information for fluid systems outside

containment and RSB for systems inside containment, with regard to identification of the systems in which failures should be postulated.

(2) A summary of the data developed to select postulated break locations including, for each point, the calculated stress intensity, the calculated cumulative usage factor, and the calculated primary plus secondary stress range as delineated in References 3 and 4 and Branch Technical Position MEB 3-1.

2. Analyses of pipe motion caused by the dynamic effects of postulated failures are reviewed. These analyses should show that pipe motions will not be such as to result in unacceptable impact upon, or overstress of, any structure, system, or component important to safety to the extent that essential functions would be impaired or precluded. The analysis methods used should be adequate to determine the resulting loadings in terms of the kinetic energy or momentum induced by the impact of the whipping pipe, if unrestrained, upon a protective barrier or a component important to safety and to determine the dynamic response of the restraints induced by the impact and rebound, if any, of the ruptured pipe.

At the CP stage, the staff reviews the applicant's criteria, methods, and procedures used or proposed for dynamic analyses by comparing them to the criteria which follow. At the OL stage, the analyses are reviewed in accordance with these criteria.

a. Dynamic Analysis Criteria

An analysis of the dynamic response or static equivalent thereof of the pipe run or branch should be performed for each longitudinal and circumferential postulated piping break.

The loading condition of a pipe run or branch, prior to the postulated rupture, in terms of internal pressure, temperature, and inertial effects should be consistent with the limiting upset plant operating condition.

In the case of a circumferential rupture, the need for a pipe whip dynamic analysis may be governed by considerations of the available driving energy as discussed in position 4.c of Reference 2.

Dynamic analysis methods used for calculating piping and restraint system responses to the jet thrust developed following the postulated rupture should adequately account for the following effects: (a) mass inertia and stiffness properties of the system, (b) impact and rebound, (c) elastic and inelastic deformation of piping and restraints, and (d) support boundary conditions.

The design strain limit for restraints should not exceed 0.5 of the ultimate uniform strain of the materials of the restraints. The method of dynamic analysis used should be capable of determining the inelastic behavior of the piping and restraint system within these design limits.

A 10% increase of minimum specified design yield strength (S_y) may be used in the analysis to account for strain rate effects.

Dynamic analysis methods and procedures presented should include:

(1) A representative mathematical model of the piping system or piping and restraint system.

(2) The analytical method of solution selected.

(3) Solutions for the most severe responses among the piping breaks analyzed.

(4) Solutions with demonstrable accuracy or justifiable conservatism.

The extent of mathematical modeling and analysis should be governed by the method of analysis selected.

b. Dynamic Analysis Models for Piping Systems

Acceptable models for the analysis of ASME Class 1, 2, and 3 piping systems and other piping systems which must be designed to seismic Category I standards include the following:

(1) Lumped Parameter Analysis Model: Lumped mass points are interconnected by springs to take into account inertia and stiffness properties of the system, and time histories of responses are computed by numerical integration, taking into account clearances at restraints and inelastic effects. In the calculation, the maximum possible initial clearance should be used to account for the most adverse dynamic effects of pipe whip.

(2) Energy Balance Analysis Model: Kinetic energy generated during the first quarter cycle movement of the ruptured pipe and imparted to the piping and restraint system through impact is converted into equivalent strain energy. In the calculation, the maximum possible initial clearance at restraints should be used to account for the most adverse dynamic effects of pipe whip. Deformations of the pipe and the restraint should be compatible with the level of absorbed energy. The energy absorbed by the pipe deformation may be deducted from the total energy imparted to the system. For applications where pipe rebound may occur upon impact on the restraint, an amplification factor of 1.2 should be used to establish the magnitude of the forcing function in order to determine the maximum reaction force of the restraint beyond the first quarter cycle of response. Amplification factors other than 1.2 may be used if justified by more detailed dynamic analysis.

(3) Static Analysis Model: The jet thrust force is represented by a conservatively amplified static loading, and the ruptured system is analyzed statically. An amplification factor can be used to establish the magnitude of the forcing

function. However, the factor should be based on a conservative value obtained by comparison with factors derived from detailed dynamic analyses performed on comparable systems.

(4) Other models may be considered if justified.

c. Dynamic Analysis Models for Jet Thrust Forces

(1) The time-dependent function representing the thrust force caused by jet flow from a posulated pipe break or crack should include the combined effects of the following: the thrust pulse resulting from the sudden pressure drop at the initial moment of pipe rupture; the thrust transient resulting from wave propagation and reflection; and the blowdown thrust resulting from build-up of the discharge flow rate, which may reach steady state if there is a fluid energy reservoir having sufficient capacity to develop a steady jet for a significant interval. Alternately, a steady state jet thrust function may be used, as outlined in (4), below.

(2) A rise time not exceeding one millisecond should be used for the initial pulse, unless longer crack propagation times or rupture opening times can be substantiated by experimental data or analytical theory.

(3) The time variation of the jet thrust forcing function should be related to the pressure, enthalpy, and volume of fluid in the upstream reservoir, and the capability of the reservoir to supply a high energy flow stream to the break area for a significant interval. The shape of the transient function may be modified by considering the break area and the system flow conditions, the piping friction losses, the flow directional changes, and the application of flow limiting devices.

(4) The jet thrust force may be represented by a steady state function if the energy balance model or the static model is used in the subsequent pipe motion analysis. In either case, a step function amplified as indicated in 2.b(2) or 2.b(3), above, is acceptable. The function should have a magnitude not less than

$$T = KpA$$

where

p = system pressure prior to pipe break

A = pipe break area, and

K = thrust coefficient.

To be acceptable, K values should not be less than 1.26 for steam-saturated water, or steam-water mixtures, or 2.0 for subcooled, nonflashing water.

3. Analyses of jet impingement forces are reviewed. These analyses should show that jet impingement loadings on nearby safety-related structures, systems, and components will not be such as to impair or preclude essential functions. Assumptions that are acceptable in modeling jet impingement forces are:

a. The jet area expands uniformly at a half angle not exceeding 10 degrees.

b. The impinging jet proceeds along a straight path.

c. The total impingement force acting on any cross-sectional area of the jet is time and distance invariant, with a total magnitude equivalent to the jet thrust force as defined in 2.c(4), above.

d. The impingement force is uniformly distributed across the cross-sectional area of the jet, and only the portion intercepted by the target is considered.

e. The break opening may be assumed to be a circular orifice of cross-sectional flow area equal to the effective flow area of the break.

f. Jet expansion within a zone of five pipe diameters from the break location is acceptable if substantiated by a valid analysis or testing, i.e., Moody's expansion model (Ref. 7). However, jet expansion is applicable to steam or water-steam mixtures only, and should not be applied to cases of subcooled water blowdown.

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information has been provided and that his review supports conclusions of the following type, to be included in the staff's safety evaluation report:

"The proposed design of piping restraints and measures to deal with jet impingement effects upon the reactor coolant pressure boundary and other safety-related systems provide adequate protection for the containment structure, reactor coolant pressure boundary elements, and other systems important to safety.

"The provisions for protection against dynamic effects associated with pipe ruptures of the reactor coolant pressure boundary inside containment and the resulting discharging fluid provide adequate assurance that design basis loss-of-coolant accidents will not be aggravated by sequential failures of safety-related piping, and emergency core cooling system performance will not be degraded by these dynamic effects.

"The proposed piping arrangement and applicable design considerations for high and moderate energy fluid systems inside and outside of containment, other than the reactor coolant pressure boundary, will provide adequate assurance that the unaffected system components, and those systems important to safety which are in close proximity to the systems in which postulated pipe failures are assumed to occur, will be protected. The design will be of a nature to mitigate the consequences of a pipe break so that the reactor can be safely shut down and maintained in a safe shutdown condition in the event of a postulated failure of a pipe carrying a high or moderate energy fluid inside or outside of containment."

V. REFERENCES

1. 10 CFR Part 50, Appendix A, General Design Criterion 4, "Environmental and Missile Design Bases."

2. Regulatory Guide 1.46, "Protection Against Pipe Whip Inside Containment."

3. Attachment to letter from A. Giambusso, December 1972, "General Information Required for Consideration of the Effects of a Piping System Break Outside Containment," Appendix B to BTP APCSB 3-1, attached to Standard Review Plan 3.6.1.

4. Letter from J. F. O'Leary, July 12, 1973, and attachment entitled,"Criteria for Determination of Postulated Break and Leakage Locations in High and Moderate Energy Fluid Piping Systems Outside of Containment Structures," Appendix C to BTP APCSB 3-1, attached to Standard Review Plan 3.6.1.

5. Branch Technical Position MEB 3-1, "Postulated Break and Leakage Locations in Fluid System Piping Outside Containment," attached to this plan.

6. Branch Technical Position APCSB 3-1, "Protection Against Postulated Piping Failures in Fluid Systems Outside Containment," attached to Standard Review Plan 3.6.1.

7. F. J. Moody, "Prediction of Blowdown and Jet Thrust Forces," ASME Paper 69 HT-31, August 6, 1969.

BRANCH TECHNICAL POSITION MEB 3-1

POSTULATED BREAK AND LEAKAGE LOCATIONS IN FLUID SYSTEM PIPING OUTSIDE CONTAINMENT

A. BACKGROUND

This position is intended to be used in conjunction with Branch Technical Position APCSB 3-1, attached to Standard Review Plan 3.6.1. The two positions together form an acceptable design approach for assuring that a plant can be safely shut down in the event of a piping failure outside containment. The background for this position is, therefore, the same as that for BTP APCSB 3-1 and reference should be made to that BTP for background information.

B. BRANCH TECHNICAL POSITION

1. High-Energy Fluid System Piping

a. Fluid Systems Separated from Essential Systems and Components

For the purpose of satisfying the separation provisions of plant arrangement as specified in B.1.a of Branch Technical Position (BIP) APCSB 3-1, a review of the piping layout and plant arrangement drawings should clearly show the effects of postulated piping breaks at any location are isolated or physically remote from essential systems and components.[1/] At the designer's option, break locations as determined from I.C. and I.D of this postion may be assumed for this purpose.

b. Fluid System Piping In Containment Penetration Areas

Breaks need not be postulated in those portions of piping identified in B.2.c of BTP APCSB 3-1 provided they meet the requirements of the ASME Code, Section III, Subarticle NE-1120 and the following additional design requirements:

(1) The following design stress and fatigue limits should not be exceeded:

For ASME Code, Section III, Class 1 Piping

(a) The maximum stress range should not exceed $2.4S_m$

(b) The maximum stress range between any two load sets (including the zero load set) should be calculated by Eq. (10) in Paragraph NB-3653, ASME Code, Section III, for normal and upset plant conditions and an operating basis earthquake (OBE) event transient.

If the calculated maximum stress range of Eq. (10) exceeds the limit of B.1.b(1)(a) but is not greater than $3S_m$, the limit of B.1.b(1)(c) should be met.

[1/] Definitions of underlined phrases are given in Appendix A to Branch Technical Position APCSB 3-1, attached to Standard Review Plan 3.6.1.

If the calculated maximum stress range of Eq. (10) exceeds $3S_m$, the stress ranges calculated by both Eq. (12) and Eq. (13) in Paragraph NB-3653 should meet the limit of B.1.b(1)(a) and the limit of B.1.b(1)(c).

(c) The cumulative usage factor should be less than 0.1 if consideration of fatigue limits is required according to B.1.b(1)(b).

(d) The maximum stress, as calculated by Eq. (9) in Paragraph NB-3652 under the loadings resulting from a postulated piping failure beyond these portions of piping should not exceed $2.25S_m$ except that following a failure outside containment, the pipe between the outboard isolation valve and the first restraint may be permitted higher stresses provided a plastic hinge is not formed and operability of the valves with such stresses is assured in accordance with the requirements specified in SRP 3.9.3. Primary loads include those which are deflection limited by whip restraints.

For ASME Code, Section III, Class 2 Piping

(e) The maximum stress ranges as calculated by the sum of Eq. (9) and (10) in Paragraph NC-3652, ASME Code, Section III, considering normal and upset plant conditions (i.e., sustained loads, occasional loads, and thermal expansion) and an OBE event should not exceed $0.8(1.2S_h + S_A)$.

(f) The maximum stress, as calculated by Eq. (9) in Paragraph NC-3652 under the loadings resulting from a postulated piping failure of fluid system piping beyond these portions of piping should not exceed $1.8S_h$.

Primary loads include those which are deflection limited by whip restraints. The exceptions permitted in (d) may also be applied provided that when the piping between the outboard isolation valve and the restraint is constructed in accordance with the Power Piping Code ANSI B31.1 (see APCSB 3-1 B-2.C.4), the piping shall either be of seamless construction with full radiography of all circumferential welds, or all longitudinal and circumferential welds shall be fully radiographed.

(2) Welded attachments, for pipe supports or other purposes, to these portions of piping should be avoided except where detailed stress analyses, or tests, are performed to demonstrate compliance with the limits of B.1.b(1).

(3) The number of circumferential and longitudinal piping welds and branch connections should be minimized. Where guard pipes are used, the enclosed portion of fluid system piping should be seamless construction unless specific access provisions are made to permit inservice volumetric examination of the longitudinal welds.

(4) The length of these portions of piping should be reduced to the minimum length practical.

(5) The design of pipe anchors or restraints (e.g., connections to containment penetrations and pipe whip restraints) should not require welding directly to the outer surface of the piping (e.g., flued integrally-forged pipe fittings may be used) except where such welds are 100 percent volumetrically examinable in service and a detailed stress analysis is performed to demonstrate compliance with the limits of B.1.b(1).

(6) Guard pipes provided for those portions of piping identified in B.2.c(2) of BTP APCSB 3-1 should be constructed in accordance with the rules of Class MC, Subsection NE of the ASME Code, Section III, where the guard pipe is part of the containment boundary. In addition, the entire guard pipe assembly should be designed to meet the following requirements and tests:

(a) The design pressure and temperature should not be less than the maximum operating pressure and temperature of the enclosed pipe under normal plant conditions.

(b) The design stress limits of Paragraph NE-3131(c) should not be exceeded under the loading associated with containment design pressure and temperature in combination with the safe shutdown earthquake.

(c) Guard pipe assemblies should be subjected to a single pressure test at a pressure not less than its design pressure.

c. Fluid Systems Enclosed Within Protective Structures

(1) With the exceptions of those portions of piping identified in B.1.b, breaks in Class 2 and 3 piping (ASME Code, Section III) should be postulated at the following locations in those portions of each piping and branch run within a protective structure or compartment designed to satisfy the plant arrangement provisions of B.1.b or B.1.c of BTP APCSB 3-1.

(a) At terminal ends of the run if located within the protective structure. Terminal ends are identified in APCSB 3-1 B.2.C.(3).

(b) At intermediate locations selected by one of the following criteria:

(i) At each pipe fitting (e.g., elbow, tee, cross, flange, and non-standard fitting), welded attachment, and valve. Where the piping contains no fittings, welded attachments, or valves, at one location at each extreme of the piping within the protective structure. (A terminal end, as determined by B.1.c(1)(a), may be considered as one of these extremes.)

(ii) At each location where the stresses[2/] exceed $0.8(1.2S_h + S_A)$ but at not less than two separated locations chosen on the basis of highest stress.[3/] Where the piping consists of a straight run without fittings, welded attachments, and valves, and all stresses are below $0.8(1.2S_h + S_A)$, a minimum of one location chosen on the basis of highest stress.

(2) Breaks in non-nuclear class piping should be postulated at the following locations in each piping or branch run:

(a) At terminal ends of the run if located within the protective structure.

(b) At each intermediate pipe fitting, welded attachment, and valve.

If a structure separates a high energy line from an essential component, that separating structure should be designed to withstand the consequences of the pipe break in the high energy line which produces the greatest effect at the structure irrespective of the fact that the above criteria might not require such a break location to be postulated.

(3) Applicable to (1) and (2) above:
If a structure separates a high energy line from an essential component, that separating structure should be designed to withstand the consequences of the pipe break in the high energy line which produces the greatest effect at the structure irrespective of he fact that the above criteria might not require such a break location to be postualted.

d. Fluid Systems Not Enclosed Within Protective Structures

(1) With the exceptions of those portions of piping identified in B.1.b, breaks in Class 2 and 3 piping (ASME Code, Section III) should be postulated at the following locations in those portions of each piping and branch run routed outside of, but alongside, above, or below, a protective structure or compartment containing essential systems and components and designed to satisfy the plant arrangement provisions of B.1.b or B.1.c of BTP APCSB 3-1.

Such piping should be considered as located adjacent to a protective structure if the distance between the piping and structure is insufficient to preclude impairment of the integrity of the structure from the effects of a postulated piping failure assuming the piping is unrestrained.

[2/] Stresses under normal and upset plant conditions, and an OBE event as calculated by Eq. (9) and (10), Para. NC-3652 of the ASME Code, Section III.

[3/] Select two locations with at least 10% difference in stress, or, if stresses differ by less than 10%, two locations separated by a change of direction of the pipe run.

(a) At terminal ends of the run if located adjacent to the protective structure. Terminal ends are identified in APCSB 3-1 B.2.C. (3).

(b) At intermediate locations selected by one of the following criteria:

(i) At each pipe fitting (e.g., elbow, tee, cross, flange, and non-standard fitting), welded attachment, and valve. Where the piping contains no fittings, welded attachments, or valves, at one location at each extreme of the piping run adjacent to the protective structure.

(ii) At each location where the stresses[4/] exceed $0.8(1.2S_h + S_A)$ but at not less than two separated locations chosen on the basis of highest stress.[5/] Where the piping consists of a straight run without fittings, welded attachment, or valves, and all stresses are below $0.8(1.2S_h + S_A)$, a minimum of one location chosen on the basis of highest stress.

(2) Breaks in non-nuclear class piping should be postulated at the following locations in each piping or branch run:

(a) At terminal ends of the run if located adjacent to the protective structure.

(b) At each intermediate pipe fitting, welded attachment, and valve.

(3) Applicable to (1) and (2) above:
If a structure separates a high energy line from an essential component, that separating structure should be designed to withstand the consequences of the pipe break in the high energy line which produces the greatest effect at the structure irrespective of the fact that the above criteria might not require such a break location to be postulated.

e. The designer should identify each piping run he has considered to postulate the break locations required by B.1.c and B.1.d above. In complex systems such as those containing arrangements of headers and parallel piping running between headers, the designer should identify and include all such piping within a designated run in order to postulate the number of breaks required by these criteria.

2. Moderate-Energy Fluid System Piping

a. Fluid Systems Separated from Essential Systems and Components
For the purpose of satisfying the separation provisions of plant arrangement as specified in B.1.a of BTP APCSB 3-1, a review of the piping layout and plant

4/ op. cit., p. 3.6.2-11, Footnote 2.
5/ op. cit., p. 3.6.2-11, Footnote 3.

arrangement drawings should clearly show that the effects of through-wall leakage cracks at any location in piping designed to seismic and non-seismic standards are isolated or physically remote from essential systems and components.

b. Fluid System Piping Between Containment Isolation Valves

Leakage cracks need not be postulated in those portions of piping identified in B.2.c. of BTP APCSB 3-1 provided they meet the requirements of the ASME Code, Section III, Subarticle NE-1120, and are designed such that the maximum stress range does not exceed $0.4(1.2S_h + S_A)$ for ASME Code, Section III, Class 2 piping.

c. Fluid Systems Within, or Outside and Adjacent to, Protective Structures

i. Through-wall leakage cracks should be postulated in seismic Category I fluid system piping located within, or outside and adjacent to, protective structures designed to satisfy the plant arrangement provisions of B.1.b. or B.1.c of BTP APCSB 3-1, except (1) where exempted by B.2.b and B.2.d, or (2) where the maximum stress range in these portions of Class 2 or 3 piping (ASME Code, Section III), or non-nuclear piping is less than $0.4(1.2S_h + S_A)$. The cracks should be postulated to occur individually at locations that result in the maximum effects from fluid spraying and flooding, with the consequent hazards or environmental conditions developed.

ii. Through-wall leakage cracks should be postulated in fluid system piping designed to non-seismic standards as necessary to satisfy B.3.d of BTP APCSB 3-1.

d. Moderate-Energy Fluid Systems in Proximity to High-Energy Fluid Systems

Cracks need not be postulated in moderate-energy fluid system piping located in an area in which a break in high-energy fluid system piping is postulated, provided such cracks would not result in more limiting environmental conditions than the high-energy piping break. Where a postulated leakage crack in the moderate-energy fluid system piping results in more limiting environmental conditions than the break in proximate high-energy fluid system piping, the provisions of B.2.c should be applied.

e. Fluid Systems Qualifying as High-Energy or Moderate-Energy Systems

Through-wall leakage cracks instead of breaks may be postulated in the piping of those fluid systems that qualify as high-energy fluid systems for only short operational periods[6/] but qualify as moderate-energy fluid systems for the major operational period.

[6/] An operational period is considered "short" if the fraction of time that the system operates within the pressure-temperature conditions specified for high-energy fluid systems is about 2 percent of the time that the system operates as a moderate-energy fluid system (e.g., systems such as the reactor decay heat removal system qualify as moderate-energy fluid systems; however, systems such as auxiliary feedwater systems operated during PWR reactor startup, hot standby, or shutdown qualify as high-energy fluid systems).

3. Type of Breaks and Leakage Cracks in Fluid System Piping

a. Circumferential Pipe Breaks

The following circumferential breaks should be postulated in high-energy fluid system piping at the locations specified in B.1 of this position:

(1) Circumferential breaks should be postulated in fluid system piping and branch runs exceeding a nominal pipe size of 1 inch, except where the maximum stress range[7] exceeds the limits specified in B.1.c(1)(b)(ii) and B.1.d(1)(b)(ii) but the circumferential stress range is at least 1.5 times the axial stress range. Instrument lines, one inch and less nominal pipe or tubing size should meet the provisions of Regulatory Guide 1.11.

(2) Where break locations are selected without the benefit of stress calculations, breaks should be postulated at the piping welds to each fitting, valve, or welded attachment. Alternatively, a single break location at the section of maximum stress range may be selected as determined by detailed stress analyses (e.g., finite element analyses) or tests on a pipe fitting.

(3) Circumferential breaks should be assumed to result in pipe severance and separation amounting to at least a one-diameter lateral displacement of the ruptured piping sections unless physically limited by piping restraints, structural members, or piping stiffness as may be demonstrated by inelastic limit analysis (e.g., a plastic hinge in the piping is not developed under loading).

(4) The dynamic force of the jet discharge at the break location should be based on the effective cross-sectional flow area of the pipe and on a calculated fluid pressure as modified by an analytically or experimentally determined thrust coefficient. Limited pipe displacement at the break location, line restrictions, flow limiters, positive pump-controlled flow, and the absence of energy reservoirs may be taken into account, as applicable, in the reduction of jet discharge.

(5) Pipe whipping should be assumed to occur in the plane defined by the piping geometry and configuration, and to cause pipe movement in the direction of the jet reaction.

b. Longitudinal Pipe Breaks

The following longitudinal breaks should be postulated in high-energy fluid system piping at the locations of the circumferential breaks specified in B.3.a:

[7] op. cit., p. 3.6.2-11, Footnote 2.

(1) Longitudinal breaks in fluid system piping and branch runs should be postulated in nominal pipe sizes 4-inch and larger, except where the maximum stress range[8] exceeds the limits specified in B.1.c(1)(b)(ii) and B.1.d(1)(b)(ii) but the axial stress range is at least 1.5 times the circumferential stress range.

(2) Longitudinal breaks need not be postulated at:

(a) Terminal ends provided the piping at the terminal ends contains no longitudinal pipe welds (if longitudinal welds are used, the requirements of B.3.b(1) apply).

(b) At intermediate locations where the criterion for a minimum number of break locations must be satisfied.

(3) Longitudinal breaks should be assumed to result in an axial split without pipe severance. Splits should be oriented (but not concurrently) at two diam[illegible]ly-opposed points on the piping circumference such that the jet reaction [illegible]auses out-of-plane bending of the piping configuration. Alternatively, a single split may be assumed at the section of highest tensile stress as determined by detailed stress analysis (e.g., finite element analysis).

(4) The dynamic force of the fluid jet discharge should be based on a circular or elliptical (2D x 1/2D) break area equal to the effective cross-sectional flow area of the pipe at the break location and on a calculated fluid pressure modified by an analytically or experimentally determined thrust coefficient as determined for a circumferential break at the same location. Line restrictions, flow limiters, positive pump-controlled flow, and the absence of energy reservoirs may be taken into account, as applicable, in the reduction of jet discharge.

(5) Piping movement should be assumed to occur in the direction of the jet reaction unless limited by structural members, piping restraints, or piping stiffness as demonstrated by inelastic limit analysis.

c. Through-Wall Leakage Cracks

The following through-wall leakage cracks should be postulated in moderate-energy fluid system piping at the locations specified in B.2 of this position:

(1) Cracks should be postulated in moderate-energy fluid system piping and branch runs exceeding a nominal pipe size of 1 inch.

[8] op. cit., p. 3.6.2-11, Footnote 2.

(2) Fluid flow from a crack should be based on a circular opening of area equal to that of a rectangle one-half pipe-diameter in length and one half pipe wall thickness in width.

(3) The flow from the crack should be assumed to result in an environment that wets all unprotected components within the compartment, with consequent flooding in the compartment and communicating compartments. Flooding effects should be determined on the basis of a conservatively estimated time period required to effect corrective actions.

C. REFERENCES

The references for this position are the same as for BTP APCSB 3-1, attached to Standard Review Plan 3.6.1.

NUREG-75/087

U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

SECTION 3.7.1 SEISMIC INPUT

REVIEW RESPONSIBILITIES

Primary - Structural Engineering Branch (SEB)

Secondary - Site Analysis Branch (SAB)

I. AREAS OF REVIEW

The following areas relating to the seismic input are reviewed:

1. Design Response Spectra

The seismic input, as defined by the design response spectra corresponding to the specified ground acceleration for a site is reviewed. A response spectrum is a plot of the maximum response of a family of single-degree-of-freedom damped oscillators with different frequency characteristics when the base of the oscillator is subjected to vibratory motion indicated by an appropriate time motion record. The response spectra are usually displayed on tripartite log-log graph paper. When obtained from a recorded earthquake record, the response spectrum tends to be irregular, with a number of peaks and valleys. A design response spectrum is a relatively smooth plot, obtained from a number of individual response spectra derived from records of past earthquakes. For high frequencies, spectral acceleration approaches the bound set by the maximum ground acceleration. For intermediate frequencies, spectral velocity is amplified relative to the ground velocity. For low frequencies, spectral displacement is amplified relative to the ground displacement.

Design response spectra for the operating basis earthquake (OBE) and safe shutdown earthquake (SSE) (Ref. 1) are reviewed. The design response spectra in the free field applied at the finished grade or at the various foundation levels of Category I structures are reviewed. Where applicable, the basis for any response spectra that differ from those of Regulatory Guide 1.60 (Ref. 2) is reviewed.

Site Analysis Branch (SAB) is responsible for reviewing the proposed values of the SSE and OBE ground acceleration appropriate for the site (see Standard Review Plan 2.5.2).

2. Design Time History

For the time history analyses, a comparison of the response spectra obtained in the free field at the finished grade level and at the foundation level (obtained from an appropriate time history at the base of the soil-structure interaction system) with

the design response spectra is reviewed for each of the damping values to be used in the design of structures, systems, and components. Alternatively if the design response spectra for the OBE and SSE are applied at the foundation levels of Category I structures in the free field, a comparison of the free field response spectra at the foundation level derived from an actual or synthetic time history, applied at the base of the soil-structure interaction system, with the design response spectra is reviewed for each of the damping values to be used in the analysis.

The current practice in the design of nuclear power plants is to use the response spectrum technique for seismic design of buildings and structures. In this technique, the input for the dynamic analysis of the model of major building structural elements is usually given in terms of a design spectrum appropriate for the seismic characteristics of the plant site. The analysis of interior equipment or component may also be based on the design response spectrum. However, such analysis requires an integrated model of the building and interior equipment which may not be practical for the many components which must be considered. In addition, care is required in modeling to assure accuracy of results where orders of magnitude differences exist in stiffness and mass characteristics between the building and component elements of the model.

For the analysis of interior equipment, where the equipment analysis is decoupled from the building, a compatible time history is needed for computation of the time-history response of each floor. The design floor spectra for equipment are obtained from this time history information.

In addition to the comparison of the response spectra derived from the time-history with the design response spectra, the period intervals at which the spectra values are calculated are also reviewed.

3. Critical Damping Values

The specific percentage of critical damping values used for Category I structures, systems, components, and soil are reviewed for both the OBE and the SSE. Critical damping is the amount of damping that would completely eliminate vibration. Although the use of critical damping is of little practical importance in itself, it assumes great significance as a measure of the damping capacity of a structure. Damping is conveniently expressed in the form of some percentage of critical damping.

Vibrating structures have energy losses which depend on numerous factors, such as material characteristics, stress levels, and geometric configuration. This dissipation of energy, or damping effect, occurs because a part of the excitation input is transformed into heat, sound waves, and other energy forms. The response of a system to dynamic loads is a function of the amount and type of damping existing in the system. A knowledge of appropriate values to represent this characteristic is essential for obtaining realistic results in dynamic analysis.

In practical seismic analysis, which usually employs linear methods of analysis, damping is also used to account for many nonlinear effects such as changes in boundary conditions,

joint slippage, plastic hinges, concrete cracking, gaps, and other effects which tend to alter response amplitude. In real structures, it is often impossible to separate "true" material damping from system damping, which is the measure of the energy dissipation, from the nonlinear effects. Overall structural damping used in design is normally determined by observing experimentally the total response of the structure.

Only the overall damping used for Category I structures, systems, components, and soil are reviewed. Where applicable, the basis for any damping values that differ from those given in Regulatory Guide 1.61 (Ref. 3) is reviewed.

4. Supporting Media for Category I Structures

The description of the supporting media for each Category I structure is reviewed, including foundation embedment depth, depth of soil over bedrock, soil layering characteristics, width of the structural foundation, total structural height, and soil properties to permit evaluation of the applicability of finite element or lumped spring approaches for soil-structure interaction analysis.

SAB is responsible for evaluating the physical properties of foundation soil and rock (see Standard Review Plans 2.5.1 and 2.5.4).

II. ACCEPTANCE CRITERIA

The acceptance criteria for the areas of review described in Section I are as follows:

1. Design Response Spectra

Design response spectra for the OBE and SSE are considered to be acceptable if the associated amplification factors are in accordance with Regulatory Guide 1.60, "Design Response Spectra for Nuclear Power Plants," for all damping values.

As noted in Regulatory Guide 1.60, there are site circumstances where the design response spectra are more appropriately developed to suit the particular site characteristics. Design response spectra based upon site-dependent analysis must be derived considering in situ variable soil properties, a representative number of site earthquake records, vertical amplification, possible slanted soil layers, and the influence of any predominant soil layers. The finite element approach or equivalent should be used to consider variable soil properties and nonlinear stress-strain relations in the soil media. The procedures used to obtain site-dependent design response spectra are reviewed on a case-by-case basis.

It should be noted that to be acceptable the design response spectra should be specified for three mutually orthogonal directions; two horizontal and one vertical. Since the two horizontal spectra have an equal probability of occurrence in any horizontal direction, current practice is to assume that the maximum accelerations in the two horizontal directions are equal, while the maximum vertical acceleration is 2/3 of the maximum horizontal acceleration.

2. Design Time History

In developing the design time history to be used at the base of the soil-structure interaction system, the following represents an acceptable procedure:

a. The design response spectra are defined for the free field and applied at the proposed finished grade level of the site.

b. Using an appropriate analysis method, with appropriate soil properties, obtain a time history at the base of the idealized soil profile.* One acceptable method for deconvolution analysis of the design response spectra at finished grade is a combined application of the SHAKE and LUSH computer codes (Refs. 4, 5). Use of other equivalent computer codes and analysis techniques is also acceptable. When the time history obtained from these methods is applied at the base of the idealized soil profile and the soil-structure interaction system, the resulting free field vibratory ground motion at finished grade level should give response spectra that envelop the design response spectra. This time history should appropriately account for variation in the soil properties at the site. In addition, when the time history obtained is applied at the base of the idealized soil profile, using appropriate soil properties, the vibratory motion calculated at the elevation of Category I structural foundations should, in general, give response spectra at all frequencies (.2 cps to 50 cps), not less than 60% of the design response spectra. The same limitation applies to the response spectra obtained at the foundation level in the free field for the soil-structure interaction system. Response spectral values in the idealized soil profile at the foundation level and those at the foundation level of the interaction system that are less than 60% of the corresponding design response spectral values may be acceptable provided they can be justified. The justification will be reviewed on a case-by-case basis.

c. The time history developed in item 2.b. above should be used at the base of the soil-structure interaction system, with appropriate soil properties, for subsequent soil-structure interaction analysis. The analysis method used should account for the strain dependency of soil modulus and damping. The peaks in the floor response spectra obtained from such a time history need be broadened by only $\pm$ 10% of the frequencies corresponding to the peaks.

d. An alternate and acceptable procedure is to apply the design response spectra at the foundation level of Category I structures in the free field. In this case, the design time history for use in the seismic analysis is acceptable if the response spectra in the free field at the foundation level obtained from the time history envelop the design response spectra for all damping values actually used in the analyses. The peaks in the floor response spectra obtained from such a time history should be broadened by a minimum of $\pm$ 15% of the frequencies corresponding to the peaks.

*Note: The idealized soil profile is the soil-structure interaction system without the structure.

The frequency intervals at which the spectra values are calculated from the design time history are to be small enough such that any reduction in these intervals does not result in more than 10% change in the computed spectra. Table 3.7.1-1 provides an acceptable set of frequencies at which the response spectra should be calculated. Another acceptable method is to choose a set of frequencies such that each frequency is within 10% of the previous one.

The acceptance criterion for meeting the spectra-enveloping requirement is that no more than five points of the spectra obtained from the time history should fall below, and no more than 10% below, the design response spectra.

Table 3.7.1-1
Suggested Frequency Intervals for Calculation of Response Spectra

Frequency Range (hertz)	Increment (hertz)
0.2 - 3.0	.10
3.0 - 3.6	.15
3.6 - 5.0	.20
5.0 - 8.0	.25
8.0 - 15.0	.50
15.0 - 18.0	1.0
18.0 - 22.0	2.0
22.0 - 34.0	3.0

3. Critical Damping Values

The specific percentage of critical damping values used in the analyses of Category I structures, systems, and components are considered to be acceptable if they are in accordance with Regulatory Guide 1.61, "Damping Values for Seismic Design of Nuclear Power Plants." Damping values in this guide are based upon the current (1973) state of the art. Higher damping values may be used in a dynamic seismic analysis if documented test data are provided to support them. These values would be reviewed and accepted by the staff on a case-by-case basis. The damping value for soil must be based upon actual measured values or other pertinent laboratory data considering variation in soil properties and strains within the soil.

4. Supporting Media for Category I Structures

To be acceptable, the description of supporting media for each Category I structure must include foundation embedment depth, depth of soil over bedrock, width of the structural foundation, total structural height, and soil properties such as shear wave velocity, shear modulus, and density as a function of depth.

III. REVIEW PROCEDURES

For each area of review, the following review procedure is followed. The reviewer will select and emphasize material from the procedures given below as may be appropriate for a particular case.

1. Design Response Spectra

   Design response spectra for the OBE and SSE for all damping values are checked to assure that the spectra are in accordance with the acceptance criteria as given in Section II. Any differences between the regulatory guide spectra and the proposed response spectra which have not been adequately justified are identified and the applicant is informed of the need for additional technical justification.

   Design response spectra based upon site-dependent analyses are reviewed to assure that the procedure used to develop these spectra considers in situ variable soil properties, a representative number of site earthquake records, vertical amplification, possible slanted soil layers, nonlinear stress-strain relations, and the influence of possibly predominant soil layers.

2. Design Time History

   Methods of defining the design time history are reviewed to ascertain that the acceptance criteria of Section II.2 are met.

3. Critical Damping Values

   The specific percentage of critical damping values for the OBE and SSE used in the analyses of Category I structures, systems, and components are checked to assure that the damping values are in accordance with the acceptance criteria as given in Section II.3. Any differences in damping values which have not been adequately justified are identified and the applicant is informed of the need for additional technical justification.

4. Supporting Media for Category I Structures

   The description of the supporting media is reviewed to verify that sufficient information, as specified in the acceptance criteria of Section II.4 is included. Any deficiency in the required information is identified and a request for additional information is transmitted to the applicant.

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information has been provided and that his evaluation supports conclusions of the following type, to be included in the staff's safety evaluation report:

"The seismic design response spectra (OBE and SSE) applied in the design of seismic Category I structures, systems, and components comply with the recommendations of Regulatory Guide 1.60, 'Design Response Spectra for Nuclear Power Plants.' The specific percentage of critical damping values used in the seismic analysis of Category I structures, systems, and components are in conformance with Regulatory Guide 1.61, 'Damping Values for Seismic Analysis of Nuclear Power Plants.' The synthetic time

history used for seismic design of Category I plant structures, systems, and components is adjusted in amplitude and frequency content to obtain reponse spectra that envelop the design response spectra specified for the site. Conformance with the recommendations of Regulatory Guides 1.60 and 1.61 assures that the seismic inputs to Category I structures, systems, and components are adequately defined so as to form a conservative basis for the design of such structures, systems, and components to withstand seismic loadings."

