UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

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DRAFT

June 19, 1995

MEMORANDUM TO: The Chairman
Commissioner Rogers
Commissioner de Planque
Commissioner Jackson

FROM: James M. Taylor
Executive Director for Operations

SUBJECT: SUMMARY OF PUBLIC COMMENTS RECEIVED ON PROPOSED REVISION OF PARTS 50, 52, AND 100

Attached is a brief history and summary of the public comments received on the second proposed revision of 10 CFR Parts 50, 52, and 100, and a listing of the commentors. As you may recall, the first proposed revision was issued for public comment in October 1992, and subsequently withdrawn. The second proposed revision was issued for public comment in the Federal Register on October 17, 1994 (59 FR 52255). The availability of draft guidance documents for public comment was published on February 28, 1995 (60 FR 10810). The comment period expired May 12, 1995; sixteen commentors responded to these announcements.

In the nonseismic area, several felt that the second proposed revision was an improvement since concerns regarding numerical values of population density and exclusion area distance in the rule had been satisfactorily addressed.

There was general agreement that the use of total effective dose equivalent (TEDE) is warranted. Differences of opinion were expressed on the numerical dose value proposed as an acceptance criterion and on the proposed use of the maximum dose received in any two-hour time period for evaluation purposes.

Most of the comments in the seismic area were supportive of the staff proposal. Many of the comments consisted primarily of editorial and technical suggestions that would clarify the rule or supporting guidance documents. A few of the comments are of a more substantive nature requiring a careful assessment of their implications.

The staff sees no unresolvable points of contention. The staff will be evaluating and resolving these comments, and plans to recommend a final rule to the Commission by the end of October 1995.

Attachment: As stated

cc: SECY OGC OCA OPA ACRS

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SUMMARY OF PUBLIC COMMENTS

10 CFR Parts 50, 52 and 100

Reactor Site Criteria Including Seismic and Earthquake Engineering Criteria for Nuclear Power Plants and Proposed Denial of Petition from Free Environment, Inc. et al.

## BACKGROUND

The first proposed revision to these regulations was published for public comment on October 20, 1992, (57 FR 47802). The availability of the draft regulatory guides and standard review plan sections that were developed to provide guidance on meeting the proposed regulations was published on November 25, 1992, (57 FR 55601). Because of the substantive nature of the changes to be made in response to public comments the proposed regulations and draft guidance documents were withdrawn and replaced with the second proposed revision of the regulations published for public comment on October 17, 1994. (FR 59 52255). The availability of the draft guidance documents was published on February 28, 1995, (FR 60 10810). The public comment period ended May 12, 1995.

The proposed regulatory action would apply to applicants who apply for a construction permit, operating license, preliminary design approval, final design approval, manufacturing license, early site permit, design certification, or combined license on or after the effective date of the final regulations.

Because the revised criteria presented in the proposed regulation would not be applied to existing plants, the licensing bases for existing nuclear power plants must remain part of the regulations. Therefore, the non-seismic and seismic reactor site criteria for current plants would be retained as Subpart A and Appendix A to 10 CFR Part 100, respectively. The proposed revised reactor site criteria would be added as Subpart B in 10 CFR Part 100 and would apply to site applications received on or after the effective date of the final regulations. Non-seismic site criteria would be added as a new §100.21 to Subpart B in 10 CFR Part 100. The criteria on seismic and geologic siting would be added as a new §100.23 to Subpart B in 10 CFR Part 100.

Criteria not associated with the selection of the site or establishment of the Safe Shutdown Earthquake Ground Motion (SSE) have been placed into 10 CFR Part 50. This action is consistent with the location of other design requirements in 10 CFR Part 50. The dose calculations and the earthquake engineering criteria would be located in 10 CFR Part 50 (§50.34(a) and Appendix S, respectively). Because Appendix S is not self executing, applicable sections of Part 50 (§50.34 and §50.54) are revised to reference Appendix S. The proposed regulation would also make conforming amendments to 10 CFR Part 52. Section 52.17(a)(1) would be amended to reflect changes in 50.34(a)(1) and 10 CFR Part 100.

The following draft regulatory guides and standard review plan sections were developed to provide prospective licensees with the necessary guidance for implementing the proposed regulation:

1. DG-1032, "Identification and Characterization of Seismic Sources and Determination of Shutdown Earthquake Ground Motions." The draft guide provides general guidance and recommendations, describes acceptable procedures and provides a list of references that present acceptable methodologies to identify and characterize capable tectonic sources and seismogenic sources.

2. DG-1033, Third Proposed Revision 2 to Regulatory Guide 1.12, "Nuclear Power Plant Instrumentation for Earthquakes." The draft guide describes seismic instrumentation type and location, operability, characteristics, installation, actuation, and maintenance that are acceptable to the NRC staff.

3. DG-1034, "Pre-Earthquake Planning and Immediate Nuclear Power Plant Operator Post-Earthquake Actions." The draft guide provides guidelines that are acceptable to the NRC staff for a timely evaluation of the recorded seismic instrumentation data and to determine whether or not plant shutdown is required.

4. DG-1035, "Restart of a Nuclear Power Plant Shut Down by a Seismic Event." The draft guide provides guidelines that are acceptable to the NRC staff for performing inspections and tests of nuclear power plant equipment and structures prior to restart of a plant that has been shut down because of a seismic event.

5. Draft Standard Review Plan Section 2.5.1, Proposed Revision 3, "Basic Geologic and Seismic Information." The draft describes procedures to assess the adequacy of the geologic and seismic information cited in support of the applicant's conclusions concerning the suitability of the plant site.

6. Draft Standard Review Plan Section 2.5.2, Second Proposed Revision 3 "Vibratory Ground Motion." The draft describes procedures to assess the ground motion potential of seismic sources at the site and to assess the adequacy of the SSE.

7. Draft Standard Review Plan Section 2.5.3, Proposed Revision 3, "Surface Faulting." The draft describes procedures to assess the adequacy of the applicant's submittal related to the existence of a potential for surface faulting affecting the site.

8. DG-4003, Second Proposed Revision 2 to Regulatory Guide 4.7, "General Site Suitability Criteria for Nuclear Power Plants." This guide discusses the major site characteristics related to public health and safety and environmental issues that the NRC staff considers in determining the suitability of sites.

SUMMARY OF COMMENTS ON REACTOR SITING CRITERIA (NONSEISMIC)

Eight organizations or individuals commented on the nonseismic aspects of the proposed revisions. The first proposed revision issued for comment in October 1992 elicited strong comments in regard to proposed numerical values of population density and a minimum distance to the exclusion area boundary (EAB) in the rule. This second proposed revision would delete these from the rule by providing guidance on population density in a Regulatory Guide and determining the distance to the EAB by the use of source term and dose calculations. Several commentors representing the nuclear industry and international nuclear organizations stated that this was a significant improvement over the first proposed revision, while the only public interest group commented that the NRC had retreated from decoupling siting and design in response to the comments of foreign entities.

Most comments on the second proposed revision centered on the use of total effective dose equivalent (TEDE), the proposed single numerical dose acceptance criterion of 25 rem TEDE, the evaluation of the maximum dose in any two-hour period, and the question of whether an organ capping dose should be adopted. Virtually all agreed that the concept of TEDE was appropriate and should be used. However, there were differing views on the proposed numerical dose of 25 rem and the proposed use of the maximum two-hour period to evaluate the dose. Virtually all industry commentors felt that the proposed numerical value was too low and that a "sliding" two-hour window for dose evaluation was confusing and inappropriate. All industry commentors opposed the use of an organ capping dose. The only public interest group that commented did not object to the use of TEDE, and believed the proposed dose value of 25 rem to be appropriate, but favored an organ capping dose. A summary of each commentors remarks follows.

Adams Atomic Engines, Inc.

There is no need to have a rule based on a traditional requirement to keep nuclear power plants far from population centers. Remote siting criteria are no longer necessary. The proposed rule has the potential for negative economic, environmental, and safety impacts on the general public, reactor suppliers and power plant operators.

The source term should be based on a maximum credible accident instead of an assumed "substantial meltdown of the core".

Ohio Citizens for Responsible Energy, Inc. (OCRE)

The proposed rule is unacceptable with respect to the nonseismic criteria. The NRC has retreated from decoupling siting and design as proposed in October 1992, in response to comments from foreign entities.

OCRE believes that the footnote in the proposed section 100.21(h) about considering economic factors is improper under Atomic Energy Act, since NRC may not consider costs to licensees.

OCRE has no objection to the use of TEDE; this is necessary if the new source term is to be used. The appropriate acceptance value is 25 rem. The NRC should also adopt an organ "capping" dose. No more than 35% of the total dose should be from a single organ.

Northeast Utilities Service System

Adopting dose criteria in terms of TEDE is consistent with recent guidance (ICRP, EPA). TEDE captures the overall potential health consequences, and is the most practical approach for limiting the combined effect to all organs. 25 rem is appropriate and consistent with the value established in other guidance documents, such as EPA 400, as an acceptable exposure to an individual.

An organ "capping" dose is not necessary since design basis accidents do not involve only iodine.

The requirement to determine the maximum two-hour period (for dose calculation) is not practical, nor necessary. It unduly complicates the radiological analysis. The concept also questions the resulting conclusion. If an individual received less than 25 rem from an exposure from 30 minutes to 2 hours and 30 minutes, what about the dose received before the 30 minute period?

ABB Combustion Engineering Nuclear Systems

Expresses concern that a site approved under present Part 100 for a currently operating reactor might not be approved for an advanced light water reactor (ALWR) under the proposed Part 100. This presents a quandary since ALWR has improved safety features.

ABB-CE fully supports the use of TEDE. The proposed dose limit was first estimated at 27 rem. NRC staff adjusted this downward to 25 rem without explanation. The value of 27 rem is more appropriate; however, the development of a more technically justifiable criterion should be pursued.

It is not clear that cancer risk is the best parameter for maintaining same level of protection. Offsite dose limit does not represent an acceptable dose to any member of the public, but is a "figure of merit". The activity corresponding to the current 300 rem thyroid and 25 rem whole body should be calculated for conservative weather conditions. ABB also strongly believes that the dose acceptance criterion should also reflect consideration of any contribution from the additional nuclides identified in NUREG-1465.

Advanced Light Water Reactor (ALWR) Program

The ALWR Program supports the use of TEDE, but not a dose acceptance criterion of 25 rem. Based on organ weighting factors given in ICRP 26, 25 rem whole body and 300 rem thyroid are equivalent to a value of 34 rem TEDE. Based on organ weighting factors given in ICRP 60 (using a revised thyroid weighting factor), the current dose criteria are equivalent to 40 rem TEDE.

There is no need for an organ capping dose, since iodine is unlikely to be present by itself.

ALWR suggests as an alternate criterion that the dose at the exclusion area boundary (EAB) should not exceed 40 rem TEDE over a 24 hour period. Also proposes significant changes in the way that meteorology dispersion factors (X/Q) are calculated, since the present approach in Reg. Guide 1.145 is overly conservative.

If 2 hour dose calculation is retained, it should begin with the start of the accident which should be defined as no later than the start of the gap release to the containment. While this does not tie the dose calculation to the declaration of a General Emergency, it reflects that reality far better than a sliding 2 hour window.

Nuclear Energy Institute (NEI)

NRC staff is to be congratulated for carefully considering and responding to complex public comments on the first proposed revision. Many troubling aspects of the first proposed revision have been addressed in a forthright and appropriate manner.

NEI supports the use of TEDE. This will support a uniform and consistent implementation of realistic source terms. A value of 25 rem is more restrictive than the current dose criteria, and does not represent the total stochastic risk, because the value of 300 rem thyroid is about 9 rem TEDE. NRC should determine the appropriate numerical value utilizing the total stochastic risk implied by the current criteria, and this should be incorporated without additional conservatism or adjustment.

An organ capping dose limit is not practical nor necessary.

There is little justification for changing the 0 to 2 hour dose calculation, at least partly because other aspects of the calculation (e.g., meteorology) are not yet clear as to how they would be calculated.

NEI supports NRC's proposed approach with regard to population density criteria, as stated in draft Regulatory Guide DG-4004. NEI supports the concept of environmental justice, but expresses concerns regarding subjective phrases and potential implementation. Recommends that the environmental justice provision be deleted from this revision of the Guide until more detailed guidance becomes available.

Morgan, Lewis and Bockius

The 1994 proposed rule is a major improvement over the previous version.

Morgan, Lewis and Bockius expresses concerns in regard to 2 changes. The proposed dose acceptance criterion of 25 rem TEDE could make NRCs accident dose limits significantly more restrictive, without any showing that these are necessary to protect public health and safety.

The proposed change from an immediate 2 hour period to a moving 2 hour period will impose another unidentified penalty, depending upon the design. Also, the change is contrary to common sense, since it requires that during an accident a member of the public will move toward the plant rather than away from it.

Westinghouse Electric Corporation

The proposed "sliding dose window" is not linked to any specific occurrence, and ignores any dose accumulated during the time between accident initiation and the two hour interval of highest dose. A more reasonable approach would be to replace it with a time interval of two hours starting with the onset of core damage plus the time interval between accident initiation and the onset of core damage. Westinghouse also proposes consideration of an additional dose criterion that the 24 hour dose at the exclusion area boundary (EAB) should not exceed twice the acceptable 2 hour dose at the same location.

Endorses the use of TEDE, but believes that the risk associated with the current dose limits would support a significantly higher numerical dose value than the value of 25 rem proposed. There is no need for an organ "capping" dose, which would result in an unnecessary complication without reducing risk to the public.

SUMMARY OF COMMENTS ON SEISMIC AND EARTHQUAKE ENGINEERING CRITERIA

A total of eleven individuals or organizations commented on the proposed revisions. A general assessment of the comments is that most are supportive of the staff positions. Many of the commentors have provided editorial and technical suggestions that would clarify the rulemaking. A few commentors provided more substantive comments requiring a careful assessment of their implications. The following is a summary of each commentor's input with focus principally on their recommendations.

American Society of Civil Engineers (Washington Office)

The seismic design and engineering criteria of ASCE Standard 4, "Seismic Analysis of Safety-Related Nuclear Structures and Commentary on Standard for Seismic Analysis of Safety-Related Nuclear Structures," should be incorporated by reference into the regulation.

G.C. Slagis Associates

Comments are limited to pressure-retaining components to the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Section III rules. Questions the soundness of only the Safe Shutdown Earthquake Ground Motion (SSE) being used for design, that is, the elimination of Operating Basis Earthquake Ground Motion (OBE) response analyses. Also, provided technical comments on supplementary NRC staff positions on fatigue analysis (positions established in certification review of ALWRs) and post-earthquake inspections (DG-1034, DG-1035).

Wais and Associates

Commends the NRC staff for adopting the probabilistic seismic hazard approach versus the deterministic approach for the Central and Eastern United States.

Site investigations are performed at four levels with the amount of detail based on distance from the site. Recommends reducing the outer area of geological and seismic investigations (DG-1032) and not restricting the updating of the LLNL and EPRI probabilistic seismic hazard databases to only situations that lead to higher hazard estimates (DG-1032). Questions the logic used to define the reference probability for the SSE exceedance level (Appendix B to DG-1032). Also questions the need for seismic instrumentation (DG-1033), and the need for plant shutdown if the OBE is exceeded and no damage is apparent (DG-1034).

ABB-CE

Agrees with the NRC staff's proposal to not require explicit design analysis of the OBE if its peak acceleration is less than one-third of the SSE.

Department of Energy (Office of Civilian Radioactive Waste Management)

Requests an explicit statement whether or not the proposed regulations apply to the Mined Geologic Disposal System (MGDS) and a Monitored Retrievable Storage (MRS) facility. Site investigations are performed at four levels with the amount of detail based on distance from the site. Recommends that the stated outer area of investigations should be reduced and that the applicant should justify its rationale for the area of investigations considered (DG-1032, SRP Sections 2.5.1 and 2.5.2).

Nuclear Energy Institute

Congratulates the NRC staff for carefully considering and responding to the voluminous and complex comments that were provided on the earlier proposed rulemaking package and considers that the seismic portion of the proposed rulemaking package is nearing maturity and with the inclusion of industry's comments, has the potential to satisfy the objectives of predictable licensing and stable regulations.

Supports the regulation format, that is, prescriptive guidance is located in regulatory guides or standard review plan sections not the regulation.

Supports the removal of the requirement from the first proposed rulemaking that both deterministic and probabilistic evaluations must be conducted to determine site suitability and seismic design requirements for the site. However, does not agree with the NRC staff's deterministic check of the seismic sources and parameters used in the LLNL and EPRI probabilistic seismic hazard analyses (DG-1032). Also, does not support the NRC staff's deterministic check of the applicants submittal (SRP Section 2.5.2).

The regulation and guidance documents should state that if an ALWR is to be sited at an existing nuclear power plant site, only confirmatory investigations of foundation conditions are required (Regulatory Guide 1.132). Also, state that for existing sites east of approximately 105° west longitude a 0.3g standardized design level is acceptable.

For nuclear power plants founded on rock sites the licensee should have the option to use the containment basemat data (instead of free-field data) to determine OBE exceedance (DG-1034).

Provided over 60 specific technical or editorial comments on the seismic portion of the rulemaking (regulation, regulatory guides and standard review plan sections).

Morgan, Lewis & Bockius

Concerned with the emphasis on the probabilistic analysis to establish the SSE. Although Section 100.23 states that a suitable sensitivity analysis can be used to address uncertainties in the SSE, DG-1032 contains no discussion for addressing uncertainties in the SSE except for performing a probabilistic seismic hazard analysis (PSHA). Also, there is no clear statement in DG-1032 that if a PSHA is performed no further analysis is necessary or if a suitable sensitivity analysis is performed a PSHA is not necessary.

Yankee Atomic Electric Company

At existing Eastern United States sites (rock or soil) or at rock Eastern United States sites not located in areas of high seismicity (for example, Charleston, South Carolina, New Madrid, Missouri, Attica, New York) a 0.3g standardized ALWR design is acceptable and only evaluations of foundation conditions at the site are required (Regulatory Guide 1.132) but not geologic/geophysical seismological investigations. For other sites a DG-1032 review is required.

Proposes an alternative to DG-1032 that incorporates soil amplification into the probabilistic analysis, does not allow scaling of the SRP Section 2.5.2 site specific spectra to define the SSE, but allows the scaling of broad-banded spectra to define the SSE.

Kinemetrics, Inc.

In general, agrees with Draft Regulatory Guide DG-1033. However, cannot comply with the battery capacity recommendations in the draft guide. Also, recommends that Regulatory Position 4.3 of an earlier draft regulatory guide (DG-1016) addressing the interconnection of instrumentation for common starting and common timing be reinstated in the final guide.

TU Electric

The recommendation for fatigue analysis in Regulatory Position 1.2 of DG-1035 should be limited to ASME Code Class 1 components and systems. Also, clarify Regulatory Position 1.3 in DG-1035, the analysis recommendation for non-safety related systems and components.

Westinghouse Electric Corporation

Supports NRC staff decision to move guidance material from the rule to regulatory guides. Supports NRC staff decision to eliminate the "dual" deterministic and probabilistic analyses from the proposed rule. Concerned that retaining deterministic evaluations in SRP Section 2.5.2 will lead to confusion as to whether future licensees will also need to perform a deterministic analysis even though such an analysis is only recommended for NRC staff to perform as a "sanity" check. Shares NEI's concern with respect to the type of analyses needed to construct a new plant on an existing approved site, using the proposed rule and associated regulatory guides.

List of Commentors on Proposed Revision of Parts 50, 52, and 100*

| Number | Commentor | Nonseismic | Seismic | Both |
|---|---|---|---|---|
| 1 | Adams Atomic Engines, Inc. | X | | |
| 2 | American Society of Civil Engineers (ASCE) | | X | |
| 3 | Ohio Citizens for Responsible Energy (OCRE) | X | | |
| 4 | G. C. Slagis Associates | | X | |
| 5 | Northeast Utilities | X | | |
| 6 | Wais and Associates | | X | |
| 7 | Wais and Associates | | X | |
| 8 | ABB Combustion Engineering Nuclear Systems | | | X |
| 9 | Advanced Light Water Reactor Program (ALWR) | X | | |
| 10 | U. S. Department of Energy | | X | |
| 11 | Nuclear Energy Institute (NEI) | | | X |
| 11A | Nuclear Energy Institute -- Supplementary | X | | |
| 12 | Morgan, Lewis and Bockius | | | X |
| 13 | Yankee Atomic Electric Company | | X | |
| 14 | Kinemetrics, Inc. | | X | |
| 15 | TU Electric | | X | |
| 16 | Westinghouse Electric Corporation | | | X |

* Does not include requests for extension of the comment period.

# DRAFT COMMISSION PAPER

19

COMM-PPR.R6 3/14/96

FOR: The Commissioners

FROM: James M. Taylor
Executive Director for Operations

SUBJECT: REVISIONS TO 10 CFR PART 100 AND 10 CFR PART 50, AND NEW APPENDIX S TO 10 CFR PART 50

PURPOSE:

To obtain Commission approval to publish a final rule revising requirements for reactor siting in 10 CFR Part 100 and 10 CFR Part 50, including a new Appendix S to 10 CFR Part 50, for use by future applicants.

SUMMARY:

This paper and accompanying enclosures present, for Commission approval, a draft final rule revising 10 CFR Part 100 and 10 CFR Part 50, and a final new Appendix S to 10 CFR Part 50. These would amend the Commission's regulations regarding reactor siting for future nuclear power plant applicants by describing basic reactor site criteria and to reflect advancements in the earth sciences and earthquake engineering.

The revised Part 100 consists of two subparts. To preserve the licensing basis for existing plants, Subpart A and Appendix A to Part 100 would be identical to the present rule. Subpart B, applicable to future plants, would contain basic nonseismic site criteria, without numerical values, in a new proposed Section 100.21, "Nonseismic Siting Criteria." Seismic criteria would appear in a new Section 100.23, "Geologic and Seismic Siting Factors." Revisions to 10 CFR Part 50 would contain source term and dose criteria (Section 50.34) and earthquake engineering criteria (new Appendix S).

The revision to 10 CFR 50.34 reflects the staff recommendation and rationale for the revised dose criteria to be used to judge the applicability of plant designs. This paper also contains a differing view on this section provided by the Office of Nuclear Regulatory Research.

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BACKGROUND:

On April 12, 1962, the Atomic Energy Commission (AEC) issued 10 CFR Part 100, "Reactor Site Criteria" (27 FR 3509). On November 13, 1973, the AEC issued Appendix A to 10 CFR Part 100, "Seismic and Geologic Siting Criteria for Nuclear Power Plants," (38 FR 31279).

A proposed rule to revise Part 100, Appendix A to Part 100, and sections of Part 50 was published for comment on October 20, 1992 (57 FR 47802). The proposed rule change combined two separate initiatives dealing with non-seismic and seismic issues, and included a minimum distance to the exclusion area boundary of 0.4 miles, guideline limits for population density, and required both probabilistic and deterministic seismic hazard evaluations. The comment period, extended twice, expired on June 1, 1993. Extensive comments, both domestic and international, were received.

The Commission was briefed on August 3, 1993, on the status of the proposed rule and the nature of the comments received. In an SRM dated August 12, 1993, the Commission raised several concerns regarding the prescriptive aspects of the proposed revisions to Part 100 as well as its form and content. In response, the staff prepared an options paper, SECY-94-017, dated January 26, 1994. In an SRM dated March 28, 1994, the Commission approved the staff recommendations; however, due to the substantive nature of the changes to be made to the rule the Commission stated that both parts were to be resubmitted for Commission review and reissued for public comment prior to the final rulemaking. Outlines of the draft regulatory guides and standard review plan section were to be submitted to the Commission for review, to demonstrate how the basic site criteria are to be implemented. The draft regulatory guides and standard review plan section were to be issued for public comment after receiving Commission approval of the outlines.

The second proposed revision to these regulations was published for public comment on October 17, 1994 (59 FR 52255). On February 8, 1995, the NRC stated (60 FR 7467) that it intended to extend the comment period to allow interested persons adequate time to provide comments on staff guidance documents. On February 28, 1995, the availability of the five draft regulatory guides and three draft standard review plan sections that were developed to provide guidance on meeting the proposed regulations was published (60 FR 10 80) and the comment period for the proposed rule was extended to May 12, 1995 (60 FR 10810).

Included in this package are the Federal Register notice for the final rule (Attachment 1), the resolution of public comments on the reactor site criteria and seismic and earthquake engineering criteria (Attachments 2 and 3), the ACRS letter on the rulemaking (Attachment 4), a draft public announcement (Attachment 5), and the draft congressional letters (Attachment 6).

DISCUSSION:

NON-SEISMIC ASPECTS;

Proposed rule

The proposed rule issued for comment on October 17, 1994 (FR 59 52255) would retain the use of source term and dose calculations (relocating these to Part 50) to determine the distance to the exclusion area boundary (EAB) and the size of the outer radius of the low population zone (LPZ). The proposed dose criteria would require that an individual located at any point on the boundary of the exclusion area for any two-hour period following the onset of the postulated fission product release not receive a dose in excess of 25 rem total effective dose equivalent (TEDE). Similarly, an individual located at the outer boundary of the LPZ for the entire period of the cloud passage (taken to be 30 days) must not receive a dose in excess of 25 rem TEDE.

Section 100.21 proposed to contain basic site criteria without any numerical values. With regard to population density, the proposed rule stated that:

> Reactor sites should be located away from very densely populated centers. Areas of low population density are, generally, preferred. However, in determining the acceptability of a particular site located away from a very densely populated center but not in an area of low density, consideration will be given to safety, environmental, economic, or other factors, which may result in the site being found acceptable.

Revision 2 of Regulatory Guide 4.7 would contain guidance on preferred population density as follows:

> A reactor preferably should be located such that at the time of initial site approval and within about 5 years thereafter, the population density, including weighted transient population, averaged over any radial distance out to 20 miles (cumulative population at a distance divided by the circular area at that distance) does not exceed 500 persons per square mile. A reactor should not be located at a site whose population density is well in excess of the above value.
>
> If the population density of the proposed site exceeds, but is not well in excess of the above preferred value, an analysis of alternative sites should be conducted for the region of interest with particular attention to alternative sites having lower population density. However, consideration will be given to other factors, such as safety, environmental, or economic considerations, which may result in the site with the higher population density being found acceptable. Examples of such factors include, but are not limited to, the higher population density site having superior seismic characteristics, better access to skilled labor for construction, better rail or highway access, shorter transmission line requirements, or less environmental impact upon undeveloped areas, wetlands, or endangered species.

Public Comments:

Eight organizations or individuals commented on the nonseismic aspects of the second proposed revision. A summary of the public comments received was transmitted to the Commission in a memorandum dated June 19, 1995. The first proposed revision issued for comment in October 1992 elicited strong comments in regard to proposed numerical values of population density and a minimum distance to the exclusion area boundary (EAB) in the rule. The second proposed revision would delete these from the rule by providing guidance on population density in a Regulatory Guide and determining the distance to the EAB and LPZ by use of source term and dose calculations. The rule would contain basic site criteria, without any numerical values.

Several commentors representing the nuclear industry and international nuclear organizations stated that the second proposed revision was a significant improvement over the first proposed revision, while the only public interest group commented that the NRC had retreated from decoupling siting and design in response to the comments of foreign entities.

Most comments on the second proposed revision centered on the use of total effective dose equivalent (TEDE), the proposed single numerical dose acceptance criterion of 25 rem TEDE, the evaluation of the maximum dose in any two-hour period, and the question of whether an organ capping dose should be adopted.

Virtually all commentors supported the concept of TEDE and its use. However, there were differing views on the proposed numerical dose of 25 rem and the proposed use of the maximum two-hour period to evaluate the dose. Virtually all industry commenters felt that the proposed numerical value of 25 rem TEDE was too low and that it represented a "ratchet" since the use of the current dose criteria plus organ weighting factors would suggest a value of 34 rem TEDE. In addition, all industry commenters believed the "sliding" two-hour window for dose evaluation to be confusing, illogical and inappropriate. They favored a rule that was based upon a two hour period after the onset of fission product release, similar in concept to the existing rule. All industry commenters opposed the use of an organ capping dose. The only public interest group that commented did not object to the use of TEDE, favored the proposed dose value of 25 rem, and supported an organ capping dose.

Final Rule:

10 CFR 100.34

No changes in the final rule are proposed as compared with the proposed rule. The final rule would require, as in the proposed rule, that an individual located at any point on the boundary of the exclusion area for any two hour period following onset of the postulated fission product release, not receive a radiation dose in excess of 25 rem total effective dose equivalent (TEDE). Similarly, an individual located at the outer boundary of the low population zone (LPZ), who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage) not receive a dose in excess of 25 rem TEDE. The staff recommends adoption of a dose acceptance criterion of 25 rem TEDE based upon consideration of the risk of latent cancer fatality, as noted in the Statement of Considerations that accompanied the proposed rule. Since the TEDE concept accounts for the contribution from all body organs, the staff recommends that no additional organ "capping" dose be required. With respect to the two hour evaluation period, the Office of Nuclear Reactor Regulation (NRR) continues to support the regulatory approach for the two hour dose evaluation period that was articulated in the proposed revision published on October 17, 1994.

In licensing reactor designs, the Office of Nuclear Reactor Regulation (NRR) assesses the performance of engineered safety features (ESFs) by calculating (1) the 2-hour *projected* dose from a *postulated* Design Basis Accident (DBA) to a *hypothetical* individual at any location at or beyond the EAB and (2) the *projected* dose over the course of the accident at the outer boundary of the LPZ. These DBA assessments are surrogates for evaluating the accident mitigation capability included in the design.

The current assessment of accident mitigation systems has been linked directly to the instantaneous release to, and mixing of, fission products in containment based on the 1962 report, TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites." Improved understanding of severe accidents, published in 1995 as NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants," indicates that fission product releases to containment do not occur instantaneously and the bulk of the releases may not take place for an hour or more. The updated insights reduce, but do not eliminate, uncertainties in the timing, magnitude, and chemical form of severe accident source terms. The staff supports the use of updated source term insights and the use of updated radiobiological insights [i.e., the shift to a total effective dose equivalent (TEDE) criterion] for the review of accident mitigation capability (i.e., DBAs). Use of the updated insights will result in more realistic analytical treatment of the delay, reduction, and removal of fission products in containment from engineered and natural processes.

There are important licensing implications in the decision to consider one dose evaluation period approach over another (any 2-hour period or first 2-hour period). The approach in the 1994 proposed revision, that the dose criterion not be exceeded for "any 2 hour period," considers the worst 2 hours of offsite exposure. Under the "current licensing framework" (Part 100 and TID source term insights), the worst 2-hour evaluation period <u>is</u> the first 2-hour evaluation period because the fission products are assumed to be released instantaneously into containment; consequently, the release to the environment and the EAB dose rate decrease monotonically over time. Therefore, arguments can be made that either the worst-2 hour or the first-2 hour approach, in conjunction with the "proposed licensing framework" (updated source term and radiobiological insights), would be consistent with prior practice. From the radiological perspective, independent of which 2-hour dose evaluation period is considered, the proposed licensing framework will result in a relaxation of accident mitigation systems design requirements when compared to the current licensing framework. This is primarily because of the change in the dose criteria (from the 25 rem to the whole body or 300 rem to the thyroid criteria to a single 25 rem TEDE criterion) reflecting updated radiobiological insights. The thyroid dose exposure guideline at the EAB generally has been the limiting standard for DBA siting analyses.

The timing, magnitude and chemical form of the source terms of NUREG-1465 used in DBA calculations result in the gap release period lasting for about 30 minutes followed by the early in-vessel release period lasting for about one and one half hours (1.3 hours for a PWR and 1.5 hours for a BWR). The buildup of radioactive material into the large containment volume is gradual and peaks at the end of the early in-vessel release period. The gap activity does not significantly contribute to the TEDE dose when compared to the activity released during the early in-vessel phase (i.e., typically less than 10% of the integrated TEDE dose). With the first-2 hour approach, a high concentration of radioactive material is present in containment for only the last part of the 2-hour period and the dose evaluation period would end when the release to the environment would be at or near its peak value. With the worst-2 hour approach, the evaluation period brackets those two hours when the containment concentration (and, therefore, the release to the environment) is at its peak.

The staff believes that it is important to select that period of the DBA for which accident mitigation features are most severely challenged to take account of uncertainties in source terms and accident phenomenology. For source term constructions other than NUREG-1465 (e.g., EPRI source term), the first-2 hour dose evaluation period could end as fission product releases are continuing and containment concentrations have not yet peaked. The staff believes that the worst-2 hour approach provides a consistent regulatory scheme to judge the performance of accident mitigation capability under its greatest DBA challenge. The worst-2 hour assessment is not sensitive to the calculation of initial delay times and accident scenario progression, is easy to perform and to reproduce, and it is an improvement over the current approach that uses the instantaneous release to containment.

Sensitivity studies performed by the staff indicate that the first-2 hour standard can be met without any accident mitigation system beyond containment (depending on assumptions about the removal of fission products from natural processes). Reliance solely on preventive systems for public protection is inconsistent with the Commission's traditional defense-in-depth philosophy and guidance on achieving the balance between accident prevention and mitigation. For those reactor designs with effective accident mitigation systems (e.g., evolutionary reactor designs), the worst-2 hour and first-2 hour evaluation values converge.

The significant differences arise between the worst-2 hour and first-2 hour approaches for those designs that would rely on reducing fission product releases using passive features. Determinations need to be made regarding which updated source term insights are directly transferrable for advanced reactor design reviews. Differences in applicant and staff views on passive fission product removal rates need to be reconciled where experience and experimental data is limited. Evolutionary and advanced LWR designs subject to the 10 CFR Part 52 approval process establish, by rule, a site dilution parameter ($\chi/Q$) that reflects the performance of the design-specific accident preventive and mitigative features to assure that the dose criteria are not exceeded. Under certain design/site combinations and a first-2 hour approach, sliding the two hour evaluation period a short time into the accident scenario could result in an exceedance of the dose criterion. The worst-2 hour approach should preclude such cliffs which are dependent upon accident progression and removal rate assumptions.

A number of the positions discussed above were provided to the Commission as the staff developed its implementation plans for use of the updated source term insights in licensing (see SECYs 94-302 and 95-172, and Memoranda dated September 6, 1994 and July 21, 1995). These positions include:

(1) The evaluation of accident mitigation capability must assess the effectiveness of the design and accompanying ESFs to delay, reduce, and remove radioactive material prior release to the public.

(2) Risk insights should be used with the proposed licensing framework to reduce unnecessary conservatisms to the extent that it complements the NRC's deterministic approach and defense-in-depth philosophy. Risk assessments were at the foundation of the updated source term and radiobiological insights.

(3) The EAB 2-hour dose standard, consistent with historical precedent, should continue to be used as a surrogate to evaluate the performance of accident mitigation capability for Design Basis Accidents.

The staff believes that (1) the proposed licensing framework would provide a relaxation of ESF performance requirements commensurate with updated source term and radiobiological insights, (2) the regulatory requirements for determination of in-containment radioactive material during the 2-hour dose evaluation period should be consistent and capable of handling designs substantially different from those analyzed in NUREG-1465, (3) the analysis should be easy to perform and reproducible with confidence, and (4) the technical bases and analytical methods should be defensible. For these reasons, the staff recommends the worst-2 hour approach for the dose evaluation period.

The Office of Nuclear Regulatory Research (RES) has a differing view with regard to the time period over which the dose is to be evaluated to an individual at the exclusion area boundary and is providing it to the Commission for consideration. RES recommends that the final rule be modified from any two hour period after release of fission products (referred to as the "worst" two hours) to a period of two hours commencing with fuel failure plus the time period from accident initiation until fuel failure begins (referred to as the "first" two hours).

RES believes that the use of the worst two hour period in the dose calculation is not justified by risk considerations (i.e., not consistent with the intent of the Commission's August 16, 1995, Policy Statement on the use of PRA Methods in Nuclear Regulatory Activities or the Regulatory Analysis Guidelines - NUREG/BR-0058, Rev. 2) and could lead to increased costs for future licensees with no commensurate gain in safety. In addition, RES believes that the two hour period should be tied to the early stages of the plant and engineered safety feature performance (thus providing a means of early public protection) and not arbitrarily to any period during the duration of the accident which tends to duplicate the function of the 30-day LPZ dose criteria. Use of the worst two-hour dose also tends to remove some of the incentive for designers to develop designs which delay the onset of core damage, since credit for this delay would be limited to radioactive decay. In RES' view, this is not consistent with the Commission's Policy Statement on Advanced Reactors of July 8, 1986 which encouraged future designs to have, among other attributes, longer time constants prior to reaching safety system challenge. These items are discussed further below.

Since use of the first two hours results in an evaluation period having a somewhat lower integrated concentration of fission products within containment than the worst two hour period, as was proposed, it would result in a lower calculated dose. The difference in calculated dose between the first two hours vs. the worst two hours depends primarily upon the efficacy and rapidity of any fission product cleanup systems incorporated in the plant design. For current designs employing active fission product cleanup systems, the difference in calculated dose is essentially negligible; however, for designs relying upon passive fission product removal, the difference in calculated dose is expected to be about a factor of two. Hence, the impact of use of the first two hours for dose evaluation could be a design with a somewhat higher allowable containment leak rate or slightly smaller distance to the exclusion area boundary than one where the dose was evaluated for the worst two hours. From a risk standpoint, these factors are essentially negligible, and do not justify increased costs associated with greater engineered safety feature (ESF) performance or a larger distance to the exclusion area boundary. This was demonstrated in information supporting the recent change to 10 CFR 50, Appendix J (60 FR 49495) where the Commission accepted the potential for an increase in containment leak rate on the basis of risk and benefit considerations.

The use of the "first" two hours reflects recent research results on fission product timing and appearance within containment as reflected in "Accident Source Terms for Light-Water Nuclear Power Plants," NUREG-1465, issued in 1995. This time period allows for the release into containment of the "gap" and "early in-vessel" release phases and provides a substantial fission product release for plant and site evaluation that is fully consistent with the intent of the regulation, as noted in Footnote 1 of 10 CFR 100, that the postulated accident "result in substantial meltdown of the core with subsequent release of appreciable quantities of fission products." RES believes that the use of the "worst" two hours for the dose evaluation does not give appropriate consideration to these research insights regarding timing and appearance of fission products. Use of the "first" two hours also provides a clear incentive to a designer to delay the release of fission products into containment, thereby promoting designs with enhanced safety characteristics.

RES also believes that the use of the "first" two hours adds credibility and enhances licensee and public understanding with regard to the actions of an individual presumed to be located at the exclusion boundary. The focus of the two hour dose should be to assess the degree of protection provided to the public in the initial stages of an accident. The use of the "first" two hours keeps this focus. The use of the worst two hours does not and tends to duplicate the purpose of the 30 day dose.

Although the revised dose evaluation in 10 CFR 50.34 is intended for future plants, there is a staff concern that a current licensee might seek to use it to remove or disable existing fission product cleanup systems. This could markedly change the risk profile of the plant from that which was licensed. RES considers that any proposed changes in the plant configuration based upon revised source terms and dose criteria must be examined from an overall integrated risk perspective to provide assurance that the safety margin of the design has not been unduly reduced. Appropriate language can be developed and included in the Statement of Considerations to provide such assurance.

10 CFR 100.21

No comments were received that proposed changes to the regulation and no changes are recommended by the staff in the final rule.

Regulatory Guide 4.7

One comment, while supporting the concept of environmental justice, expressed concern regarding subjective phrases and potential implementation, and recommended that the environmental justice provision be deleted from this version of the Guide until more detailed guidance becomes available. The staff recognizes that detailed implementation guidance may not yet be available in this area, but recommends that the environmental justice provision be retained in issuing this Guide in final form.

SEISMIC ASPECTS:

Proposed Rule:

Because no significant changes were made to the regulations published for public comment this discussion will focus on the differences between the current (Appendix A to Part 100) and final regulations (Section 100.23 to Part 100 and Appendix S to Part 50) and staff resolution of the public comments.

Final Rule:

Because the criteria presented in the regulation will not be applied to existing plants, the licensing bases for existing nuclear power plants must remain part of the regulations. Therefore, the criteria on seismic and geologic siting are designated as a new Section 100.23 to 10 CFR Part 100 and added to the existing body of regulations. In addition, earthquake engineering criteria are located in 10 CFR Part 50, in a new Appendix S. Since Appendix S is not self executing, applicable sections of Part 50 (§50.8 and §50.34) are revised to reference Appendix S. Conforming amendments to 10 CFR Parts 52 and 100 are also made. Sections 52.17(a)(1), 52.17(a)(1)(vi), 100.8, and 100.20(c)(1) and (3) are amended to note Section 100.23 to Part 100 or Appendix S to Part 50.

Geologic and Seismic Siting

The regulations and guidance documents reflect new information and research results, and comments from the public. In response to the August 12, 1993, SRM pertaining to the prescriptive aspects of the first proposed revisions to Part 100 as well as its form and content, the final regulation only contains the basic requirements; the detailed guidance similar to that contained in Appendix A to 10 CFR Part 100 has been removed to guidance documents. Thus, the new regulation (Section 100.23 to Part 100) contains: (a) required definitions, (b) a requirement to determine the geological, seismological, and engineering characteristics of the proposed site, and (c) requirements to determine the Safe Shutdown Earthquake Ground Motion (SSE), to determine the potential for surface deformation, and to determine the design bases for seismically induced floods and water waves. Detailed guidance, that is, procedures acceptable to the NRC staff for meeting the requirements, is contained in Regulatory Guide 1.165, "Identification and Characterization of Seismic Sources and Determination of Safe Shutdown Earthquake Ground Motion," (Draft was DG-1032). NRC staff review guidelines is provided in Standard Review Plan (SRP) Section 2.5.2, "Vibratory Ground Motion," Revision 3. Two other SRP sections, 2.5.1, "Basic Geologic and Seismic Information," and 2.5.3, "Surface Faulting," are also revised to assure consistency among the rule, SRP Section 2.5.2, and Regulatory Guide 1.165.

The existing approach for determining a Safe Shutdown Earthquake Ground Motion (SSE) for a nuclear reactor site, embodied in Appendix A to 10 CFR Part 100, relies on a "deterministic" approach. Using this deterministic approach, an applicant develops a single set of earthquake sources, develops for each source a postulated earthquake to be used as the source of ground motion that can affect the site, locates the postulated earthquake according to prescribed rules, and then calculates ground motions at the site.

Although this approach has worked reasonably well for the past two decades, in the sense that SSEs for plants sited with this approach are judged to be suitably conservative, the approach has not explicitly recognized uncertainties in geosciences parameters. Because of the uncertainty about earthquake phenomena (especially in the eastern United States), there have often been differences of opinion and differing interpretations among experts as to the largest earthquakes to be considered and ground-motion models to be used, thus often making the licensing process relatively cumbersome.

Over the past decade, analysis methods for incorporating these different interpretations have been developed and used. These "probabilistic" methods have been designed to allow explicit incorporation of different models for zonation, earthquake size, ground motion, and other parameters. The advantage of using these probabilistic methods is their ability to not only incorporate different models and different data sets, but also to weight them using judgments as to the validity of the different models and data sets, and thereby providing an explicit expression for the uncertainty in the ground motion estimates and a means of assessing sensitivity to various input parameters. Another advantage of the probabilistic method endorsed in Regulatory Guide 1.165 is the target exceedance probability is set by examining the design bases of more recently licensed nuclear power plants resulting in a more uniform level of safety from site to site.

The revision to the regulation now explicitly recognizes that there are inherent uncertainties in establishing the seismic and geologic design parameters and allows for the option of using a probabilistic seismic hazard methodology capable of propagating uncertainties as a means to address these uncertainties. The rule further recognizes that the nature of uncertainty and the appropriate approach to account for it depend greatly on the tectonic regime and parameters, such as, the knowledge of seismic sources, the existence of historical and recorded data, and the understanding of tectonics. Therefore, methods other than the probabilistic methods, such as sensitivity analyses, may be adequate to account for uncertainties for some sites.

The key elements of the approach exemplified in Regulatory Guide 1.165 and Standard Review Plan Section 2.5.2 are:

a. Conduct site-specific and regional geoscience investigations.
These investigations are performed to determine specific characteristics of the proposed site, such as, the presence or absence of potential seismic sources, capable faults at or near the site, characterization of the rock and soil strata, earthquake history of the site and environs, etc. In addition to characterizing the site, these data are needed to verify that regional characteristics used in the Lawrence Livermore National Laboratory (LLNL) or the Electric Power Research Institute (EPRI) probabilistic seismic hazard assessments (PSHA) are valid for the proposed site.

b. Target exceedance probability is set by examining the design bases of more recently licensed nuclear power plants.
The target exceedance probability is the median annual probability of exceeding the Safe Shutdown Earthquake (SSE) for operating nuclear power plant that were designed to Regulatory Guide 1.60 or to a similar spectrum. This value has been determined to be 1E-5/year.

c. Determine if information from geoscience investigations change probabilistic results.
The applicant conducts an evaluation that demonstrates that the data obtained from the site investigations (Step a. above) do not provide information that would necessitate revision of the seismic sources used in the existing seismic hazard studies and their characteristics or attenuation models.

d. Conduct probabilistic seismic hazard analysis and determine ground motion level corresponding to the target exceedance probability.
The applicant conducts a LLNL or EPRI PSHA for the proposed site to obtain a seismic hazard curve, ground acceleration or spectral amplitude vs. annual probability of exceedance. The hazard curve median is deaggregated to determine a seismic event described by an average earthquake magnitude and distance (distance from earthquake to the nuclear power plant site) which contributes most to the ground motion level corresponding to the target exceedance

probability. This magnitude and distance is then used in subsequent steps to determine site-specific spectral shape.

e. Dete[illegible]ne site-specific spectral shape and scale this shape to the ground motion level determined above.
The applicant will use the seismic event of magnitude and distance determined in Step d to develop site-specific spectral shapes in accordance with SRP 2.5.2 procedures and additional guidance provided in the regulatory guide. The SRP procedures, in part, are based on use of seismic recorded motions or ground motion models appropriate for the event, region and site under consideration.

f. NRC staff review of ground motion.
The NRC staff will review the applicant's proposed SSE ground motion to assure that it takes into account all available data including insights and information gained from previous licensing experience.

g. Update the data base and reassess probabilistic methods at least every ten years.
To keep the regulatory guidance on the probabilistic methods and their seismic hazard data base current, the NRC would reassess them at least every ten years and update them as appropriate.

The results of the regional and site-specific investigations must be considered in the application of the probabilistic method. The current probabilistic methods (the NRC sponsored study conducted by LLNL or the EPRI seismic hazard study), are regional studies without detailed information on any specific location. The specific applicant's geosciences investigations are used to update the database used by the probabilistic hazard methodology to assure that all appropriate information is incorporated.

It is also necessary to incorporate local site geological factors such as stratigraphy and to account for site-specific geotechnical properties in establishing the design basis ground motion. In order to incorporate local site factors and advances in ground motion attenuation models, ground motion estimates are determined using the procedures that are outlined in Standard Review Plan Section 2.5.2.

The NRC staff's approach to evaluating an application is described in SRP Section 2.5.2. This review takes into account the information base developed in licensing more than 100 plants. Although the premise in establishing the target exceedance probability is that the current design levels are adequate, a staff review assures that there is consistency with previous licensing decisions and that the scientific basis for decisions are clearly understood. This review approach will also assist in assessing the fairly complex regional probabilistic modeling which incorporates multiple hypotheses and a multitude of parameters. Furthermore, this process should provide a clear basis for the staff's decisions and facilitate communication with nonexperts.

Earthquake Engineering

Criteria not associated with the selection of the site or establishment of the Safe Shutdown Earthquake Ground Motion (SSE) have been placed into Part 50. This action is consistent with the location of other design requirements in Part 50. The regulation is a new Appendix S, "Earthquake Engineering Criteria for Nuclear Power Plants," to Part 50.

In the current regulation, Appendix A to Part 100, the Operating Basis Earthquake Ground Motion (OBE), the vibratory ground motion that will assure safe continued operation, is one-half the SSE. In Appendix S this requirement has been deleted and replaced with two options: (1) applicant selection of an OBE that is either one-third of the SSE or less, or (2) a value greater than one-third of the SSE. With the OBE level set at one-third or less of the SSE, only the SSE is used for design; the OBE only serves the function of an inspection and shutdown level. If the OBE is greater than one-third of the SSE, the current practice of using both the OBE and SSE for design continues; and in addition, the OBE serves the function of an inspection and shutdown level. This change responds to one of the major criticisms with the existing regulations, that the OBE controls the design of some parts of the plant.

For new applicationsthe regulation would treat plant shutdown associated with vibratory ground motion exceeding the OBE (or significant plant damage) as a condition in every operating license. Section 50.54 is revised accordingly. Related plant shutdown and OBE exceedance guidelines for operating plants are being developed separately by NRR.

Procedures acceptable to the NRC staff for meeting the requirements in the new regulation will be contained in three regulatory guides, (a) Regulatory Guide 1.12, "Nuclear Power Plant Instrumentation for Earthquakes," Revision 2 (Draft was DG-1033), (b) Regulatory Guide 1.166, "Pre-Earthquake Planning and Immediate Nuclear Power Plant Operator Postearthquake Actions" (Draft was DG-1034), and (c) Regulatory Guide 1.167, "Restart of a Nuclear Power Plant Shut Down by a Seismic Event" (Draft was DG-1035).

Public Comments

Seven letters were received addressing either the regulations or both the regulations and the draft guidance documents. An additional five letters were received addressing only the guidance documents, for a total of twelve comment letters.

10 CFR 100.23

No changes were made to the regulation as a result of the public comments. In general, the commentors were supportive of the regulation, specifically, the removal of prescriptive guidance from the regulation and locating it in regulatory guides or standard review plan sections and the removal of the requirement from the first proposed rulemaking (57 FR 47802) that both deterministic and probabilistic evaluations must be conducted to determine site suitability and seismic design requirements for the site.