Alternatively, if a site-dependent analysis is used to develop the shape of the design response spectra, the language of the evaluation findings should be similar to the following:

"The site-dependent analysis has used a finite element approach to develop the seismic design response spectra from site-related information, including site time histories. This approach, used in lieu of the response spectra specified in Regulatory Guide 1.60, is acceptable since the free field response spectra at finished grade level (or at the structural foundation level) include consideration of appropriate amplification factors based upon an acceptable set of site earthquake records, and the analysis has taken into account actual soil properties at the site and includes consideration of appropriate damping values corresponding to the calculated soil stress levels. The specific percentage of critical damping values used in the seismic analysis of Category I structures, systems, and components are in conformance with Regulatory Guide 1.61, 'Damping Values for Seismic Analysis of Nuclear Power Plants.'

"The use of the site-dependent analysis and the damping values of Regulatory Guide 1.61 assures that the seismic inputs to Category I structures, systems, and components are adequately defined so as to form a conservative basis for the design of such structures, systems, and components to withstand seismic loadings."

V. REFERENCES

1. 10 CFR Part 100, Appendix A, "Seismic and Geologic Siting Criteria for Nuclear Power Plants."

2. Regulatory Guide No. 1.60, "Design Response Spectra for Nuclear Power Plants."

3. Regulatory Guide No. 1.61, "Damping Values for Seismic Analysis for Nuclear Power Plants."

4. Per B. Schnabel, J. Lysmer, and H. B. Seed, "SHAKE - A Computer Program for Earthquake Response Analysis of Horizontally Layered Sites," EERC 72-12, Earthquake Engineering Research Center, University of California, Berkeley (1972).

5. J. Lysmer, T. Udaka, H. B. Seed, and R. Hwang, "LUSH - A Computer Program for Complex Response Analysis of Soil-Stucture Systems," Draft Report, Earthquake Engineering Research Center, University of California, Berkeley (1974).

NUREG-75/087

U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

SECTION 3.8.1 CONCRETE CONTAINMENT

REVIEW RESPONSIBILITIES

Primary - Structural Engineering Branch (SEB)

Secondary - None

I. AREAS OF REVIEW

The following areas relating to concrete containments or to concrete portions of steel/concrete containments, as applicable, are reviewed.

1. Description of the Containment

The descriptive information, including plans and sections of the structure, is reviewed to establish that sufficient information is provided to define the primary structural aspects and elements relied upon to perform the containment function. In particular, the type of concrete containment is identified and its structural and functional characteristics are examined. Among the various types of concrete containments reviewed are:

a. Reinforced and prestressed concrete BWR containments utilizing the pressure-suppression concept, including the Mark I (modified lightbulb/torus), the Mark II (over/under) and the Mark III , ith horizontal venting between a centrally-located cylindrical drywell and a surrounding suppression pool).
b. Reinforced concrete PWR containments utilizing the pressure-suppression concept with ice-condenser elements.
c. Reinforced concrete PWR containments designed to function under sub-atmospheric conditions.
d. Reinforced and prestressed concrete PWR dry containments designed to function at atmospheric conditions.
e. Reinforced and prestressed concrete PWR or BWR containments utilizing special features or modifications of the above-listed types.

Various geometries have been utilized for these containments. The geometry most commonly encountered is an upright cylinder topped with a dome and supported on a flat concrete base mat. Although applicable to any geometry, the specific provisions of this review plan are best suited to the cylindrical type containment topped by a dome. If containments with

other types of geometry are reviewed, the necessary modifications to this plan are made on a case-by-case basis.

The geometry of the containment is reviewed, including sketches showing plan views at various elevations and sections in at least two orthogonal directions. The arrangement of the containment and the relationship and interaction of the shell with its surrounding structures and with its interior compartment walls and floors are reviewed to determine the effect which these structures could have upon the design boundary conditions and expected structural behavior of the containment when subjected to design loads.

General information related to the containment shell is reviewed including the following:

a. The base foundation slab, including the main reinforcement; the floor liner plate and its anchorage and stiffening system; the methods by which the interior structures are anchored through the liner plate and into the slab, if applicable.
b. The cylindrical wall, including the main reinforcement and prestressing tendons, if any; the wall liner plate and its anchorage and stiffening system; the major penetrations and the reinforcement surrounding them including the equipment and personnel hatches and major pipe penetrations; major structural attachments to the wall which penetrate the liner plate such as beam seats, pipe restraints and crane brackets; and external supports, if any, attached to the wall to support external structures such as enclosure buildings.
c. The dome and the ring girder, if any, including the main reinforcement and prestressing tendons; the liner plate and its anchorage and stiffening systems; and any major attachments to the liner plate made from the inside.
d. Steel components of concrete containments that resist pressure and are not backed by structural concrete are covered by Standard Review Plan 3.8.2.

2. Applicable Codes, Standards, and Specifications

Information pertaining to design codes, standards, specifications, regulations, general design criteria and regulatory guides, and other industry standards that are applied in the design fabrication, construction, testing, and in-service surveillance of the containment, is reviewed. The specific edition, date, or addenda identified for each document are reviewed.

3. Loads and Loading Combinations

Information pertaining to the applicable design loads and various combinations thereof is reviewed with emphasis on the extent of compliance with Article CC-3000 of the proposed "Standard Code for Concrete Reactor Vessels and Containments," ACI-ASME (ACI-359), (Ref. 1), (hereafter "the Code"). The loads normally applicable to concrete containments include the following:

a. Those loads encountered during preoperational testing.
b. Those loads encountered during normal plant startup, operation, and shutdown, including dead loads, live loads, thermal loads due to operating temperature, and hydrostatic loads such as those present in pressure suppression containments utilizing water.

c. Those loads to be sustained during severe environmental conditions, including those induced by the design wind and the operating basis earthquake specified for the plant site.
d. Those loads to be sustained during extreme environmental conditions, including those induced by the design basis tornado and the safe shutdown earthquake specified for the plant site.
e. Those loads to be sustained during abnormal plant conditions, which include loss-of-coolant accidents (LOCA). The main abnormal plant condition for containment design is the design basis LOCA. Also considered are other accidents involving various high-energy pipe ruptures. Loads induced on the containment by such accidents include elevated temperatures and pressures and possibly localized loads such as jet impingement and associated missile impact.
f. Those loads to be sustained, if applicable, after abnormal plant conditions including flooding of the containment subsequent to a LOCA for fuel recovery.

The various combinations of the above loads that are normally postulated and reviewed include the following:

Testing loads; normal operating loads; normal operating loads with severe environmental loads; normal operating loads with extreme environmental loads; normal operating loads with abnormal loads; normal operating loads with severe environmental and abnormal loads; normal operating loads with extreme environmental and abnormal loads; and post-LOCA flooding loads with severe environmental loads, if applicable.

The loads and load combinations described above are generally applicable to all containments. However, other site-related design loads might be applicable also. Such loads, which are not normally combined with abnormal loads, are reviewed on a case-by-case basis. They include those loads induced by floods, potential aircraft crashes, explosive hazards in proximity to the site and projectiles and missiles generated from activities of nearby military installations.

4. Design and Analysis Procedures

The design and analysis procedures utilized for the containment are reviewed with emphasis on the extent of compliance with Article CC-3000 of the Code, particularly with respect to the following:

a. Assumptions on boundary conditions.
b. Treatment of axisymmetric and non-axisymmetric loads.
c. Treatment of transient and localized loads.
d. Treatment of the effects of creep, shrinkage, and cracking of the concrete.
e. A description of the computer programs utilized in the design and analyses.
f. The treatment of the effects of seismically-induced tangential (membrane) shears.
g. The evaluation of the effects of variations in specified physical properties of materials on analytical results.
h. The treatment of the large, thickened penetration regions.

i. The treatment of the steel liner plate and its anchors. Steel penetration closures are covered by Standard Review Plan 3.8.2.

5. Structural Acceptance Criteria

The design limits imposed on the various parameters that serve to quantify the structural behavior of the containment are reviewed, with emphasis on the extent of compliance with Article CC-3000 of the Code, specifically with respect to allowable stresses, strains, gross deformations, and other parameters that identify quantitatively the margins of safety. For each load combination specified, the proposed allowable limits are compared with the acceptable limits delineated in Section II.5 of this plan. Included in these allowable limits are the following major parameters:

a. Compressive stresses in concrete, including membrane, membrane plus bending, and localized stresses.
b. Shear stresses in concrete, particularly those tangential (membrane) stresses induced by lateral loads.
c. Tensile stresses in reinforcement.
d. Tensile stresses in prestressing tendons.
e. Tensile or compressive stress/strain limits in the liner plate, including membrane and membrane plus bending.
f. Force/displacement limits in the liner plate anchors, including those induced by strains in the adjacent concrete.

6. Materials, Quality Control, and Special Construction Techniques

Information provided on materials that are used in construction of the containment is reviewed with emphasis on the extent of compliance with Article CC-2000 of the Code. Among the major materials of construction that are reviewed are the following:

a. The concrete ingredients.
b. The reinforcing bars and splices.
c. The prestressing system.
d. The liner plate.
e. The liner plate anchors and associated hardware.
f. The structural steel used for embedments such as beam seats and crane brackets.
g. The corrosion-retarding compounds used for the prestressing tendons.

The quality control program that is proposed for the fabrication and construction of the containment is reviewed with emphasis on the extent of compliance with Articles CC-4000 and CC-5000 of the Code, including the following:

Examination of the materials including tests to determine the physical properties of concrete, reinforcing steel, mechanical splices, the liner plate and its anchors, and the prestressing system, if any; placement of concrete; and erection tolerances of the liner plate, reinforcement, and prestressing system.

Special, new or unique construction techniques, if proposed, such as slip forming, are reviewed on a case-by-case basis to determine their effects on the structural integrity of the completed containment.

7. Testing and In-service Surveillance Requirements

The preoperational structural testing program for the completed containment and for individual components, such as personnel and equipment locks and hatches, is reviewed including the objectives of the test program and acceptance criteria, with emphasis on the extent of compliance with Article CC-3000 of the Code. In-service surveillance programs such as the periodic surveillance and inspection of the prestressing tendons, if any, are also reviewed, including the applicable Technical Specifications, at the operating license stage. Special testing and in-service surveillance requirements proposed for new or previously untried design approaches are also reviewed on a case-by-case basis.

II. ACCEPTANCE CRITERIA

The Regulatory acceptance criteria for the areas of review are as follows:

1. Description of the Containment

The descriptive information in the safety analysis report (SAR) is considered acceptable if it meets the minimum requirements set forth in Section 3.8.1.1 of the "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," Revision 2 (Ref. 2). If the concrete containment has new or unique features that are not specifically covered in the "Standard Format...," the reviewer determines that the information necessary to accomplish a meaningful review of the structural aspects of these new or unique features is presented, as appropriate.

2. Applicable Codes, Standards, and Specifications

The design, materials, fabrication, erection, inspection, testing, and in-service surveillance of concrete containments are covered by codes, standards, specifications, and guides that are either applicable in their entirety or in part. The following codes and guides are acceptable.

Code	Title
ACI/ASME (ACI-359)*	Proposed Standard Code for Concrete Reactor Vessels and Containments, ASME Boiler and Pressure Vessel Code, Section III, Division 2, Issued for Interim Trial-Use and Comment, April 1973

Regulatory Guides	Title
1.10	Mechanical (Cadweld) Splices for Reinforcing Bars of Category I Concrete Structures
1.15	Testing of Reinforcing Bars for Category I Concrete Structures
1.18	Structural Acceptance Test for Concrete Primary Reactor Containments
1.19	Nondestructive Examination of Primary Containment Liner Welds
1.35	Inservice Surveillance of Ungrouted Tendons in Prestressed Concrete Containment Structures
1.55	Concrete Placement in Category I Structures

*Issued as an interim code for trial use and comments and subject to revisions prior to publication as a mandatory code. Any proposed use of amendments to the April version of the interim code will be reviewed on a case-by-case basis.

3. Loads and Loading Combinations

The specified loads and load combinations are acceptable if found to be in accordance with Article CC-3000 of the Code with the exceptions listed below taken to the requirements specified in Table CC-3200-1.

a. Y_j, jet impingement loads, and Y_m, impact loads of missiles associated with accidents, should be included.

b. The 6th combination, representing "abnormal" load conditions, need not include Y_r in combination with 1.5P.

c. In the 7th, 8th, and 9th combinations, representing "abnormal/severe environmental" and abnormal/extreme environmental" load conditions, the "and/or" between R_a and Y_r should be deleted and, in addition to R_a and Y_r, the combinations should include Y_j and Y_m.

d. The maximum values of P_a, T_a, R_a, Y_r, Y_j, and Y_m should be applied simultaneously, where appropriate, unless a time-history analysis is performed to justify doing otherwise.

Where post-LOCA flooding is a design consideration, the following combination should also be considered in the factored load category:

1.0 D + 1.0 L + 1.0 F + 1.0 Feqo, where D, L, Feqo are as defined in the Code and F is the load generated by the post-LOCA flooding of the containment.

4. Design and Analysis Procedures

The procedures of design and analysis utilizes for the concrete containment, including the steel liner, are acceptable if found in accordance with those stipulated in Article CC-3300 of the Code. In particular, for the areas of review outlined in Section 1.4 of this plan, the following procedures are, in general, acceptable:

a. Assumptions on boundary conditions

The boundary conditions depend on the methods of analysis to be used and the portions of the containment shell to be separately analyzed. If the analysis is to be accomplished through the use of the finite element technique, and is to include the foundation media, the boundary would be the demarcation lines separating the foundation mass taken into consideration in the analysis from the surrounding media. The boundaries of the foundation mass considered have to be so selected that any further extension of the boundaries will not affect the results by more than 15 percent.

If only the containment shell and its foundation mat are taken into consideration in the analysis, then the bottom of the foundation slab is the boundary of the analytical model. The foundation media should be represented by appropriate soil springs.

If separate analyses of the containment shell and the base mat are to be used, it is considered acceptable if strain compatibility of the bottom portion of the shell with the base mat is maintained.

b. Axisymmetric and non-axisymmetric loads

Even with the large penetrations and buttresses that may be utilized in the shell, the overall behavior of the shell has been shown to be axisymmetric under pressure. Therefore, it is acceptable if such an assumption is made with respect to the containment geometry. However, for loads such as those induced by wind, tornadoes, earthquakes, and pipe rupture, the non-axisymmetric effect of these loads should be considered.

c. Transient and localized loads

During normal operation, a linear temperature gradient across the containment wall thickness may develop. After the loss-of-coolant accident (LOCA), however, the sudden increase in temperature in the steel liner and the adjacent concrete may produce a non-linear transient temperature gradient across the containment wall thickness. Effects of such transient loads should be considered.

In a PWR ice-condenser containment, non-axisymmetric and transient pressure loads resulting from compartmentation inside the containment will develop after a LOCA.

For the effects of such localized and transient loads, the overall behavior of the containment structure should first be determined. A portion of the containment shell, within which the localized or transient load is located, should then be analyzed, using the results obtained from the analysis of the overall vessel behavior as boundary conditions.

d. Creep, shrinkage, and cracking of concrete

Creep and shrinkage values for concrete should be established by tests performed on the concrete which is to be used in the containment structure, or from data obtained on completed containments constructed of the same kind of concrete. In establishing these values, consideration should be given to the differences in the environment between the test samples and the actual concrete in the structure. Cracking of the concrete may be considered in either of the following two ways: (i) the moments, forces, and shears under load may be obtained on the basis of an uncracked section for all loading combinations. In sizing the reinforcing steel required, however, the concrete shall not be relied upon for resisting tension. Thermal moments may be modified to take creep and cracking into consideration. (ii) For axisymmetrical loadings, cracking of the concrete may be considered through the use of computer programs which are capable of treating such cracking by an iterative process. However, for non-axisymmetric loadings, most of the computer programs available do not have the capability of considering cracking, since the structure itself becomes non-axisymmetric when concrete cracking is to be considered iteratively. Accordingly, if the concrete is cracked under any load combination involving axisymmetric and non-axisymmetric loadings, a method should be described for considering cracking. Such methods are reviewed on a case-by-case basis.

e. Computer programs

The computer programs used in the design and analysis should be described and validated by any of the following procedures or criteria:

(i) The computer program is a recognized program in the public domain and has had sufficient history of use to justify its applicability and validity without further demonstration.

(ii) The computer program solution to a series of test problems has been demonstrated to be substantially identical to those obtained by a similar and independently-written and recognized program in the public domain. The test problems should be demonstrated to be similar to or within the range of applicability of the problems analyzed by the public domain computer program.

(iii) The computer program solution to a series of test problems has been demonstrated to be substantially identical to those obtained from classical solutions or from accepted experimental tests, or to analytical results published in technical literature. The test problems should be demonstrated to be similar to or within the range of applicability of the classical problems analyzed to justify acceptance of the program.

A summary comparison should be provided for the results obtained in the validation of each computer program.

f. Tangential shear

Design and analysis procedures for tangential shear are acceptable if in accordance with those contained in Article CC-3000 of the Code. The exceptions taken by the Regulatory staff to the provisions of this Article, as contained in Section II.5 of this plan, are to be noted.

g. Variation in physical material properties

For considering the effects of possible variations in the physical properties of materials on the analytical results, the upper and lower bounds of these properties should be used in the analysis, wherever critical. Among the physical properties that may be critical include the soil modulus, and modulus of elasticity and Poisson's ratio of concrete.

h. Thickened penetrations

The effect of the large, thickened penetration regions on the overall behavior of the containment may be treated in the same manner as for localized loads discussed in item (c).

i. Steel liner plate and anchors

For the design and analysis of the liner plate and its anchorage system, the procedures furnished are found adequate and acceptable if in accordance with the provisions of Subarticle CC-3600 of the Code. In general, the liner plate analysis should consider deviations in geometry due to fabrication and erection tolerances, and variations of the assumed physical properties of the liner and anchor material. Since the liner plate is usually anchored at relatively closely spaced intervals, the analysis procedures are acceptable if based on either the classical plate or beam theory. Since the concrete shell is much stiffer than the liner plate, the strains in the liner will essentially follow those in the concrete. The strains in the concrete under the various load combinations as obtainable from the analysis of the shell are thus imposed on the liner plate and the resulting strains and stresses in the liner and its anchors should be lower than the allowable limits defined in Tables CC-3700-1 and CC-3700-2 of the Code.

5. Structural Acceptance Criteria

a. For the structural portions of the containment, the specified allowable limits for stresses and strains are acceptable if they are in accordance with Subsection CC-3400 of the Code but with the following exceptions:

CC-3411.5

- Under no conditions shall the tangential shear stress carried by the concrete, v_c, exceed 40 psi and 60 psi for the 7th and 9th combinations of Table CC-3200-1, representing abnormal/severe environmental and abnormal/extreme environmental conditions, respectively.
- For prestressed concrete, the principal tensile stress shall not exceed $4\sqrt{f'c}$.

C-3421.1

- The footnote on page 196 indicates that the 33-1/3% increase in allowable stresses is permitted only temperature loads and not for seismic or wind loads.

CC-3422.1

- Item (c) should be deleted.

CC-3422.2

- The footnote on page 197 should be deleted.

b. For the liner plate and its anchorage system, the specified limits for stresses and strains are acceptable if in accordance with Tables CC-3700-1 and CC-3700-2 of the Code, respectively.

6. Materials, Quality Control, and Special Construction Techniques

a. The specified materials of construction are acceptable if in accordance with Article CC-2000 of the Code with the following exceptions:

(i) CC-2243.3 permits the chloride and nitrate content of constituents of cement grout for prestressing tendons to reach maximum limits of 300 ppm and 100 ppm, respectively. It is understood that these are total allowable limits in the grout, not just for the mixing water, and ACI 318 (commentary 3.4.1) is the apparent source of the 300 ppm chloride limit. However, in view of the cautions approach taken in the wording of 3.4.1 and 3.6.1 of the ACI 318-71 commentary, the potential sensitivity of the prestressed steels to chloride ions, and since local municipal treated potable water contains in the order of only 12 ppm each of chlorides and nitrates, reconsideration of the 300 ppm and 100 ppm limits are recommended. Consideration should be given to the possibility of chloride and nitrate content of all the grout constituents should be individually determined and the possible concentration levels evaluated.

(ii) In Tables I-1.1 and I-1.2, the inclusion of deformed bars as acceptable materials for prestressing systems should be deleted, since neither ASTM acceptance nor sufficient user justification data have been secured, as per the requirements of the Code.

b. Quality control programs are acceptable if in accordance with applicable portions of Articles CC-4000 and CC-5000 of the Code as augmented by Regulatory Guides 1.10 for Cadweld reinforcement splicing (Ref. 3), 1.15 for testing of reinforcing bars (Ref. 4), 1.19 for the nondestructive examination of the liner plate welds (Ref. 5), and 1.55 for concrete placement (Ref. 6).

c. Special construction techniques, if any, are reviewed on a case-by-case basis.

7. Testing and In-service Surveillance Requirements

a. Procedures for the post-construction preoperational structural proof test proposed for the containment are acceptable if found in accordance with those delineated in Article CC-6000 of the Code as augmented by the provisions delineated in Regulatory Guide 1.18 (Ref. 7).

b. For prestressed concrete containments, in-service surveillance requirements for the tendons, as presented in the Technical Specifications of the Operating License, are acceptable if in accordance with Regulatory Guides 1.35 for ungrouted tendons (Ref. 8) and 1.___ for grouted tendons (Ref. 9), respectively.

III. REVIEW PROCEDURES

The reviewer selects and emphasizes material from the review procedures described below as may be appropriate for a particular case.

1. Description of the Containment

After the type of containment and its functional characteristics are identified, information on similar and previously licensed applications is obtained for reference. Such information, which is available in safety analysis reports and amendments of previous license applications, enables identification of differences for the case under review. These differences require additional scrutiny and evaluation. New and unique features that have not been used in the past are of particular interest and are examined in greater detail. The information furnished in the SAR is reviewed for completeness in accordance with the "Standard Format...," Revision 2. A decision is then made with regard to the sufficiency of the descriptive information provided in the SAR. Any additional required information not provided is requested from the applicant at an early stage of the review process.

2. Applicable Codes, Standards, and Specifications

The list of codes, standards, guides, and specifications are checked against the list in Section II.2 of this plan. The reviewer assures himself that the applicable edition and stated effective addenda are utilized.

3. Loads and Loading Combinations

The reviewer verifies that the loads and load combinations, as described by the applicant, are as conservative as those referenced in Section II.3 of this plan. Loading conditions that are unique to the site, such as potential aircraft crashes, and that are not specifically covered in Section II.3, are treated on a case-by-case basis. Any deviations from the acceptance criteria for loads and load combinations that have not been adequately justified are identified as unacceptable and this information is transmitted to the applicant for further consideration.

4. Design and Analysis Procedures

The reviewer assures himself that the applicant has committed to utilize design and analysis procedures delineated in Article CC-3000 of the Code. Any exceptions to these procedures are reviewed and evaluated on a case-by-case basis. In particular, the areas of review contained in Section I.4 of this plan are evaluated for conformance with the acceptance criteria.

5. Structural Acceptance Criteria

The limits on allowable stresses and strains in the concrete, reinforcement, liner plate and its anchors, and in components of the prestressing system, if any, are reviewed and compared with the acceptable limits referenced in Section II.5 of this plan. Where the applicant proposes to exceed some of these limits for some of the load combinations and at some localized points on the structure, the justification provided to show that the structural integrity of the containment will not be affected is evaluated. If such justification is unacceptable, the applicant is required to submit additional justification or otherwise comply with the acceptance criteria delineated in Section II.5 of this plan.

6. Materials, Quality Control, and Special Construction Techniques

The information provided on materials, quality control programs, and special construction techniques, if any, is reviewed and compared with that referenced in Section II.6 of this plan. If a material not used in previously licensed applications is utilized, the applicant is requested to provide sufficient test and user data to establish the acceptability of the material. Similarly, any new quality control programs or construction techniques are reviewed and evaluated to assure that there will be no degradation of structural quality that might affect the structural integrity of the containment, the liner plate, and its anchorage system.

7. Testing and In-service Surveillance Requirements

The initial structural overpressure test program is reviewed and compared with that indicated as acceptable in Section II.7 of this plan. Proposed deviations are considered on a case-by-case basis. In-service surveillance programs, particularly for the prestressing tendons, if any, as presented in the Technical Specifications of the Operating License, are similarly reviewed.

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information has been provided to satisfy the requirements of this review plan, and concludes that his evaluation is sufficiently complete and adequate to support the following type of conclusive statement to be included in the staff's Safety Evaluation Report:

"The criteria used in the analysis, design, and construction of the concrete containment structure to account for anticipated loadings and postulated conditions that may be imposed upon the structure during its service lifetime are in conformance with established criteria, codes, standards, guides, and specifications acceptable to the Regulatory staff.

"The use of these criteria as defined by applicable codes, standards, guides, and specifications; the loads and loading combinations; the design and analysis procedures; the structural acceptance criteria; the materials, quality control programs, and special construction techniques; and the testing and in-service surveillance requirements provide reasonable assurance that, in the event of winds, tornadoes, earthquakes and various postulated accidents occurring within and outside the containment, the structure will withstand the specified design conditions without impairment of structural integrity or safety function. Conformance with these criteria constitutes an acceptable basis for satisfying, in part, the requirements of General Design Criteria 2, 4, 16, and 50."

V. REFERENCES

1. ASME Boiler and Pressure Vessel Code, Section III, Division 2 (ACI-359), "Proposed Standard Code for Concrete Reactor Vessels and Containments," issued for interim trial use and comment, April 1973, American Society of Mechanical Engineers.

2. Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," Revision 2 (in preparation).

3. Regulatory Guide 1.10, "Mechanical (Cadweld) Splices in Reinforcing Bars of Category I Concrete Structures."

4. Regulatory Guide 1.15, "Testing of Reinforcing Bars for Category I Concrete Structures."

5. Regulatory Guide 1.19, "Nondestructive Examination of Primary Containment Liner Welds."

6. Regulatory Guide 1.55, "Concrete Placement in Category I Structures."

7. Regulatory Guide 1.18, "Structural Acceptance Test for Concrete Primary Reactor Containments."

8. Regulatory Guide 1.35, "In-service Surveillance of Ungrouted Tendons in Prestressed Concrete Containments."

9. Regulatory Guide 1.__, "In-service Surveillance in Prestressed Concrete Containments with Grouted Tendons," (in preparation).

10. 10 CFR Part 50, Appendix A, General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena."

11. 10 CFR Part 50, Appendix A, General Design Criterion 4, "Environmental and Missile Design Bases."

12. 10 CFR Part 50, Appendix A, General Design Criterion 16, "Containment Design."

13. 10 CFR Part 50, Appendix A, General Design Criterion 50, "Containment Design Basis."

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U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

SECTION 3.8.2 STEEL CONTAINMENT

REVIEW RESPONSIBILITIES

Primary - Structural Engineering Branch (SEB)

Secondary - Materials Engineering Branch (MTEB)

I. AREAS OF REVIEW

The following areas relating to steel containments or to other Class MC steel portions of steel/concrete containments, as applicable, are reviewed.

1. Description of the Containment

The descriptive information, including plans and sections of the structure, is reviewed to establish that sufficient information is provided to define the primary structural aspects and elements relied upon to perform the containment function. In particular, the type of steel containment is identified and its structural and functional characteristics are examined. Among the various types of steel containments reviewed are:

a. Steel BWR containments utilizing the pressure-suppression concept, including the Mark I (lightbulb/torus), the Mark II (over/under) and the Mark III (with horizontal venting between a centrally-located cylindrical drywell and a surrounding suppression pool).
b. Steel PWR containments utilizing the pressure-suppression concept with ice-condenser elements.
c. Steel PWR dry containments.

Various geometries have been utilized for these containments. The geometry most commonly encountered, however, is an upright cylinder topped with a dome and supported on either a flat concrete base mat covered with a liner plate, or on a concrete foundation built around the bottom portion of the steel shell, which is an inverted dome. Although applicable to any geometry, the specific provisions of this review plan are best suited to the cylindrical-type steel containment surrounded by a Category I concrete shield building. If containments with other types of geometry are reviewed, the necessary modifications to this plan are made on a case-by-case basis.

The geometry of the containment is reviewed, including sketches showing plan views at various elevations and sections in at least two orthogonal directions. The arrangement of the containment and the relationship and interaction of the shell with its surrounding shield building and with its interior compartments, walls and floors, are reviewed to determine the effect which these structures could have upon the design boundary conditions and the expected behavior of the shell when subjected to the design loads.

General information related to the containment shell is reviewed including the following:

a. The foundation of the steel containment including the following:
   (i) If the bottom of the steel containment is continuous through an inverted dome, the method by which the inverted dome and its supports are anchored to the concrete foundation, which is covered by Standard Review Plan 3.8.5, is reviewed.
   (ii) If the bottom of the steel containment is not continuous, and where a concrete base slab topped with a liner plate is used for a foundation, the extent of descriptive information reviewed for the foundation is contained and is reviewed as stated in Section I.1 of Standard Review Plan 3.8.1. Further, the method of anchorage of the steel cylindrical shell walls in the concrete base slab is reviewed, particularly the connection between the floor liner plate and the steel shell.
b. The cylindrical portion of the shell is reviewed including major structural attachments such as beam seats, pipe restraints, crane brackets, and shell stiffeners, if any, in the hoop and vertical directions.
c. The dome of the steel containment including any reinforcement at the dome/cylinder junction, penetrations or attachments made on the inside such as supports for containment spray piping, and any stiffening of the dome.
d. Major penetrations or portions thereof, of steel or concrete containments, to the limits defined by Figure NE-1132.1 of Subsection NE of the ASME Boiler and Pressure Vessel Code (hereafter "the Code"), Section III, Division 1 (Ref. 1), and portions of the penetrations that are intended to resist pressure but are not backed by structural concrete, including those of sleeved and unsleeved piping penetrations, mechanical systems penetrations such as fuel transfer tubes, electrical penetrations, and access openings such as the equipment hatch and personnel locks.

2. Applicable Codes, Standards, and Specifications

The information pertaining to design codes, standards, specifications, regulations, general design criteria and regulatory guides, and other industry standards that are used in the design, fabrication, construction, testing, and in-service surveillance of the steel containment, is reviewed. The specific edition, date, or addenda identified for each document are also reviewed.

3. Loads and Loading Combinations

Information pertaining to the applicable design loads and various load combinations is reviewed with emphasis on the extent of compliance with Subsection NE of the Code, Section III, Division 1, and with Regulatory Guide 1.57 (Ref. 2). The loads normally applicable to steel containments include the following:

a. Those loads encountered during preoperational testing.
b. Those loads encountered during normal plant startup, operation, and shutdown, including dead loads, live loads, thermal loads due to operating temperature, and hydrostatic loads such as those present in pressure-suppression containments utilizing water.
c. Those loads to be sustained during severe environmental conditions, including those induced by design wind (if not protected by a shield building) and the operating basis earthquake.
d. Those loads to be sustained during extreme environmental conditions, including those induced by the design basis tornado (if not protected by a shield building) and the safe shutdown earthquake specified for the plant site.
e. Those loads to be sustained during abnormal plant conditions, which include loss-of-coolant accidents (LOCA). The main abnormal plant condition for containment design is the design basis LOCA. Also to be considered are other accidents involving various high-energy pipe ruptures. Loads induced on the containment by such accidents include elevated temperatures and pressures and possibly localized loads such as jet impingement and associated missile impact. Also included are external pressure loads generated by events inside or outside the containment.
f. Those loads to be sustained, if applicable, after abnormal plant conditions, including flooding of the containment subsequent to a LOCA for fuel recovery.

The various combinations of the above loads that are normally postulated and reviewed include the following: Testing loads; normal operating loads; normal operating loads with severe environmental loads; normal operating loads with severe environmental loads and abnormal loads; normal operating loads with extreme environmental loads and abnormal loads; and post-LOCA flooding loads with severe environmental loads, if applicable. Specific and more detailed information on these combinations are delineated in Section II.3 of this plan.

Unless the steel containment is protected by a shield building, other site-related design loads might also be applicable, including those described in Section I.3 of Standard Review Plan 3.8.1.

4. Design and Analysis Procedures

The design and analysis procedures utilized for the steel containment are reviewed with emphasis on the extent of compliance with Subsection NE of the Code, Section III, Division 1. Particular emphasis is placed on the following subjects:

a. Treatment of non-axisymmetric and localized loads.
b. Treatment of local buckling effects.
c. The computer programs utilized in the design and analysis.

5. Structural Acceptance Criteria

The design limits imposed on the various parameters that serve to quantify the structural behavior of the containment are reviewed, specifically with respect to allowable stresses, strains, and gross deformations, with emphasis on the extent of compliance with Subsection NE of the Code, Section III, Division 1, and with Regulatory

Guide 1.57. For each specified load combination, the proposed allowable limits are compared with the acceptable limits delineated in Section II.5 of this plan. Included in these allowable limits are the following major parameters:

a. Primary stresses, including general membrane, local membrane, and bending plus local membrane stresses.
b. Primary and secondary stresses.
c. Peak stresses.
d. Buckling criteria.

6. Materials, Quality Control, and Special Construction Techniques

Information provided on the materials that are to be used in the construction of the steel containment is reviewed with emphasis on the extent of compliance with Article NE-2000 of Subsection NE of the Code, Section III, Division 1. Among the major materials that are reviewed are the following:

a. Steel plates used as shell components.
b. Structural steel shapes used for stiffeners, beam seats, and crane brackets.

Corrosion and corrosion protection procedures are reviewed by the Materials Engineering Branch (MTEB).

The quality control program that is proposed for the fabrication and construction of the containment is reviewed with emphasis on the extent of compliance with Article NE-5000 of Subsection NE of the Code, Section III, Division 1, including the following:

a. Nondestructive examination of the materials, including tests to determine their physical properties.
b. Welding procedures.
c. Erection tolerances.

Special construction techniques, if proposed, are reviewed on a case-by-case basis to determine their effects on the structural integrity of the completed containment.

7. Testing and In-service Surveillance Program

The preoperational structural test programs for the completed containment and for individual class MC components reviewed, including the objectives of the test, and the acceptance criteria with emphasis on the extent of compliance with Article NE-6000 of Subsection NE of the Code, Section III, Division 1. Structural tests for components such as personnel and equipment locks are also reviewed.

In-service surveillance programs, if any, of components relied upon for containment structural integrity, are reviewed. Any in-service surveillance required in special areas subject to corrosion is reviewed by the Materials Engineering Branch (MTEB).

Special testing and in-service surveillance requirements proposed for new or previously untried design approaches are reviewed.

II. ACCEPTANCE CRITERIA

The acceptance criteria for the areas of review are as follows:

1. Description of the Containment

The descriptive information in the safety analysis report (SAR) is considered acceptable if it meets the minimum requirements set forth in Section 3.8.2.1 of the "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," Revision 2 (Ref. 3).

If the steel containment has new or unique features that are not specifically covered in the "Standard Format...," the reviewer determines that the information necessary to accomplish a meaningful review of the structural aspects of these new or unique features is presented.

2. Applicable Codes, Standards, and Specifications

The design, materials, fabrication, erection, inspection, testing, and in-service surveillance of steel containments are covered by codes, standards, and specifications which are either applicable in their entirety or in part. The following codes and guides are acceptable.

Code	Title
ASME	Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NE, "Class MC Components"
Regulatory Guides	
1.57	Design Limits and Loading Combinations for Metal Primary Reactor Containment System Components

3. Loads and Loading Combinations

Subsection NE of the Code, Section III, Division 1 and Regulatory Guide 1.57 are not explicit with respect to the loads and load combinations which should be considered in the design of steel containments. The specified loads and load combinations are acceptable if found to be in accordance with the following:

a. Loads

- D --- Dead loads.
- L --- Live loads.
- P_t --- Test pressure.
- T_t --- Test temperature.
- T_o --- Thermal effects and loads during startup, normal operating or shutdown conditions, based on the most critical transient or steady-state condition.
- R_o --- Pipe reactions during startup, normal operating or shutdown conditions, based on the most critical transient or steady state-condition.
- E --- Loads generated by the operating basis earthquake.
- E' --- Loads generated by the safe shutdown earthquake.
- P_e --- Design external pressure.
- T_e --- Thermal loads under thermal conditions during event causing external pressure.

R_e --- Pipe reactions under thermal conditions during event causing external pressure.

P_a --- Pressure equivalent static load generated by the postulated design basis accident.

T_a --- Thermal loads under thermal conditions generated by the postulated design basis accident and including T_o.

R_a --- Pipe reactions under thermal conditions generated by the postulated design basis accident and including R_o.

Y_r --- Equivalent static load on the structure generated by the reaction on the broken pipe during the design basis accident.

Y_j --- Jet impingement equivalent static load on the structure generated by the broken pipe during the design basis accident.

Y_m --- Missile impact equivalent static load on the structure generated by or during the design basis accident, such as pipe whipping.