A suggestion that for existing sites east of approximately 105$^0$ west longitude (the Rocky Mountain front), a 0.3g standardized design level be codified was not adopted. The NRC has determined that the use of a spectral shape anchored to 0.3g peak ground acceleration as a standardized design level would be appropriate for existing sites based on the current state of knowledge. However, as new information becomes available it may not be appropriate for future licensing decisions. Pertinent information such as that described in Regulatory Guide 1.165 (Draft was DG-1032) is needed to make that assessment. Therefore, it is not appropriate to codify the request.

The suggestion to change the regulation to enable an applicant for an operating license already holding a construction permit to apply the amended methodology and criteria in Subpart B to Part 100 was not incorporated. The NRC will address this request on a case-by-case basis rather than through a generic change to the regulations. This situation pertains to a limited number of facilities in various stages of construction. Some of the issues that must be addressed by the applicant and NRC during the operating license review include differences between the design bases derived from the current and amended regulations (Appendix A to Part 100 and Section 100.23, respectively), and earthquake engineering criteria such as, OBE design requirements and OBE shutdown requirements.

An explicit statement whether or not Section 100.23 to Part 100 applies to the Mined Geologic Disposal System (MGDS) and a Monitored Retrievable Storage (MRS) facility was not added to the regulation or Supplemental Information Section of the rule. Presently, NUREG-1451, "Staff Technical Position on Investigations to Identify Fault Displacement Hazards and Seismic Hazards at a Geologic Repository," notes that Appendix A to 10 CFR Part 100 does not apply to a geologic repository. Section 72.102(b) requires that, for an MRS located west of the Rocky Mountain front or in areas of known potential seismic activity in the east, the seismicity be evaluated by the techniques of Appendix A to 10 CFR Part 100. The applicability of Section 100.23 to other than power reactors, if considered appropriate by the NRC, would be a separate rulemaking. That rulemaking would clearly state the applicability of Section 100.23 to an MRS or other facility. In addition, NUREG-1451 will remain the NRC staff technical position on seismic siting issues pertaining to a MGDS until it is superseded through a rulemaking, revision of NUREG-1451, or other appropriate mechanism.

<u>Appendix S to 10 CFR Part 50</u>

Support for the NRC position pertaining to the elimination of the Operating Basis Earthquake Ground Motion (OBE) response analyses has been documented in various NRC publications such as SECY-79-300, SECY-90-016, SECY-93-087, and NUREG-1061. The final safety evaluation reports related to the certification of the System 80+ and the Advanced Boiling Water Reactor design (NUREG-1462 and NUREG-1503, respectively) have already adopted the single earthquake design philosophy. In addition, similar activities are being done in foreign countries, such as, Germany. However, one commentor expressed concern about the elimination of OBE response analyses of pressure-retaining components designed to the ASME Boiler and Pressure Vessel Section III rules. Positions pertaining to the elimination of the OBE were proposed in SECY-93-087. Commission approval is documented in a memorandum from Samuel J. Chilk to James M. Taylor, Subject: SECY-93-087 - Policy, Technical and Licensing Issues

Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs, dated July 21, 1993. Item V(B)(5), "Value of the Operating Basis Earthquake Ground Motion (OBE) and Required OBE Analysis," to the supplemental information to the regulations was slightly modified to address the noted concerns.

The regulation was not changed to incorporate by reference the American Society of Civil Engineers (ASCE) Standard 4, "Seismic Analysis of Safety-Related Nuclear Structures and Commentary on Standard for Seismic Analysis of Safety-Related Nuclear Structures." In response to the August 12, 1993, SRM pertaining to the prescriptive aspects of the first proposed revisions to Part 100 as well as its form and content, the final regulation contains only the basic requirements;the detailed guidance is provided in regulatory guides and standard review plan sections. ASCE Standard 4 is cited in the 1989 revision of Standard Review Plan Sections 3.7.1, 3.7.2, and 3.7.3.

The reference to aftershocks in Paragraph IV(b), Surface Deformation was deleted. Paragraphs VI(a)(1), "Safe Shutdown Earthquake," and VI(b)(3) of Appendix A to Part 100 contain the phrase "including aftershocks." In the proposed regulation the "including aftershocks" phrase was only removed from the Safe Shutdown Earthquake Ground Motion requirements (Paragraph IV(a)(1) of Appendix S to Part 50).

Guidance Documents

Many of the commentors have provided editorial and technical suggestions that would clarify the documents. A few commentors provided more substantive comments requiring a careful assessment of their implications. For example, the Staff clarified the procedure in SRP Section 2.5.2 used to assess the adequacy of an applicants submittal. Also, Regulatory Guide 1.165 (Draft was DG-1032) discusses how uncertainties in the SSE can be addressed through a suitable sensitivity analysis. In general, no technical changes were made to the staff positions described in the draft guidance documents.

It is anticipated that the availability of the related regulatory guidance and standard review plan sections will be published in the Federal Register coincident with the effective date of the final regulations.

RECOMMENDATIONS:

That the Commission:

1. Approve publication of the Revisions to the Regulatory Requirements for Reactor Siting (Seismic and Nonseismic) and Earthquake Engineering Criteria in 10 CFR Parts 100 and 50 (Attachment 1) as a final rule

2. Certify that this rule will not have a significant economic effect on a substantial number of small entities pursuant to the Regulatory Flexibility Act of 1980 (5 U.S.C. 605(b)).

3. Note:

a. The final rule will be published in the Federal Register and become effective 30 days after publication.

b. The reporting and recordkeeping requirements contained in this regulation have been approved by the Office of Management and Budget, OMB approval Numbers 3150-0093 and 3150-0011.

c. A public announcement (Attachment 5) will be issued when the notice of rulemaking is sent to the Office of the Federal Register.

d. The appropriate Congressional committees will be informed (Attachment 6).

f. Copies of the Federal Register notice will be distributed to all power reactor licensees. The notices will be sent to other interested parties upon request.

g. The Chief Counsel for Advocacy of the Small Business Administration will be notified of the Commission's determination, pursuant to the Regulatory Flexibility Act of 1980 (5 U.S.C. 605 (b)), that this rule will not have a significant economic effect on a substantial number of small entities.

h. The availability of the final regulatory guides and standard review plan sections will be published in the Federal Register subsequent to the effective date of the final rule.

i. A copy of "Resolution of Public Comments on the Proposed Seismic and Earthquake Engineering Criteria for Nuclear Power Plants" (Attachment 2), will be placed in the Public Document Room and sent to interested parties upon request.

Attachments:
1. Federal Register Notice of Rulemaking
2. Resolution of Public Comments on the Proposed Seismic and Earthquake Engineering Criteria for Nuclear Power Plants
3. ACRS Letter
4. Draft Public Announcement
5. Draft Congressional Letters
6. Regulatory Analysis
7. Environmental Assessment

FEDERAL REGISTER NOTICE OF RULEMAKING

NUCLEAR REGULATORY COMMISSION

10 CFR Parts 50, 52 and 100

RIN 3150-AD93

# Reactor Site Criteria

# Including Seismic and Earthquake Engineering Criteria for

# Nuclear Power Plants

# and Proposed Denial of Petition from Free Environment, Inc. et. al.

**AGENCY:** Nuclear Regulatory Commission.

**ACTION:** Final rule and denial of petition from Free Environment, Inc. et.al.

**SUMMARY:** The Nuclear Regulatory Commission (NRC) is amending its regulations to update the criteria used in decisions regarding power reactor siting, including geologic, seismic, and earthquake engineering considerations for future nuclear power plants. The rule would allow NRC to benefit from experience gained in the application of the procedures and methods set forth in the current regulation and to incorporate the rapid advancements in the earth sciences and earthquake engineering. This rule primarily consists of two separate changes, namely, the source term and dose considerations, and the seismic and earthquake engineering considerations of reactor siting. The Commission is also denying the remaining issue in petition (PRM-50-20) filed by Free Environment, Inc. et. al.

**EFFECTIVE DATE:** (30 days after publication in the Federal Register).

**FOR FURTHER INFORMATION CONTACT:** Dr. Andrew J. Murphy, Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission, Washington, DC 20555, telephone (301) 415-6010, concerning the seismic and earthquake engineering aspects and Mr. Leonard Soffer, Office of the Executive Director for Operations, U.S. Nuclear Regulatory Commission, Washington DC 20555, telephone (301) 415-1722, concerning other siting aspects.

SUPPLEMENTARY INFORMATION:



## I. Background

The present regulation regarding reactor site criteria (10 CFR Part 100) was promulgated April 12, 1962 (27 FR 3509). NRC staff guidance on exclusion area and low population zone sizes as well as population density was issued in Regulatory Guide 4.7, "General Site Suitability Criteria for Nuclear Power Stations," published for comment in September 1974. Revision 1 to this guide was issued in November 1975. On June 1, 1976, the Public Interest Research Group (PIRG) filed a petition for rulemaking (PRM-100-2) requesting that the NRC incorporate minimum exclusion area and low population zone distances and population density limits into the regulations. On April 28, 1977, Free Environment, Inc. et. al., filed a petition for rulemaking (PRM-50-20). The remaining issue of this petition requests that the central Iowa nuclear project and other reactors be sited at least 40 miles from major population centers. In August 1978, the Commission directed the NRC staff to develop a general policy statement on nuclear power reactor siting. The "Report of the Siting Policy Task Force" (NUREG-0625) was issued in August 1979 and provided recommendations regarding siting of future nuclear power reactors. In the 1980 Authorization Act for the NRC, the Congress directed the NRC to decouple siting from design and to specify demographic criteria for siting. On July 29, 1980 (45 FR 50350), the NRC issued an Advance Notice of Proposed Rulemaking (ANPRM) regarding revision of the reactor site criteria, which discussed the recommendations of the Siting Policy Task Force and sought public comments. The proposed rulemaking was deferred by the Commission in December 1981 to await development of a Safety Goal and improved research on accident source terms. On August 4, 1986 (51 FR 23044), the NRC issued its Policy Statement on Safety Goals that stated quantitative health objectives with regard to both prompt and latent cancer fatality risks. On December 14, 1988 (53 FR 50232), the NRC denied PRM-100-2 on the basis that it would unnecessarily restrict NRC's regulatory siting policies and would not result in a substantial increase in the overall protection of \ ublic health and safety. B-ause of possible renewed interest in power reactor siting, the NRC is proceed..g with a rulemaking in this area. The Commission proposes to address the remaining issue in PRM-50-20 as part of this rulemaking action.

Appendix A, "Seismic and Geologic Siting Criteria for Nuclear Power Plants," to 10 CFR Part 100 was originally issued as a proposed regulation on November 25, 1971 (36 FR 22601), published as a final regulation on November 13, 1973 (38 FR 31279), and became effective on December 13, 1973. There have been two amendments to 10 CFR Part 100, Appendix A. The first amendment, issued November 27, 1973 (38 FR 32575), corrected the final regulation by adding t.. legend under the diagram. The second amendment res..ted from a petition rulemaking (PRM 100-1) requesting that an opinion be issued that would inte..ret and clarify Appendix A with respect to the determination of the Safe Shutdown Earthquake. A notice of filing of the petition was published on May 14, 1975 (40 FR 20983). The substance of the petitioner's proposal was accepted and published as an immediately effective final regulation on January 10, 1977 (42 FR 2052).

The first proposed revision to these regulations was published for public comment on October 20, 1992, (57 FR 47802). The availability of the five draft regulatory guides and the standard review plan section that were developed to provide guidance on meeting the proposed regulations was published on November 25, 1992, (57 FR 55601). The comment period for the proposed regulations was extended two times. First, the NRC staff initiated an extension (58 FR :71) from February 17, 1993 to March 24, 1993, to be consistent with the comment period on the draft regulatory guides and standard review plan section. Second, in response to a request from the public, the comment period was extended to June 1, 1993 (58 FR 16377).

The second proposed revision to these regulations was published for public comment on October 17, 1994 (59 FR 52255). The NRC stated on February 8, 1995, (60 FR 7467) that it intended to extend the comment period to allow interested persons adequate time to provide comments on staff guidance documents. On February 28, 1995, the availability of .. five draft regulatory guides and three standard review plan sect.. ..t were developed to provide guidance on meeting the proposed regulation. .. published (60 FR 10880) and the comment period for the proposed rule was ..ended to May 12, 1995 (60 FR 10810).

## II. Objectives

The objectives of this regulatory action are to --

1. State basic site criteria for future sites that, based upon experience and importance to risk, have been shown as key to protecting public health and safety;

2. Provide a stable regulatory basis for seismic and geologic siting and applicable earthquake engineering design of future nuclear power plants that will update and clarify regulatory requirements and provide a flexible structure to permit consideration of new technical understandings; and

3. Relocate source term and dose requirements that apply primarily to plant design into 10 CFR Part 50.

## III. Genesis

The regulatory action reflects changes that are intended to (1) benefit from the experience gained in applying the existing regulation and from research; (2) resolve interpretive questions; (3) provide needed regulatory flexibility to incorporate state-of-the-art improvements in the geosciences and earthquake engineering; and (4) simplify the language to a more "plain English" text.

The regulatory action would apply to applicants who apply for a construction permit, operating license, preliminary design approval, final design approval, manufacturing license, early site permit, design certification or combined license on or after the effective date of the final regulations. However, if the construction permit was issued prior to the effective date of the final regulations the operating license applicant shall comply with the seismic and geologic siting criteria and the earthquake engineering criteria in Appendix A to 10 CFR Part 100.

Criteria not associated with the selection of the site or establishment of the Safe Shutdown Earthquake Ground Motion (SSE) have been placed into 10 CFR Part 50. This action is consistent with the location of other design requirements in 10 CFR Part 50.

Because the revised criteria presented in the regulation would not be applied to existing plants, the licensing bases for existing nuclear power plants must remain part of the regulations. Therefore, the non-seismic and seismic reactor site criteria for current plants would be retained as Subpart A and Appendix A to 10 CFR Part 100, respectively. The revised reactor site criteria would be added as Subpart B in 10 CFR Part 100 and would apply to site applications received on or after the effective date of the final regulations. Non-seismic site criteria would be added as a new §100.21 to Subpart B in 10 CFR Part 100. The criteria on seismic and geologic siting would be added as a new §100.23 to Subpart B in 10 CFR Part 100. The dose calculations and the earthquake engineering criteria would be located in 10 CFR Part 50 (§50.34(a) and Appendix S, respectively). Because Appendix S is not self executing, applicable sections of Part 50 (§50.34 and §50.54) are revised to reference Appendix S. The regulation would also make conforming amendments to 10 CFR Part 52. Section 52.17(a)(1) would be amended to reflect changes in 50.34(a)(1) and 10 CFR Part 100.

## IV. Alternatives

The first alternative considered by the Commission was to continue using current regulations for site suitability determinations. This is not considered an acceptable alternative. Accident source terms and dose calculations currently primarily influence plant design requirements rather than siting. It is desirable to state basic site criteria which, through importance to risk, have been shown to be key to assuring public health and safety. Further, significant advances in understanding severe accident behavior, including fission product release and transport, as well as in the earth sciences and in earthquake engineering have taken place since the promulgation of the present regulation and deserve to be reflected in the regulations.

The second alternative considered was replacement of the existing regulation with an entirely new regulation. This is not an acceptable alternative because the provisions of the existing regulations form part of the licensing bases for many of the operating nuclear power plants and others that are in various stages of obtaining operating licenses. Therefore, these provisions should remain in force and effect.

The approach of establishing the revised requirements in new sections to 10 CFR Part 100 and relocating plant design requirements to 10 CFR Part 50 while retaining the existing regulation was chosen as the best alternative. The public will benefit from a clearer, more uniform, and more consistent licensing process that incorporates updated information and is subject to fewer interpretations. The NRC staff will benefit from improved regulatory implementation (both technical and legal), fewer interpretive debates, and increased regulatory flexibility. Applicants will derive the same benefits in addition to avoiding licensing delays caused by unclear regulatory requirements.

## V. MAJOR CHANGES

### *A. Reactor Siting Criteria (Nonseismic).*

Since promulgation of the reactor site criteria in 1962, the Commission has approved more than 75 sites for nuclear power reactors and has had an opportunity to review a number of others. In addition, light-water commercial power reactors have accumulated about 1800 reactor-years of operating experience in the United States. As a result of these site reviews and operational experience, a great deal of insight has been gained regarding the design and operation of nuclear power plants as well as the site factors that influence risk. In addition, an extensive research effort has been conducted to understand accident phenomena, including fission product release and transport. This extensive operational experience together with the insights gained from recent severe accident research as well as numerous risk studies on radioactive material releases to the environment under severe accident conditions have all confirmed that present commercial power reactor design, construction, operation and siting is expected to effectively limit risk to the public to very low levels. These risk studies include the early "Reactor Safety Study" (WASH-1400), published in 1975, many Probabilistic Risk Assessment (PRA) studies conducted on individual plants as well as several specialized studies, and the recent "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," (NUREG-1150), issued in 1990. Advanced reactor designs currently under review are expected to result in even lower risk and improved safety compared to existing plants. Hence, the substantial base of knowledge regarding power reactor siting, design, construction and operation reflects that the primary factors that determine public health and safety are the reactor design, construction and operation.

Siting factors and criteria, however, are important in assuring that radiological doses from normal operation and postulated accidents will be acceptably low, that natural phenomena and potential man-made hazards will be appropriately accounted for in the design of the plant, and that site characteristics are amenable to the development of adequate emergency plans to protect the public and adequate security measures to protect the plant. The Commission has also had a long standing policy of siting reactors away from densely populated centers, and is continuing this policy in this rule.

The Commission is incorporating basic reactor site criteria in this rule to accomplish the above purposes. The Commission is retaining source term and dose calculations to verify the adequacy of a site for a specific plant, but source term and dose calculations are relocated to Part 50, since experience has shown that these calculations have tended to influence plant design aspects such as containment leak rate or filter performance rather than siting. No specific source term is referenced in Part 50. Rather, the source term is required to be one that is "... assumed to result in substantial meltdown of the core with subsequent release into the containment of appreciable quantities of fission products." Hence, this guidance can be utilized with the source term currently used for light-water reactors, or used in conjunction with revised accident source terms.

The relocation of source term and dose calculations to Part 50 represent a partial decoupling of siting from accident source term and dose calculations. The siting criteria are envisioned to be utilized together with standardized plant designs whose features will be certified in a separate design certification rulemaking procedure. Each of the standardized designs will specify an atmospheric dilution factor that would be required to be met, in order to meet the dose criteria at the exclusion area boundary. For a given standardized design, a site having relatively poor dispersion characteristics would require a larger exclusion area distance than one having good dispersion characteristics. Additional design features would be discouraged in a standardized design to compensate for otherwise poor site conditions.

Although individual plant tradeoffs will be discouraged for a given standardized design, a different standardized design could require a different atmospheric dilution factor. For custom plants that do not involve a standardized design, the source term and dose criteria will continue to provide assurance that the site is acceptable for the proposed design.

Rationale for Individual Criteria

A. Exclusion Area. An exclusion area surrounding the immediate vicinity of the plant has been a requirement for siting power reactors from the very beginning. This area provides a high degree of protection to the public from a variety of potential plant accidents and also affords protection to the plant from potential man-related hazards. The Commission considers an exclusion area to be an essential feature of a reactor site and is retaining this requirement, in Part 50, to verify that an applicant's proposed exclusion area distance is adequate to assure that the radiological dose to an individual will be acceptably low in the event of a postulated accident. However, as noted above, if source term and dose calculations are used in conjunction with standardized designs, unlimited plant tradeoffs to compensate for poor site conditions will not be permitted. For plants that do not involve standardized designs, the source term and dose calculations will provide assurance that the site is acceptable for the proposed design.

The present regulation requires that the exclusion area be of such size that an individual located at any point on its boundary for two hours immediately following onset of the postulated fission product release would not receive a total radiation dose in excess of 25 rem to the whole body or 300 rem to the thyroid gland. A footnote in the present regulation notes that a whole body dose of 25 rem has been stated to correspond numerically to the once in a lifetime accidental or emergency dose to radiation workers which could be disregarded in the determination of their radiation exposure status (NBS Handbook 69 dated June 5, 1959). However, the same footnote also clearly states that the Commission's use of this value does not imply that it considers it to be an acceptable limit for an emergency dose to the public under accident conditions, but only that it represents a reference value to be used for evaluating plant features and site characteristics intended to mitigate radiological consequences of accidents in order to provide assurance low risk to the public under postulated accidents. The Commission, based upon extensive experience in applying this criterion, and in recognition of the conservatism of the assumptions in its application (a large fission product release within containment associated with major core damage, maximum allowable containment leak rate, a postulated single failure of any of the fission product cleanup systems, such as the containment sprays, adverse site meteorological dispersion characteristics, an individual presumed to be located at the boundary of the exclusion area at the centerline of the plume for two hours without protective actions), believes that this criterion has clearly resulted in an adequate level of protection. As an illustration of the conservatism of this assessment, the maximum whole body dose received by an actual individual during the Three Mile Island accident in March 1979, which involved major core damage, was estimated to be about 0.1 rem.

The proposed rule considered two changes in this area.

First, the Commission proposed that the use of different doses for the whole body and thyroid gland be replaced by a single value of 25 rem, total effective dose equivalent (TEDE).

The proposed use of the total effective dose equivalent, or TEDE, was noted as being consistent with Part 20 of the Commission's regulations and was also based upon two considerations. First, since it utilizes a risk consistent methodology to assess the radiological impact of all relevant nuclides upon all body organs, use of TEDE promotes a uniformity and consistency in assessing radiation risk that may not exist with the separate whole body and thyroid organ dose values in the present regulation. Second, use of TEDE lends itself readily to the application of updated accident source terms, which can vary not only with plant design, but in which additional nuclides besides the noble gases and iodine are predicted to be released into containment.

The Commission considered the current dose criteria of 25 rem whole body and 300 rem thyroid with the intent of selecting a TEDE numerical value equivalent to the risk implied by the current dose criteria.

The Commission proposed to use the risk of latent cancer fatality as the appropriate risk measure since quantitative health objectives (QHOs) for it have been established in the Commission's Safety Goal policy. Although the supplementary information in the proposed rule noted that the current dose criteria are equivalent in risk to 27 rem TEDE the Commission proposed to use 25 rem TEDE as the dose criterion for plant evaluation purposes, since this value is essentially the same level of risk as the current criteria.

However, the Commission specifically requested comments on whether the current dose criteria should be modified to utilize the total effective dose equivalent, or TEDE, concept, on whether a TEDE value of 25 rem (consistent with latent cancer fatality), or 34 rem (consistent with latent cancer incidence), or some other value should be used, and whether the dose criterion should also include a "capping" limitation, that is, an additional requirement that the dose to any individual organ not be in excess of some fraction of the total.

Based on the comments received, there was a general consensus that the use of the TEDE concept was appropriate, and a nearly unanimous opinion that no organ "capping" dose was required, since the TEDE concept itself provided the appropriate risk weighting for all body organs.

With regard to the value to be used as the dose criterion, a number of comments [illegible] received that the proposed value of 25 rem TEDE represented a more restrictive criterion than the current values of 25 rem whole body and 300 rem to the thyroid gland. These commenters noted that the use of organ weighting factors of 1 for the whole body and 0.03 for the thyroid (as listed in ICRP Report 26), would yield a value of 34 rem TEDE for a whole body and thyroid doses of 25 and 300 rem, respectively. As noted in the discussion accompanying the proposed rule, the Commission was aware of this argument and noted that this represented the use of risk of latent cancer incidence, rather than the risk of latent cancer fatality, as for the proposed rule.

After careful consideration, the Commission has decided to adopt a value of 25 rem TEDE as the dose acceptance criterion for the final rule. The bases for this decision follows. First, the Commission has generally based its regulations on the risk of latent cancer fatality, rather than latent cancer incidence. A strict numerical calculation suggests a value of 27 rem TEDE, as noted in the discussion that accompanied the proposed rule. The Commission concludes that a value of 27 rem is sufficiently close to a value of 25 rem, and that the use of 27 rather than 25 implies an unwarranted numerical precision. The argument that 25 rem TEDE represents a undue tightening of the dose criterion, while true in theory, is not true in practice. A review of dose analyses for operating plants has shown that the thyroid dose limit of 300 rem has been the limiting dose criterion in licensing reviews, and that all operating plants would be able to meet a dose criterion of 25 rem TEDE. Finally, the Commission notes that the value of 25 rem TEDE is currently recommended as an acceptable dose for emergency radiation workers. While the Commission does not, as noted above, regard this dose value as one that is acceptable for members of the public under accident conditions, the Commission concludes that it provides a useful perspective with regard to doses that ought not to be exceeded, even for radiation workers under emergency conditions. In adopting this value, the Commission also rejects the view, advanced by some, that the dose calculation is merely a "reference" value that bears no relation to what might be experienced by an actual person in an accident. Although the Commission considers it highly unlikely, because of the conservative assumptions made, that an actual person would receive such a dose, under accident conditions, it is conceivable.

The second change proposed in this area was in regard to the time period that a hypothetical individual is assumed to be at the exclusion area boundary. While the duration of the time period remains at a value of two hours, the proposed rule stated that this time period not be fixed in regard to the appearance of fission products within containment, but that various two-hour periods be examined with the objective that the dose to an individual not be in excess of 25 rem TEDE for any two-hour period after the appearance of fission products within containment. The Commission proposed this change to reflect improved understanding of fission product release into the containment under severe accident conditions. For an assumed instantaneous release of fission products, as contemplated by the present rule, the two hour period that commences with the onset of the fission product release clearly results in the highest dose to a hypothetical individual offsite. Improved understanding of severe accidents shows that fission product releases to the containment do not occur instantaneously, and that the bulk of the releases may not take place for about an hour or more. Hence, the two-hour period commencing with the onset of fission product release may not represent the highest dose that an individual could be exposed to over any two-hour period. As a result, the Commission proposed that various two-hour periods be examined to assure that the dose to a hypothetical individual at the exclusion area boundary would not be in excess of 25 rem TEDE over any two-hour period after the onset of fission product release.

A number of comments received in regard to this proposed criterion stated that so-called "sliding" two-hour window for dose evaluation at the exclusion area boundary was confusing, illogical and inappropriate. Several commenters felt it was difficult to ascertain which two hour period represented the maximum. Others expressed the view that the significance of such a calculation was not clearly stated nor understood. For example, one comment expressed the view that a dose evaluated for a "sliding" two-hour period was logically inconsistent since it implied either that an individual was not at the exclusion area boundary prior to the accident, and approached close to the plant after initiation of the accident, contrary to what might be expected, or that the individual was, in fact, located at the exclusion area boundary all along, in which case the dose contribution received prior to the "maximum" two hour value was being ignored.

Although the Commission recognizes that evaluation of the dose to a hypothetical individual over any two-hour period may not be entirely consistent with the actions of an actual individual in an accident, the intent is to assure that the short-term dose to an individual will not be in excess of the acceptable value, even where there is some variability in the time that an individual might be located at the exclusion area boundary. In addition, the dose calculation should not be taken too literally with regard to the actions of a real individual, but rather is intended primarily as a means to evaluate the effectiveness of the plant design and site characteristics in mitigating postulated accidents.

For these reasons, the Commission is retaining the requirement, in the final rule, that the dose to an individual located at the nearest exclusion area boundary over any two-hour period after the appearance of fission products in containment, should not be in excess of 25 rem total effective dose equivalent (TEDE).

B. Site Dispersion Factors Site dispersion factors have been utilized to provide an assessment of dose to an individual as a result of a postulated accident. Since the Commission is requiring that a verification be made that the exclusion area distance is adequate to assure that the guideline dose to a hypothetical individual will not be exceeded under postulated accident conditions, as well as to assure that radiological limits are met under normal operating conditions, the Commission is requiring that the atmospheric dispersion characteristics of the site be evaluated, and that site dispersion factors based upon this evaluation be determined and used in assessing radiological consequences of normal operations as well as accidents.

C. Low Population Zone. The present regulation requires that a low population zone (LPZ) be defined immediately beyond the exclusion area. Residents are permitted in this area, but the number and density must be such that there is a reasonable probability that appropriate protective measures could be taken in their behalf in the event of a serious accident. In addition, the nearest densely populated center containing more than about 25,000 residents must be located no closer than one and one-third times the outer boundary of the LPZ. Finally, the dose to a hypothetical individual located at the outer boundary of the LPZ over the entire course of the accident [illegible] not be in excess of the dose values given in the regulation.

While the Commission considers that the siting functions intended for the LPZ, namely, a low density of residents and the feasibility of taking protective actions, have been accomplished by other regulations or can be accomplished by other guidance, the Commission continues to believe that a requirement that limits the radiological consequences over the course of the accident provides a useful evaluation of the plant's long-term capability to mitigate postulated accidents. For this reason, the Commission is retaining the requirement that the dose consequences be evaluated at the outer boundary of the LPZ over the course of the postulated accident and that these not be in excess of 25 rem TEDE.

D. Physical Characteristics of the Site It has been required that physical characteristics of the site, such as the geology, seismology, hydrology, meteorology characteristics be considered in the design and construction of any plant proposed to be located there. The final rule requires that these characteristics be evaluated and that site parameters, such as design basis flood conditions or tornado wind loadings be established for use in evaluating any plant to be located on that site in order to ensure that the occurrence of such physical phenomena would pose no undue hazard.

E. Nearby Transportation Routes, Industrial and Military Facilities As for natural phenomena, it has been a long-standing NRC staff practice to review man-related activities in the site vicinity to provide assurance that potential hazards associated with such facilities or transportation routes will pose no undue risk to any plant proposed to be located at the site. The final rule codifies this practice.

F. Adequacy of Security Plans The rule requires that the characteristics of the site be such that adequate security plans and measures for the plant could be developed. The Commission envisions that this will entail a small secure area considerably smaller than that envisioned for the exclusion area.

G. Adequacy of Emergency Plans The rule also requires that the site characteristics be such that adequate plans to carry out protective measures for members of the public in the event of emergency could be developed.

H. Siting Away From Densely Populated Centers

Population density considerations beyond the exclusion area have been required since issuance of Part 100 in 1962. The current rule requires a "low population zone" (LPZ) beyond the immediate exclusion area. The LPZ boundary must be of such a size that an individual located at its outer boundary must not receive a dose in excess of the values given in Part 100 over the course of the accident. While numerical values of population or population density are not specified for this region, the regulation also requires that the nearest boundary of a densely populated center of about 25,000 or more persons be located no closer than one and one-third times the LPZ outer boundary. Part 100 has no population criteria other than the size of the LPZ and the proximity of the nearest population center, but notes that "where very large cities are involved, a greater distance may be necessary."

Whereas the exclusion area size is based upon limitation of individual risk, population density requirements serve to set societal risk limitations and reflect consideration of accidents beyond the design basis, or severe accidents. Such accidents were clearly a consideration in the original issuance of Part 100, since the Statement of Considerations (27 FR 3509; April 12, 1962) noted that:

> "Further, since accidents of greater potential hazard than those commonly postulated as representing an upper limit are conceivable, although highly improbable, it was considered desirable to provide for protection against excessive exposure doses to people in large centers, where effective protective measures might not be feasible... Hence, the population center distance was added as a site requirement."

Limitation of population density beyond the exclusion area has the following benefits:

(a) it facilitates emergency preparedness and planning; and

(b) it reduces potential doses to large numbers of people and reduces property damage in the event of severe accidents.

Although the Commission's Safety Goal policy provides guidance on individual risk limitations in the form of the Quantitative Health Objectives (QHO), it provides no guidance with regard to societal risk limitations and therefore cannot be used to ascertain whether a particular population density would meet the Safety Goal.

However, results of severe accident risk studies, particularly those obtained from NUREG-1150, can provide useful insights for considering potential criteria for population density. Severe accidents having the highest consequences are those where core-melt together with early bypass of or containment failure occurs. Such an event would likely lead to a "large release" (without defining this precisely). Based upon NUREG-1150, the probability of a core-melt accident together with early containment failure or bypass for some current generation LWRs is estimated to be between $10^{-5}$ and $10^{-6}$ per reactor year. For future plants, this value is expected to be less than $10^{-6}$ per reactor year.

If a reactor was located nearer to a large city than current NRC practice permitted, the likelihood of exposing a large number of people to significant releases of radioactive material would be about the same as the probability of a core-melt and early containment failure, that is, less than $10^{-6}$ per reactor year for future reactor designs. It is worth noting that events having the very low likelihood of about $10^{-6}$ per reactor year or lower have been regarded in past licensing actions to be "incredible", and as such, have not been required to be incorporated into the design basis of the plant. Hence, based solely upon accident likelihood, it might be argued that siting a reactor nearer to a large city than current NRC practice would pose no undue risk.

If, however, a reactor were sited away from large cities, the likelihood of the city being affected would be reduced because of two factors. First, because the wind is expected to blow in all directions with roughly the same frequency, the likelihood that radioactive material would actually be carried towards the city is reduced significantly because it is likely that the wind will blow in a direction away from the city. Second, the radiological dose consequences would also be reduced with distance because the radioactive material becomes increasingly diluted by the atmosphere and the inventory becomes depleted due to the natural processes of fallout and rainout before reaching the city. Analyses indicate that if a reactor were located at distances ranging from 10 to about 20 miles away from a city, depending upon its size, the likelihood of exposure of large numbers of people within the city would be reduced by factors of ten to one hundred or more compared with locating a reactor very close to a city.

In summary, next-generation reactors are expected to have risk characteristics sufficiently low that the safety of the public is reasonably assured by the reactor and plant design and operation itself, resulting in a very low likelihood of occurrence of a severe accident. Such a plant can satisfy the QHOs of the Safety Goal with a very small exclusion area distance (as low as 0.1 miles). The consequences of design basis accidents, analyzed using revised source terms and with a realistic evaluation of engineered safety features, are likely to be found acceptable at distances of 0.25 miles or less. With regard to population density beyond the exclusion area, siting a reactor closer to a densely populated city than is current NRC practice would pose a very low risk to the populace.

Nevertheless, the Commission concludes that defense-in-depth considerations and the additional enhancement in safety to be gained by siting reactors away from densely populated centers should be maintained.

The Commission is incorporating a two-tier approach with regard to population density and reactor sites. The rule requires that reactor sites be located away from very densely populated centers, and that areas of low population density are, generally, preferred. The Commission believes that a site not falling within these two categories, although not preferred, can be found acceptable under certain conditions.

The Commission is not establishing specific numerical criteria for evaluation of population density in siting future reactor facilities because the acceptability of a specific site from the standpoint of population density must be considered in the overall context of safety and environmental considerations. The Commission's intent is to assure that a site that has significant safety, environmental or economic advantages is not rejected solely because it has a higher population density than other available sites. Population density is but one factor that must be balanced against the other advantages and disadvantages of a particular site in determining the site's acceptability. Thus, it must be recognized that sites with higher population density, so long as they are located away from very densely populated centers, can be approved by the Commission if they present advantages in terms of other considerations applicable to the evaluation of proposed sites.

On April 28, 1977, Free Environment, Inc. et. al., filed a petition for rulemaking (PRM-50-20) requesting, among other things, that "the central Iowa nuclear project and other reactors be sited at least 40 miles from major population centers." The petitioner also stated that "locating reactors in sparsely-populated areas ...has been endorsed in non-binding NRC guidelines for reactor siting." The petitioner did not specify what constituted a major population center. The only NRC guidelines concerning population density in regard to reactor siting are in Regulatory Guide 4.7, issued in 1974, and revised in 1975, prior to the date of the petition. This guide states population density values of 500 persons per square mile out to a distance of 30 miles from the reactor, not 40 miles.

Regulatory Guide 4.7 does provide effective separation from population centers of various sizes. Under this guide, a population center of about 25,000 or more residents should be no closer than 4 miles (6.4 km) from a reactor because a density of 500 persons per square mile within this distance would yield a total population of about 25,000 persons. Similarly, a city of 100,000 or more residents should be no closer than about 10 miles (16 km); a city of 500,000 or more persons should be no closer than about 20 miles (32 km), and a city of 1,000,000 or more persons should be no closer than about 30 miles (50 km) from the reactor.

The Commission has examined these guidelines with regard to the Safety Goal. The Safety Goal quantitative health objective in regard to latent cancer fatality states that, within a distance of ten miles (16 km) from the reactor, the risk to the population of latent cancer fatality from nuclear power plant operation, including accidents, should not exceed one-tenth of one percent of the likelihood of latent cancer fatalities from all other causes. In addition to the risks of latent cancer fatalities, the Commission has also investigated the likelihood and extent of land contamination arising from the release of long-lived radioactive species, such as cesium-137, in the event of a severe reactor accident.

The results of these analyses indicate that the latent cancer fatality quantitative health objective noted above is met for current plant designs. From analysis done in support of this proposed change in regulation, the likelihood of permanent relocation of people located more than about 20 miles (32 km) from the reactor as a result of land contamination from a severe accident is very low. A revision of Regulatory Guide 4.7 which incorporated this finding that population density guidance beyond 20 miles was not needed in the evaluation of potential reactor sites was issued for comment at the time of the proposed rule. No comments were received on this aspect of the guide.

Therefore, the Commission concludes that the NRC staff guidance in Regulatory Guide 4.7 provide a means of locating reactors away from population centers, including "major" population centers, depending upon their size, that would limit societal consequences significantly, in the event of a severe accident. The Commission finds that granting of the petitioner's request to specify population criteria out to 40 miles would not substantially reduce the risks to the public. As noted, the Commission also believes that a higher population density site could be found to be acceptable, compared to a lower population density site, provided there were safety, environmental or economic advantages to the higher population site. Granting o' the petitioner's request would neglect this possibility and would make population density the sole criterion of site acceptability. For these reasons, the Commission has decided not to adopt the proposal by Free Environment, Incorporated.

The Commission also notes that future population growth around a nuclear power plant site, as in other areas of the region, is expected but cannot be predicted with great accuracy, particularly in the long-term. Population growth in the site vicinity will be periodically factored into the emergency plan for the site, but since higher population density sites are not unacceptable, per se, the Commission does not intend to consider license conditior. . restrictions upon an operating reactor solely upon the basis that the ation density around it may reach or exceed levels that were not expected e time of site approval. Finally, the Commission wishes to emphasize t population considerations as well as other siting requirements apply only for the initial siting for new plants and will not be used in evaluating applications for the renewal of existing nuclear power plant licenses.

Change to 10 CFR Part 50

The change to 10 CFR Part 50 relocates from 10 CFR Part 100 the dose requirements for each applicant at specified distances. Because these requirements affect reactor design rather than siting, they are more appropriately located in 10 CFR Part 50.

These requirements apply to future applicants for a construction permit, design certification, or an operating license. The Commission will consider after further experience in the review of certified designs whether more specific requirements need to be developed regarding revised accident source terms and severe accident insights.

*B. Seismic and E. thquake Engineering Criteria.*

The following major changes to Appendix A, "Seismic and Geologic Siting Criteria for Nuclear Power Plants to 10 CFR Part 100, are associated with the seismic and earthquake engine ing criteria rule making. These changes reflect new information and research results, and incorporate the intentions of this regulatory action as defined in Section III of this rule. Much of the following discussion remains unchanged from that issued for public comment (59 FR 52255) because there were no comments which necessitated a major change to the regulations and supporting documentation.

1. Separate Siting from Design.

Criteria not associated with site suitability or establishment of the Safe Shutdown Earthquake Ground Motion (SSE) have been placed into 10 CFR Part 50. This action is consistent with the location of other design requirements in 10 CFR Part 50. Because the revised criteria presented in the regulation will not be applied to existing plants, the licensing basis for existing nuclear power plants must remain part of the regulations. The criteria on seismic and geologic siting would be designated as a new Section 100.23 to Subpart B in 10 CFR Part 100. Criteria on earthquake engineering would be designated as a new Appendix S, "Earthquake Engineering Criteria for Nuclear Power Plants," to 10 CFR Part 50.

2. Remove Detailed Guidance from the Regulation.

Appendix A to 10 CFR Part 100 contains both requirements and guidance on how to satisfy the requirements. For example, Section IV, "Required Investigations," of Appendix A, states that investigations are required for vibratory ground motion, surface faulting, and seismically induced floods and water wave[illegible] Appendix A then provides detailed guidance on what constitutes an accep[illegible] investigation. A similar situation exists in Section V, "Seismic [illegible]eologic Design Bases," of Appendix A.

Geo[illegible]ce assessments require considerable latitude in judgment. This latitude in judgment is needed because of limitations in data and the state-of-the-art of geologic and seismic analyses and because of the rapid evolution taking place in the geosciences in terms of accumulating knowledge and in modifying concepts. This need appears to have been recognized when the existing regulation was developed. The existing regulation states that it is based on limited geophysical and geological information and will be revised as necessary when more complete information becomes available.

However, having geoscience assessments detailed and cast in a regulation has created difficulty for applicants and the staff in terms of inhibiting the use of needed latitude in judgment. Also, it has inhibited flexibility in applying basic principles to new situations and the use of evolving methods of analyses (for instance, probabilistic) in the licensing process.

The final regulation is streamlined, becoming a new section in Subpart B to 10 CFR Part 100 rather than a new appendix to Part 100. Also, the level of detail presented in the final regulation is reduced considerably. Thus, the final regulation contains: a) required definitions, b) a requirement to determine the geological, seismological, and engineering characteristics of the proposed site, and c) requirements to determine the Safe Shutdown Earthquake Ground Motion (SSE), to determine the potential for surface deformation, and to determine the design bases for seismically induced floods and water waves. The guidance documents describe how to carry out these required determinations. The key elements of the approach to determine the SSE are presented in the following section. The elements are the guidance that is described in Regulatory Guide1.165, "Identification and Characterization of Seismic Sources and Determination of Safe Shutdown Earthquake Ground Motions."

3. Uncertainties and Probabilistic Methods

The existing approach for determining a Safe Shutdown Earthquake Ground Motion (SSE) for a nuclear reactor site, embodied in Appendix A to 10 CFR Part 100, relies on a "deterministic" approach. Using this deterministic approach, an applicant develops a single set of earthquake sources, develops for each source a postulated earthquake to be used as the source of ground motion that can affect the site, locates the postulated earthquake according to prescribed rules, and then calculates ground motions at the site.

Although this approach has worked reasonably well for the past two decades, in the sense that SSEs for plants sited with this approach are judged to be suitably conservative, the approach has not explicitly recognized uncertainties in geosciences parameters. Because of uncertainties about earthquake phenomena (especially in the eastern United States), there have often been differences of opinion and differing interpretations among experts as to the largest earthquakes to be considered and ground-motion models to be used, thus often making the licensing process relatively unstable.

Over the past decade, analysis methods for incorporating these different interpretations have been developed and used. These "probabilistic" methods have been designed to allow explicit incorporation of different models for zonation, earthquake size, ground motion, and other parameters. The advantage of using these probabilistic methods is their ability to not only incorporate different models and different data sets, but also to weight them using judgments as to the validity of the different models and data sets, and thereby providing an explicit expression for the uncertainty in the ground motion estimates and a means of assessing sensitivity to various input parameters. Another advantage of the probabilistic method is the target exceedance probability is set by examining the design bases of more recently licensed nuclear power plants.

The final regulation explicitly recognizes that there are inherent uncertainties in establishing the seismic and geologic design parameters and allows for the option of using a probabilistic seismic hazard methodology capable of propagating uncertainties as a means to address these uncertainties. The rule further recognizes that the nature of uncertainty and the appropriate approach to account for it depend greatly on the tectonic regime and parameters, such as, the knowledge of seismic sources, the existence of historical and recorded data, and the understanding of tectonics. Therefore, methods other than the probabilistic methods, such as sensitivity analyses, may be adequate for some sites to account for uncertainties.

Methods acceptable to the NRC staff for implementing the regulation are described in Regulatory Guide 1.165, "Identification and Characterization of Seismic Sources and Determination of Safe Shutdown Earthquake Ground Motion." The key elements of this approach are:

- Conduct site-specific and regional geoscience investigations,
- Target exceedance probability is set by examining the design bases of more recently licensed nuclear power plants,
- Conduct probabilistic seismic hazard analysis and determine ground motion level corresponding to the target exceedance probability
- Determine if information from the regional and site geoscience investigations change probabilistic results,
- Determine site-specific spectral shape and scale this shape to the ground motion level determined above,
- NRC staff review using all available data including insights and information from previous licensing experience, and
- Update the data base and reassess probabilistic methods at least every ten years.

Thus, the approach requires thorough regional and site-specific geoscience investigations. Results of the regional and site-specific investigations must be considered in applications of the probabilistic method. The current probabilistic methods, the NRC sponsored study conducted by Lawrence Livermore National Laboratory (LLNL) or the Electric Power Research Institute (EPRI) seismic hazard study, are regional studies without detailed information on any specific location. The regional and site-specific investigations provide detailed information to update the database of the hazard methodology as necessary.

It is also necessary to incorporate local site geological factors such as structural geology, stratigraphy, and topography and to account for site-specific geotechnical properties in establishing the design basis ground motion. In order to incorporate local site factors and advances in ground motion attenuation models, ground motion characteristics are determined using the procedures outlined in Standard Review Plan Section 2.5.2, "Vibratory Ground M[illegible]." Revision 3.

Th[illegible] staff's review approach to evaluate ground motion estimates is describe[illegible] SRP Section 2.5.2, Revision 3. This review takes into account the information base developed in licensing more than 100 plants. Although the basic premise in establishing the target exceedance probability is that the current design levels are adequate, a staff review further assures that there is consistency with previous licensing decisions and that the scientific bases for decisions are clearly understood. This review approach will also assess the fairly complex regional probabilistic modeling, which incorporates multiple hypotheses and a multitude of parameters. Furthermore, the NRC staff's Safety Evaluation Report should provide a clear basis for the staff's decisions and facilitate communication with nonexperts.

4. Safe Shutdown Earthquake.

The existing regulation (10 CFR Part 100, Appendix A, Section V(a)(1)(iv)) states "The maximum vibratory accelerations of the Safe Shutdown Earthquake at each of the various foundation locations of the nuclear power plant structures at a given site shall be determined ..." The location of the seismic input motion control point as stated in the existing regulation has led to confrontations with many applicants that believe this stipulation is inconsistent with good engineering fundamentals.

The final regulation moves the location of the seismic input motion control point from the foundation-level to the free-field at the free ground surface. The 1975 version of the Standard Review Plan placed the control motion in the free-field. The final regulation is also consistent with the resolution of Unresolved Safety Issue (USI) A-40, "Seismic Design Criteria" (August 1989), that resulted in the revision of Standard Review Plan Sections 2.5.2, 3.7.1, 3.7.2, and 3.7.3. The final regulation also requires that the horizontal component of the Safe Shutdown Earthquake Ground Motion in the free-field at the foundation level of the structures must be an appropriate response spectrum considering the site geotechnical properties, with a peak ground acceleration of at least 0.1g.

5. Value of the Operating Basis Earthquake Ground Motion (OBE) and Required OBE Analyses.

The existing regulation (10 CFR Part 100, Appendix A, Section V(a)(2)) states that the maximum vibratory ground motion of the OBE is at least one half the maximum vibratory ground motion of the Safe Shutdown Earthquake ground motion. Also, the existing regulation (10 CFR Part 100, Appendix A, Section VI(a)(2)) states that the engineering method used to insure that structures, systems, and components are capable of withstanding the effects of the OBE shall involve the use of either a suitable dynamic analysis or a suitable qualification test. In some cases, for instance piping, these multi-facets of the OBE in the existing regulation made it possible for the OBE to have more design significance than the SSE. A decoupling of the OBE and SSE has been suggested in several documents. For instance, the NRC staff, SECY-79-300, suggested that a compromise is required between design for a broad spectrum of unlikely events and optimum design for normal operation. Design for a single limiting event (the SSE) and inspection and evaluation for earthquakes in excess of some specified limit (the OBE), when and if they occur, makes the most sound regulatory approach. NUREG-1061, "Report of the U.S. Nuclear Regulatory Commission Piping Review Committee," Vol.5, April 1985, (Table 10.1) ranked a decoupling of the OBE and SSE as third out of six high priority changes. In SECY-90-016, "Evolutionary Light Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements," the NRC staff states that it agrees that the OBE should not control the design of safety systems. Furthermore, the final safety evaluation reports related to the certification of the System 80+ and the Advanced Boiling Water Reactor design (NUREG-1462 and NUREG-1503, respectively) have already adopted the single earthquake design philosophy.

Activities equivalent to OBE-SSE decoupling are also being done in foreign countries. For instance, in Germany their new design standard requires only one design basis earthquake (equivalent to the SSE). They require an inspection-level earthquake (for shutdown) of 0.4 SSE. This level was set so that the vibratory ground motion should not induce stresses exceeding the allowable stress limits originally required for the OBE design.

The final regulation allows the value of the OBE to be set at (i) one-third or less of the SSE, where OBE requirements are satisfied without an explicit response or design analyses being performed, or (ii) a value greater than one-third of the SSE, where analysis and design are required. There are two issues the applicant should consider in selecting the value of the OBE: first, plant shutdown is required if vibratory ground motion exceeding that of the OBE occurs (discussed below in Item 6, Required Plant Shutdown), and second, the amount of analyses associated with the OBE. An applicant may determine that at one-third of the SSE level, the probability of exceeding the OBE vibratory ground motion is too high, and the cost associated with plant shutdown for inspections and testing of equipment and structures prior to restarting the plant is unacceptable. Therefore, the applicant may voluntarily select an OBE value at some higher fraction of the SSE to avoid plant shutdowns. However, if an applicant selects an OBE value at a fraction of the SSE higher than one-third, a suitable analysis shall be performed to demonstrate that the requirements associated with the OBE are satisfied. The design shall take into account soil-structure interaction effects and the expected duration of the vibratory ground motion. The requirement associated with the OBE is that all structures, systems, and components of the nuclear power plant necessary for continued operation without undue risk to the health and safety of the public shall remain functional and within applicable stress, strain and deformation limits when subjected to the effects of the OBE in combination with normal operating loads.

As stated above, it is determined that if an OBE of one-third or less of the SSE is used, the requirements of the OBE can be satisfied without the applicant performing any explicit response analyses. In this case, the OBE serves the function of an inspection and shutdown earthquake. Some minimal design checks and the applicability of this position to seismic base isolation of buildings are discussed below. There is high confidence that, at this ground-motion level with other postulated concurrent loads, most critical structures, systems, and components will not exceed currently used design limits. This is ensured, in part, because PRA insights will be used to support a margins-type assessment of seismic events. A PRA-based seismic margins analysis will consider sequence-level High Confidence, Low Probability of Failures (HCLPFs) and fragilities for all sequences leading to core damage or containment failures up to approximately one and two-thirds the ground motion acceleration of the design basis SSE (Reference: Item II.N, Site-Specific Probabilistic Risk Assessment and Analysis of External Events, memorandum from Samuel J. Chilk to James M. Taylor, Subject: SECY-93-087 - Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advance Light-Water Reactor (ALWR) Designs, dated July 21, 1993).

There are situations associated with current analyses where only the OBE is associated with the design requirements, for example, the ultimate heat sink (see Regulatory Guide 1.27, "Ultimate Heat Sink for Nuclear Power Plants"). In these situations, a value expressed as a fraction of the SSE response would be used in the analyses. Section VII of this final rule identifies existing guides that would be revised technically to maintain the existing design philosophy.

In SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advance Light-Water Reactor (ALWR) Designs," the NRC staff requested Commission approval on 42 technical and policy issues pertaining to either evolutionary LWRs, passive LWRs, or both. The issue pertaining to the elimination of the OBE is designated I.M. The NRC staff identified actions necessary for the design of structures, systems, and components when the OBE design requirement is eliminated. The staff clarified that guidelines should be maintained to ensure the functionality of components, equipment, and their supports. In addition, the staff clarified how certain design requirements are to be considered for buildings and structures that are currently designed for the OBE, but not the SSE. Also, the NRC staff has evaluated the effect on safety of eliminating the OBE from the design load combinations for selected structures, systems, and components and has developed proposed criteria for an analysis using only the SSE. Commission approval is documented in the Chilk to Taylor memorandum dated July 21, 1993, cited above.

More than one earthquake response analysis for a seismic base isolated nuclear power plant design may be necessary to ensure adequate performance at all earthquake levels. Decisions pertaining to the response analyses associated with base isolated facilities will be handled on a case by case basis.

6. Required Plant Shutdown.

The current regulation (Section V(a)(2)) states that if vibratory ground motion exceeding that of the OBE occurs, shutdown of the nuclear power plant will be required. The supplementary information to the final regulation (published November 13, 1973; 38 FR 31279, Item 6e) includes the following statement: "A footnote has been added to §50.36(c)(2) of 10 CFR Part 50 to assure that each power plant is aware of the limiting condition of operation which is imposed under Section V(2) of Appendix A to 10 CFR Part 100. This limitation requires that if vibratory ground motion exceeding that of the OBE occurs, shutdown of the nuclear power plant will be required. Prior to resuming operations, the licensee will be required to demonstrate to the Commission that no functional damage has occurred to those features necessary for continued operation without undue risk to the health and safety of the public." At that time, it was the intention of the Commission to treat the OBE as a limiting condition of operation. From the statement in the Supplementary Information, the Commission directed applicants to specifically review 10 CFR Part 100 to be aware of this intention in complying with the requirements of 10 CFR 50.36. Thus, the requirement to shut down if an OBE occurs was expected to be implemented by being included among the technical specifications submitted by applicants after the adoption of Appendix A. In fact, applicants did not include OBE shutdown requirements in their technical specifications.

The final regulation treats plant shutdown associated with vibratory ground motion exceeding the OBE or significant plant damage as a condition in every operating license. A new §50.54(ff) is added to the regulations to require a process leading to plant shutdown for licensees of nuclear power plants that comply with the earthquake engineering criteria in Paragraph IV(a)(3) of Appendix S, "Earthquake Engineering Criteria for Nuclear Power Plants," to 10 CFR Part 50. Immediate shutdown could be required until it is determined that structures, systems, and components needed for safe shutdown are still functional.

Regulatory Guide 1.166, "Pre-Earthquake Planning and Immediate Nuclear Power Plant Operator Post-Earthquake Actions," provides guidance acceptable to the NRC staff for determining whether or not vibratory ground motion exceeding the OBE ground motion or significant plant damage had occurred and the timing of nuclear power plant shutdown. The guidance is based on criteria developed by the Electric Power Research Institute (EPRI). The decision to shut down the plant should be made by the licensee within eight hours after the earthquake. The data from the seismic instrumentation, coupled with information obtained from a plant walk down, are used to make the determination of when the plant should be shut down, if it has not already been shut down by operational perturbations resulting from the seismic event. The guidance in Regulatory Guide 1.166 is based on two assumptions, first, that the nuclear power plant has operable seismic instrumentation, including the equipment and software required to process the data within four hours after an earthquake, and second, that the operator walk down inspections can be performed in approximately four to eight hours depending on the number of personnel conducting the inspection. The regulation also includes a provision that requires the licensee to consult with the Commission and to propose a plan for the timely, safe shutdown of the nuclear power plant if systems, structures, or components necessary for a safe shutdown or to maintain a safe shutdown are not available. (This unavailability may be due to earthquake related damage.)

Regulatory Guide 1.167, "Restart of a Nuclear Power Plant Shut Down by a Seismic Event," provides guidelines that are acceptable to the NRC staff for performing inspections and tests of nuclear power plant equipment and structures prior to plant restart. This guidance is also based on EPRI reports. Prior to resuming operations, the licensee must demonstrate to the Commission that no functional damage has occurred to those features necessary for continued operation without undue risk to the health and safety of the public. The results of post-shutdown inspections, operability checks, and surveillance tests must be documented in written reports and submitted to the Director, Office of Nuclear Reactor Regulation. The licensee shall not resume operation until authorized to do so by the Director, Office of Nuclear Reactor Regulation.

7. Clarify interpretations.

Section 100.23 to 10 CFR Part 100 resolves questions of interpretation. As an example, definitions and required investigations stated in the final regulat[illegible] not contain the phrases in APpendix A to Part 100 that were more applica[illegible] only the western part of the United States.

Th[illegible]itutional definition for "safety-related structures, systems, and compo[illegible]s" is drawn from Appendix A to Part 100 under III(c) and VI(a). With the relocation of the earthquake engineering criteria to Appendix S to Part 50 and the relocation and modification to dose guidelines in §50.34(a)(1), the definition of safety-related structures, systems, and components is included in Part 50 definitions with references to both the Part 100 and Part 50 dose guidelines.

## VI. Related Regulatory Guides and Standard Review Plan Sections

The NRC is developing the following regulatory guides and standard review plan sections to provide prospective licensees with the necessary guidance for implementing the final regulation. The notice of availability for these materials will be published in a later issue of the Federal Register.

1. Regulatory Guide 1.165, "Identification and Characterization of Seismic Sources and Determination of Shutdown Earthquake Ground Motions." The guide provides general guidance and recommendations, describes acceptable procedures and provides a list of references that present acceptable methodologies to identify and characterize capable tectonic sources and seismogenic sources. Section V.B.3 of this rule describes the key elements.
2. Regulatory Guide 1.12, "Nuclear Power Plant Instrumentation for Earthquakes," Revision 2. The guide describes seismic instrumentation type and location, operability, characteristics, installation, actuation, and maintenance that are acceptable to the NRC staff.
3. Regulatory Guide 1.166, "Pre-Earthquake Planning and Immediate Nuclear Power Plant Operator Post-Earthquake Actions." The guide provides guidelines that are acceptable to the NRC staff for a timely evaluation of the recorded seismic instrumentation data and to determine whether or not plant shutdown is required.
4. Regulatory Guide 1.167, "Restart of a Nuclear Power Plant Shut Down by a Seismic Event." The guide provides guidelines that are acceptable to the NRC staff for performing inspections and tests of nuclear power plant equipment and structures prior to restart of a plant that has been shut down because of a seismic event.
5. Standard Review Plan Section 2.5.1, Revision 3, "Basic Geologic and Seismic Information." This SRP Section describes procedures to assess the

adequacy of the geologic and seismic information cited in support of the applicant's conclusions concerning the suitability of the plant site.

6. Standard Review Plan Section 2.5.2, Revision 3 "Vibratory Ground Motion." This SRP Section describes procedures to assess the ground motion potential of seismic sources at the site and to assess the adequacy of the SSE.

7. Standard Review Plan Section 2.5.3, Revision 3, "Surface Faulting." This SRP Section describes procedures to assess the adequacy of the applicant's submittal related to the existence of a potential for surface faulting affecting the site.

8. [illegible]latory Guide 4.7, "General Site Suitability Criteria for Nuclear Power P[illegible]" Revision 2. This guide discusses the major site characte[illegible]cs related to public health and safety and environmental issues that the NRC staff considers in determining the suitability of sites.

## VII. Future Regulatory Action

Several existing regulatory guides will be revised to incorporate editorial changes or maintain the existing design or analysis philosophy. These guides will be issued as final guides without public comment subsequent to the publication of the final regulations.

The following regulatory guides will be revised to incorporate editorial changes, for example to reference new sections to Part 100 or Appendix S to Part 50. No technical changes will be made in these regulatory guides.

1. 1.57, "Design Limits and Loading Combinations for Metal Primary Reactor Containment System Components."
2. 1.59, "Design Basis Floods for Nuclear Power Plants."
3. 1.60, "Design Response Spectra for Seismic Design of Nuclear Power Plants."
4. 1.83, "Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes."
5. 1.92, "Combining Modal Responses and Spatial Components in Seismic Response Analysis."
6. 1.102, "Flood Protection for Nuclear Power Plants."
7. 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes."
8. 1.122, "Development of Floor Design Response Spectra for Seismic Design of Floor—Supported Equipment or Components."

The following regulatory guides will be revised to update the design or analysis philosophy, for example, to change OBE to a fraction of the SSE:

1. 1.27, "Ultimate Heat Sink for Nuclear Power Plants."
2. 1.100, "Seismic Qualification of Electric and Mechanical Equipment for Nuclear Power Plants."
3. 1.124, "Service Limits and Loading Combinations for Class 1 Linear—Type Component Supports."
4. 1.130, "Service Limits and Loading Combinations for Class 1 Plate—and—Shell—Type Component Supports."

5. 1.132, "Site Investigations for Foundations of Nuclear Power Plants."
6. 1.138, "Laboratory Investigations of Soils for Engineering Analysis and Design of Nuclear Power Plants."
7. 1.142, "Safety-Related Concrete Structures for Nuclear Power Plants (Other than Reactor Vessels and Containments)."
8. 1.143, "Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants."

M nd conforming changes to other Regulatory Guides and standard review ections as a result of changes in the nonseismic criteria are also pla. If substantive changes are made during the revisions, the applicable guides will be issued for public comment as draft guides.

## VIII. Referenced Documents

An interested person may examine or obtain copies of the documents referenced in this rule as set out below.

Copies of NUREG-0625, NUREG-1061, NUREG-1150, NUREG-1451, NUREG-1462, NUREG-1503, and NUREG/CR-2239 may be purchased from the Superintendent of Documents, U.S. Government Printing Office, Mail Stop SSOP, Washington, DC 20402-9328. Copies are also available from the National Technical Information Service, 5285 Port Royal Road, Springfield, VA 22161. A copy is also available for inspection and copying for a fee in the NRC Public Document Room, 2120 L Street, NW. (Lower Level), Washington, DC.

Copies of issued regulatory guides may be purchased from the Government Printing Office (GPO) at the current GPO price. Information on current GPO prices may be obtained by contacting the Superintendent of Documents, U.S. Government Printing Office, Mail Stop SSOP, Washington, DC 20402-9328. Issued guides may also be purchased from the National Technical Information Service on a standing order basis. Details on this service may be obtained by writing NTIS, 5826 Port Royal Road, Springfield, VA 22161.

SECY 79-300, SECY 90-016, SECY 93-087, and WASH-1400 are available for inspection and copying for a fee at the NRC Public Document Room, 2120 L Street, NW. (Lower Level), Washington, DC.

## IX. Summary of Comments on the Proposed Regulations.

### *A. Reactor Siting Criteria (Nonseismic).*

Eight organizations or individuals commented on the nonseismic aspects of the second proposed revision. The first proposed revision issued for comment in October 1992 elicited strong comments in regard to proposed numerical values of population density and a minimum distance to the exclusion area boundary (EAB) in the rule. The second proposed revision would delete these from the rule by providing guidance on population density in a Regulatory Guide and determining the distance to the EAB and LPZ by use of source term and dose calculations. The rule would contain basic site criteria, without any numerical values.

Several commentors representing the nuclear industry and international nuclear organizations stated that the second proposed revision was a significant improvement over the first proposed revision, while the only public interest group commented that the NRC had retreated from decoupling siting and design in response to the comments of foreign entities.

Most comments on the second proposed revision centered on the use of total effective dose equivalent (TEDE), the proposed single numerical dose acceptance criterion of 25 rem TEDE, the evaluation of the maximum dose in any two-hour period, and the question of whether an organ capping dose should be adopted.

Virtually all commentors supported the concept of TEDE and its use. However, there were differing views on the proposed numerical dose of 25 rem and the proposed use of the maximum two-hour period to evaluate the dose. Virtually all industry commenters felt that the proposed numerical value of 25 rem TEDE was too low and that it represented a "ratchet" since the use of the current dose criteria plus organ weighting factors would suggest a value of 34 rem TEDE. In addition, all industry commenters believed the "sliding" two-hour window for dose evaluation to be confusing, illogical and inappropriate. They favored a rule that was based upon a two hour period after the onset of fission product release, similar in concept to the existing rule. All industry commenters opposed the use of an organ capping dose. The only public interest group that commented did not object to the use of TEDE, favored the proposed dose value of 25 rem, and supported an organ capping dose.

*B. Seismic and Earthquake Engineering Criteria.*

Seven letters were received addressing either the regulations or both the regulations and the draft guidance documents identified in Section VI (except DG-4003). An additional five letters were received addressing only the guidance documents, for a total of twelve comment letters. A document, "Resolution of Public Comments on the Proposed Seismic and Earthquake Engineering Criteria for Nuclear Power Plants," is available explaining the NRC's disposition of the comments received on the regulations. A copy of this document has been placed in the NRC Public Document Room, 2120 L Street NW. (Lower Level), Washington, DC. Single copies are available from Dr. Andrew J. Murphy, Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission, Washington, DC 20555, telephone (301) 415-6010. A second document, "Resolution of Public Comments on Draft Regulatory Guides and Standard Review Plan Sections Pertaining to the Proposed Seismic and Earthquake Engineering Criteria for Nuclear Power Plants," will explain the NRC's disposition of the comments received on the guidance documents. The Federal Register notice announcing the avaliability of the guidance documents will also discuss how to obtain copies of the comment resolution document.

A summary of the major comments on the proposed regulations follows.

## Supplemental Information

### Section III, Genesis (Application)

The Department of Energy (Office of Civilian Radioactive Waste Management), requests an explicit statement whether or not Section 100.23 to Part 100 applies to the Mined Geologic Disposal System (MGDS) and a Monitored Retrievable Storage (MRS) facility. The NRC has noted in NUREG-1451, "Staff Technical Position on Investigations to Identify Fault Displacement Hazards and Seism Hazards at a Geologic Respository," that Appendix A to 10 CFR Part 100 does apply to a geologic repository. NUREG-1451 also notes that the contemp revisions to Part 100 would also not be applicable to a geologic reposito Section 72.102(b) requires that, for an MRS located west of the Rocky Mou n front or in areas of known potential seismic activity in the east, the seismicity be evaluated by the techniques of Appendix A to 10 CFR Part 100.

Although Appendix A to 10 CFR Part 100 is titled "Seismic and Geologic Siting Criteria for Nuclear Power Plants," it is also referenced in two other parts of the regulation. They are (1) Part 40, "Domestic Licensing of Source Material," Appendix A, "Criteria Relating to the Operation of Uranium Mills and the Disposition of Tailings or Waste Produced by the Extraction or Concentration of Source Material from Ores Processed Primarily for Their Source Material Content," Section I, Criterion 4(e), and (2) Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste," Paragraphs (a)(2), (b) and (f)(1) of §72.102.

The referenced applicability of Section 100.23 to other than power reactors, if considered appropriate by the NRC, would be a separate rulemaking. That rulemaking would clearly state the applicability of Section 100.23 to an MRS or other facility. In addition, NUREG-1451 will remain the NRC staff technical position on seismic siting issues pertaining to an MGDS until it is superceded through a rulemaking, revision of NUREG-1451, or other appropriate mechanism.

### Section V(B)(5), "Value of the Operating Basis Earthquake Ground Motion (OBE) and Required OBE Analysis."

One commentor, ABB Combustion Engineering Nuclear Systems, specifically stated that they agree with the NRC's proposal to not require explicit design analysis of the OBE if its peak acceleration is less than one-third of the Safe Shutdown Earthquake Ground Motion (SSE). The only negative comments, from C.C. Slagis Associates, stated that the proposed rule in the area of required OBE analysis is not sound, not technically justified, and not appropriate for the design of pressure-retaining components. The following are specific comments (limited to the design of pressure-retaining components to the ASME Boiler and Pressure Vessel Section III rules) that pertain to the supplemental information to the proposed regulations, item V(B)(5), "Value of the Operating Basis Earthquake Ground Motion (OBE) and Required OBE Analysis."

(1) Disagrees with the statement in SECY-79-300 that design for a single limiting event and inspection and evaluation for earthquakes in excess of some specified limit may be the most sound regulatory approach. It is not feasible to inspect for cyclic damage to all the pressure-retaining components.

Visually inspecting for permanent deformation, or leakage, or failed component supports is certainly not adequate to determine cyclic damage.

The NRC agrees. Postearthquake inspection and evaluation guidance is described in Regulatory Guide 1.167 (Draft was DG-1035), "Restart of a Nuclear Power Plant Shut Down by an Seismic Event." The guidance is not limited to visual inspections; it includes inspections, tests, and analyses including fatigue analysis.

(2) Disagrees with the NRC statement in SECY-090-016 that the OBE should not control design. There is a problem with the present requirements. Requiring design for five OBE events at ½ SSE is unrealistic for most (all?) sites and requires an excessive and unnecessary number of seismic supports. The solution is to properly define the OBE magnitude and the number of events expected during the life of the plantand to require design for that loading. OBE may or may not control the design. But you cannot assume, before you have the seismicity defined and before you have a component design, that OBE will not govern the design.

The NRC has concluded that design requirements based on an estimated OBE magnitude at the plant site and the number of events expected during the plant life will lead to low design values that will not control the design, thus resulting in unnecessary analyses.

(3) It is not technically justified to assume that Section III components will remain within applicable stress limits (Level B limits) at one-third the SSE. The Section III acceptance criteria for Level D (for an SSE) is completely different than that for Level B (for an OBE). The Level D criteria is based on surviving the extremely-low probability SSE load. Gross structural deformations are possible, and it is expected that the component will have to be replaced. Cyclic effects are not considered. The cyclic effects of the repeated earthquakes have to be considered in the design of the component to ensure pressure boundary integrity throughout the life, especially if the SSE can occur after the lower level earthquakes.

In SECY-93-087, Issue I.M, "Elimination of Operating-Basis Earthquake," the NRC recognizes that a designer of piping systems considers the effects of primary and secondary stresses and evaluates fatigue caused by repeated cycles of loading. Primary stresses are induced by the inertial effects of vibratory motion. The relative motion of anchor points induces secondary stresses. The repeating seismic stress cycles induce cyclic effects (fatigue). However, after reviewing these aspects, the NRC concludes that, for primary stresses, if the OBE is established at one-third the SSE, the SSE load combinations control the piping design when the earthquake contribution dominates the load combination. Therefore, the NRC concludes that eliminating the OBE piping stress load combination for primary stresses in piping systems will not significantly reduce existing safety margins.

Eliminating the OBE will, however, directly affect the current methods used to evaluate the adequacy of cyclic and secondary stress effects in the piping design. Eliminating the OBE from the load combination could cause uncertainty in evaluating the cyclic (fatigue) effects of earthquake-induced motions in piping systems and the relative motion effects of piping anchored to equipment and structures at various elevations because both of these effects are currently evaluated only for OBE loadings. Accordingly, to account for earthquake cycles in the fatigue analysis of piping systems, the staff proposes to develop guidelines for selecting a number of SSE cycles at a fraction of the peak amplitude of the SSE. These guidelines will provide a level of fatigue design for the piping equivalent to that currently provided in Standard Review Plan Section 3.9.2.

Positions pertaining to the elimination of the OBE were proposed in SECY-93-087. Commission approval is documented in a memorandum from Samuel J. Chilk to James M. Taylor, Subject: SECY-93-087 - Policy, Technical and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs, dated July 21, 1993.

(4) There is one major flaw in the "SSE only" design approach. The equipment designed for SSE is limited to the equipment necessary to assure the integrity of the reactor coolant pressure boundary, to shutdown the reactor, and to prevent or mitigate accident consequences. The equipment designed for SSE is only part of the equipment "necessary for continued operation without undue risk to the health and safety of the public." Hence, by this rule, it is possible that some equipment necessary for continued operation will not be designed for SSE or OBE effects.

The NRC does not agree that the design approach is flawed. It is not possible that some equipment necessary for continued safe operation will not be designed for SSE or OBE effects. General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena," of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50 requires that nuclear power plant structures, systems, and components important to safety be designed to withstand the effects of earthquakes without loss of capability to perform their safety functions. The criteria in Appendix S to 10 CFR Part 50 implement General Design Criterion 2 insofar as it requires structures, systems, and components important to safety to withstand the effects of earthquakes. Regulatory Guide 1.29, "Seismic Design Classification," describes a method acceptable to the NRC for identifying and classifying those features of light-water-cooled nuclear power plants that should be designed to withstand the effects of the SSE. Currently, components which are designed for OBE only include components such as waste holdup tanks. As noted in Section VII, Future Regulatory Actions, regulatory guides related to these components will be revised to provide alternative design requirements.

Section 100.23 to 10 CFR Part 100

The Nuclear Energy Institute (NEI) congratulated the NRC staff for carefully considering and responding to the voluminous and complex comments that were provided on the earlier proposed rulemaking package (57 FR 47802) and considered that the seismic portion of the proposed rulemaking package is nearing maturity and with the inclusion of industry's comments (which were principally on the guidance documents), has the potential to satisfy the objectives of predictable licensing and stable regulations.

Both NEI and Westinghouse Electric Corporation support the regulation format, that is, prescriptive guidance is located in regulatory guides or standard review plan sections not the regulation.

NEI and Westinghouse Electric Corporation support the removal of the requirement from the first proposed rulemaking (57 FR 47802) that both deterministic and probabilistic evaluations must be conducted to determine site suitability and seismic design requirements for the site. [Note: the commentors do not agree with the NRC staff's deterministic check of the seismic sources and parameters used in the LLNL and EPRI probabilistic seismic hazard analyses (Regulatory Guide 1.165, draft was DG-1032). Also, they do not support the NRC staff's deterministic check of the applicants submittal (SRP Section 2.5.2). These items are addressed in the document pertaining to comment resolution of the draft regulatory guides and standard review plan sections.]

NEI, Westinghouse Electric Corporation, and Yankee Atomic Electric Corporation recommend that the regulation should state that for existing sites east of the Rocky Mountain Front (east of approximately 105° west longitude), a 0.3g standardized design level is acceptable at these sites given confirmatory foundations evaluations [Regulatory Guide 1.132, but not the geologic, geophysical, seismological investigations in Regulatory Guide 1.165].

The NRC has determined that the use of a spectral shape anchored to 0.3g peak ground acceleration as a standardized design level would be appropriate for existing central and eastern U.S. sites based on the current state of knowledge. However, as new information becomes available it may not be appropriate for future licensing decisions. Pertinent information such as that described in Regulatory Guide 1.165 (Draft was DG-1032) is needed to make that assessment. Therefore, it is not appropriate to codify the request.

NEI recommended a rewording of Paragraph (a), Applicability. Although unlikely, an applicant for an operating license already holding a construction permit may elect to apply the amended methodology and criteria in Subpart B to Part 100.

The NRC will address this request on a case-by-case basis rather than through a generic change to the regulations. This situation pertains to a limited number of facilities in various stages of construction. Some of the issues that must be addressed by the applicant and NRC during the operating license review include differences between the design bases derived from the current and amended regulations (Appendix A to Part 100 and Section 100.23, respectvely), and earthquake engineering criteria such as, OBE design requirements and OBE shutdown requirements.

### Appendix S to 10 CFR Part 50

Support for the NRC position pertaining to the elimination of the Operating Basis Earthquake Ground Motion (OBE) response analyses has been documented in various NRC publications such as SECY-79-300, SECY-90-016, SECY-93-087, and NUREG-1061. The final safety evaluation reports related to the certification of the System 80+ and the Advanced Boiling Water Reactor design (NUREG-1462 and NUREG-1503, respectively) have already adopted the single earthquake design philosophy. In addition, similar activities are being done in foreign countries, for instance, Germany. (Additional discussion is provided in Section V(B)(5) of this rule).

The American Society of Civil Engineers (ASCE) recommended that the seismic design and engineering criteria of ASCE Standard 4, "Seismic Analysis of Safety-Related Nuclear Structures and Commentary on Standard for Seismic Analysis of Safety-Related Nuclear Structures," be incorporated by reference into Appendix S to 10 CFR Part 50.

The Commission has determined that new regulations will be more streamlined containing only basic requirements with guidance being provided in regulatory guides and, to some extent, in standard review plan sections. Both the NRC and industry have experienced difficulties in applying prescriptive regulations such as Appendix A to 10 CFR Part 100 because they inhibit the use of needed latitude in judgement. Therefore, it is common NRC practice not to reference publications such as ASCE Standard 4 (an analysis, not design standard) in its regulations. Rather, publications such as ASCE Standard 4 are cited in regulatory guides and standard review plan sections. ASCE Standard [illegible] cited in the 1989 revision of Standard Review Plan Sections 3.7.1, 3[illegible] and 3.7.3.

The [illegible]artment of Energy stated that the required consideration of aftershocks in Paragraph IV(B), Surface Deformation, is confusing and recommended that it be deleted.

The NRC agrees. The reference to aftershocks in Paragraph IV(b) has been deleted. Paragraphs VI(a), Safe Shutdown Earthquake, and VI(B)(3) of Appendix A to Part 100 contain the phrase "including aftershocks." The "including aftershocks" phrase was removed from the Safe Shutdown Earthquake Ground Motion requirements in the proposed regulation. The recommended change will make Paragraphs IV(a)(1), "Safe Shutdown Earthquake Ground Motion," and IV(b), "Surface Deformation, of Appendix S to 10 CFR Part 50 consistent.

## X. Finding of No Significant Environmental Impact: Availability

The Commission has determined under the National Environmental Policy Act of 1969, as amended, and the Commission's regulations in Subpart A of 10 CFR Part 51, that this regulation is not a major Federal action significantly affecting the quality of the human environment and therefore an environmental impact statement is not required.

The revisions associated with the reactor siting criteria in 10 CFR Part 100 and the relocation of the plant design requirements from 10 CFR Part 100 to 10 CFR Part 50 have been evaluated against the current requirements. The Commission has concluded that relocating the requirement for a dose calculation to Part 50 and adding more specific site criteria to Part 100 does not decrease the protection of the public health and safety over the current regulations. The amendments do not affect nonradiological plant effluents and have no other environmental impact.

The addition of §100.23 to 10 CFR Part 100, and the addition of Appendix S to 10 CFR Part 50, will not change the radiological environmental impact offsite. Onsite occupational radiation exposure associated with inspection and maintenance will not change. These activities are principally associated with base line inspections of structures, equipment, and piping, and with maintenance of seismic instrumentation. Base line inspections are needed to differentiate between pre-existing conditions at the nuclear power plant and earthquake related damage. The structures, equipment and piping selected for these inspections are those routinely examined by plant operators during normal plant walkdowns and inspections. Routine maintenance of seismic instrumentation ensures its operability during earthquakes. The location of the seismic instrumentation is similar to that in the existing nuclear power plants. The amendments do not affect nonradiological plant effluents and have no other environmental impact.

The environmental assessment and finding of no significant impact on which this determination is based are available for inspection at the NRC Public Document Room, 2120 L Street NW. (Lower Level), Washington, DC. Single copies of the environmental assessment and finding of no significant impact are available from Mr. Leonard Soffer, Office of the Executive Director for Operations, U.S. Nuclear Regulatory Commission, Washington, DC 20555, telephone (301) 415-1722, or Dr. Andrew J. Murphy, Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission, Washington, DC 20555, telephone (301) 415-6010.

### XI. Paperwork Reduction Act Statement

This final rule amends information collection requirements that are subject to the Paperwork Reduction Act of 1995 (44 U.S.C. 3501 et seq.). These requirements were approved by the Office of Management and Budget, approval numbers 3150-0011 and 3150-0093.

The public reporting burden for this collection of information is estimated to average 800,000 hours per response, including the time for reviewing instructions, searching existing data sources, gathering and maintaining the data needed, and completing and reviewing the collection of information. Send comments on any aspect of this collection of information, including suggestions for reducing the burden, to the Information and Records Management Branch (T-6 F33), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202, (3150-0011 and 3150-0093), Office of Management and Budget, Washington, DC 20503.

### Public Protection Notification

The NRC may not conduct or sponsor, and a person is not required to respond to, a collection of information unless it displays a currently valid OMB control number.

### XII. Regulatory Analysis

The Commission has prepared a regulatory analysis on this regulation. The analysis examines the costs and benefits of the alternatives considered by the Commission. Interested persons may examine a copy of the regulatory analysis at the NRC Public Document Room, 2120 L Street NW. (Lower Level), Washington, DC. Single copies of the analysis are available from Mr. Leonard Soffer, Office of the Executive Director for Operations, U.S. Nuclear Regulatory Commission, Washington, DC 20555, telephone (301) 415-1722, or Dr. Andrew J. Murphy, Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission, Washington, DC 20555, telephone (301) 415-6010.

### XIII. Regulatory Flexibility Certification

As required by the Regulatory Flexibility Act of 1980, 5 U.S.C. 605(b), the Commission certifies that this regulation does not have a significant economic impact on a substantial number of small entities. This regulation affects only the licensing and operation of nuclear power plants. The companies that own these plants do not fall within the definition of "small entities" set forth in the Regulatory Flexibility Act or the size standards established by the NRC (April 11, 1995; 60 FR 18344).

## XIV. Backfit Analysis

Th[illegible] has determined that the backfit rule, 10 CFR 50.109, does not apply to [illegible] regulation, and therefore, a backfit analysis is not required for this regulation because these amendments do not involve any provisions that would impose backfits as defined in 10 CFR 50.109(a)(1). The regulation would apply only to applicants for future nuclear power plant construction permits, preliminary design approval, final design approval, manufacturing licenses, early site reviews, operating licenses, and combined operating licenses.

### List of Subjects

10 CFR Part 50 — Antitrust, Classified information, Criminal penalty, Fire protection, Intergovernmental relations, Nuclear power plants and reactors, Radiation protection, Reactor siting criteria, Reporting and recordkeeping requirements.

10 CFR Part 52 — Administrative practice and procedure, Antitrust, Backfitting, Combined license, Early site permit, Emergency planning, Fees, Inspection, Limited work authorization, Nuclear power plants and reactors, Probabilistic risk assessment, Prototype, Reactor siting criteria, Redress of site, Reporting and recordkeeping requirements, Standard design, Standard design certification.

10 CFR Part 100 — Nuclear power plants and reactors, Reactor siting criteria.

For the reasons set out in the preamble and under the authority of the Atomic Energy Act of 1954, as amended, the Energy Reorganization Act of 1974, as amended, and 5 U.S.C. 553, the NRC is proposing to adopt the following amendments to 10 CFR Parts 50, 52 and 100.

### PART 50 — DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES

1. The authority citation for Part 50 continues to read as follows:

AUTHORITY: Secs. 102, 103, 104, 105, 161, 182, 183, 186, 189, 68 Stat. 936, 937, 938, 948, 953, 954, 955, 956, as amended, sec. 234, 83 Stat. 1244, as amended (42 U.S.C. 2132, 2133, 2134, 2135, 2201, 2232, 2233, 2236, 2239, 2282); secs. 201, as amended, 202, 206, 88 Stat. 1242, as amended, 1244, 1246, (42 U.S.C. 5841, 5842, 5846).

Section 50.7 also issued under Pub. L. 95-601, sec. 10, 92 Stat. 2951 as amended by Pub. L. 102-486, sec. 2902, 106 Stat. 3123, (42 U.S.C. 5851). Section 50.10 also issued under secs. 101, 185, 68 Stat. 936, 955 as amended (42 U.S.C. 2131, 2235), sec. 102, Pub. L. 91-190, 83 Stat. 853 (42 U.S.C. 4332). Sections 50.13, 50.54(dd) and 50.103 also issued under sec. 108, 68 Stat. 939, as amended (42 U.S.C. 2138). Sections 50.23, 50.35, 50.55, and 50.56 also issued under sec. 185, 68 Stat. 955 (42 U.S.C. 2235). Sections 50.33a, 50.55a and Appendix Q also issued under sec. 102, Pub. L. 91-190, 83 Stat. 853 (42 U.S.C. 4332). Sections 50.34 and 50.54 also issued under sec. 204, 88 Stat. 1245 (42 U.S.C. 5844). Sections 50.58, 50.91 and 50.92 also issued under Pub. L. 97-415, 96 Stat. 2073 (42 U.S.C. 2239). Section 50.78 also issued under sec. 122, 68 Stat. 939 (42 U.S.C. 2152). Sections 50.80 — 50.81 also issued under sec. 184, 68 Stat. 954, as amended (42 U.S.C. 2234). Appendix F also issued under sec. 187, 68 Stat. 955 (42 U.S.C. 2237).

* * * * *

2. Section §50.2 is revised by adding in alphabetical order the definitions for <u>Committed dose equivalent</u>, <u>Committed effective dose equivalent</u>, <u>Deep-dose equivalent</u>, <u>Exclusion area</u>, <u>Low population zone</u>, <u>Safety-related structures, systems, and components</u> and <u>Total effective dose equivalent</u> to read as follows:

§ 50.2 Definitions.

* * * * *

<u>Committed dose equivalent</u> means the dose equivalent to organs or tissues of reference that will be received from an intake of radioactive material by an individual during the 50-year period following the intake.

* * * * *

<u>Committed effective dose equivalent</u> is the sum of the products of the weighting factors applicable to each of the body organs or tissues that are irradiated and the committed dose equivalent to these organs or tissues.

* * * * *

<u>Deep-dose equivalent</u>, which applies to external whole-body exposure, is the dose equivalent at a tissue depth of 1 cm ($1000 mg/cm^2$).

* * * * *

<u>Exclusion area</u> means that area surrounding the reactor, in which the reactor licensee has the authority to determine all activities including exclusion or removal of personnel and property from the area. This area may be traversed by a highway, railroad, or waterway, provided these are not so close to the facility as to interfere with normal operations of the facility and provided appropriate and effective arrangements are made to control traffic on the highway, railroad, or waterway, in case of emergency, to protect the public health and safety. Residence within the exclusion area shall normally be prohibited. In any event, residents shall be subject to ready removal in case of necessity. Activities unrelated to operation of the reactor may be permitted in an exclusion area under appropriate limitations,

provided that no significant hazards to the public health and safety will result.

* * * * *

Low population zone means the area immediately surrounding the exclusion area which contains residents, the total number and density of which are such that there is a reasonable probability that appropriate protective measures could be taken in their behalf in the event of a serious accident. These guides do not specify a permissible population density or total population within this zone because the situation may vary from case to case. Whether a specific number of people can, for example, be evacuated from a specific [illegible] or instructed to take shelter, on a timely basis will depend on many fac[illegible]uch as location, number and size of highways, scope and extent of advanc[illegible]nning, and actual distribution of residents within the area.

* * * * *

Safety-related Structures Systems and Components means those structures, systems, and components that are relied on to remain functional during and following design basis (postulated) events to assure:

(1) The integrity of the reactor coolant pressure boundary,

(2) The capability to shutdown 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 applicable guideline exposures set forth in § 50.34(a)(1) or § 100.11 of this chapter.

* * * * *

Total effective dose equivalent (TEDE) means the sum of the deep-dose equivalent (for external exposures) and the committed effective dose equivalent (for internal exposures).

* * * * *

3. In §50.8, paragraph (b) is revised to read as follows:

**§ 50.8 Information collection requirements: OMB approval.**

* * * * *

(b) The approved information collection requirements contained in this part appear in §§50.30, 50.33, 50.33a, 50.34, 50.34a, 50.35, 50.36, 50.36a, 50.48, 50.49, 50.54, 50.55, 50.55a, 50.59, 50.60, 50.61, 50.63, 50.64, 50.65, 50.71, 50.72, 50.80, 50.82, 50.90, 50.91, and Appendices A, B, E, G, H, I, J, K, M, N, O, Q, R, and S.

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4. In §50.34, footnotes 6, 7, and 8 are redesignated as footnotes 8, 9 and 10 and paragraph (a)(1) is revised and paragraphs (a)(12), (b)(10), and (b)(11) are added to read as follows:

§ 50.34 Contents of applications; technical information.

(a) * * *

(1) Stationary power reactor applicants for a construction permit pursuant to this part, or a design certification or combined license pursuant to Part 52 of this chapter who apply on or after [INSERT EFFECTIVE DATE OF THE FINAL RULE], shall comply with paragraph (a)(1)(ii) of this section. All other applicants for a construction permit pursuant to this part or a design certification or combined license pursuant to Part 52 of this chapter, shall comply with paragraph (a)(1)(i) of this section.

(i) A description and safety assessment of the site on which the facility is to be located, with appropriate attention to features affecting facility design. Special attention should be directed to the site evaluation factors identified in Part 100 of this chapter. The assessment must contain an analysis and evaluation of the major structures, systems and components of the facility which bear significantly on the acceptability of the site under the site evaluation factors identified in Part 100 of this chapter, assuming that the facility will be operated at the ultimate power level which is contemplated by the applicant. With respect to operation at the projected initial power level, the applicant is required to submit information prescribed in paragraphs (a)(2) through (a)(8) of this section, as well as the information required by this paragraph, in support of the application for a construction permit, or a design approval.

(ii) A description and safety assessment of the site and a safety assessment of the facility. It is expected that reactors will reflect through their design , construction and operation an extremely low probability for accidents that could result in the release of significant quantities of radioactive fission products. The following power reactor design characteristics and proposed operation will be taken into consideration by the Commission:

(A) Intended use of the reactor including the proposed maximum power level and the nature and inventory of contained radioactive materials;

(B) The extent to which generally accepted engineering standards are applied to the design of the reactor;

(C) The extent to which the reactor incorporates unique, unusual or enhanced safety features having a significant bearing on the probability or consequences of accidental release of radioactive materials;

(D) The safety features that are to be engineered into the facility and those barriers that must be breached as a result of an accident before a release of radioactive material to the environment can occur. Special attention must be directed to plant design features intended to mitigate the radiological consequences of accidents. In performing this assessment, an applicant shall assume a fission product release[6] from the core into the containment assuming that the facility is operated at the ultimate power level

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[6] The fission product release assumed for this evaluation should be based upon a major accident, hypothesized for purposes of site analysis or postulated from considerations of possible accidental events. Such accidents have generally been assumed to result in substantial meltdown of the core with subsequent release into the containment of appreciable quantities of fission products.

contemplated. The applicant shall perform an evaluation and analysis of the postulated fission product release, using the expected demonstrable containment leak rate and any fission product cleanup systems intended to mitigate the consequences of the accidents, together with applicable site characteristics, including site meteorology, to evaluate the offsite radiological consequences. Site characteristics must comply with Part 100 of this chapter. The evaluation must determine that:

(1) An individual located at any point on the boundary of the exclusion area for any 2 hour period following the onset of the postulated fission product release, would not receive a radiation dose in excess of 25 rem[7] total effective dose equivalent (TEDE).

(2) An individual located at any point on the outer boundary of the low population zone, who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage) would not receive a radiation dose in excess of 25 rem total effective dose equivalent (TEDE).

(E) With respect to operation at the projected initial power level, the applicant is required to submit information prescribed in paragraphs (a)(2) through (a)(8) of this section, as well as the information required by this paragraph, in support of the application for a construction permit, or a design approval.

* * * * *

(12) On or after [INSERT EFFECTIVE DATE OF THE FINAL RULE], stationary power reactor applicants who apply for a construction permit pursuant to this part, or a design certification or combined license pursuant to Part 52 of this chapter, as partial conformance to General Design Criterion 2 of Appendix A to this part, shall comply with the earthquake engineering criteria in Appendix S of this part.

(b) * * *

(10) On or after [INSERT EFFECTIVE DATE OF THE FINAL RULE], stationary power reactor applicants who apply for an operating license pursuant to this part, or a design certification or combined license pursuant to Part 52 of this chapter, as partial conformance to General Design Criterion 2 of Appendix A to this part, shall comply with the earthquake engineering criteria of Appendix S to this part. However, if the construction permit was issued prior to [INSERT EFFECTIVE DATE OF THE FINAL RULE], the stationary power reactor applicant shall comply with the earthquake engineering criteria in Section VI of Appendix A to Part 100 of this chapter.

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[7] A whole body dose of 25 rem has been stated to correspond numerically to the once in a lifetime accidental or emergency dose for radiation workers which, according to NCRP recommendations at the time could be disregarded in the determination of their radiation exposure status (see NBS Handbook 69 dated June 5, 1959). However, its use is not intended to imply that this number constitutes an acceptable limit for an emergency dose to the public under accident conditions. Rather, this dose value has been set forth in this section as a reference value, which can be used in the evaluation of plant design features with respect to postulated reactor accidents, in order to assure that such designs provide assurance of low risk of public exposure to radiation, in the event of such accidents.

(11) On or after [INSERT EFFECTIVE DATE OF THE FINAL RULE], stationary power reactor applicants who apply for an operating license pursuant to this Part, or a combined license pursuant to Part 52 of this chapter, shall provide a description and safety assessment of the site and of the facility as in §50.34(a)(1)(ii) of this part.

* * * * *

5. In §50.54, paragraph (ff) is added to read as follows:

§50.54 Conditions of licenses.

* * * * *

(ff) For licensees of nuclear power plants that have implemented the earthquake engineering criteria in Appendix S of this part, plant shutdown is required as provided in Paragraph IV(a)(3) of Appendix S. Prior to resuming operations, the licensee shall demonstrate to the Commission that no functional damage has occurred to those features necessary for continued operation without undue risk to the health and safety of the public and the licensing basis is maintained.

6. Appendix S to Part 50 is added to read as follows:

APPENDIX S TO PART 50 - EARTHQUAKE ENGINEERING CRITERIA FOR NUCLEAR POWER PLANTS

General Information

This appendix applies to applicants for a design certification or combined license pursuant to Part 52 of this chapter or a construction permit or operating license pursuant to Part 50 of this chapter on or after [INSERT EFFECTIVE DATE OF THE FINAL RULE]. However, if the construction permit was issued prior to [INSERT EFFECTIVE DATE OF THE FINAL RULE], the operating license applicant shall comply with the earthquake engineering criteria in Section VI of Appendix A to 10 CFR Part 100.

I. Introduction

Each applicant for a construction permit, operating license, design certification, or combined license is required by §50.34(a)(12), (b)(10), and General Design Criterion 2 of Appendix A to this Part to design nuclear power plant structures, systems, and components important to safety to withstand the effects of natural phenomena, such as earthquakes, without loss of capability to perform their safety functions. Also, as specified in § 50.54(ff), nuclear power plants that have implemented the earthquake engineering criteria described herein must shut down if the criteria in Paragraph IV(a)(3) of this appendix are exceeded.

These criteria implement General Design Criterion 2 insofar as it requires structures, systems, and components important to safety to withstand the effects of earthquakes.

## II. Scope

The evaluations described in this appendix are within the scope of investigations permitted by §50.10(c)(1).

## III. Definitions

As used in these criteria:

Combined license means a combined construction permit and operating license with conditions for a nuclear power facility issued pursuant to Subpart C of Part 52 of this chapter.

Design Certification means a Commission approval, issued pursuant to Subpart B of Part 52 of this chapter, of a standard design for a nuclear power facility. A design so approved may be referred to as a "certified standard design."

The Operating Basis Earthquake Ground Motion (OBE) is the vibratory ground motion for which those features of the nuclear power plant necessary for continued operation without undue risk to the health and safety of the public will remain functional. The Operating Basis Earthquake Ground Motion is only associated with plant shutdown and inspection unless specifically selected by the applicant as a design input.

A response spectrum is a plot of the maximum responses (acceleration, velocity, or displacement) of idealized single-degree-of-freedom oscillators as a function of the natural frequencies of the oscillators for a given damping value. The response spectrum is calculated for a specified vibratory motion input at the oscillators' supports.

The Safe Shutdown Earthquake Ground Motion (SSE) is the vibratory ground motion for which certain structures, systems, and components must be designed to remain functional.

The structures, systems, and components required to withstand the effects of the Safe Shutdown Earthquake Ground Motion or surface deformation 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, or

(3) The capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures comparable to the guideline exposures of §50.34(a)(1)(ii).

Surface deformation is distortion of geologic strata at or near the ground surface by the processes of folding or faulting as a result of various earth forces. Tectonic surface deformation is associated with earthquake processes.

## IV. Application To Engineering Design

The following are pursuant to the seismic and geologic design basis requireme... of §100.23 of this chapter:

(a) ...bratory Ground Motion.

(... ...e Shutdown Earthquake Ground Motion. The Safe Shutdown Earthqua... ...und Motion must be characterized by free-field ground motion response ...ra at the free ground surface. In view of the limited data available on vibratory ground motions of strong earthquakes, it usually will be appropriate that the design response spectra be smoothed spectra. The horizontal component of the Safe Shutdown Earthquake Ground Motion in the free-field at the foundation level of the structures must be an appropriate response spectrum with a peak ground acceleration of at least 0.1g.

The nuclear power plant must be designed so that, if the Safe Shutdown Earthquake Ground Motion occurs, certain structures, systems, and components will remain functional and within applicable stress, strain, and deformation limits. In addition to seismic loads, applicable concurrent normal operating, functional, and accident-induced loads must be taken into account in the design of these safety-related structures, systems, and components. The design of the nuclear power plant must also take into account the possible effects of the Safe Shutdown Earthquake Ground Motion on the facility foundations by ground disruption, such as fissuring, lateral spreads, differential settlement, liquefaction, and landsliding, as required in §100.23 to Part 100 of this chapter.

The required safety functions of structures, systems, and components must be assured during and after the vibratory ground motion associated with the Safe Shutdown Earthquake Ground Motion through design, testing, or qualification methods.

The evaluation must take into account soil-structure interaction effects and the expected duration of vibratory motion. It is permissible to design for strain limits in excess of yield strain in some of these safety-related structures, systems, and components during the Safe Shutdown Earthquake Ground Motion and under the postulated concurrent loads, provided the necessary safety functions are maintained.

(2) **Operating Basis Earthquake Ground Motion.**

(i) The Operating Basis Earthquake Ground Motion must be characterized by response spectra. The value of the Operating Basis Earthquake Ground Motion must be set to one of the following choices:

(A) One-third or less of the Safe Shutdown Earthquake Ground Motion design response spectra. The requirements associated with this Operating Basis Earthquake Ground Motion in Paragraph (a)(2)(i)(B)(*I*) can be satisfied without the applicant performing explicit response or design analyses, or

(B) A value greater than one-third of the Safe Shutdown Earthquake Ground Motion design response spectra. Analysis and design must be performed to demonstrate that the requirements associated with this Operating Basis Earthquake Ground Motion in Paragraph (a)(2)(i)(B)(*I*) are satisfied. The design must take into account soil-structure interaction effects and the duration of vibratory ground motion.

(*I*) When subjected to the effects of the Operating Basis Earthquake Ground Motion in combination with normal operating loads, all structures, systems, and components of the nuclear power plant necessary for continued operation without undue risk to the health and safety of the public must remain functional and within applicable stress, strain, and deformation limits.

(3) **Required Plant Shutdown.** If vibratory ground motion exceeding that of the Operating Basis Earthquake Ground Motion or if significant plant damage occurs, the licensee must shut down the nuclear power plant. If systems, structures, or components necessary for the safe shutdown of the nuclear power plant are not available after the occurrence of the Operating Basis Earthquake Ground Motion, the licensee must consult with the Commission and must propose a plan for the timely, safe shutdown of the nuclear power plant. Prior to resuming operations, the licensee must demonstrate to the Commission that no functional damage has occurred to those features necessary for continued operation without undue risk to the health and safety of the public.

(4) **Required Seismic Instrumentation.** Suitable instrumentation must be provided so that the seismic response of nuclear power plant features important to safety can be evaluated promptly after an earthquake.

(5) **Surface Deformation.** The potential for surface deformation must be taken into account in the design of the nuclear power plant by providing reasonable assurance that in the event of deformation, certain structures, systems, and components will remain functional. In addition to surface deformation induced loads, the design of safety features must take into account seismic loads and applicable concurrent functional and accident-induced loads. The design provisions for surface deformation must be based on its postulated occurrence in any direction and azimuth and under any part of the nuclear power plant, unless evidence indicates this assumption is not appropriate, and must take into account the estimated rate at which the surface deformation may occur.

(c) **Seismically Induced Floods and Water Waves and Other Design Conditions.** Seismically induced floods and water waves from either locally or distantly generated seismic activity and other design conditions determined pursuant to §100.23 of this chapter must be taken into account in the design of the nuclear power plant so as to prevent undue risk to the health and safety of the public.