F_L --- Loads generated by the post-LOCA flooding of the containment, if any.

b. Loading Combinations

(1) ---- $D + L + P_t + T_t$

(2) ---- $D + L + T_o + R_o$

(3) ---- $D + L + T_o + R_o + E$

(4) ---- $D + L + T_a + R_a + P_a + E$

(5) ---- $D + L + T_e + R_e + P_e + E$

(6) ---- $D + L + T_a + R_a + P_a + E'$

(7) ---- $D + L + T_e + R_e + P_e + E'$

(8) ---- $D + L + T_a + R_a + P_a + Y_r + Y_j + Y_m + E'$

(9) ---- $D + L + F_L + E$

4. Design and Analysis Procedures

Design and analysis procedures for steel containments are covered in Article NE-3000 of Subsection NE of the Code, Section III, Division 1. The procedures given in the Code, as augmented by the applicable provisions of Regulatory Guide 1.57, constitute an acceptable basis for design and analysis. Moreover, for the specific areas of review described in Section I.4 of this plan, the following criteria are acceptable:

a. Treatment of non-axisymmetric and localized loads

For most containments, the major non-axisymmetric loads which apply are the horizontal seismic loads. Other possible non-axisymmetric and localized loads are those induced by pipe rupture such as reactions, jet impingement forces, and missiles. For the PWR ice-condenser containment, the design basis accident may result in a non-axisymmetric pressure load due to compartmentalization of the containment interior. For such localized loads, the analyses should include a determination of the local effects of the loads. These effects should then be superimposed on the overall effects. For the overall effects of non-axisymmetric loads on shells of revolution, an acceptable general procedure is to expand the load by a Fourier series. Other methods are reviewed on a case-by-case basis for applicability to a large thin shell.

b. Treatment of local buckling effects

Localized pressure loads, such as those encountered in PWR ice-condenser containments, require consideration of local buckling of the shell. An acceptable approach to the problem is to perform a non-linear dynamic analysis. If a static analysis is performed, an appropriate dynamic load factor should be used to obtain the effective static load.

c. Computer programs

The computer programs used in the design and analysis should be described and validated by any of the procedures or criteria described in Section II.4.e of Standard Review Plan 3.8.1.

5. Structural Acceptance Criteria

Stresses at various locations of the shell of the containment for various design loads are determined by analysis. Total stresses for the combination of loads delineated in Section II.3 of this plan are acceptable if found to be within limits defined by various sections of the Code, Section III, Subsection NE, as augmented by Regulatory Guide 1.57. An acceptable interpretation of these limits is contained in Table 3.8.2-1 where the notation is in accordance with the Code.

6. Materials, Quality Control, and Special Construction Techniques

a. The materials of construction are acceptable if in accordance with Article NE-2000 of Subsection NE of the Code, Section III, Division 1. Acceptance criteria for corrosion protection are established by the Materials Engineering Branch (MTEB).

b. Quality control programs are acceptable if in accordance with Articles NE-4000 and NE-5000 of Subsection NE of the Code, Section III, Division 1.

c. Special construction techniques, if any, are reviewed on a case-by-case basis.

7. Testing and In-service Surveillance Requirements

a. Procedures for the preoperational structural proof test are acceptable if found in accordance with Article NE-6000 of Subsection NE of the Code, Section III, Division 1.

b. In-service surveillance requirements for steel containments have not yet been established by the Code and they are currently under development. Acceptance criteria for in-service surveillance programs in areas subject to corrosion are established by the Materials Engineering Branch (MTEB), as required.

III. REVIEW PROCEDURES

The reviewer selects and emphasizes material from the review procedures described below as may be appropriate for a particular case.

1. Description of the Containment

After the type of containment and its functional characteristics are identified, information on similar and previously-licensed applications is obtained for reference. Such information, which is available in safety analysis reports and

TABLE 3.8.2-1

STRESS LIMITS FOR STEEL CONTAINMENTS

Section II.3.b Combination No.		Primary Stresses			Primary & Secondary Stresses	Peak Stresses	Buckling
		Gen. Memb. P_m	Local Memb. P_L	Bend + Local Memb. $P_B + P_L$			Note (3)
(1)		$.9S_y$	$1.25S_y$	$1.25S_y$	$3S_m$	Consider for Fatigue Analysis	125% of Allow. Given by NE-3133
(2) & (3)		S_m	$1.5S_m$	$1.5S_m$	$3S_m$	Consider for Fatigue Analysis	Allow. Given by NE-3133
(4) & (5)		S_m	$1.5S_m$	$1.5S_m$	N/A	N/A	Allow. Given by NE-3133
(6) & (7)	Not integral and continuous	S_m	$1.5S_m$	$1.5S_m$	N/A	N/A	Allow. Given by NE-3133
	Integral and continuous	The Greater of $1.2S_m$ or S_y	The Greater of $1.8S_m$ or $1.5S_y$	The Greater of $1.8S_m$ or $1.5S_y$	N/A	N/A	120% of Allow. Given by NE-3133
(8)	Not integral and continuous	The Greater of $1.2S_m$ or S_y	The Greater of $1.8S_m$ or $1.5S_y$	The Greater of $1.8S_m$ or $1.5S_y$	N/A	N/A	120% of Allow. Given by NE-3133
	Integral and continuous	85% of Stress Intensity Limits of Appendix F			N/A	N/A	85% of Allow. Given by F-1325 of App. F
(9)		$1.5S_m$	The Greater of $1.8S_m$ or $1.5S_y$	The Greater of $1.8S_m$ or $1.5S_y$	N/A	N/A	120% of Allow. Given by NE-3133

NOTES: (1) Thermal stresses need not be considered in computing P_m, P_L and P_B.
(2) Thermal effects are considered in:
(a) Specifying stress intensity limits as a function of temperature.
(b) Analyzing effects of cyclic operation (NB-3222.4).
(3) If a detailed analysis considering inelastic behavior is performed for checking instability (buckling), such an analysis should demonstrate that the applied stress is less than 50% of the critical buckling stress. Designs utilizing vertical stiffeners are permitted. The allowable axial compressive stress may be determined by considering the effects of circumferential stiffener spacing and the effects of water, if present.

amendments of previous license applications, enables identification of differences for the case under review which require additional scrutiny and evaluation. New and unique features that have not been used in the past are of particular interest and are thus examined in greater detail. The information furnished in the SAR is reviewed for completeness in accordance with the "Standard Format...," Revision 2. A decision is then made with regard to the sufficiency of the descriptive information provided. Any additional required information not provided is requested from the applicant at an early stage of the review process.

2. Applicable Codes, Standards, and Specifications

The list of codes, standards, guides, and specifications is checked against the list in Section II.2 of this plan. The reviewer assures himself that the applicable edition and effective addenda are utilized.

3. Loads and Loading Combinations

The reviewer verifies that the loads and load combinations are as conservative as those specified in Section II.3 of this plan. Loading conditions that are unique and that are not specifically covered in Section II.3, are treated on a case-by-case basis. Any deviations from the acceptance criteria for loads and load combinations that have not been adequately justified are identified as unacceptable and transmitted to the applicant for further consideration.

4. Design and Analysis Procedures

The reviewer assures himself that the applicant is committed to the design and analysis procedures delineated in Article NE-3000 of Subsection NE of the Code, Section III, Division 1. Any exceptions to these procedures are reviewed and evaluated on a case-by-case basis. In particular, the areas of review contained in Section I.4 of this plan are evaluated for conformance with the acceptance criteria.

5. Structural Acceptance Criteria

The limits on allowable stresses in the steel shell and its components are reviewed and compared with the acceptable limits specified in Section II.5 of this plan. Where the applicant proposes to exceed some of these limits for some of the load combinations and at some localized points of the structure, the justification provided to show that the structural integrity of the containment will not be affected is reviewed and evaluated. If such justification is unacceptable, the applicant is required to comply with the acceptance criteria delineated in Section II.5 of this plan.

6. Materials, Quality Control, and Special Construction Techniques

The information provided on materials, quality control programs, and special construction techniques, if any, is compared with that referenced in Section II.6 of this plan. If a material not covered by the Code is utilized, the applicant is requested to provide sufficient test and user data to establish the acceptability of the material. Similarly, any new quality control programs or construction techniques

are reviewed and evaluated to assure that there will be no degradation of structural quality that might affect the structural integrity of the containment and its various components.

7. Testing and In-service Surveillance Requirements

The initial structural overpressure test program is reviewed and compared with that indicated as acceptable in Section II.7 of this plan. Any proposed deviations are considered on a case-by-case basis. In-service surveillance programs, if any, as presented in the Technical Specifications of the Operating License, are similarly reviewed.

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information has been provided in accordance with the requirements of this review plan, and concludes that his evaluation is sufficiently complete to support the following type of conclusive statement to be included in the staff's safety evaluation report:

"The criteria used in the analysis, design, and construction of the steel containment structure to account for anticipated loadings and postulated conditions that may be imposed upon the structure during its service lifetime are in conformance with established criteria, codes, standards, and guides acceptable to the Regulatory staff.

"The use of these criteria as defined by applicable codes, standards, and guides; the loads and loading combinations; the design and analysis procedures; the structural acceptance criteria; the materials, quality control programs, and special construction techniques; and the testing and in-service surveillance requirements, provide reasonable assurance that, in the event of earthquakes and various postulated accidents occuring within and outside the containment, the structure will withstand the specified conditions without impairment of structural integrity or safety function. A Category I concrete shield building protects the steel containment from the effects of wind and tornadoes and various postulated accidents occurring outside the shield building. Conformance with these criteria constitutes an acceptable basis for satisfying in part the requirements of General Design Criteria 2, 4, 16, and 50."

V. REFERENCES

1. ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NE, "Class MC Components," American Society of Mechanical Engineers.
2. Regulatory Guide 1.57, "Design Limits and Loading Combinations for Metal Primary Reactor Containment System Components."
3. Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," Revision 2 (in preparation).
4. 10 CFR Part 50, Appendix A, General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena."
5. 10 CFR Part 50, Appendix A, General Design Criterion 4, "Environmental and Missile Design Bases."

6. 10 CFR Part 50, Appendix A, General Design Criterion 16, "Containment Design."
7. 10 CFR Part 50, Appendix A, General Design Criterion 50, "Containment Design Basis."

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OFFICE OF NUCLEAR REACTOR REGULATION

SECTION 3.8.3 CONCRETE AND STEEL INTERNAL STRUCTURES OF STEEL OR CONCRETE CONTAINMENTS

REVIEW RESPONSIBILITIES

Primary - Structural Engineering Branch (SEB)

Secondary - Mechanical Engineering Branch (MEB)
Containment Systems Branch (CSB)

I. AREAS OF REVIEW

The following areas relating to the containment internal structures are reviewed:

1. Description of the Internal Structures

The descriptive information including plans and sections of the various internal structures is reviewed to establish that sufficient information is provided to define the primary structural aspects and elements relied upon to perform the safety-related functions of these structures. The internal structures have several safety-related functions for which their structural integrity is important. By providing support during normal operation and seismic disturbances, they should prevent the occurrence of a loss of coolant accident (LOCA). If such an accident does occur, however, they should act to mitigate its consequences by protecting the containment and other engineered safety features from the effects induced by the accident such as jet forces and whipping pipes.

The major containment internal structures that are reviewed, together with the primary structural function of each structure, and the extent of descriptive information required for each structure, are indicated below. For equipment supports that are not covered by this plan, reference is made to Standard Review Plan 3.9.3.

For PWR Dry Containment Internal Structures

a. Reactor Supports

The PWR vessel should be supported and restrained to resist normal operating loads, seismic loads, and loads induced by postulated pipe rupture including the loss of coolant accident. The support and restraint system should limit the movement of the vessel to within allowable limits under the applicable combinations of loadings.

The support system should nevertheless minimize resistance to the thermal movements expected during operation.

With these functional requirements in mind, the general arrangement and principal features of the reactor vessel linear supports are reviewed with emphasis on methods of transferring loads from the vessel to the support and eventually to the structure and its foundations. Shell-type supports and component standard supports are reviewed by the Mechanical Engineering Branch (MEB). The definition of linear, shell type, and standard supports is in accordance with Subsection NF of the ASME Boiler and Pressure Vessel Code, Section III, Division 1 (Ref. 1). Where uplift supports are utilized, the method of anchoring such supports in the concrete is also reviewed.

b. Steam Generator Supports

Steam generators should be supported and restrained to resist normal operating loads, seismic loads, and loads induced by pipe rupture. The support system should prevent the rupture of the primary coolant pipes due to a postulated rupture in steam or feedwater pipes and vice versa. The system should nevertheless minimize resistance to the thermal movements expected during operation.

With these functional requirements in mind, the general arrangement and principal features of the steam generator linear supports are reviewed with emphasis on methods of transferring loads from the vessel to the support and eventually to the structure and its foundations. Shell-type supports, standard supports, and mechanical restraints such as hydraulic snubbers are reviewed by the Mechanical Engineering Branch (MEB).

c. Reactor Coolant Pump Supports

Reactor coolant pumps should be supported and restrained to prevent excessive deflections during normal operating, seismic, and pipe rupture conditions. Under LOCA loads, the pump should not become a missile and should not generate missiles that might damage other safety-related components. The pump support system should also minimize resistance to thermal movements expected during operation.

With these functional requirements in mind, the general arrangement and principal features of the pump linear supports are reviewed with emphasis on methods of transferring loads from the pump to the support and eventually to the structure and its foundations. Shell-type supports, standard supports, and mechanical restraints such as hydraulic snubbers are reviewed by the Mechanical Engineering Branch (MEB).

d. Primary Shield Wall and Reactor Cavity

The primary shield wall forms the reactor cavity and usually supports and restrains the reactor vessel. It is usually a thick wall that surrounds the reactor vessel and may be anchored through the liner plate to the containment base slab.

The general arrangement and principal features of the wall and cavity are reviewed including the main reinforcement and anchorage system.

e. Secondary Shield Walls

The secondary shield walls surround the primary loops, forming the steam generator compartments, and protecting the containment from the effects of pipe rupture accidents inside the compartment. They may also support intermediate floors and the operating floor. The general arrangement and principal features of these walls are reviewed with emphasis on the method of structural framing and expected behavior under compartment pressure loads and jet forces, particularly those associated with the LOCA.

f. Other Interior Structures

The other major interior structures of PWR dry containments that are reviewed in a similar manner are the pressurizer linear supports, refueling pool walls, the operating floor, other intermediate floors, and the polar crane supporting elements.

For PWR Ice-condenser Containment Internal Structures

In PWR plants where the ice-condenser containment system is utilized, in addition to the applicable structures reviewed in dry PWR containments, the following elements are also reviewed:

a. The Divider Barrier

In the PWR ice-condenser containment system, which utilizes the pressure-suppression concept, the divider barrier surrounds the reactor coolant system. The upper portion of the divider barrier is nearly surrounded by the ice-condenser which is bounded by the containment shell on the outside and by the divider barrier wall on the inside. Several venting doors connect the space inside the divider barrier to the ice-condenser.

In the event of a LOCA, the divider barrier will contain the steam released from the reactor coolant system and, temporarily acting as a pressure-retaining envelope, will channel the steam through the venting doors and into the ice-condenser. The ice will condense the steam and the energy released to the containment will thus be minimized.

Following such a LOCA and before blowdown is completed, the divider barrier will be subjected to a differential pressure and possibly jet forces, and any structural failure in its boundary may result in steam bypassing the ice-condenser and flowing directly into the containment, possibly generating a containment pressure higher than that for which it has been designed.

With this functional requirement in mind, the general arrangement and principal features of the divider barrier are reviewed with emphasis on structural framing and expected behavior when subjected to the design loads.

b. Ice-Condenser

A major feature of the ice-condenser containment is the ice-condenser which contains the baskets of ice forming the heat sink essential for pressure suppression. The structurally significant components of the ice-condenser that are reviewed are the vent doors, ice baskets, brackets, couplings and lattice framings, lower and upper supports, and insulating and cooling panels.

The general arrangment and principal features of these major components are reviewed with emphasis on the structural framing, supports, and expected behavior when subjected to design loads.

For BWR Containment Internal Structures

Since it is expected that future BWR applications will utilize the Mark III containment concept, this Standard Review Plan is oriented towards and based on this type of containment. For other types of BWR containments, modifications to this plan are made on a case-by-case basis.

Among the major Mark III containment internal structures that are reviewed, together with the primary structural function of each structure, and the extent of descriptive information required for each structure, are the following:

a. Drywell

In the BWR Mark III containment system, which utilizes the pressure-suppression concept, the drywell surrounds the reactor coolant system. The lower portion of the drywell is surrounded by the suppression pool which is bounded by the containment shell on the outside and by a weir wall located just inside the drywell wall. Several vent holes connect the drywell to the suppression pool. In the event of a loss-of-coolant accident, the drywell will contain the steam released from the reactor coolant system and, temporarily acting as a pressure-retaining envelope, will channel the steam through the vent holes and into the suppression pool. The pool water will condense the steam and the energy released to the containment will thus be minimized.

Following such a LOCA and before blowdown is completed, the drywell will be subjected to a differential pressure and possibly jet forces, and any structural failure in its boundary would result in steam bypassing the suppression pool and flowing directly into the containment, possibly generating a containment pressure higher than that for which it has been designed.

With this functional requirement in mind, the general arrangement and principal features of the drywell are reviewed with emphasis on structural framing and expected behavior under loads. Since the drywell geometrically resembles, to a certain degree, a containment, the descriptive information reviewed is similar to that reviewed for containments as delineated in Section I.1 of Standard Review Plan 3.8.1. The major components of the drywell that are so reviewed, other than the main body of the drywell, include the bottom vent region, the roof and drywell head, and major penetrations.

b. Weir Wall

The weir wall forms the inner boundary of the suppression pool and is located inside the drywell. It completely surrounds the lower portion of the reactor coolant system. The general arrangement and principal features of the weir wall are reviewed with emphasis on structural framing and behavior under loads.

c. Refueling Pool and Operating Floor

The refueling pool walls are located on top of the drywell. The outer walls form a rectangular pool that is usually subdivided by two interior crosswalls. The base slab of the pool is common to the drywell roof slab. The pool may be filled continuously with water for shielding purposes during operation.

The general arrangement and principal features of the refueling pool are reviewed with emphasis on structural framing and behavior under loads.

The operating floor is intended to provide laydown space for refueling operations and is usually a combination of reinforced concrete and structural steel framing. The containment walls and the refueling pool walls may support the floor.

The general arrangement and principal features of the operating floor are reviewed.

d. Reactor and Recirculation Pump Supports

The support systems of the BWR vessel and recirculation pumps have the same functions as the support systems for PWR vessels and pumps are similarly reviewed.

e. Reactor Pedestal

The reactor pedestal is usually a cylindrical structure located below and supporting the reactor vessel, which is anchored to the top of the pedestal.

The general arrangement and principal features of the reactor pedestal are reviewed with emphasis on structural framing, main reinforcement and the manner in which the pedestal is anchored to the containment base slab.

f. Reactor Shield Wall

This is usually a cylindrical wall surrounding the reactor vessel for radiation shielding purposes. It is supported on the reactor pedestal. The wall may be lined on both surfaces with steel plates which also may act as the main structural components of the wall. The wall may also be utilized as an anchor for pipe restraints.

The general arrangement and principal features of the wall are reviewed with particular emphasis on structure framing and behavior under loads.

g. Other Interior Structures

The other major interior structures constructed of reinforced concrete or structural steel or combinations thereof that are also reviewed in a similar manner are the floors located inside the drywell and in the annulus between the drywell and the containment, and the polar crane supporting elements. The general arrangement and principal features of these structures are reviewed.

2. Applicable Codes, Standards, and Specifications

The information pertaining to design codes, standards, specifications, regulations, general design criteria and regulatory guides, and other industry standards that are applied in the design, fabrication, construction, testing, and surveillance of the containment internal structures, is reviewed. The specific edition, date, or addenda identified for each document are also reviewed.

3. Loads and Loading Combinations

Information pertaining to the applicable design loads and various load combinations thereof is reviewed. The loads normally applicable to containment internal structures include the following:

a. Those loads encountered during normal plant startup, operation, and shutdown, including dead loads, live loads, thermal loads due to operating temperature, and hydrostatic loads such as in refueling and pressure suppression pools.
b. Those loads to be sustained during severe environmental conditions, including those induced by the operating basis earthquake (OBE) specified for the plant site.
c. Those loads to be sustained during extreme environmental conditions, including those induced by the safe shutdown earthquake (SSE) specified for the plant site.
d. Those loads to be sustained during abnormal plant conditions. The most critical abnormal plant condition during which most of the containment internal structures have to perform their primary function is the design basis LOCA. Ruptures of other high-energy pipes should also be considered. Time-dependent and dynamic loads induced by such accidents include elevated temperatures and differential pressures across compartments, jet impingement, impact forces associated with the postulated ruptures of piping, and loads applicable to some structures such as pool swell loads in the BWR Mark III containment and drag forces in the PWR ice-condenser containment.

The various combinations of the above loads that are normally postulated and reviewed include the following: normal operating loads; normal operating loads with severe environmental loads; normal operating loads with extreme environmental loads; normal operating loads with abnormal loads; normal operating loads with severe environmental and abnormal loads; and normal operating with extreme environmental and abnormal loads.

In addition, the following information is reviewed:

a. The extent to which the applicant's criteria comply with the "Building Code Requirements for Reinforced Concrete," ACI 318-71 (Ref. 2) for concrete, and with the AISC "Specification for Design," Fabrication and Erection of Structural Steel for Buildings" (Ref. 3) for steel, as applicable.
b. For concrete portions of the divider barrier of the PWR ice-condenser containment and for concrete portions of the drywell of the Mark III BWR containment, the extent to which the applicant's loading criteria comply with Article CC-3000 of the proposed "Standard Code for Concrete Reactor Vessels and Containments," ACI-ASME (ACI-359) (Ref. 4). For steel pressure-resisting portions of these two structures, the extent to which the applicant's loading criteria comply with Article NE-3000 of Subsection NE of the ASME Code, Section III, Div. 1, (Ref. 5) as augmented by Regulatory Guide 1.57 (Ref. 9).
c. For steel linear supports of the reactor coolant system, the extent to which the applicant's criteria comply with Subsection NF of the ASME Code, Section III, Division 1 (Ref. 1).

4. Design and Analysis Procedures

The design and analysis procedures utilized for the containment internal structures are reviewed with emphasis on the extent of compliance with the applicable codes as indicated in Section I.3, including those applicable to the following areas:

For PWR Dry Containment Internal Structures

a. Reactor Coolant System Supports

The support system for the reactor vessel, steam generators, and reactor coolant pumps, as described in Section I of this plan, should be designed to resist various combinations of loadings, including normal operating loads, seismic loads, and loss of coolant and other pipe rupture accident loads.

Analytical procedures for determining normal operating loads and accident loads are reviewed by the Mechanical Engineering Branch (MEB).

Analytical procedures for determining seismic loads are as described in Standard Review Plan 3.7.3.

After the procedures for determining individual loads and combinations thereof are so reviewed, the design and analysis methods utilized for the linear supports are reviewed including the type of analysis (elastic or plastic), the methods of load transfer, and the assumptions on boundary conditions. Specifically, the extent of compliance with design and analysis procedures delineated in Subsection NF of the ASME Section III Code, Division 1 (Ref. 1), is reviewed.

b. Primary Shield Wall and Reactor Cavity

The primary shield wall should withstand all the applicable loads including those transmitted through the reactor supports. It is subjected to most of the loads described in Section I.3 of this plan and should be designed and analyzed for all the applicable load combinations. During normal plant operation, a thermal gradient across the wall is generated by the attenuation heat of gamma and neutron radiation originating from the reactor core. Insulation and cooling systems may be provided to reduce the severity of this gradient by limiting the rise in temperature to an acceptable level.

Procedures for determining seismic loads on the primary shield wall are reviewed in accordance with Standard Review Plan 3.7.2.

Loss of coolant accident loads that are applicable to the primary shield wall include a differential pressure created across the reactor cavity by a pipe break in the vicinity of the reactor nozzles. Such a transient pressure may act on the entire cavity or on portions thereof. Procedures for determining such pressures are reviewed by the Containment Systems Branch (CSB).

Other loss of coolant accident loads that apply are those transmitted to the wall through the reactor supports including pipe rupture reaction forces which may induce simultaneous shear forces, torsional moments, and bending moments at the base of the wall. Further, the elevated temperature within and around the primary shield

created by the accident may produce transient thermal gradients across the thick wall. Design and analysis procedures for such accident effects are accordingly reviewed.

c. Secondary Shield Walls

The secondary shield walls surrounding the primary loops and supporting the operating floor should be designed for loads similar to those applicable to the primary shield wall including loads of fluid jets from a postulated break of a primary pipe which can impinge on these walls. The analytical techniques utilized for these walls are reviewed including their structural framing and behavior under loads. Where elasto-plastic behavior is assumed and the ductility of the walls is relied upon to absorb the energy associated with jet loads, the procedures and assumptions are reviewed with particular emphasis on such areas as modeling techniques, boundary conditions, force-time functions, and assumed ductility. For the time-dependent differential pressure, however, elastic behavior is required and the methods of determining an equivalent static load are accordingly reviewed.

d. Other Interior Structures

Most of the other interior structures that are also reviewed are combinations of slabs, walls, beams and columns, classified as Category I structures and subject to most of the loads and combinations described in Section I.3 of this plan. Analytical techniques for these structures are reviewed on the same basis as for the structures described above.

For PWR Ice-Condenser Containment Internal Structures

a. Divider Barrier

Since the divider barrier has to maintain a certain degree of leak-tightness during a LOCA and is thus a critical structure with respect to the proper functioning of the containment, it is treated on the same basis as the containment.

The loads that usually govern the design of the divider barrier are those induced by the LOCA, including the time-dependent differential pressure across the barrier and any concurrent concentrated jet impingement loads. As the divider barrier is typically a combination of walls and slabs framed together, the design and analysis procedures are of the conventional type. They are accordingly reviewed with emphasis on the assumed boundary conditions and behavior under loads. Since the differential pressure and jet impingement loadings are dynamic impulsive loads that vary with time, the techniques utilized to determine their equivalent static loads are reviewed.

b. Ice-Condenser

The design of the ice-condenser and its various components may be based on a combination of analysis and testing. The analytical and testing procedures that are reviewed include those for the ice baskets and brackets (couplings); the lattice frames and columns including attachments; the supporting structures comprising the lower supports; the wall panels and cooling duct and supports of various auxiliary components.

The ice-condenser and its components should be analyzed or tested for various loads and combinations thereof including dead and live loads, thermal loads induced by differential thermal expansion within the various elements, seismic loads and loads induced by the loss-of-coolant accident. Accident loads include pressure differential drag loads and loads induced by the change of momentum of the flowing steam.

Elastic analysis is usually utilized for the ice-condenser and its components. However, plastic analysis may also be used as an alternate. Accordingly, the load factors that are applied to each of the applicable loads and the basis and justification of these load factors are reviewed.

Where experimental verification of the design using simulated load conditions is used, the procedures used to account for similitude relationships which exist between the actual component and the test model are reviewed to assure that the results obtained from the test are a conservative representation of the load carrying capability of the actual component under the postulated loading.

For BWR Containment Internal Structures

a. Drywell

The drywell, which has to maintain a certain degree of leak-tightness during a LOCA, is critical with respect to the proper functioning of the containment. Accordingly, and since it geometrically resembles a containment, the design and analysis procedures utilized for the drywell are reviewed on a basis similar to those of containments as described in Section I.4 of Standard Review Plans 3.8.1 and 3.8.2 for concrete and steel portions, respectively.

b. Weir Wall

One of the major loads to which the weir wall may be subjected is a jet impingement load induced by a pipe rupture in a nearby recirculation loop. Under such a concentrated load, the weir wall should not deform to an extent that might impair or degrade the pressure-suppression performance. Accordingly, the procedures utilized to analyze the wall for such dynamic time-dependent loads are reviewed with particular emphasis on modeling techniques, assumptions on boundary conditions, and behavior under loads.

c. Refueling Pool and Operating Floor

In the BWR Mark III containments reviewed recently, the refueling pool is continuously filled with water to provide biological shielding above the reactor. The operating floor, which may be supported on the walls of the refueling pool on one side and on the containment shell on the other side, is a combination of reinforced concrete and structural steel. The design and analysis procedures for the refueling pool and the operating floor are of the conventional type and are accordingly reviewed, with particular emphasis on the structural framing and behavior under loads. In cases where the floor beams are supported vertically on the containment shell, they should be laterally isolated to minimize interaction between the containment and its interior.

d. Reactor and Recirculation Pump Supports

The design and analysis procedures utilized for the reactor and recirculation pump supports are reviewed in a similar manner to that for PWR reactor and pump supports, as already described in this plan.

e. Reactor Pedestal

The reactor pedestal supports the reactor and has to withstand the loads transmitted through the reactor supports. It is thus subjected to most of the loads described in Section I.3 of this plan and is designed and analyzed for all the applicable load combinations.

Because of the similarity in geometry and function of the BWR reactor pedestal to the PWR primary shield wall, the design and analysis procedures are similar and are reviewed accordingly as has already been discussed in this plan.

f. Reactor Shield Wall

This cylindrical wall, which surrounds the reactor and provides biological shielding, is also subjected to most of the loads described in Section I.3 of this plan. In most cases, the wall is utilized to anchor pipe restraints placed around the reactor coolant system piping. Moreover, a pipe rupture in the vicinity of the reactor nozzles may pressurize the space within the wall. The wall is usually lined on both faces with steel plates which may constitute the major structural elements relied upon to resist the design loads.

The analytical and design techniques utilized to determine the effect of the design loads on the wall are reviewed with particular emphasis on the assumed boundary conditions and the behavior of the wall under loads.

g. Other Interior Structures

There are several platforms within the BWR Mark III containment some of which are inside the drywell and the others outside in the annulus between the drywell and the containment. Platforms inside the drywell are usually of structural steel and their main structural function is to provide foundations for the pipe restraints inside the drywell. Platforms outside the drywell are usually combinations of steel and concrete and have to be designed to resist the various applicable loads particularly the effects of pool swell during a loss-of-coolant accident. The analytical procedures for determining pool swell loads are reviewed by the Containment Systems Branch (CSB). Design and analysis procedures for these platforms are reviewed with particular emphasis on the framing and structural behavior under loads.

5. Structural Acceptance Criteria

The design limits imposed on the various parameters that serve to quantify the structural behavior of the various interior structures of the containment are reviewed, specifically with respect to stresses, strains, deformations, and factors of safety against structural failure, with emphasis on the extent of compliance with the applicable codes as indicated in Section I.3 of this plan.

6. Materials, Quality Control, and Special Construction Techniques

Information provided on the materials that are used in the construction of the containment internal structures is reviewed. Among the major materials of construction that are reviewed are the concrete ingredients, reinforcing bars and splices, and structural steel and various supports and anchors.

The quality control program that is proposed for the fabrication and construction of the containment interior structures is reviewed including nondestructive examination of the materials to determine physical properties, placement of concrete, and erection tolerances.

Special, new, or unique construction techniques, if proposed, are reviewed on a case-by-case basis to determine their effects on the structural integrity of the completed interior structure.

In addition, the following information should be provided:

a. The extent to which the materials and quality control programs comply with the "Building Code Requirements for Reinforced Concrete," ACI 318-71 (Ref. 2), for concrete, and with the AISC "Specifications for Design, Fabrication and Erection of Structural Steel for Buildings,"(Ref. 3), for steel, as applicable.
b. For steel linear supports of the reactor coolant system, the extent to which the material and quality control programs comply with Subsection NF of the ASME Code, Section III, Division 1 (Ref. 1).
c. For quality control in general, the extent to which the applicant complies with ANSI N45.2.5 (Ref. 7).
d. If welding of reinforcing bars is proposed, the extent to which the applicant complies with the applicable sections of the proposed "Standard Code for Concrete Reactor Vessels and Containments," ACI-ASME (ACI-359) (Ref. 4), should be described and any exceptions taken should be justified.

7. Testing and Inservice Surveillance Programs

If applicable, any post-construction testing and in-service surveillance programs are reviewed on a case-by-case basis.

The structural test for the drywell of the BWR Mark III containment is reviewed in a similar manner to that of the containment.

II. ACCEPTANCE CRITERIA

The acceptance criteria for the areas of review are as follows:

1. Description of the Internal Structures

The descriptive information in the SAR is considered acceptable if it meets the minimum requirements set forth in Section 3.8.3.1 of the "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," Revision 2 (Ref. 8).

Deficient areas of descriptive information are identified by the reviewer and a request for additional information is initiated at the application acceptance review. New or unique design features that are not specifically covered in the "Standard Format" may require a more detailed review. The reviewer determines if additional information is required to accomplish a meaningful review of the structural aspects of such new or unique features.

2. Applicable Codes, Standards, and Specifications

The design, materials, fabrication, erection, inspection, testing, and in-service surveillance, if any, of interior structures of containments are covered by codes, standards, and guides that are either applicable in their entirety or in portions thereof. The following codes, standards, specifications, and guides are acceptable.

Code, Standard, or Specification	Title
ACI 318-71	Building Code Requirements for Reinforced Concrete
ACI/ASME (ACI-359)	Proposed Standard Code for Concrete Reactor Vessels and Containments, ASME Boiler and Pressure Vessel Code, Section III, Division 2
ASME	Boiler and Pressure Vessel Code, Section III, Subsections NE and NF
AISC	Specification for the Design, Fabrication and Erection of Structural Steel for Buildings
ANSI N45.2.5	Supplementary Quality Assurance Requirements for Installation, Inspection and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants
Regulatory Guides	
1.10	Mechanical (Cadweld) Splices in Reinforcing Bars of Category I Concrete Structures
1.15	Testing of Reinforcing Bars for Category I Concrete Structures
1.55	Concrete Placement in Category I Structures

3. Loads and Load Combinations

With the exception of the divider-barrier and ice-condenser elements of the ice-condenser PWR containment, the drywell of the BWR Mark III containment, and the steel linear supports of the reactor coolant system, the loads and load combinations for all other containment interior structures described in Section I.1 of this plan, are acceptable if found in accordance with the following:

Loads, Definitions, and Nomenclature

All the major loads to be encountered or to be postulated are listed below. All the loads listed, however, are not necessarily applicable to all the interior structures. Loads and the applicable load combinations for which each structure has to be designed will depend on the conditions to which that particular structure could be subjected.

Normal loads, which are those loads to be encountered during normal plant operation and shutdown, include:

D --- Dead loads or their related internal moments and forces, including any permanent equipment loads and hydrostatic loads. For equipment supports, it also includes static and dynamic head and fluid flow effects.

L --- Live loads or their related internal moments and forces, including any movable equipment loads and other loads which vary with intensity and occurrence. For equipment supports, it also includes loads due to vibration and any support movement effects.

T_o --- Thermal effects and loads during normal operating or shutdown conditions, based on the most critical transient or steady state condition.

R_o --- Pipe reactions during normal operating or shutdown conditions, based on the most critical transient or steady state condition.

Severe environmental loads include:

E --- Loads generated by the operating basis earthquake.

Extreme environmental loads include:

E' --- Loads generated by the safe shutdown earthquake.

Abnormal loads, which are those loads generated by a postulated high-energy pipe break accident, include:

P_a --- Pressure equivalent static load within or across a compartment generated by the postulated break, and including an appropriate dynamic load factor to account for the dynamic nature of the load.

T_a --- Thermal loads under thermal conditions generated by the postulated break and including T_o.

R_a --- Pipe reactions under thermal conditions generated by the postulated break and including R_o.

Y_r --- Equivalent static load on the structure generated by the reaction on the broken high-energy pipe during the postulated break, and including an appropriate dynamic load factor to account for the dynamic nature of the load.

Y_j --- Jet impingement equivalent static load on a structure generated by the postulated break, and including an appropriate dynamic load factor to account for the dynamic nature of the load.

Y_m --- Missile impact equivalent static load on a structure generated by or during the postulated break, as from pipe whipping, and including an appropriate dynamic load factor to account for the dynamic nature of the load.

In determining an appropriate equivalent static load for Y_r, Y_j, and Y_m, elasto-plastic behavior may be assumed with appropriate ductility ratios, provided excessive deflections will not result in loss of function of any safety-related system.

Load Combinations for Concrete Structures

For concrete interior structures, the load combinations are acceptable if found in accordance with the following:

a. For service load conditions, either the working stress design (WSD) method or the strenght design method may be used.

(i) If the WSD method is used, the following load combinations should be considered:

(1) $D + L$

(2) $D + L + E$

If thermal stresses due to T_o and R_o are present, the following combinations should be also considered:

(1a) $D + L + T + R_o$

(2a) $D + L + T_o + R_o + E$

Both cases of L having its full value or being completely absent should be checked.

(ii) If the strength design method is used, the following load combinations should be considered:

(1) $1.4D + 1.7L$

(2) $1.4D + 1.7L + 1.9E$

If thermal stresses due to T_o and R_o are present, the following combinations should also be considered:

(1b) $(0.75)\ (1.4D + 1.7L + 1.7T_o + 1.7R_o)$

(2b) $(0.75)\ (1.4D + 1.7L + 1.9E + 1.7T_o + 1.7R_o)$

b. For factored load conditions, which represent extreme environmental, abnormal, abnormal/severe environmental, and abnormal/extreme environmental conditions, the strength design method should be used and the following load combinations should be considered:

(3) $D + L + T_o + R_o + E'$

(4) $D + L + T_a + R_a + 1.5\ P_a$

(5) $D + L + T_a + R_a + 1.25\ P_a + 1.0\ (Y_r + Y_j + Y_m) + 1.25E$

(6) $D + L + T_a + R_a + 1.0\ P_a + 1.0\ (Y_r + Y_j + Y_m) + 1.0\ E'$

In combinations (4), (5), and (6), the m imum values of P_a, T_a, R_a, Y_j, Y_r, and Y_m, including an appropriate dynamic load factor, should be used unless a time-history analysis is performed to justify otherwise. Combinations (5) and (6) and the corresponding structural acceptance criteria of Section II.5 of this plan should first be satisfied without Y_r, Y_j, and Y_m. When considering these loads, local section strength capacities may be exceeded under these concentrated loads, provided there will be no loss of function of any safety-related system.

Both cases of L having its full value or being completely absent should be checked.

Load Combinations for Steel Structures

For steel interior structures, the load combinations are acceptable if found in accordance with the following:

a. For service load conditions, either the elastic working stress design methods for Part 1 of AISC, or the plastic design methods of Part 2 of AISC, may be used.

(i) If the elastic working stress design methods are used:

(1) $D + L$

(2) $D + L + E$

If thermal stresses due to T_o and R_o are present, the following combinations should also be considered:

(1a) $D + L + T_o + R_o$

(2a) $D + L + T_o + R_o + E$

(ii) If the plastic design methods are used:

(1) $1.7D + 1.7L$

(2) $1.7D + 1.7L + 1.7E$

If thermal stresses due to T_o and R_o are present, the following combinations should also be considered:

(1b) $1.3\ (D + L + T_o + R_o)$

(2b) $1.3\ (D + L + E + T_o + R_o)$

b. For factored load conditions, the following load combinations should be considered:

(i) If the elastic working stress design methods are used:

(3) $D + L + T_o + R_o + E'$

(4) $D + L + T_a + R_a + P_a$

(5) $D + L + T_a + R_a + P_a + 1.0\ (Y_j + Y_r + Y_m) + E$

(6) $D + L + T_a + R_a + P_a + 1.0\ (Y_j + Y_r + Y_m) + E'$

(ii) If the plastic design methods are used:

(3) $D + L + T_o + R_o + E'$

(4) $D + L + T_a + R_a + 1.5\ P_a$

(5) $D + L + T_a + R_a + 1.25\ P_a + 1.0\ (Y_j + Y_r + Y_m) + 1.25\ E$

(6) $D + L + T_a + R_a + 1.0\ P_a + 1.0\ (Y_j + Y_r + Y_m) + 1.0\ E'$

In the above combinations, thermal loads can be neglected when it can be shown that they are secondary and self-limiting in nature.

In combinations (4), (5), and (6), the maximum values of P_a, T_a, R_a, Y_j, Y_r, and Y_m, including an appropriate dynamic load factor, should be used unless a time-history analysis is performed to justify otherwise. Combinations (5) and (6) and the corresponding structural acceptance criteria of Section II.5 of this plan should first be satisfied without Y_r, Y_j, and Y_m. When considering these loads, however, local section strengths or stresses may be exceeded under these concentrated loads, provided there will be no loss of function of any safety-related system.

For the divider barrier, ice-condenser elements, the Mark III containment drywell, and for the steel linear supports of the reactor coolant system, the loading criteria are acceptable if found in accordance with the following:

a. <u>Divider barrier</u>

As the structural integrity of the divider barrier and, to a certain extent, its leak-tight integrity as well, are important to the proper functioning of the ice-condenser containment system, it is treated for design purposes similar to the containment itself.

Accordingly, for concrete pressure-resisting portions of the divider barrier, the loads and load combinations of Article CC-3000 of ACI-359 (Ref. 4) will apply, with the following exceptions.

For Table CC-3200-1

(i) Jet impingement loads, Y_j, and impact loads of missiles associated with the loss-of-coolant accident, Y_m, should be included.

(ii) The 6th combination, representing abnormal conditions, need not include Y_r in combination with 1.5 P_a,

(iii) In the 7th, 8th, and 9th combinations, representing abnormal/severe environmental and abnormal/extreme environmental load conditions, the "and/or" between R_a and Y_r should be deleted and, in addition to R_a and Y_r, the combinations should include Y_j and Y_m.

(iv) It should be indicated that the maximum values of P_a, T_a, R_a, Y_r, Y_j, and Y_m, including an appropriate dynamic load factor, should be applied simultaneously, unless a time-history analysis is performed to justify otherwise.

Steel portions of the divider barrier which resist the design differential pressure and are not backed by concrete, such as penetrations, hatches, locks and guard pipes, should be designed in accordance with the appropriate sections of Subsection NE of the ASME Code, Section III, Division 1, (Ref. 5) together with the applicable loads, load combinations, and acceptance criteria of Regulatory Guide 1.57, (Ref. 9). Specifically, the load combinations of Section II.3 of Standard Review Plan 3.8.2 apply.

b. Ice-condenser Elements

In the ice-condenser containment system the structural integrity of the ice baskets, ice bed framing, and their supports, is important to the functional integrity of the containment system. The major loads that are applicable to the ice-condenser elements are: D, L, E, E', and P_a. For this structure, P_a is the LOCA pressure load induced by drag and change in momentum of the flowing air and steam. Load combinations for the ice-condenser elements are acceptable if found in accordance with the following:

(i) For service load conditions, if elastic working stress design methods are used:

(1) D + L

(2) D + L + E

(ii) For service load conditions, if plastic design methods are used:

(1) 1.7 D + 1.7 L

(2) 1.7 D + 1.7 L + 1.7 E

(iii) For service load conditions, if an experimental test verification of the design is used:

(1) 1.9 D + 1.9 L

(2) 1.9 D + 1.9 L + 1.9 E

If thermal stresses are significant and have to be considered, an acceptable procedure for accounting for such thermal loads is contained in item (a) of Subarticle NF-3231.1 of Subsection NF of the ASME Code, Section III, Division 1 (Ref. 1).

(iv) For factored load conditions, if elastic working stress design methods are used:

(3) $D + L + E'$

(4) $D + L + P_a$

(5) $D + L + P_a + E'$

(v) For factored load conditions, if plastic design methods are used:

(3) $1.3\ D + 1.3\ L + 1.3\ E'$

(4) $1.3\ D + 1.3\ L + 1.3\ P_a$

(5) $1.2\ D + 1.2\ L + 1.2\ P_a + 1.2\ E'$

(vi) For factored load conditions, if an experimental test verification of the design is used:

(3) $1.4\ D + 1.4\ L + 1.4\ E'$

(4) $1.4\ D + 1.4\ L + 1.4\ P_a$

(5) $1.3\ D + 1.3\ L + 1.3\ P_a + 1.3\ E'$

c. BWR Mark III Containment Drywell

As the structural integrity of the drywell and, to a certain extent, its leak-tight integrity as well, are critically important to the proper functioning of the Mark III pressure-suppression system, the drywell is treated, for design and testing purposes only, similar to the containment itself.

Accordingly, for concrete pressure-resisting portions of the drywell, the loads and loading combinations of Article CC-3000 of ACI-359 (Ref. 4) will apply, with the exceptions listed for concrete portions of the PWR ice-condenser divider barrier.

For steel components of the drywell that resist pressure and are not backed by concrete, such as the drywell head, the appropriate sections of Subsection NE of the ASME Code, Section III, Division 1, (Ref. 5) should be used together with the applicable loads, load combinations, and acceptance criteria of Regulatory Guide 1.57 (Ref. 9). Specifically, the load combinations of Section II.3 of Standard Review Plan 3.8.2 apply.

For the lower vent portion of the drywell:

(i) If the main reinforcement of the drywell is carried down between the vent holes and the reinforced concrete section is relied upon for structural purposes, the criteria that apply to concrete portions of the drywell as described above will apply.

(ii) If the main reinforcement of the drywell is terminated above the vent holes and two steel plates lining both faces of the drywell are alone utilized for structural purposes, the criteria that apply to steel portions of the drywell as described above will apply.

(iii) If other structural systems are used in the vent region, the loads and load combinations are reviewed and judged on a case-by-case basis.

d. Reactor Coolant System Supports

Steel linear supports for the reactor vessel, steam generators, reactor coolant pumps, and recirculation pumps, as described in Section I of this plan, are governed by Subsection NF of the ASME Code, Section III, Division 1. This Code does not explicitly delineate load combinations for the design of these supports. Accordingly, the following combinations should be satisfied as a minimum:

Load Combinations

(1) If the elastic method of analysis of paragraph NF-3231.1 of Subsection NF of the ASME Code, Section III, Division 1, is used, the following combinations should be satisfied as a minimum:

(i) $D + L + E$

(ii) $D + L + E' + P_a + Y_r + Y_j + Y_m$

In addition, the conditions of item (a) of paragraph NF-3231.1 shall be satisfied.

(2) If the limit method of analysis of paragraph NF-3231.2 of Subsection NF is used, the following combinations should be satisfied as a minimum:

(i) $1.7\ (D + L + E)$

(ii) $1.0\ (D + L + E' + P_a + Y_r + Y_j + Y_m)$

4. Design and Analysis Procedures

The design and analysis procedures utilized for the interior structures of the containment are acceptable if found in accordance with the following:

For PWR Dry Containment Internal Structures

a. Reactor Coolant System Supports

The linear support systems for the reactor vessel, steam generators, and reactor coolant pumps, as described in Section I of this plan, should be analyzed for and designed to resist various combinations of loadings as indicated in Section II.3 of this plan. Design and analysis procedures for such supports are acceptable if in accordance with Subsection NF of the ASME Section III Code, Division 1, (Ref. 1), particularly with Appendix XIII.

b. Primary Shield Wall and Reactor Cavity

The design and analysis procedures utilized for the shield wall are acceptable if in accordance with the ACI 318-71 Code (Ref. 2). This code is mostly based on the strength design method. However, the use of Section 8.10 of the Code, which is based on the working stress design method where actual elastic/linear stresses in the concrete and reinforcement are determined and compared with their corresponding allowables, is considered acceptable.

Analyses for loss-of-coolant accident loads applicable to the primary shield wall, such as for the cavity differential pressure combined with pipe rupture reaction forces, are acceptable if these loads are treated as dynamic time-dependent loads whereby either a detailed time-history analysis is performed, or a static analysis utilizing the peak of the forcing function amplified by an appropriately chosen dynamic load factor is utilized. Elastic behavior of the wall should be maintained under the differential pressure. However, for the concentrated accident loads such as Y_r and Y_j, elasto-plastic behavior may be assumed as long as the deflections are limited to maintain functional requirements. Simplified methods for determining effective dynamic load factors for elastic behavior are acceptable if in accordance with recognized dynamic analysis methods.

c. Secondary Shield Walls

Design and analysis procedures utilized for the secondary shield walls are acceptable if in accordance with conventional beam/slab design and analysis procedures described in the ACI 318-71 Code.

Similar to the primary shield wall, the secondary shield walls are also subject to dynamic loss-of-coolant accident loads and the same methods described in paragraph b. above are, therefore, applicable and acceptable.

d. Other Interior Structures

Most of the other interior structures that are reviewed are combinations of reinforced concrete and steel slabs, walls, beams, and columns, which are classified as Category I structures subject to the loads and load combinations described in Section II.3 of this plan. Analytical techniques for these structures are acceptable if found in accordance with those described in the ACI 318-71 Code for concrete and with those in the AISC specifications for steel.

For PWR Ice-condenser Containment Internal Structures

a. Divider Barrier

The most important loads that usually govern the design of the divider barrier are those induced by the loss-of-coolant accident, including the differential pressure across the barrier and any concentrated jet impingement loads. As the divider barrier is a combination of walls and slabs framed together, the design and analysis procedures are acceptable if in accordance with those contained in Section 8.10 of the ACI 318-71 Code for the concrete portions of the divider barrier. These methods are based on the elastic/linear working stress design method where actual stresses are determined.

For steel portions of the divider barrier that resist pressure but are not backed by structural concrete, the design and analysis procedures are acceptable if found in accordance with the applicable provisions of Subsection NE of the ASME Code, Section III, Division 1.

b. Ice-condenser Elements

The design and analysis procedures for the ice-condenser and its various components are acceptable if in accordance with either the elastic/linear design method of Part 1 of the AISC Specifications or with the plastic design method of Part 2 of the same Specifications. For components where experimental testing is utilized to verify the design, the testing procedures are acceptable if in accordance with recognized prototype or model testing procedures where the effect of scaling and similitude are taken into consideration.

For BWR Containment Internal Structures

a. Drywell

The design and analysis procedures utilized for concrete portions of the drywell are acceptable if in accordance with Section II.4 of Standard Review Plan 3.8.1. For steel portions of the drywell that resist pressure but are not backed by structural concrete, the design and analysis procedures are acceptable if found in

accordance with the applicable provisions of Subsection NE of the ASME Code, Section III, Division 1.

b. Weir Wall

One of the major loads to which the weir wall may be subjected is a jet impingement load induced by a pipe rupture in a nearby recirculation loop. The deflection of the wall under such a load must be limited so as not to impair the pressure-suppression performance. The procedures utilized to analyze the wall for such a dynamic time-dependent load are acceptable if a detailed time-history dynamic analysis is performed or if an equivalent static analysis is performed utilizing the peak of the jet load amplified by an appropriately chosen dynamic load factor.

c. Refueling Pool and Operating Floor

The refueling pool and the operating floor, which may be supported on the walls of the refueling pool on one side and on the containment shell on the other side, are a combination of reinforced concrete and structural steel. The design and analysis procedures are acceptable if found in accordance with conventional methods described in the ACI 318-71 Code for concrete and in the AISC Specifications for structural steel.

d. Reactor Supports

The linear support system for the reactor vessel, described in Section I of this plan, should be designed to resist various combinations of loadings as indicated in Section II.3 of this plan. Among the major loads that should be considered are normal operating loads, seismic loads, and loss-of-coolant accident loads.

Design and analysis procedures are acceptable if in accordance with those delineated in Subsection NF of the ASME Section III Code, Division 1, particularly with Appendix XVII.

e. Reactor Pedestal

The reactor pedestal, which supports the reactor and has to withstand the loads transmitted through the reactor supports, should be subjected to most of the loads described in Section II.3 and should be designed for all the applicable load combinations.

The design and analysis procedures are acceptable if found to be similar to those referenced for the primary shield wall of PWR containments in paragraph (b) under PWR dry containments.

f. Reactor Shield Wall

This cylindrical wall, which surrounds the reactor and provides biological shielding, should be subjected to most of the loads described in Section II.3 of this plan. In most cases, the wall is utilized to anchor most of the pipe restraints placed around the reactor coolant system piping. A pipe rupture in the vicinity of the reactor nozzles may pressurize the space within the wall. The wall may be lined on both faces with steel plates which may constitute the major structural elements relied upon to resist the design loads.

Similar to the reactor pedestal, the biological shield wall is also subjected to dynamic loss-of-coolant accident loads and the same methods are, therefore, applicable and acceptable.

g. Miscellaneous Platforms

Platforms inside the drywell are usually of structural steel and their main structural function is to provide foundations for the pipe restraints inside the drywell. Platforms outside the drywell are usually combinations of steel and concrete. The analytical and design procedures for these platforms are acceptable if in accordance with the ACI 318-71 Code for reinforced concrete, and with the AISC Specifications for structural steel. Of particular interest are the dynamic loads induced on these floors by pool swell during a LOCA.

Computer programs used in the design and analysis of containment interior structure should be described and validated by any of the procedures described in Section II.4.e of Standard Review Plan 3.8.1.

5. Structural Acceptance Criteria

With the exception of the divider barrier and ice-condenser elements of the ice-condenser PWR containment, the drywell of the BWR Mark III containment, and the steel linear supports of the reactor coolant system, the structural acceptance criteria for all other interior structures of the containment described in Section I.1 of this plan are acceptable if found in accordance with the following:

For each of the loading combinations delineated in the beginning of Section II.3 of this plan, the following defines the allowable limits which constitute the structural acceptance criteria:

In Combinations for Concrete Internal Structures	Limit
(a)(i) 1, 2	S(1)
(a)(i) 1a, 2a	1.3 S
(a)(ii) 1, 2	U(2)
(a)(ii) 1b, 2b	U
(b) 3, 4, 5, 6	U

In Combinations for Steel Internal Structures	Limit
(a)(i) 1, 2	S
(a)(i) 1a, 2a	1.5 S
(a)(ii) 1, 2	Y(3)
(a)(ii) 1b, 2b	Y
(b)(i) 3, 4, 5(4)	1.6 S
(b)(i) 6(4)	1.7 S
(b)(ii) 3, 4, 5, 6	.9 Y

Notes

(1) S --- For concrete structures, S is the required section strength based on the working stress design method and the allowable stresses defined in Section 8.10 of ACI 318-71.

For structural steel, S is the required section strength based on the elastic design methods and the allowable stresses defined in Part 1 of the AISC "Specification for the Design, Fabrication and Erection of Structural Steel for Buildings," February 12, 1969.

The 33% increase in allowable stresses for concrete and steel due to seismic loadings is not permitted.

(2) U --- For concrete structures, U is the section strength required to resist design loads based on the strength design methods described in ACI 318-71.

(3) Y --- For structural steel, Y is the section strength required to resist design loads and based on plastic design methods described in Part 2 of the AISC "Specification for the Design, Fabrication and Erection of Structural Steel for Buildings," February 12, 1969.

(4) --- For these two combinations, in computing the required section strength, S, the plastic section modulus of steel shapes may be used.

For the divider barrier, ice-condenser elements, the drywell, and the linear steel supports of the reactor coolant system, the structural acceptance criteria are acceptable if found in accordance with the following:

a. Divider barrier

(i) For concrete portions of the divider barrier, the specified limits for stresses and strains are acceptable if found in accordance with Subsection CC-3400 of the ACI-359 Code, but with the following exceptions:

CC-3421.1

- The footnote on page 196 should be revised to indicate that the 33-1/3% increase in allowable stresses is permitted only for temperature loads and not for seismic loads.

CC-3422.1

- Item (c) should be deleted.

CC-3422.2

- The footnote on page 197 should be deleted.

(ii) For steel portions of the divider barrier which resist the design differential pressure and are not backed by concrete, the design should be similar to that of steel containments. Accordingly, the load combinations and stress limits of Section II.3 of Standard Review Plan 3.8.2 apply.

b. Ice-condenser Elements

For load combinations delineated in Section II.3 of this plan for the ice-condenser elements, the stress limits are acceptable if found in accordance with the following:

For Combinations:		Limit
(i)	(1), (2)	S(1)
(ii)	(1), (2)	Y(2)
(iii)	(1)	C(3)
(iv)	(3), (4)	1.3S
(iv)	(5)	1.6S
(v)	(3), (4), (5)	Y
(vi)	(3), (4), (5)	C

Notes

(1) S --- As defined in "Notes" under first tables in II.5 above.

(2) Y --- As defined in "Notes" under first tables in II.5 above.

(3) C --- Where experimental testing is used for verification of the design, C shall be the ultimate load carrying capacity of the member. Size effects and any similitude relationship which may exist between the actual component and the test model shall be accounted for in the evaluation of C.

c. BWR Mark III Containment Drywell

(i) For concrete portions of the drywell, the acceptance criteria of paragraph (a)(i) as described for the divider barrier apply.

(ii) For steel portions of the drywell that resist pressure and are not backed by structural concrete, the acceptance criteria of paragraph (a)(ii) as described for the divider barrier apply.

(iii) For the lower vent portion of the drywell:

- If the main reinforcement of the drywell is carried down between the vent holes and the reinforced concrete section is relied upon for structural purposes, the structural acceptance criteria is the same as for (i) above.
- If the main reinforcement of the drywell is terminated above the vent holes and two steel plates lining both faces of the wall are utilized for structural purposes, the acceptance criteria for (ii) above will apply.
- If other structural systems are used in the vent region, the acceptance criteria are reviewed on a case-by-case basis.

d. Reactor Coolant System Supports

The structural acceptance criteria for the steel linear supports of the reactor coolant system are acceptable if found in accordance with the following:

For load combinations delineated in paragraph (d) of Section II.3 of this plan for the reactor coolant system linear supports, the following acceptance criteria will apply:

(1) If the elastic analysis method is used:

Combination	Allowable Limits
(i)	Limits of XVII-2000 of Appendix XVII of the ASME Code, Section III.
(ii)	Limits of F-1370 of Appendix F of the ASME Code, Section III.

(2) If the limit method of analysis is used:

Combination	Allowable Limits
(i)	Limits of XVII-400 of Appendix XVII of the ASME Code, Section III.
(ii)	Same as above.

6. Materials, Quality Control, and Special Construction Techniques

The specified materials of construction and quality control programs are acceptable if in accordance with the applicable code or standard as indicated in Section I.6 of this plan.

Special construction techniques, if any, are treated on a case-by-case basis.

7. Testing and In-service Surveillance Requirements

Each BWR Mark III containment drywell should be subjected to a structural proof test. Such a test is acceptable if in accordance with the following:

a. The drywell should be subjected to an acceptance test that increases the drywell internal pressure in three or more approximately equal pressure increments from atmospheric pressure to at least the design pressure. The drywell should be depressurized in the same number of increments. Measurements should be recorded at atmospheric pressure and at each pressure level of the pressurization and depressurization cycles. At each level, the pressure should be held constant for at least one hour before the deflections and strains are recorded.

b. So that the overall deflection pattern can be determined in prototype drywells, radial deflections should be measured at least at three points along each of at least three meridians equally spaced around the drywell, including locations with varying stiffness characteristics. Radial deflections should be measured at the lower vent region, at about mid-height and at near the top of the cylindrical wall. The measurement points may be relocated depending on the distribution of stresses and deformations anticipated in each particular design.

c. In prototype drywells only, strain measurements sufficient to permit an evaluation of strain distribution should be recorded at least at two opposing meridians at the following locations on the wall:

(1) at the bottom of the wall, and

(2) at mid-height of the wall.

These strain measurements should be made at least at three positions within the wall section; one at the center and one each near the inner and outer surfaces.

d. In nonprototype drywells, deflection and strain measurements need not be made if strain levels have been correlated with deflection measurements during the acceptance test of a prototype drywell if measured strains and deflections are within the predefined tolerances of their predicted response.

e. Any reliable system of displacement meters, optical devices, strain gauges, or other suitable apparatus may be used for the measurements.

f. If the test pressure drops due to unexpected conditions to or below the next lower pressure level, the entire test sequence should be repeated. Significant deviations from the previous test should be recorded and evaluated.

g. If any significant modifications or repairs are made to the drywell following and because of the initial test, the test should be repeated.

h. A description of the proposed acceptance test and instrumentation requirements should be included in the preliminary safety analysis report.

i. The following information should be submitted prior to the performance of the test:

(i) The numerical values of the predicted responses of the structure which will be measured.

(ii) The tolerances to be permitted on the predicted responses.

(iii) The bases on which the predicted responses and the tolerances thereon were established.

i. The following information should be included in the final test report:

(i) A description of the actual test and instrumentation.

(ii) A comparison of the test measurements with the allowable limits (predicted response plus tolerance) for deflections and strains.

(iii) An evaluation of the accuracy of the measurements.

(iv) An evaluation of any deviations (i.e., test results that exceed the allowable limits), the disposition of the deviations, and the need for corrective measures.

(v) A discussion of the calculated safety margin provided by the structure as deduced from the test results.

For steel linear supports of the reactor coolant system, testing and in-service surveillance requirements are acceptable if in accordance with Subsection NF of the ASME Section III Code, Division I.

III. REVIEW PROCEDURES

The reviewer selects and emphasizes material from the review procedures described below, as may be appropriate for a particular case.

1. Description of the Internal Structures

After each structure and its functional characteristics are identified, information on similar structures of previously licensed applications is obtained for reference. Such information, which is available in safety analysis reports and amendments of licensed plants enables identification of differences for the case under review which require additional scrutiny. New or unique features that have not been used in the past are of particular interest. The information furnished in the SAR is reviewed for sufficiency in accordance with the "Standard Format..." Revision 2. A decision is then made with regard to the sufficiency of the descriptive information provided in the SAR. Any additional required information is requested from the applicant at an early stage of the review process.

2. Applicable Codes, Standards, and Specifications

The list of codes, standards, guides, and specifications is checked against the list in Section II.2 of this plan. The reviewer assures himself that the applicable edition and stated effective addenda are utilized.

3. Loads and loading Combinations

The reviewer verifies that the loads and load combinations are as conservative as those specified in Section II.3 of this plan. Any deviations from the acceptance critieria for loads and load combinations that have not been adequately justified are

identified as unacceptable and transmitted to the applicant for further consideration.

4. Design and Analysis Procedures

The reviewer familiarizes himself with the design and analysis procedures that are generally utilized for the type of structure being reviewed. Since the assumptions made on the expected behavior of the structure and its various elements under loads may be significant, the reviewer determines that they are conservative. The behavior of the structure under various loads and the manner in which these loads are treated in conjunction with other coexistent loads, are reviewed to establish compliance with procedures delineated in Section II.4 of this plan.

5. Structural Acceptance Criteria

The limits on allowable stresses and strains in the concrete, reinforcement, structural steel, etc., are compared with those specified in Section II.5 of this plan. Where the applicant proposes to exceed some of these limits for some of the load combinations and at some localized points on the structure, the justification provided to show that the functional integrity of the structure will not be affected is evaluated. If such justification is not acceptable, a request for the required additional justification and bases is made.

6. Materials, Quality Control, and Special Construction Techniques

The information provided on materials, quality control programs, and special construction techniques, if any, is reviewed and compared with that specified in Section II.6 of this plan. If a new material not used in prior license applications is utilized, the applicant is requested to provide sufficient test and user data to establish the acceptability of such a material. Similarly, any new quality control programs or construction techniques are reviewed and evaluated to assure that there will be no degradation of structural quality that might affect the structural integrity of the structure.

7. Testing and In-service Surveillance Requirements

Procedures for the structural test of the BWR Mark III containment drywell are reviewed and compared with the procedures described in Section II.7 of this plan. Any other proposed testing andin-service surveillance programs are reviewed on a case-by-case basis.

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information has been provided in accordance with the requirements of this review plan, and concludes that his evaluation is sufficiently complete and adequate to support the following type of conclusive statement to be included in the staff's safety evaluation report:

"The criteria used in the design, analysis, and construction of the containment internal structures to account for anticipated loadings and postulated conditions that may be imposed upon the structures during their service lifetime are in conformance with established criteria, and with codes, standards, and specifications acceptable to the Regulatory staff.

"The use of these criteria as defined by applicable codes, standards, and specifications; the loads and loading combinations; the design and analysis procedures; the structural acceptance criteria; the materials, quality control programs, and special construction techniques; and the testing and in-service surveillance requirements provide reasonable assurance that, in the event of earthquakes and various postulated accidents occurring within the containment, the interior structures will withstand the specified design conditions without impairment of structural integrity or the performance of required safety functions. Conformance with these criteria constitutes an acceptable basis for satisfying in part the requirements of General Design Criteria 2 and 4."

V. REFERENCES

1. ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NF, "Requirements for Component Supports," American Society of Mechanical Engineers.

2. ACI 318-1971, "Building Code Requirements for Reinforced Concrete," American Concrete Institute (1971).

3. AISC, "Specification for Design, Fabrication and Erection of Structural Steel for Buildings," American Institute of Steel Construction (1969).

4. ASME Boiler and Pressure Vessel Code, Section III, Division 2 (ACI-359), "Proposed Standard Code for Concrete Reactor Vessels and Containments," issued for interim trial use and comment, April 1973, American Society of Mechanical Engineers.

5. ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NE, "Class MC Components," American Society of Mechanical Engineers.

6. Regulatory Guide 1.55, "Concrete Placement in Category I Structures."

7. ANSI N45.2.5, "Supplementary Quality Assurance Requirements for Installation, Inspection and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants," Draft 3, Revision 1, January 1974, American National Standards Institute.

8. Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," Revision 2 (in preparation).

9. Regulatory Guide 1.57, "Design Limits and Loading Combinations for Metal Primary Reactor Containment System Components."

10. 10 CFR Part 50, Appendix A, General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena.

11. 10 CFR Part 50, Appendix A, General Design Criterion 4, "Environmental and Missile Design Bases."

U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

SECTION 3.8.4 OTHER SEISMIC CATEGORY I STRUCTURES

REVIEW RESPONSIBILITIES

Primary - Structural Engineering Branch (SEB)

Secondary - None

I. AREAS OF REVIEW

The following areas relating to all seismic Category I structures and other safety-related structures that may not be classified as seismic Category I, other than the containment and its interior structures, are reviewed.

1. Description of the Structures

The descriptive information, including plans and sections of each structure, is reviewed to establish that sufficient information is provided to define the primary structural aspects and elements relied upon for the structure to perform its safety-related function. Also reviewed is the relationship between adjacent structures including the separation provided or structural ties, if any. Among the major plant structures that are reviewed, together with the descriptive information reviewed for each, are the following:

a. Containment Enclosure Building

The containment enclosure building, which may surround all or part of the primary concrete or steel containment structure, is primarily intended to reduce leakage during and after a loss-of-coolant accident (LOCA) from within the containment. Concrete enclosure buildings also protect the primary containment, which may be of steel or concrete, from outside hazards.

The enclosure building is usually either a concrete structure or a structural steel and metal siding building.

Where it is a concrete structure, it usually has the geometry of the containment and, as applicable, the descriptive information reviewed is similar to that of a concrete containment as contained in Section I.1 of Standard Review Plan 3.8.1.

Where it is a structural steel and metal siding building, the following items are reviewed: general arrangement of the building including its foundations, wall, and roof; any bracing and lateral ties provided for the stability of the building; the roof supports which may bear on the dome of the containment; and major corner and siding joint connections.

b. Auxiliary Building

The auxiliary building, which is usually adjacent to the containment and which may be shared by the two containments in 2-unit plants, is usually of reinforced concrete and structural steel construction. The general arrangement of the structural walls, columns, floors, roof, and any removable sections, is reviewed.

c. Fuel Storage Building

The fuel storage building, which may be independent or part of the auxiliary building, is also of reinforced concrete and structural steel. It houses the new fuel storage area and the spent fuel pool. In addition to the information reviewed for the auxiliary building, the general arrangement of the spent fuel pool is reviewed including its foundations and walls.

d. Control Building

The control room is located in most plants within the auxiliary building. However, where it is located in a separate building, usually called the control building, the building is reviewed as a separate structure. To provide missile protection and shielding, this building is usually of reinforced concrete and the descriptive information reviewed is similar to that reviewed for the auxiliary building.

e. Diesel Generator Building

The emergency diesel generators are, in some plants, located within the auxiliary building. However, they may also be located in a separate building called the diesel generator building. Again, this is usually a reinforced concrete structure and the descriptive information reviewed is similar to that reviewed for the auxiliary building.

f. Other Structures

In most plants, there are several miscellaneous seismic Category I structures and other structures that may be safety-related but, because of other design provision, may not be classified as seismic Category I. These structures are usually either of reinforced concrete or structural steel, or a combination thereof. The descriptive information reviewed for such structures is similar to that reviewed for the auxiliary building. Among such structures are: pipe and electrical conduit tunnels, waste storage facilities, stacks, intake structures, pumping stations, and cooling towers.

Further, the reviewer may encounter special safety-related structures such as emergency cooling water tunnels, embankments, concrete dams, and water wells. Such structures are reviewed on a case-by-case basis. The descriptive information provided is reviewed to understand the structural behavior of these structures, specifically during seismic events and plant process conditions during which such structures are required to remain functional.

2. Applicable Codes, Standards, and Specifications

Information pertaining to design codes, standards, specifications, regulations, general design criteria and regulatory guides, and other industry standards that are applied in the design, fabrication, construction, testing, and surveillance of seismic Category I structures, is reviewed.

3. Loads and Load Combinations

Information pertaining to the applicable design loads and various combinations thereof, is reviewed. The loads normally applicable to seismic Category I structures include the following:

a. Those loads encountered during normal plant startup, operation, and shutdown, including dead loads, live loads, thermal loads due to operating temperature, and hydrostatic loads such as those in spent fuel pools.
b. Those loads to be sustained during severe environmental conditions, including those induced by the operating basis earthquake (OBE) and the design wind specified for the plant.
c. Those loads to be sustained during extreme environmental conditions, including those induced by the safe shutdown earthquake (SSE) and the design tornado specified for the plant.
d. Those loads to be sustained during abnormal plant conditions. Such abnormal plant conditions include the postulated rupture of high-energy piping. Loads induced by such an accident may include elevated temperatures and pressures within or across compartments, and possibly jet impingement and impact forces associated with such ruptures.

The various combinations of the above loads that are normally postulated and reviewed include normal operating loads; normal operating loads with severe environmental loads; normal operating loads with extreme environmental loads; normal operating loads with abnormal loads; normal operating loads with severe environmental and abnormal loads; and normal operating loads with extreme environmental and abnormal loads.

The loads and load combinations described above are generally applicable to all types of structures. However, other site-related loads might also be applicable. Such loads, which are not normally combined with abnormal loads, include those induced by floods, potential aircraft crashes, explosive hazards in proximity to the site, and projectiles and missiles generated from activities of nearby military installations.

4. Design and Analysis Procedures

The design and analysis procedures utilized for Category I structures are reviewed with emphasis on the extent of compliance with the ACI-318-71 Code (Ref. 1) for concrete structures and with the AISC Specifications (Ref. 2) for steel structures, including the following areas:

a. General assumptions on boundary conditions.
b. The expected behavior under loads and the methods by which vertical and lateral loads and forces are transmitted from the various elements to their supports and eventually to the foundation of the structure.
c. The computer programs that are utilized.

Any new or unique procedures used in the design and analysis are reviewed on a case-by-case basis.

5. Structural Acceptance Criteria

The design limits imposed on the various parameters that serve to quantify the structural behavior of each structure and its components are reviewed, specifically with respect to stresses, strains, gross deformations, and factors of safety against structural failure. For each load combination specified, the specified allowable limits are compared with the acceptable limits delineated in Section II.5 of this plan.

6. Materials, Quality Control, and Special Construction Techniques

Information on the materials that are used in the construction of Category I structures is reviewed. Among the major materials of construction that are reviewed are the concrete ingredients, the reinforcing bars and splices, and the structural steel and anchors.

The quality control program that is proposed for the fabrication and construction of Category I structures is reviewed including nondestructive examination of the materials to determine physical properties, placement of concrete, and erection tolerances.

Special construction techniques, if proposed, are reviewed on a case-by-case basis to determine their effects on the structural integrity of the completed structure.

In addition, the information contained in items a, c, and d of Section I.6 of Standard Review Plan 3.8.3, is also reviewed.

7. Testing and In-service Surveillance Programs

If applicable, any post-construction testing and in-service surveillance programs are reviewed on a case-by-case basis.

II. ACCEPTANCE CRITERIA

The acceptance criteria for the areas of review, described in Section I of this plan are as follows:

1. Description of the Structure

The descriptive information in the SAR is considered acceptable if it meets the minimum requirements set forth in Section 3.8.4.1 of the "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," Revision 2 (Ref 3).

Deficient areas of descriptive information are identified by the reviewer and a request for additional information is initiated at the application acceptance review. New or unique design features that are not specifically covered in the "Standard Format...," require a more detailed review. The reviewer determines the additional information that may be required to accomplish a meaningful review of the structural aspects of such new or unique features.

2. Applicable Codes, Standards, and Specifications

The design, materials, fabrication, erection, inspection, testing, and surveillance, if any, of Category I structures are covered by codes, standards, and guides that are either applicable in their entirety or in portions thereof. A list of such documents is contained in Section II.2 of Standard Review Plan 3.8.3.

3. Loads and Load Combinations

The specified loads and load combinations are acceptable if found to be in accordance with the following:

Loads, Definitions, and Nomenclature

All the major loads to be encountered or to be postulated in a nuclear power plant are listed below. All the loads listed, however, are not necessarily applicable to all the structures and their elements. Loads and the applicable load combinations for which each structure has to be designed will depend on the conditions to which that particular structure may be subjected.

Normal loads, which are those loads to be encountered during normal plant operation and shutdown, include:

- D --- Dead loads or their related internal moments and forces including any permanent equipment loads and hydrostatic loads.
- L --- Live loads or their related internal moments and forces including any movable equipment loads and other loads which vary with intensity and occurrence, such as soil pressure.
- T_o --- Thermal effects and loads during normal operating or shutdown conditions, based on the most critical transient or steady-state condition,
- R_o --- Pipe reactions during normal operating or shutdown conditions, based on the most critical transient or steady-state condition,

Severe environmental loads include:

- E --- Loads generated by the operating basis earthquake.
- W --- Loads generated by the design wind specified for the plant.

Extreme environmental loads include:

- E' --- Loads generated by the safe shutdown earthquake.
- W_t --- Loads generated by the design tornado specified for the plant. Tornado loads include loads due to the tornado wind pressure, the tornado-created differential pressure, and to tornado-generated missiles.

Abnormal loads, which are those loads generated by a postulated high-energy pipe break accident, include:

- P_a --- Pressure equivalent static load within or across a compartment generated by the postulated break, and including an appropriate dynamic load factor to account for the dynamic nature of the load.
- T_a --- Thermal loads under thermal conditions generated by the postulated break and including T_o.

R_a --- Pipe reactions under thermal conditions generated by the postulated break and including R_o.

Y_r --- Equivalent static load on the structure generated by the reaction on the broken high-energy pipe during the postulated break, and including an appropriate dynamic load factor to account for the dynamic nature of the load.

Y_j --- Jet impingement equivalent static load on a structure generated by the postulated break, and including an appropriate dynamic load factor to account for the dynamic nature of the load.

Y_m --- Missile impact equivalent static load on a structure generated by or during the postulated break, as from pipe whipping, and including an appropriate dynamic load factor to account for the dynamic nature of the load.

In determining an appropriate equivalent static load for Y_r, Y_j, and Y_m, elasto-plastic behavior may be assumed with appropriate ductility ratios, provided excessive deflections will not result in loss of function of any safety-related system.