**PART 52 — EARLY SITE PERMITS; STANDARD DESIGN CERTIFICATIONS; AND COMBINED LICENSES FOR NUCLEAR POWER PLANTS**

7. The authority citation for Part 52 continues to read as follows:

AUTHORITY: Secs. 103, 104, 161, 182, 183, 186, 189, 68 Stat. 936, 948, 953, 954, 955, 956, as amended, sec. 234, 83 Stat. 1244, as amended (42 U.S.C. 2133, 2201, 2232, 2233, 2236, 2239, 2282); secs. 201, 202, 206, 88 Stat. 1242, 1244, 1246, as amended (42 U.S.C. 5841, 5842, 5846).

8. In §52.17, the introductory text of paragraph (a)(1) and paragraph (a)(1)(vi) are revised to read as follows:

§52.17 Contents of applications.

(a)(1) The application must contain the information required by § 50.33(a)-(d), the information required by § 50.34 (a)(12) and (b)(10), and to the exter pproval of emergency plans is sought under paragraph (b)(2)(ii) of this sec the information required by § 50.33 (g) and (j), and § 50.34 (b)(6)(v he application must also contain a description and safety assessmen the site on which the facility is to be located. The assessment must contain an analysis and evaluation of the major structures, systems, and components of the facility that bear significantly on the acceptability of the site under the radiological consequence evaluation factors identified in § 50.34(a)(1) of this chapter. Site characteristics must comply with Part 100 of this chapter. In addition, the application should describe the following:

* * * * *

(vi) The seismic, meteorological, hydrologic, and geologic characteristics of the proposed site;

* * * * *

PART 100 -- REACTOR SITE CRITERIA

9. The authority citation for Part 100 continues to read as follows:

AUTHORITY: Secs. 103, 104, 161, 182, 68 Stat. 936, 937, 948, 953, as amended (42 U.S.C. 2133, 2134, 2201, 2232); sec. 201, as amended, 202, 88 Stat. 1242, as amended, 1244 (42 U.S.C. 5841, 5842).

10. The table of contents for Part 100 is revised to read as follows:

PART 100 - REACTOR SITE CRITERIA

**Subpart B — Evaluation Factors for Stationary Power Reactor Site Applications on or after [EFFECTIVE DATE OF THE FINAL RULE].**

100.20 Factors to be considered when evaluating sites.
100.21 Non-seismic site criteria.
100.23 Geologic and seismic siting criteria.

APPENDIX A — Seismic and Geologic Siting Criteria for Nuclear Power Plants.

11. Section 100.1 is revised to read as follows:

**§ 100.1 Purpose.**

(a) The purpose of this part is to establish approval requirements for proposed sites for stationary power and testing reactors subject to Part 50 or Part 52 of this chapter.

(b) There exists a substantial base of knowledge regarding power reactor siting, design, construction and operation. This base reflects that the primary factors that determine public health and safety are the reactor design, construction and operation.

(c) Siting factors and criteria are important in assuring that radiological doses from normal operation and postulated accidents will be acceptably low, that natural phenomena and potential man-made hazards will be appropriately accounted for in the design of the plant, and that the site characteristics are amenable to the development of adequate emergency plans to protect the public and adequate security measures to protect the plant.

(d) This approach incorporates the appropriate standards and criteria for approval of stationary power and testing reactor sites. The Commission intends to carry out a traditional defense-in-depth approach with regard to reactor siting to ensure public safety. Siting away from densely populated centers has been and will continue to be an important factor in evaluating applications for site approval.

12. Section 100.2 is revised to read as follows:

**§ 100.2 Scope.**

The siting requirements contained in this part apply to applications for site approval for the purpose of constructing and operating stationary power and testing reactors pursuant to the provisions of Parts 50 or 52 of this chapter.

13. Section 100.3 is revised to read as follows:

**§ 100.3 Definitions.**

As used in this part:

<u>Combined license</u> means a combined construction permit and operating license with conditions for a nuclear power facility issued pursuant to Subpart C of Part 52 of this chapter.

Early Site Permit means a Commission approval, issued pursuant to subpart A of Part 52 of this chapter, for a site or sites for one or more nuclear power facilities.

Exclusion area means that area surrounding the reactor, in which the reactor licensee has the authority to determine all activities including exclusion or removal of personnel and property from the area. This area may be traversed by a highway, railroad, or waterway, provided these are not so close to the facility as to interfere with normal operations of the facility and provided appropriate and effective arrangements are made to control traffic on the highway, railroad, or waterway, in case of emergency, to protect the public health and safety. Residence within the exclusion area shall normally be prohibited. In any event, residents shall be subject to ready removal in case of necessity. Activities unrelated to operation of the reactor may be permitted in an exclusion area under appropriate limitations, provided that no significant hazards to the public health and safety will result.

Low population zone means the area immediately surrounding the exclusion area which contains residents, the total number and density of which are such that there is a reasonable probability that appropriate protective measures could be taken in their behalf in the event of a serious accident. These guides do not specify a permissible population density or total population within this zone because the situation may vary from case to case. Whether a specific number of people can, for example, be evacuated from a specific area, or instructed to take shelter, on a timely basis will depend on many factors such as location, number and size of highways, scope and extent of advance planning, and actual distribution of residents within the area.

Population center distance means the distance from the reactor to the nearest boundary of a densely populated center containing more than about 25,000 residents.

Power reactor means a nuclear reactor of a type described in §50.21(b) or 50.22 of this chapter designed to produce electrical or heat energy.

A Response spectrum is a plot of the maximum responses (acceleration, velocity, or displacement) of idealized single-degree-of-freedom oscillators as a function of the natural frequencies of the oscillators for a given damping value. The response spectrum is calculated for a specified vibratory motion input at the oscillators' supports.

The Safe Shutdown Earthquake Ground Motion is the vibratory ground motion for which certain structures, systems, and components must be designed pursuant to Appendix S to Part 50 of this chapter to remain functional.

Surface deformation is distortion of geologic strata at or near the ground surface by the processes of folding or faulting as a result of various earth forces. Tectonic surface deformation is associated with earthquake processes.

Testing reactor means a testing facility as defined in §50.2 of this chapter.

14. Section 100.4 is added to read as follows:

§100.4 Communications.

Except where otherwise specified in this part, all correspondence, reports, applications, and other written communications submitted pursuant to 10 CFR 100 should be addressed to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555, and copies sent to the appropriate Regional

Office and Resident Inspector. Communications and reports may be delivered in person at the Commission's offices at 2120 L Street, NW., Washington, DC, or at 11555 Rockville Pike, Rockville, Maryland.

15. Section 100.8 is revised to read as follows:

**§ 100.8 Information collection requirements: OMB approval.**

(a) The Nuclear Regulatory Commission has submitted the information collection requirements contained in this part to the Office of Management and Budget (OMB) for approval as required by the Paperwork Reduction Act of 1980 (44 U.S.C. 3501 et seq.). OMB has approved the information collection requirements contained in this part under control number 3150-0093.

(b) The approved information collection requirements contained in this part appear in §100.23 and Appendix A.

16. A heading for Subpart A is added directly before §100.10 to read as follows:

Subpart A — Evaluation Factors for Stationary Power Reactor Site Applications before [EFFECTIVE DATE OF THIS REGULATION] and for Testing Reactors.

17. Section 100.10 is revised to read as follows:

§100.10 Factors to be considered when evaluating sites.

Factors considered in the evaluation of sites include those relating both to the proposed reactor design and the characteristics peculiar to the site. It is expected that reactors will reflect through their design, construction and operation an extremely low probability for accidents that could result in release of significant quantities of radioactive fission products. In addition, the site location and the engineered features included as safeguards against the hazardous consequences of an accident, should one occur, should insure a low risk of public exposure. In particular, the Commission will take the following factors into consideration in determining the acceptability of a site for a power or testing reactor:

(a). Characteristics of reactor design and proposed operation including--

(1) Intended use of the reactor including the proposed maximum power level and the nature and inventory of contained radioactive materials;

(2) The extent to which generally accepted engineering standards are applied to the design of the reactor;

(3) The extent to which the reactor incorporates unique or unusual features having a significant bearing on the probability or consequences of accidental release of radioactive materials;

(4) The safety features that are to be engineered into the facility and those barriers that must be breached as a result of an accident before a release of radioactive material to the environment can occur

(b) Population density and use characteristics of the site environs, including the exclusion area, low population zone, and the population center distance.

(c) Physical characteristics of the site, including seismology, meteorology, geology, and hydrology.

(1) Appendix A to Part 100, "Seismic and Geologic Siting Criteria for Nuclear Power Plants," describes the nature of investigations required to obtain the geologic and seismic data necessary to determine site suitability and to provide reasonable assurance that a nuclear power plant can be constructed and operated at a proposed site without undue risk to the health and safety of the public. It describes procedures for determining the quantitative vibratory ground motion design basis at a site due to earthquakes and describes information needed to determine whether and to what extent a nuclear power plant need be designed to withstand the effects of surface faulting.

(2) Meteorological conditions at the site and in the surrounding area should be considered.

(3) Geological and hydrological characteristics of the proposed site may have a bearing on the consequences of an escape of radioactive material from the facility. Special precautions should be planned if a reactor is to be located at a site where a significant quantity of radioactive effluent might accidentally flow into nearby streams or rivers or might find ready access to underground water tables.

(d) Where unfavorable physical characteristics of the site exist, the proposed site may nevertheless be found to be acceptable if the design of the facility includes appropriate and adequate compensating engineering safeguards.

13. Section 100.11 is revised to read as follows:

§100.11 Determination of exclusion area, low population zone, and population center distance.

(a) As an aid in evaluating a proposed site, an applicant should assume a fission product release[1] from the core, the expected demonstrable leak rate from the containment and the meteorological conditions pertinent to his site to derive an exclusion area, a low population zone and population center distance. For the purpose of this analysis, which shall set forth the basis for the numerical values used, the applicant should determine the following:

(1) An exclusion area of such size that an individual located at any point on its boundary for two hours immediately following onset of the postulated fission product release would not receive a total radiation dose to the whole

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[1]The fission product release assumed for these calculations should be based upon a major accident, hypothesized for purposes of site analysis or postulated from considerations of possible accidental events, that would result in potential hazards not exceeded by those from any accident considered credible. Such accidents have generally been assumed to result in substantial meltdown of the core with subsequent release of appreciable quantities of fission products.

body in excess of 25 rem[2] or a total radiation dose in excess of 300 rem to the thyroid from iodine exposure.

(2) A low population zone of such size that an individual located at any point on its outer boundary who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage) would not receive a total radiation dose to the whole body in excess of 25 rem or a total radiation dose in excess of 300 rem to the thyroid from iodine exposure.

(3) A population center distance of at least one and one-third times the distance from the reactor to the outer boundary of the low population zone. In applying this guide, the boundary of the population center shall be determined upon consideration of population distribution. Political boundaries are not controlling in the application of this guide. Where very large cities are involved, a greater distance may be necessary because of total integrated population dose consideration.

(b) For sites for multiple reactor facilities consideration should be given to the following:

(1) If the reactors are independent to the extent that an accident in one reactor would not initiate an accident in another, the size of the exclusion area, low population zone and population center distance shall be fulfilled with respect to each reactor individually. The envelopes of the plan overlay of the areas so calculated shall then be taken as their respective boundaries.

(2) If the reactors are interconnected to the extent that an accident in one reactor could affect the safety of operation of any other, the size of the exclusion area, low population zone and population center distance shall be based upon the assumption that all interconnected reactors emit their postulated fission product releases simultaneously. This requirement may be reduced in relation to the degree of coupling between reactors, the probability of concomitant accidents and the probability that an individual would not be exposed to the radiation effects from simultaneous releases. The applicant would be expected to justify to the satisfaction of the Commission the basis for such a reduction in the source term.

(3) The applicant is expected to show that the simultaneous operation of multiple reactors at a site will not result in total radioactive effluent releases beyond the allowable limits of applicable regulations.

NOTE: For further guidance in developing the exclusion area, the low population zone, and the population center distance, reference is made to Technical Information Document 14844, dated March 23, 1962, which contains a procedural method and a sample calculation that result in distances roughly reflecting current siting practices of the Commission. The calculations described in Technical Information Document 14844 may be used as a point of departure for

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[2] The whole body dose of 25 rem referred to above corresponds numerically to the once in a lifetime accidental or emergency dose for radiation workers which, according to NCRP recommendations may be disregarded in the determination of their radiation exposure status (see NBS Handbook 69 dated June 5, 1959). However, neither its use nor that of the 300 rem value for thyroid exposure as set forth in these site criteria guides are intended to imply that these numbers constitute acceptable limits for emergency doses to the public under accident conditions. Rather, this 25 rem whole body value and the 300 rem thyroid value have been set forth in these guides as reference values, which can be used in the evaluation of reactor sites with respect to potential reactor accidents of exceedingly low probability of occurrence, and low risk of public exposure to radiation.

consideration of particular site requirements which may result from evaluation of the characteristics of a particular reactor, its purpose and method of operation.

Copies of Technical Information Document 14844 may be obtained from the Commission's Public Document Room, 2120 L Street NW.(Lower Level), Washington, DC, or by writing the Director of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, DC 20555.

19. Subpart B (§§100.20 - 100.23) is added to read as follows:

**Subpart B — Evaluation Factors for Stationary Power Reactor Site Applications on or After [EFFECTIVE DATE OF THE FINAL RULE].**

**§100.20 Factors to be considered when evaluating sites.**

The Commission will take the following factors into consideration in determining the acceptability of a site for a stationary power reactor:

(a) Population density and use characteristics of the site environs, including the exclusion area, the population distribution, and site-related characteristics must be evaluated to determine whether individual as well as societal risk of potential plant accidents is low, and that site-related characteristics would not prevent the development of a plan to carry out suitable protective actions for members of the public in the event of emergency.

(b) The nature and proximity of man-related hazards (e.g., airports, dams, transportation routes, military and chemical facilities) must be evaluated to establish site parameters for use in determining whether a plant design can accommodate commonly occurring hazards, and whether the risk of other hazards is very low.

(c) Physical characteristics of the site, including seismology, meteorology, geology, and hydrology.

(1) §100.23, "Geologic and seismic siting factors," of this part describes the criteria and nature of investigations required to obtain the geologic and seismic data necessary to determine the suitability of the proposed site and the plant design bases.

(2) Meteorological characteristics of the site that are necessary for safety analysis or that may have an impact upon plant design (such as maximum probable wind speed and precipitation) must be identified and characterized.

(3) Factors important to hydrological radionuclide transport (such as soil, sediment, and rock characteristics, adsorption and retention coefficients, ground water velocity, and distances to the nearest surface body of water) must be obtained from on-site measurements. The maximum probable flood along with the potential for seismically induced floods discussed in §100.23 (d)(3) of this part must be estimated using historical data.

§ 100.21 Non-seismic siting criteria.

Applications for site approval for commercial power reactors shall demonstrate that the proposed site meets the following criteria:

(a) Every site must have an exclusion area and a low population zone, as defined in §100.3;

(b) The population center distance, as defined in §100.3, must be at least one and one-third times the distance from the reactor to the outer boundary of the low population zone. In applying this guide, the boundary of the population center shall be determined upon consideration of population distribution. Political boundaries are not controlling in the application of this guide;

(c) Site atmospheric dispersion characteristics must be evaluated and dispersion parameters established such that:

(1) Radiological effluent release limits associated with normal operation from the type of facility proposed to be located at the site can be met for any individual located offsite; and

(2) Radiological dose consequences of postulated accidents shall meet the criteria set forth in §50.34(a)(1) of this chapter for the type of facility proposed to be located at the site;

(d) The physical characteristics of the site, including meteorology, geology, seismology, and hydrology must be evaluated and site parameters established such that potential threats from such physical characteristics will pose no undue risk to the type of facility proposed to be located at the site;

(e) Potential hazards associated with nearby transportation routes, industrial and military facilities must be evaluated and site parameters established such that potential hazards from such routes and facilities will pose no undue risk to the type of facility proposed to be located at the site;

(f) Site characteristics must be such that adequate security plans and measures can be developed;

(g) Site characteristics must be such that adequate plans to take protective actions for members of the public in the event of emergency can be developed:

(h) Reactor sites should be located away from very densely populated centers. Areas of low population density are, generally, preferred. However, in determining the acceptability of a particular site located away from a very densely populated center but not in an area of low density, consideration will be given to safety, environmental, economic, or other factors, which may result in the site being found acceptable[3].

§ 100.23 Geologic and seismic siting factors.

This section sets forth the principal geologic and seismic considerations that guide the Commission in its evaluation of the suitability of a proposed site

[3] Examples of these factors include, but are not limited to, such factors as the higher population density site having superior seismic characteristics, better access to skilled labor for construction, better rail and highway access, shorter transmission line requirements, or less environmental impact on undeveloped areas, wetlands or endangered species, etc. Some of these factors are included in, or impact, the other criteria included in this section.

and adequacy of the design bases established in consideration of the geologic and seismic characteristics of the proposed site, such that, there is a reasonable assurance that a nuclear power plant can be constructed and operated at the proposed site without undue risk to the health and safety of the public. Applications to engineering design are contained in Appendix S to Part 50 of this chapter.

(a) **Applicability**. The requirements in paragraphs (c) and (d) of this section apply to applicants for an early site permit or combined license pursuant to Part 52 of this chapter, or a construction permit or operating license for a nuclear power plant pursuant to Part 50 of this chapter on or after [INSERT EFFECTIVE DATE OF THE FINAL RULE]. However, if the construction permit was issued prior to [INSERT EFFECTIVE DATE OF THE FINAL RULE], the operating license applicant shall comply with the seismic and geologic siting criteria in Appendix A to Part 100 of this chapter.

(b) **Commencement of construction**. The investigations required in paragraph (c) of this section are within the scope of investigations permitted by § 50.10(c)(1) of this chapter.

(c) **Geological, seismological, and engineering characteristics**. The geological, seismological, and engineering characteristics of a site and its environs must be investigated in sufficient scope and detail to permit an adequate evaluation of the proposed site, to provide sufficient information to support evaluations performed to arrive at estimates of the Safe Shutdown Earthquake Ground Motion, and to permit adequate engineering solutions to actual or potential geologic and seismic effects at the proposed site. The size of the region to be investigated and the type of data pertinent to the investigations must be determined based on the nature of the region surrounding the proposed site. Data on the vibratory ground motion, tectonic surface deformation, nontectonic deformation, earthquake recurrence rates, fault geometry and slip rates, site foundation material, and seismically induced floods and water waves must be obtained by reviewing pertinent literature and carrying out field investigations. However, each applicant shall investigate all geologic and seismic factors (for example, volcanic activity) that may affect the design and operation of the proposed nuclear power plant irrespective of whether such factors are explicitly included in this section.

(d) **Geologic and seismic siting factors**. The geologic and seismic siting factors considered for design must include a determination of the Safe Shutdown Earthquake Ground Motion for the site, the potential for surface tectonic and nontectonic deformations, the design bases for seismically induced floods and water waves, and other design conditions as stated in paragraph (d)(4) of this section.

(1) **Determination of the Safe Shutdown Earthquake Ground Motion**. The Safe Shutdown Earthquake Ground Motion for the site is characterized by both horizontal and vertical free-field ground motion response spectra at the free ground surface. The Safe Shutdown Earthquake Ground Motion for the site is determined considering the results of the investigations required by paragraph (c) of this section. Uncertainties are inherent in such estimates. These uncertainties must be addressed through an appropriate analysis, such as a probabilistic seismic hazard analysis or suitable sensitivity analyses. Paragraph IV(a)(1) of Appendix S to Part 50 of this chapter defines the minimum Safe Shutdown Earthquake Ground Motion for design.

(2) Determination of the potential for surface tectonic and nontectonic deformations. Sufficient geological, seismological, and geophysical data must be provided to clearly establish whether there is a potential for surface deformation.

(3) Determination of design bases for seismically induced floods and water waves. The size of seismically induced floods and water waves that could affect a site from either locally or distantly generated seismic activity must be determined.

(4) Determination of siting factors for other design conditions. Siting factors for other design conditions that must be evaluated include soil and rock stability, liquefaction potential, natural and artificial slope stability, cooling water supply, and remote safety-related structure siting. Each applicant shall evaluate all siting factors and potential causes of failure, such as, the physical properties of the materials underlying the site, ground disruption, and the effects of vibratory ground motion that may affect the design and operation of the proposed nuclear power plant.

Dated at Rockville, Maryland, this ___ day of ____ .

For the Nuclear Regulatory Commission.

John C. Hoyle,
Acting Secretary of the Commission.

# RESOLUTION OF PUBLIC COMMENTS

# SEISMIC AND EARTHQUAKE ENGINEERING RULE

# RESOLUTION OF PUBLIC COMMENTS

# ON THE PROPOSED

# SEISMIC AND EARTHQUAKE ENGINEERING CRITERIA

# FOR NUCLEAR POWER PLANTS

Section 100.23, Geologic and Seismic Siting Factors
to 10 CFR Part 100

and

Appendix S, Earthquake Engineering Criteria for Nuclear Power Plants
to 10 CFR Part 50

October 17, 1994 Publication

85

COMMENT RESOLUTION

Section 100.23, Geologic and Seismic Siting Factors
to 10 CFR Part 100

and

Appendix S, Earthquake Engineering Criteria for Nuclear Power Plants
to 10 CFR Part 50

BACKGROUND

The first proposed revision of the Reactor Site Criteria Including Seismic and Earthquake Engineering Criteria for Nuclear Power Plants (10 CFR Parts 50, 52 and 100) was published for public comment on October 20, 1992, (57 FR 47802). The availability of the draft regulatory guides and standard review plan section that were developed to provide guidance on meeting the proposed regulations was published on November 25, 1992, (57 FR 55601). Because of the substantive nature of the changes to be made in response to public comments the proposed regulations and draft guidance documents were withdrawn and replaced with the second proposed revision of the regulations published for public comment on October 17, 1994, (59 FR 52255). The availability of the draft guidance documents was published on February 28, 1995, (60 FR 10810).

Forty letters (References 1 through 40) contain comments on the October 1992 publication of Proposed Appendix B, "Criteria for the Seismic and Geologic Siting of Nuclear Power Plants on or After [Effective Date of the Final Rule]," to 10 CFR Part 100, "Reactor Site Criteria," and/or the first Proposed Appendix S, "Earthquake Engineering Criteria for Nuclear Power Plants," to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities." The Federal Register Notice published on October 17, 1994 (59 FR 52555) containing Proposed Section 100.23, "Geologic and Seismic Siting Factors," to 10 CFR Part 100 (replacement of Proposed Appendix B to 10 CFR Part 100) and the second Proposed Appendix S, "Earthquake Engineering Criteria for Nuclear Power Plants," to 10 CFR Part 50 reflect the only documentation pertaining to NRC staff evaluation and implementation of all comments provided in References 1 to 40.

The resolution of comments contained below relate to the October 17, 1994 publication.

## RESOLUTION OF COMMENTS ON SUPPLEMENTAL INFORMATION

### Applicability

1a. "The proposed regulatory action would apply to applicants who apply for a construction permit, operating license, preliminary design approval, final design approval, manufacturing license, early site permit, design certification, or combined license ..." This statement does not explicitly indicate whether or not the proposed revisions would apply to the Mined Geologic Disposal System (MGDS). (Reference 41)

1b "The proposed regulatory action would apply to applicants who apply for a construction permit, operating license, preliminary design approval, final design approval, manufacturing license, early site permit design certification, of combined license ..." This statement does not explicitly indicate whether or not the proposed revisions would apply to a Monitored Retrievable Storage (MRS) facility. (Reference 41)

Response. Although Appendix A to 10 CFR Part 100 is titled "Seismic and Geologic Siting Criteria for Nuclear Power Plants," it is also referenced in two other parts of the regulation. They are (1) Part 40, "Domestic Licensing of Source Material," Appendix A, "Criteria Relating to the Operation of Uranium Mills and the Disposition of Tailings or Waste Produced by the Extraction or Concentration of Source Material from Ores Processed Primarily for Their Source Material Content," Section I, Criterion 4(e), and (2) Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste," Paragraphs (a)(2), (b) and (f)(1) of §72.102.

The referenced applicability of Section 100.23 to other than power reactors, if considered appropriate by the NRC, would be a separate rulemaking. That rulemaking would clearly state the applicability of Section 100.23 to a MRS or other facility. In addition, NUREG-1451 will remain the NRC staff technical position on seismic siting issues pertaining to a MGDS until it is superseded through a rulemaking, revision of NUREG-1451, or other appropriate mechanism.

Section V(D)(3) "Uncertainties and Probabilistic Methods"

1. It is stated that "Because so little is known about earthquake phenomena..." Use of the expression "so little is known" creates a false impression of the current state of knowledge about earthquake phenomena. Although our understanding of earthquake phenomena remains uncertain, quantum advances in knowledge have been made during the past 25 years. With these very significant advances, geoscientists now have much more confidence than previously in expressions of uncertainty regarding interpretations of inputs to a probabilistic seismic hazard analyses; and these can be fully accounted for in the uncertainty in the seismic hazard results. The language of the regulation should reflect these very positive developments. (Reference 41)

   Response: The statement will be revised to put less emphasis on the negative as follows: "Because of uncertainties about earthquake phenomena (especially in the eastern United States), there have often been differences of opinion and differing....."

2. The key elements of the NRC's proposed balanced approach are listed. The wording of the fourth element should be revised to indicate that the geoscience investigations refer to site-specific data, or new regional data, or a combination of the two. (Reference 41)

   Response: It refers to both regional and site investigations. The element will be revised to: "Determine if information from the regional and site geoscience investigations....."

Section V(B)(5), "Value of the Operating Basis Earthquake Ground Motion (OBE) and Required OBE Analysis."

Does not support the NRC staff's position to not require explicit design analysis for the Operating Basis Earthquake Ground Motion (OBE). The staff's position is not sound, not technically justified, and not appropriate for the design of Section III pressure-retaining components. It is not possible to inspect to verify that cyclic fatigue effects for the OBE are insignificant. There is no technical basis to state that OBE should not control the design of safety systems. It is not technically justified to assume that Section III components will remain within applicable stress limits at one-third of the SSE. Equipment necessary for continued operation, but not required for safe shutdown, is not required to be designed for OBE nor SSE.

The following specific comments [1 through 7] pertain to the supplemental information to the proposed regulations, item V(B)(5), "Value of the Operating Basis Earthquake Ground Motion (OBE) and Required OBE Analysis." Comments are limited to the design of pressure-retaining components to the ASME Boiler and Pressure Vessel Section III rules. (Reference 42)

1. Regarding the soundness of SSE only design:

> "For instance, the NRC staff, SECY-79-300, suggested that design for a single limiting event and inspection and evaluation for earthquakes in excess of some specified limit may be the most sound regulatory approach."

This is not a sound regulatory approach if it is not feasible to inspect for cyclic damage to all the pressure-retaining components. It is not feasible to inspect. Many components are not accessible. Even if accessible, the components may be covered with insulation. Even if there is not insulation or the insulation is removed, it is not feasible to inspect to determine the amount of the fatigue life used by the OBE cyclic loads. It is not feasible to inspect for crack initiation on the inside of the component in all critical areas. Even if it were feasible to inspect for cracks, it is possible to have an unacceptable amount of fatigue life used by the OBE without crack initiation. Visually inspecting for permanent deformation, or leakage, or failed component supports is certainly not adequate to determine cyclic damage.

Response. SECY-79-300, "Identification of Issues Pertaining to Seismic and Geologic Siting Regulation, Policy, and Practice for Nuclear Power Plants," informed the Commission of the status of the staff's reassessment of Appendix A, "Seismic and Geologic Siting Criteria for Nuclear Power Plants," to 10 CFR Part 100, "Reactor Site Criteria." The cited statement appeared in an enclosure (Enclosure B, Section 2.4) discussing issues arising from engineering requirements in Appendix A, procedures for providing an interface of these requirements with geologic and seismic input, and with matters involving scientific and engineering conservatism. In a related area (Enclosure A, Section 2.4), the NRC staff informed the Commission about problems in applying the Appendix A requirement that the plant must be shut down and inspected if ground motion in excess of that corresponding to the OBE occurs because there is no definitive shutdown guidance or inspection criteria.

The proposed regulations is similar to the statement in SECY-79-300 in that it allowed plants to be designed for a single limiting event (the SSE) and inspected and evaluated for earthquake in excess of some specified limit (the OBE) when and if it occurred. Also, the proposed regulation allowed for the plant to be designed at both the SSE and OBE levels. Earlier concerns expressed in SECY-79-300 regarding OBE exceedance and shutdown/restart guidelines have been resolved. A criterion to determine OBE exceedance is described in Regulatory Guide

1.166, "Pre-Earthquake Planning and Immediate Nuclear Power Plant Operator Postearthquake Actions," (Draft was DG-1034). Postearthquake inspection and evaluation guidance is described in Regulatory Guide 1.167, "Restart of a Nuclear Power Plant Shut Down by an Seismic Event," (Draft was DG-1035). The guidance is not limited to visual inspections, it includes inspections, tests, and analyses including fatigue analysis.

2. Regarding OBE controlling design:

> "In SECY-90-016, "Evolutionary Light Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements," the NRC staff states that it agrees that the OBE should not control the design of safety systems."

There is no technical basis for stating that the OBE should not control the design of safety systems. Based on my knowledge of current plant designs, I can state that if there are five OBE's of the magnitude of one-half the SSE expected to occur in the life of the plant, then OBE will control the design of the piping systems. And in this case, OBE should control the design. The cyclic effects of the repeated earthquakes have to be considered in the design of the component to ensure pressure boundary integrity throughout the life, especially if the SSE can occur after the lower level earthquakes.

The appropriate action is to define the magnitude of the OBE that is expected to occur, and to require the component manufacturer to design for the OBE. It appears that NRC is assuming the liability for the proper design of a pressure-retaining component for a lower level earthquake. It should be the N certificate holder's responsibility to provide a component that is structurally and functionally adequate for both the OBE and the SSE.

Response. The NRC staff agrees that the cyclic effects of repeated earthquakes have to be considered in the design of the components to ensure pressure boundary integrity. The NRC staff has identified actions necessary for the design of structures, systems, and components when the OBE design requirement is eliminated (these actions include fatigue analysis). A discussion pertaining to these actions (provided in SECY-93-087, Issue I.M), is included within supplemental information item V(B)(5) of the proposed regulation. The guidelines in SECY-93-087 provide a level of fatigue design for the piping equivalent to that currently provided in the Standard Review Plan Section 3.9.2.

Also, The NRC staff has concluded that design requirements based on an estimated OBE magnitude at the plant site and the number of events expected during the plant life will lead to low design values that will not control the design thus resulting in unnecessary analyses.

3. Regarding explicit response or design analyses:

> "The proposed regulation would allow the value of the OBE to be set at (i) one-third or less of the SSE, where OBE requirements are satisfied without an explicit response or design analysis.. "

The OBE requirements are -- "... components .... shall remain functional and within applicable stress, strain and deformation limits when subjected to the effects of the OBE in combination with normal operating loads."

It is not technically justified to assume that Section III components will remain within applicable stress limits (Level B limits) at one-third the SSE. The Section III acceptance criteria for Level D (for an SSE) is completely different than that for Level B (for an OBE). The Level D criteria is based on surviving the extremely-low probability SSE load. Gross structural deformations are possible, and it is expected that the component will have to be replaced. Cyclic effects are not considered. For Level B, the component must be designed to withstand the cyclic effects of the earthquake load and all other cyclic Level A and B loads without damage requiring repair.

In order for the assumption to be valid -- that at one-third SSE, the Level B criteria is satisfied for a component designed for the SSE -- the cyclic fatigue damage from the OBE must be insignificant. It is highly improbable that the fatigue damage from the OBE will be insignificant unless the component is designed for the OBE.

Response. The following is extracted from SECY-93-087, "Policy, Technical and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs," Issue I.M, "Elimination of Operating-Basis Earthquake."

> "A designer of piping systems considers the effects of primary and secondary stresses and evaluates fatigue caused by repeated cycles of loading. Primary stresses are induced by the inertial effects of vibratory motion. The relative motion of anchor points induces secondary stresses. The repeating seismic stress cycles induce cyclic effects (fatigue).
>
> After reviewing these aspects, the staff concludes that, for primary stresses, if the OBE is established at one-third the

SSE, the SSE load combinations control the piping design when the earthquake contribution dominates the load combination. Therefore, the staff concludes that eliminating the OBE piping stress load combination for primary stresses in piping systems will not significantly reduce existing safety margins.

Eliminating the OBE will, however, directly affect the current methods used to evaluate the adequacy of cyclic and secondary stress effects in the piping design. Eliminating the OBE from the load combination could cause uncertainty in evaluating the cyclic (fatigue) effects of earthquake-induced motions in piping systems and the relative motion effects of piping anchored to equipment and structures at various elevations because both of these effects are currently evaluated only for OBE loadings

Accordingly, to account for earthquake cycles in the fatigue analysis of piping systems, the staff proposes to develop guidelines for selecting a number of SSE cycles at a fraction of the peak amplitude of the SSE. These guidelines will provide a level of fatigue design for the piping equivalent to that currently provided in the standard review plan (SRP) (NUREG-0800)."

Positions pertaining to the elimination of the Operating Basis Earthquake were proposed in SECY-93-087. Commission approval is documented in a memorandum from Samuel J. Chilk to James M. Taylor, Subject: SECY-93-087 - Policy, Technical and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs, dated July 21, 1993.

4. Regarding the OBE and PRA insights:

> "There is high confidence that, at this ground-motion level with other postulated concurrent loads, most critical structures, systems, and components will not exceed currently used design limits. This is ensured, in part, because PRA insights will be used to support a margins-type assessment of seismic events."

This technical position is not valid for Section III pressure-retaining components. As stated under comment 3, cyclic effects are not considered for the SSE. There is no possible way to predetermine that the cyclic effects at one-third SSE are insignificant without evaluating specific configurations. To say that PRA insights from a margins-type assessment will ensure that Level B design limits will be satisfied at one-third SSE is completely wrong.

Response. See response to comment 3.

5. Regarding NRC proposed criteria:

> "Also, the NRC staff has evaluated the effect on safety of eliminating the OBE from the design load combinations for selected structures, systems, and components and has developed proposed criteria for an analysis using only the SSE."

The proposed criteria referred to is the proof that "SSE only" is not a prudent regulatory approach. In order to ensure that the OBE requirements are satisfied at one-third SSE, the NRC staff is requiring a fatigue evaluation for two SSE's for the ABWR. This may be more restrictive than designing for five OBE's at one-third SSE. Consider what has happened. The NRC staff realized that it is not sufficient for Section III components to be designed only for the SSE. They are requiring an explicit fatigue analysis so that the OBE requirements will be satisfied. The bottom line is that the NRC staff, in implementing "SSE only," have required an explicit for an equivalent OBE loading. A better approach would be to design for the OBE.

Response. The proposed criteria is a prudent regulatory approach. On the basis of analysis, tests, and engineering judgement, the NRC staff has determined the design produced using SSE load combinations, in general, envelop the load combinations produced using the OBE. For specific situations such as piping, where eliminating the OBE will directly affect the current methods used to evaluate the adequacy of cyclic and secondary stress effects in the piping design procedures have been developed (see response to comment 3).

6. Regarding required plant shutdown:

> "Prior to resuming operations, the licensee will be required to demonstrate to the Commission that no functional damage has occurred to those features necessary for continued operation without undue risk to the health and safety of the public."

If the applicant does not do an analysis and design for one-third SSE, the applicant is required to shutdown and inspect if the one-third SSE occurs. Obviously, the assumption is that the applicant can inspect to determine if there is damage to the Section III components. It is not possible to inspect to determine if there is cyclic damage to the Section III pressure-retaining components. The damage that has to be assessed is the effect of the cyclic loads on the life of the component. You are not inspecting for permanent deformations, leaks, or bent or failed supports. If these conditions occur at one-third SSE, then the plant seismic design is obviously deficient. You need to determine that the cyclic effects are not significant. This is impossible to determine by inspection. The question that has to be answered it whether the fatigue usage factor from the OBE is acceptable. The acceptability of the fatigue usage factor for a specific component is dependant on the severity of all the other cyclic loads on the component. The cyclic effects from the OBE for a component with high fatigue damage from service conditions, a pressurizer surge line or a nozzle subject to flow stratification effects for example, would have to be insignificant. The fatigue "damage" from the OBE cannot be determined by inspection. Analysis is the only method to verify that the OBE cyclic effects are within acceptable limits. The only reasonable approach is to perform the OBE fatigue analyses as part of the component design process.

Response. Postearthquake inspection and evaluation guidance is described in Draft Regulatory Guide DG-1035, "Restart of a Nuclear Power Plant Shut Down by an Seismic Event." The guidance is not limited to visual inspections, it includes inspections, tests, and analyses including fatigue analysis.

7. Regarding equipment seismic design:

> "The Operating Basis Earthquake Ground Motion (OBE) is the vibratory ground motion for which those features of the nuclear power plant necessary for continued operation without undue risk to the health and safety of the public will remain functional."
>
> "The Safe Shutdown Earthquake Ground Motion (SSE) is the vibratory ground motion for which certain structures, systems, and components must be designed to remain functional." [Three types of equipment are described.]

There is one major flaw in the "SSE only" design approach. The equipment designed for SSE is limited to the equipment necessary to assure the integrity of the reactor coolant pressure boundary, to shutdown the reactor, and to prevent or mitigate accident consequences. The equipment designed for SSE is only part of the equipment "necessary for continued operation without undue risk to the health and safety of the public." Hence, by this rule, it is possible that some equipment necessary for continued operation will not be designed for SSE or OBE effects.

I am disappointed that a proposed rule would be published with flaws in the technical logic. Perhaps the approach of designing for the SSE only is adequate for building structures designed to AISC rules, but this approach is certainly not adequate for Section III pressure-retaining components. There appears to be a lack of understanding of the Section III design requirements and the significance of seismic loads. To assume that the component stresses will be within the Section III Level B code requirements at 1/3 the SSE if the component is designed for the SSE is not valid. To assume that an applicant can properly inspect the safety related components after an OBE earthquake to determine that the ability of the components to function for the remaining life has not been impaired is unreasonable. The potential problem is detrimental impact on the fatigue life from the cyclic OBE loading. There is no feasible way to inspect for detrimental impact on fatigue life.

It is not prudent to design only for SSE, and to assume that there will be no cyclic damage from the OBE. I see no reason to compromise the seismic design of the plant. It is inappropriate to assume that design for OBE is not required without even knowing the component configuration.

We do have a problem in the industry with the present requirements. Requiring design for five OBE events at ½ SSE is unrealistic for most (all?) sites and requires an excessive and unnecessary number of seismic supports. The solution is to properly define the OBE magnitude and the number of events expected during the life of the plant. And to require design for that loading. OBE may or may not control the design. But you cannot assume, before you have the seismicity defined and before you have a component design, that OBE will not govern the design.

The problem with not designing for OBE can be simply stated. The pressure-retaining component may be designed to the fatigue limit for other Level A and B loads (for example, thermal transients). In this situation, OBE stresses above the endurance limit reduce the operational life of the component. It is highly improbable that OBE stresses will be below the endurance limit. The only way to accept the OBE stress cycles is to accept lower margins of safety. This is compromising the design of the plant, and is unnecessary. Design for OBE, if the OBE magnitude is reasonably defined, will not result in an excessive number of seismic supports.

The rule refers to "new information and research results." The newest information and research results is the Northridge earthquake and the Kobe earthquake. In the Northridge earthquake, steel building members cracked and this behavior was unexpected. In the Kobe earthquake, a seismically designed elevated highway toppled over, and this behavior was unexpected. What I have learned from these events and earlier earthquakes, is that our understanding of seismic response is limited. Conventional wisdom is that ductile steel piping systems will not fail in a single earthquake event. But in a recent NRC/EPRI program on dynamic reliability, undegraded piping components failed in a single earthquake event. The loadings were extreme in most cases, but the failure in a single event was not expected.

The intent of the rule making, to uncouple the OBE and the SSE, is a necessary change in the seismic requirements.

Response. It is not possible that some equipment necessary for continued safe operation will not be designed for SSE or OBE effects. General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena," of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50 requires that nuclear power plant structures, systems, and components important to safety be designed to withstand the effects of earthquakes without loss of capability to perform their safety functions. The criteria in Appendix S to 10 CFR Part 50 implement General Design Criterion 2 insofar as it requires structures, systems, and components important to safety to withstand the effects of earthquakes. Regulatory Guide 1.29, "Seismic Design Classification," describes a method acceptable to the NRC staff for identifying and classifying those features of light-water-cooled nuclear power plants that should be designed to withstand the effects of the SSE.

Currently, components which are designed for OBE only include components such as waste holdup tanks. As noted in the Supplemental Information, Section VII, Future Regulatory Actions, regulatory guides related to these components will be revised to provide alternative design requirements.

See response to comments 3 and 5 for discussions on stress limits and fatigue.

## RESOLUTION OF COMMENTS ON SECTION 100.23

(a) Applicability.

1. The language relevant to an applicant under Part 50 appears to be intended to avoid "backfitting" the new criteria in lieu of that used to obtain the construction permit originally. Unfortunately, the words *shall comply* unnecessarily imposes retention of the original Appendix A criteria on such applicants. Although unlikely, an applicant already holding a construction permit may elect to apply the new methodology and criteria. Replace "shall comply" with "may elect to demonstrate compliance with the seismic and geologic siting criteria in Subpart A or B to Part 100 of this Chapter." (Reference 43)

   Response. The NRC will address this request on a case-by-case basis rather than through a generic change to the regulations. This situation pertains to a limited number of facilities in various stages of construction. Some of the issues that must be addressed by the applicant and NRC during the operating license review include differences between the design bases derived from the current and amended regulations (Appendix A to Part 100 and Section 100.23, respectively), and earthquake engineering criteria such as, OBE design requirements and OBE shutdown requirements.

<u>(d)(1) Determination of the Safe Shutdown Earthquake Ground Motion</u>.

1. Determination of the SSE is based upon an evaluation that includes investigation of geological and seismological information and the results of a probabilistic seismic hazard analysis. Addressing uncertainties is an inherent part of the process.

   Based upon prior licensing decisions and scientific evaluations (Systematic Evaluation Program, Appendix A evaluations, LLNL, and EPRI) it seems reasonable to only perform detailed confirmatory site investigations (Regulatory Guide 1.132) at existing sites. Standardized 0.3g advanced plant designs are sufficiently robust to bound the seismic design attributes of all nuclear power plants at current sites. Inclusion of these simplified requirements for existing sites represents a significant step toward predictable and cost-effective licensing.

   Revise to read (substitution in italics): "Determination of the Safe Shutdown Earthquake Ground Motion. The Safe Shutdown Earthquake Ground Motion for the site is characterized by both horizontal spectra and vertical free-field ground motion response spectra at the free ground surface. *The Safe Shutdown Earthquake Ground Motion for the site is based upon the investigations required by paragraph (c) of this section and the results of a probabilistic seismic hazard analysis. Seismological and geological uncertainties are inherent in these determinations and are captured by the probabilistic analysis. Suitable sensitivity analyses may also be used to evaluate uncertainties.* Paragraph IV (a)(1) of Appendix S to Part 50 of this Chapter defines the minimum Safe Shutdown Earthquake Ground Motion for design. *Based upon prior scientific findings and licensing decisions at existing nuclear power plant sites east of the Rocky Mountain Front (east of approximately 105 west longitude), a 0.3g Standardized design level is acceptable at these sites given confirmatory foundation evaluations.*" (Reference 43)

   <u>Response</u>. (1) Determination of the Safe Shutdown Earth Ground Motion. Your recommended rewording is another way of saying the same thing, but places less emphasis on site-specific investigations relative to the PSHA than the current wording. We regard the current wording as better reflecting the proper priorities. Site specific investigations (regional and site geological, seismological, geophysical, and

geotechnical) are of prime importance in deriving the bases for the SSE. It must not be forgotten that if all of the data that is needed about a site to determine the SSE could be obtained through site-specific investigations, a PSHA would not be necessary. However, because of uncertainties, at the present time, more reliance must be placed on PSHA's than may be necessary in the future when more information is available.

Paragraph IV(a)(1) of Appendix S to Part 50. Investigations at most of the existing sites will more than likely be c[illegible]ry if the initial investigations were thorough, and there has n[illegible] too much lag time since the initial investigations were accompli[illegible]d and the results reviewed by the NRC. However, in many cases it may be necessary to carry out more extensive investigations than are usually considered as "confirmatory" investigations because: (1) the state-of-the-science is rapidly changing as new information is derived from every earthquake that occurs, and from ongoing research; (2) applicants may elect not to use the standard design plant and justify an SSE different than 0.3g; and (3) it will often be necessary, even for standard design sites, to determine a site-specific SSE as the design basis for other, non-standard design, safety-related structures, systems or components such as dams, reservoirs, intake and discharge facilities, etc.

The current wording in the proposed regulation most accurately represents the NRC staff's position on this issue.

2. Proposes that at existing eastern U.S. sites (rock or soil), or at eastern U.S. rock sites not located in areas of high seismicity (for example, Charleston, South Carolina, New Madrid, Missouri, Attica, New York) a 0.3g standardized ALWR design is acceptable and only evaluations of foundation conditions at the site are required (Regulatory Guide 1.132), but not geologic/geophysical seismological investigations. For other sites a DG-1032 review is required.

   Proposes that 10 CFR Part 100 Section 100.23(d)(1) be modified to reflect this consideration as follows:

   "Determination of the Safe Shutdown Earthquake Ground Motion. The Safe Shutdown Earthquake Ground Motion for the site is characterized by both horizontal and vertical free-field ground motion response spectra at the free ground surface. The Safe shutdown Earthquake Ground Motion for the site is based upon the investigations required by paragraph (c) of this section and the results of a probabilistic seismic hazard analysis. Seismological and geologic uncertainties are inherent in these determinations and are captured by the probabilistic analysis. Suitable sensitivity analyses may also be used to evaluate uncertainties. Paragraph IV(a) (1) of Appendix S to Part 50 of this Chapter defines the minimum Safe Shutdown Earthquake Ground Motion for design. Based upon prior scientific findings and licensing decisions at existing nuclear power plant sites east of the Rocky Mountain Front (east of approximately 105 west longitude) a 0.3g Standardized design level is acceptable at these sites given confirmatory foundation evaluations. For rock sites not in areas of known seismic activity including but not limited to the regions around New Madrid, MO, Charleston, SC, and Attica, New York, a 0.3g Standardized design level is acceptable given confirmatory foundation evaluations at the site." (Reference 44)

Response. Although some of the suggested wording may improve the readability of the text, the staff does not agree with the basic philosophy of the recommended modification for the following reasons:

1. The suggested modification brings back a prescriptive element which we have tried to eliminate in revising the siting document. It is more appropriate to include such a modification in Regulatory Guide 1.165 (Draft was DG-1032). The staff's position regarding the application of the 0.3g ALWR design is addressed in the main body of the draft guide, and in Appendix D.

2. A standard design of 0.3g does not preclude the need to conduct a thorough regional and site area investigation. The standard plant is designed for 0.3g, but other safety related components aren't part of the standard design plan. Such components include emergency cooling ponds and associated dams levees, spillways, etc., and they will have to be designed to the appropriate level based on regional and site geological, seismological, geophysical, and geotechnical investigations.

3. The level of investigations for a standard design plant or any additional unit sited on a previously validated site depends on when that site was previously validated, the complexity of the geology and seismology of the region and site, the advent of new information or hypotheses about regional tectonics, and the kinds of methods used and the thoroughness applied in using those

methods in the original investigations and analyses. The investigations can range anywhere between a literature review to a very extensive investigation program.

4. The discovery of the Meers Fault and the paleoseismic evidence for a large prehistoric earthquake in the Wabash Valley are examples in the central and eastern U.S. of the occurrences of events of great significance to the seismic hazard to those regions that were unknown until regional investigations were performed. Thus, we expect that evidence for similar, currently unknown tectonic structures or events is present in the CEUS.

Based on the above factors, the level of investigations could vary considerably, therefore, it would be inappropriate to make the modifications recommended.

## RESOLUTION OF COMMENTS ON APPENDIX S TO PART 50

### General Information

1. Mandate the retrofit of existing nuclear power plants in extremely active seismic zones with the most recent ASCE seismic design and engineering criteria. The requirements should be phased in a manner to take effect at individual reactors at the time of relicensing to ease the financial impact on the licensees. (Reference 45)

Response. This regulation is applicable to applicants for a design certification, combined license, construction permit or operating license on or after the effective date of the final rule. Because the requested change pertains to existing (operating) nuclear power plants it is beyond the scope of this rulemaking. The regulations pertaining to relicensing are contained in 10 CFR Part 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants." Further, If the NRC staff were to change the licensing bases for operating plants the burden would be on the staff to ensure that the backfit requirements stated in Section 50.109, "Backfitting," to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," are met.

2. There are several phrases that are used in the regulation that should be modified to make the regulation more stable from a licensing point of view. The following phrases and others that are similar in nature should be modified: (Reference 46)

2a. "... certain structures, systems, and components ..." should read: "... certain structures, systems, and components as identified in Regulatory Guides XXX ..." By referencing the regulatory guides, the vagueness of the statement is eliminated from the rule and the description of the structures, systems and components can be changed, if necessary, via changes to the regulatory guides."

Response. Regulatory guides are issued to describe and make available to the public such information as methods acceptable to the NRC staff for implementing specific parts of the Commission's regulations, techniques used by staff in evaluating specific problems or postulated accidents, and guidance to applicants. The Introduction section of the guide cites the applicable regulations pertaining to the guidance. Regulatory guides are not cited in regulations. The regulation was not changed.

2b. "... without loss of capability to perform their safety functions" should read: "... without loss of capability to perform their intended functions." The components perform a function and not a "safety" function -- components may be part of a safety system or a non-safety system. There are other sentences which have a similar phraseology -- for example, item c below. These sentences should be similarly modified.

Response. The term "safety function" is synonymous with terminology codified in other regulations; for example, General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena," of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50. The regulation was not changed.

2c. "The required safety functions of structures, systems, and components must be assured ..." should read: "The required functions of structures, systems, and components must be assured per the guidelines provided in Regulatory Guide XXX ..." The change shows that the regulatory guide contains guidance as to how a future license applicant can provide "assurance."