Load Combinations for Concrete Structures

For concrete structures, the load combinations are acceptable if found in accordance with the following:

a. For service load conditions, either the working stress design (WSD) method or the strength design method may be used.

(i) If the WSD method is used, the following load combinations should be considered:

(1) $D + L$

(2) $D + L + E$

(3) $D + L + W$

If thermal stresses due to T_o and R_o are present, the following combinations should also be considered:

(1a) $D + L + T_o + R_o$

(2a) $D + L + T_o + R_o + E$

(3a) $D + L + T_o + R_o + W$

Both cases of L having its full value or being completely absent should be checked.

(ii) If the strength design method is used, the following load combinations should be considered:

(1) $1.4\ D + 1.7\ L$

(2) $1.4\ D + 1.7\ L + 1.9\ E$

(3) $1.4\ D + 1.7\ L + 1.7\ W$

If thermal stresses due to T_o and R_o are present the following combinations should also be considered:

(1b) $(0.75)\ (1.4\ D + 1.7\ L + 1.7\ T_o + 1.7\ R_o)$

(2b) $(0.75)\ (1.4\ D + 1.7\ L + 1.9\ E + 1.7\ T_o + 1.7\ R_o)$

(3b) $(0.75)\ (1.4\ D + 1.7\ L + 1.7\ W + 1.7\ T_o + 1.7\ R_o)$

Both cases of L having its full value or being completely absent should be checked. In addition, the following combinations should be considered:

(2b') $1.2\ D + 1.9\ E$

(3b') $1.2\ D + 1.7\ W$

Where soil and hydrostatic pressures are present, in addition to all the above combinations where they have been included in L and D respectively, the requirements of Sections 9.3.4 and 9.3.5 of ACI-318-71 (Ref. 1) should also be satisfied.

b. For factored load conditions, which represent extreme environmental, abnormal, abnormal/severe environmental and abnormal/extreme environmental conditions, the strength design method should be used and the following load combinations should be considered.

(4) $D + L + T_o + R_o + E'$

(5) $D + L + T_o + R_o + W_t$

(6) $D + L + T_a + R_a + 1.5\ P_a$

(7) $D + L + T_a + R_a + 1.25\ P_a + 1.0\ (Y_r + Y_j + Y_m) + 1.25\ E$

(8) $D + L + T_a + R_a + 1.0\ P_a + 1.0\ (Y_r + Y_j + Y_m) + 1.0\ E'$

In combinations (6), (7), and (8), the maximum values of P_a, T_a, R_a, Y_j, Y_r, and Y_m, including an appropriate dynamic load factor, should be used unless a time-history analysis is performed to justify otherwise. Combinations (5), (7), and (8) and the corresponding structural acceptance criteria of Section II.5 of this plan should be satisfied first without the tornado missile load in (5) and without Y_r, Y_j, and Y_m in (7) and (8). When considering these concentrated loads, local section strength capacities may be exceeded provided there will be no loss of function of any safety-related system.

Both cases of L having its full value or being completely absent should be checked.

Load Combinations for Steel Structures

For steel structures, the load combinations are acceptable if found in accordance with the following:

a. For service load conditions, either the elastic working stress design methods of Part 1 of the AISC specifications, or the plastic design methods of Part 2 of the AISC specifications, may be used.

(i) If the elastic working stress design methods are used, the following load combinations should be considered:

(1) $D + L$

(2) $D + L + E$

(3) $D + L + W$

If thermal stresses due to T_o and R_o are present, the following combinations should also be considered:

(1a) $D + L + T_o + R_o$

(2a) $D + L + T_o + R_o + E$

(3a) $D + L + T_o + R_o + W$

Both cases of L having its full value or being completely absent should be checked.

(ii) If plastic design methods are used, the following load combinations should be considered:

(1) $1.7\ D + 1.7\ L$

(2) $1.7\ D + 1.7\ L + 1.7\ E$

(3) $1.7\ D + 1.7\ L + 1.7\ W$

If thermal stresses due to T_o and R_o are present, the following combinations should also be considered:

(1b) $1.3\ (D + L + T_o + R_o)$

(2b) $1.3\ (D + L + E + T_o + R_o)$

(3b) $1.3\ (D + L + W + T_o + R_o)$

Both cases of L having its full value or being completely absent should be checked.

b. For factored load conditions, the following load combinations should be considered:

(i) If elastic working stress design methods are used:

(4) $D + L + T_o + R_o + E'$

(5) $D + L + T_o + R_o + W_t$

(6) $D + L + T_a + R_a + P_a$

(7) $D + L + T_a + R_a + P_a + 1.0\ (Y_j + Y_r + Y_m) + E$

(8) $D + L + T_a + R_a + P_a + 1.0\ (Y_j + Y_r + Y_m) + E'$

(ii) If plastic design methods are used:

(4) $D + L + T_o + R_o + E'$

(5) $D + L + T_o + R_o + W_t$

(6) $D + L + T_a + R_a + 1.5\ P_a$

(7) $D + L + T_a + R_a + 1.25\ P_a + 1.0\ (Y_j + Y_r + Y_m) + 1.25\ E$

(8) $D + L + T_a + R_a + 1.0\ P_a + 1.0\ (Y_j + Y_r + Y_m) + E'$

In the above factored load combinations, thermal loads can be neglected when it can be shown that they are secondary and self-limiting in nature and where the material is ductile.

In combinations (6), (7), and (8), the maximum values of P_a, T_a, R_a, Y_j, Y_r, and Y_m, including an appropriate dynamic load factor, should be used unless a time-history analysis is performed to justify otherwise. Combinations (5), (7), and (8) and the corresponding structural acceptance criteria of Section II.5 of this plan should be first satisfied without the tornado missile load in (5) and without Y_r, Y_j, and Y_m in (7) and (8). When considering these concentrated loads, local section strengths may be exceeded provided there will be no loss of function of any safety-related system.

4. Design and Analysis Procedures

The design and analysis procedures utilized for Category I structures, including assu..., ions on boundary conditions and expected behavior under loads, are acceptable if found in accordance with the following:

a. For concrete structures, the procedures are in accordance with ACI-318-71, "Building Code Requirements for Reinforced Concrete," (Ref. 1).

b. For steel structures, the procedures are in accordance with the AISC "Specification...," (Ref. 2).

Computer programs are acceptable if the validation provided is found in accordance with procedures delineated in Section II.4.e of Standard Review Plan 3.8.1.

5. Structural Acceptance Criteria

For each of the loading combinations delineated in Section II.3 of this plan, the following defines the allowable limits which constitute the structural acceptance criteria.

In Combinations for Concrete	Limit
a(i)1, 2, 3	S(1)
a(i)1a, 2a, 3a	1.3 S
a(ii)1, 2, 3	U(2)
a(ii)1b, 2b, 3b	U
a(ii)2b', 3b'	U
(b)4, 5, 6, 7, 8	U

In Combinations for Steel	Limit
a(i)1, 2, 3	S
a(i)1a, 2a, 3a	1.5 S
a(ii)1, 2, 3	Y(3)
a(ii)1b, 2b, 3b	Y
b(i)4, 5, 6, 7(4)	1.6 S
b(i)8(4)	1.7 S
b(ii)4, 5, 6, 7, 8	.9 Y

NOTES

(1) S --- For concrete structures, S is the required section strength based on the working stress design method and the allowable stresses defined in Section 8.10 of ACI-318-71.

For structural steel, S is the required section strength based on the elastic design methods and the allowable stresses defined in Part 1 of the AISC "Specification for the Design, Fabrication and Erection of Structural Steel for Buildings," February 12, 1969.

The 33% increase in allowable stresses for concrete and steel due to seismic or wind loadings is not permitted.

(2) U --- For concrete structures, U is the section strength required to resist design loads based on the strength design methods described in ACI-318-71.

(3) Y --- For structural steel, Y is the section strength required to resist design loads and based on plastic design methods described in Part 2 of the AISC "Specification for the Design, Fabrication and Erection of Structural Steel for Buildings," February 12, 1969.

(4) --- For these two combinations, in computing the required section strength, S, the plastic section modulus of steel shapes may be used.

6. Materials, Quality Control, and Special Construction Techniques

For Category I structures outside the containment, the acceptance criteria for materials, quality control, and any special construction techniques are in accordance with the codes and standards indicated in Section I.6 of Standard Review Plan 3.8.3, as applicable.

7. Testing and Inservice Surveillance Requirements

At present there are no special testing or in-service surveillance requirements for Category I structures outside the containment. However, where some requirements become necessary for special structures, such requirements are reviewed on a case-by-case basis.

III. REVIEW PROCEDURES

The reviewer selects and emphasizes material from the review procedures described below, as may be appropriate for a particular case.

1. Description of the Structures

After the type of structure and its functional characteristics are identified, information on similar and previously licensed plants is obtained for reference. Such information, which is available in safety analysis reports and amendments of previous license applications, enables identification of differences for the case under review. These differences require additional scrutiny and evaluation. New and unique features that have not been used in the past are of particular interest and are thus examined in greater detail. The information furnished in the SAR is reviewed for completeness in accordance with the "Standard Format...,"Revision 2. A decision is then made with regard to the sufficiency of the descriptive information provided. Any additional required information not provided is requested from the applicant at an early stage of the review process.

2. Applicable Codes, Standards, and Specifications

The list of codes, standards, guides, and specifications is compared with the list referenced in Section II.2 of this plan. The reviewer assures himself that the appropriate code or guide is utilized and that the applicable edition and stated effective addenda are acceptable.

3. Loads and Load Combinations

The reviewer verifies that the loads and load combinations are as conservative as those specified in Section II.3 of this plan. Any deviations from the acceptance criteria for loads and load combinations that have not been adequately justified are identified as unacceptable and transmitted to the applicant.

4. Design and Analysis Procedures

The reviewer assures himself that for the design and analysis procedures, the applicant is utilizing the ACI-318-71 Code and the AISC Specifications for concrete and steel structures, respectively.

Any computer programs that are utilized in the design and analysis of the structure are reviewed to verify their validity in accordance with the acceptance criteria delineated in Section II.4.e of Standard Review Plan 3.8.1.

5. Structural Acceptance Criteria

The limits on allowable stresses and strains in the concrete, reinforcement, and structural steel are compared with the corresponding allowable stresses specified in Section II.5 of this plan. If the applicant proposes to exceed some of these limits for some of the load combinations and at some localized points on the structure, the justification that the structural integrity of the structure will not be affected is evaluated. If such justification is determined to be inadequate, the proposed deviations are identified and transmitted to the applicant with a request for the required additional justification and bases.

6. Materials, Quality Control, and Special Construction Techniques

The materials, quality control procedures, and any special construction techniques are compared with those referenced in Section II.6 of this plan. If a new material not used in prior licensed cases is utilized, the applicant is requested to provide sufficient test and user data to establish the acceptability of such a material. Similarly, any new quality control procedures or construction techniques are evaluated to assure that there will be no degradation of structural quality that might affect the structural integrity of the structure.

7. Testing and In-service Surveillance Programs

Any testing and in-service surveillance programs are reviewed on a case-by-case basis.

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information has been provided to satisfy the requirements of this review plan, and concludes that his evaluation is sufficiently complete and adequate to support the following type of conclusive statement to be included in the staff's safety evaluation report:

"The criteria used in the analysis, design, and construction of all the plant Category I structures to account for anticipated loadings and postulated conditions that may be

imposed upon each structure during its service lifetime are in conformance with established criteria, codes, standards, and specifications acceptable to the Regulatory staff.

"The use of these criteria as defined by applicable codes, standards, and specifications; the loads and loading combinations; the design and analysis procedures; the structural acceptance criteria; the materials, quality control, and special construction techniques; and the testing and in-service surveillance requirements provide reasonable assurance that, in the event of winds, tornadoes, earthquakes and various postulated accidents occurring within the structures, the structures will withstand the specified design conditions without impairment of structural integrity or the performance of required safety functions. Conformance with these criteria, codes, specifications, and standards constitutes an acceptable basis for satisfying, in part, the requirements of General Design Criteria 2 and 4."

V. REFERENCES

1. ACI-318-71, "Building Code Requirements for Reinforced Concrete," American Concrete Institute (1971).

2. AISC, "Specification for Design, Fabrication and Erection of Structural Steel for Buildings," American Institute of Steel Construction (1969).

3. Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," Revision 2 (in preparation).

4. 10 CFR Part 50, Appendix A, General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena."

5. 10 CFR Part 50, Appendix A, General Design Criterion 4, "Environmental and Missile Design Bases."

U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

SECTION 3.8.5 FOUNDATIONS

REVIEW RESPONSIBILITIES

Primary - Structural Engineering Branch (SEB)

Secondary - None

I. AREAS OF REVIEW

The following areas relating to the foundations of all seismic Category I structures and other safety-related structures are reviewed.

1. Description of the Foundations

The descriptive information, including plans and sections of each foundation, is reviewed to establish that sufficient information is provided to define the primary structural aspects and elements relied upon to perform the foundation function. Also reviewed is the relationship between adjacent foundations, including the methods of separation provided where such separation is utilized to minimize seismic interaction between the buildings. In particular, the type of foundation is identified and its structural characteristics are examined. Among the various types of foundations that are reviewed are mat-foundations and footings, including individual column footings, combined footings supporting more than one column, and wall footings supporting bearing walls.

Other types of foundations that may also be utilized are pile foundations, caisson foundations, combinations of footings, retaining walls, abutments, and rock anchor systems. These foundation types are reviewed on a case-by-case basis.

The major plant Category I foundations that are reviewed, together with the descriptive information reviewed for each, are listed below:

a. Containment Structure Foundation

The most commonly used type of foundation for both concrete and steel containments is a mat foundation, where a flat thick slab supports the containment, its interior structures, and a shield building surrounding the containment, if any. For some PWR containments the base mat has a central depression forming the reactor cavity. The general arrangement of the containment base slab is reviewed as described in Section 1.1 of Standard Review Plan 3.8.1, with particular emphasis on methods of

transferring horizontal shears, such as those seismically-induced, to the foundation media. Where shear keys are utilized for such purposes, the general arrangement of the keys is reviewed. Where waterproofing membranes are utilized, their effect on the shear resistance of the foundation is reviewed. In prestressed concrete containments, where a tendon inspection gallery is utilized, arrangement of the gallery and means of either isolating it from the remainder of the base slab or of relying upon it for some function such as resisting shears, are reviewed.

b. Containment Enclosure Building Foundation

Where the containment enclosure building is constructed of reinforced concrete, it is usually supported on the same mat foundation supporting the containment.

Where it is a structural steel and metal siding building, it may surround only the exposed portion of the containment. In such a situation, the enclosure building columns are founded on individual or combined footings at grade level, on the roof of buildings adjacent to or surrounding the containment, on the dome of the containment, and possibly on brackets anchored on the exterior face of the cylindrical wall of the containment. General arrangement of such foundations is reviewed with particular emphasis on methods of isolating the enclosure building from other buildings in a lateral direction, where this is preferable to minimize seismic interaction.

c. Auxiliary Building Foundation

The auxiliary building foundation is typically of a mat type, particularly where the supporting foundation media is soil.

The general arrangement of the foundation is reviewed, again with particular emphasis on methods of transferring loads from the structure to the foundation media.

d. Other Category I Foundations

The foundations for other Category I structures, which may be one or a combination of several foundation types, are reviewed to an extent similar to that of the containment foundation. Among Category I structures the foundations of which are so reviewed, are: fuel storage buildings, control buildings, diesel generator buildings, intake structures, and cooling towers. Also reviewed are foundations of safety-related structures which, because of other design provisions, are not classified as seismic Category I.

2. Applicable Codes, Standards, and Specifications

Information pertaining to design codes, standards, specifications, regulations, general design criteria and regulatory guides, and other industry standards that are applied in the design, fabrication, construction, testing, and surveillance of seismic Category I foundations is reviewed.

3. Loads and Load Combinations

Information pertaining to the applicable design loads and their various combinations is reviewed. The loads normally applicable to Category I foundations are the same as those applicable to the structures which the foundations support. These loads are described in Section I.3 of Standard Review Plan 3.8.4.

4. Design and Analysis Procedures

The design and analysis procedures utilized for Category I foundations are reviewed with emphasis on the extent of compliance with the ACI-318-71 Code (Ref. 1) for concrete structures, and with the AISC Specifications (Ref. 2) for steel structures, including the following areas:

a. The assumptions made on boundary conditions and the expected behavior of each foundation when subjected to the various design loads.
b. The methods by which lateral loads and forces and overturning moments thereof are transmitted from the structure to the foundation media. Such forces are mainly generated by the environmental and abnormal plant conditions such as wind, tornadoes, earthquakes, and pipe ruptures. Methods of determining overturning moments due to the three components of the earthquake are also reviewed.
c. The computer programs that are utilized in the design and analysis of foundations.

5. Structural Acceptance Criteria

The design limits imposed on the various parameters that serve to quantify the structural behavior of each foundation are reviewed with emphasis on the extent of compliance with the ACI-318-71 Code for concrete structures, specifically with respect to stresses, strains, deformations, and factors of safety against overturning and sliding, as applicable.

6. Materials, Quality Control, and Special Construction Techniques

Information on the materials that are used in the construction of Category I foundations is reviewed. Among the major materials of construction that are reviewed are the following:

a. The concrete ingredients.
b. The reinforcing bars and mechanical splices.
c. The structural steel.
d. Rock anchors, including any prestressing system.

The quality control program that is proposed for the fabrication and construction of Category I foundations is reviewed, including the following: nondestructive examination of the materials to determine physical properties, placement of concrete, and erection tolerances.

Special construction techniques, if proposed, are reviewed on a case-by-case basis to determine their effects on the structural integrity of the completed foundation.

In addition, the information contained in items a, c, and d of Section I.6 of Standard Review Plan 3.8.3, is also reviewed.

7. Testing and In-service Surveillance Programs

If applicable, any post-construction testing and in-service surveillance programs for foundations, such as monitoring potential settlements and displacements, are reviewed on a case-by-case basis.

II. ACCEPTANCE CRITERIA

The acceptance criteria for the areas of review described in Section I of this plan are as follows:

1. Description of the Foundation

   The descriptive information in the SAR is considered acceptable if it meets the minimum requirements set forth in Section 3.8.5.1 of the "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," Revision 2 (Ref. 4).

   Deficient areas of descriptive information are identified by the reviewer and a request for additional information is initiated, at the application acceptance review, if possible. New or unique design features that are not specifically covered in the "Standard Format...," require a more detailed review. The reviewer determines the additional information that may be required to accomplish a meaningful review of the structural aspects of such new or unique foundation features.

2. Applicable Codes, Standards, and Specification:

   The design, materials, fabrication, erection, inspection, testing, and surveillance, if any, of Category I foundations are covered by codes, standards, and guides that are either applicable in their entirety or in portions thereof. A list of such documents is contained in Section II.2 of the Standard Review Plan 3.8.3. In addition, the documents listed in Section II.2 of Standard Review Plan 3.8.1 are acceptable for the containment foundation.

3. Loads and Load Combinations

   The specified loads and load combinations utilized in the design of Category I foundations are acceptable if found to be in accordance with those combinations referenced in Section II.3 of Standard Review Plan 3.8.1 for the containment foundation, and with those combinations listed in Section II.3 of Standard Review Plan 3.8.4 for all other Category I foundations.

   In addition to the load combinations referenced above, the combinations utilized to check against sliding and overturning due to earthquakes, winds, and tornadoes, and against floatation due to floods, are found acceptable if in accordance with the following:

   a. $D + H + E$
   b. $D + H + W$
   c. $D + H + E'$
   d. $D + H + W_t$
   e. $D + F'$

   where D, E, W, E', W_t are as defined in Standard Review Plan 3.8.4, H is the lateral earth pressure, and F' is the bouyant force of the design basis flood. Justification should be provided for including live loads or portions thereof in these combinations.

4. Design and Analysis Procedures

   The design and analysis procedures utilized for Category I foundations are acceptable if found in accordance with the following:

a. For Category I concrete foundations other than the containment foundations, the procedures are in accordance with the ACI-318-71, "Building Code Requirements for Reinforced Concrete," (Ref. 1).

b. For Category I steel foundations, the procedures are in accordance with the AISC "Specifications...," (Ref. 2).

c. For the containment foundation, the design and analysis procedures referenced in Section II.4 of Standard Review Plan 3.8.1 are acceptable.

For determining the overturning moment due to an earthquake, the three components of the earthquake should be combined in accordance with methods described in Standard Review Plan 3.7.2. Computer programs are acceptable if the validation provided is found in accordance with procedures delineated in Section II.4.e of Standard Review Plan 3.8.1.

5. Structural Acceptance Criteria

For each of the loading combinations referenced in Section II.3 of this plan, the allowable limits which constitute the acceptance criteria are referenced in Section II.5 of Standard Review Plan 3.8.1 for the containment foundation, and are listed in Section II.5 of Standard Review Plan 3.8.4 for all other foundations. In addition, for the five additional load combinations delineated in Section II.3 of this plan, the factors of safety against overturning, sliding, and floatation are acceptable if found in accordance with the following:

	Minimum Factors of Safety		
For Combination	Overturning	Sliding	Floatation
a.	1.5	1.5	--
b.	1.5	1.5	--
c.	1.10	1.1	--
d.	1.10	1.1	--
e.	--	--	1.1

6. Materials, Quality Control, and Special Construction Techniques

For the containment foundation, the acceptance criteria for materials, quality control, and any special construction techniques are referenced in Section II.6 of Standard Review Plan 3.8.1. For all other Category I foundations, the acceptance criteria are similar to those referenced in Section II.6 of Standard Review Plan 3.8.4.

7. Testing and In-service Surveillance Requirements

At present there are no special testing or in-service surveillance requirements for Category I foundations other than those required for the containment foundation, which are covered in Section II.7 of Standard Review Plan 3.8.1. However, should some requirements become necessary for special foundations, they will be reviewed on a case-by-case basis.

III. REVIEW PROCEDURES

The reviewer selects and emphasizes material from the review procedures described below, as may be appropriate for a particular case.

1. Description of the Foundations

After the type of foundation and its structural characteristics are identified, information on similar and previously licensed plants is obtained for reference. Such information, which is available in safety analysis reports and amendments of license applications enables identification of differences for the case under review. These differences require additional scrutiny and evaluation. New and unique features that have not been used in the past are examined in greater detail. The information furnished in the SAR is reviewed for sufficiency in accordance with the "Standard Format...," Revision 2. A decision is then made with regard to the sufficiency of the descriptive information provided. Any additional required information is requested from the applicant at an early stage of the review process.

2. Applicable Codes, Standards, and Specifications

The list of codes, standards, guides, and specifications is compared with the list referenced in Section II.2 of this plan. The reviewer assures himself that the appropriate code or guide is utilized and that the applicable edition and stated effective addenda are acceptable.

3. Loads and Load Combinations

The reviewer verifies that the loads and load combinations are as conservative as those referenced and specified in Section II.3 of this plan. Any deviations from the acceptance criteria for loads and load combinations that have not been adequately justified are identified as unacceptable and transmitted to the applicant.

4. Design and Analysis Procedures

The reviewer assures himself that for the design and analysis procedures, the applicant is utilizing the procedures in the applicable code as delineated in Section II.4 of this plan.

Any computer programs that are utilized in the design and analysis of the foundation are reviewed to verify their validity in accordance with the acceptance criteria delineated in Section II.4.e of Standard Review Plan 3.8.1.

5. Structural Acceptance Criteria

The limits on allowable stresses and strains in the concrete, reinforcement, and structural steel, and on factors of safety for overturning, sliding, and floatation are compared with the corresponding allowable values specified in Section II.5 of this plan. If the applicant proposes to deviate from these limits for some of the load combinations and at some localized points, the justification that the structural integrity of the foundation will not be affected is evaluated. If such justification is determined to be inadequate, a request for the required additional justification and bases is made.

6. Materials, Quality Control, and Special Construction Techniques

The materials, quality control procedures, and any special construction techniques are compared with those referenced in Section II.6 of this plan. If a new material not

used in prior licensed cases is utilized, the applicant is requested to provide sufficient test and user data to establish the acceptability of such a material. Similarly, any new quality control procedures or construction techniques are evaluated in detail to assure that there will be no degradation of structural quality that might affect the structural integrity of the foundation.

7. Testing and In-service Surveillance Programs

For the containment foundation, testing and in-service surveillance programs are reviewed in accordance with Section II.7 of Standard Review Plan 3.8.1 for concrete containments. Any testing and in-service surveillance programs for other foundations are reviewed on a case-by-case basis.

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information has been provided to satisfy the requirements of this review plan, and concludes that his evaluation is sufficiently complete and adequate to support the following type of conclusive statement to be included in the staff's safety evaluation report:

"The criteria used in the analysis, design, and construction of all the plant Category I foundations to account for anticipated loadings and postulated conditions that may be imposed upon each foundation during its service lifetime are in conformance with established criteria, codes, standards, and specifications acceptable to the Regulatory staff.

"The use of these criteria as defined by applicable codes, standards, and specifications; the loads and loading combinations; the design and analysis procedures; the structural acceptance criteria; the materials, quality control, and special construction techniques; and the testing and in-service surveillance requirements provide reasonable assurance that, in the event of winds, tornadoes, earthquakes, and various postulated events, Category I foundations will withstand the specified design conditions without impairment of structural integrity and stability or the performance of required safety functions. Conformance with these criteria, codes, specifications, and standards constitutes an acceptable basis for satisfying in part the requirements of General Design Criteria 2 and 4."

V. REFERENCES

1. ACI-318-71, "Building Code Requirements for Reinforced Concrete," American Concrete Institute (1971).

2. AISC, "Specification for Design, Fabrication and Erection of Structural Steel for Building," American Institute of Steel Construction (1969).

3. ASME Boiler and Pressure Vessel Code, Section III, Division 2 (ACI-359), "Proposed Standard for Concrete Reactor Vessels and Containments, issued for interim trial use and comment, April 1973, American Society of Mechanical Engineers.

4. Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," Revision 2 (in preparation).

5. 10 CFR Part 50, Appendix A, General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena."

6. 10 CFR Part 50, Appendix A, General Design Criterion 4, "Environmental and Missile Design Bases."

U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

SECTION 3.9.1 SPECIAL TOPICS FOR MECHANICAL COMPONENTS

REVIEW RESPONSIBILITIES

Primary - Mechanical Engineering Branch (MEB)

Secondary - Reactor Systems Branch (RSB)

I. AREAS OF REVIEW

Information concerning design transients and methods of analysis for seismic Category I components, including both those designated as Class 1, 2, 3, or CS under the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section III (hereafter "the Code"), and component supports, reactor internals, and other components not covered by the Code, is given in the applicant's safety analysis report (SAR) and is reviewed by the MEB to assure conformance with the requirements of General Design Criteria 14 and 15. Certain aspects of dynamic system analysis methods are discussed in SRP section 3.9.2 as well as in this SRP section. The following specific subjects are reviewed under this SRP section:

1. Transients which are used in the design and fatigue analyses of all Code Class 1 and CS components, and of component supports and reactor internals. The Reactor Systems Branch confirms on request the acceptability of the listed transients and the number of cycles and events expected over the service lifetime of the plant. The Structural Engineering Branch confirms the seismic cyclic ground input loading as described in SRP section 3.7.3. The method used to determine the seismic cyclic loading used for fatigue analysis of appropriate components and supports will be reviewed.

2. Descriptions of all computer programs which will be used in analyses of seismic Category I Code and non-Code items listed in this SRP section.

3. Descriptions of any experimental stress analysis programs which will be used in lieu of theoretical stress analyses.

4. Descriptions of the analysis methods which will be used if the applicant elects to use elastic-plastic stress analysis methods in the design of any of the above-noted components.

II. ACCEPTANCE CRITERIA

The acceptance criteria for the areas of review are as follows:

1. The applicant shall provide a complete list of transients to be used in the design and fatigue analysis of all Code Class 1 and CS components, and of component supports and reactor internals within the reactor coolant pressure boundary. The number of events for each transient shall be included along with assurance that the number of load and stress cycles per event is properly taken into account and that the method used to determine the number of cycles is acceptable. All transients such as startup and shutdown operations, power level changes, emergency and recovery conditions, switching operations (i.e., startup or shutdown of one or more coolant loops), control system or other system malfunctions, component malfunctions, transients resulting from single operator errors, inservice hydrostatic tests, seismic and design basis events, that are contained in the Code-required "Design Specifications" for the components of the reactor coolant pressure boundary shall be specified. All transients or combinations of transients shall be categorized with respect to the plant/system operating conditions identified as "normal," "upset," "emergency," "faulted," or "testing".

   The section of the applicant's SAR which pertains to transients will be acceptable if the transient conditions selected for equipment fatigue evaluation are based upon a conservative estimate of the magnitude and frequency of the temperature and pressure conditions resulting from those transients. To a large extent the selection of these specific transient conditions is based upon engineering judgment and experience. Some guidance on the selection of these transients and combinations can be found in References 5 and 6. Transients, and their inclusion in the design and service loading combinations must provide a complete basis for design of the reactor coolant pressure boundary for all conditions and events expected over the service lifetime of the plant to satisfy, in part, the requirements of General Design Criteria 14 and 15.

2. A list of computer programs that will be used in dynamic and static analyses to determine the structural and functional integrity of seismic Category I Code and non-Code items and the analyses to determine stresses shall be provided, including a brief description of each program and the extent of its application. The design control measures, as required by Appendix B of 10 CFR Part 50, that will be employed to demonstrate the applicability and validity of these computer programs should meet one of the following criteria:

   a. The computer program is recognized and widely used, with a sufficient history of successful use to justify its applicability and validity without further demonstration by the applicant. The dated program version that will be used, the software or operating system, and the hardware configuration must be specified to be accepted by virtue of its history of use.

b. The computer program solutions to a series of test problems with accepted results have been demonstrated to be substantially identical to those obtained by a similar program which meets the criteria of (a) above. The test problems shall be demonstrated to be similar to or within the range of applicability for the problems analyzed by the computer program to justify acceptance of the program.

c. The program solutions to a series of test problems are substantially identical to those obtained by hand calculations or from accepted experimental tests or analytical results published in technical literature. The test problems shall be demonstrated to be similar to the problems analyzed to justify acceptance of the program.

A summary comparison of the results obtained from the use of each computer program under options (b) or (c) above with either the results derived from a similar program meeting option (a), or a previously approved computer program, or results from the test problems of option (c) shall be provided. They should include representative comparisons of responses due to static and/or dynamic loading, preferably in graphical form.

3. If experimental stress analysis methods are used in lieu of analytical methods, for any seismic Category I Code or non-Code items, the section of the SAR discussing the experimental stress analysis methods will be acceptable if the information provided meets the provisions of Appendix II of Reference 4, and as in the case of analytical methods, if the information provided is sufficiently detailed to show the validity of the design to meet the provisions of the Code-required "Design Specifications."

4. When service limit D is specified by the applicant for Code Class 1 and CS components, and for component supports, reactor internals, and other non-Code items, the methods of analysis used to calculate the stresses and deformations shall conform to the methods outlined in Appendix F of Reference 4, subject to deformation constraints discussed in III.4 below.

III. REVIEW PROCEDURES

The reviewer will select and emphasize material from the procedures described below, as may be appropriate for a particular case.

1. The list of transients, the number of events estimated for each transient presented in the applicant's SAR, and the method used to determine this number are compared to the same information on similar and previously licensed applications and to the acceptance criteria outlined in II above. Any deviations from previous accepted practice are noted and the applicant is required to justify these deviations. The MEB verifies that each transient has been properly categorized with respect to the

plant/system operating conditions of design, i.e., "normal." "upset," "emergency," "faulted" and "testing."

Any deviations that have not been justified to the satisfaction of the staff are identified and the finding is transmitted to the applicant with a request that, unless conformance with the MEB acceptance criteria is agreed upon, additional technical justification be submitted.

2. The information pertaining to computer programs which is presented in the applicant's SAR is reviewed as follows:

   a. The list of programs is evaluated to determine that the applicant has adequately described each program with respect to the type of analysis that is performed and the specific components to which the program is applied.

   b. The design control measures, which are required by 10 CFR Part 50, Appendix B, are reviewed for each program. The procedures outlined in subsection II.2.a, b, or c must be met for each program. Verification by the applicant that he has met the requirements of at least one of the above paragraphs is acceptable.

   c. The summary comparison of the results obtained from the use of each program which is not recognized and widely used (See subsection II.2) with either the results derived from a similar recognized and widely used program, a previously approved computer program, or results from test problems is reviewed and evaluated. Numerical results so derived should compare favorably enough to provide confidence in the validity of the program.

   Any deviations that have not been justified to the satisfaction of the staff are identified and the finding is transmitted to the applicant with a request that, unless conformance with the MEB acceptance criteria is agreed upon, additional technical justification be submitted.

3. If the applicant elects to use experimental stress analysis techniques in lieu of theoretical stress analyses, sufficient information must be presented in the SAR to demonstrate that the requirements of Appendix II to Reference 4, as they apply to the conditions set forth in the "Design Specifications" have been met.

4. If the applicant employs an elastic-plastic method of analysis to evaluate the design of safety-related Code or non-Code items for the faulted plant condition (NB-3225 and Appendix F of Reference 4), the review covers the following points:

   a. The applicant must demonstrate that the stress-strain relationship for component materials that will be used in the analysis is valid. The ultimate strength values at service temperature must be justified.

b. The analytical procedures to be used in the analysis are reviewed to determine the validity of the analysis. If a computer program is used, the applicable requirements of II.2 above shall be met.

c. If elastic, elastic-plastic or limit analysis methods are used for components in conjunction with elastic or elastic-plastic system analyses, the basis upon which these procedures are used are reviewed. The applicant shall provide assurance that the calculated item or item support deformations and displacements do not violate the corresponding limits and assumptions on which the methods used for the system analysis are based. (For example, current small deformation methods of analysis typically tend to have acceptable effective strain limits up to 5 percent and large deformation methods up to 15 percent.)

Any deviations that have not been justified to the satisfaction of the staff are identified and the finding is transmitted to the applicant with a request that, unless conformance with the MEB acceptance criteria is agreed upon, additional technical justification be submitted.

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information has been provided in accordance with this SRP section, and that his evaluation supports conclusions of the following type, to be included in the staff's safety evaluation report:

"The criteria used in the methods of analysis that the applicant has employed in the design of all seismic Category I ASME Code Class 1, 2, 3, and CS components, component supports, reactor internals and other non-Code items are in conformance with established technical positions and criteria which are acceptable to the Regulatory staff and satisfy the applicable portions of General Design Criteria 14 and 15.

"The use of these criteria in defining the applicable transients, computer codes used in analyses, analytical methods, and experimental stress analysis methods provides assurance that the stresses, strains, and displacements calculated for the above-noted items are as accurate as the current state-of-the-art permits and are adequate for the design of these items."

V. REFERENCES

1. 10 CFR Part 50, Appendix A, General Design Criterion 14, "Reactor Coolant Pressure Boundary."

2. 10 CFR Part 50, Appendix A, General Design Criterion 15, "Reactor Coolant System Design."

3. 10 CFR Part 50, Appendix B, "Quality Assurance Requirements for Nuclear Power Plants and Fuel Reprocessing Plants."

4. ASME Boiler and Pressure Vessel Code, Section III, Division I, "Nuclear Power Plant Components," American Society of Mechanical Engineers.

5. Regulatory Guide 1.68, "Initial Test Programs for Water-Cooled Reactor Power Plants."

6. Standard Review Plan Section 3.9.3, "ASME Code Class 1, 2, 3 Components, Component Supports, and Core Support Structures".

U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

SECTION 3.9.2 DYNAMIC TESTING AND ANALYSIS OF SYSTEMS, COMPONENTS, AND EQUIPMENT

REVIEW RESPONSIBILITIES

Primary - Mechanical Engineering Branch (MEB)

Secondary - Reactor Systems Branch (RSB)

I. AREAS OF REVIEW

MEB reviews the criteria, testing procedures, and dynamic analyses employed to assure the structural and functional integrity of piping systems, mechanical equipment, reactor internals, and their supports under vibratory loadings, including those due to fluid flow and postulated seismic events to assure conformance with General Design Criteria 1, 2, 4, 14, and 15. The staff review covers the following specific areas:

1. Piping vibration, thermal expansion, and dynamic effect testing should be conducted during startup testing. The systems to be monitored should include (a) ASME Code Class 1, 2, and 3 systems, (b) other high-energy piping systems inside Seismic Category I Structures, (c) high-energy portions of systems whose failure could reduce the functioning of any Seismic Category I plant feature to an unacceptable safety level, and (d) Seismic Category I portions of moderate-energy piping systems located outside containment. The supports and restraints necessary for operation during the life of the plant are considered to be parts of the piping system. The purpose of these tests is to confirm that these piping systems, restraints, components, and supports have been adequately designed to withstand flow-induced dynamic loadings under the steady-state and operational transient conditions anticipated during service and to confirm that normal thermal motion is not restrained. The test program description should include a list of different flow modes, a list of selected locations for visual inspections and other measurements, the acceptance criteria, and possible corrective actions if excessive vibration or indications of normal thermal motion restraint occurs.