Response. See response to comments 2(a) and 2(b). The regulation was not changed.

Definitions

1. The parenthetical phrase in the definition of response spectrum should be changed to (acceleration, velocity, and displacement) [not "or" displacement]. Displacement is also involved in a response spectrum. (Reference 41)

Response. There are situations where it is only necessary for the response spectrum plot to show one of the three parameters depicted; for example, a plot of accelerations and frequencies. The definition was not changed.

Safe Shutdown Earthquake Ground Motion

1. Incorporate the seismic design and engineering criteria of ASCE Standard 4, "Seismic Analysis of Safety-Related Nuclear Structures and Commentary on Standard for Seismic Analysis of Safety-Related Nuclear Structures," into Part 100 to strengthen the basis for the requirements. (Reference 45)

Response. The supplemental information to the proposed regulations, item VB(2), "Remove Detailed Guidance from the Regulation," cites that the current regulation (Appendix A to 10 CFR Part 100) is too detailed, containing both requirements and guidance to satisfy the requirements. It further notes that having detailed assessments cast in a regulation has caused difficulty for applicants and the NRC staff in terms of inhibiting the use of needed latitude in judgement. Also, it has inhibited flexibility in applying basic principals to new situations and the use of evolving methods of analysis (for instance, probabilistic) in the licensing process. Therefore, the Commission has determined that new regulations will be more streamlined containing only basic requirements with guidance being provided in regulatory guides and, to some extent, in standard review plan sections. Therefore, it is common NRC practice not to reference publications such as ASCE Standard 4 (an analysis, not design standard) in its regulations. Rather, publications such as ASCE Standard 4 are cited in regulatory guides and standard review plan sections. ASCE Standard 4 is cited in the 1989 revision of Standard Review Plan Sections 3.7.1, 3.7.2, and 3.7.3.

Operating Basis Earthquake Ground Motion

1. Supports the NRC staff's position to not require explicit design analysis for the Operating Basis Earthquake Ground Motion (OBE) if its

peak acceleration is less than one-third of the Safe Shutdown Earthquake Ground Motion (SSE). The OBE for ABB-CE's System 80+™ is less than one-third of the SSE. The supporting analysis has already been reviewed and approved by the NRC staff in NUREG-1462, "Final Safety Evaluation Report Related to the Certification of the System 80+ Design." (Reference 47)

Surface Deformation

1. There is no definite indication of the type of deformation that must be considered. A clear distinction should be made between tectonic and non-tectonic deformation; and the design actions appropriate for both provided. (Reference 41)

   Response. The definition of surface deformation in Appendix S to 10 CFR Part 50 addresses tectonic surface deformation as a subset of surface deformation. Therefore, it is not necessary for the discussion in the regulation (Paragraph IV(b)) to distinguish between surface tectonic and nontectonic deformations. In addition, Section 100.23(d), "Geologic and Seismic Siting Factors," to 10 CFR Part 100 requires, in part, that the geologic and seismic siting factors considered for design include the potential for surface tectonic and nontectonic deformations.

   With regard to including a discussion on design actions appropriate for both surface tectonic and nontectonic deformations, the Commission has determined that new regulations will be more streamlined containing only basic requirements; guidance will be provided in regulatory guides and, to some extent, in standard review plan sections as appropriate. Therefore, design actions will not be provided in the regulation. The response to comment C1 contains additional discussion on the removal of detailed guidance from the regulation.

2. The required consideration of aftershocks is confusing and not needed. It has been recognized from early in the NRC's implementation of seismic design requirements that design for the SSE is more than adequate to account for any vibratory ground motion due to aftershocks. Alternatively, clarifying language should be added indicating aftershocks are fully considered in SSE design. (Reference 41)

   Response. The reference to aftershocks will be deleted. One of the changes to the Appendix A to Part 100, Safe Shutdown Earthquake requirements was the deletion of the phrase "including aftershocks."

The recommended change will make the aftershock requirements in Paragraphs IV(b), "Surface Deformation, and IV(a)(1), "Safe Shutdown Earthquake Ground Motion," of Appendix S to 10 CFR Part 50 consistent.

3. When surface deformation is identified as a hazard at a site, the determination of appropriate design parameters will specifically include a determination of its spatial characteristics. The requirement to postulate the occurrence of the load in any direction and azimuth and under any part of the nuclear plant is inappropriate, and should be removed. (Reference 41)

Response. The regulation specifically states if and how spatial characteristics for surface deformation must be considered in design. The same requirements are contained in Paragraph VI(b)(3) of Appendix A to Part 100 (effective December 1973). A technical justification stating why it is inappropriate to require the postulated occurrence of the load in any direction and azimuth and under any part of the nuclear plant was not provided. The regulation was not changed.

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REGULATORY GUIDE 1.12, REVISION 2
DRAFT WAS DG-1033

(SEISMIC INSTRUMENTATION)

REGULATORY GUIDE 1.12
(Draft was DG-1033)

# NUCLEAR POWER PLANT INSTRUMENTATION FOR EARTHQUAKES

## A. <u>INTRODUCTION</u>

In 10 CFR Part 20, "Standards for Protection Against Radiation," licensees are required to make every reasonable effort to maintain radiation exposures as low as is reasonably achievable. Paragraph IV(a)(4) of ~~Proposed~~ Appendix S, "Earthquake Engineering Criteria for Nuclear Power Plants," to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," ~~would~~ requires that suitable instrumentation must be provided so that the seismic response of nuclear power plant features important to safety can be evaluated promptly after an earthquake. Paragraph IV(a)(3) of ~~Proposed~~ Appendix S to 10 CFR Part 50 ~~would~~ requires shutdown of the nuclear power plant if vibratory ground motion exceeding that of the operating basis earthquake ground motion (OBE) occurs.[1]

This guide ~~is being developed to~~ describes seismic instrumentation acceptable to the NRC staff for satisfying the requirements of Parts 20 and ~~50 and the Proposed~~ Appendix S to Part 50.

~~Regulatory guides are issued to describe and make available to the public such information as methods acceptable to the NRC staff for implementing specific parts of the Commission's regulations, techniques used by staff in evaluating specific problems or postulated accidents, and guidance to applicants. Regulatory guides are not substitutes for regulations, and compliance with regulatory guides is not required. Regulatory guides are issued in draft form for public comment to involve the public in the early stages of developing the regulatory positions. Draft regulatory guides have not received complete staff review and do not represent official NRC staff positions.~~



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[1] ~~Guidance is being developed in Draft~~ Regulatory Guide ~~DG-1034~~ 1.166, "Pre- Earthquake Planning and Immediate Nuclear Power Plant Operator Postearthquake Actions," ~~on~~ provides criteria for plant shutdown.

Any information collection activities mentioned in this ~~draft~~ regulatory guide are contained as requirements in ~~the proposed amendments to~~ 10 CFR Part 50, which provides ~~that would provide~~ the regulatory basis for this guide. The ~~proposed amendments have been submitted to~~ information collection requirements in 10 CFR Part 50 have been approved by the Office of Management and Budget ~~for clearance that may be appropriate under the Paperwork Reduction Act. Such clearance, if obtained, would also apply to any information collection activities mentioned in this guide~~, Approval No. 3150-0011.

## B. DISCUSSION

When an earthquake occurs, it is important to take prompt action to assess the effects of the earthquake at the nuclear power plant. This assessment includes both an evaluation of the seismic instrumentation data and a plant walkdown. Solid-state digital time-history accelerographs installed at appropriate locations will provide time-history data on the seismic response of the free-field, containment structure, and other Seismic Category I structures. The instrumentation should be located so that a comparison and evaluation of such response may be made with the design basis and so that occupational radiation exposures associated with their location, installation, and maintenance are maintained as low as reasonably achievable (ALARA).

Instrumentation is provided in the free-field and foundation level and at elevation in Seismic Category I structures. Free-field instrumentation data ~~would~~ will be used to compare measured response to the engineering evaluations used to determine the design input motion to the structures and to determine whether the OBE has been exceeded (see ~~Draft~~ Regulatory Guide ~~DG-1034~~ 1.166). ~~Foundation level instrumentation would provide data on the actual seismic input to the containment and other buildings and would quantify differences between the vibratory ground motion at the free field and at the foundation level.~~ The instruments located at the foundation level and at elevation in the structures measure responses that are the input to the equipment or piping and ~~would~~ will be used in long-term evaluations (see ~~Draft~~ Regulatory Guide ~~DG-1035~~ 1.167, "Restart of a Nuclear Power Plant Shut Down by a Seismic Event"). Foundation-level instrumentation will provide data on the actual seismic input to the containment and other Seismic Category I structures and will be used to quantify differences between the vibratory ground motion at the free-field and at the foundation level. Instrumentation is not located on equipment, piping, or supports since experience has shown that data obtained at these locations are obscured by vibratory motion associated with normal plant operation.

The guidance ~~being developed~~ in ~~Draft~~ Regulatory Guide ~~DG-1034~~ 1.166 is based on the assumption that the nuclear power plant has operable seismic instrumentation, including the equipment and software needed to process the data within 4 hours after an earthquake. This is necessary to determine whether plant shut down is required. This determination will be made by comparing the recorded data against OBE exceedance criteria and the results of the plant walkdown inspections that take place within 8 hours of the event

It may not be necessary for identical nuclear power units on a given site to each be provided with seismic instrumentation if essentially the same seismic response at each of the units is expected from a given earthquake.

An evaluation of seismic instrumentation noted that instruments have been out of service during plant shutdown and sometimes during plant operation. The instrumentation system should be operable and operated at all times. If the seismic instrumentation or data processing hardware and software necessary to determine whether the OBE has been exceeded is inoperable, the guidelines in Appendix A to ~~Draft~~ Regulatory Guide ~~DG-1034~~ 1.166 ~~would~~ should be used.

The characteristics, installation, activation, remote indication, and maintenance of the instrumentation are described in this guide to help ensure (1) that the data provided are comparable with the data used in the design of the nuclear power plant, (2) that exceedance of the OBE can be determined, and (3) that the equipment will perform as required.

It is important that all of the significant ground motion associated with an [illegible] is recorded. This is accomplished by specifying how long before [illegible] the actuation of the seismic trigger the data should be recorded. [illegible] for the pre-event memory should be correlated with the maximum distance to an potential epicenter that can affect a specific site. The "P" wave may not be recorded at a 3 second setting. Also, when an event occurs at some distance and the trigger threshold limit is not exceeded until 15 or 20 seconds into the event, a part of the record, albeit for a low event, is lost. A 30 second value may be more appropriate and is within the capabilities of current digital time-history accelerographs at no aditional cost.

The appendix to this guide provides definitions to be used with this guidance.

~~Holders of an operating license or construction permit issued prior to the implementation date to be specified in the active guide may voluntarily implement the methods to be described in the active guide and the methods being developed in Draft Regulatory Guides DG 1034, "Pre Earthquake Planning and Immediate Nuclear Power Plant Operator Postearthquake Actions," and DG 1035, "Restart of a Nuclear Power Plant Shut Down by a Seismic Event."~~

## C. REGULATORY POSITION

The type, locations, operability, characteristics, installation, actuation, remote indication, and maintenance of seismic instrumentation described below are acceptable to the NRC staff for satisfying the requirements in 10 CFR Part 20~~, 10 CFR 50.65(b)(2),~~ and Paragraph IV(a)(4) of ~~Proposed~~ Appendix S to 10 CFR Part 50 for ensuring the safety of nuclear power plants.

### 1. SEISMIC INSTRUMENTATION TYPE AND LOCATION

1.1 Solid-state digital instrumentation that will enable the processing of data at the plant site within 4 hours of the seismic event should be used.

1.2 A triaxial time-history accelerograph should be provided at each of the following locations:

1. Free-field.

2. Containment foundation.

3. Two elevations (excluding the foundation) on a structure internal to the containment.

4. An independent Seismic Category I structure foundation where the response is different from that of the containment structure.

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5. An elevation (excluding the foundation) on the independent Seismic Category I structures selected in 4 above.

6. If seismic isolators are used, instrumentation should be placed on both the rigid and isolated portions of the same or an adjacent structure, as appropriate, at approximately the same elevations.

<u>1.3</u> The specific locations for instrumentation should be determined by the nuclear plant designer to obtain the most pertinent information consistent with maintaining occupational radiation exposures ALARA for the location, installation, and maintenance of seismic instrumentation. In general:

<u>1.3.1</u> The free-field sensors should be located and installed so that their recorded motion will be of the ground surface and the effects that are associated with ~~certain surface~~ features, buildings, and components ~~will be absent from on~~ the recorded ground motion will be insignificant.

<u>1.3.2</u> The in-structure instrumentation should be placed at locations that have been modeled as mass points in the building dynamic analysis so that the measured motion can be directly compared with the design spectra. The instrumentation should not be located on a secondary structural frame member that is not modeled as a mass point in the building dynamic model.

<u>1.3.3</u> A design review of the location, installation, and maintenance of proposed instrumentation for maintaining exposures ALARA should be performed by the facility in the planning stage in accordance with Regulatory Guide 8.8, "Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations Will Be As Low As Is Reasonably Achievable."

<u>1.3.4</u> Instrumentation should be placed in a location with as low a dose rate as is practical, consistent with other requirements.

1.3.5 Instruments should be selected to require minimal maintenance and in-service inspection, as well as minimal time and numbers of personnel to conduct installation and maintenance.

## 2. INSTRUMENTATION AT MULTI-UNIT SITES

Instrumentation in addition to that installed for a single unit will not be required if essentially the same seismic response is expected at the other units based on the seismic analysis used in the seismic design of the plant. However, if there are separate control rooms, annunciation should be provided to both control rooms as specified in Regulatory Position 7.1

## 3. SEISMIC INSTRUMENTATION OPERABILITY

The seismic instrumentation should operate during all modes of plant operation, including periods of plant shutdown. The maintenance and repair procedures should provide for keeping the maximum number of instruments in service during plant operation and shutdown.

## 4. INSTRUMENTATION CHARACTERISTICS

4.1 The design should include provisions for in-service testing. The instruments should be capable of periodic channel checks during normal plant operation.

4.2 The instruments should have the capability for in-place functional testing.

4.3 Instrumentation that has sensors located in inaccessible areas should contain provisions for data recording in an accessible location, and the instrumentation should provide an external remote alarm to indicate actuation.

4.4 ~~After actuation, the~~ **The** instrumentation should record**, at a minimum,** ~~the~~ 3 seconds of low amplitude motion prior to seismic trigger actuation, continue to record the motion during the period in which the earthquake motion exceeds the seismic trigger threshold, and continue to record low amplitude motion for a minimum of 5 seconds beyond the last exceedance of the seismic trigger threshold.

4.5 The instrumentation should be capable of recording 25 minutes of sensed motion.

4.6 The battery should be of sufficient capacity to power the instrumentation ~~and~~ to sense and record (see Regulatory Position 4.5) 25 minutes of motion~~, with no battery charger,~~ over a period of not less than the channel check test interval (Regulatory Position 8.2). This can be accomplished by providing enough battery capacity for a minimum of 25 minutes of system operation at any time over a 24 hour period, without recharging, in combination with a battery charger whose line power is connected to an uninteruptable power supply or a line source with an alarm that is checked, at least every 24 hours. Other combinations of larger battery capacity and alarm intervals may be used.

4.7 Acceleration Sensors

4.7.1 The dynamic range should be 1000:1 zero to peak, or greater; for example, 0.001g to 1.0g.

4.7.2 The frequency range should be 0.20 Hz to 50 Hz or an equivalent demonstrated to be adequate by computational techniques applied to the resultant accelerogram.

4.8 Recorder

4.8.1 The sample rate should be at least 200 samples per second in each of the three directions.

4.8.2 The bandwidth should be at least from 0.20 Hz to 50 Hz.

4.8.3 The dynamic range should be 1000:1 or greater and be able to record at least 1.0g ~~0~~ zero to peak.

4.9 **Seismic Trigger.** The actuating level should be adjustable and within the range of 0.001g to 0.02g.

5. INSTRUMENTATION INSTALLATION

5.1 The instrumentation should be designed and installed so that the mounting is rigid.

5.2 The instrumentation should be oriented so that the horizontal axes are parallel to the orthogonal horizontal axes assumed in the seismic analysis.

5.3 Protection against accidental impacts should be provided.

6. INSTRUMENTATION ACTUATION

6.1 Both vertical and horizontal input vibratory ground motion should actuate the same time-history accelerograph. One or more seismic triggers may be used to accomplish this.

6.2 Spurious triggering should be avoided.

6.3 The seismic trigger mechanisms of the time-history accelerograph should be set for a threshold ground acceleration of not more than 0.02g.

7. REMOTE INDICATION

~~Activation~~ triggering of the free-field or any foundation-level time-history accelerograph should be annunciated in the control room. If there is more than one control room at the site, annunciation should be provided to each control room.

8. MAINTENANCE

8.1 The purpose of the maintenance program is to ensure that the equipment will perform as required. As stated in Regulatory Position 3, the maintenance and repair procedures should provide for keeping the maximum number of instruments in service during plant operation and shutdown.

8.2 Systems are to be given channel checks every 2 weeks for the first 3 months of service after startup. Failures of devices normally occur during initial operation. After the initial 3-month period and 3 consecutive successful checks, monthly channel checks are sufficient. The monthly channel check is to include checking the batteries. The channel functional test should be performed every 6 months. Channel calibration should be performed during each refueling outage at a minimum.

## D. IMPLEMENTATION

The purpose of this section is to provide guidance to applicants and licensees regarding the NRC staff's plans for using this regulatory guide. ~~This proposed revision has been released to encourage public participation in its development.~~ Except in those cases in which the applicant proposes an acceptable alternative method for complying with the specified portions of the Commission's regulations, ~~the method to be described in the active~~ this guide ~~reflecting public comments~~ will be used in the evaluation of applications for construction permits, operating licenses, combined licenses, or design certification submitted after ~~the implementation date to be specified in the active guide~~ EFFECTIVE DATE OF THE FINAL RULE. This guide ~~would~~ will not be used in the evaluation of an application for an operating license submitted after ~~the implementation date to be specified in the active guide~~ EFFECTIVE DATE OF THE FINAL RULE if the construction permit was issued prior to that date.

[redacted] an operating license or construction permit issued prior to EFFECTIV[redacted]F THE FINAL RULE may voluntarily implement the methods described in this guide in combination with the methods in Regulatory Guides 1.166, "Pre-Earthquake Planning and Immediate Nuclear Power Plant Operator Postearthquake Actions," and 1.167, "Restart of a Nuclear Power Plant Shut Down by a Seismic Event." Other implementation strategies, such as a voluntary implementation of portions of the cited regulatory guides, will be evaluated by the NRC staff on a case-by-case basis.

# APPENDIX

## DEFINITIONS

Acceleration Sensor. An instrument capable of sensing absolute acceleration and transmitting the data to a recorder.

Accessible Instruments. Instruments or sensors whose locations permit ready access during plant operation without violation of applicable safety regulations, such as Occupational Safety and Health Administration (OSHA), or regulations dealing with plant security or radiation protection safety.

Channel Calibration (Primary Calibration). The determination and, if required, adjustment of an instrument, sensor, or system such that it responds within a specific range and accuracy to an acceleration, velocity, or displacement input, as applicable, or responds to an acceptable physical constant.

Channel Check. The qualitative verification of the functional status of the instrument sensor. This check is an "in-situ" test and may be the same as a channel functional test.

Channel Functional Test (Secondary Calibration). The determination without adjustment that an instrument, sensor, or system responds to a known input of such character that it will verify the instrument, sensor, or system is functioning in a manner that can be calibrated.

Containment - See Primary Containment and Secondary Containment.

Nonaccessible Instruments. Instruments or sensors in a location that does not permit ready access during plant operation because of a risk of violating applicable plant operating safety regulations, such as OSHA, or regulations dealing with plant security or radiation protection safety.

Operating Basis Earthquake Ground Motion (OBE). The vibratory ground motion for which those features of the nuclear power plant necessary for continued operation without undue risk to the health and safety of the public will remain functional. The value of the OBE is set by the applicant.

specified portions of the Commission's regulations, the method ~~to be~~ described in ~~the active~~ [redacted] guide ~~reflecting public comments~~ will be used in the evaluation of applications for construction permits, operating licenses, combined licenses, or design certification submitted after ~~the implementation date to be specified in the active guide~~ EFFECTIVE DATE OF THE FINAL RULE. This guide ~~would~~ will not be used in the evaluation of an application for an operating license submitted after ~~the implementation date to be specified in the active guide~~ EFFECTIVE DATE OF THE FINAL RULE if the construction permit was issued prior to that date.

Holders of an operating license or construction permit issued prior to EFFECTIVE DATE OF THE FINAL RULE may voluntarily implement the methods described in this guide in combination with the methods in Regulatory Guides 1.12, "Nuclear Power Plant Instrumentation for Earthquakes," Revision 2, and 1.167, "Restart of a Nuclear Power Plant Shut Down by a Seismic Event." Other implementation strategies, such as a voluntary implementation of portions of the cited regulatory guides, will be evaluated by the NRC staff on a case-by-case basis.

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2. Plan views and vertical sections showing the location of each seismic instrument and the orientation of the instrument axis with respect to a plant reference axis.

3. A complete service history of each seismic instrument. The service history should include information such as dates of servicing, description of completed work, and calibration records and data (where applicable). The documentation and retention of these data should be commensurate with the recordkeeping for other plant equipment.

4. A suitable earthquake time-history (e.g., the October 1987 Whittier, California, earthquake) or manufacture's calibration standard and the corresponding response spectrum and cumulative absolute velocity (CAV) (see Regulatory Positions 4.1 and 4.2). ~~The response spectrum and CAV should be calculated after~~ After the initial installation and each servicing of the free-field instrumentation the response spectrum and CAV should be calculated and filed (see Regulatory Position 4.3).

1.2 <u>Planning for Postearthquake Inspections</u>

The pre-earthquake actions, that is, the selection of equipment and structures for inspections, and the content of the baseline inspections as described in Sections 5.3.1 ~~and 5.3.2.1~~ of EPRI NP-6695, "Guidelines for Nuclear Plant Response to an Earthquake," are acceptable to the NRC staff for satisfying the ~~proposed~~ requirements in Paragraph IV(a)(3) of ~~Proposed~~ Appendix S to 10 CFR Part 50 for ensuring the safety of nuclear power plants.

2. <u>IMMEDIATE POSTEARTHQUAKE ACTIONS</u>

The guidelines for immediate postearthquake actions specified in Sections 4.3.1 (with the exception specified below) and 4.3.2 ~~(including Section 5.3.2.1 and items 7 and 8 of Table 5.1)~~ of EPRI NP-6695 are acceptable to the NRC staff for satisfying the requirements ~~proposed~~ in Paragraph IV(a)(3) of ~~Proposed~~ Appendix S to 10 CFR Part 50.

In Section 4.3.1, a check of the neutron flux monitoring sensors for changes should be added to the specific control room board checks.

## 3. EVALUATION OF GROUND MOTION RECORDS

### 3.1 Data Identification

A record collection log should be maintained at the plant, and all data should be identifiable and traceable with respect to:

1. The date and time of collection,

2. The make, model, serial number, location, and orientation of the instrument (sensor) from which the record was collected.

### 3.2 Data Collection

3.2.1 Only personnel trained in the operation of the instrument should collect the data.

3.2.2 The steps for removing and storing records from each seismic instrument should be planned and performed in accordance with established procedures.

3.2.3 Extreme caution should be exercised to prevent accidental damage to the recording media and instruments during data collection and subsequent handling.

3.2.4 As data are collected and the instrumentation is inspected, notes should be made regarding the condition of the instrument and its installation, for example, instrument flooded, mounting surface tilted, fallen objects that struck the instrument or the instrument mounting surface.

3.2.5 For validation of the collected data, the information described in Regulatory Position 1.1(4) should be ~~added to the record without affecting the previously recorded data provided~~.

3.2.6 If the instrument's operation appears to have been normal, the instrument should remain in service without readjustment or change that would defeat attempts to obtain postevent calibration.

### 3.3 Record Evaluation

Records should be analyzed according to the manufacturer's specifications and the results of the analysis should be evaluated. Any record anomalies, invalid data, and nonpertinent signals should be noted, along with any known causes.

## 4. DETERMINING OBE EXCEEDANCE

The evaluation to determine whether the OBE was exceeded should be performed using data obtained from the three components of the free-field ground motion (i.e., two horizontal and one vertical). The evaluation may be performed on uncorrected earthquake records. It was found in a study of uncorrected versus corrected earthquake records (see EPRI NP-5930) that the use of uncorrected records is conservative. The evaluation should consist of a check of the response spectrum~~, and~~ CAV~~ limit~~, and the operability of the instrumentation. This evaluation should take place within 4 hours of the earthquake.

### 4.1 Response Spectrum Check

4.1.1

The OBE response spectrum check is performed using the lower of:

1. The spectrum used in the certified standard design, or

2. A spectrum other than (1) used in the design of any Seismic Category I structure.

4.1.2

The OBE response spectrum is exceeded if any one of the three components (two horizontal and one vertical) of the 5 percent damped free-field ground motion response spectra is larger than:

1. The corresponding design response spectral acceleration (OBE spectrum if used, otherwise 1/3 of the safe shutdown earthquake (SSE) spectrum) or 0.2g, whichever is greater, for frequencies between 2 to 10 Hz, or

2. The corresponding design response spectral velocity (OBE spectrum if used, otherwise 1/3 of the SSE spectrum) or a spectral velocity of 6 inches per second (15.24 centimeters per second), whichever is greater, for frequencies between 1 and 2 Hz.

4.2 Cumulative Absolute Velocity (CAV) ~~Limit Check~~

For each component of the free-field ground motion, the CAV should be calculated as follows: (1) the absolute acceleration (g units) time-history is divided into 1-second intervals, (2) each 1-second interval that has at least 1 exceedance of 0.025g is integrated over time, (3) all the integrated values are summed together to arrive at the CAV. The CAV ~~limit check~~ is exceeded if any CAV calculation is greater than 0.16 g-second. Additional information on how to determine the CAV is provided in EPRI TR-100082.

4.3 Instrument Operability Check

After an earthquake at the plant site, the response spectrum and CAV should be calculated using the same input as that used in ~~the calibration standard (see~~ Regulatory Position 1.1(4)~~)~~ and the results should be compared with the latest filed data to demonstrate that the time-history analysis hardware and software were functioning properly. The results of this compari[redacted]ld be reported to the NRC.

4.4 Inoperable Instrumentation or Data Processing Hardware or Software

If the response spectrum and the CAV (Regulatory Positions 4.1 and 4.2) can not be obtained because the seismic instrumentation is inoperable, data from the instrumentation are destroyed, or the data processing hardware or software is inoperable, the criteria in Appendix A to this guide should be used to determine whether the OBE has been exceeded.

## 5. CRITERIA FOR PLANT SHUTDOWN

If the OBE is exceeded or significant plant damage occurs, the plant must be shut down unless a plan for the timely, safe shutdown of the nuclear power plan' has been proposed by the licensee and accepted by the NRC staff.

### 5.1 OBE Exceedance

If the response spectrum check and the CAV ~~limit~~ check (performed or calculated in accordance with Regulatory Positions 4.1 and 4.2) were exceeded, the OBE was exceeded and plant shutdown is required. If either ~~limit~~ check does not exceed the criterion, the earthquake motion did not exceed the OBE. If only one ~~limit~~ check can be ~~checked~~ performed, the other ~~limit~~ check is assumed to be exceeded; if neither check can be performed see Regulatory Position 4.4. The determination of whether or not the OBE has been exceeded should be performed even if the plant automatically trips off-line as a result of the earthquake.

### 5.2 Damage

The plant should be shut down if the walkdown inspections performed in accordance with Regulatory Position 2 discover damage. This evaluation should take place within 8 hours of the earthquake occurrence.

### 5.3 Continued Operation

If the OBE was not exceeded and the walkdown inspection indicates no damage to the nuclear power plant, shutdown of the plant is not required. The plant may continue to operate (or restart following a post-trip review, if it tripped off-line because of the earthquake).

## 6. PRE-SHUTDOWN INSPECTIONS

The pre-shutdown inspections described in Section 4.3.4 ~~(including all subsections)~~ of EPRI NP-6695, "Guidelines for Nuclear Plant Response to an Earthquake," with the exceptions specified below are acceptable to the NRC staff for satisfying the requirements ~~proposed~~ in Paragraph IV(a)(3) of

~~Proposed~~ Appendix S to 10 CFR Part 50 for ensuring the safety of nuclear power plants.

6.1 <u>Shutdown Timing</u>

Delete the last sentence in the first paragraph of Section 4.3.4.

6.2 <u>Safe Shutdown Equipment</u>

In Section 4.3.4.1, a check of the containment isolation system should be added to the minimum list of equipment to be inspected.

6.3 <u>Orderly Plant Shutdown</u>

The following paragraph in Section 4.3.4 of EPRI NP-6695 is printed here to emphasize that the plant should shut down in an orderly manner.

> "Prior to initiating plant shutdown following an earthquake, visual inspections and control board checks of safe shutdown systems should be performed by plant operations personnel, and the availability of off-site and emergency power sources should be determined. The purpose of these inspections is to determine the effect of the earthquake on essential safe shutdown equipment which is not normally in use during power operation so that any resets or repairs required as a result of the earthquake can be performed, or alternate equipment can be readied, prior to initiating shutdown activities. In order to ascertain possible fuel and reactor internal damage, the following checks should be made, if possible, before plant shutdown is initiated . . . . "

## D. <u>IMPLEMENTATION</u>

The purpose of this section is to provide guidance to applicants and licensees regarding the NRC staff's plans for using this regulatory guide.

~~This proposed revision has been released to encourage public participation in its development.~~ Except in those cases in which the applicant proposes an acceptable alternative method for complying with the

Primary Containment. The principal structure of a unit that acts as the barrier, after the fuel cladding and reactor pressure boundary, to control the release of radioactive material. The primary containment includes (1) the containment structure and its access openings, penetrations, and appurtenances, (2) the valves, pipes, closed systems, and other components used to isolate the containment atmosphere from the environment, and (3) those systems or portions of systems that, by their system functions, extend the containment structure boundary (e.g., the connecting steam and feedwater piping) and provide effective isolation.

Recorder. An instrument capable of simultaneously recording the data versus time from an acceleration sensor or sensors.

Secondary Containment. The structure surrounding the primary containment that acts as a further barrier to control the release of radioactive material.

Seismic Isolator. A device (for instance, laminated elastomer and steel) installed between the structure and its foundation to reduce the acceleration of the isolated structure, as well as the attached equipment and components.

Seismic Trigger. A device that starts the time-history accelerograph.

Time-History Accelerograph. An instrument capable of sensing and permanently recording the absolute acceleration versus time. The components of the time-history accelerograph (acceleration sensor, recorder, seismic trigger) may be assembled in a self-contained unit or may be separately located.

Triaxial. Describes the function of an instrument or group of instruments in three mutually orthogonal directions, one of which is vertical.

REGULATORY ANALYSIS

A separate regulatory analysis was not prepared for this regulatory guide. The ~~draft~~ regulatory analysis, "~~Proposed~~ Revision of 10 CFR Part 100 and 10 CFR Part 50," was prepared for the ~~proposed~~ amendments, and it provides the regulatory basis for this guide and examines the costs and benefits of the rule as implemented by the guide. A copy of the ~~draft~~ regulatory analysis is available for inspection and copying for a fee at the NRC Public Document Room, 2120 L Street NW. (Lower Level), Washington, DC, as ~~Enclosure 2 to Secy 94 194~~ LATER.

<u>APPENDIX A</u>

INTERIM OPERATING BASIS EARTHQUAKE EXCEEDANCE GUIDELINES

This regulatory guide is based on the assumption that the nuclear power plant has operable seismic instrumentation and equipment (hardware and software) to process the data. If the seismic instrumentation or data processing equipment is inoperable, the following should be used to determine whether the operating basis earthquake ground motion (OBE) has been exceeded:

1. For plants at which instrumentally determined data are available only from an instrument installed on a foundation, the cumulative absolute velocity (CAV) ~~limit check~~ (see Regulatory Position 4.2 of this guide) is not applicable. In this case, the determination of OBE exceedance is based on a response spectrum check similar to that described in Regulatory Position 4.1 of this regulatory guide. A comparison is made between the foundation-level design response spectra and data obtained from the foundation-level instruments. If the response spectrum check at any foundation is exceeded, the OBE is exceeded and the plant must be shut down. At this instrument location it is inappropriate to use the 0.2g spectral acceleration limit or the 6 inches per second (15.24 centimeters per second) spectral velocity limit stated in Regulatory Position 4.1.2.

2. For plants at which no free-field or foundation-level instrumental data are available, <u>or</u> the data processing equipment is inoperable and the response spectrum check and the CAV ~~limit check~~ can not be determined (Regulatory Positions 4.1 and 4.2), the OBE will be considered to have been exceeded and the plant must be shut down if one of the following applies:

    1. The earthquake resulted in Modified Mercalli Intensity (MMI) VI or greater within 5 km of the plant,

    2. The earthquake was felt within the plant and was of magnitude 6.0 or greater, or

3. The earthquake was of magnitude 5.0 or greater and occurred within 200 km of the plant.

A postearthquake plant walkdown should be conducted (see Regulatory Position 2 of this guide).

If plant shutdown is warranted under the above guidelines, the plant should be shut down in an orderly manner (see Regulatory Position 6 of this guide).

Note: The determinations of epicentral location, magnitude, and intensity by the U.S. Geological Survey, National Earthquake Information Center, will usually take precedence over other estimates; however, regional and local determinations will be used if they are considered to be more accurate. Also, higher quality damage reports or a lack of damage reports from the nuclear power plant site or its immediate vicinity will take precedence over more distant reports.

# APPENDIX B

## DEFINITIONS

Certified Standard Design. A Commission approval, issued pursuant to Subpart B of 10 CFR Part 52, of a standard design for a nuclear power facility.

Design Response Spectra. Response spectra used to design Seismic Category I structures, systems, and components.

Operating Basis Earthquake Ground Motion (OBE). The vibratory ground motion for which those features of the nuclear power plant necessary for continued operation without undue risk to the health and safety of the public will remain functional. The value of the OBE is set by the applicant.

Spectral Acceleration. The acceleration response of a linear oscillator with prescribed frequency and damping.

Spectral Velocity. The velocity response of a linear oscillator with prescribed frequency and damping.

<u>REGULATORY ANALYSIS</u>

A separate regulatory analysis was not prepared for this regulatory guide. The ~~draft~~ regulatory analysis, "~~Proposed~~ Revisions of 10 CFR Part 100 and 10 CFR Part 50," was prepared for the ~~proposed~~ amendments, and it provides the regulatory basis for this guide and examines the costs and benefits of the rule as implemented by the guide. A copy of the ~~draft~~ regulatory analysis is available for inspection and copying for a fee at the NRC Public Document Room, 2120 L Street NW. (Lower Level), Washington, DC, as ~~Enclosure 2 to Secy-94-194~~ LATER.

REGULATORY GUIDE 1.166
DRAFT WAS DG-1034

(PLANT SHUTDOWN)

REGULATORY GUIDE 1.166
(Draft was DG-1034)

# PRE-EARTHQUAKE PLANNING AND IMMEDIATE NUCLEAR POWER PLANT OPERATOR POSTEARTHQUAKE ACTIONS

## A. INTRODUCTION

Paragraph IV(a)(4) of ~~Proposed~~ Appendix S, "Earthquake Engineering Criteria for Nuclear Power Plants," to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," ~~would~~ requires that suitable instrumentation[1] be provided so that the seismic response of nuclear power plant features important to safety can be evaluated promptly. Paragraph IV(a)(3) of ~~Proposed~~ Appendix S to 10 CFR Part 50 ~~would~~ requires shutdown of the nuclear power plant if vibratory ground motion exceeding that of the operating basis earthquake ground motion (OBE) or significant plant damage occurs. If systems, structures, or components necessary for the safe shutdown of the nuclear power plant ~~would~~ are not ~~be~~ available after occurrence of the OBE, the licensee ~~would be required to~~ must consult with the NRC and must propose a plan for the timely, safe shutdown of the nuclear power plant. ~~Proposed~~ Paragraph 50.54(ff) to 10 CFR Part 50 ~~would~~ requires licensees of nuclear power plants that have adopted the earthquake engineering criteria in ~~Proposed~~ Appendix S to 10 CFR Part 50 to shut down the plant if the criteria in Paragraph IV(a)(3) of ~~Proposed~~ Appendix S are exceeded.

This guide ~~is being developed to~~ provides guidance acceptable to the NRC staff for a timely evaluation after an earthquake of the recorded instrumentation data and for determining whether plant shutdown ~~would be~~ is required by ~~the proposed amendments to~~ 10 CFR Part 50.

~~Regulatory guides are issued to describe and make available to the public such information as methods acceptable to the NRC staff for implementing specific parts of the Commission's regulations, techniques used by the staff in evaluating specific problems or postulated accidents, and guidance to applicants. Regulatory guides are not substitutes for regulations, and compliance with regulatory guides is not required. Regulatory guides are issued in draft form for public comment to involve the public in the early~~

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[1] ~~Guidance is being developed in Draft Regulatory Guide DG-1033, the Third Proposed Revision 2 to~~ Regulatory Guide 1.12, "Nuclear Power Plant Instrumentation for Earthquakes," Revision 2, ~~to~~ describes seismic instrumentation acceptable to the NRC staff.

130

~~stages of developing the regulatory positions. Draft regulatory guides have not received complete staff review and do not represent official NRC staff positions.~~

Any information collection activities mentioned in this ~~draft~~ regulatory guide are contained as requirements in ~~the proposed amendments to~~ 10 CFR Part 50 ~~that would provide~~ , which provides the regulatory basis for this guide. The ~~proposed amendments have been submitted to~~ information collection requirements in 10 CFR Part 50 have been approved by the Office of Management and Budget ~~for clearance that may be appropriate under the Paperwork Reduction Act. Such clearance, if obtained, would also apply to any information collection activities mentioned in this guide~~, Approval No. 3150-0011.

## B. DISCUSSION

When an earthquake occurs, ground motion data are recorded by the seismic instrumentation.[1] These data are used to make a rapid determination of the degree of severity of the seismic event. The data from the nuclear power plant's free-field seismic instrumentation, coupled with information obtained from a plant walkdown, are used to make the initial determination of whether the plant must be shut down, if it has not already been shut down by operational perturbations resulting from the seismic event. If on the basis of these initial evaluations (instrumentation data and walkdown) it is concluded that the plant shutdown criteria have not been exceeded, it is presumed that the plant will not be shut down (or could restart following a post-trip review, if it tripped off-line because of the earthquake). Guidance ~~is being developed~~ on postshutdown inspections and plant restart~~;~~ is contained [redacted] ~~see Draft~~ Regulatory Guide ~~DG-1035,~~ 1.167, "Restart of a Nuclear Power Plant Shut Down by a Seismic Event." The Electric Power Research Institute has developed guidelines that will enable licensees to quickly identify and assess earthquake effects on nuclear power plants. These guidelines are in EPRI NP-5930, "A Criterion for Determining Exceedance of the Operating Basis Earthquake," July 1988[2]; EPRI NP-6695, "Guidelines for Nuclear Plant Response to an Earthquake," December 1989[2]; and EPRI TR-100082, "Standardization of Cumulative Absolute Velocity," December 1991.[2]

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[2]EPRI reports may be obtained from the Electric Power Research Institute, Research Reports Center, P.O. Box 50490, Palo Alto, CA 94303

This regulatory guide is based on the assumption that the nuclear power plant has operable seismic instrumentation, including the computer equipment and software required to process the data within 4 hours after an earthquake. This is necessary because the decision to shut down the plant will be made, in part, by comparing the recorded data against OBE exceedance criteria. The decision to shut down the plant is also based on the results of the plant walkdown inspections that take place within 8 hours of the event. If the seismic instrumentation or data processing equipment is inoperable, the guidelines in Appendix A to this guide would be used to determine whether the OBE has been exceeded.

Because free-field seismic instrumentation data are used in the plant shutdown determination, it is important to ascertain that the time-history analysis hardware and software were functioning properly. Therefore, the response spectrum and cumulative absolute velocity (CAV) should be calculated using a suitable earthquake time-history or manufacturer's calibration standard after the initial installation and each servicing of the free-field instrumentation. After an earthquake at the plant site, the response spectrum and CAV should be calculated using the time-history or calibration standard that was used during the last servicing (or initial instrumentation installation if no servicing has been performed) and the results compared with the latest data on file at the plant.

Because earthquake-induced vibration of the reactor vessel could lead to changes in neutron fluxes, a prompt check of the neutron flux monitoring sensors would provide an indication that the reactor is stable.

Shutdown of the nuclear power plant ~~would be~~ is required if the vibratory ground motion experienced exceeds that of the OBE. ~~Two criteria~~ A criterion for determining exceedance of the OBE (based on data recorded in the free-field) ~~are~~ is provided in EPRI NP-5930: a threshold response spectrum ordinate ~~criterion~~ check and a cumulative absolute velocity (CAV) ~~criterion~~ check. Seismic Category I structures at the nuclear power plant site may be designed using different ground motion response spectra; for example, one used for the certified standard design and another for site-specific applications. The spectrum ordinate criterion is based on the lowest spectrum used in the design of the Seismic Category I structures. A procedure to standardize the calculation of the CAV is provided in EPRI TR-100082. A spectral velocity threshold has also been recommended by EPRI since some structures have fundamental frequencies below the range specified in EPRI NP-5930. The NRC

5 132

staff now recommends 1.0 to 2.0 Hz for the range of the spectral velocity limit since some structures have fundamental frequencies below 1.5 Hz. ~~The former range was~~ This is instead of the 1.5 to 2.0 Hz range proposed by EPRI.

Since the containment isolation valves may have malfunctioned during an earthquake, inspection of the containment isolation system is necessary to ensure continued containment integrity.

The NRC staff does not endorse the philosophy discussed in EPRI NP-6695, Section 4.3.4 (first paragraph, last sentence), pertaining to plant shutdown considerations following an earthquake based on the need for continued power generation in the region. If the licensee determines that plant shutdown is required by the NRC's regulations, but the licensee does not consider it prudent to do so, the licensee would be required to consult with the NRC and propose a plan for the timely, safe shutdown of the nuclear power plant.

Appendix B to this guide provides definitions to be used with this guidance.

~~Holders of an operating license or construction permit issued prior to the implementation date to be specified in the active guide may voluntarily implement the methods to be described in the active guide and the methods being developed in Draft Regulatory Guides DG 1033, "Nuclear Power Plant Instrumentation for Earthquakes," and DG 1035, "Restart of a Nuclear Power Plant Shut Down by a Seismic Event."~~

## C. REGULATORY POSITION

### 1. BASE-LINE DATA

#### 1.1 Information Related to Seismic Instrumentation

A file containing information on all the seismic instrumentation should be kept at the plant. The file should include:

1. Information on each instrument type such as make, model, and serial number; manufacturers' data sheet; list of special features or options; performance characteristics; examples of typical instrumentation readings and interpretations; operations and maintenance manuals; repair procedures (manufacturers' recommendations for repairing common problems); and a list of any special requirements, e.g., maintenance, operational, installation.

REGULATORY GUIDE 1.167
DRAFT WAS DG-1035

(PLANT RESTART)

REGULATORY GUIDE 1.167
(Draft was DG-1035)

# RESTART OF A NUCLEAR POWER PLANT SHUT DOWN BY A SEISMIC EVENT

## A. INTRODUCTION

Paragraph IV(a)(3) of ~~Proposed~~ Appendix S, "Earthquake Engineering Criteria for Nuclear Power Plants," to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," ~~would~~ requires shutdown of the nuclear power plant if vibratory ground motion exceeding that of the operating basis earthquake ground motion (OBE) occurs or if significant plant damage occurs.[1] Prior to resuming operations, the licensee must demonstrate to the NRC that no functional damage has occurred to those features necessary for continued operation without undue risk to the health and safety of the public.

This guide ~~is being developed to~~ provides guidance acceptable to the NRC staff for performing inspections and tests of nuclear power plant equipment and structures prior to restart of a plant that has been shut down by a seismic event.

~~Regulatory guides are issued to describe and make available to the public such information as methods acceptable to the NRC staff for implementing specific parts of the Commission's regulations, techniques used by the staff in evaluating specific problems or postulated accidents, and guidance to applicants. Regulatory guides are not substitutes for regulations, and compliance with regulatory guides is not required. Regulatory guides are issued in draft form for public comment to involve the public in the early stages of developing the regulatory positions. Draft regulatory guides have not received complete staff review and do not represent official NRC staff positions.~~

Any information collection activities mentioned in this ~~draft~~ regulatory guide are contained as requirements in ~~the proposed amendments to~~ 10 CFR Part 50 ~~that would provide~~ , which provides the regulatory basis for this guide. The ~~proposed amendments have been submitted to~~ information collection requirements in 10 CFR Part 50 have been approved by the Office of Management

134

[1] ~~Guidance is being developed in Draft~~ Regulatory Guide ~~DG-1034~~ 1.166, "Pre-Earthquake Planning and Immediate Nuclear Power Plant Operator Postearthquake Actions," ~~to~~ provides criteria for plant shutdown.

and Budget ~~for clearance that may be appropriate under the Paperwork Reduction Act. Such clearance, if obtained, would also apply to any information collection activities mentioned in this guide~~, Approval No. 3150-0011.

## B. DISCUSSION

Data from seismic instrumentation[2] and a walkdown of the nuclear power plant are used to make the initial determination of whether the plant must be shut down after an earthquake, if the plant has not already shut down from operational perturbations resulting from the seismic event.[1]

The Electric Power Research Institute has developed guidelines that will enable licensees to quickly identify and assess earthquake effects on nuclear power plants in EPRI NP-6695, "Guidelines for Nuclear Plant Response to an Earthquake,"[3] December 1989. This regulatory guide addresses sections of EPRI NP-6695 that relate to postshutdown inspection and tests, inspection criteria, inspection personnel, documentation, and long-term evaluations.

EPRI NP-6695 has been supplemented to add inspections and tests as a basis for acceptance of stresses in excess of Service Level C and to recommend that engineering evaluations of components with calculated stresses in excess of service Level D focus on areas of high stress and include fatigue analyses.

~~Holders of an operating license or construction permit issued prior to the implementation date to be specified in the active guide may voluntarily implement the methods to be described in the active guide and the methods being developed in Draft Regulatory Guides DG 1033, "Nuclear Power Plant Instrumentation for Earthquakes," and DG 1034, "Pre Earthquake Planning and Immediate Nuclear Power Plant Operator Postearthquake Action."~~

## C. REGULATORY POSITION

After a plant has been shut down by an earthquake, the guidelines for inspections and tests of nuclear power plant equipment and structures that are

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[2]~~Guidance is being developed in Draft~~ Regulatory Guide ~~DG 1033~~ 1.12, ~~the third Proposed Revision 2 to Regulatory Guide 1.12,~~ "Nuclear Power Plant Instrumentation for Earthquakes," Revision 2, ~~that will~~ describes seismic instrumentation acceptable to the NRC staff.

[3]EPRI reports may be obtained from the Electric Power Research Institute, Research Reports Center, P.O. Box 50490, Palo Alto, CA 94303.

depicted in EPRI NP-6695 in Figure 3-2 and specified in Sections 5.3.2 ~~(including Tables 2-1, 2-2, and 5-1)~~, 5.3.3 ~~(includes Table 5-1)~~, and 5.3.4; the documentation to be submitted to the NRC specified in Section in 5.3.5; and the long-term evaluations that are specified in Section 6.3 ~~(all sections and subsections)~~, with the exceptions specified below, ~~would be~~ are acceptable to the NRC staff for satisfying the requirements ~~proposed~~ in Paragraph IV(a)(3) of ~~the Proposed~~ Appendix S to 10 CFR Part 50.

1. EXCEPTIONS TO SECTION 6.3.4.1 OF EPRI NP-6695

1.1 Item (1) should read:

If the calculated stresses from the actual seismic loading conditions are less than the allowables for emergency conditions (e.g., ASME Code Level C Service Limits or equivalent) or original design bases, the item is considered acceptable, provided the results of inspections and tests (Section 5.3.2) show no damage.

1.2 The second dashed statement of Item (3) should read:

-- An engineering evaluation of the effects of the calculated stresses on the functionality of the item. This evaluation should address all locations where stresses exceed faulted allowables and should include fatigue analysis for ASME Code Class 1 components and systems.

~~1.3 The last paragraph should read:~~

~~Reanalysis of safety related piping systems is not considered necessary unless there is observed damage to the piping systems. Experience has shown that piping systems designed to the ASME Code are not damaged by inertia loads resulting from an earthquake. If damage occurs, it will most likely occur in the piping supports or as damage to the pipe at fixed supports caused by relative support displacements. These types of damage would be detected by the plant walkdown inspections and post-shutdown inspections described in Sections 4 and 5 of this report. In general, piping reanalysis should be performed on a sampling basis to verify the adequacy of piping and to assess the need for supplemental nondestructive examination of potential high strain areas.~~

2. LONG-TERM EVALUATIONS

Coincident with the long-term evaluations, the plant should be restored to its current licensing basis. Exceptions to this must be approved by the Director, Office of Nuclear Reactor Regulation.

## D. IMPLEMENTATION

The purpose of this section is to provide guidance to applicants and licensees regarding the NRC staff's plans for using this regulatory guide. ~~This draft guide has been released to encourage public participation in its development.~~ Except in those cases in which the applicant proposes an acceptable alternative method for complying with the specified portions of the Commission's regulations, the method ~~to be~~ described in ~~the active~~ this guide ~~reflecting public comments~~ will be used in the evaluation of applications for construction permits, operating licenses, combined licenses, or design certification submitted after ~~the implementation date to be specified in the active guide~~ EFFECTIVE DATE OF THE FINAL RULE. This guide ~~would~~ will not be used in the evaluation of an application for an operating license submitted after ~~the implementation date to be specified in the active guide~~ EFFECTIVE DATE OF THE FINAL RULE if the construction permit was issued prior to that date.

Holders of an operating license or construction permit issued prior to EFFECTIVE DATE OF THE FINAL RULE may voluntarily implement the methods [redacted] Guides 1.12, [redacted] 2, and 1.166, [redacted] Postear[redacted] Other implementation strategies, [redacted] voluntary implem[redacted] of the cited regulatory guides, will be evaluated by the [redacted] a case-by-case basis.

## REGULATORY ANALYSIS

A separate regulatory analysis was not prepared for this regulatory guide. The ~~draft~~ regulatory analysis, "~~Proposed~~ Revision of 10 CFR Part 100 and 10 CFR Part 50," was prepared for the ~~proposed~~ amendments, and it provides the regulatory basis for this guide and examines the costs and benefits of the rule as implemented by the guide. A copy of the ~~draft~~ regulatory analysis is available for inspection and copying for a fee at the NRC Publ'c Document Room, 2120 L Street NW. (Lower Level), Washington, DC, as ~~Se~~ ~~-94-194-~~ LATER.

REGULATORY GUIDE 1.165
DRAFT WAS DG-1032

(SEISMIC SOURCES)

~~U.S. NUCLEAR REGULATORY COMMISSION~~ ~~February 1995~~
~~OFFICE OF NUCLEAR REGULATORY RESEARCH~~ ~~Division 1~~
~~Task DG-1032~~

~~DRAFT REGULATORY GUIDE~~

~~Contact: A.J. Murphy (301)415-6010~~

~~DRAFT~~ REGULATORY GUIDE ~~DG-1032~~1.165
(~~Previously issued~~ was Draft DG-10~~15~~32)

# IDENTIFICATION AND CHARACTERIZATION OF SEISMIC SOURCES AND DETERMINATION OF SAFE SHUTDOWN EARTHQUAKE GROUND MOTION

## A. INTRODUCTION

~~The NRC has recently proposed amendments to~~ 10 CFR Part 100, "Reactor Site Criteria," ~~in the Federal Register on October 17, 1994 (59 FR 52255). In the proposed~~ Section 100.23, "Geologic and Seismic Siting Factors," paragraph (c), "Geological, Seismological, and Engineering Characteristics," ~~would~~ requires that the geological, seismological, and engineering characteristics of a site and its environs be investigated in sufficient scope and detail to permit an adequate evaluation of the proposed site, to provide sufficient information to support evaluations performed to arrive at estimates of the Safe Shutdown Earthquake Ground Motion (SSE), and to permit adequate engineering solutions to actual or potential geologic and seismic effects at the proposed site. Data on the vibratory ground motion, tectonic surface deformation, nontectonic deformation, earthquake recurrence rates, fault geometry and slip rates, site foundation material, and seismically induced floods, water waves, and other siting factors ~~would~~ **will** be obtained by reviewing pertinent literature and carrying out field investigations.

~~In the proposed~~ Section 100.23, paragraph (d), "Geologic and Seismic Siting Factors," ~~would~~ requires that the geologic and seismic siting factors considered for design include a determination of the SSE for the site, the potential for surface tectonic and nontectonic deformations, the design bases for seismically induced floods and water waves, and other design conditions.

~~This regulatory guide is being issued in draft form to involve the public in the early stages of the development of a regulatory position in this area. It has not received complete staff review and does not represent an official NRC staff position.~~

~~Public comments are being solicited on the draft guide (including any implementation schedule) and its associated regulatory analysis or value/impact statement. Comments should be accompanied by appropriate supporting data. Written comments may be submitted to the Rules Review and Directives Branch, DFIPS, Office of Administration, U.S. Nuclear Regulatory Commission, Washington, DC 20555. Copies of comments received may be examined at the NRC Public Document Room, 2120 L Street NW., Washington, DC. Comments will be most helpful if received by May 12, 1995.~~

139

~~Requests for single copies of draft guides (which may be reproduced) or for placement on an automatic distribution list for single copies of future guides in specific divisions should be made in writing to the U.S. Nuclear Regulatory Commission, Washington, DC 20555, Attention: Office of Administration, Distribution and Mail Services Section.~~

~~In the proposed~~ Section 100.23, paragraph (d)(1), "Determination of the Safe Shutdown Earthquake Ground Motion," ~~would~~ requires that uncertainty inherent in estimates of the SSE be addressed through an appropriate analysis, such as a probabilistic seismic hazard analysis or suitable sensitivity analysis.

This guide ~~is~~ **has** ~~being~~ **been** developed to provide general guidance on procedures acceptable to the NRC staff to (1) conduct geological, geophysical, seismological, and geotechnical investigations, (2) identify and characterize seismic sources, (3) conduct probabilistic seismic hazard analyses, and (4) determine the SSE for satisfying the requirements of ~~the proposed~~ Section 100.23.

This guide contains several appendices that address the objectives stated above. Appendix A contains a list of definitions of pertinent terms. Appendix B describes the procedure used to determine the reference probability for the SSE exceedance level **that is acceptable to the staff**. Appendix C discusses the development of a seismic hazard information base and the determination of the probabilistic ground motion level and controlling earthquakes. Appendix D discusses site-specific geological, seismological, and geophysical investigations. Appendix E describes a method to confirm the adequacy of existing seismic sources and source parameters as the basis for determining the SSE for a site. Appendix F describes procedures to determine the SSE.

~~Regulatory guides are issued to describe and make available to the public such information as methods acceptable to the NRC staff for implementing specific parts of the Commission's regulations, techniques used by the staff in evaluating specific problems or postulated accidents, and guidance to applicants. Regulatory guides are not substitutes for regulations, and compliance with regulatory guides is not required. Regulatory guides are issued in draft form for public comment to involve the public in the early stages of developing the regulatory positions. Draft regulatory guides have not received complete staff review and do not represent official NRC staff positions.~~

Any information collection activities mentioned in this regulatory guide are contained as requirements ~~in the proposed amendments to~~ **in** 10 CFR Part 100 ~~that would~~ **which** provides the regulatory basis for this guide. The ~~proposed amendments have been submitted to the~~ **information collection requirements** in **10 CFR Part 100 have been approved by the** Office of Management and Budget ~~for~~

~~clearance that may be appropriate under the Paperwork Reduction Act. Such clearance, if obtained, would also apply to any information collection activities mentioned in this guide~~. **Approval No. 3150-0093.**

## B. DISCUSSION

### BACKGROUND

A probabilistic seismic hazard analysis (PSHA) has been identified in ~~the proposed~~ Section 100.23 as ~~one of the~~ **a** means to ~~address~~ **determine the SSE and account for** uncertainties in ~~estimates of the SSE~~ **the seismological and geological evaluations**. The ~~proposed~~ rule further recognizes that the nature of uncertainty and the appropriate approach to account for it depend on the tectonic regime and parameters such as the knowledge of seismic sources, the existence of historical and recorded data, and the **level of** understanding of **the** tectonics. Therefore, methods other than probabilistic methods such as sensitivity analyses may be adequate for some sites to account for uncertainties.

Appendix A, "Seismic and Geologic Siting Criteria for Nuclear Power Plants," to 10 CFR Part 100 is primarily based on a deterministic methodology. Past licensing experience in applying Appendix A has demonstrated the need to formulate procedures that quantitatively incorporate uncertainty **(including alternative scientific interpretations)** in the evaluation of seismic hazards. A **single** deterministic representation of seismic sources and ground motions at a site ~~does~~ **may** not explicitly provide a quantitative representation of the uncertainties in ~~scientific interpretations of~~ geological, seismological, and geophysical data **and alternative scientific interpretations**.

Probabilistic procedures were developed during the past 10-15 years specifically for nuclear power plant seismic hazard assessments in the Central and Eastern United States (CEUS) (the area east of the Rocky Mountains), also referred to as the Stable Continent Region (SCR). These procedures provide a structured approach for decision making with respect to the SSE when performed together with site-specific investigations. A PSHA provides a framework to address the uncertainties associated with the identification and characterization of seismic sources by incorporating multiple interpretations of seismological parameters. **A PSHA also provides an evaluation of the likelihood of SSE recurrence during the design life time of a given facility,** given the recurrence interval and recurrence pattern of pertinent seismic sources. Within the framework of a probabilistic analysis, uncertainties in the characterization of seismic sources and ground motions are identified and incorporated in the procedure at each step of the process for estimating the SSE. The role of ~~site-specific regional and site~~ geological, seismological, and geophysical investigations is to develop geosciences information about the site for use in the detailed design analysis of the facility, as well as to ensure that the seismic hazard analysis is based on up-to-date information.

Experience in performing seismic hazard evaluations in active plate margin regions in the Western United States (for example, the San Gregorio-Hosgri fault zone and the Cascadia Subduction Zone) has also identified uncertainties associated with the characterization of seismic sources (Refs. 1, 2, and 3). Sources of uncertainty include fault geometry, rupture segmentation, rupture extent, seismic-activity rate, ground motion and earthquake occurrence modeling. As is the case for sites in the CEUS, alternative hypotheses and parameters must be considered to account for these uncertainties.

Uncertainties associated with the identification and characterization of seismic sources in tectonic environments in both the CEUS and the Western United States should be evaluated. Therefore, the same basic approach can be applied to determine the SSE.

## APPROACH

The process to determine the SSE at a site ~~should~~is general includes:

1. Site- and region-specific geological, seismological, geophysical and geotechnical investigations, and
2. A probabilistic seismic hazard assessment.

## CENTRAL AND EASTERN UNITED STATES

The CEUS is considered to be that part of the United States east of the Rocky Mountain front, or east of Longitude 105° West (Refs. 4 and 5). To determine the SSE in the CEUS, an accepted PSHA methodology with a range of credible alternative input interpretations should be used. For sites in the CEUS, the seismic hazard methods, the data developed, and seismic sources identified by Lawrence Livermore National Laboratory (LLNL) (Refs. 4, 5, and 6) and the Electric Power Research Institute (EPRI) (Ref. 7) have been reviewed and accepted by the staff. The LLNL and EPRI studies developed data bases and scientific interpretations of available information and determined seismic sources and source characterizations for the CEUS (e.g., earthquake occurrence rates, estimates of maximum magnitude).

In the CEUS, characterization of seismic sources is more problematic than in the active plate-margin region because there is generally no clear association between seismicity and known tectonic structures or near-surface geology. In general, the observed geologic structures were generated in response to tectonic forces that no longer exist and bear little or no correlation with current tectonic forces. ~~Thus, there is greater uncertainty in making judgments about the CEUS than there is for active plate margin regions, and~~ Therefore, it is important to account for this uncertainty by the use of multiple alternative models.

The identification of seismic sources and reasonable alternatives in the CEUS considers hypotheses presently advocated for the occurrence of earthquakes in the CEUS (for example, the reactivation of favorably oriented zones of weakness or the local amplification and release of stresses concentrated around a geologic structure). In tectonically active areas of the CEUS, such as the New Madrid Seismic Zone, where geological, seismological, and geophysical evidence suggest the nature of the sources that generate the earthquakes ~~in that region~~, it may be more appropriate to evaluate those seismic sources by using procedures similar to those normally ~~applicable~~ applied in the Western United States.

WESTERN UNITED STATES

The Western United States is considered to be that part of the United States that lies west of the Rocky Mountain front, or west of approximately 105° West Longitude. For the Western United States, an information base of earth science data and scientific interpretations of seismic sources and source characterizations (e.g., geometry, seismicity parameters) comparable to the CEUS as documented in the LLNL and EPRI studies does not exist. For this region, specific interpretations on a site-by-site basis should be applied (Ref. 1).

The active plate margin region includes **for example**, coastal California, Oregon, and Washington. For the active plate margin region, where earthquakes can often be correlated with known tectonic structures, those structures should be assessed for their earthquake and surface deformation potential. In this region, at least three types of sources exist: (1) faults that are known to be at or near the surface, (2) buried (blind) sources that may often be manifested as folds at the earth's surface, and (3) subduction zone sources, such as those in the Pacific Northwest. The nature of surface faults can be evaluated by conventional surface and near-surface investigation techniques to assess ~~strike~~ **orientation** geometry, sense of displacements, length of rupture, Quaternary history, etc.

Buried (blind) faults are often ~~accompanied by coseismic~~ **associated with** surficial deformation such as folding, uplift, or subsidence. The surface expression of blind faulting can be detected by mapping the uplifted or down-dropped geomorphological features or stratigraphy, survey leveling, and geodetic methods. The nature of the structure at depth can often be evaluated by core borings and geophysical techniques.

**Continental** United States subduction zones are located in the Pacific Northwest and Alaska. Seismic sources associated with subduction zones are sources within the overriding plate, **on** the interface between the subducting and overriding lithospheric plates, and ~~intraslab sources~~ in the interior of the downgoing oceanic slab. The characterization of subduction zone seismic sources includes consideration of the following: three-dimensional geometry of the subducting plate, rupture segmentation of subduction zones, geometry of historical ruptures, constraints on the up-dip and down-dip extent of rupture, and comparisons with other subduction zones worldwide.

The Basin and Range region of the Western United States, and to a lesser extent the Pacific Northwest and the Central United States, ~~include~~ **exhibit** temporal clustering of earthquakes. Temporal clustering is best exemplified by the rupture histories within the Wasatch fault zone in Utah and the Meers fault in central Oklahoma, where several large late Holocene coseismic faulting events occurred at relatively close intervals (hundreds to thousands of years) that were preceded by long periods of quiescence that lasted thousands to tens of thousand years. Temporal clustering should be considered in these regions or wherever paleoseismic evidence indicates that it has occurred.

## C. REGULATORY POSITION

### 1. GEOLOGICAL, GEOPHYSICAL, SEISMOLOGICAL, AND GEOTECHNICAL INVESTIGATIONS

1.1 Comprehensive geological, seismological, geophysical, and geotechnical investigations of the site and regions around the site should be performed. **For existing nuclear power plant sites where additional units are planned, the geosciences technical information used originally to validate those sites may be inadequate, depending on how much new or additional information has become available since the initial investigations and analyses were performed, the quality of the investigations performed at the time, and the complexity of the site and regional geology and seismology. This technical information should be utilized along with all other available information to plan and determine the scope of additional investigations.** The~~se~~ investigations **described in this regulatory guide** are performed primarily to gather information needed to confirm the suitability of the site and to gather data pertinent to the safe design and construction of the nuclear power plant. Appropriate geological, seismological, and geophysical investigations are described in Appendix D to this ~~draft~~ guide. Geotechnical investigations are described in Regulatory Guide 1.132, "Site Investigations for Foundations of Nuclear Power Plants" (Ref. 8). Another important purpose for the site-specific investigations is to determine whether there are new data or interpretations that are not adequately incorporated in the existing PSHA databases. Appendix E describes a method to evaluate new information derived from the site-specific investigations in the context of the PSHA.

These investigations should be performed at four levels, with the degree of their **detail** based on distance from the site, the nature of the Quaternary tectonic **regime**, the geological complexity of the site and region, the existence of potential seismic sources, the potential for surface deformations, etc. **A more detailed discussion of the areas and levels of investigations and the bases for them is presented in Appendix D to this regulatory guide.** The levels of investigation are:

1. Regional geological and seismological investigations ~~such as geological reconnaissances and literature reviews should be~~ **are not expected to be extensive nor in great detail, but should include literature reviews, the study of maps and**

**remote sensing data, and, if necessary, ground truth reconnaissances** conducted within a radius of 320 km (200 miles) of the site to identify seismic sources (seismogenic and capable tectonic sources).

2. Geological, seismological, and geophysical investigations should be carried out within a radius of 40 km (25 miles) in greater detail than the regional investigations to identify and characterize the seismic and surface deformation potential of any capable tectonic sources and the seismic potential of seismogenic sources, or to demonstrate that such structures are not present. Sites with capable tectonic or seismogenic sources within a radius of 40 km (25 miles) may require more extensive geological and seismological investigations and analyses (similar in detail to investigations and analysis usually preferred within an 8-km (5-mile) radius).

3. Detailed geological, seismological, geophysical, and geotechnical investigations should be conducted within a radius of 8 km (5 miles) of the site, as appropriate, to evaluate the potential for tectonic deformation at or near the ground surface and to assess the ground motion transmission characteristics of soils and rocks in the site vicinity. Investigations should include monitoring by a network of seismic stations.

4. Very detailed geological, geophysical, and geotechnical engineering investigations should be conducted within the site ~~(~~[radius of approximately 1 km **(0.5 miles)**] to assess specific soil and rock characteristics as described in Regulatory Guide 1.132 (Ref. 8).

<u>1.2</u> The areas of investigations may be expanded beyond those specified above in regions that include capable tectonic sources, relatively high seismicity, ~~or~~ complex geology**, or which have experienced a large geologically recent earthquake**.

1.3 It should be demonstrated that deformation features discovered during **construction**, particularly faults, do not have the potential to compromise the safety of the plant. The two-step licensing practice of requiring applicants to acquire a Construction Permit (CP), and then during construction apply for an Operating License (OL), has been ~~expanded~~ **modified** to allow for an alternative procedure. The requirements and procedures applicable to NRC's issuance of combined licenses for nuclear power facilities are in 10 CFR 52.71. Applying the combined licensing procedure to a site could result in the award of a license prior to **the start of** construction. During the construction of nuclear power plants licensed in the past two decades, previously unknown faults were often discovered in site excavations. Before **issuing** an OL ~~would be issued~~, it was necessary to demonstrate that the faults in the excavation posed no hazard to the facility. Under the combined license procedure, these kinds of features should be mapped and assessed as to their rupture and ground motion generating potential while the excavations' walls and bases are exposed. Therefore, a commitment should be made, in documents (Safety Analysis Reports) supporting the license application, **to geologically map all excavations and** to notify the NRC staff when excavations are open for inspection ~~and to geologically map all excavations~~.

1.4 Sufficient data to clearly justify all conclusions should be presented. Because engineering solutions cannot always be **satisfactorally** demonstrated for the effects of permanent ground displacement, it is prudent to avoid a site that has a potential for surface or near-surface deformation. Such sites normally will require extensive additional investigations.

1.5 For the site and **for** the area surrounding the site, the lithologic, stratigraphic, hydrologic, and structural geologic conditions should be characterized. The investigations should include the measurement of the static and dynamic engineering properties of the materials underlying the site and an evaluation of physical evidence concerning the behavior during prior earthquakes of the surficial materials and the substrata underlying the site. The properties needed to assess the behavior of the underlying material during earthquakes, including the potential for liquefaction, and the characteristics of the underlying material in transmitting earthquake ground motions to the foundations of the plant (such as seismic wave velocities, density, water content, porosity, elastic moduli, and strength) should be measured.

## 2. SEISMIC SOURCES SIGNIFICANT TO THE SITE SEISMIC HAZARD

**2.1 For sites located in the CEUS, when the EPRI and LLNL PSHA methodologies are used to determine the SSE, it still may be necessary to investigate and characterize potential seismic sources that were previously unknown or uncharacterized, and perform sensitivity analyses to assess their significance to the seismic hazard estimate. ~~However, it is expected that newly discovered seismic sources along with their uncertainties are enveloped by the data base of the PSHA method used.~~ The results of investigations discussed in Regulatory Position 1 are to be used, in accordance with Appendix E, to determine whether updating of the LLNL or EPRI seismic sources and their characterization is needed. The guidance in Subsections 2.2 and 2.3 below and Appendix D of this guide may be used if additional seismic sources are to be developed as a result of investigations.**

2.~~1~~**2 When the LLNL and EPRI methods are not used or applicable, this and the following Subsection 2.3 provide general guidance for identification and characterization of seismic sources. The uncertainties in the characterization of seismic sources should be addressed as appropriate.** A seismic source is a general term referring to both seismogenic sources and capable tectonic sources. The main distinction between these two types of seismic sources is that a seismogenic source would not cause surface displacement, but a capable tectonic source causes surface or near-surface displacement.

Identification and characterization of seismic sources should be based on regional and site geological and geophysical data, historical and instrumental seismicity data, the regional stress field, and geological evidence of prehistoric earthquakes. Investigations to identify seismic sources are described in Appendix D. The bases for the identification of seismic sources should be documented. A general list of characteristics to be evaluated for a seismic source is presented in Appendix D.

2.~~2~~3 As part of the seismic source characterization, the seismic potential ~~(magnitude and recurrence rate)~~ for each source should be

~~determined~~evaluated. Typically, characterization of the seismic potential consists of four equally important elements:

1) Selection of a model for the spatial distribution of earthquakes in a source.

2) Selection of a model for the temporal distribution of earthquakes in a source.

3) Selection of a model for relative frequency of earthquakes of various magnitudes including an estimate for the largest earthquake that could occur in the source under the current tectonic regime.

4) A complete description of the uncertainty.

For example, in the LLNL study a truncated exponential model was used for the distribution of magnitudes given that an earthquake has occurred in a source. A stationary Poisson process is used to model the spatial and temporal occurrences of earthquakes in a source.

For a general discussion of evaluating the earthquake potential and characterizing the uncertainty refer to the Senior Seismic Hazard Analysis Committee Report (1993) (Ref. 9).

2.3.1 For sites in the CEUS, when the LLNL or EPRI method is not used or not applicable (such as in the New Madrid Seismic Zone, etc.), then it is necessary to evaluate the seismic potential for each source. The seismic sources and data that have been accepted by the NRC in past lecensing decision may be used along with the data gathered as the result of the investigation carried our as described in Section 1.

Generally, the seismic sources for the CEUS are area sources because there is uncertainty about the underlying causes of earthquakes due the lack of active surface faulting, a low rate of seismic activity and a short historical record. The assessment of earthquake recurrence for CEUS area sources commonly relies heavily on catalogs of observed seismicity. Because these catalogs are too short and incomplete it is difficult to obtain reliable estimates of the rate of activity. Considerable care must be taken to correct for incompleteness and to model the uncertainty in the rate of earthquake recurrence. To completely characterize the seismic potential for a source it is also necessary to estimate the largest earthquake magnitude that a seismic source is capable of generating under the current tectonic regime. This estimated magnitude defines the upper-bound of the earthquake recurrence relationship.

The assessment of earthquake potential for area sources is particularly difficult because the physical constraint most important to the assessment - the dimensions of fault rupture - is not known. As a result, the primary methods for assessing maximum earthquakes for area sources usually include a consideration of the historical seismicity record, the pattern and rate of seismic activity, the Quaternary (2 million years and younger), characteristics of the source, the current stress regime (and how it aligns with known tectonic structures), paleoseismic data and analogies to other sources in regions considered tectonically similar to the CEUS. Because of the shortness of the historical catalog and low rate of seismic activity considerable judgement is needed. It is important to characterize the large uncertainties in the assessment of the earthquake potential.

~~For sites located in the CEUS (when the LLNL or EPRI method is not used or not applicable), the seismic sources and data that have been accepted by the NRC staff in past licensing decisions may be used to estimate seismic potential. It is necessary to use a variety of approaches to estimate the maximum magnitude for a seismic source in the CEUS because there is uncertainty about the underlying causes of earthquakes because of due to the lack of active surface faulting. Also, there is a short historical record and low seismicity rate. The determination of the maximum magnitude for each identified seismic source is based on the maximum historical earthquake, the pattern and rate of seismic activity, the Quaternary (2 million years and younger) characteristics of the source, the current stress regime (and how it aligns with the known tectonic structures in the source), and paleoseismic data. These seismic sources and their parameters should be used to judge the adequacy of seismic sources and parameters used in the LLNL or EPRI PSHA.~~

2.~~2~~3.2 For sites located within the Western United States, earthquakes can often be associated with known tectonic structures. For faults, the ~~maximum magnitude~~ earthquake **potential** is related to the characteristics of the estimate~~d rupture, such as the length or the amount of fault displacement~~ **for the future rupture, such as the [illegible] rupture area, or the length, or the amount of fault displacement.** The following empirical relations can be used to estimate the earthquake potential from fault behavior data and also to estimate the amount of displacement that might be expected for a given magnitude. It is prudent to use several of these different relations to obtain an estimate of the earthquake magnitude.

1. Surface rupture length versus magnitude (Refs. ~~9-12~~ **10-13**).

2. Subsurface rupture length versus magnitude (Ref. 143).

3. Rupture area versus magnitude (Ref. 154).

4. Maximum and average displacement versus magnitude (Ref. 143).

5. Slip rate versus magnitude (Ref. 165).

~~Fault hazard analyses in the Western United States using these and other methods should consider the frequency of occurrence and calculated slip rates on faults based on the geochronology of strata and crosscutting relationships. Additionally, the phenomenon of temporal clustering should be considered when there is geological evidence of its past occurrence.~~

**When such correlations as references 9-15 are used the earthquake potential is often evaluated as the mean of the distribution. The difficult issue is the evaluation of the appropriate rupture dimension to be used. This is a judgemental process based on geological data for the fault in question and the behavior of other regional fault systems of the same type.**

**The other elements of the recurrence model are generally obtained using catalogs of seismicity, fault slip rate and other data. In some cases , it may be appuopriate to use recurrence models with memory, All the sources of uncertainty must be appropriately modeled. Additionally, the phenomenon of**

temporal clustering should be considered when there is geological evidence of its past occurrence.

2.23.3 For sites near subduction zones, such as in the Pacific Northwest and Alaska, the maximum magnitude must be assessed for subduction zone seismic sources. Worldwide observations indicate that the largest known earthquakes are associated with the plate interface, although intraslab earthquakes may also have large magnitudes. The assessment of plate interface earthquakes can be based on estimates of the expected dimensions of rupture or analogies to other subduction zones worldwide.

## 3. PROBABILISTIC SEISMIC HAZARD ANALYSIS (PSHA) PROCEDURES

A PSHA should be performed for the site as it allows the use of multiple models to estimate the likelihood of earthquake ground motions occurring at a site, and a PSHA systematically takes into account uncertainties that exist in various parameters (such as seismic sources, maximum earthquakes, and ground motion attenuation). Alternative hypotheses are considered in a quantitative fashion in a PSHA~~. The PSHA~~, and also can be ~~also~~ used to evaluate the hazard sensitivity to the varying significant parameters and to identify the relative contribution of each seismic source to the hazard. Ref. 9 provides guidance on how to conduct a PSHA.

The following steps describe a PSHA procedure that is acceptable to the NRC staff. The details of the calculational aspects of deriving controlling earthquakes from the PSHA are included in Appendix C.

1. Perform regional and site geological, seismological, and geophysical investigations in accordance with Regulatory Position 1 and Appendix D.

2. For CEUS sites, perform an evaluation of LLNL or EPRI seismic sources in accordance with Appendix E to determine whether they are consistent with the site-specific data gathered in Step 1 or require updating.

   The PSHA should only be updated if ~~it will lead to higher hazard estimates.~~ the new information indicates that the current version significantly under estimates the hazard and there is a strong technical basis that supports such a revision. It may be possible to justify a lower hazard estimate with an exceptionally strong technical basis. However, it is expected that large uncertainties in estimating seismic hazard in the CEUS will continue to exist in the future, and substantial delays in the licensing process will result in trying to address them with respect to a specific site. For these reasons the staff discourages efforts to justify a lower hazard estimate. For most cases, limited scope sensitivity studies should be sufficient to demonstrate that the existing data base in the PSHA envelops the findings from site-specific investigations. In general, the significant revisions to the LLNL and EPRI data base is to be only undertaken periodically (every ten years), or when there is an important new finding or occurrence that has, based on sensitivity studies, resulted in a significant increase in the hazard estimate. The overall revision of the data base will also require a reexamination of the reference probability discussed in Appendix B and used in Step 4 below. Any significant update should follow the guidance of Ref. 9.

3. Perform the LLNL or EPRI probabilistic seismic hazard analysis (for CEUS sites only) using original or updated sources as determined in Step 2 or a site-specific PSHA for sites in other parts of the country (Ref. 9). The ground motion estimates should be made for rock conditions in the free-field or by assuming hypothetical rock conditions for a nonrock site to develop the seismic hazard information base discussed in Appendix C.

4. Using the reference probability (1E-5 per yr) described in Appendix B, ~~which is applicable to all sites,~~ determine 5% of critically damped median spectral ground motion levels for the average of 5 and 10 Hz, $S_{a,5-10}$, and for the average of 1 and 2.5 Hz, $S_{a,1-2.5}$. Appendix B discusses situations in

which an alternative reference probability may be more appropriate. The alternative reference probability is reviewed and accepted on a case-by-case basis. Appendix B also describes a procedure that should be used when a general revision to the reference probability is needed.

5. Deaggregation **of the median probabilistic** ~~the~~ hazard **characterization** in accordance with Appendix C to determine the controlling earthquakes (i.e., magnitudes and distances). Document the hazard information base as discussed in Appendix C.

4. <u>PROCEDURES FOR DETERMINING THE SSE</u>

After completing the PSHA (See Regulatory Position 3) and determining **the** controlling earthquakes, the following procedure should be used to determine the SSE. Appendix F contains an additional discussion of some of the characteristics of the SSE.

1. With the controlling earthquakes determined as described in Regulatory Position 3 and by using the procedures in ~~Draft~~ Standard Review Plan (SRP) Section 2.5.2 (which may include the use of ground motion models not included in the probabilistic seismic hazard analysis but that are more appropriate for the source, region, and site under consideration or that represent the latest scientific development), develop 5% of critical damping response spectral <u>shapes</u> for the actual or assumed rock conditions. **The same controlling earthquakes are also used to derive vertical response spectral shape.**

2. Use $S_{a,5-10}$ to scale the response spectrum shape corresponding to the controlling earthquake. If, as described in Appendix C, there is a controlling earthquake for $S_{a,1-2.5}$, determine that the $S_{a,5-10}$ scaled response spectrum also envelopes the ground motion spectrum for the controlling earthquake for $S_{a,1-2.5}$. Otherwise, modify the shape to envelope the low-

frequency spectrum or use two spectra in the following steps. See additional discussion in Appendix F. For **a** ~~the~~ rock site go to Step 4.

3. For the nonrock sites, perform a site-specific soil amplification analysis considering uncertainties in site-specific geotechnical properties and parameters to determine response spectra at the free ground surface in the free-field for the actual site conditions.

4. Compare the smooth SSE spectrum or spectra used in design (e.g., 0.3g, broad-band spectra used in Advanced Light Water Reactor designs) with the spectrum or spectra determined in Step 2 for rock sites or determined in Step 3 for the nonrock sites to assess the adequacy of the SSE spectrum or spectra.

   **For situations where site specific design response spectra are needed,** ~~T~~ **t**o obtain an adequate design SSE based on the site-specific response spectrum or spectra, develop a smooth spectrum or spectra or use a standard broad band shape that envelopes the spectra of Step 2 or Step 3.

   Additional discussion of this step is provided in Appendix F.

## D. IMPLEMENTATION

The purpose of this section is to provide guidance to applicants and licensees regarding the NRC staff's plans for using this regulatory guide.

This proposed revision has been released to encourage public participation in its development. Except in those cases in which the applicant proposes an acceptable alternative method for complying with the specified portions of the Commission's regulations, the method to be described in the active guide reflecting public comments will be used in the evaluation of applications for construction permits, operating licenses, early site permits, or combined licenses submitted after the implementation date to be specified in the active guide. This guide ~~would~~ **will** not be used in the evaluation of an application for an operating license submitted after the implementation date to be specified in the active guide if the construction permit was issued prior to that date.

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[1]Copies are available for inspection or copying for a fee from the NRC Public Document Room at 2120 L Street NW., Washington, DC; the PDR's mailing address is Mail Stop LL-6, Washington, DC 20555; telephone (202)634-3273; fax (202)634-3343.

[2]Copies are available for inspection or copying for a fee from the NRC Public Document Room at 2120 L Street NW., Washington, DC; the PDR's mailing address is Mail Stop LL-6, Washington, DC 20555; telephone (202)634-3273; fax (202)634-3343. Copies may be purchased at current rates from the U.S. Government Printing Office, P.O. Box 37082, Washington, DC 20402-9328 (telephone (202)512-2249); or from the National Technical Information Service by writing NTIS at 5285 Port Royal Road, Springfield, VA 22161.

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## APPENDIX A

## DEFINITIONS

Controlling Earthquakes --The controlling earthquakes are the earthquakes used to determine spectral shapes or to estimate ground motions at the site. There may be several controlling earthquakes for a site. ~~In~~ As a result of the probabilistic seismic hazard analysis (PSHA), the controlling earthquakes are characterized as mean magnitudes and distances derived from a deaggregation analysis of the median estimate of the PSHA. ~~The controlling earthquakes are the earthquakes used to determine spectral shapes or to estimate ground motions at the site. There may be several controlling earthquakes for a site.~~

Earthquake Recurrence-- Earthquake recurrence Earthquake recurrence is the frequency of recurrence of earthquakes having various magnitudes. Recurrence relationships or curves are developed for each seismic source, and reflect the frequency of occurrence (usually expressed on an annual basis) of magnitudes up to the maximum, including measures of uncertainty.

Intensity -- The intensity of an earthquake is a measure of vibratory ground motion effects on humans, human-built structures, and on the earth's surface at a particular location. Intensity is described by a numerical value on the Modified Mercalli scale.

Magnitude -- An earthquake's magnitude is a measure of the strength of the earthquake as determined from seismographic observations.

Maximum Magnitude-- The maximum magnitude is the upper-bound to recurrence curves.

Nontectonic Deformation -- Nontectonic deformation is distortion of surface or near-surface soils or rocks that is not directly attributable to tectonic activity. Such deformation includes features associated with subsidence, karst terrane, glaciation or deglaciation, and growth faulting.

Safe Shutdown Earthquake Ground Motion (SSE) -- The Safe Shutdown Earthquake Ground Motion is the free-field vibratory ground motion for which certain structures, systems, and components ~~would be~~ are designed, pursuant to ~~the proposed~~ Appendix S to 10 CFR Part 50, to remain functional.

Seismic Potential--A model giving the complete description of the future earthquake activity in a seismic source zone. The model includes a relation giving the frequency (rate) of earthquakes of any magnitude, an estimate of the largest earthquake that could occur under current tectonic regime and a complete description of the uncertainty. A typical model used for PSHA is the use of a truncated exponential model for the magnitude distribution and a stationary Poisson process for the temporal and spatial occurrence of earthquakes.

Seismic Source -- A "seismic source" is a general term referring to both seismogenic sources and capable tectonic sources.

Capable Tectonic Source -- A "capable tectonic source" is a tectonic structure that can generate both vibratory ground motion and tectonic surface deformation such as faulting or folding at or near the earth's surface in the present seismotectonic regime. It is described by at least one of the following characteristics:

a. Presence of surface or near-surface deformation of landforms or geologic deposits of a recurring nature within the last approximately 500,000 years or at least once in the last approximately 50,000 years.

b. A reasonable association with one or more large earthquakes or sustained earthquake activity that are usually accompanied by significant surface deformation.

c. A structural association with a capable tectonic source having characteristics of section a in this paragraph such that movement on one could be reasonably expected to be accompanied by movement on the other.

In some cases, the geological evidence of past activity at or near the ground surface along a particular capable tectonic source may be obscured at a particular site. This might occur, for example, at a site having a deep overburden. For these cases, evidence may exist elsewhere along the structure from which an evaluation of its characteristics in the vicinity of the site can be reasonably based. Such evidence is to be used in determining whether the structure is a capable tectonic source within this definition.

Notwithstanding the foregoing paragraphs, ~~structural~~ the association of a structure with the geological structures~~al~~ ~~features~~ that are ~~geologically old (~~at least pre-Quaternary~~)~~, such as many of those found in the Central and Eastern region of the United States will, in the absence of conflicting evidence, demonstrate that the structure is not a capable tectonic source within this definition.

Seismogenic Source -- A "seismogenic source" is a portion of the earth that ~~has~~ we assumed has uniform earthquake potential (same expected maximum earthquake and recurrence frequency~~ of recurrence~~) distinct from ~~other~~ the seismicity of the surrounding regions. A seismogenic source will generate vibratory ground motion but is assumed not to cause surface displacement. Seismogenic sources cover a wide range of possibilities from a well-defined tectonic structure to simply a large region of diffuse seismicity (seismotectonic province) thought to be characterized by the same earthquake recurrence model. A seismogenic source is also characterized by its involvement in the current tectonic regime (the Quaternary, or approximately the last 2 million years).

Stable Continental Region -- A "stable continental region" (SCR) is composed of continental crust, including continental shelves, slopes, and attenuated continental crust, and excludes active plate boundaries and zones of currently active tectonics directly influenced by plate margin processes. It exhibits no significant deformation associated with the major Mesozoic-to-Cenozoic (last 240 million years) orogenic belts. It excludes major zones of Neogene (last 25 million years) rifting, volcanism, or suturing.

Stationary Poisson Process--A probabilistic model of the occurrence of an event over time (space) characterized by the following properties: (1) the occurrence of the event in small interval is constant over time (space), (2) the occurrence of two (or more) events in a small interval is "negligible", and (3) the occurrence of the event in non-overlapping intervals is independent. This

Tectonic Structure -- A tectonic structure is a large-scale dislocation or distortion, usually within the earth's crust. Its extent may be on the order of tens of meters (yards) to hundreds of kilometers (miles).

# APPENDIX B

## REFERENCE PROBABILITY FOR THE EXCEEDANCE LEVEL OF THE SAFE SHUTDOWN EARTHQUAKE GROUND MOTION

### B.1 INTRODUCTION

This appendix describes the procedure used by the NRC staff to determine the reference probability, an annual probability of exceeding the Safe Shutdown Earthquake Ground Motion (SSE) at future nuclear power plant sites, that is acceptable to the NRC staff. The reference probability is used in Appendix C in conjunction with the probabilistic seismic hazard analysis (PSHA).

### B.2 REFERENCE PROBABILITY FOR THE SSE

The reference probability is the annual probability level such that 50% of a set of currently operating plants (selected by the NRC, see Table B.1) has an annual median probability of exceeding the SSE that is below this level. The reference probability is determined for the annual probability of exceeding the average of the 5 and 10 Hz SSE response spectrum ordinates associated with 5% of critical damping.

### B.3 PROCEDURE TO DETERMINE THE REFERENCE PROBABILITY

The **following** procedure was used to determine the reference probability and should **be used** in the future if general revisions to PSHA methods or data bases **result** in significant changes in hazard predictions for the selected plant sites in Table B.1.

The reference probability is calculated using the Lawrence Livermore National Laboratory (LLNL) methodology and results (Refs. B.1 and B.2) but is also considered applicable for the Electric Power Research Institute (EPRI) study (Refs. B.3 and B.4). This reference probability is also to be used in conjunction with sites not in the Central and Eastern United States (CEUS) and for sites for which LLNL and EPRI methods and data have not been used or are not available. **However,** ~~T~~the final SSE ground motion at a higher reference probability **may be more** appropriate and acceptable[1] **for some sites** considering **the** slope characteristics of the site hazard curves, the overall uncertainty in calculations (i.e., differences between mean and median hazard estimates), and the knowledge of the seismic sources that contribute to the hazard. Reference B.4 includes a procedure to determine an alternative reference probability on the risk-based considerations; its application will also be reviewed on a case-by-case basis.

### B.3.1 Selection of Current Plants for Reference Probability Calculations

Table B.1 identifies plants, along with their site characteristics, used in calculating the reference probability. These plants represent relatively recent designs that used Regulatory Guide 1.60, "Design Response Spectra for Seismic Design of Nuclear Power Plants" (Ref. B.5), or similar spectra as their design bases. The use of these plants should ensure an adequate level of conservatism in determining an SSE consistent with recent licensing decisions.

### B.3.2 Procedure To Establish Reference Probability

Step 1

Using **LLNL, EPRI, or a comparable methodology that is acceptable to the NRC staff,** ~~an accepted methodology,~~ calculate the seismic hazard results for the site for spectral responses at 5 and 10 Hz (as stated earlier, the staff used the LLNL methodology and associated results as documented in Refs. B.1 and B.2).

Step 2

Calculate the ~~median~~ composite annual probability of exceeding the SSE for spectral responses at 5 and 10 Hz using median hazard estimates. The composite annual probability is determined as:

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[1] The use of a higher reference probability will be reviewed and accepted on a case-by-case basis.

$$\text{Composite probability} = 1/2(a1) + 1/2(a2)$$

where a1 and a2 represent median annual probabilities of exceeding SSE spectral ordinates at 5 and 10 Hz, respectively. The procedure is illustrated in Figure B-1.

Step 3

Figure B-2 illustrates the distribution of median probabilities of exceeding the SSEs for the plants in Table B.1 based on the LLNL methodology (Refs. B.1 and B.2). The reference probability is simply the median probability of this distribution.

For the LLNL methodology, this reference probability is 1E-5/yr and, as stated earlier, is also to be used in conjunction with the current EPRI methodology (Ref. B.3) or for sites not in the CEUS.

Table B.1 Plants/Sites Used in Determining Reference Probability

| Plant/Site Name | Soil Condition Primary/Secondary* |
|---|---|
| Limerick | Rock |
| Shearon Harris | Sand - S1 |
| Braidwood | Rock |
| River Bend | Deep Soil |
| Wolf Creek | Rock |
| Watts Bar | Rock |
| Vogtle | Deep Soil |
| Seabrook | Rock |
| Three Mile Is. | Rock/Sand - S1 |
| Catawba | Rock/Sand - S1 |
| Hope Creek | Deep Soil |
| McGuire | Rock |
| North Anna | Rock/Sand - S1 |
| Summer | Rock/Sand - S1 |
| Beaver Valley | Sand - S1 |
| Byron | Rock |
| Clinton | Till - T3 |
| Davis Besse | Rock |
| LaSalle | Till - T2 |
| Perry | Rock |
| Bellefonte | Rock |
| Callaway | Rock/Sand - S1 |
| Commanche Peak | Rock |
| Grand Gulf | Deep Soil |
| South Texas | Deep Soil |
| Waterford | Deep Soil |
| Millstone 3 | Rock |
| Nine Mile Point | Rock/Sand - S1 |
| Brunswick | Sand - S1 |

* If two soil conditions are listed, the first is the primary and the second is the secondary soil condition. See Ref. B.1 for a discussion of soil conditions.

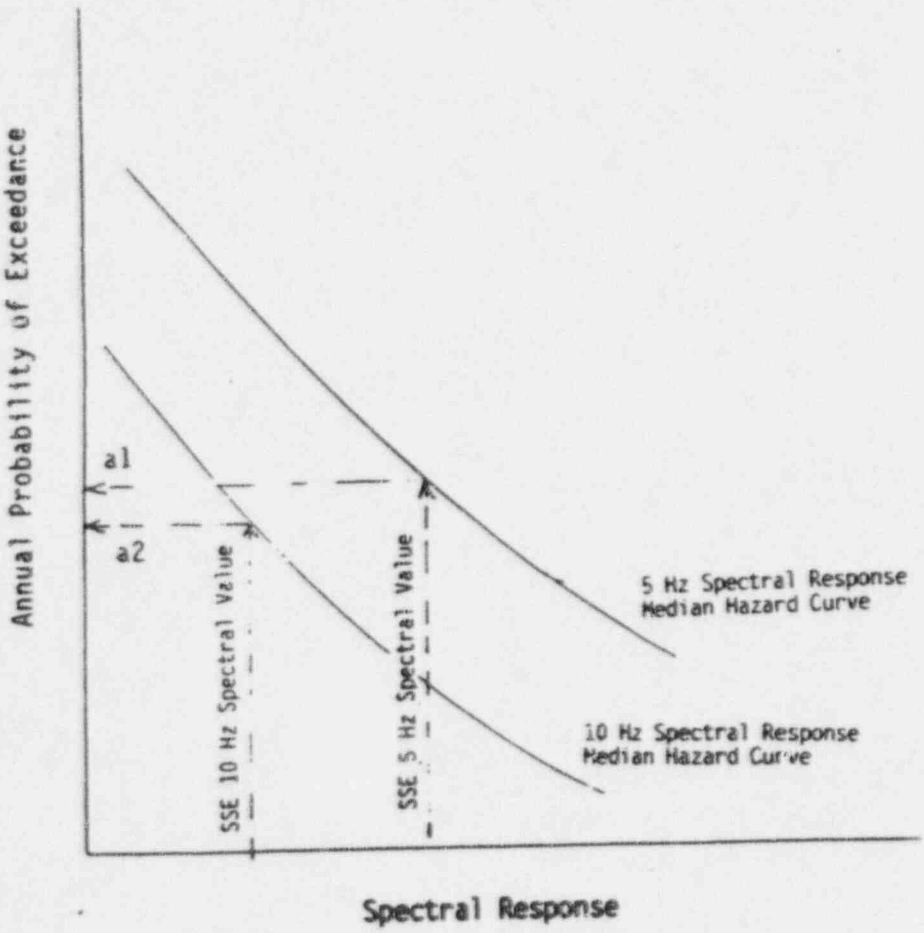
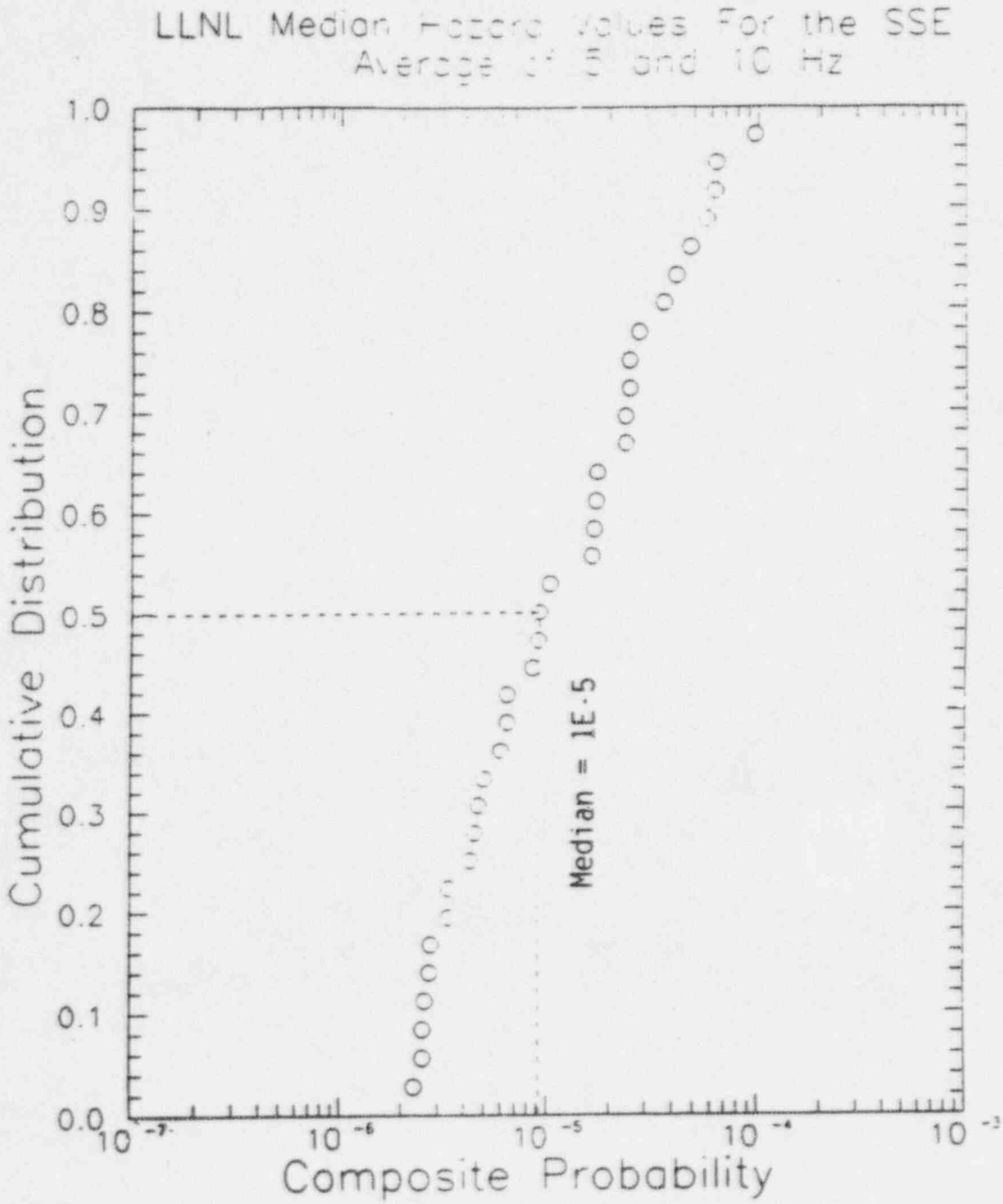



**Figure B.1 Procedure to Compute Probability of Exceeding Design Basis**

Comp. Prob. = 1/2(a1) + 1/2(a2)




**Probability of Exceeding SSE**

**Figure B.2 Probability of Exceeding SSE using Median LLNL Hazard Estimates**

## REFERENCES


B.1 D.L. Bernreuter et al., "Seismic Hazard Characterization of 69 Nuclear Plant Sites East of the Rocky Mountains," NUREG/CR-5250, January 1989.[2]

B.2 P. Sobel, "Revised Livermore Seismic Hazard Estimates for Sixty-Nine Nuclear Power Plant Sites East of the Rocky Mountains," NUREG-1488, USNRC, April 1994.[2]

B.3 Electric Power Research Institute, "Probabilistic Seismic Hazard Evaluations at Nuclear Power Plant Sites in the Central and Eastern United States: Resolution of the Charleston Earthquake Issue," Report NP-6395-D, April 1989.

B.4 Attachment to Letter from D. J. Modeen, Nuclear Energy Institute, to A.J. Murphy, USNRC, Subject: Seismic Siting Decision Process, May 25, 1994.[3]

B.5 USNRC, "Design Response Spectra for Seismic Design of Nuclear Power Plants," Regulatory Guide 1.60.[2]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[2] Copies are available for inspection or copying for a fee from the NRC Public Document Room at 2120 L Street NW., Washington, DC; the PDR's mailing address is Mail Stop LL-6, Washington, DC 20555; telephone (202)634-3273; fax (202)634-3343. Copies may be purchased at current rates from the U.S. Government Printing Office, P.O. Box 37082, Washington, DC 20402-9328 (telephone (202)512-2249); or from the National Technical Information Service by writing NTIS at 5285 Port Royal Road, Springfield, VA 22161.