2. Seismic qualification testing of safety-related mechanical equipment is required to assure its ability to function during and after a postulated seismic occurrence. At the construction permit (CP) stage, the staff review covers the following specific areas:

a. The criteria for seismic qualification such as the deciding factors for choosing test or analysis, the considerations defining the input motion, and the steps to demonstrate adequacy of the seismic qualification program.

b. The methods and procedures used to assure seismic Category I mechanical equipment operability during and after the safe shutdown earthquake (SSE), and to assure structural and functional integrity of the equipment after several occurrences of the operating basis earthquake. Included are mechanical equipment such as fans, pump drives, heat exchanger tube bundles, valve actuators, battery and instrument racks, control consoles, cabinets, panels, and cable trays.

c. The methods and procedures of analysis or testing for the supports for the seismic Category I mechanical equipment listed above, and the procedures used to account for the possible amplification of loads (amplitude and frequency content) under seismic conditions.

At the operating license (OL) stage, the staff reviews the results of tests and analyses to assure the proper implementation of the criteria established in the CP review, and to demonstrate adequate seismic qualification.

3. Dynamic responses of structural components within the reactor vessel caused by steady-state and operational flow transient conditions should be analyzed for prototype (first of a design) reactors. Generally, this analysis is not required for non-prototypes except that segments of an analysis may be necessary if there are substantial deviations from the prototype internals design. The purpose of this analysis is to predict the vibration behavior of the components, so that the input forcing functions and the level of response can be estimated. Before conducting the analyses, the specific locations for calculated responses, the considerations in defining the mathematical models, the interpretation of analytical results, the acceptance criteria, and the methods of verifying predictions by means of tests should be determined. If the reactor internal structures are a non-prototype design, reference should be made to the results of tests and analyses for the prototype reactor and a brief summary of the results should be given.

4. Flow-induced vibration testing of reactor internals should be conducted during the preoperational and startup test program. The purpose of this test is to demonstrate that flow-induced vibrations similar to those expected during operation will not cause unanticipated flow-induced vibrations of significant magnitude or structural damage. The test program description should include a list of flow modes, a list of sensor types and locations, a description of test procedures and methods to be used to process and interpret the measured data, a description of the visual inspections to be made, and a comparison of the test results with the analytical predictions. If the reactor internal structures are a non-prototype design, reference should be

made to the results of tests and analyses for the prototype reactor and a brief summary of the results should be given.

5. Dynamic system analyses should be performed to confirm the structural design adequacy and ability, with no loss of function, of the reactor internals and unbroken loops of the reactor coolant piping to withstand the loads from a loss-of-coolant accident (LOCA) in combination with the SSE. The staff review covers the methods of analysis, the considerations in defining the mathematical models, the descriptions of the forcing functions, the calculational scheme, the acceptance criteria, and the interpretation of analytical results.

6. A discussion should be provided which describes the methods to be used to correlate results from the reactor internals vibration test with the analytical results from dynamic analyses of the reactor internals under steady-state and operational flow transient conditions.

   In addition, test results from previous plants of similar characteristics may be used to verify the mathematical models used for the loading condition of LOCA in combination with the SSE by comparing such dynamic characteristics as the natural frequencies. The staff review covers the methods to be used for comparison of test and analytical results and for verification of the analytical models.

Computer programs used in the analyses discussed in this plan are reviewed in accordance with SRP Section 3.9.1.

The RSB verifies on request that (1) the various flow modes to be used to conduct the vibration test of the reactor internals are representative of the steady-state and operational transient conditions anticipated for the reactor during its service, and (2) that an acceptable hydraulic analysis has been used to determine the loads acting on the reactor coolant system piping and the reactor internals.

II. ACCEPTANCE CRITERIA

To fulfill in part the design requirements for safety-related structures, systems, and components set forth in General Design Criteria 1, 2, 4, 14, and 15, the acceptance criteria for the areas of MEB review are as follows:

1. Vibration, thermal expansion, and dynamic effects testing should be conducted during startup functional testing for specified high-and moderate-energy piping, and their supports and restraints. The purpose of these tests is to confirm that the piping, components, restraints, and supports have been designed to withstand the dynamic loadings and operational transient conditions that will be encountered during service as required by the Code and to confirm that no unacceptable restraint of normal thermal motion occurs.

An acceptable test program to confirm the adequacy of the designs should consist of the following:

a. A list of systems that will be monitored.

b. A listing of the different flow modes of operation and transients such as pump trips, valve closures, etc. to which the components will be subjected during the test. (For additional guidance see Reference 8.) For example, the transients associated with the reactor coolant system heatup tests should include, but not necessarily be limited to:

   (1) Reactor coolant pump start.

   (2) Reactor coolant pump trip.

   (3) Operation of pressure-relieving valves.

   (4) Closure of a turbine stop valve.

c. A list of selected locations in the piping system at which visual inspections and measurements (as needed) will be performed during the tests. For each of these selected locations, the deflection (peak-to-peak) or other appropriate criteria, to be used to show that the stress and fatigue limits are within the design levels, should be provided.

d. A list of snubbers on systems which experience sufficient thermal movement to measure snubber travel from cold to hot position.

e. A description of the thermal motion monitoring program, i.e., verification of snubber movement, adequate clearances and gaps, including acceptance criteria and how motion will be measured.

f. If vibration is noted beyond the acceptance levels set by the criteria of c., above, corrective restraints should be designed, incorporated in the piping system analysis, and installed. If, during the test, piping system restraints are determined to be inadequate or are damaged, corrective restraints should be installed and another test should be performed to determine that the vibrations have been reduced to an acceptable level. If no snubber piston travel is measured at those stations indicated in d., above, a description should be provided of the corrective action to be taken to assure that the snubber is operable.

2. A test program is required to confirm the ability of all seismic Category I mechanical equipment to function as needed during and after an earthquake of magnitude up to and including the SSE.

a. Analysis without testing is acceptable if structural integrity alone can assure the intended function. When a complete seismic test is impracticable, a combination of test and analysis is acceptable.

b. Equipment should be tested in the operational condition. Loadings simulating those of plant normal operation, such as thermal and flow-induced loadings, if any, should be concurrently superimposed upon the seismic loading. Operability should be verified during and after the test.

c. The characteristics of the seismic input motion should be specified by one of the following:

(1) Response spectrum.

(2) Power spectral density function.

(3) Time history.

Such characteristics, as derived from the structure or system seismic analysis, should be representative of the seismic input motion at the equipment mounting locations.

d. The test input motion should be characterized in the same manner as the seismic input motion, and the conservatism in amplitude and frequency content should be demonstrated.

e. Seismic excitations generally have a broad frequency content. Multi-Frequency input motion should be used in the testing. However, single frequency input, such as sine "beats," may be applicable provided one of the following conditions are met:

(1) The characteristics of the seismic input motion indicate that the motion is dominated by one frequency (e.g., by structural filtering effects).

(2) The anticipated response of the equipment is adequately represented by one mode.

(3) The test input motion has sufficient intensity and duration to excite all modes to the required amplitudes, such that the testing response spectra will envelope the corresponding response spectra of the individual modes.

The test input motion should be applied to one vertical axis and one principal horizontal axis (or two orthogonal horizontal axes) simultaneously unless it can be demonstrated that the equipment response in the vertical direction is not sensitive to the vibratory motion in the horizontal direction, and vice

versa. The time phasing of the inputs in the vertical and horizontal directions must be such that a purely rectilinear resultant input is avoided. An acceptable alternative is to have vertical and horizontal inputs in-phase, and then repeated with inputs 180 degrees out-of-phase. In addition, the test must be repeated with the equipment rotated 90 degrees horizontally.

g. Dynamic coupling between the equipment and related sytems, if any, such as connected piping and other mechanical components, should be considered.

h. The fixture design should meet the following requirements:

(1) Simulate the actual service mounting.

(2) Cause no extraneous dynamic coupling to the test item.

i. The in situ application of vibratory devices to superimpose the seismic vibratory loadings on a complex active device for operability testing is acceptable if it is shown that a meaningful test can be made in this way.

j. The test program may be based upon selectively testing a representative number of mechanical components according to type, load level, size, etc., on a prototype basis.

k. Analyses or tests should be performed for all supports of mechanical equipment to assure their structural capability to withstand seismic excitation. The analytical results must include the following:

(1) The required input motions to the mounted equipment should be obtained and characterized in the manner as stated in subsection 2.c, above.

(2) The combined stresses of the support structures should be within the limits of the ASME Code, Subsection NF, "Component Support Structures."

l. Supports should be tested with equipment installed or with an equivalent mass that simulates the equipment dynamic coupling to the support. If the equipment is installed in a nonoperating condition for the support test, the response at the equipment mounting location should be characterized in the manner as stated in subsection 2.c, above. In such a case, the equipment should be tested separately for operability and the actual input to the equipment should be more conservative in amplitude and frequency content than the monitored response.

m. The requirements of subsections 2.c, 2.d, 2.e, 2.f, and 2.h, above, are applicable when tests are conducted on equipment supports.

3. The following guidelines, in addition to Regulatory Guide 1.20 (Reference 7), apply to the analytical solutions to predict vibrations of reactor internals for prototype plants. Generally, this analysis is required only for prototype designs.

a. The results of vibration calculations for a prototype reactor should consist of the following:

(1) Dynamic responses to operating transients at critical locations of the internal structures should be determined and, in particular, at the locations where vibration sensors will be mounted on the reactor internals. For each location, the maximum response, the modal contribution to the total response, and the response causing the maximum stress amplitude should be calculated.

(2) The dynamic properties of internal structures, including the natural frequencies, the dominant mode shapes, and the damping factors should be characterized. If analyses are performed on a component structural element basis, the existence of dynamic coupling among component structure elements should be investigated.

(3) The response characteristics, such as the dependence on hydrodynamic excitation forces, the flow path configuration, coolant recirculation pump frequencies, and the natural frequencies of the internal structures, should be identified.

(4) Acceptance criteria for allowable responses should be established, as should criteria for the location of vibration sensors. Such criteria should be related to the Code allowable stresses, strains, and limits of deflection that are established to preclude loss of function with respect to the reactor core structures and fuel assemblies.

b. The forcing functions should account for the effects of transient flow conditions and the frequency content. Acceptable methods for formulating forcing functions for vibration prediction include the following:

(1) Analytical method: based on standard hydrodynamic theory, the governing differential equations for vibratory motions should be developed and solutions obtained with appropriate boundary conditions and parameters. This method is acceptable where the geometry along the fluid flow paths is mathematically tractable.

(2) Test-analysis combination method: based on data obtained from plant tests or scaled model tests (e.g., velocity or pressure distribution data), forcing functions should be formulated which will include the effects of complex flow path configurations and wide variations of pressure distributions.

(3) Response-deduction method: based on a derivation of response characteristics from plant or scaled model test data, forcing functions should be formulated. However, since such functions may not be unique, the computational procedures and the basis for the selection of the representative forcing functions should be described.

c. Acceptable methods of obtaining dynamic responses for vibration predictions are as follows:

(1) Force-response computations are acceptable if the characteristics of the forcing functions are predetermined on a conservative basis and the mathematical model of the reactor internals is appropriately representative of the design.

(2) If the forcing functions are not predetermined, either a special analysis of the response signals measured from reactor internals of similar design may be performed to predict amplitude and modal contributions, or parameter studies useful for extrapolating the results from tests of internals or components of similar designs based on composite statistics may be used.

d. Vibration predictions should be verified by test results. If the test results differ substantially from the predicted response behavior, the vibration analysis should be appropriately modified to improve the agreement with test results and to validate the analytical method as appropriate for predicting responses of the prototype unit, as well as of other units where confirmatory tests are to be conducted.

4. The preoperational vibration test program for the internals of a prototype (first of a design) reactor should conform to the requirements for a prototype test, as specified in Regulatory Guide 1.20, including vibration prediction, vibration monitoring, data reduction, and surface inspection. The test program should include, but not necessarily be limited to the following:

a. The vibration testing should be conducted with the fuel elements in the core or with dummy elements which provide equivalent dynamic effects and flow characteristics. Testing without fuel elements in the core may be acceptable if it can be demonstrated that testing in this mode is conservative.

b. A brief description of the vibration monitoring instrumentation should be provided, including instrument types and diagrams of locations, which should include the locations having the most severe vibratory motions or having the most effect on safety functions.

c. The planned duration of the test for the normal operation modes to assure that all critical components are subjected to at least 10^6 cycles of vibration should be provided. For instance, if the lowest response frequency of the core internal structures is 10 Hz, a total test duration of 1.2 days or more will be acceptable.

d. Testing should include all of the different flow modes of normal operation and upset transients. The proposed set of flow modes are acceptable if they provide a conservative basis for determining the dynamic response of the reactor internals and are reviewed by RSB on request.

e. The methods and procedures to be used to process the test data to obtain a meaningful interpretation of the core structure vibration behavior should be provided. Vibration interpretation should include the amplitude, frequency content, stress state, and the possible effects on safety functions.

f. Vibration predictions, test acceptance criteria and bases, and permissible deviations from the criteria should be provided before the test.

g. Visual and nondestructive surface inspections should be performed after the completion of the vibration tests. The inspection program description should include the areas subject to inspection, the methods of inspection, the design access provisions to the reactor internals, and the equipment to be used for performing such inspections. These inspections should be conducted preferably following the removal of the internals from the reactor vessel. Where removal is not feasible, the inspections should be performed by means of equipment appropriate for in situ inspection. The areas inspected should include all load-bearing interfaces, core restraint devices, high stress locations, and locations critical to safety functions.

For internals of subsequent reactors that have the same design, size, configuration, and operating conditions as the prototype reactor internals, the vibration test program should conform to the requirements of the appropriate non-prototype program as specified in Regulatory Guide 1.20.

5. Dynamic system analyses should be performed to confirm the structural design adequacy of the reactor internals and the reactor coolant piping (unbroken loops) to withstand the dynamic loadings of the most severe LOCA in combination with the SSE. Where a substantial separation between the forcing frequencies of the LOCA (or SSE) loading and the natural frequencies of the internal structures can be demonstrated, the analysis may treat the loadings statically.

The most severe dynamic effects from LOCA loadings are generally found to result from a postulated double-ended rupture of a primary coolant loop near a reactor vessel inlet or outlet nozzle with the reactor in the most critical normal operating

mode. However, all other postulated break locations should be evaluated and the location producing the controlling effects should be identified.

Mathematical models used for dynamic system analysis for LOCA in combination with the SSE effects should include the following:

a. Modeling should include reactor internals and dynamically related piping, pipe supports, components, and fluid-structure interaction effects when applicable. Typical diagrams and the basis of modeling should be developed and described.

b. Mathematical models should be representative of system structural characteristics, such as the flexibility, mass inertia effect, geometric configuration, and damping (including possible coexistence of viscous and Coulomb damping).

c. Any system structural partitioning and directional decoupling employed in the dynamic system modeling should be justified.

d. The effects of flow upon the mass and flexibility properties of the system should be discussed.

Typical diagrams and the basis for postulating the LOCA-induced forcing function should be provided, including a description of the governing hydrodynamic equations and the assumptions used for mathematically tractable flow path geometries, tests for determining flow coefficients, and any semiempirical formulations and scaled model flow testing for determining pressure differentials or velocity distributions. The acceptabilty of the hydraulic analysis, as reviewed by RSB on request, is based on established engineering practice and generic topical reviews performed by the staff.

The methods and procedures used for dynamic system analyses should be described, including the governing equations of motion and the computational scheme used to derive results. Time domain forced-response computation is acceptable for both LOCA and SSE analyses. The response spectrum modal analysis method may be used for SSE analysis.

The stability of elements in compression, such as the core barrel and the control rod guide tubes under outlet pipe rupture loadings should be investigated.

Either response spectra or time histories may be used for specifying seismic input motions of the SSE at the reactor core supports.

The criteria for acceptance of the analytical results are as provided in SRP Sections 3.9.3 and 3.9.5.

6. Regarding the correlation to be made of tests and analyses of reactor internals, a discussion covering the following items should be provided:

   a. Comparison of the measured response frequencies with the analytically obtained natural frequencies of the reactor internals for possible verification of the mathematical model used in the analysis.

   b. Comparison of the analytically obtained mode shapes with the shape of measured motion for possible identification of the modal combination or verification of a specific mode.

   c. Comparison of the response amplitude time variation and the frequency content obtained from test and analysis for possible verification of the postulated forcing function.

   d. Comparison of the maximum responses obtained from test and analysis for possible verification of stress levels.

   e. Comparison of the mathematical model used for dynamic system analysis under operational flow transients and under the combined LOCA and SSE loadings, to note similarities.

III. REVIEW PROCEDURES

The reviewer will select and emphasize material from the procedures described below, as may be appropriate for a particular case.

General Design Criteria 1 2, 4, 14, and 15 state that all structures, system and components important to safety should be designed and tested to assure that safety functions can be performed in the event of operational transients, earthquakes, and LOCA loadings.

Under these guidelines, the staff reviews the treatment of dynamic responses of safety-related piping systems and reactor internal structures by the following procedures:

1. During the CP stage, the PSAR is reviewed to assure that the applicant is provided a commitment to conduct a piping steady-state vibration, thermal expansion and operational transient test program. The applicants program description should be sufficiently comprehensive to contain all the elements of an acceptable program as described in subsection II.1.

   During the OL stage, the FSAR is reviewed to assure that the applicants PSAR commitment is fulfilled and the program is developed in sufficient detail. The reviewer should be assured that the applicants program is sufficiently developed to:

(a) Establish the rationale and bases for the acceptance criteria and selection of locations to monitor pipe motions.

(b) Provide the displacement or other appropriate limits at locations to be monitored.

(c) Describe the techniques and instruments (as needed) for monitoring or measuring pipe motions.

(d) Assure that the NRC will be provided documentation of any corrective action resulting from the test and conformation by additional testing that substantiates effectiveness of the corrective action.

2. At the CP stage, the staff reviews the program which the applicant has described in the preliminary safety analysis report (PSAR) for the seismic qualification of all seismic Category I mechanical equipment. The program is measured against the requirements listed in the acceptance criteria subsection of this SRP section. Of particular interest are the proper use of test and analytical procedures. Equipment which is too complex for reliable mathematical modeling should be tested unless the analytical procedures and corresponding design are demonstrated to be conservative. Both the test and the analysis methods are reviewed for assurance that all important modes of response have been excited in tests or considered in analyses. Proper application of test input motions so as to envelop the required input, whether in terms of response spectra, power spectral density, or time history, and in all necessary directions, is verified. The use or treatment of supports is also reviewed.

   At the OL stage, the staff reviews the program again as described by the applicant in the Final Safety Analysis Report (FSAR). In addition, the FSAR is reviewed for documentation of successful implementation of the seismic qualification program, including test and analysis results. Also, the acceleration levels used in the tests and in the analyses are reviewed for assurance that they equal or exceed the acceleration at the equipment mounting locations derived from structural response studies of the plant structure as built or as designed.

3. At the CP stage, the applicant should commit to performing an analysis of the vibration of the reactor internal structures if they are designated as a prototype design. A brief description of the methods and procedures to be used for the analysis should be provided.

   At the OL stage, a detailed dynamic analysis should be provided for a prototype design, to be used for vibration prediction prior to the performance of preoperational vibration tests. Acceptance of the analysis is based on the technical soundness of the analytical method and procedures used and the degree of conformance to the acceptance criteria listed above. In addition, the analysis is verified by correlation with the test results when these are available.

For both CP and OL stages, if the reactor internal structures are a non-prototype design, then reference should be made to the reactor which is prototypical of the reactor being reviewed. A brief summary of test and analysis results for the prototype should be given, Alternatively, the information may be contained in another applicable document, such as a topical report, to which reference should be made.

4. At the CP stage, the staff review of the program for preoperational vibration testing of reactor internals for flow-induced vibrations includes the following matters:

   a. The applicant should clarify his intention to perform either a prototype test or a non-prototype test.

   b. If the plant is designated as a prototype, a brief description of the preoperational vibration test program should be provided. The staff review will be based on the conformance of this program to the requirements as listed in subsection II.4, above.

   c. If the plant is a non-prototype, the applicant should identify the existing pla similar design that is the prototype plant. The staff reviews the validity of the designated prototype, including any design difference of reactor internal structures from the prototype plant to verify that any design modifications do not substantially alter the behavior of the flow transients and the response of the reactor internals. Additional detailed analysis, scaled model tests, or installation of some instrumentation during the confirmatory test may be required in order to complete the review. In addition, the applicant should commit to performing the prototype test if adequate test results are not obtained on a timely basis for the designated prototype.

   At the OL stage, the staff review includes the following procedures:

   a. A detailed preoperational vibration test program and the tentative schedule to perform the test are reviewed. If elements of the program differ substantially from the guidelines specified in Regulatory Guide 1.20, discussion of the need and justification for the differences should be given. On request, RSB verifies that the flow modes to be used are acceptable.

   b. For a prototype plant, the review covers the acceptability of vibration prediction, the visual surface inspection procedures, the details of instrumentation for vibration monitoring, the methods and procedures to process the test results, and possible supplementary tests, such as component vibration tests, flow tests, and scaled model tests.

   c. For a non-prototype plant, the staff verifies the applicability of the designated prototype, including the design similarity of the reactor internal structures to the prototype. Additional detailed analysis, scaled model tests, or

vibration monitoring in the confirmatory tests may be needed in order to complete the review.

5. In the CP stage review of the dynamic analysis of the reactor internals and unbroken loops of the reactor coolant piping under faulted condition loadings, the applicant commits to perform this analysis or identifies the applicable document, generally in form of a topical report, containing the required information. A brief description of the scope and methods of analysis should be provided.

   In the OL review, the staff reviews the detailed information to confirm that an adequate analysis has been made of the capability of reactor internal structures and unbroken loops to withstand dynamic loads from the most severe LOCA in combination with the safe shutdown earthquake. The staff review covers the analytical methods and procedures, the basis of the forcing functions, the mathematical models to represent the dynamic system, and the stability investigations for the core barrel and essential compressive elements. Acceptance of the analysis is based on (1) the technical soundness of the analytical methods used, (2) the degree of conformance to the acceptance criteria listed above, and (3) verification that stresses under the combined loads are within allowable limits of the applicable code and deformations are within the limits set to assure the ability of reactor internal structures and piping to perform needed safety functions. On request, RSB verifies that an acceptable hydraulic analysis has been used.

6. MEB reviews the program which the applicant has committed to implement as part of the preoperational test procedure, principally to correlate the test measurements with the analytically predicted flow-induced dynamic response of the reactor internals. MEB reviews the applicant's statements in this areas to assure that there is a commitment to submit a report on a timely basis. The report should summarize the analyses and test results so that MEB can review the compatibility of the results from tests and analyses, the consistency between mathematical models used for different loadings, and the validity of the interpretation of the test and analysis results.

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information has been provided and that the review supports conclusions of the following type, to be included in the staff's safety evaluation report:

"The vibration, thermal expansion, and dynamic effects test program which will be conducted during startup and initial operation on specified high- and moderate-energy piping, and all associated systems, restraints and supports is an acceptable program. The tests provide adequate assurance that the piping and piping restraints of the system have been designed to withstand vibrational dynamic effects due to valve closures, pump trips, and other operating modes associated with the design

basis flow conditions. In addition, the tests provide assurance that adequate clearances and free movement of snubbers exist for unrestrained thermal movement of piping and supports during normal system heatup and cooldown operations. The planned tests will develop loads similar to those experienced during reactor operation. Compliance with this test program constitutes an acceptable basis for fulfilling, in part, the requirements of General Design Criteria 14 and 15.

"The capability of safety-related mechanical equipment to perform necessary protective actions in the event of a safe shutdown earthquake (SSE) is essential for plant safety. The qualification testing program which will be implemented for seismic Category I mechanical equipment provides adequate assurance that such equipment will function properly under the loads from vibratory forces imposed by the safe shutdown earthquake and under the conditions of post-earthquake operation. This program constitutes an acceptable basis for satisfying, in part, the requirements of General Design Criterion 2.

"The preoperational vibration program planned for the reactor internals provides an acceptable basis for verifying the design adequacy of these internals under test loading conditions comparable to those that will be experienced during operation. The combination of tests, predictive analysis, and post-test inspection provide adequate assurance that the reactor internals will, during their service lifetime, withstand the flow-induced vibrations of reactor operation without loss of structural integrity. The integrity of the reactor internals in service is essential to assure the proper positioning of reactor fuel assemblies and unimpaired operation of the control rod assemblies to permit safe reactor operation and shutdown. The conduct of the preoperational vibration tests is in conformance with the provisions of Regulatory Guide 1.20 and constitutes an acceptable basis for demonstrating design adequacy of the reactor internals, and satisfies the applicable requirements of General Design Criteria 1 and 4.

"The dynamic system analysis to be performed provides an acceptable basis for confirming the structural design adequacy of the reactor internals and unbroken piping loops to withstand the combined dynamic loads of postulated loss of coolant accidents (LOCA) and the safe shutdown earthquake (SSE) and the combined loads of a postulated main steam line rupture and SSE (for a BWR). The analysis provides adequate assurance that the combined stresses and strains in the components of the reactor coolant system and reactor internals will not exceed the allowable design stress and strain limits for the materials of construction, and that the resulting deflections or displacements at any structural elements of the reactor internals will not distort the reactor internals geometry to the extent that core cooling may be impaired. The methods used for component analysis have been found to be compatible with those used for the systems analysis. The proposed combinations of component and system analyses are, therefore, acceptable. The assurance of structural integrity of the reactor internals under LOCA conditions for the most adverse postulated loading event provides added confidence that the design will withstand a

spectrum of lesser pipe breaks and seismic loading events. Accomplishment of the dynamic system analysis constitutes an acceptable basis for satisfying the applicable requirements of General Design Criteria 2 and 4."

For the FSAR, the review should provide justification for a finding similar to that stated above with the phrase "will be implemented" modified to read "has been implemented."

V. REFERENCES

1. 10 CFR Part 50, Appendix A, General Design Criterion 1, "Quality Standards and Records."

2. 10 CFR Part 50, Appendix A, General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena."

3. 10 CFR Part 50, Appendix A, General Design Criterion 4, "Environmental and Missile Design Bases."

4. 10 CFR Part 50, Appendix A, General Design Criterion 14, "Reactor Coolant Pressure Boundary."

5. 10 CFR Part 50, Appendix A, General Design Criterion 15, "Reactor Coolant System Design."

6. ASME Boiler and Pressure Vessel Code, Section III, "Nuclear Power Plant Components," American Society of Mechanical Engineers.

7. Regulatory Guide 1.20, "Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing."

8. Regulatory Guide 1.68, "Preoperational and Initial Startup Test Programs for Water-Cooled Power Reactors."

NUREG-75/087

U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

SECTION 3.9.3 ASME CODE CLASS 1, 2, AND 3 COMPONENTS, COMPONENT SUPPORTS, AND CORE SUPPORT STRUCTURES

REVIEW RESPONSIBILITIES

Primary - Mechanical Engineering Branch (MEB)

Secondary - Reactor Systems Branch (RSB)
Auxiliary and Power Conversion Systems Branch (APCSB)

I. AREAS OF REVIEW

Information is presented in the applicant's safety analysis report (SAR) and is reviewed by the MEB concerning the structural integrity and operability of pressure-retaining components, their supports, and core support structures which are designed in accordance with the rules of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section III (hereafter "the Code").

The staff review covers the following specific areas:

1. Loading Combinations, Design Transients, and Stress Limits

The design loading combinations (e.g., design loads or anticipated operational loads including design transients in combination with loads calculated to result from postulated accidents and seismic events) specified for Code constructed items designated as Code Class 1, 2, 3 and CS are reviewed to determine that they have been appropriately categorized with respect to "normal," "upset," "emergency," or "faulted" plant conditions. In addition, the staff review determines that the design stress limits and deformation criteria associated with each of the plant operating conditions and appropriate component operating conditions comply with the applicable limits specified in the Code and other criteria. Design stress limits which allow inelastic deformation of Code Class 1, 2, 3 and CS items are evaluated as are the justifications for the proposed design procedures. Piping which is "field run" should be included. Internal parts of components such as valve discs and seats and pump shafting subjected to dynamic loading during operation of the component should be included.

2. Pump and Valve Operability Assurance Programs

The component operability assurance program is intended to assure the operability of Code Class 1, 2, and 3 active valves, 2 inches and greater in nominal pipe size, and the ability of active pumps to function under plant conditions where their operation is relied upon for plant shutdown or for mitigating the consequences of an accident. The program is evaluated with respect to test and analytical methods and combinations thereof. The test program may include prototype testing, either individually under

simulated test conditions in the shop, or in situ after installation. The staff review covers the following specific information and provisions of the component operability assurance program:

a. A listing of active Class 1, 2, and 3 valves and pumps identified by system and "active" function. The Auxiliary and Power Conversion Systems Branch and the Reactor Systems Branch confirm the acceptability of the listing for Class 1, 2 and 3 pumps and valves.

b. The components, in terms of size, type, design, and manufacturer, for which one prototype test is proposed to confirm operability.

c. The components for which prototype test results are available, from applications for other plants or other sources, and the comparisons that show that the test conditions are equivalent to the plant design conditions.

d. The identification of combinations of plant conditions and loads which the active component is expected to withstand during the "active" function (such conditions are generally specified in the component design specification, as required by Code rules).

e. The test conditions and loads that will be imposed on components to confirm operability, and the comparisons to show that these are representative of plant conditions and loads (where more than one set of conditions may be applicable, the most adverse or bounding combinations should be evaluated).

f. The extent to which analytical methods will be used in lieu or in partial fulfillment of the provisions of the component operability assurance program.

3. Design and Installation of Pressure Relief Devices

The design and installation criteria applicable to the mounting of pressure relief devices (safety valves and relief valves) for the overpressure protection of Code Class 1 and Class 2 components are reviewed. The review includes evaluation of the applicable loading conditions and design stress criteria as related to the normal, upset, and emergency plant operating conditions. The design review extends to consideration of the means provided to accommodate the rapidly applied reaction force when a safety valve or relief valve opens, and the transient fluid-induced loads applied to the piping downstream of a safety or relief valve in a closed discharge piping system.

The design of safety and relief valve systems is reviewed with respect to the load combinations imposed on the safety or relief valves, upstream piping or header, downstream or vent piping, and system supports.

The loading combinations should identify the most severe combination of the applicable loads due to internal fluid pressure, dead weight of valves and piping, thermal load under heatup, steady state and transient valve operation, reaction forces when valves are discharging (thrust, bending, and torsion), and seismic forces, i.e., operating basis earthquake (OBE) and safe shutdown earthquake (SSE).

The structural response of the piping and support system is reviewed with particular attention to the dynamic or time-history analyses employed in evaluating the

appropriate support and restraint stiffness effects under dynamic loadings when valves are discharging.

Where the use of hydraulic snubbers is proposed, the snubber performance characteristics are reviewed to assure that their effects have been considered in the analyses under steadystate valve operation and repetitive load applications caused by cyclic valve opening and closing during the course of a pressure transient.

The Auxiliary and Power Conversion Systems Branch verifies that the number and size of valves specified for the steam and feedwater systems have adequate pressure relieving capacity as confirmed by their review and evaluation of the "Overpressure Protection Analysis" that has been prepared in accordance with the requirements of the Code.

The Reactor Systems Branch verifies that the number and size of valves specified for the reactor coolant pressure boundary have adequate pressure relieving capacity as confirmed by their review and evaluation of the "Report on Overpressure Protection" that has been prepared in accordance with the requirements of the Code. The design criteria for pressure-relieving devices which may have an active function during and after a faulted plant condition are judged also against the requirements of the component operability assurance program.

4. Component Supports

The review of information submitted by the applicant includes an evaluation of Code Class 1, 2, and 3 component supports. The review includes an assessment of the design and structural integrity of the supports and their effect on the operability of active components. The review addresses three types of supports: plate and shell, linear, and component standard types, and their function.

Nuclear power plant component supports are those metal supports which are designed to transmit loads from the pressure-retaining boundary of the component.

Linear supports covered in this plan are those which are not included in Standard Review Plan 3.8.3.

II. ACCEPTANCE CRITERIA

The criteria by which the areas of review defined in Section I are judged to be acceptable are as follows:

1. Loading Combinations, Design Transients, and Stress Limits

The plant and component operating conditions, design transients, and design loading combinations considered for each system should be sufficiently defined to provide the basis for design of Code Class 1, 2, 3 and CS items for all conditions and events expected over the service lifetime of the plant and should satisfy the requirements of General Design Criteria 1, 2, and 4.

The acceptability of the combination of loading conditions and design transients applicable to the design of Code constructed items within a system, including the

categorization of the appropriate plant and component operating condition for each initiating event (i.e., LOCA, SSE, pipebreak, etc.) which may be used with each loading combination, is judged by comparison with the positions stated in Reference 5, and with appropriate standards acceptable to the staff developed by professional societies and standards organizations. When these combinations have been established, the corresponding stress limits which may be applied to the design of Code constructed items are as specified in the appropriate subsections of Division 1 of Section III of the ASME Code. The need for more conservative stress limits for active components and their supports should be considered in the context and with the other features of the operability assurance program.

2. Pump and Valve Operability Assurance Program

The operation of certain pumps and valves is relied upon to shut down the plant or mitigate the consequences of an accident. These are termed "active" pumps and valves. Certain of these active pumps and valves may be required to function coincidentally with the postulated accident or event. Other active pumps and valves may be required to function only after a postulated accident or event has occurred. Acceptable procedures for demonstrating the operability of active pumps and valves during or after postulated accidents or natural events follow:

a. Pumps and Valves Whose Operability is Required During an Accident or Event

This section presents acceptable procedures for demonstrating the operability of active pumps and valves during accident or event conditions. The pump or valve includes the pressureretaining body, all internal structures, and all appurtenances necessary for component operation. The most desirable operability assurance program consists of testing the pump or valve under simulated accident or event loadings (pressure, external loads due to SSE, etc.) and environmental conditions (temperature, humidity, etc.). When this approach is not practicable, other conservative procedures may be employed. These include more elementary testing or a combination of testing and analysis. In addition, design of the pump and valve supports must be considered and accounted for in the testing and analysis to demonstrate operability. The design specification must be written to include the requirements for operability under the accident conditions; assurance of this must be provided in the SAR. Design stress limits discussed in II.1 are acceptable for active components and their supports if considered in the operability assurance program. The following programs provide an acceptable approach to demonstrate the operability of active pumps and valves requires to operate during an accident or event.

(1) Testing

The following features should be incorporated into a test program:

(a) An individual pump or valve is tested in the manufacturer's shop, or in situ following installation in the system provided the test conditions simulate those conditions under which the "active" function is required.

(b) The pump or valve is tested in the operational mode.

(c) The test program is based upon selectively testing a representative number of pumps or valves according to type, load level, size, etc. on a prototype basis. Pumps or valves that can be demonstrated to be equivalent (e.g., similar nondestructive examination program, materials, weldments, pressure, and temperature) to a prototype pump or valve, may be exempted from testing provided the test results of the prototype pump or valve are documented and available, and the loading conditions for the exempted pump or valve are equivalent to or less severe than those imposed during testing of the prototype pump or valve.

(d) The characteristics of the required seismic or accident input motion are properly specified as obtained from the system dynamic analysis and are representative of the input motion at the component mounting locations. The characteristics of the required input motion are specified by response spectrum, power spectral density function, or time history. Such characteristics, as derived from the structures or systems analysis, are representative of the input motion at the equipment mounting locations. Seismic excitation generally has a broad frequency content. Random vibration input motion should be used. However, single frequency input motions, such as sine "beats," are acceptable provided the characteristics of the required input motion indicate that the motion is dominated by one frequency (e.g., by structural filtering effects), the anticipated response of the equipment is adequately represented by one mode, or the input has sufficient intensity and duration to excite all modes to the required amplitudes such that the testing response spectra will envelope the corresponding response spectra of the individual modes.

(e) Seismic or accident input motion is applied to one vertical axis and one principal horizontal axis (or two orthogonal horizontal axes) simultaneously, unless it can be demonstrated that the equipment response in the vertical direction is not sensitive to vibratory motion in the horizontal direction, and vice versa. In the case of a single frequency input motion, the time phasing of the inputs in the vertical and horizontal directions must be such that a purely rectilinear resultant input is avoided.

(f) The characteristics of applicable environments such as temperature (at the accident condition) are taken into account.

(g) The fixture design simulates the actual service mounting (same stiffness characteristics) and causes no extraneous dynamic coupling to the test item.

(h) End loads are properly taken into account.

(i) Dynamic coupling to other related systems, if any, such as connected piping and other mechanical components, is considered.

The in situ application of vibratory devices to superimpose vibratory loadings on a complex active device is acceptable for operability assurance when it is shown that a meaningful test can be made in this

way, with due regard being given to the effects on other parts of the system.

If the dynamic testing of a pump or valve assembly proves to be impracticable, static testing (static application of loads) of the assembly is acceptable provided that the end loadings are conservatively applied and are equal to or greater than accident loads, all dynamic amplification effects are accounted for, the component is in the operating mode during and after the application of loads, and an adequate analysis is made to show the validity of the static application of loads.

(2) A Combination of Test and Analysis

(a) When complete testing is not practicable, a combination of test and analysis is acceptable. Simple and passive elements, such as valve and pump bodies and their related piping and supports may be analyzed to confirm structural integrity under accident loadings. However, complex active devices such as pump motors, valve operator and gate or disk assemblies, and other electrical, mechanical, pneumatic, or hydraulic appurtenances which are vital to the pump or valve operation must be tested for operability in accordance with the section above, or the Institute of Electrical and Electronics Engineers (IEEE) Standard IEEE 344-1975, as appropriate.

(b) The following analyses are acceptable provided they are correlated to classical problems, elementary laboratory tests, or in situ tests:

i. An analysis is performed to determine the seismic input to the valve or pump;

ii. An analysis is performed to determine the system natural frequencies and the movement of the pump or valve during the SSE.

iii. An analysis is performed to determine the pressure differential and the impact energy of a valve disc during a LOCA, and to verify the design adequacy of the disc.

iv. An analysis is performed to determine the forcing functions of the axial and radial loads imposed on a pump rotor due to a LOCA, such that combined LOCA and SSE effects on the shaft and rotor assembly can be evaluated.

v. An analysis is performed to determine the speed of the pump shaft as a result of postulated accidents and to compare it with the design critical speed.

vi. An analysis is performed to verify the design adequacy of the wall thickness of valve and pump pressurecontaining bodies.

vii. An analysis is performed to determine the natural frequencies of a pump shaft and rotor assembly to ascertain whether they are within the frequency range of the seismic excitations. If the minimum natural frequency of the assembly is beyond the excitation

frequencies, a static deflection analysis for the shaft is acceptable to account for SSE effects. If the assembly natural frequencies are close to the excitation frequencies, an acceptable dynamic analysis must be performed to determine the structural response of the assembly to the excitation frequencies.

(3) Design Adequacy of Pump and Valve Supports

(a) Analyses or tests are performed for all supports of pumps and valves to ensure their structural capability to withstand seismic excitation.

(b) The analytical results must include the required input motions to the mounted equipment which should be obtained and characterized in the manner specified by one of the following:

i. Response spectrum.
ii. Power spectral density function.
iii. Time history.

Such characteristics, as derived from the structures or systems seismic analysis, should be representative of the input motion at the equipment mounting locations. The analytical results must also show that the combined stresses of the support structures are within the limits of the Code, Subsection NF, "Component Supports."

(c) The support is tested with the pump or valve installed or with equivalent mass inertia effects. If the equipment is inoperative during the support test, the response at the equipment mounting locations is monitored and characterized in the manner stated in (b) above. In such a case the equipment is tested separately and the actual input to the equipment should be more conservative in amplitude and frequency content than the monitored response in the support test.

b. Pumps and Valves Whose Operability is Required After an Accident or Event

This section presents acceptable procedures for demonstrating the operability of active pumps and valves that are not required to operate coincident with an accident or event, but are required to operate following the accident or event. The applicant must identify those active pumps and valves considered to meet this description and justify such classification. Components that may operate or may inadvertently be operated coincident with an accident or event should meet the requirements of pumps and valves whose operability is required during an accident or event, unless the applicant can demonstrate by test or analysis that operation coincident with an accident or event will not impair the ability of the component to perform its required operation following an accident or event.

An acceptable operability assurance program for active pumps and valves whose operability is required only after an accident or event consists of design integrity and testing phases.

(1) Design Integrity

The integrity of active pumps and valves, whose operability is required only after an accident or event, is established by including in the design specification the requirement that the loads due to the accident (emergency or faulted plant conditions) shall be considered as normal loads for the active pump or valve. Design stress limits discussed in II.1 above are acceptable for active components and their supports if considered in the operability assurance program.

(2) Testing

Operability assurance testing of active pumps and valves, whose operability is required after an accident or event, is required only for the component appurtenances vital to the operation of the component, such as operators, motors, switches, relays, etc. The testing of such items may be accomplished independently of the component provided all coupling effects are identified and properly factored into the tests as boundary conditions. Such qualification testing should be in accordance with the requirements of II.2.a(1), above, or IEEE Std 344-1975, as appropriate.

c. Design Specifications

The design specification is the document by which the component (pump or valve) designer is guided relative to the parameters employed to describe the environment in which the component must perform its function. Consequently, it is essential that for "active" pumps and valves, the environment in which the component must perform its function to shut the plant down or mitigate the effects of an accident is adequately specified as a design requirement. Therefore, the applicant shall provide assurance that the following items are included in the design specifications of "active" pumps and valves:

(1) External loads expressed as flange end loadings associated with the accident condition for which the pump or valve must operate; i.e., the loading combinations associated with the faulted plant condition, and with due regard for the proper representation of the supports, if any, become the design loads for the active component. The design loads must be equal to or less than the end loads specified by the component manufacturer as permitted for normal operation.

(2) All other relevant environmental conditions, such as temperature, humidity, etc., associated with the accident condition are specified as a normal design condition.

(3) Operating clearances or deformation limits necessary to assure operation are specified and maintained for the accident condition in which the component must operate. Excessive rubbing (other than ordinary seal rub) on rotating parts is not acceptable for active pumps under the accident conditions.

(4) All test conditions, including loadings and environmental conditions, are specified and operability requirements stated.

(5) The operability requirements during or after the accident or event are clearly stated.

3. Design and Installation of Pressure Relief Devices

Acceptable design criteria for pressure relief stations in open discharge systems are specified in Regulatory Guide 1.67, "Installation of Overpressure Protection Devices."

As indicated in Code Case 1569, the rules for acceptable design procedures for systems where the pressure-relieving devices discharge into closed systems or systems with long discharge pipes have not reached the stage of final codification. However, for these closed or quasi-closed systems, the safety analysis report must include a commitment to perform a conservative dynamic analysis of the system, including mounting pipe runs or headers where applicable, relief device mountings, and discharge piping systems. The SAR must also include a description of the calculational procedures, computer programs, and other methods to be used in the analysis. The analysis must include the time history or equivalent effects of changes of momentum due to fluid flow changes of direction. The fluid states considered must include postulated water slugs where water seals are used. Stress computations and stress limits must be in accord with applicable rules of the Code.

4. Component Supports

To be acceptable, the component support designs should provide adequate margins of safety under all plant operating conditions.

The acceptability of the combinations of loading conditions and design transients applicable to the design of component supports within a system, including the categorization of the appropriate plant and component support operating condition for each initiating event, (i.e., LOCA, SSE, pipe break, etc.) which may be used with each loading combination, is judged by comparison with the positions stated in Reference 5, and with appropriate standards acceptable to the staff developed by professional societies and standards organizations. When these conditions have been established, the corresponding stress limits which may be applied to the design of component supports are as specified in Subsection NF of Division 1 of Section III of the ASME Code. The need for more conservative stress limits for active component supports should be considered in context with the other features of the operability assurance program.

In addition, if the component support affects the operability requirements of the supported component, then deformation limits should also be specified. The deformation limits for active component supports should be compatible with the operability requirements of the components supported. In establishing allowable deformations, the possible movements of the support base structures must be taken into account.

III. REVIEW PROCEDURES

The reviewer will select and emphasize material from the procedures described below, as may be appropriate for a particular case.

For each area of review, the following review procedures apply:

1. Loading Combinations, Design Transients, and Stress Limits

The objectives in reviewing the loading combinations and stress limits employed by the applicant in the design of Code Class 1, 2, 3, and CS items are to confirm that each of the plant operating conditions have been included, that the loading combinations and design transients applicable to the design of Code constructed items and the categorization of proposed operating conditions are appropriate, that the design stress levels associated with each imposed loading combination are low enough to provide adequate margins with respect to the structural integrity of the item, and that for active components and their supports, stress levels are considered in the operability assurance program. The review conducted during the CP stage determines that the objectives have been addressed and are being implemented in the design in the form of a commitment by the applicant that specific design criteria will be utilized by checking actual summary analysis results, the OL stage review verifies that the design criteria have been utilized and that components have been designed to meet the objectives. To assure that these objectives are met, the review is performed as follows:

a. The applicant's proposed combination of plant operating conditions and appropriate compensating conditions in terms of anticipated transients and design basis events is reviewed for completeness and for categorization as normal, upset, emergency, or faulted.

b. The combination of design loading conditions, including procedures for combination, proposed by the applicant for each Code constructed item are reviewed to determine if they are adequate. This aspect of the review is made by comparison with the loading combinations set forth in Regulatory Guide 1.48. Deviations from the guide are evaluated on a case-by-case basis by questions addressed to the applicant to determine the rationale and justification for exceptions. Final determination is based on engineering judgment and past experience with prior applications.

c. The design stress limits selected by the applicant for each plant and item operating condition as established in (b) are reviewed to determine if they meet those specified in the appropriate subsection of Division 1 of the Code, and in Regulatory Guide 1.48. Deviations from Regulatory Guide 1.48 may be permitted provided justification is presented by the applicant. The acceptability determination is based on considerations of adequate margins of safety.

d. Analytical methods for components including their internal parts subjected to the faulted component operating condition dynamic loading should meet the criteria set forth in Section 5 of Standard Review Plan 3.9.2 as prescribed for reactor internals.

2. Pump and Valve Operability Assurance Program

The objective of the review of the pump and valve operability assurance program is to determine whether the program submitted will assure the operability of a component which

is required to function to shut down the plant or mitigate the consequences of an accident. During the CP stage, a commitment to adopt a program which satisfactorily meets the acceptance criteria is required. At the OL stage, it is verified that the detailed procedures actually meet this objective. To assure the achievement of the objective, the review is performed as follows:

a. The applicant's program is reviewed to determine if it consists of tne proper combination of test and analysis.

b. The test and analysis methods and programs are reviewed by comparing the information submitted in the SAR with the acceptance criteria delineated in Section II.2 of this review plan. In those cases that are not directly comparable, the reviewer determines whether an acceptable level of assurance of operability has been reached.

3. Design and Installation of Pressure Relief Devices

The objective of the review of the design and installation of pressure relief devices is to assure the adequacy of the design and installation, so that there is assurance of the integrity of the pressure relieving devices and associated piping during the functioning of one or more of the relief devices. In the CP review, it is determined whether there is reasonable assurance that the final design will meet these objectives. At the OL stage, the final design is reviewed to determine that the objectives have been met.

The review is performed as follows:

a. The design of the pressure-retaining boundary of the device is reviewed by comparison with the Code. Since explicit rules are not yet available within the Code for the design of safety and pressure relief valves, the design is reviewed on the basis of reference to sections of the Code on vessels, piping, and line valves, and on experience with similar installations and good engineering design practice.

   Allowable stress limits are compared with those for the appropriate class of construction in the Code. Deviations are identified and the applicant is requested to provide justification. Stress limits and loading combinations for the various plant operating conditions are covered under the subsections entitled "Loading Combinations, Design Transients, and Stress Limits" in this plan.

b. The design of the installation is reviewed for structural adequacy to withstand the dynamic effects of relief valve operation. The applicant should include and discuss: reaction force, valve opening sequence, valve opening time, method of analysis, and magnitude of a dynamic load factor (if used). In reaching an acceptance determination, the reviewer compares the submission with the requirements in II.3, above.

   Where deviations occur, they are identified and the justification is evaluated. Valve opening sequence effects must consider the worst combination possible and

forcing functions must be justified with valve opening time data. The review is based in part on comparisons with prior acceptable designs tested in operating plants.

4. Component Supports

The objective in the review of component supports is to determine that adequate attention has been given the various aspects of design and analysis, so that there is assurance as to support structural integrity and as to operability of active components that interact with component supports.

The structural integrity and the effects on operability of the three types of component supports described in I.4 are reviewed against the criteria and guidelines of II.1 and II.4 of this plan.

Also, the ASME Code provides rules for the construction requirements for metal supports which are intended to transmit loads from the pressure-retaining barrier of the component, as defined in Subsection NF of the Code, to the load-carrying structural member, whether concrete or structural steel.

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information has been provided in accordance with the requirements of this review plan, and that his evaluation supports conclusions of the following type, to be included in the staff's safety evaluation report:

"The specified design basis combinations of loadings as applied to safety-related ASME Code Class 1, 2, and 3 pressure-retaining components in systems designed to meet seismic Category I standards are such as to provide assurance that in the event of an earthquake affecting the site, or an upset, emergency, or faulted plant transient occurring during normal plant operation, the resulting combined stresses imposed on system components will not exceed allowable stress and strain limits for the materials of construction. Limiting the stresses under such loading combinations provides a conservative basis for the design of system components to withstand the most adverse combination of loading events without loss of structural integrity. The design load combinations and associated stress and deformation limits specified for ASME Code Class 1, 2, and 3 components constitute an acceptable basis for design in satisfying applicable portions of General Design Criteria 1, 2, and 4.

"The component operability assurance program for ASME Code Class 1, 2, and 3 active valves and pumps provides adequate assurance of the capability of such active components (a) to withstand the imposed loads associated with normal, upset, emergency, and faulted plant and component operating conditions without loss of structural integrity, and (b) to perform necessary "active" functions (e.g., valve closure or opening, pump operation) under accident conditions and conditions expected when plant shutdown is required. The specified component operability assurance test program constitutes an acceptable basis for satisfying applicable portions of General Design Criteria 1, 2, and 4 and is acceptable to the staff.

"The criteria used in the design and installation of ASME Class 1, 2, and 3 safety and relief valves provide adequate assurance that, under discharging conditions, the resulting stresses will not exceed allowable stress and strain limits for the materials of construction. Limiting the stresses under the loading combinations associated with the actuation of these pressure relief devices provides a conservative basis for the design and installation of the devices to withstand these loads without loss of structural integrity or impairment of the overpressure protection function. The criteria used for the design and installation of ASME Class 1, 2, and 3 overpressure relief devices constitute an acceptable basis for meeting the applicable requirements of General Design Criteria 1, 2, 4, 14, and 15 and are consistent with those specified in Regulatory Guide 1.67.

"The specified design basis loading combinations used for the design of safety-related ASME Code Class 1, 2, and 3 component supports in systems classified as seismic Category I provide assurance that in the event of an earthquake or an upset, emergency, or faulted plant transient, the resulting combined stresses imposed on system components will not exceed allowable stress and strain limits for the materials of construction. Limiting the stresses under such loading combinations provides a conservative basis for the design of support components to withstand the most adverse combination of loading events without loss of structural integrity or supported component operability. The design load combinations and associated stress and deformation limits specified for ASME Code Class 1, 2, and 3 component supports constitute an acceptable basis for satisfying applicable portions of General Design Criteria 1, 2, and 4."

Class CS component evaluation findings are covered in Standard Review Plan 3.9.5 in connection with reactor internals.

V. REFERENCES

1. 10 CFR § 50.55a, "Codes and Standards."

2. 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants."

3. ASME Boiler and Pressure Vessel Code, Section III, Division 1, "Nuclear Power Plant Components," American Society of Mechanical Engineers.

4. IEEE Std 344-1975, "Guide for Seismic Qualification of Class I Electric Equipment for Nuclear Power Generating Stations," Institute of Electrical and Electronics Engineers.

5. Regulatory Guide 1.48, "Design Limits and Loading Combinations for Seismic Category I Fluid System Components."

6. Regulatory Guide 1.67, "Installation of Overpressure Protection Devices."

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SECTION 3.9.4 CONTROL ROD DRIVE SYSTEMS

REVIEW RESPONSIBILITIES

Primary - Mechanical Engineering Branch (MEB)

Secondary - Reactor Systems Branch (RSB)
Materials Engineering Branch (MTEB)

I. AREAS OF REVIEW

Information in the areas noted below is provided in the applicant's safety analysis report and is reviewed by the MEB in accordance with this plan. This information pertains to the reactor control rod drive system (CRDS), which is considered to extend to the coupling interface with the reactivity control elements in the reactor pressure vessel. For electromagnetic systems, the review under this plan is limited to just the control rod drive mechanism (CRDM) portion of the CRDS. For hydraulic systems, the review covers the CRDM and also the hydraulic control unit, the condensate supply system, and the scram discharge volume. For both types of systems, the CRDM housing should be treated as part of the reactor coolant pressure boundary (RCPB); the relevant mechanical engineering information may be presented in this section or by reference to the sections on the RCPB.

If other types of CRDS are proposed or if new features that are not specifically mentioned here are incorporated in CRDS of current types, information should be supplied for the new systems or new features similar to that described below.

1. The descriptive information, including design criteria, testing programs, drawings, and a summary of the method of operation of the control rod drives, is reviewed to permit an evaluation of the adequacy of the system to perform its mechanical function properly.

2. A review is performed of information pertaining to design codes, standards, specifications, and standard practices, as well as to General Design Criteria, Regulatory Guides, and branch positions that are applied in the design, fabrication, construction, and operation of the CRDS.

The various criteria, described in general terms above, should be supplied along with the names of the apparatus to which they apply. Pressurized parts of the system are reviewed to determine the extent to which the applicant complies with the Class 1 requirements of Section III of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (hereafter "the Code") for those portions which are not part of the reactor coolant pressure boundary, and with other specified parts of Section III, or other sections of the Code for pressurized portions which are not part of the reactor coolant pressure boundary. The MEB reviews the non-pressurized portions of the control rod drive system to determine the acceptability of design margins for allowable values of stress, deformation, and fatigue used in the analyses. If an experimental testing program is used in lieu of analysis, the program is reviewed to determine whether it adequately covers the areas of concern in stress, deformation, and fatigue.

3. Information is reviewed which pertains to the applicable design loads and their appropriate combinations, to the corresponding design stress limits, and to the corresponding allowable deformations. The deformations are of interest in the present context only in those instances where a failure of movement could be postulated due to excessive deformation and such movement would be necessary for a safety-related function.

   If the applicant selects an experimental testing option in lieu of establishing a set of stress and deformation allowables, a detailed description of the testing program must be provided for review.

   In the preliminary safety analysis report (PSAR), the load combinations, design stress limits, and allowable deformations criteria should be provided for review.

   In the final safety analysis report (FSAR), the actual design should be compared with the design criteria and limits to demonstrate that the criteria and limits have not been exceeded.

   Loadings imposed during normal plant operation and startup and shutdown transients include but are not limited to pressure, deadweight, temperature effects, and anticipated operational occurrences. Loadings associated with specific seismic and other dynamic events are then combined with the above plant-type loads. Each set of combined loads has a selected stress or deformation limit. The selection of a specific limit is influenced by the probability of the postulated event occurring and the need to assure operation during and after the event.

4. The portion of the SAR is reviewed that describes plans for the conduct of an operability assurance program or that references previous test programs or standard industry procedures for similar apparatus. For example, the life cycle test program for the CRDS is reviewed. The operability assurance program is reviewed to ascertain coverage of the following:
   a. Life cycle test program.
   b. Proper service environment imposed during test.

c. Mechanism functional tests.

d. Program results.

II. ACCEPTANCE CRITERIA

The acceptance criteria for the areas of review are the following:

1. The descriptive information is determined to be sufficient provided the minimum requirements for such information meet Section 3.9.4 of Reference 14.

2. Construction (as defined in NA-1110 of Section III of the ASME Code, Reference 10) should meet the following codes and standards utilized by the nuclear industry which have been reviewed and found acceptable:

   a. Pressurized Portions of Equipment Classified as Quality Group A, B, C (Regulatory Guide 1.26)

      Section III of the ASME Code, Class 1, 2, or 3 as appropriate (Ref. 10).

   b. Pressurized Portions of Equipment Classified as Quality Group D (Regulatory Guide 1.26)

      (1) Section VIII, Division 1 of the ASME Code for vessels and pump casings (Ref. 10).

      (2) Applicable to Piping Systems (American National Standards Institute, ANSI)[1/]:

      B16.5 Steel Pipe Flanges and Flanged Fittings (Ref. 16).
      B16.9 Steel Butt Welding Fittings (Ref. 17).
      B16.11 Steel Socket Welding Fittings (Ref. 18).
      B16.25 Butt Welding Ends (Ref. 19).
      B31.1 Piping (Ref. 20).
      SP-25 Standards (Ref. 21).
      SP-66 Valves (Ref. 22).

   c. Non-Pressurized Equipment (Non-ASME Code)

      Design margins presented for allowable stress, deformation, and fatigue should be equal to or greater than those for other plants of similar design having a period of successful operation. Justification of any decreases should be provided.

3. For the various plant operating conditions defined in NB-3113 of Section III of the ASME Code (Ref. 10), load combination sets are as given in Standard Review Plan 3.9.3 (Ref. 15). The stress limits applicable to pressurized and non-pressurized portions of the control rod drive systems should be as given in Reference 15 for each loading set.

4. The operability assurance program will be acceptable provided the observed performance as to wear, functioning times, latching, and overcoming a stuck rod meet system design requirements.

III. REVIEW PROCEDURES

The reviewer will select and emphasize material from the procedures described below as may be appropriate for a particular case.

1/ This list can be extended by a staff review and acceptance of other ANSI & MSS Standards in the piping system area.

1. The objectives of the review are to determine that design, fabrication, and construction of the control rod drive mechanisms provide structural adequacy and that suitable life cycle testing programs have been utilized to prove operability under service conditions.

   In the construction permit (CP) review, it should be determined that the design criteria utilize proper load combinations, stress and deformation limits, and that operability assurance is provided by reference to a previously accepted testing program or that a commitment is made to perform a testing program which includes the essential elements listed below. In the operating license (OL) review, the results of any testing program not previously reviewed should be evaluated.

2. The design criteria presented should be evaluated for both the internal pressure-containing portions and other portions of the CRDS. These include the CRDM housing, hydraulic control unit, condensate supply system and scram discharge volume, and portions such as the cylinder, tube, piston, and collect assembly.

   Of particular concern are any new and unique features which have not been used in the past. Pressure-containing components are checked to ensure that they meet the design requirements of the codes and criteria which have been accepted by the Reactor Systems Branch, and are identified in Standard Review Plan 3.2.2. The review of the functional design of reactivity control systems, including control rod drive systems, is the responsibility of RSB (See SRP 4.5). The loading combinations for the various plant operating conditions are checked for consistency with Reference 15; given these loading combinations, the stress limits of the appropriate code should not be exceeded, or the limits in Reference 15 should not be exceeded if not specified in the listed design code. Exceptions taken by the applicant to any of the accepted codes, standards, or AEC criteria must be identified and the basis clearly justified so that evaluation is possible. Engineering judgment, experience, comparisons with earlier cases and design margins, and consultation with supervisors permit the reviewer to reach a decision on the acceptability of any exceptions posed by the applicant.

   The choice of materials of construction for unpressurized equipment that is not governed by accepted codes or standards is reviewed bv the MTEB.

3. Loading combinations are defined as those loadings associated with plant operations which are expected to occur one or more times during the lifetime of the plant and include but are not limited to loss of power to all recirculation pumps, tripping of the turbine generator set, isolation of the main condenser, and loss of all offsite power, combined with loadings caused by natural or accident events. The load combinations which are postulated to occur are specified for each of the plant operating conditions as defined in Paragraph NB-3113 of the ASME Code (Ref. 10). These load combinations are defined in Reference 15 and are compared by the reviewer with those provided by the applicant.

The design stress limits, including fatigue limits, and deformation limits as appropriate to the components of the control rod drive mechanism are compared by the reviewer with those of specified codes, previously designed and successfully operating systems, or with the results of scale model and prototype testing programs.

4. The control rod drive mechanisms of a new design or configuration should be subjected to a life cycle test program to determine the ability of the drives to function over the full range of temperatures, pressures, loadings, and misalignment expected in service. The tests should include functional tests to determine times of rod insertion and withdrawal, latching operation, scram operation and time, system valve operation and scram accumulator leakage for hydraulic CRDS, ability to overcome a stuck rod condition, and wear. Rod travel and number of trips expected during the mechanism operational life should be duplicated in the tests.

   The reviewer checks the elements of the test program to be sure all required parameters have been included and finally reviews the test results to determine acceptability Excessive wear, malfunction of components, operating times beyond determined limits, scram accumulator leakage, etc., all would be cause for retesting.

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information has been provided to satisfy the requirements of this review plan, and that his evaluation is sufficiently complete and adequate to support conclusions of the following type, to be included in the staff's safety evaluation report:

> "The design criteria and the testing program conducted in verification of the mechanical operability and life cycle capabilities of the reactivity control system are in conformance with established criteria, codes, standards, and specifications acceptable to the Regulatory staff. The use of these criteria provide reasonable assurance that the system will function reliably when required, and form an acceptable basis for satisfying the mechanical reliability stipulations of General Design Criterion 27."

V. REFERENCES

1. 10 CFR Part 50, Appendix A, General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena."

2. 10 CFR Part 50, Appendix A, General Design Criterion 14, "Reactor Coolant Pressure Boundary."

3. 10 CFR Part 50, Appendix A, General Design Criterion 15, "Reactor Coolant System Design."

4. 10 CFR Part 50, Appendix A, General Design Criterion 20, "Protection System Functions."

5. 10 CFR Part 50, Appendix A, General Design Criterion 26, "Reactivity Control System Redundancy and Capability."

6. 10 CFR Part 50, Appendix A, General Design Criterion 29, "Protection Against Anticipated Operational Occurrences."

7. 10 CFR Part 50, Appendix A, General Design Criterion 30, "Quality of Reactor Coolant Pressure Boundary."

8. 10 CFR Part 50, Appendix A, General Design Criterion 31, "Fracture Prevention of Reactor Coolant Pressure Boundary."

9. 10 CFR Part 50, Appendix A, General Design Criterion 32, "Inspection of Reactor Coolant Pressure Boundary."

10. ASME Boiler and Pressure Vessel Code, Sections III and VIII, American Society of Mechanical Engineers.

11. Regulatory Guide 1.26, "Quality Group Classifications and Standards."

12. Regulatory Guide 1.29, "Seismic Design Classification."

13. Regulatory Guide 1.48, "Design Limits and Loading Combinations for Seismic Category I Fluid System Components."

14. Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," Revision 2.

15. Standard Review Plan 3.9.3, "ASME Code Class 1, 2, and 3 Components, Component Supports, and Core Support Structures."

16. ANSI B 16.5, "Steel Pipe Flanges and Flanged Fittings," American National Standard Institute.

17. ANSI B 16.9, "Wrought Steel Butt Welding Fittings," American National Standard Institute.

18. ANSI B 16.11, "Steel Fittings Steel Welding and Threaded," American National Standard Institute.

19. ANSI B 16.25, "Butt Welding Ends - Pipe, Valves, Flanges, and Fittings," American National Standard Institute.

20. ANSI B 31.1, "Power Piping," American National Standard Institute.

21. MSS-SP-25, "Marking for Va'ves, Fittings, Flanges, and Unions," Manufacturers Standardization Society.

22. MSS-SP-66, "Pressure-Temperature Ratings for Steel Butt Welding End Valves," Manufacturers Standardization Society.

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SECTION 3.9.5 REACTOR PRESSURE VESSEL INTERNALS

REVIEW RESPONSIBILITIES

Primary - Mechanical Engineering Branch (MEB)

Secondary - None

I. AREAS OF REVIEW

For the purpose of this SRP section, the term "reactor internals" refers to all structural and mechanical elements inside the reactor pressure vessel with the exception of the following:

Fuel assemblies, control rod drive mechanisms (defined and reviewed in SRP Section 3.9.4), and reactivity control elements out to the CRDM coupling interface. The CRDM guide tubes are considered to be part of the reactor internals.

In-core instrumentation (in-core instrumentation support structures are considered part of the reactor internals).

The Materials Engineering Branch reviews the reactor internals materials in SRP Section 4.5.2. The Core Performance Branch reviews the fuel assembly design in SRP Section 4.2.

The staff reviews the following specific areas to assure conformance with the requirements of General Design Criteria 1, 2, 4 and 10:

1. The physical or design arrangements of all reactor internal structures, components, assemblies and systems. This should include the manner of positioning and securing such items within the reactor pressure vessel, the manner of providing for axial and lateral retention and support of the internals assemblies and components, and the manner of accommodating dimensional changes due to thermal and other effects.

2. The plant and system operating conditions and design basis events which provide the basis for the design of the reactor internals.

3. The specific design and service loading combinations and the appropriate method of combination of these loads. The design and service stress limits associated with these various loading combinations are reviewed, including maximum allowable

stress, deflection limits, and cycling and fatigue limits, as specified in Table I of Branch Technical Position MEB 3-2 of SRP Section 3.9.3. The extent of compliance with subsection NG of the ASME Code is reviewed. Details of dynamic analysis, input forcing functions, and response loadings are discussed in SRP Section 3.9.2.

II. ACCEPTANCE CRITERIA

Compliance with the following criteria constitutes an acceptable basis for satisfying the applicable portions of General Design Criteria 1, 2, 4 and 10.

A discussion of the acceptance criteria for the design and service loading combinations and stress limits applicable to reactor internals is presented in SRP Section 3.9.3 (Ref. 7).

The design and construction of the core support structures should conform to the requirements of subsection NG, "Core Support Structures," of Section III of the ASME Code (Ref. 5).

The design and service loadings, stress limits, and analyses that provide the basis for the design of reactor internals other than the core support structures should meet the guidelines of NG-3000 and be constructed so as not to adversely affect the integrity of the core support structures (NG-1122).

Deformation limits for reactor internals should be established by the applicant and presented in his Safety Analysis Report. The basis for these limits should be included. The stresses associated with these displacements should not exceed the specified design limits. The requirements for dynamic analysis of these components are discussed in SRP Section 3.9.2.

III. REVIEW PROCEDURES

The reviewer will select and emphasize material from the procedures described below as may be appropriate for a particular case.

The configuration and general arrangement of all mechanical and structural internal elements covered by this SRP section are reviewed and compared to those of previously licensed similar plants. Any significant changes in design are noted and the applicant is asked to verify that these changes do not affect the flow-induced vibration test results required by SRP Section 3.9.2.

With respect to the design and analysis of these components, a statement by the applicant that they are designed in accordance with subsection NG, "Core Support Structures," of the ASME Code, Section III, and meet the criteria for design and service loads and limits in SRP Section 3.9.3, is acceptable. In lieu of such a commitment, the reviewer must determine that the design and analysis of these components are consistent with the requirements discussed in II, above. This is accomplished by requiring that the applicant describe the design procedures and criteria used in the design of these components.

This includes a list of the design and service stress limits used for all of the applicable loading conditions.

The deformation limits specified for these components are reviewed to verify that the applicant has stated that these deflections will not interfere with the functioning of related components, e.g., control rods and standby cooling systems, and that the stresses associated with these displacements are less than the specified limits for the core support structures.

At the operating license stage, the calculated stresses and deformations are reviewed to determine that they do not exceed the specified limits.

Any deviations that have not been adequately justified are identified and findings to that effect are transmitted to the applicant with a request for conformance with the requirements discussed in subsection II, or for additional technical justification.

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information has been provided in accordance with this SRP section, and that his evaluation supports conclusions of the following type, to be included in the staff's Safety Evaluation Report:

> "The design procedures and criteria that the applicant has used for the reactor internals are in conformance with established technical procedures, positions, standards, and criteria which are acceptable to the staff.
>
> "The specified transients, design and service loadings, and combination of loadings as applied to the design of the reactor internals structures and components provide reasonable assurance that in the event of an earthquake or of a system transient during normal plant operation, the resulting deflections and associated stresses imposed on these structures and components would not exceed allowable stresses and deformation limits for the materials of construction. Limiting the stresses and deformations under such loading combinations provides an acceptable basis for the design of these structures and components to withstand the most adverse loading events which have been postulated to occur during service lifetime without loss of structural integrity or impairment of function. In addition, the design procedures and criteria used by the applicant in the design of the reactor internals constitutes an acceptable basis for satisfying the applicable requirements of General Design Criteria 1, 2, 4 and 10."

V. REFERENCES

1. 10 CFR Part 50, Appendix A, General Design Criterion 1, "Quality Standards and Records."

2. 10 CFR Part 50, Appendix A, General Design Criterion 2, "Design Basis for Protection Against Natural Phenomena."

3. 10 CFR Part 50, Appendix A, General Design Criterion 4, "Environmental and Missile Design Bases."

4. 10 CFR Part 50, Appendix A, General Design Criterion 10, "Reactor Design."

5. ASME Boiler and Pressure Vessel Code, Section III, Division 1, "Nuclear Power Plant Components," American Society of Mechanical Engineers.

6. SRP Section 3.9.2, "Dynamic Testing and Analysis of Systems, Components, and Equipment."

7. SRP Section 3.9.3, "ASME Code Class 1, 2, and 3 Components, Component Supports, and Core Support Structures."

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SECTION 3.9.6 INSERVICE TESTING OF PUMPS AND VALVES

REVIEW RESPONSIBILITIES

Primary - Mechanical Engineering Branch (MEB)

Secondary - Reactor Systems Branch (RSB)
Auxiliary Systems Branch (ASB)
Containment Systems Branch (CSB)

I. AREAS OF REVIEW

The MEB reviews the following areas of the applicant's safety analysis report (SAR) that cover the inservice testing of pumps and valves designated as Class 1, 2, or 3 under Section III of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (hereafter "the Code"), to assure conformance with 10 CFR 50 Appendix A, General Design Criteria 37, 40, 43, and 46 and 10 CFR 50.55a(g):

1. Inservice Testing of Pumps

   a. The descriptive information in the SAR covering the inservice test program is reviewed for those Code Class 1, 2, and 3 system pumps provided with an emergency power source. Upon request the Auxiliary Systems Branch verifies the code class designations for each listed pump and the completeness of the list.

   b. Reference values for testing for speed, pressure, flow rate, vibration, and bearing temperature at normal pump operating conditions are reviewed.

   c. The pump test schedule, included in the plant technical specifications, is reviewed.

   d. The methods described in the SAR for measuring the reference values and inservice values for the pump parameters above are reviewed.

2. Inservice Testing of Valves

   The descriptive information in the SAR covering the inservice test program of all Code Class 1, 2, and 3 valves is reviewed. This review does not include those valves defined in IWV-1200 of Section XI of the Code. Upon request the Auxiliary Systems Branch verifies the classification of each listed valve and the completeness of the list.

3. Relief Requests

10 CFR 50.55 a(g) requires a nuclear power facility to periodically update its inservice testing program to meet the requirements of future revisions of Section XI of the ASME Code. However, if it proves impractical to implement these criteria, the applicant is allowed to submit requests for relief from Section XI requirements on a case by case basis. Accordingly, any requests for relief are reviewed by the staff to determine if the proposed exceptions to Section XI will degrade the overall plant safety. Due consideration is given to the burden upon the applicant that could result if the criteria of Section XI were imposed on the facility. Upon request, RSB, ASB, and CSB review the system aspects of these relief requests.

II. ACCEPTANCE CRITERIA

The acceptance criteria for the areas of review described in subsection I are as follows. Compliance with these criteria constitutes an acceptable basis for satisfying the applicable portions of General Design Criteria 37, 40, 43, 46 and 10 CFR 50.55a(g).

For those areas of review identified in subsection I as being the responsibility of other branches, the acceptance criteria and their methods of application are contained in the SRP sections corresponding to those branches.

1. Inservice Testing of Pumps

a. The scope of the applicant's test program is acceptable if it is in agreement with IWP-1000 of Section XI of the Code. Since the pump test program is based on the detection of changes in the hydraulic and mechanical condition of a pump relative to a reference test specified in IWP-3000, the establishment of a reference set of parameters and a consistent test method is a basic criterion of the program.

b. The pump test program is acceptable if it meets the requirements for establishing reference values and the periodic testing schedule of IWP-3000 of Section XI of the Code. The allowable ranges of inservice test quantities, corrective actions, and bearing temperature tests are established by IWP-3000 and IWP-4000. The pump test schedule in the plant technical specification is required to comply with these rules.

c. The test frequencies and durations in the plant technical specifications are acceptable if the provisions of IWP-3000 of Section XI of the Code are met. If a pump is normally operated more frequently than once a month, and at the reference conditions, it need not be specially tested. Otherwise, pumps must be tested each month during plant operation, and during shutdown periods if practical. The pumps must be run for at least five minutes under conditions as stable as the system permits. Bearing temperatures must be measured once a year for the duration specified in IWP-3000.

d. The methods of measurement are acceptable if the test program meets the requirements of IWP-4000 of Section XI of the Code with regard to instruments, pressure measurements, temperature measurements, rotational speed, vibration measurement, and flow measurements.

2. Inservice Testing of Valves

a. To be acceptable, the SAR valve test list must contain all Code Class 1, 2, and 3 valves except those defined in IWV-1200. The SAR valve list must include a valve categorization which complies with the provisions of IWV-2100 of Section XI of the Code. Each specific valve to be tested by the rules of Subsection IWV is listed in the SAR by type, valve identification number, code class, and IWV-2100 valve category.

b. The valve test procedures in the plant technical specifications are acceptable if the provisions of IWV-3000 of Section XI of the Code are met with regard to preservice and periodic inservice valve testing.

3. Information Required for Review of Relief Requests

a. Identify component for which relief is requested:

(1) Name and number as given in FSAR
(2) Function
(3) ASME Section III Code Class
(4) For valve testing, also specify the ASME Section XI valve category as defined in IWV-2000

b. Specifically identify the ASME Code requirement that has been determined to be impractical for each component.

c. Provide information to support the determination that the requirement in (b) is impractical; i.e., state and explain the basis for requesting relief.

d. Specify the inservice testing that will be performed in lieu of the ASME Code Section XI requirements.

e. Provide the schedule for implementation of the procedure(s) in (d).

Requests for relief from Section XI requirements will be granted by the staff if the applicant has adequately demonstrated either of the following:

a. Compliance with the code requirements would result in hardships or unusual difficulties without a compensating increase in the level of safety, and noncompliance will provide an acceptable level of quality and safety.

b. Proposed alternatives to the code requirements or portions thereof will provide an acceptable level of quality and safety.