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# APPENDIX C

## DETERMINATION OF CONTROLLING EARTHQUAKES AND DEVELOPMENT OF SEISMIC HAZARD INFORMATION BASE

### C.1 INTRODUCTION

This appendix elaborates on the steps described in Regulatory Position 3 of **this regulatory guide** ~~Draft Regulatory Guide DG-1032~~ to determine the controlling earthquakes used to define the Safe Shutdown Earthquake Ground Motion (SSE) at the site and to develop a seismic hazard information base. The information base summarizes the contribution of individual magnitude and distance ranges to the seismic hazard and the magnitude and distance values of the controlling earthquakes at the average of 1 and 2.5 Hz and the average of 5 and 10 Hz. They are developed for the ground motion level corresponding to the reference probability as defined in Appendix B to this regulatory guide.

The spectral ground motion levels, as determined from a probabilistic seismic hazard analysis (PSHA), are used to scale a response spectrum shape. A site-specific response spectrum shape is determined for the controlling earthquakes and local site conditions. Regulatory Position 4 and Appendix F to this regulatory guide describe a procedure to determine the SSE using the controlling earthquakes **and results from the PSHA.**

### C.2 PROCEDURE TO DETERMINE CONTROLLING EARTHQUAKES

The following is an approach acceptable to the NRC staff for determining the controlling earthquakes and developing a seismic hazard information base. This procedure is based on a de-aggregation of the probabilistic seismic hazard in terms of earthquake magnitudes and distances. Once the controlling earthquakes have been obtained, the SSE response spectrum can be determined according to the procedure described in Appendix F to this regulatory guide.

#### Step 1

~~(a)~~ Perform a site-specific PSHA using the Lawrence Livermore National Laboratory (LLNL) or Electric Power Research Institute (EPRI) methodologies for Central and Eastern United States (CEUS) sites or perform a site-specific

PSHA for sites not in the CEUS or for sites for which LLNL or EPRI methods and data are not **applicable** ~~available~~, for actual or assumed rock conditions. The hazard assessment **(mean, median, 85th percentile, and 15th percentile)** should be performed for spectral accelerations at 1, 2.5, 5, 10, and 25 Hz, and the peak ground acceleration. A lower-bound magnitude of 5.0 is recommended. ~~The PSHA should include an uncertainty assessment.~~

~~(b) Determine the following parameters as part of the assessment for each ground motion measure:~~

- ~~Total hazard in terms of the median (50th percentile), mean, 85th, and 15th percentile hazard curves.~~

- ~~De-aggregated median hazard results for a matrix of magnitude-distance pairs discussed in Step 3. As a part of the information base, de-aggregated results for mean hazard results may also be useful.~~

~~These results obtained from the de-aggregation of the median hazard are used to determine the SSE and to develop the seismic hazard information base.~~

<u>Step 2</u>

(a) Using the reference probability as defined in Appendix B to this regulatory guide, determine the ground motion levels for the spectral accelerations at 1, 2.5, 5, and 10 Hz from the total <u>median</u> hazard obtained in Step 1.

(b) Calculate the average **of the** ground motion level for the 1 and 2.5 Hz and the **5 and** 10 Hz spectral acceleration pairs.

~~Steps 3 to 5 describe the procedure to develop the seismic hazard information base for each ground motion level determined in Step 2. This information base will consist of:~~

- ~~Fractional contribution of each magnitude-distance pair to the total median seismic hazard.~~

- ~~Magnitudes and distances of the controlling earthquakes.~~

- ~~The ground motion levels for the spectral accelerations at 1, 2.5, 5, and 10 Hz defined in Step 2.~~

- ~~The average of the ground motion levels listed above at the 1 and 2.5 Hz, $S_{a1-2.5}$, and 5 and 10 Hz, $S_{a5-10}$, spectral accelerations corresponding to the reference probability.~~

Step 3

**Perform a complete probabilistic seismic hazard analysis ~~is performed~~ for each of the magnitude-distance bins described in Table C.3.**

Step 4

Using the de-aggregated median hazard results from Step ~~1~~ **3**, at the ground motion levels obtained from Step 2 calculate the fractional contribution to the total median hazard of earthquakes in a selected set of magnitude and distance bins (Section C.3 provides magnitude and distance bins to be used in conjunction with the LLNL and EPRI methods) for the average of 1 and 2.5 Hz and 5 and 10 Hz. The median annual probability of exceeding the ground motion levels calculated in Step ~~1~~2 for each magnitude and distance bin and ground motion measure is denoted by $H_{mdf}$.

The fractional contribution of each magnitude and distance bin to the total hazard for the average of 1 and 2.5 Hz, $P(m,d)_1$, is computed according to:

$$P(m,d)_1 = \frac{\dfrac{\left(\sum_{f=1,2} H_{mdf}\right)}{2}}{\sum_m \sum_d \dfrac{\left(\sum_{f=1,2} H_{mdf}\right)}{2}} \qquad (Equation\ 1)$$

where f = 1 and f = 2 represent the ground motion measure at 1 and 2.5 Hz, respectively.

The fractional contribution of each magnitude and distance bin to the total hazard for the average of 5 and 10 Hz, $P(m,d)_2$, is computed according to:

$$P(m,d)_2 = \frac{\dfrac{(\sum_{f=1,2} H_{mdf})}{2}}{\sum_m \sum_d \dfrac{(\sum_{f=1,2} H_{mdf})}{2}} \qquad \textit{(Equation 2)}$$

where f = 1 and f = 2 represent the ground motion measure at 5 and 10 Hz, respectively.

Step ~~4~~5

Review the magnitude-distance distribution for the average of 1 and 2.5 Hz to determine whether the contribution to the hazard for distances of 100 km or greater is substantial (on the order of 5% or greater).

If the contribution to the hazard for distances of 100 km or greater exceeds 5%, additional calculations are needed to determine the controlling earthquakes using the magnitude-distance distribution for distances greater than 100 km (63 mi). This distribution, $P_{>100}(m,d)_1$, is defined by:

$$P{>}100\ (m,d)_1 = \frac{P(m,d)_1}{\sum_m \sum_{d>100} P(m,d)_1} \qquad \textit{(Equation 3)}$$

The purpose of this calculation is to identify a distant, larger event that may control low-frequency content of a response spectrum.

The distance of 100 km is chosen for CEUS sites. However, **for all sites** ~~CEUS sites and sites not in the CEUS~~ the results of full magnitude-distance distribution should be carefully examined to ensure that proper controlling earthquakes are clearly identified.

Step ~~5~~6

Calculate the mean magnitude and distance of the controlling earthquake associated with the ground motions determined in Step 2 for the average of 5 and 10 Hz. The following relation is used to calculate the mean magnitude using results of the entire magnitude-distance bins matrix:

$$M_c\ (5\text{-}10\ Hz) = \sum_m m \sum_d P(m,d)_2 \qquad \text{(Equation 4)}$$

where m is the central magnitude value for each magnitude bin.

The mean distance of the controlling earthquake is determined using results of the entire magnitude-distance bins matrix:

$$Ln\ \{D_c\ (5\text{-}10\ Hz)\} = \sum_d Ln(d) \sum_m P(m,d)_2 \qquad \text{(Equation 5)}$$

where d is the centroid distance value for each distance bin.

Step ~~6~~7

If the contribution to the hazard calculated in Step ~~4~~5 for distances of 100 km or greater exceeds 5% for the average of 1 and 2.5 Hz, calculate the mean magnitude and distance of the controlling earthquakes associated with the ground motions determined in Step 2 for the average of 1 and 2.5 Hz. The following relation is used to calculate the mean magnitude using calculations based on magnitude-distance bins greater than distances of 100 km as discussed in Step 4:

$$M_c\ (1\text{-}2.5\ Hz) = \sum_m m \sum_{d>100} P{>}100\,(m,d)_1 \qquad \text{(Equation 6)}$$

where m is the central magnitude value for each magnitude bin.

The mean distance of the controlling earthquake is based on magnitude-distance bins greater than distances of 100 km as discussed in Step 4 and determined according to:

$$Ln\,\{D_c\,(1\text{-}2.5\ Hz)\} = \sum_{d>100} Ln(d) \sum_{m} P{>}100\,(m,d)_2 \qquad \text{(Equation 7)}$$

where d is the centroid distance value for each distance bin.

Step ~~7~~8

Determine the SSE response spectrum using the procedure described in Appendix F of this regulatory guide.

## C.3 EXAMPLE FOR A CEUS SITE

To illustrate the procedure in Section C.2, calculations are shown here for a CEUS site using the 1993 LLNL hazard results (Refs. C.1 and C.2). It must be emphasized that the recommended magnitude and distance bins and procedure used to establish controlling earthquakes were developed for application in the CEUS where the nearby earthquakes generally control the response in the 5 to 10 Hz frequency range and larger but distant events can control the lower frequency range. For other situations, alternative binning schemes as well as a study of contributions from various bins will be necessary to identify controlling earthquakes consistent with the distribution of the seismicity.

Step 1

The 1993 LLNL seismic hazard methodology (Ref. C.1 and C.2) was used to determine the hazard at the site. A lower bound magnitude of 5.0 was used in this analysis. The analysis was performed for spectral acceleration at 1, 2.5, 5, and 10 Hz.

Step 2

The hazard curves at 1, 2.5, 5, and 10 Hz obtained in Step 1 are assessed at the reference probability value of 1E-5/yr, as defined in Appendix B to this regulatory guide. The corresponding ground motion level values are given in Table C.1.

Table C.1

Ground Motion Levels

| Frequency (Hz) | 1 | 2.5 | 5 | 10 |
|---|---|---|---|---|
| Spectral Acc. (cm/s/s) | 88 | 258 | 351 | 551 |

The average of the ground motion levels at the 1 and 2.5 Hz, $S_{a1-2.5}$, and 5 and 10 Hz, $S_{a5-10}$, are given in Table C.2.

Table C.2

Average Ground Motion Values

| | |
|---|---|
| $S_{a1-2.5}$ (cm/s/s) | 173 |
| $S_{a5-10}$ (cm/s/s) | 451 |

Step 3

The median seismic hazard is de-aggregated for the matrix of magnitude and distance bins as given in Table C.3.

Table C.3

Recommended Magnitude and Distance Bins

| Distance Range of Bin (km) | Magnitude Range of Bin | | | | |
|---|---|---|---|---|---|
| | 5 - 5.5 | 5.5 - 6 | 6 -6.5 | 6.5 - 7 | >7 |
| 0-15 | | | | | |
| 15-25 | | | | | |
| 25-50 | | | | | |
| 50-100 | | | | | |
| 100-200 | | | | | |
| 200-300 | | | | | |
| > 300 | | | | | |

A complete probabilistic hazard analysis was performed for each bin to determine the contribution to the hazard from all earthquakes within the bin, e.g., all earthquakes with magnitudes 6 to 6.5 and distance 25 to 50 km from the site. The hazard values corresponding to the ground motion levels defined in step 2 for the spectral acceleration at 1, 2.5, 5, and 10 Hz are listed in

Tables C.4-C.7.

Table C.4

Median Exceeding Probability Values for Spectral Accelerations at 1 Hz (88 cm/s/s)

| Distance Range of Bin (km) | Magnitude Range of Bin | | | | |
|---|---|---|---|---|---|
| | 5 - 5.5 | 5.5 - 6 | 6 -6.5 | 6.5 - 7 | >7 |
| 0-15 | 1.98E-08 | 9.34E-08 | 1.14E-08 | 0 | 0 |
| 15-25 | 4.03E-09 | 2.58E-08 | 2.40E-09 | 0 | 0 |
| 25-50 | 1.72E-09 | 3.03E-08 | 2.14E-08 | 0 | 0 |
| 50-100 | 2.35E-10 | 1.53E-08 | 7.45E-08 | 2.50E-08 | 0 |
| 100-200 | 1.00E-11 | 2.36E-09 | 8.53E-08 | 6.10E-07 | 0 |
| 200-300 | 0 | 1.90E-11 | 1.60E-09 | 1.84E-08 | 0 |
| > 300 | 0 | 0 | 8.99E-12 | 1.03E-11 | 1.69E-10 |

Table C.5

**Median Exceeding Probability Values for Spectral Accelerations at 2.5 Hz (258 cm/s/s)**

| Distance Range of Bin (km) | Magnitude Range of Bin | | | | |
|---|---|---|---|---|---|
| | 5 - 5.5 | 5.5 - 6 | 6 -6.5 | 6.5 - 7 | >7 |
| 0-15 | 2.24E-07 | 3.33E-07 | 4.12E-09 | 0 | 0 |
| 15-25 | 5.39E-08 | 1.20E-07 | 1.08E-08 | 0 | 0 |
| 25-50 | 2.60E-08 | 1.6[illegible]E-07 | 6.39E-08 | 0 | 0 |
| 50-100 | 3.91E-09 | 6.27E-08 | 1.46E-07 | 4.09E-08 | 0 |
| 100-200 | 1.50E-10 | 7.80E-09 | 1.07E-07 | 4.75E-07 | 0 |
| 200-300 | 7.16E-14 | 2.07E-11 | 7.47E-10 | 5.02E-09 | 0 |
| > 300 | 0 | 1.52E-14 | 4.94E-13 | 9.05E-15 | 2.36E-15 |

Table C.6

**Median Exceeding Probability Values for Spectral Accelerations at 5 Hz (351 cm/s/s)**

| Distance Range of Bin (km) | Magnitude Range of Bin | | | | |
|---|---|---|---|---|---|
| | 5 - 5.5 | 5.5 - 6 | 6 -6.5 | 6.5 - 7 | >7 |
| 0-15 | 4.96E-07 | 5.85E-07 | 5.16E-[illegible] | 0 | 0 |
| 15-25 | 9.39E-08 | 2.02E-07 | 1.36E-08 | 0 | 0 |
| 25-50 | 2.76E-08 | 1.84E-07 | 7.56E-08 | 0 | 0 |
| 50-100 | 1.23E-08 | 3.34E-08 | 9.98E-08 | 2.85E-08 | 0 |
| 100-200 | 8.06E-12 | 1.14E-09 | 2.54E-08 | 1.55E-07 | 0 |
| 200-300 | 0 | 2.39E-13 | 2.72E-11 | 4.02E-10 | 0 |
| > 300 | 0 | 0 | 0 | 0 | 0 |

Table C.7

**Median Exceeding Probability Values for Spectral Accelerations at 10 Hz (551 cm/s/s)**

| Distance Range of Bin (km) | Magnitude Range of Bin | | | | |
|---|---|---|---|---|---|
| | 5 - 5.5 | 5.5 - 6 | 6 -6.5 | 6.5 - 7 | >7 |
| 0-15 | 1.11E-06 | 1.12E-06 | 8.30E-08 | 0 | 0 |
| 15-25 | 2.07E-07 | 3.77E-07 | 3.12E-08 | 0 | 0 |
| 25-50 | 4.12E-08 | 2.35E-07 | 1.03E-07 | 0 | 0 |
| 50-100 | 5.92E-10 | 2.30E-08 | 6.89E-08 | 2.71E-08 | 0 |
| 100-200 | 1.26E-12 | 1.69E-10 | 6.66E-09 | 5.43E-08 | 0 |
| 200-300 | 0 | 3.90E-15 | 6.16E-13 | 2.34E-11 | 0 |
| > 300 | 0 | 0 | 0 | 0 | 0 |

<u>**Step 4**</u>

Using de-aggregated median hazard results, the fractional contribution of each magnitude-distance pair to the total hazard is determined.

Tables C.48 and C.59 show $P(m,d)_1$ and $P(m,d)_2$ for the average of 1 and 2.5 Hz and 5 and 10 Hz, respectively.

Table C.48
$P(m,d)_1$ for Average Spectral Accelerations 1 and 2.5 Hz
Corresponding to the Reference Probability

| Distance Range of Bin (km) | Magnitude Range of Bin | | | | |
|---|---|---|---|---|---|
| | 5 - 5.5 | 5.5 - 6 | 6 - 6.5 | 6.5 - 7 | >7 |
| 0-15 | 0.083 | 0.146 | 0.018 | 0.000 | 0.000 |
| 15-25 | 0.020 | 0.050 | 0.005 | 0.000 | 0.000 |
| 25-50 | 0.009 | 0.067 | 0.029 | 0.000 | 0.000 |
| 50-100 | 0.001 | 0.027 | 0.075 | 0.022 | 0.000 |
| 100-200 | 0.000 | 0.003 | 0.066 | 0.370 | 0.000 |
| 200-300 | 0.000 | 0.000 | 0.001 | 0.008 | 0.000 |
| > 300 | 0.000 | 0.000 | 0.000 | 0.000 | 0.000 |

Table C.59

$P(m,d)_2$ for Average Spectral Accelerations 5 and 10 Hz
Corresponding to the Reference Probability

| Distance Range of Bin (km) | Magnitude Range of Bin | | | | |
|---|---|---|---|---|---|
| | 5 - 5.5 | 5.5 - 6 | 6 - 6.5 | 6.5 - 7 | >7 |
| 0-15 | 0.289 | 0.306 | 0.024 | 0.000 | 0.000 |
| 15-25 | 0.054 | 0.104 | 0.008 | 0.000 | 0.000 |
| 25-50 | 0.012 | 0.075 | 0.032 | 0.000 | 0.000 |
| 50-100 | 0.001 | 0.010 | 0.030 | 0.010 | 0.000 |
| 100-200 | 0.000 | 0.001 | 0.006 | 0.038 | 0.000 |
| 200-300 | 0.000 | 0.000 | 0.000 | 0.000 | 0.000 |
| > 300 | 0.000 | 0.000 | 0.000 | 0.000 | 0.000 |

Step 45

Because the contribution of the distance bins greater than 100 km in Table C.48 **contains** ~~does account for~~ more than 5% of the total hazard for the average of 1 and 2.5 Hz, the controlling earthquake for the spectral average of 1 and 2.5 Hz will be calculated using magnitude-distance bins for distance greater than 100 km. Table C.610 shows $P_{>100}(m,d)_1$ for the average of 1-2.5 Hz.

Table C.610

$P_{>100}(m,d)_1$ for Average Spectral Accelerations 1 and 2.5 Hz
Corresponding to the Reference Probability

| Distance Range of Bin (km) | Magnitude Range of Bin | | | | |
|---|---|---|---|---|---|
| | 5 - 5.5 | 5.5 - 6 | 6 - 6.5 | 6.5 - 7 | >7 |
| 100-200 | 0.000 | 0.007 | 0.147 | 0.826 | 0.000 |
| 200-300 | 0.000 | 0.000 | 0.002 | 0.018 | 0.000 |
| > 300 | 0.000 | 0.000 | 0.000 | 0.000 | 0.000 |

Figures C.1 to C.3 show the above information in terms of the relative percentage contribution.

Steps ~~5~~6 and ~~6~~7

To compute the controlling magnitudes and distances at 1-2.5 Hz and 5-10 Hz for the example site, the values of $P_{>100}(m,d)_1$ and $P(m,d)_2$ are used with m and d values corresponding to the mid-point of the magnitude of the bin (5.25, 5.75, 6.25, 6.75, 7.3) and centroid of the ring area (10, 20.4, 38.9, 77.8, 155.6, 253.3, and somewhat arbitrarily 350 km). Note that the mid-point of the last magnitude bin may change because this value is dependent on the maximum magnitudes used in the hazard analysis. For this example site, the controlling earthquake characteristics (magnitudes and distances) are given in Table C.~~7~~11.

Table C.~~7~~11

Magnitudes and Distances of Controlling Earthquakes from the LLNL Probabilistic Analysis

| 1-2.5 Hz | 5 - 10 Hz |
|---|---|
| $M_c$ and $D_c$ > 100 km | $M_c$ and $D_c$ |
| 6.7 and 157 km | 5.7 and 17 km |

Step ~~7~~8

The SSE response spectrum is determined by the procedures described in Appendix F.

## C.4 SITES NOT IN THE CEUS

The determination of the controlling earthquakes and the seismic hazard information base for sites not in the CEUS is also carried out using the procedure described in Section C.2 of this appendix. However, because of differences in seismicity rates and ground motion attenuation at these sites, alternative magnitude-distance bins may have to be used. In addition, as discussed in Appendix B, an alternative reference probability may also have to be developed, particularly for sites in the active plate margin region and for sites at which a known tectonic structure dominates the hazard.

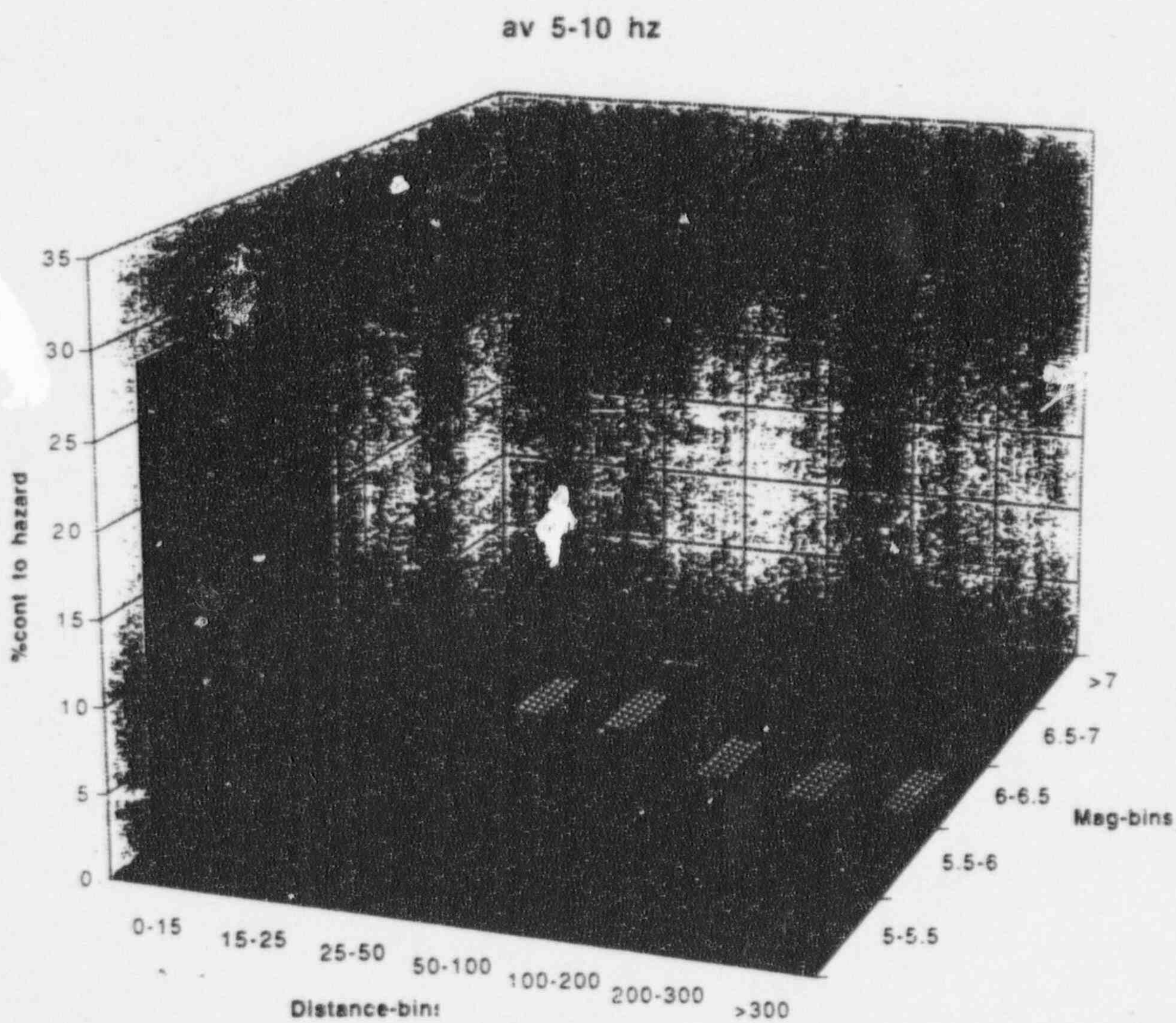


Figure C.1 Full Distribution for Average of 5 and 10Hz

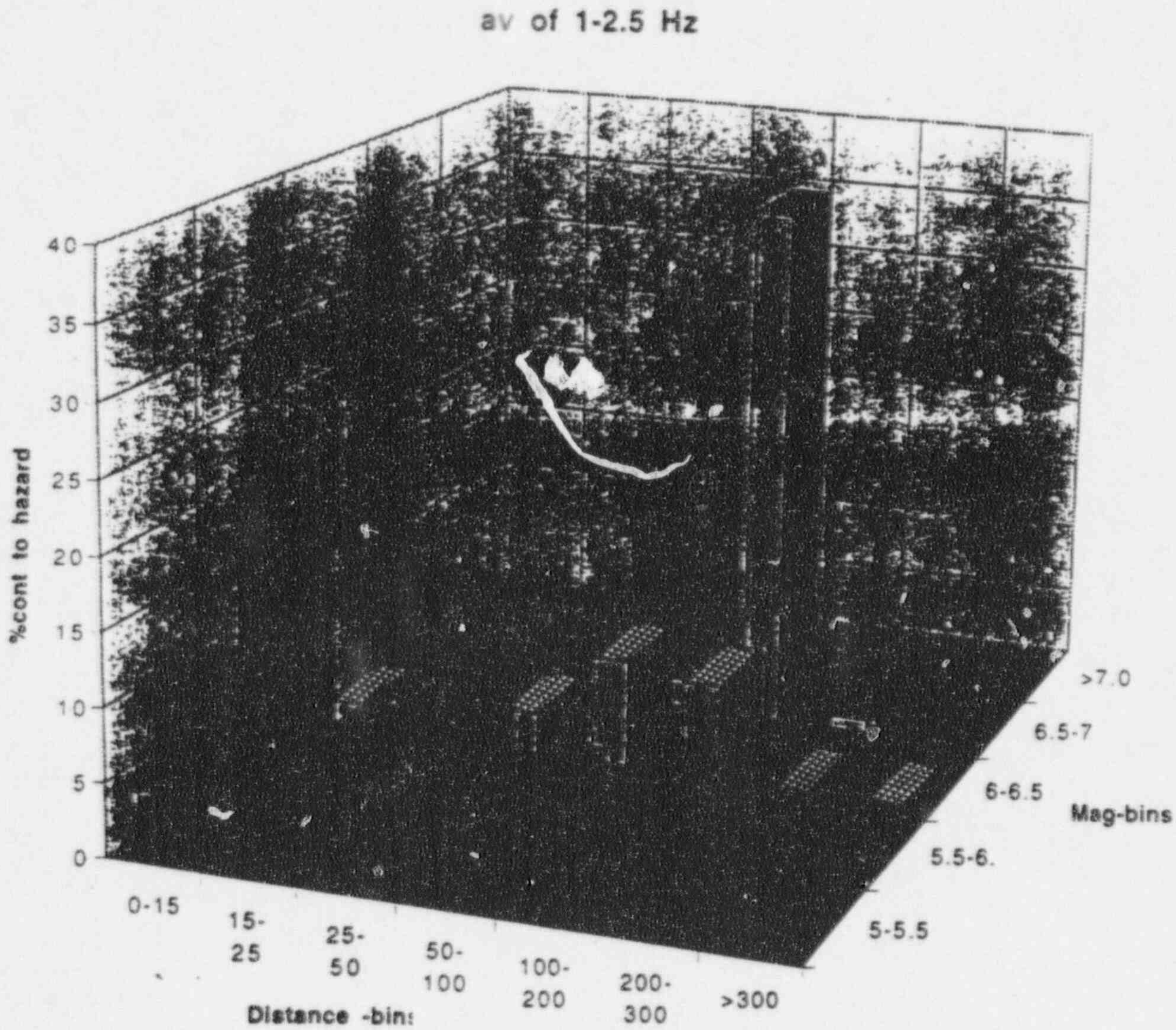


Figure C.2 Full Distribution for Average of 1 and 2.5Hz

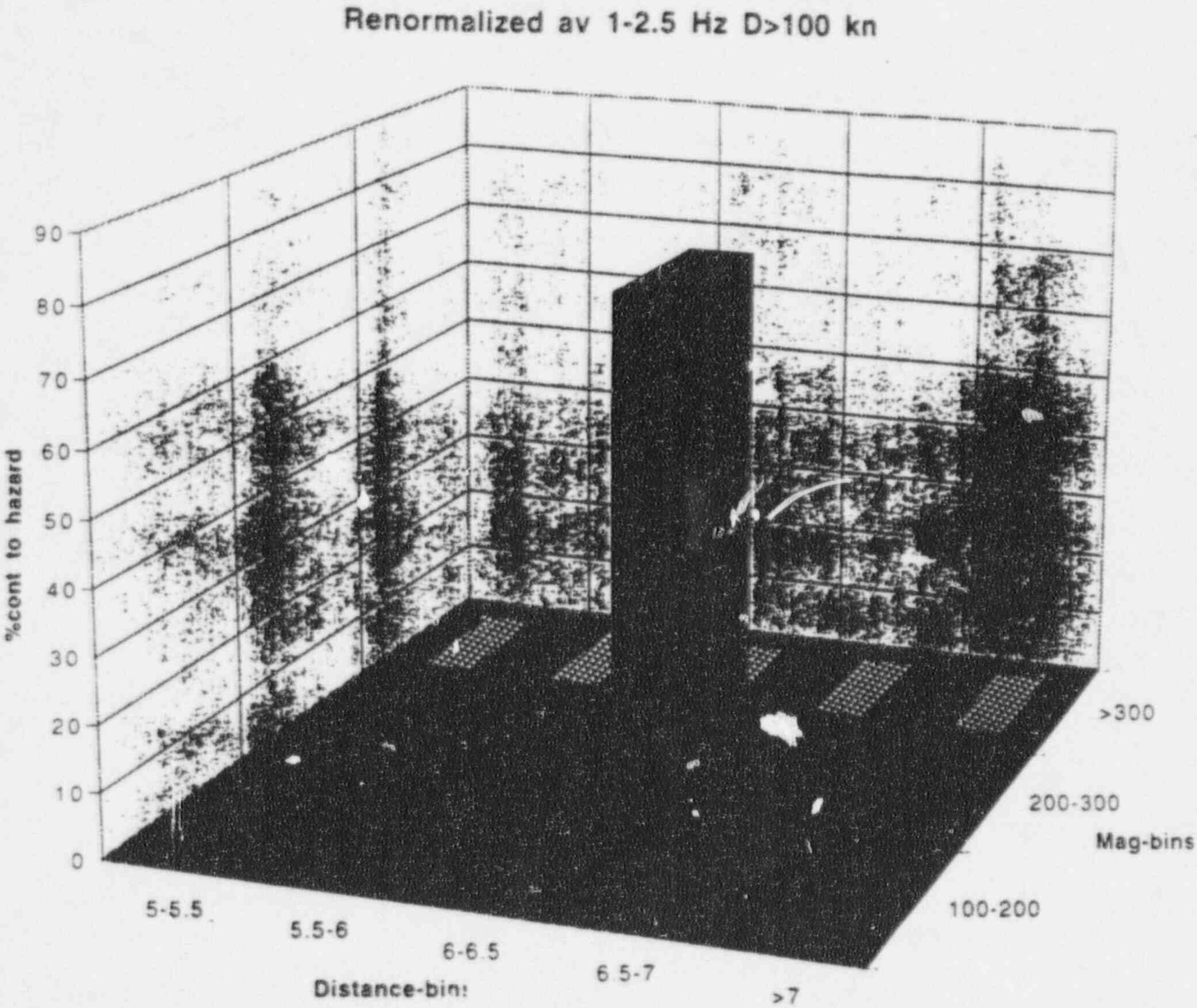


Figure C.3 Renormalized Hazard Distribution for Distances > 100km for average of 1 and 2.5Hz

## REFERENCES

C.1 P. Sobel, "Revised Livermore Seismic Hazard Estimates for Sixty-Nine Nuclear Power Plant Sites East of the Rocky Mountains," NUREG-1488, USNRC, April 1994.[1]

C.2 J.B. Savy et al., "Eastern Seismic Hazard Characterization Update," UCRL-ID-115111, Lawrence Livermore National Laboratory, June 1993 (Accession number 9310190318 in NRC's Public Document Room).

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[1]Copies are available for inspection or copying for a fee from the NRC Public Document Room at 2120 L Street NW., Washington, DC; the PDR's mailing address is Mail Stop LL-6, Washington, DC 20555; telephone (202)634-3273; fax (202)634-3343. Copies may be purchased at current rates from the U.S. Government Printing Office, P.O. Box 37082, Washington, DC 20402-9328 (telephone (202)512-2249); or from the National Technical Information Service by writing NTIS at 5285 Port Royal Road, Springfield, VA 22161.

APPENDIX D

GEOLOGICAL, SEISMOLOGICAL, AND GEOPHYSICAL INVESTIGATIONS TO CHARACTERIZE SEISMIC SOURCES

## D.1 INTRODUCTION

**As characterized for use in PSHA's,** ~~S~~seismic sources are ~~areas~~ zones within which future earthquakes are likely to occur at ~~similar~~ **the same** recurrence rates. Geological, seismological, and geophysical investigations provide the information needed to identify and characterize source parameters, such as size and geometry, and to estimate earthquake recurrence rates and maximum magnitudes. The amount of data available about earthquakes and their causative sources varies substantially between the Western United States (west of the Rocky Mountain front) and the Central and Eastern United States (CEUS), or stable continental region (SCR) (east of the Rocky Mountain front). Furthermore, there are variations in the amount and quality of data within these regions. In active tectonic regions ~~the focus will be on the identification of~~ **there are** both capable tectonic sources and seismogenic sources**, and because of their relatively high activity rate they may be more readily identified**. In the CEUS, identifying seismic sources is less certain because of the difficulty in correlating earthquake activity with known tectonic structures and the lack of adequate knowledge about earthquake causes**, and the relatively lower activity rate**.

In the CEUS, several significant tectonic structures exist and some of these have been interpreted as potential seismogenic sources (e.g., New Madrid fault zone, Nemaha Ridge, and Meers fault). There is no single recommended procedure to follow to characterize maximum magnitude associated with such candidate seismogenic sources; therefore, it is most likely that the determination of the properties of the seismic source will be inferred rather than demonstrated by strong correlations with seismicity or geologic data. Moreover, it is not generally known what relationships exist between observed tectonic structures in a seismic source within the CEUS and the current earthquake activity that may be associated with that source. Generally, the observed tectonic structure resulted from ancient tectonic forces that are no longer present.~~, thus a structure's extent may not be a very meaningful~~

~~indicator of the size of future earthquakes associated with the source.~~ The historical seismicity record, the results of regional and site studies, and judgment play key roles. If, on the other hand, strong correlations and data exist suggesting a relationship between seismicity and seismic sources, approaches used for more active tectonic regions can be applied.

The primary objective of geological, seismological, and geophysical investigations is to develop an up-to-date, site-specific earth science data base that supplements existing information (Ref. D.1). In the CEUS the results of these investigations will also be used to assess whether new data and their interpretation are consistent with the information used as the basis for accepted probabilistic seismic hazard studies. If the new data are consistent with the existing earth science data base, ~~development of new seismic sources~~ **modification of the hazard analysis** is not required. For sites in the CEUS where there is significant new information (see Appendix E) provided by the site investigation, and for sites in the Western United States, site-specific seismic sources are **to be** determined. It is anticipated that for most sites in the CEUS, new information will have been adequately bounded by existing seismic source interpretations.

The following is a general list of characteristics to be determined for a seismic source for site-specific source interpretations:

- Source zone geometry (location and extent, both surface and subsurface).

- ~~Description of Quaternary (last 2 million years) displacements (sense of slip on the fault, fault length and width, area of the fault plane, age of displacements, estimated displacement per event, estimated magnitude per offset, and displacement history or uplift rate of seismogenic folds).~~

- Historical and instrumental seismicity associated with each source.

- Paleoseismicity.

- Relationship of the potential seismic source to other potential seismic sources in the region.

- Seismic potential ~~Maximum magnitude earthquake that can be generated by~~ of the seismic source, based on the source's known characteristics, including seismicity.

- Recurrence model (Frequency of earthquake occurrence versus magnitude).

- Other factors that will be evaluated, depending on the geologic setting of a site, such as:

  - **Quaternary (last 2 million years) displacements (sense of slip on faults, fault length and width, area of the fault plane, age of displacements, estimated displacement per event, estimated magnitude per offset, segmentation, orientations of regional tectonic stresses with respect to faults, and displacement history or uplift rates of seismogenic folds).**

  - Effects of human activities such as withdrawal of fluid from or addition of fluid to the subsurface, extraction of minerals, or the construction of dams and reservoirs.

  - Volcanism. Volcanic hazard is not addressed in this regulatory guide. It will be considered on a case-by-case basis in regions where this hazard exists.

  - ~~Other factors that can contribute to characterization of seismic sources such as strike and dip of tectonic structures, orientations of regional and tectonic stresses, fault segmentation (along both strike and downdip), etc.~~

D.2. <u>INVESTIGATIONS TO EVALUATE SEISMIC SOURCES</u>

D.2.1 <u>General</u>

Investigations of the site and region around the site are necessary to identify both seismogenic sources and capable tectonic sources and to determine their potential for generating earthquakes and causing surface deformation. If it is determined that surface deformation need not be taken into account at the site, sufficient data to clearly justify the determination should be presented in the application for early site review, construction permit, operating license, or combined license. Generally, any tectonic deformation at the earth's surface within 40 km (25 miles) of the site will require ~~adequate~~ **detailed** examination to determine its significance. Potentially active tectonic deformation within the seismogenic zone beneath a site will have to be assessed using geophysical and seismological methods to determine its significance.

Engineering solutions are generally available to mitigate the potential vibratory effects of earthquakes through design. However, ~~adequate~~ engineering solutions cannot always be demonstrated **to be adequate** for mitigation of the effects of permanent ground displacement phenomena such as surface faulting or folding, subsidence, or ground collapse. For this reason, it is prudent to select an alternative site when the potential for permanent ground displacement exists at the proposed site (Ref. D.2).

In most of the CEUS, as determined from instrumentally ~~determined~~ **located** earthquake hypocenters, tectonic structures at seismogenic depths ~~often~~ **seldom** bear ~~no~~ **any** relationship to geologic structures exposed at the ground surface. Possible geologically young fault displacements either do not extend to the ground surface or there is insufficient geologic material of the appropriate age available to date the faults. Capable tectonic sources are not always exposed at **the** ground surface in the Western United States **(WUS)** as demonstrated by the buried (blind) reverse causative faults of the 1983 Coalinga, 1988 Whittier Narrows, 1989 Loma Prieta, and 1994 Northridge earthquakes. These factors emphasize the need to not only conduct thorough investigations at the ground surface but also in the subsurface to identify structures at seismogenic depths.

The level of detail for investigations should be governed by knowledge of the current and late Quaternary tectonic regime and the geological complexity of the site and region. The investigations should be based on increasing the amount of detailed information as they proceed from the regional level down to the site area (e.g., 320 km to 8 km distance from the site). Whenever faults or other structures are encountered at a site (including sites in the CEUS) either in outcrop or excavations, it is necessary to perform many of the investigations described below to determine whether or not they are capable tectonic sources.

The investigations for determining seismic sources should be divided into three levels, Regional, Site Vicinity, and Site Area. Regional investigations should extend to a distance of 320 km (200 mi) from the site~~,~~ ~~and data should be presented at a scale of 1:500,000 or smaller~~. Site vicinity investigations should be conducted to a distance of 40 km (25 mi) from the site. Investigations of the site area should extend out to a radius of 8 km (5 mi). The specific site should be investigated in detail to a distance of at least 1 km (0.66 mi).

The regional investigations [within a radius of 320 km (200 mi) of the site], should be planned to identify seismic sources and describe the Quaternary tectonic regime. The data should be presented at a scale of 1:500,000 or smaller. The investigations are not expected to be extensive or in detail, but should include a comprehensive literature review supplemented by focused geological reconnaissances based on the results of the literature study (including topographic, geologic, aeromagnetic, and gravity maps, and airphotos). Some detailed investigations at specific locations within the region may be necessary if potential capable tectonic sources, or seismogenic sources that may be significant for determining the SSE, are identified.

The large size of the area for the regional investigations is recommended because of the possibility that all significant seismic sources, or alternate configurations, may not have been enveloped by the LLNL/EPRI data base. Thus, it will increase the chances of: (1) identifying evidence for unkown seismic sources that might extend close enough for earthquake ground motions generated by that source to affect the site, and (2) increase the likelihood of confirming the PSHA's database. Furthermore, because of the relatively aseismic nature of the CEUS, the area should be large enough to include as many historical and instrumentally recorded earthquakes for analysis as reasonably possible. The specified area of study is expected to be large enough to incorporate any previously identified sources that could be analogous to sources that may underlie or be relatively close to the site. In past licensing activities of sites in the CEUS, it has often been necessary, because of the absence of datable horizons overlying bedrock, to extend investigations out many tens or hundreds of kilometers from the site along a structure, or to an outlying analogous structure, in order to locate overlying datable strata or unconformities so that geochronological methods could be applied. This procedure has also been used to estimate the age of a an undatable seismic source in the site vicinity by relating its time of last activity to that of a similar, previously evaluated structure, or a known tectonic episode, the evidence of which may be many tens or hundreds of miles away.

In the WUS it is also often necessary to extend the investigations to great distances (up to hundreds of kilometers) to characterize a major tectonic structure, such as the San Gregorio-Hosgri Fault Zone, the Juan de Fuca Subduction Zone, etc. On the other hand, in the WUS, it is not usually necessary to extend the regional investigations that far in all directions. For example, for a site such as Diablo Canyon, which is near the San Gregorio-Hosgri Fault, it would not be necessary to extend the regional investigations to the east beyond the dominant San Andreas Fault which is about 75 km (45 ~~km~~ mi) from the site; nor to the west beyond the Santa Lucia Banks Fault, which is about 45 km (27 mi). Justification for using lesser distances should be provided.

Reconnaissance level investigations, which may need to be supplemented at specific locations by more detailed explorations such as geologic mapping, geophysical surveying, borings, and trenching, should be conducted in the site vicinity to a distance of 40 km (25 mi) from the site; the data should be presented at a scale of 1:50,000 or smaller.

Detailed investigations should be carried out in the site area within a radius of 8 km (5 mi) from the site and the resulting data should be presented at a scale of 1:5000 or smaller. The level of investigations in the site vicinity should delineate the geologic regime and the potential for tectonic deformation at or near the ground surface. The investigations should use the methods described in subsections D.2.2 and D.2.3 that are appropriate for the tectonic regime to characterize seismic sources.

The site vicinity and site area investigations may be asymmetrical and may cover a larger area than those described above in regions of late Quaternary activity, regions with high rates of historical seismic activity (felt or instrumentally recorded data), or sites that are located near a capable tectonic source such as a fault zone.

Data from investigations at the site (approximately 1 square kilometer) should be presented at a scale of 1:500 or smaller. Important aspects of the site investigations are the excavation and logging of exploratory trenches and the mapping of the excavations for the plant structures, particularly ~~those~~ plant structures that are characterized as Seismic Category I. In addition to geological, geophysical, and seismological investigations, ~~considerable~~ detailed geotechnical engineering investigations as described in Regulatory Guide 1.132 (Ref. D.3) should be conducted at the site.

The investigations needed to assess the ~~integrity~~ suitability of the site with respect to effects of potential ground motions and surface deformation should include determination of (1) the lithologic, stratigraphic, geomorphic, hydrologic, geotechnical, and structural geologic characteristics of the site and the area surrounding the site, including its seismicity and geological history, (2) geological evidence of fault offset or other distortion such as folding at or near ground surface within the site area (8 km radius), and (3) whether or not any faults or other tectonic structures, any part of which are within a radius of 8 km (5 mi) from the site, are capable tectonic sources. This information will be used to evaluate tectonic structures underlying the site area, whether buried or expressed at the surface, with regard to their potential for generating earthquakes and for causing surface deformation at or near the site. Theis part of the evaluation should also consider the possible effects caused by human activities such as withdrawal of fluid from or addition of fluid to the subsurface, extraction of minerals, or the loading effects of dams and reservoirs.

### D.2.2 Reconnaissance Investigations, Literature Review, and Other Sources of Preliminary Information

Regional literature and reconnaissance-level investigations can be planned based on reviews of available documents and the results of previous investigations. Possible sources of information may include universities, consulting firms, and government agencies. A detailed list of possible sources of information is given in Regulatory Guide 1.132 (Ref. D.3).

### D.2.3 Detailed Site Vicinity and Site Area Investigations

The following methods are suggested but they are not all-inclusive and investigations should not be limited to them. Some procedures will not be applicable to every site, and situations will occur that require investigations that are not included in the following discussion. It is anticipated that new technologies will be available in the future that will be applicable to these investigations.

### D.2.3.1 Surface Investigations

Surface exploration needed to assess the neotectonic regime and the geology of the area around the site is dependent on the site location and may be carried out with the use of any appropriate combination of the ~~following~~ geological, geophysical, seismological, and geotechnical engineering techniques **summarized in the following paragraphs and Ref. D.3**~~, but~~. **However,** not all **of these methods** ~~will~~ **must** be carried out at a given site.

D.2.3.1.1. Geological interpretations of aerial photographs and other remote-sensing imagery, as appropriate for the particular site conditions, to assist in identifying rock outcrops, faults and other tectonic features, fracture traces, geologic contacts, lineaments, soil conditions, and evidence of landslides or soil liquefaction.

D.2.3.1.2. Mapping of topographic, geologic, geomorphic, and hydrologic features at scales and **with** contour intervals suitable for analysis, stratigraphy (particularly Quaternary), surface tectonic structures such as fault zones, and Quaternary geomorphic features. For offshore sites, coastal sites, or sites located near lakes or rivers, this includes topography, geomorphology (particularly mapping marine and fluvial terraces), bathymetry, geophysics (such as seismic reflection), and hydrographic surveys to the extent needed for evaluation.

D.2.3.1.3. Identification and evaluation of vertical crustal movements by (1) geodetic land surveying to identify and measure short-term crustal movements (Refs. D.4 and D.5) and (2) geological analyses such as analysis of regional dissection and degradation patterns, marine and lacustrine terraces and shorelines, fluvial adjustments such as changes in stream longitudinal profiles or terraces, and other long-term changes such as elevation changes across lava flows (Ref. D.6).

D.2.3.1.4. Analysis of offset, displaced, or anomalous landforms such as displaced stream channels or changes in stream profiles or the upstream migration of knickpoints (Refs. D.7 - D.12); abrupt changes in fluvial deposits or terraces; changes in paleochannels across a fault (Refs. D.11 and

D.12); or uplifted, downdropped, or laterally displaced marine terraces (Ref. D.12).

D.2.3.1.5. Analysis of Quaternary sedimentary deposits within or near tectonic zones, such as fault zones, including (1) fault-related or fault-controlled deposits including sag ponds, graben fill deposits, and colluvial wedges formed by the erosion of a fault paleoscarp and (2) non-fault-related, but offset, deposits including alluvial fans, debris cones, fluvial terrace, and lake shoreline deposits.

D.2.3.1.6. Identification and analysis of deformation features caused by vibratory ground motions, including seismically induced liquefaction features (sand boils, explosion craters, lateral spreads, settlement, soil flows), mud volcanoes, landslides, rockfalls, deformed lake deposits or soil horizons, shear zones, cracks or fissures (Refs. D.13 and D.14).

D.2.3.1.7. ~~Estimation of the ages of~~ **Analysis of** fault displacements, **such as** by ~~analysis~~ **the interpretion** of the morphology of topographic fault scarps associated with or produced by surface rupture. Fault scarp morphology is useful in estimating age of last displacement **(in conjunction with the appropriate geochronological methods described in Subsection D.2.4**, approximate size of the earthquake, recurrence intervals, slip rate, and the nature of the causative fault at depth (Refs. D.15 - D.18).

D.2.3.2 Seismological Investigations

D.2.3.2.1. Listing of all historically reported earthquakes having Modified Mercalli Intensity (MMI) greater than or equal to IV or magnitude greater than or equal to 3.0 that can reasonably be associated with seismic sources, any part of which is within a radius of 320 km (200 miles) of the site (the site region). The earthquake descriptions should include the date of occurrence and measured or estimated data on the highest intensity, magnitude, epicenter, depth, focal mechanism, and stress drop. Historical seismicity includes both historically reported and instrumentally recorded data. For pre-instrumentally recorded data, intensity should be converted to magnitude, the procedure used to convert it to magnitude should be clearly documented, and epicenters should be determined based on intensity distributions. **Methods** to convert intensity values to magnitudes in the CEUS are described in References D.1, D.19, D.20, and D.21.

D.2.3.2.2. Seismic monitoring in the site area should be established as soon as possible after site selection. For sites in **both** the CEUS **and WUS**, a single large dynamic range, broad-band seismograph**, and a network of short period instruments to locate events should be deployed around the site area.** ~~may be adequate. For sites in the Western United States~~**~~WUS~~**~~, a network of at least five such seismographs would be deployed within 25 km (15 mi) surrounding the site~~.

~~The primary purposes of seismic monitoring are to obtain data from distant earthquakes, to determine site response,~~ **The data obtained by monitoring current seismicity will be used, along with the much larger data base acquired from site investigations, to evaluate site response** and **to** provide **information about whether there are** ~~assurance that there are no~~ significant sources of earthquakes within the site vicinity**, or to provide data by which an existing source can be characterized**. ~~For sites in the Western United States seismic monitoring could help locate any ongoing seismicity that may indicate capable faulting within the site vicinity.~~

Monitoring should be initiated **as soon as practicable at the site, preferably at least** ~~up to~~ five years prior to construction of a nuclear unit at a site and should continue ~~for at least five years following initiation of plant operation~~ **at least until the free field seismic monitoring strong ground motion instrumentation described in Regulatory Guide 1.12 is operational**.