III. REVIEW PROCEDURES

The reviewer will select and emphasize material from the procedures described below as may be appropriate for a particular case. For each area of review, the following review procedures are followed:

1. Inservice Testing of Pumps

   a. The scope of the applicant's program is reviewed for agreement with subsection II.1.a. The program is acceptable if a preservice test program is used to established reference values. The periodic inservice program must verify the reference values within acceptable limits. Upon request ASB verifies the acceptability of the pump list.

   b. The pump test program procedures must agree with the requirements of subsection II.1.b. The program is best presented in tabular form in the plant technical specifications.

   c. The inservice test frequencies and test durations in the plant technical specifications are reviewed for agreement with subsection II.1.c.

   d. The test results described in the SAR are reviewed for agreement with subsection II.1.d. The SAR need only provide the necessary information to permit a conclusion that the methods of measurement and the data acquisition system will provide the needed data. The reviewer does not approve or disapprove the instruments or methods proposed or used.

2. Inservice Testing of Valves

   a. The SAR valve test list and categorization are reviewed for agreement with subsection II.2.a. Upon request ASB verifies the acceptability of the valve test list and categorization.

   b. The valve test program is acceptable if the procedures follow the rules of subsection II.2.b for preservice and periodic inservice testing.

3. Relief Requests

   Requests for relief from Section XI requirements are reviewed to determine that sufficient information has been provided and that the acceptance criteria of subsection II.3 have been met. If necessary, the secondary reviewers are requested to provide input on the system aspects of the relief requests.

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information is provided in accordance with the requirements of this SRP section and that his evaluation supports conclusion of the following type, to be included in the staff's safety evaluation report:

> "To ensure that all ASME Code Class 1, 2, and 3 pumps and valves will be in a state of operational readiness to perform necessary safety functions throughout the life of the plant, a test program is provided which includes baseline preservice testing and periodic inservice testing. The program provides for both functional testing of the components in the operating state and for visual inspection for leaks and other signs of distress.
>
> "The applicant has stated that the inservice test program for all Code Class 1, 2, and 3 pumps and valves meets the requirements of 10 CFR 50.55a(g).
>
> "Compliance with these requirements constitute an acceptable basis for satisfying the applicable portions of General Design Criteria 37, 40, 43, and 46."

V. REFERENCES

1. 10 CFR Part 50, Appendix A, General Design Criterion 37, "Testing of Emergency Core Cooling System."

2. 10 CFR Part 50, Appendix A, General Design Criterion 40, "Testing of Containment Heat Removal System."

3. 10 CFR Part 50, Appendix A, General Design Criterion 43, "Testing of Containment Atmosphere Cleanup Systems."

4. 10 CFR Part 50, Appendix A, General Design Criterion 46, "Testing of Cooling Water System."

5. ASME Boiler and Pressure Vessel Code, Section III and Section XI, Subsections IWP and IWV, American Society of Mechanical Engineers.

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SECTION 3.10 SEISMIC QUALIFICATION OF CATEGORY I INSTRUMENTATION AND ELECTRICAL EQUIPMENT

REVIEW RESPONSIBILITIES

Primary - Mechanical Engineering Branch (MEB)

Secondary - Instrumentation and Control Systems Branch (ICSB)
Power Systems Branch (PSB)

I. AREAS OF REVIEW

Information concerning the methods of test and analysis employed to assure the operability of essential instrumentation and electrical equipment in the event of an earthquake should be provided in the applicant's safety analysis report (SAR) and is reviewed by the MEB to assure conformance with the requirements of General Design Criterion 2. Systems and components that must retain structural integrity, remain leaktight, or continue to function in the event of an earthquake, in order to assure safe operation or shutdown of the plant, must be designed to seismic Category I requirements.

At the construction permit (CP) stage, the staff review covers the following specific areas:

1. The criteria for seismic qualification, such as the deciding factors for choosing between tests or analyses, the considerations in defining the seismic input motion, and the demonstration of adequacy of the seismic qualification program.

2. The methods and procedures, including tests and analyses, used to assure the operability of seismic Category I instrumentation and electrical equipment in the event of a safe shutdown earthquake (SSE) and to assure structural integrity and operability of the equipment after occurrence of the operating basis earthquake (OBE). Instrumentation and electrical equipment that must be designed to seismic Category I requirements include the reactor protection system, engineered safety feature circuits, emergency power systems, and all auxiliary safety-related electrical systems.

3. The methods and procedures of analysis or testing of the supports for the seismic Category I instrumentation and electrical equipment, and the procedures used to

account for possible amplification of vibratory motion (amplitude and frequency content) under seismic conditions. Supports include items such as battery racks, instrument racks, control consoles, cabinets, panels, and cable trays.

At the operating license (OL) stage, the staff reviews the results of tests and analyses to assure the proper implementation of criteria established in the CP review, and to demonstrate adequate seismic qualification.

The ICSB and PSB verify that all of the seismic Category I instrumentation, controls and electrical equipment and supports are included in the seismic qualification program, that the electrical performance aspects of these items are included in the seismic qualification testing, and that the equipment mounting during the test adequately simulates the actual service mounting. For these considerations, ICSB has responsibility for instrumentation and control systems while PSB has responsibility for electrical power systems.

II. ACCEPTANCE CRITERIA

The acceptance criteria for the areas of review designated in subsection I are as follows. Conformance with these criteria satisfies the applicable portions of General Design Criterion 2.

1. For plants for which the CP application, including the preliminary safety analysis report (PSAR), was docketed before October 27, 1972, the seismic qualification of Category I instrumentation, electrical equipment, and supports should meet the requirements of IEEE Std. 344-1971 (Ref. 3). For these plants an additional staff review will be conducted to assure that such components do have adequate margin to perform their intended design functions during the seismic event. Special emphasis should be placed on the effects of possible multi-mode response and simultaneous vertical and horizontal excitations on the component operability. The following requirements should be met:

   a. Seismic Qualification for Equipment Operability

      (1) Tests or analyses are required to confirm the operability of all seismic Category I electrical equipment and instrumentation during and after an earthquake of magnitude up to and including the SSE. (The analysis method is not recommended for complex equipment that cannot be modeled accurately enough to predict its response correctly for functional verification.) The secondary reviewers verify the completeness of the seismic qualification program based upon the acceptance criteria given in the SRP sections for which they have primary responsibility.

      Designs and equipment that have been previously qualified by means of tests and analyses equivalent to those described here may be acceptable provided that proper documentation of such tests and analyses is submitted.

(2) Single frequency input excitations, such as continuous single frequency sinusoidal motions or sine beat motions, may be used; however, multi-frequency input excitations as delineated in IEEE Std. 344-1975 (Ref. 4) are preferable and should be utilized whenever possible. In either case, the maximum input motion acceleration should equal or exceed the maximum seismic acceleration expected at the equipment mounting location. See subsection II.1.b(3) below for a discussion of the participation of the equipment supports.

(3) For single frequency input excitation, the discrete frequencies at which the test input motion is applied should cover the range 1-33 Hz. If resonant frequencies of the equipment and equipment supports are identified by prior analysis or "sweep" testing or both, tests conducted only at the resonant frequencies are acceptable. However, if multi-frequency input excitations are used, the level of response spectrum derived from the test input should envelope the corresponding response spectrum level required for seismic qualification at the component mounting location.

(4) Equipment should be tested in the operational condition. Procedures for monitoring the equipment under test and the acceptability of the signals so obtained for functional verification are reviewed by ICSB and PSB in accordance with acceptance criteria given in the SRP sections for which they have primary responsibility.

(5) The test motion may be applied to one vertical and two orthogonal horizontal axes separately. However, biaxial input with simultaneous vertical and horizontal excitations as delineated in IEEE Std. 344-1975 is preferable and should be utilized whenever possible.

(6) The test program may be based upon selectively testing a representative number of mechanical components according to type, load level, size, etc. on a prototype basis.

b. Seismic Design Adequacy of Supports

(1) Analyses or tests should be performed for all supports of seismic Category I electrical equipment and instrumentation to assure their structural capability to withstand seismic excitation.

(2) The analytical results should include the maximum accelerations and associated frequencies at the equipment mounting location, and the combined stresses of the support structures should be within the limits of the ASME Code, Section III, Subsection NF, "Component Support Structures" (Ref. 2) or other appropriate limits which are acceptable to the staff.

(3) Supports should be tested with equipment installed or with a dummy simulating the equivalent inertial mass effects and dynamic coupling to the support. If the equipment is installed in a nonoperational mode for the support test, the response at the equipment mounting location should be monitored such that the maximum accelerations and associated frequencies can be defined. In such a case, equipment should be tested separately for operability and the actual input motion to the equipment should be more conservative in amplitude and frequency content than the monitored response.

(4) The requirements of subsection II.1.a(2), (3), and (5), above, are applicable when tests are conducted on the equipment supports.

2. For plants for which the CP application was docketed after October 27, 1972, the seismic qualification of Category I instrumentation, electrical equipment, and supports should meet the requirements of IEEE Std. 344-1975 and Regulatory Guide 1.100, and special emphasis should be placed on the following (also see Refs. 4 and 6):

a. Seismic Qualification for Equipment Operability

(1) Tests and analyses are required to confirm the operability of all seismic Category I electrical equipment and instrumentation during and after an earthquake of magnitude up to and including the OBE and SSE. Prior to SSE qualification, it should be demonstrated that the equipment can withstand the OBE excitation without loss of structural integrity. Analyses alone, without testing, are acceptable as a basis for seismic qualification only if the necessary functional operability of the instrumentation or equipment is assured by its structural integrity alone. When complete seismic testing is impractical, a combination of tests and analyses is acceptable. The secondary reviewers verify the completeness of the seismic qualification program based upon the acceptance criteria given in the SRP sections for which they have primary responsibility.

Designs and equipment that have been previously qualified by means of tests and analyses equivalent to those described here are acceptable provided that proper documentation of such tests and analyses is submitted.

(2) The characteristics of the required seismic input motion should be specified by response spectrum or time history methods. These characteristics, derived from the structures or systems seismic analysis, should be representative of the seismic input motion at the equipment mounting locations.

(3) Equipment should be tested in the operational condition. Operability should be verified during and after the testing.

(4) The actual test input motion should be characterized in the same manner as the required input motion, and the conservatism in amplitude and frequency content should be demonstrated (i.e., the test response spectrum (TRS) should envelope the required response spectrum (RRS) over the critical frequency range).

(5) Seismic excitation generally has a broad frequency content. Multi-frequency vibration input motion should be used. However, single frequency input motion, such as sine beats, is acceptable provided the characteristics of the required input motion indicate that the motion is dominated by one frequency (e.g., by structural filtering effects), the anticipated response of the equipment is adequately represented by one mode, or the input has sufficient intensity and duration to excite all modes simultaneously to the required amplitudes such that the test response spectra will envelope the corresponding required response spectra over the frequency range covering all the individual modes.

Components that have been previously tested to IEEE Std. 344-1971 should be requalified using multi-frequency test input motion unless justification for using a single frequency test is provided.

(6) The test input motion should be applied to one vertical axis and one principal horizontal axis (or two orthogonal horizontal axes) simultaneously unless it can be demonstrated that the equipment response in the vertical direction is not sensitive to the vibratory motion in the horizontal direction, and vice versa. The time phasing of the inputs in the vertical and horizontal directions must be such that a purely rectilinear resultant input is avoided. An acceptable alternative is to test with vertical and horizontal inputs in-phase, and then repeat the test with inputs 180 degrees out-of-phase. In addition, the test must be repeated with the equipment rotated 90 degrees horizontally.

Components that have been previously tested to IEEE Std. 344-1971 should be requalified using biaxial test input motions unless justification for using a single axis test input motion is provided.

(7) The fixture design should simulate the actual service mounting and should not cause any extraneous dynamic coupling to the test item.

(8) The in situ application of vibratory devices to superimpose the seismic vibratory motions on a complex active device for operability testing is acceptable when it is shown that a meaningful test can be made in this way.

(9) The test program may be based upon selectively testing a representative number of components according to type, load level, size, etc., on a prototype basis.

b. Seismic Design Adequacy of Supports

(1) Analyses or tests should be performed for all supports of seismic Category I electrical equipment and instrumentation to assure their structural capability to withstand seismic excitation.

(2) The analytical results should include the required input motions to the mounted equipment as obtained and characterized in the manner stated in subsection 11.2.a(2), above, and the combined stresses of the support structures should be within the limits of the ASME Code, Section III, Subsection NF, "Component Support Structures" (Ref. 2) or other appropriate limits which are acceptable to the staff. Refer to SRP Section 3.9.3.

(3) Supports should be tested with equipment installed or with a dummy simulating the equivalent equipment inertial mass effects and dynamic coupling to the support. If the equipment is installed in a nonoperational mode for the support test, the response in the test at the equipment mounting location should be monitored and characterized in the manner as stated in subsection II.2.a(2), above. In such a case, equipment should be tested separately for operability and the actual input motion to the equipment in this test should be more conservative in amplitude and frequency content than the monitored response from the support test.

(4) The requirements of subsections 11.2.a(2), (4), (5), (6), and (7), above, are applicable when tests are conducted on the equipment supports.

c. Verification That Seismic Qualification Is Performed in the Proper Sequence of the Overall Qualification Program

As defined in Part B of Regulatory Guide 1.100 (reference 6), IEEE Std 344-1975 (reference 4) is an ancilliary standard of IEEE Std 323-1974 (endorsed with exceptions, by Regulatory Guide 1.89). To assure that overall qualification has been performed properly, the staff review shall verify that the seismic testing portion has been performed in its proper sequence as indicated in Sections 5 and 6 of IEEE Std 323-1974. On request, ICSB will provide consultation on the acceptability of any qualification sequence not in compliance with IEEE 323-1974.

3. In documenting the implementation of the seismic qualification program described above, the PSAR or FSAR should contain the following:

a. A detailed description of NSSS and A/E practice followed in seismic qualification, including criteria, methods and procedures used in conducting testing and analysis. (PSAR)

b. Information regarding administrative control of component seismic qualification, especially the handling of documentation, internal acceptance review procedures, identification of the scope of NSSS and A/E suppliers, and interface problems among NSSS, A/E, equipment vendors and testing laboratories. (PSAR)

c. A brief description of NSSS and A/E testing facilities, including the capability of the facilities to test the functioning of the equipment being tested. (PSAR)

d. Lists of equipment (devices or assemblies) and support structures should be provided with the following information specified for each item: (FSAR)

(1) Description of equipment (i.e., manufacturer type, size, capacity) and/or support structure.

(2) Method of qualification used and reporting of results:

(a) Analysis or test

If by analysis -

(i) describe whether static or dynamic analysis

(ii) provide justification that the analysis assures the proper functioning of the equipment or support during the seismic event.

(iii) identify where the seismic qualification is implemented in the overall qualification program sequence.

If by testing-

(i) describe whether single or multi-frequency test

(ii) describe whether input was single or bi-axial

(iii) identify test duration

(iv) identify whether devices are mounted during the testing of assemblies or supporting structures (i.e., panels, racks, etc.) and demonstrate the validity of any tests conducted without the devices (or suitable substitutes) or with the mounted devices in inoperative condition.

(v) identify where the seismic testing is implemented in the overall qualification program sequence.

(b) Report of the g-level required and tested to if using single frequency tests. Verify that the test response spectra (TRS) envelopes the required response spectra (RRS) if using multi-frequency tests.

(c) Report of the finding of resonance frequencies.

(d) Report on functional verification.

III. REVIEW PROCEDURES

The reviewer will select and emphasize material from the procedures described below as may be appropriate for a particular case. Upon request from the primary reviewer, the secondary review branches will provide input for the areas of review designated in subsection I. The primary reviewer obtains and uses such input as necessary to assure that this review procedure is complete.

For each area of review the following review procedures are used:

1. At the CP stage, the staff reviews the program which the applicant has described in the PSAR for the seismic qualification of all Category 1 instrumentation and electrical equipment. The program is measured against the requirements listed in subsection II. Of particular interest are the proper use of test and analytical procedures. Equipment which is too complex for reliable mathematical modeling should be tested unless the analytical procedures and corresponding design are convincingly conservative. Both the test and the analysis methods are reviewed for assurance that all important modes of response will be excited in tests or considered in analysis. Proper consideration of input motions so as to bound the required input, whether in terms of response spectra or time history in all necessary directions is verified. The use or treatment of supports is also reviewed.

2. At the OL stage, the staff reviews the program again as described by the applicant in the FSAR. In addition, the FSAR is reviewed for documentation of the successful implementation of the seismic qualification program including test and analysis results. Also, the acceleration levels used in the tests and in the analyses are reviewed for assurance that they equal or exceed the levels at the equipment mounting locations derived from structural response studies of the plant structure as built or as designed.

IV. EVALUATION FINDINGS

The reviewer should verify that sufficient information has been provided and that the review supports conclusions of the following type (for a CP review), to be included in the staff's safety evaluation report:

> "The proper functioning of essential instrumentation and electrical equipment in the event of the safe shutdown earthquake (SSE) is necessary to initiate protective actions including, for example, operation of the reactor protection system, engineered safety features, and standby power systems.
>
> "The seismic qualification program which will be implemented for seismic Category I instrumentation and electrical equipment provides adequate assurance that such equipment will function properly during the excitation from vibratory forces imposed by the safe shutdown earthquake and under the conditions of post-accident operation. This program constitutes an acceptable basis for satisfying the applicable requirements of General Design Criterion 2."

At the OL stage, the review should provide justification for a finding similar to that above with the phrase "will be implemented" modified to read "has been implemented."

V. REFERENCES

1. 10 CFR Part 50, Appendix A, General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena."

2. ASME Boiler and Pressure Vessel Code, Section III, "Nuclear Power Plant Components," American Society of Mechanical Engineers.

3. IEEE Std 344-1971, "Guide for Seismic Qualification of Class I Electric Equipment for Nuclear Power Generating Stations," Institute of Electrical and Electronics Engineers.

4. IEEE Std 344-1975, "Recommended Practices for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations," Institute of Electrical and Electronics Engineers.

5. K. M. Skreiner, E. G. Fischer, S. N. Hou, and G. Shipway, "New Seismic Requirements for Class I Electrical Equipment," IEEE Paper T 74 048-5, 1974 Winter Meeting of IEEE Power Engineering Society, Institute of Electrical and Electronics Engineers.

6. NRC Regulatory Guide 1.100, "Seismic Qualification of Electric Equipment for Nuclear Power Plants "

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SECTION 3.11 ENVIRONMENTAL QUALIFICATION OF MECHANICAL AND ELECTRICAL EQUIPMENT

REVIEW RESPONSIBILITIES

Primary - Instrumentation and Control Systems Branch (ICSB)
Power Systems Branch (PSB)

Secondary - Auxiliary Systems Branch (ASB)
Containment Systems Branch (CSB)
Reactor Systems Branch (RSB)
Accident Analysis Branch (AAB)

I. AREAS OF REVIEW

The information presented in Section 3.11 of the applicant's safety analysis report (SAR) should be sufficient to support the conclusion that all items of safety-related mechanical and electrical equipment are capable of performing their design safety functions under all normal, abnormal, and accident environmental conditions. The review will be performed to assure conformance with the requirements of General Design Criteria 1, 2, 4 and 23. The "normal, abnormal, and accident environmental conditions" are deemed to include all environmental conditions which may result from any normal or abnormal mode of plant operation, design basis events, post-design basis events, and containment tests. The information presented should include identification of the safety-related equipment, and for each item of equipment, the environmental design bases, definition of normal, abnormal, and design basis environments, and documentation of the qualification tests and analyses performed to demonstrate the required environmental capability. In the preliminary safety analysis report (PSAR), this documentation may consist of a description of the tests and analyses that have been or will be performed. In the final safety analysis report (FSAR), the results of the qualification tests and analyses for each type of equipment should be provided. Seismic qualification is addressed in SRP Section 3.10.

Section 3.11 of the SAR is reviewed to determine whether the required environmental capability of all safety-related equipment, i.e., the capability to perform design safety functions under normal, abnormal and accident environments, will be or has been adequately demonstrated.

The secondary review branches will provide information to the ICSB and PSB with regard to mechanical and electrical equipment of safety-related systems within their respective

primary review responsibilities, as described below, but exclusive of any electrical equipment located in the control room or other designated electrical equipment rooms or areas (this equipment is an ICSB and PSB responsibility). The SAR sections reviewed by the branches in performance of their secondary review functions are as follows: ASB reviews Sections 3.4.1, 3.5.1.1, 3.5.2, 3.6.1, 5.4.11, and applicable sections of Chapter 9; CSB reviews Section 6.2; RSB reviews Sections 3.2.1, 3.2.2, 4.4, 6.3, and applicable sections of Chapter 15, and AAB reviews Section 6.5.2. Guidance with regard to the definition of "safety-related systems" for the purposes of this SRP section is contained in SRP Section 7.1, and the assignments of primary review responsibility for these systems are contained in the applicable SRP sections.

The ASB, CSB, and RSB confirm that the SAR identifies all safety-related equipment.

The ASB and CSB confirm the location of each item of equipment, both inside and outside the containment. Inside the containment, the location must be specified, whether inside or outside of the missile shield for pressurized water reactor (PWR) plants, or whether inside or outside of the drywell for boiling water reactor (BWR) plants.

The ASB, CSB, and RSB confirm the validity of the descriptions of the normal, abnormal and accident environments provided in the SAR. They will also confirm the acceptability of the values provided in the SAR for the length of time that equipment is required to operate in accident environments.

With regard to the environments resulting from loss of environmental control systems (ventilation, heating, air conditioning), the ASB will confirm the description of these environments as provided in the SAR for those areas which contain safety-related equipment, including electrical control and instrumentation equipment.

The QAB reviews and determines that the applicant's QA program described in Chapter 17 of the SAR satisfies the requirements of 10 CFR 50, Appendix B.

The AAB reviews the adequacy of the radiation and chemical conditions for qualification for the normal, abnormal, and accident environments.

Specific information may be requested from the MEB as needed.

II. ACCEPTANCE CRITERIA

The general requirements for environmental design and qualification of all equipment important to safety are embodied in General Design Criteria 1, 2, 4, and 23 of Appendix A to 10 CFR Part 50, and the criteria of Appendix B to 10 CFR Part 50. In addition, the requirement for environmental qualification is included in IEEE Std 279 and in IEEE Std 308, as augmented by Regulatory Guide 1.32. However, none of the above documents provide specific criteria for assessing the acceptability of an environmental design and qualification program.

Simply stated, the general requirements for environmental design and qualification are as follows: (1) The equipment shall be designed to have the capability of performing design safety functions under all normal, abnormal and accident environments. (2) The equipment environmental capability shall be demonstrated by appropriate testing and analyses. (3) A quality assurance program meeting the requirements of 10 CFR 50 Appendix B shall be established and implemented to provide assurance that all requirements have been satisfactorily accomplished. The environmental design of safety-related mechanical and electrical equipment is acceptable when it can be ascertained that all three requirements are met.

Subsection V lists the documents which provide both acceptance criteria and evaluation guidance used in the review. The most important of these documents is IEEE Std 323 (augmented by Regulatory Guide 1.89), "General Guide for Qualifying Class I Electric Equipment for Nuclear Power Generating Stations." This document, although specifically written for Class I electric equipment, contains a clear presentation of the principles and criteria that are generic to the environmental qualification process itself; therefore, IEEE Std 323 is considered applicable to the environmental qualification of other types of equipment. This document contains detailed criteria applicable to whatever method of qualification is used, i.e., type testing, analyses, operating experience, on-going qualification, or combined qualification. The environmental design and qualification of safety-related equipment is acceptable when it is ascertained that the criteria of IEEE Std 323 have been met.

IEEE Std 334, "Guide for Type Tests of Continuous-Duty Class I Motors Installed Inside the Containment of Nuclear Power Generating Stations" (augmented by Regulatory Guide 1.40); IEEE Std 382, "Guide for Type Test of Class I Electric Valve Operators for Nuclear Power Generating Stations" (augmented by Regulatory Guide 1.73); and IEEE Std 383, "Standard for Type Test of Class IE Electric Cables, Field Splices, and Connections for Nuclear Power Generating Stations" (augmented by Regulatory Guide 1.131), are specific with regard to type test qualification of the equipment identified in their titles. The detailed criteria contained in these documents should be used in conjunction with the more comprehensive criteria of IEEE Std 323 for evaluating the respective equipment environmental qualifications.

IEEE Std 317, "Electric Penetration Assemblies in Containment Structures for Nuclear Power Generating Stations" (augmented by Regulatory Guide 1.63), contains general guidance for qualification of penetration assemblies. Therefore, this document should be used in conjunction with IEEE Std 323 for evaluating the environmental qualification of this equipment.

The applicant's estimate of the chemical environment is acceptable if it reflects the chemical composition of all fluids and additives present in the primary system or added to the containment environment in the course of the accident for various modes of equipment operation. The applicant's estimate of the radiation environment is acceptable if it reflects source terms comparable to those postulated in Regulatory Guide 1.89.

For those areas of review identified in subsection I of this SRP section as being the responsibility of other branches, the acceptance criteria and their methods of application are contained in the SRP sections corresponding to those branches.

III. REVIEW PROCEDURES

This section of the review plan describes the essential elements of the review process including the use of the criteria and evaluation guides.

The review objective is to determine from the information presented in the SAR whether there is reasonable assurance that all items of safety-related electrical and mechanical equipment are capable of performing design safety functions under all normal, abnormal, and accident environmental conditions.

To achieve the objective, the review is divided into two distinct phases; the information audit phase and the evaluation phase. The audit phase is concerned with the completeness of the information presented. The evaluation phase is concerned with whether the required environmental capability will be or has been adequately demonstrated for each item of equipment. The two phases of the review process are performed as follows:

1. Information Audit Phase

   The review should determine that the following information is included:

   a. Equipment Identification

      All safety-related mechanical and electrical equipment must be identified. The equipment tabulations provided should be checked for completeness against the descriptions of safety-related systems contained in SAR Chapters 4, 5, 6, 7, 8, 9, 10, 11, 15, and 17. Definitions of the three categories of safety-related systems are contained in SRP Section 7.1.

      The ICSB is responsible for verifying the completeness of the identification of all the safety-related, control, logic and instrumentation equipment. In addition, the ICSB confirms the equipment identification inputs of the secondary review branches.

      The PSB is responsible for verifying the completeness of the identification of all the safety-related electric power system equipment, such as penetrations, valves motors, cables, motor control centers and switchgear. In addition, the PSB confirms the equipment identification inputs of the secondary review branches.

      The secondary review branches are responsible for verifying the completeness of the identification of all mechanical equipment, and all electro-mechanical equipment located outside of the control room or other designated electrical equipment areas which pertain to the safety systems within their primary review responsibilities.

b. Equipment Location

The location of each item of safety-related equipment must be identified, both inside and outside the containment. Inside the containment, the location must specify whether inside or outside of the missile shield (for PWRs) or whether inside or outside of the drywell (for BWRs). Location of equipment is required in order to establish accurate definitions of normal, abnormal, and accident environments.

The ICSB and PSB and the secondary review branches are responsible for verifying the location of the items of equipment identified by these branches in accordance with subsection III.1.a above. The equipment locations are verified by review of the descriptions of the safety-related systems and the plant layout drawings in applicable sections of the SAR.

c. Normal, Abnormal, and Accident Environmental Conditions

The normal, abnormal, and accident environmental conditions must be explicitly defined for each item of equipment. These definitions must include the following parameters: temperature, pressure, relative humidity, radiation, chemicals, and vibration (nonseismic).

For the normal environment, specific values should be provided. For the abnormal and accident environments, these parameters should be presented as functions of time and the cause of the design bases environment (loss-of-coolant accident, steam line break, or other) should be identified.

The ICSB and PSB will verify that the normal, abnormal, and accident environments have been defined as indicated above for each item of equipment in their review scope.

d. Time Required to Operate

The length of time that each item of equipment is required to operate in the abnormal and accident environments must be provided. This time shall be compared with the qualification time to ensure margin. ICSB and PSB will verify the inclusion of this information. The secondary review branches will confirm the adequacy of the specified time interval for the equipment in their respective areas of primary review responsibility.

e. Environmental Qualification

The SAR should contain a complete description of the design bases and environmental qualification tests and analyses that have been (FSAR) or will be (PSAR) performed on each item of safety-related equipment. This should include qualification for the abnormal and accident environments, qualification for extreme normal operating environments, and qualification to assure that loss of environmental control systems that are not classified as safety-related will not adversely affect the operability of safety-related equipment, particularly electrical equipment located in the control room and other control equipment rooms. The ICSB and PSB will confirm that this information is

provided. The evaluation of the adequacy of the information is addressed below.

2. Evaluation Phase

The evaluation phase of the review involves the exercise of engineering judgement to determine from the information presented, particularly that regarding environmental qualification, whether an adequate demonstration of the required environmental capabilities of safety-related equipment will be or has been made. This phase of the review is performed after it has been established (by means of the information audit phase of the review previously described) that the information content requirements for Section 3.11 of the SAR have been satisfied. Although specifically written for use in evaluating the environmental qualification of Class I electric equipment, IEEE Std 323 and Regulatory Guide 1.89 contain principles and criteria that are comprehensive and generic to the qualification process itself; therefore, they are considered applicable to the environmental qualification of other types of equipment.

This phase of the review is performed as follows:

a. ICSB verifies that for each item of safety-related equipment in their scope of review, the environmental qualification program performed (FSAR) or proposed (PSAR) meets the detailed requirements of IEEE Std 323 and Regulatory Guide 1.89, with particular emphasis on the following:

   (1) The accuracy and validity of the definitions of the normal, abnormal, and accident environments are verified by checking against the appropriate environmental control system design requirements for normal and abnormal environments, and against the accident analyses with regard to accident environments resulting from loss-of-coolant accidents (LOCA) or steam or feedwater line breaks.

   (2) Type testing, or partial type testing in conjunction with one or more of the other methods, as defined in IEEE Std 323 (augmented by Regulatory Guide 1.89), must be used for qualifying equipment for design basis environments. The qualification method used (type test, operating experience, analysis, combined qualification, or on-going qualification) should be identified. The corresponding requirements of IEEE Std 323 (augmented by Regulatory Guide 1.89) then apply.

   (3) The type test must be designed to demonstrate that the equipment performance meets or exceeds the requirements of the equipment specifications for the plant, i.e., some margin must be demonstrated as indicated in IEEE Std 323. Margin is demonstrated by increasing the levels of testing, the number of test cycles, and the test duration.

   (4) The test sequence, i.e., the order of application of the simulated environmental conditions (e.g., aging, radiation, and vibration) during testing, must constitute the most severe sequence for the item being tested.

(5) The equipment being type tested should be operated under design operating conditions and adequately monitored during testing to determine functional performance characteristics.

(6) The equipment qualified by type testing must be prototypical of the actual equipment to be used in the plant. If this is not the case, a detailed analysis must be provided to justify the qualification.

b. PSB verifies that for each item of safety-related equipment in their scope of review, the environmental qualification program performed (FSAR) or proposed (PSAR) meets the detailed requirements of IEEE Std 323 and Regulatory Guide 1.89, with particular emphasis on the items identified in SRP Section 3.11-III-2a. In addition, the criteria and detailed requirements of IEEE Stds 317, 334, 382 and 383, and Regulatory Guides 1.40, 1.63, 1.73, and 1.131, should be used in conjunction with IEEE Std 323 in evaluating the environmental qualification program.

Upon request from the primary reviewer, the secondary review branches will provide input for the areas of review stated in subsection I. The primary reviewers obtains and uses such input as required to assure that this review procedure is complete.

c. The ASB, CSB, AAB, and RSB evaluate the validity of the descriptions of the normal, abnormal and accident environments in those areas of the plant for which they have primary review responsibility. The normal and abnormal environments are evaluated by means of a review of the design of the environmental control systems (ventilation, heating, cooling, air-conditioning); the accident environments by checking against the environmental conditions described in the accident analyses. The accident environments resulting from LOCA and from steam and feedwater line breaks are the responsibility of the AAB and RSB. The secondary review branches will advise ICSB and PSB of any inadequacy in the descriptions of the normal, abnormal, and accident environments.

d. The ASB evaluates the validity of the description of the environment resulting from the loss of environmental control systems (ventilation, heating, cooling, air-conditioning) in those areas of the plant which contain safety-related equipment, including the control room and other electrical equipment rooms. This evaluation is performed by review of the design of the respective environmental contol systems and calculation of the environment resulting from failure of the systems. The ASB will advise ICSB and PSB of any inadequacy in the descriptions of the environments resulting from the loss of environmental control systems.

e. The ASB, CSB, and RSB evaluate the acceptability of values provided in the SAR for the length of time that safety-related equipment is required to operate in the accident environment. This evaluation is performed by checking against the particular system or equipment operating requirements as postulated in the accident analysis. The secondary review branches will advise ICSB and PSB if any of the equipment accident environment operating times listed in the SAR are unacceptable.

f. QAB reviews the QA program to be applied to this activity which is described in Chapter 17 of the SAR. The QAB responsibilities and review procedures are set forth in SRP Sections 17.1 and 17.2.

g. AAB reviews the environmental qualification program to verify that the tests, analysis and documentation adequately address the radiation environments in each area of the plant. The AAB will advise the ICSB and PSB if any of the radiation environments are unacceptable. In addition the AAB reviews the chemical environment inside the containment by considering the total quantity of injection liquid and the total quantity of additives (e.g., NaOH, Na_2SO_3, N_2H_2). From this information the reviewer may calculate the weight and volume percent of the additive. The pH of the resulting solution can be calculated for appropriate combinations of equipment operation using generally accepted values of dissociation constants. (This information should be cross-checked with Section 6.5.2 of the applicant's safety analysis report.) See also SRP Section 6.5.2. The AAB also reviews the radiation environment integrated over the life of the nuclear reactor plant. A radiation source term consistent with Regulatory Guide 1.89 (Ref. 16) is assumed as appropriate to the air or water environment under consideration. From the source term information, the reviewer may calculate the radiation dose rates and integrated doses in the containment, ESF filters, and in equipment rooms housing ESF components. The results are compared with those of the applicant. The evaluation findings of the chemical and radiation environmental source terms are given to ICSB and PSB when there is a disagreement with the applicant's submittal.

IV. EVALUATION FINDINGS

The review should verify that sufficient information is contained in the SAR to support conclusions of the following type, to be included in the staff's safety evaluation report:

"The applicant has identified all the safety-related mechanical and electrical equipment, defined the normal, abnormal and accident environments that this equipment may be subjected to, and described the environmental qualification program that has been (for FSAR) or will be (for PSAR) performed to demonstrate its required environmental capability. It is concluded from this information that there is assurance that all items of safety-related equipment will be capable of performing needed safety functions under normal, abnormal and accident environmental conditions."

V. REFERENCES

1. 10 CFR Part 50, Appendix A, General Design Criterion 1, "Quality Standards and Records"; General Design Criterion 2, "Design Basis for Protection against Natural Phenomena"; General Design Criterion 4, "Environmental and Missile Design Bases"; and General Design Criterion 23, "Protection System Failure Modes."

2. 10 CFR Part 50, Appendix B.

3. IEEE Std 279 (ANSI N42.7-1972), "Criteria for Protection Systems for Nuclear Power Generating Stations," Institute of Electrical and Electronics Engineers.

4. IEEE Std 379, "Standard Application of the Single Failure Criterion to Nuclear Power Generating Station Class IE Systems."

5. IEEE Std 308, "Criteria for Class IE Electric Systems for Nuclear Power Generating Stations," Institute of Electrical and Electronics Engineers.

6. *IEEE Std 317, "Electric Penetration Assemblies in Containment Structures for Nuclear Power Generating Stations," Institute of Electrical and Electronics Engineers.

7. **IEEE Std 323, "General Guide for Qualifying Class I Electric Equipment for Nuclear Power Generating Stations," Institute of Electrical and Electronics Engineers.

8. *IEEE Std 334, "Guide for Type Tests of Continuous Duty Class I Motors Installed Inside the Containment of Nuclear Power Generating Stations," Institute of Electrical and Electronics Engineers.

9. *IEEE Std 382, "Guide for Type Test of Class I Electric Valve Operators for Nuclear Power Generating Stations," Institute of Electrical and Electronics Engineers.

10. *IEEE Std 383, "Standard for Type Test of Class IE Electric Cables, Field Splices, and Connections for Nuclear Power Generating Stations," Institute of Electrical and Electronics Engineers.

11. *Regulatory Guide 1.32, "Criteria for Class IE Electric Systems for Nuclear Power Generating Stations."

12. *Regulatory Guide 1.40, "Qualification Tests of Continuous Duty Motors Installed Inside the Containment of Water-Cooled Nuclear Power Plants" (this guide supplements IEEE Std 334).

13. *Regulatory Guide 1.53, "Application of Single Failure Criterion to Nuclear Power Plant Protection Systems"

14. *Regulatory Guide 1.63, "Electric Penetration Assemblies in Containment Structures for Water-Cooled Nuclear Power Plants" (this guide supplements IEEE Std 317).

*Acceptance criteria or evaluation guidance.
**Basic acceptance criteria.

15. *Regulatory Guide 1.73, "Qualification Tests of Electric Valve Operators Installed Inside the Containment of Nuclear Power Plants" (this guide supplements IEEE Std 382).

16. *Regulatory Guide 1.89, "Qualification of Class IE Equipment for Nuclear Power Plants" (this guide supplements IEEE Std 323).

17. *Regulatory Guide 1.131, "Qualification Tests of Electric Cables, Field Spices, and Connections for Light-Water-Cooled Nuclear Reactors" (this guide supplements IEEE Std 383).

*Acceptance criteria or evaluation guidance.
**Basic acceptance criteria.