D.2.3.3 Subsurface Investigations

**Ref. [redacted] describes geological, geotechnical, and geophysical investigation techniques that can be applied to explore the subsurface beneath the site and in the region around the site.** Subsurface investigations in the site area and within the site vicinity to identify and define seismogenic sources and capable tectonic sources may include the following investigations.

D.2.3.3.1. Geophysical investigation~~s~~ **that have been useful in the past include, but are not limited to:** ~~such as air~~ magnetic and gravity surveys, seismic reflection and seismic refraction surveys, borehole geophysics, **electrical surveys,** and ground-penetrating radar **surveys**.

D.2.3.3.2. Core borings to map subsurface geology and obtain samples for testing such as ~~examining~~ **determining** the properties of the subsurface soils and rocks and geochronological analysis.

D.2.3.3.3. Excavating and logging of trenches across geological features as part of the neotectonic investigation and to obtain samples for the geochronological analysis of those features.

At some sites, deep soil, bodies of water, or other material may obscure geologic evidence of past activity along a tectonic structure. In such cases, the analysis of evidence elsewhere along the structure can be used to evaluate its characteristics in the vicinity of the site (Refs. D.12 and D.22).

### D.2.4 Geochronology

An important part of the geologic investigations to identify and define potential seismic sources is the geochronology of geologic materials. ~~The NRC is currently supporting a research project to develop a data base on which to base a future regulatory guide on geochronological methods. This guide will contain an up-to-date bibliography of state-of-the-art documents on geochronology. The availability of this guide will be published in the Federal Register.~~ An acceptable classification of dating methods is based on the rationale described in Reference D.23. The following techniques, which are presented according to that classification, are useful in dating Quaternary deposits.

#### D.2.4.1 Sidereal Dating Methods

- Dendrochronology - tree-ring analysis - age range is from modern times to several thousand years (Refs. D.24 and D.25).
- Varve chronology - 0 to 10,000 years (Ref. D.26).

#### D.2.4.2 Isotopic Dating Methods

- Radiocarbon for dating organic materials - 100 to 40,000 (up to 100,000 years using AMS) (Refs. D.27 and D.28).
- Potassium argon for dating volcanic rocks ranging in age from about 100,000 to 10 million years (Refs. D.27 and D.29).
- Argon 39 - Argon 40, for dating relatively unweathered igneous and metamorphic rocks - 100,000 to unlimited upper limit (Ref. D.30)
- Uranium series uses the relative properties of various decay products of $^{238}U$ or $^{235}U$. Ages range from 10,000 to 350,000 years (Ref. D.27). $^{235}U/^{238}U$ can yield between 40,000 and 1,000,000 years (Ref. D.31).
- Uranium Trend - for relatively undisturbed soils ranging in age from 100,000 to 900,000 years (Ref. D.32).

D.2.4.3 Cosmogenic Isotopes - for dating surficial rocks and soils. Nuclides $^{36}Cl$, $^{10}Be$, $^{21}Pb$, and $^{26}Al$ - age range varies within the Quaternary according to isotope tested (Refs. D.33 and D.34).

D.2.4.4 Radiogenic Dating Methods

- Thermoluminescence (TL) - for dating fine-grained eolian and lacustrine, and possibly alluvium and colluvium as well - age range is from 1,000 to 1,000,000 years (Refs. D.27 and D.35).
- Electron spin resonance (ESR) is used for sediments, shells, carbonates, bones, and possibly to date quartz that formed in fault gouge during the fault event - age range is from 50,000 to 500,000 years (Ref. D.36).
- Fission Track - for dating minerals such as zircon and apatite, with fissionable uranium in volcanic rocks - 100 to several million years (Refs. D.27 and D.37).

D.2.4.5 Chemical and Biological Dating Methods

- Obsidian and Tephra Hydration - age range is from 200 to several million years (Ref. D.38).
- Amino Acid Racemization - for fossils, shells, and bones - age range is from 100 to 1,000,000 years (Refs. D.39 and D.40).

- Rock varnish chemistry - cation ratio of manganese, iron, and clay coatings on desert stones - age range is 1,000 to 40,000 years (Ref. D.41). The results of this method are controversial and its use is not recommended pending further validation.

D.2.4.6 <u>Geomorphic Dating Methods</u>

- Soil profile development - for analysis of the upper few meters of stable soils - age range is from 1,000 to 1,000,000 years (Refs. D.27, D.42 through D.47).
- Rock and mineral weathering - for measuring the progression of weathering, such as thicknesses of weathering rind development on the margins of clasts, hornblende etching, limestone solutioning, etc. - age range, depending on material - 10 to 1,000,000 (Ref. D.27).
- Geomorphic position - fluvial and marine terraces, and glacial moraines - 1,000 to 1,000,000 years (Ref. D.48).
- Rate of deposition - lacustrine, playa, and sometimes alluvial deposits - tens to millions of years (Ref. D.26)
- Scarp degradation - works best in coarse unconsolidated alluvium - age range is from 2,000 to 20,000 years (Refs. D.15 and D.49).

D.2.4.7 <u>Correlation Dating Methods</u>

- Lithostratigraphy - correlation of distinctive geologic units between sites - age range is from 0 to 4.5 billion years (Ref. D.50)
- Tephrochronology - volcanic ash layers interbedded with sedimentary deposits - age range is from zero to several million years (Refs. D.51 and D.38).
- Paleomagnetism - most igneous and sedimentary rocks containing hematite and magnetite - age range is from 0 to 5,000,000 years (Ref. D.27).
- Archeology - deposits associated with archeological materials (Ref. D.52).

- Paleontology (marine and terrestial) - fossil-bearing rocks or soils - age range is from 0 to 1 billion years (Ref. D.53).
- Lichenometry - used to estimate ages from sizes of lichens growing on gravel or boulders (such as glacial deposits) (Ref. D.54).

In the CEUS, it may not be possible to reasonably demonstrate the age of last activity of a tectonic structure. In such cases the NRC staff will accept association of such structures with geologic structural features or tectonic processes that are geologically old (at least pre-Quaternary) as an age indicator in the absence of conflicting evidence.

These investigative procedures should also be applied, where possible, to characterize offshore structures (faults or fault zones, and folds, uplift, or subsidence related to faulting at depth) for coastal sites or those sites located adjacent to landlocked bodies of water. Investigations of offshore structures will rely heavily on seismicity, geophysics, and bathymetry rather than conventional geologic mapping methods that can normally be used effectively onshore. However, it is often useful to investigate similar features onshore to learn more about the significant offshore features.

### D.2.5 Distinction Between Tectonic and Nontectonic Deformation

Nontectonic deformation, like tectonic deformation, at a site can pose a substantial hazard to nuclear power plants, but there are likely to be differences in the approaches used to resolve the issues raised by the two types of phenomena. Therefore, nontectonic deformation should be distinguished from tectonic deformation at a site. In past nuclear power plant licensing activities, surface displacements caused by phenomena other than tectonic phenomena have been confused with tectonically induced faulting. Such features include faults on which the last displacement was induced by glaciation or deglaciation; collapse structures, such as found in karst terrain; and growth faulting, such as occurs in the Gulf Coastal Plain or in other deep soil regions subject to extensive subsurface fluid withdrawal.

Glacially induced faults generally do not represent a deep-seated seismic or fault displacement hazard because the conditions that created them are no longer present. However, residual stresses from Pleistocene glaciation may still be present in glaciated regions, although they are of less concern than active tectonically induced stresses. These features should be investigated with respect to their relationship to current in situ stresses.

The nature of faults related to collapse features can usually be defined through geotechnical investigations and can either be avoided or, if feasible, adequate engineering fixes can be provided.

Large, naturally occurring growth faults as found in the coastal plain of Texas and Louisiana can pose a surface displacement hazard, even though offset most likely occurs at a much less rapid rate than that of tectonic faults. They are not regarded as having the capacity to generate damaging **vibratory ground motion** ~~earthquakes~~, can often be identified and avoided in siting, and their displacements can be monitored. Some growth faults and antithetic faults related to growth faults are not easily identified; therefore, investigations described above with respect to capable faults and fault zones should be applied in regions where growth faults are known to be present. Local human-induced growth faulting can be monitored and controlled or avoided.

If questionable features cannot be demonstrated to be of non-tectonic origin they should be treated as tectonic deformation.

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## APPENDIX E

### PROCEDURE FOR THE EVALUATION OF NEW GEOSCIENCES INFORMATION OBTAINED FROM THE SITE-SPECIFIC INVESTIGATIONS

### E.1 INTRODUCTION

This appendix provides methods acceptable to the NRC staff for assessing the impact of new information obtained during site-specific investigations on the database used for the probabilistic seismic hazard analysis (PSHA).

Regulatory Position 4 in this guide describes acceptable PSHA's ~~analyses~~ that were developed by Lawrence Livermore National Laboratories (LLNL) and the Electric Power Research Institute (EPRI) to characterize the seismic hazard for nuclear power plants ~~estimate the controlling earthquakes~~ and to develop the Safe Shutdown Earthquake ground motion (SSE). The procedure to determine the SSE outlined in ~~this Draft~~ Regulatory Guide 1.165 ~~DG-1032~~ relies primarily on either the LLNL or EPRI PSHA results for the Central and Eastern United States (CEUS). It is necessary to evaluate the geological, seismological, and geophysical data obtained from the site-specific investigations to demonstrate that these data are consistent with the PSHA data bases of these two methodologies. If ~~significant differences~~ new information are identified by the site specific ~~between the~~ investigations ~~results~~ ~~that are validated by a strong technical basis~~ ~~and the PSHA data base, are identified and these differences~~ would result in a significant increase in the hazard estimate for a site, and this new information is validated by a strong technical basis, the PSHA may have to be modified to incorporate the new technical information.

In general, major recomputations of the ~~significant revisions to the~~ LLNL and EPRI data base are planned to be ~~only~~ undertaken periodically (approximately every ten years), or when there is an important new finding or occurrence that has, based on sensitivity studies, resulted in a significant increase in the hazard estimate. Using sensitivity studies, it may also be possible to justify a lower hazard estimate with and exceptionally strong technical basis. However, it is expected that large uncertainties in estimating seismic hazard in the CEUS will continue to exist in the future, and substantial delays in the licensing process will result in trying to address them with respect to a specific site.

## E.2 POSSIBLE SOURCES OF NEW INFORMATION THAT COULD AFFECT THE SSE

Types of new data that could affect the PSHA results can be put in three general categories: seismic sources, earthquake recurrence models or rates of deformation, and ground motion models.

### E.2.1 Seismic Sources

There are several possible sources of new information from the site-specific investigations that could ~~e~~affect the seismic hazard. Continued recording of small earthquakes, including microearthquakes, may indicate the presence of a localized seismic source. Paleoseismic evidence, such as paleoliquefaction features or displaced Quaternary strata, may indicate the presence of a previously unknown tectonic structure or a larger amount of activity on a known structure than was previously considered. ~~Future~~ ~~g~~Geophysical studies (aeromagnetic, gravity, and seismic reflection/refraction) ~~will probably~~ may identify crustal structures that suggest the presence of previously unknown seismic sources. In situ stress measurements and the mapping of tectonic structures in the future may indicate potential seismic sources.

Detailed local site investigations often reveal faults or other tectonic structures that were unknown, or reveal additional characteristics of known tectonic structures. Generally, based on past licensing experience in the CEUS, the discovery of such features will not require a modification of the seismic sources provided in the LLNL and EPRI studies. However, initial evidence regarding a newly discovered tectonic structure in the CEUS is often equivocal with respect to activity, and additional detailed investigations are required. By means of these detailed investigations, and based on past licensing activities, previously unidentified tectonic structures can usually be shown to be inactive or otherwise insignificant to the seismic design basis of the facility, and a modification of the seismic sources provided by the LLNL and EPRI studies will not be required. On the other hand, if the newly discovered features are relatively young, possibly associated with ~~historical~~ earthquakes that were large and ~~close to~~ could impact the hazard for the proposed facility, a modification may be required.

Of particular concern is the possible existence of previously unknown, potentially active tectonic structures that could ~~localize~~ have moderatelysized, but potentially damaging, near-field earthquakes or could cause surface displacement. Also of concern is the presence of structures that could generate larger earthquakes within the region.

Investigations to determine whether there is a possibility for permanent ground displacement are especially important in view of the provision to allow for a combined licensing procedure under 10 CFR Part 52 as an alternative to the two-step procedure of the past (Construction Permit and Operating License). In the past at numerous nuclear power plant sites, potentially significant faults were identified when excavations were made during the construction phase prior to the issuance of an operating license, and extensive additional investigations of those faults had to be carried out to properly characterize them.

### E.2.2 Earthquake Recurrence Models

There are three elements of the source zone's recurrence models that could be affected by new site-specific data: (1) the rate of occurrence of earthquakes, (2) their maximum magnitude, and (3) the form of the recurrence model, for example, a change from truncated exponential to a characteristic earthquake model. Among the new site-specific information that is most likely to have a significant impact on the hazard is the discovery of paleoseismic evidence such as extensive soil liquefaction features, which would indicate with reasonable confidence that much larger estimates of the maximum earthquake ~~would ensue~~ than those predicted by the previous studies **would ensue**. The paleoseismic data could also be significant even if the maximum magnitudes of the previous studies are consistent with the paleoseismic earthquakes if there are sufficient data to develop return period estimates significantly shorter than those previously used in the probabilistic analysis. The paleoseismic data could also indicate that a characteristic earthquake model would be more applicable than a truncated exponential model.

In the future, expanded earthquake catalogs will become available that will differ from the catalogs used by the previous studies. Generally, these new catalogues have been shown to have only minor impacts on estimates of the parameters of the recurrence models. Cases that might be significant include the discovery of records that ~~place~~ indicate earthquakes in a region that had no seismic activity in the previous catalogs, the occurrence of an earthquake larger than the largest historic earthquakes, re-evaluating the largest historic earthquake to a significantly larger magnitude, or the occurrence of one or more moderate to large earthquakes (magnitude 5.0 or greater) in the CEUS.

Geodetic measurements, particularly satellite-based networks, may provide data and interpretations of rates and styles of deformation in the CEUS that can have implications for earthquake recurrence. New hypotheses regarding present-day tectonics based on new data or reinterpretation of old data may be developed that were not considered or given high weight in the EPRI or LLNL PSHA. Any of these cases could have an impact on the estimated maximum earthquake if the result is larger than the values provided by LLNL and EPRI.

### E.2.3 Ground Motion Attenuation Models

Alternative ground motion models may be used to determine the site-specific spectral shape as discussed in Regulatory Position 4 and Appendix F of this regulatory guide. If the ground motion models used are a major departure from the original models used in the hazard analysis and are likely to have impacts on the hazard results of many sites, a reevaluation of the reference probability may be needed using the procedure discussed in Appendix B. Otherwise, a periodic (e.g., every ten years) reexamination of PSHA and the associated data base is considered appropriate to incorporate new understanding regarding ground motion models.

## E.3 PROCEDURE AND EVALUATION

The EPRI and LLNL studies provided a wide range of interpretations of the possible seismic sources for most regions of the CEUS, as well as a wide range of interpretations for all the key parameters of the seismic hazard model. The first step in comparing the new information with those interpretations is determining whether the new information is consistent with the following LLNL and EPRI parameters: (1) the range of seismogenic sources as interpreted by the seismicity experts or teams involved in the study, (2) the range of seismicity rates for the region around the site as interpreted by the seismicity experts or teams involved in the studies, and (3) the range of maximum magnitudes determined by the seismicity experts or teams. The new information is considered not significant and no further evaluation is needed if it is consistent with the assumptions used in the PSHA, no additional alternative seismic sources or seismic parameters are needed, or it supports maintaining or decreasing the site median seismic hazard.

An example is an additional nuclear unit sited near an existing nuclear power plant site that was recently investigated by state-of-the-art geosciences techniques and evaluated by current hazard methodologies. Detailed geological, seismological, and geophysical site-specific investigations would be required to update existing information regarding the new site, but it is very unlikely that significant new information would be found that would invalidate the previous PSHA.

On the other hand, after evaluating the results of the site-specific investigations if there is still uncertainty about whether the new information will affect the estimated hazard, it will be necessary to evaluate the potential impact of the new data and interpretations on the median of the range of the input parameters. Such new information may indicate the addition of a new seismic source, a change in the rate of activity, a change in the spatial patterns of seismicity, an increase in the rate of deformation, or the observation of a relationship between tectonic structures and current seismicity. The new findings should be assessed by comparing them with the specific input of each expert or team that participated in the PSHA. Regarding a new source, for example, the specific seismic source characterizations for each expert or team (such as tectonic feature being modeled, source geometry, probability of being active, maximum earthquake magnitude, or occurrence rates) should be assessed in the context of the significant new data and interpretations.

~~Usually~~ It is expected that the new information will be within the range of interpretations in the existing data base, and the data will not result in an increase in overall seismicity rate or increase in the range of maximum earthquakes to be used in the probabilistic analysis. It can then be concluded that the current LLNL or EPRI results apply. It is possible that the new data may necessitate a change in some parameter. In this case, appropriate sensitivity analyses should be performed to determine whether the new site-specific data could affect the ground motion estimates at the reference probability level.

An example is a consideration of the seismic hazard near the Wabash River Valley (Ref. E.1). Geological evidence found recently within the Wabash River Valley and several of its tributaries indicated that an earthquake much larger than any historic event had occurred several thousand years ago in the vicinity of Vincennes, Indiana. A review of the inputs by the experts and teams involved in the LLNL and EPRI PSHA's revealed that many of them had made allowance for this possibility in their tectonic models by assuming the extension of the New Madrid Seismic Zone northward into the Wabash Valley. Several experts had given strong weight to the relatively high seismicity of the area, including the number of magnitude 5 historic earthquakes that have occurred, and thus had assumed the larger event. This analysis of the source characterizations of the experts and teams resulted in the conclusion by the analysts that a new PSHA would not be necessary for this region because an event similar to the prehistoric earthquake had been considered in the existing PSHAs.

A third step would be required if the site-specific geosciences investigations revealed significant new information that would substantially affect the estimated hazard. Modification of the seismic sources would more than likely be required if the results of the detailed local and regional site investigations indicate that a previously unknown seismic source is identified in the vicinity of the site. A hypothetical example would be the recognition of geological evidence of recent activity on a fault near a nuclear power plant site in the stable continental region (SCR) similar to the evidence found on the Meers Fault in Oklahoma (Ref. E.2). If such a source is identified, the same approach used in the active tectonic regions of the Western United States should be used to assess the largest earthquake expected and the rate of activity. If the resulting maximum earthquake and the rate of activity are higher than those provided by the LLNL or EPRI experts or teams regarding seismic sources within the region in which this newly discovered tectonic source is located, it may be necessary to modify the existing interpretations by introducing the new seismic source and developing modified seismic hazard estimates for the site. The same would be true if the current ground motion models are a major departure from the original models. These occurrences would likely require performing a new PSHA using the updated data base, and may require determining the appropriate reference probability in accordance with the procedure described in Appendix B.

REFERENCES

E.1 Memorandum from A. Murphy, NRC, to L. Shao, NRC, Subject: Summary of a Public Meeting on the Revision of Appendix A, "Seismic and Geologic Siting Criteria for Nuclear Power Plants," to 10 CFR Part 100; Enclosure (Viewgraphs): NUMARC, "Development and Demonstration of Industry's Integrated Seismic Siting Decision Process," February 23, 1993.[1]

E.2 A.R. Ramelli, D.B. Slemmons, and S.J. Brocoum, "The Meers Fault: Tectonic Activity in Southwestern Oklahoma," NUREG/CR-4852, US NRC, March 1987.[2]

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[1]Copies are available for inspection or copying for a fee from the NRC Public Document Room at 2120 L Street NW., Washington, DC; the PDR's mailing address is Mail Stop LL-6, Washington, DC 20555; telephone (202)634-3273; fax (202)634-3343.

[2]Copies are available for inspection or copying for a fee from the NRC Public Document Room at 2120 L Street NW., Washington, DC; the PDR's mailing address is Mail Stop LL-6, Washington, DC 20555; telephone (202)634-3273; fax (202)634-3343. Copies may be purchased at current rates from the U.S. Government Printing Office, P.O. Box 37082, Washington, DC 20402-9328 (telephone (202)512-2249); or from the National Technical Information Service by writing NTIS at 5285 Port Royal Road, Springfield, VA 22161.

## APPENDIX F

## PROCEDURE TO DETERMINE THE SAFE SHUTDOWN EARTHQUAKE GROUND MOTION

### F.1 INTRODUCTION

This appendix elaborates on Step 4 of Regulatory Position 4 of ~~Draft Regulatory Guide DG-1032~~**this guide**, which describes an acceptable procedure to determine the Safe Shutdown Earthquake Ground Motion (SSE). The SSE is defined in terms of the horizontal and vertical free-field ground motion response spectra at the free ground surface. It is developed with consideration of local site effects and site seismic wave transmission effects. The SSE response spectrum ~~is~~ **can be** determined by scaling a **site-specific** spectral shape determined for the controlling earthquakes **or by scaling a standard broad-band spectral shape to envelope**~~to~~ the average **of the** ground motion levels for 5 and 10 Hz ($S_{a,5-10}$), and 1 and 2.5 Hz ($S_{a,1-2.5}$) **as** determined in Step C.2 of Appendix C to this guide.

It is anticipated that a regulatory guide will be developed that provides guidance on assessing site-specific effects and determining smooth design response spectra, taking into account recent developments in ground motion modeling and site amplification studies (e.g., Ref. F.1).

### F.2 DISCUSSION

For engineering purposes, it is essential that the design ground motion response spectrum be a broad-band smooth response spectrum with adequate energy in **the** frequencies of interest. In the past, it was general practice to select **a** standard broad-band spectrum, such as the spectrum in Regulatory Guide 1.60 (Ref. F.2), and ~~anchor~~ **scale** it ~~to~~ **by** a peak ground motion parameter (usually peak ground acceleration (PGA)), which is derived based on the size of the controlling earthquake. During the licensing review this spectrum was checked against site-specific spectral estimates derived using Standard Review Plan 2.5.2 procedures to be sure that the SSE design spectrum adequately enveloped the site-specific spectrum. These past practices to define the SSE are still valid and, based on this consideration, the following three possible situations are depicted in Figures F.1 to F.3.

Figure F.1 depicts a situation in which a site is to be used for a certified design with an established SSE (for instance, an Advanced Light Water Reactor with 0.3g PGA SSE). In this example, the certified design SSE spectrum compares favorably with the site-specific response spectra determined in Step 2 or 3 of Regulatory Position 4.

Figure F.2 depicts a situation in which a standard broad-band shape is selected and its amplitude is scaled so that the design SSE envelopes the site-specific spectra.

Figure F.3 depicts a situation in which a specific smooth shape for the design SSE spectrum is developed to envelope the site-specific spectra. In this case, it is particularly important to be sure that the SSE contains adequate energy in the frequency range of engineering interest and is sufficiently broad-band.

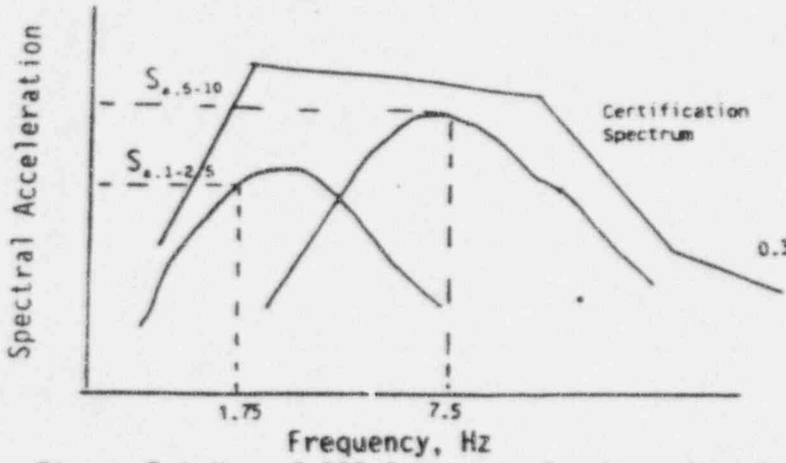


Figure F.1 Use of SSE Spectrum of a Certified Design

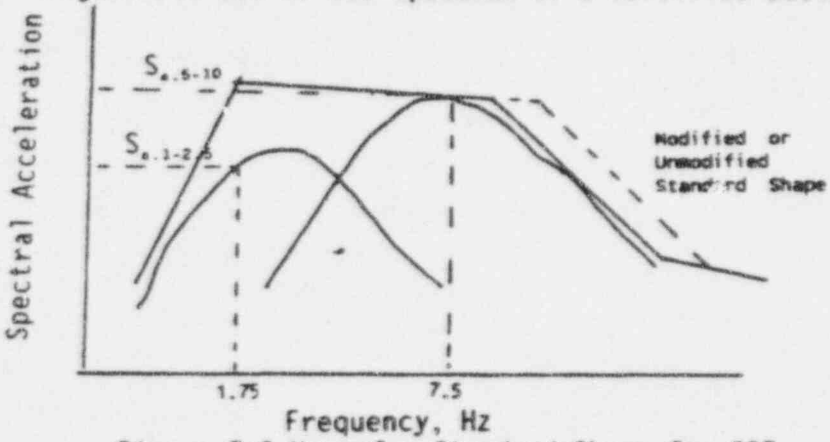


Figure F.2 Use of a Standard Shape for SSE

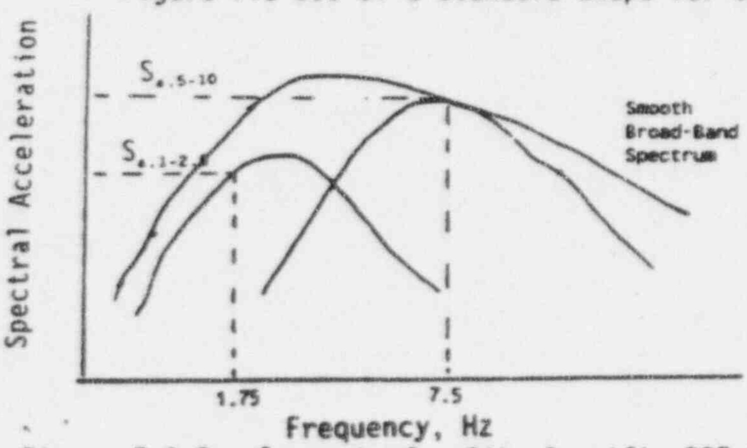


**Figure F.3 Development of a Site-Specific SSE Spectrum**

(Note: The above figures illustrate situations for a rock site, for other site conditions the SSE spectra are compared at free-field after performing site amplification studies as discussed in Step 4 of Regulatory Position 4)

## REFERENCES


F.1 Electric Power Research Institute, "Guidelines for Determining Design Basis Ground Motions," EPRI Report TR-102293, Volumes 1-4, May 1993.

F.2 USNRC, "Design Response Spectra for Seismic Design of Nuclear Power Plants," Regulatory Guide 1.60.[1]


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[1]Copies are available for inspection or copying for a fee from the NRC Public Document Room at 2120 L Street NW., Washington, DC; the PDR's mailing address is Mail Stop LL-6, Washington, DC 20555; telephone (202)634-3273; fax (202)634-3343. Copies may be purchased at current rates from the U.S. Government Printing Office, P.O. Box 37082, Washington, DC 20402-9328 (telephone (202)512-2249); or from the National Technical Information Service by writing NTIS at 5285 Port Royal Road, Springfield, VA 22161.

## REGULATORY ANALYSIS

A separate regulatory analysis was not prepared for this regulatory guide. The ~~draft~~ regulatory analysis, "~~Proposed~~ Revision of 10 CFR Part 100 and 10 CFR Part 50," was prepared for the ~~proposed~~ amendments, and it provides the regulatory basis for this guide and examines the costs and benefits of the rule as implemented by the guide. A copy of the ~~draft~~ regulatory analysis is available for inspection and copying for a fee at the NRC Public Document Room, 2120 L Street NW. (Lower Level), Washington, DC, as ~~Enclosure 2 to Secy 94 194~~ LATER.

**SRP SECTION 2.5.1, REVISION 3**

**(BASIC GEOLOGIC AND SEISMIC INFORMATION)**

U.S. NUCLEAR REGULATORY COMMISSION
STANDARD REVIEW PLAN 2.5.1 ~~February 1995~~
BASIC GEOLOGIC AND SEISMIC INFORMATION ~~Contact: A.J. Murphy~~
~~PROPOSED~~ REVISION 3 ~~(301)415-6010~~

REVIEW RESPONSIBILITIES

Primary - Civil Engineering and Geosciences Branch (ECGB)

Secondary - None

## I. <u>AREAS OF REVIEW</u>

ECGB reviews the geological, seismological, and geophysical information submitted in the applicant's early site evaluation report (ESR) or safety analysis report (SAR), Sections 2.5.1, 2.5.2 and 2.5.3. Because there is a strong overlap among these areas of review and those of geotechnical engineering and geohydrology, the reviewers of these sections of the SARs should also carefully review SRP Section 2.5.4 and Section 2.4.12, and closely coordinate their reviews and findings with those of the geotechnical engineering and the geohydrology reviewers. For example, coordination with geotechnical engineers is required when verification of geological processes affecting the site, such as the preloading history of the plant's soil foundations by means of glacial and other geologic processes, can be determined through various geotechnical testing methodologies.



~~This standard review plan is being issued in draft form to involve the public in the early stages of its development. It has not received complete staff review and does not represent an official NRC staff position.~~

~~Public comments are being solicited on this draft standard review plan, which is part of a group of drafts of regulatory guides and standard review plan sections on meeting proposed amendments to the regulations on siting nuclear power plants (59 FR 52255). Comments should be accompanied by appropriate supporting data. Written comments may be submitted to the Rules Review and Directives Branch, DFIPS, Office of Administration, U.S. Nuclear Regulatory Commission, Washington, DC 20555. Copies of comments received may be examined at the NRC Public Document Room, 2120 L Street NW., Washington, DC. Comments will be most helpful if received by May 12, 1995.~~

~~Requests for single copies of this standard review plan (which may be reproduced) will be filled while supplies last. Requests should be in writing to the U.S. Nuclear Regulatory Commission, Washington, DC 20555, Attention: Office of Administration, Distribution and Mail Services Section.~~

References 1 through 8 (regulations and regulatory guides) provide guidance to the ECGB reviewers in evaluating potential nuclear facility sites. The principal regulation that will be used by ECGB in the future to determine the scope and adequacy of the submitted geological, seismological, and geophysical information for new nuclear facility sites is 10 CFR Part 100, ~~Proposed~~ Section 100.23, "Seismic and Geologic Siting Factors" (Ref. 2). Specific guidance for implementing this regulation can be found in ~~Draft~~ Regulatory Guide ~~DG-1032~~1.165, "Identification and Characterization of Seismic Sources and Determination of Safe Shutdown Earthquake Ground Motions" (Ref. 3). Guidance regarding the geotechnical engineering aspects is found in Regulatory Guide 1.132, "Site Investigations for Foundations of Nuclear Power Plants" (Ref. 4). Additional guidance is provided to the ECGB reviewers through information published in the scientific literature. As the state of the art in the geosciences is advancing rapidly, it is the responsibility of the reviewers to stay abreast of changes by reviewing the current scientific literature on a regular basis, attending professional meetings, etc.

Using the knowledge derived from these activities and the geosciences reviewers' own aggregate academic backgrounds and experience, ECGB judges the adequacy of the geological, seismological, and geophysical information cited in support of the applicant's conclusions concerning the suitability of the plant site.

The geological, seismological, and geophysical information that must be provided by applicants for the site review to proceed is divided into the following three basic categories:

1. Tectonic or seismic information. Information regarding tectonics, (particularly Quaternary tectonics), seismicity, correlation of seismicity with tectonic structure, characterization of seismic sources, and ground motion. Seismicity and vibratory ground motions are primary review responsibilities addressed in SRP Section 2.5.2. However, the review and acceptance of the applicant's basic data-gathering processes and findings that are presented in support of these topics, and their completeness, are also integral parts of the review responsibilities covered in this section. There must be close coordination among

geologists, geophysicists, and seismologists in reviewing these sections.

Sufficient information must be provided to estimate the potential for strong earthquake ground motions or surface deformation at the site, such as the proximity and nature of potential seismic sources, Quaternary geological evidence for faulting, folding, prehistoric earthquakes (i.e., paleoliquefaction features), and other seismically induced features. A complete presentation, including supporting basic data, of the characteristics of the subsurface materials beneath the site must be provided (or cross-referenced with Standard Review Plan (SRP) Section 2.5.4) and reviewed by the staff so that an assessment of the potential for amplification of vibratory ground motion or ground failure under dynamic loading can be made. Potential ground failure modes may include liquefaction, excessive settlement, differential settlement, and those caused by high tectonic stresses. Additionally, for sites adjacent to large bodies of water, information pertinent to estimating tsunami and seiche hazards must be provided, or cross-referenced to SRP Section 2.4.12.

2. Nontectonic deformation information. Adequate information must be provided for an assessment of other nontectonic geological hazards, such as landsliding and other mass-wasting phenomena, subsidence (including differential subsidence), growth faulting, glacially induced deformation, chemical weathering, the potential for collapse or subsidence in areas underlain by carbonate rocks, evidence of preconsolidation, etc.

3. Conditions caused by human activities. Information on changes in groundwater conditions caused by the withdrawal or injection of fluids, subsidence or collapse caused by withdrawal of fluids, mineral extraction, induced seismicity and fault movement caused by reservoir impoundment, fluid injection or withdrawal must be included in the SAR or ESR and evaluated by the ECGB staff.

Acceptance Criteria related to the above conditions are presented in SAR Subsections 2.5.1.1 (Regional Geology) and 2.5.1.2 (Site Geology). This information should be reviewed in terms of the regional and site tectonics, with emphasis on the Quaternary period, structural geology, physiography, geomorphology, stratigraphy, and lithology. 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, solutioning, jointing, seismicity, and fracturing.

The information provided should be documented by appropriate references to all relevant published and unpublished materials. Illustrations such as maps and cross sections should include but should not be limited to structural, tectonic, physiographic, topographic, geologic, gravity, and magnetic maps; structural and stratigraphic sections; boring logs; and aerial photographs. Some sites may require 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.

Maps should include superimposed plot plans 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 photographs, satellite imagery, and geophysical data should be presented for evaluation. In addition, plot plans showing the locations of all plant structures, borings, trenches, profiles, etc. should be included.

The review can be brought to an earlier conclusion if the ESR or SAR contains sufficient data to allow the reviewers to make an independent assessment of the applicant's conclusions. The reviewers 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. A literature search will be conducted by the staff at the appropriate level of detail, depending on the completeness of the SAR or ESR. All pertinent data, including that which is controversial, should be presented and evaluated. The geologic terminology used should conform to standard reference works (Refs. 9 and 10).

The primary purposes for conducting the site and regional investigations are to determine the geological and seismological suitability of the site and to provide the bases for the design of the plant. A secondary goal is to determine whether there is significant new tectonic or ground motion information that could impact the seismic design bases as determined by a probabilistic seismic hazard analysis (PSHA) (Refs. 11, 12, and 13). The objective of Section 2.5.1 of the SAR is to present the results of these investigations and to describe geologic and seismic features as they affect the site under review; all data, information, discussions, interpretations, and conclusions should be directed to this objective.

## II. ACCEPTANCE CRITERIA

The applicable rules and basic acceptance criteria pertinent to the areas of this section of the SRP are given below:

1. 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants," General Design Criterion (GDC) 2, "Design Bases for Protection Against Natural Phenomena," - The criterion requires that safety-related portions of the structures, systems, and components important to safety be designed to withstand the effects of earthquakes, tsunami, and seiche without loss of capability to perform their safety functions (Ref. 1).

2. 10 CFR Part 100, ~~Proposed~~ Section 100.23, "Geologic and Seismic Siting Factors" ~~(59 FR 52255)~~ - This ~~proposed~~ section of Part 100 ~~would~~ requires that the geological, seismological, geophysical, and geotechnical engineering characteristics of a site and its environs be investigated in sufficient scope and detail to permit an adequate evaluation of the proposed site, to provide sufficient information to support evaluations performed to arrive at estimates of the Safe Shutdown Earthquake ground motion (SSE), to preclude sites with potential surface or near-surface tectonic deformation, and to permit adequate engineering solutions to actual or assumed geologic and seismic effects at the proposed site. It ~~would~~ requires the determination of the SSE, the potential for surface tectonic and nontectonic deformations, the design bases for seismically induced floods and water waves, and other design conditions (Ref. 2).

The following regulatory guides provide information, recommendations, and guidance, and in general, describe a basis acceptable to the staff for implementing the requirements of GDC 2, Part ~~100~~ **50**, and Section 100.23 of Part 100.

a. ~~Draft~~ Regulatory Guide ~~DG-1032~~**1.165**, "Identification and Characterization of Seismic Sources and Determination of Safe Shutdown Earthquake Ground Motions" (Ref.3) - This ~~proposed~~ guide describes acceptable methods to: (1) conduct geological, seismological, and geophysical investigations of the site and region around the site, (2) identify and characterize seismic sources, (3) perform probabilistic seismic hazard analyses (PSHA), and (4) determine the SSE for the site (see SRP Section 2.5.2.6 and Ref. 14).

b. Regulatory Guide 1.132, "Site Investigations for Foundations of Nuclear Power Plants" - This guide describes ~~programs of~~ site investigations related to geotechnical aspects that would normally meet the needs for evaluating the safety of the site from the standpoint of the performance of foundations under anticipated loading conditions, including earthquakes. It provides general guidance and recommendations for developing site-specific investigation programs as well as specific guidance for conducting subsurface investigations, including borings, sampling, and geophysical explorations (Ref. 4).

c. Regulatory Guide 4.7, "General Site Suitability Criteria for Nuclear Power Stations" - This guide discusses the major site characteristics related to public health and safety that the NRC staff considers in determining the suitability of sites for nuclear power stations (Ref. 5).

The information in the SAR **or ESR** must be complete and thoroughly documented, and it must be consistent with the requirements of Reference 2 and should conform to the format suggested in Reference 6. Information from varied sources, including the United States Geological Survey (USGS) and other Federal or State agencies' published and open file papers, maps, aerial photographs, geophysical data, and similar data from nongovernmental sources 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 or ESR.

The ECGB reviewers must ensure that investigations, as described in ~~Draft~~ Regulatory Guide ~~DG-1032~~1.165 and Regulatory Guide 1.132, are conducted with the appropriate level of thoroughness within the 4 areas designated in ~~Draft~~ Regulatory Guide 1.165 ~~DG-1032~~, based on distances from the site: 320 km (200 mi), 40 km (25 mi), 8 km (5 mi), and 1 km (0.6 mi). There must be sufficient information presented in the ESR or SAR on which to base a comparison between the new data derived from the regional and site investigations and that used in the tectonic and ground motion models of the probabilistic seismic hazard analysis (Ref. 3).

Specific criteria necessary to meet the relevant requirements of General Design Criterion 2, of Part 100, ~~Appendix A,~~ and ~~Proposed~~ Section 100.23 are as follows:

<u>Subsection 2.5.1.1</u>, "Regional Geology." In meeting the requirements of References 1 and 2, the subsection will be considered acceptable if a complete and documented discussion is presented of all geological, seismological, and geophysical features, as well as conditions caused by human activities. This subsection should contain a review of the regional tectonics, with emphasis on the Quaternary period, structural geology, seismology, paleoseismology, physiography, geomorphology, stratigraphy, and geologic history within a distance of 320 km (200 mi) (site region) from the site, to provide a framework within which the safety significance can be evaluated of the geology, seismology, and conditions brought about by human activities.

<u>Subsection 2.5.1.2</u>, "Site Geology." In meeting the requirements of References 1 and 2, and the regulatory positions of References 4 and 5 and certain recommendations of Reference 7, the subsection will be judged acceptable if it contains a description and evaluation of site-related geologic features, seismic conditions, and conditions caused by human activities, at appropriate levels of detail (defined by the distances of 40 km (25 mi) (site subregion),

8 km {5 mi} (site vicinity), and 1 km (site area) of the site). This subsection should contain the following general site information:

1. The structural geology of the site, specifically the identification and characterization of local seismic sources and their relationship to the regional structural geology and seismic sources.

2. The seismicity of the site, including historical and instrumentally recorded earthquakes, and whether there is a relationship to tectonic structure.

3. The geological history, particularly the Quaternary period, of the site and its relationship to the regional history.

4. Evidence of paleoseismicity or lack of it.

5. The site stratigraphy and lithology and their relationship to those of the region.

6. The engineering significance of geological 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 human activities in the area.

7. The site groundwater conditions.

III. <u>REVIEW PROCEDURES</u>

The staff review is conducted in three phases. The first phase is the acceptance review, a brief review of the SAR or ESR to evaluate its completeness and to identify obvious safety issues that could result in delays at subsequent stages of the review. The judgments on acceptance or rejection of the SAR or ESR for review are governed by two criteria: (1) adherence to the Standard Format (Ref. 6) in identifying and describing the geological, seismological, and geophysical features and the conditions resulting from human activities that affect safety of the site, and (2) provision of adequate information and documentation as described in ~~Draft~~ Regulatory Guide 1.165 ~~DG-1032~~ to allow for an independent staff review of the conclusions made therein.

After an SAR or ESR 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 carefully examines the SAR or ESR 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, such as that published in the scientific literature.

At the beginning of this phase of the review, the staff usually seeks assistance from the U.S. Geological Survey (USGS) and decides to what extent consultants should be involved. The necessary information is then made available to the USGS advisors and consultants. Advisors from the USGS and consultants are asked to perform such varied tasks as reviewing the tectonic setting of plants in regions of complex geology, evaluating the potential for surface displacement, verifying an applicant's mineral identifications and geochronology, or providing advice on the proper level of earthquake ground motion in the seismic evaluation of selected sites.

A review of relevant references is conducted by the staff, USGS advisors, and consultants. 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. Several basic general references used in the past by the staff are References 9, 15, and 16. GeoRef database (Ref. 17) and other databases, such as References 18 and 19, are used to identify specific references.

As publication usually lags behind the completion of research or construction investigation projects by months or years, the reviewers should not rely entirely on information submitted by the applicant or in the published literature. The reviewers should make an effort to identify any pertinent studies that may be under way in the site region and any preliminary findings of these studies. This may be accomplished by contacting the U.S. Geological Survey or other Federal agencies, State geological surveys, universities, and industry, to obtain current information about the site. Some pertinent information may be of a proprietary nature, and special provisions may be required to examine the data.

The staff members will conduct a geological reconnaissance of the site and region around the site as part of the second phase of the review to examine geological features, soil and rock samples from core borings or test pits, trenches excavated across the site, and actual excavations for the plant facilities, if present at this stage. This site reconnaissance is especially important in view of the revised requirement of 10 CFR Part 52 (Ref. 8), which allows for a combined license as an alternative to the previous two-step requirement of a construction permit followed by an operating license. In the previous procedure, many geologic features, such as faults (as at North Anna, Summer, Byron, Catawba, Seabrook, Watts Bar, etc.) that had the potential to impact the safety of the plant were not identified until the actual construction excavations for the plant were made. Additionally, unanticipated engineering problems have occurred during and after construction (as at North Anna, WNP-2, Nine Mile Point-2). For example, larger-than-expected settlements have frequently occurred in engineered backfill, even though the design had been approved by the staff during the construction permit review. Under 10 CFR Part 52 ~~it is possible that~~ the construction excavations for a plant will not be made until after the staff has prepared the site SER.

During the second phase of the review, 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 first round questions usually require the applicant to conduct additional investigations or to supply clarifying information. Questions may result from the reviewer's discovery of references not cited by the applicant that contain conclusions that are in conflict with those made by the applicant. When the applicant provides insufficient data to support its interpretations and conclusions and there are reasonable, technically supported, ~~and~~ more conservative alternative interpretations in the literature, the staff will request additional investigations or require that the applicant adopt the more conservative interpretation. This phase of the review will usually involve public meetings with the applicant to clarify questions and allow the applicant to present new data to justify its position. The applicant's response to questions are reviewed and any remaining issues are settled either by a second round of questions or by staff positions.

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.

A staff position is usually in the form of a requirement to provide confirmatory information or to design for a specific condition in a way that the staff considers to be sufficiently conservative and consistent with the requisites of Reference 2. When all safety issues have been resolved, the staff provides its input to the safety evaluation report (SER).

A staff position that has characterized licensing during the past two decades is that all Category 1 excavations are required to be geologically mapped by the applicant and examined by the staff before backfill is placed or concrete poured. These activities were usually accomplished before the SER was made final. This procedure should continue in the future regarding sites that are licensed under the 10 CFR Part 50 two-phase, Construction Permit and Operating Licensing, procedure.

However, ~~Under~~ the new 10 CFR Part 52 combined licensing procedure (COL), as described above, geological features such as faults that ~~were~~ are not discovered until after the construction excavations are made, and therefore after the ~~SER~~ has been ~~prepared~~ issued, ~~would~~ will not have been assessed by the staff. Likewise, unanticipated engineering problems such as the presence of liquefiable materials, excessive settlement, heave, or groundwater flow that occur~~red~~ during or following construction ~~would~~ **will** not have been evaluated by the staff. For these reasons, **there must be a commitment in the site specific portion of the SAR for a facility: (1) notify the staff immediately if previously unknown geologic features that could represent a hazard to the plant are encountered during excavation; (2) geologically map all excavations for Category 1 structures, as a minimum; and (3) notify the staff when the excavations are open for its examination and evaluation.** ~~conditions should be included in the SER that t~~The staff **should** conduct a followup site review when the excavations for the Seismic Category I ~~facilities~~ **structures** are open to confirm ~~tentative~~ **the** conclusions **that the site parameters are within the envelope of the certified design**. ~~presented in the SER., and that final conclusions by the staff are pending the results of this site review unless there is reasonable certainty that such occurrences are unlikely.~~

## IV. EVALUATION FINDINGS

If the evaluation by the staff, on completion of the review of the geological and seismological aspects of the plant site and region, confirms that the applicant has met the requirements of applicable portions of References 1 and 2. and the guidance contained in References 3, 4, 5, and 6, the conclusion in the SER states that the information provided and investigations performed support the applicant's conclusions regarding the geological and seismological integrity of the proposed nuclear power plant site. **Licensing conditions instituted by the staff to resolve** ~~Staff reservations about~~ any significant deficiency ~~presented~~ **identified** in the applicant's SAR or ESR are stated in sufficient detail to make clear the precise nature of concern **and required resolution**.

The evaluation determinations **with respect to the geological and seismological suitability of the site** are made by the staff after the early site, construction permit, or operating license reviews. **A conclusion regarding an Operating License will include an evaluation of the excavations for Category I structures.** A ~~similar~~ conclusion **regarding the geological and seismological suitability of a site** following a combined license review **will be made when the applicant has committed to mapping excavations for Category 1 facilities and notifying the staff of their availability for examination.** ~~should not be~~

~~tentative finalized until after~~ ~~t~~The staff will conduct this examination at the appropriate time after licensing ~~as the excavations for the seismic category 1 facilities and~~ to confirm ~~determines~~ that there are no previously unknown features, such as potentially active faults, evidence for strong ground motions such as late Quaternary seismically induced paleoliquefaction features, unsuitable soil zones, or cavities in the excavations. ~~There may be additional questions that arise because of this examination. However, documentation of the staff's final conclusions should be made as soon after the excavation examination as possible.~~

This final staff visit, in addition to determining whether there is any new information since the combined licensing review, ensures that the staff recommendations or ~~positions~~ conditions formulated by the staff during the combined licensing review have been implemented.

A typical staff finding at the conclusion of the ~~combined~~ licensing review follows:

> In its review of the geological and seismological aspects of the plant, the staff has considered pertinent information gathered in support of the application for a ~~combined~~ license. The information reviewed includes data from site and near-site investigations, as well as a geological reconnaissance of the site and region, an independent review of recently published literature, and discussions with knowledgeable scientists with the USGS and other Federal agencies, the State Geological Survey, local universities, consulting firms, etc.
>
> Based on its review, the staff concludes that:
>
> (1) The results of ~~G~~geological, geophysical and seismological investigations~~,~~ and other information provided by the applicant and required by ~~the Proposed~~ Section 100.23 ~~to~~ of 10 CFR Part 100~~;~~, the staff's independent review of the data and other sources of information, ~~and~~ including a geological reconnaissance of the site and region and examination of excavations for Seismic Category I structures at the site ~~by the staff~~, provide an adequate basis to establish that no capable tectonic sources or

seismogenic sources exist in the plant site area that have the potential of causing near-surface displacement or earthquakes to be centered there.

(2) Based on the results of the applicant's regional and site geological, seismological, and geophysical investigations, and the staff's independent evaluation ~~(which is conducted primarily by the reviewer of Section 2.5.2 but supported by the reviewer of this section)~~, the staff concludes that all seismic sources significant to determining the SSE for the site have been identified and appropriately characterized by the applicant in accordance with~~ Draft~~ Regulatory Guide ~~DG-1032~~1.165 and SRP Section 2.5.2.

(3) Based on the applicant's geological, geophysical, and geotechnical investigations of the site vicinity and site area, the staff concludes that the site lithology, stratigraphy, geological history, structural geology, and characteristics of the subsurface soils and rocks have been properly characterized.

(4) There is no potential for the occurrence of other geological events (such as landsliding, collapse or subsidence caused by carbonate solutioning, differential settlement) that could compromise the safety of the site; or the applicant has mitigated such occurrences and has adequately supported the engineering solutions in the SAR.

(5) There is no potential for the effects of human activity, such as subsidence caused by withdrawal or injection of fluids or collapse due to mineral extraction, that compromises the safety of the site; or the applicant has taken steps to prevent such occurrences and has adequately supported these actions in the SAR.

(6) If this is a combined license review, **the staff states that** the conclusions stated under (1) above ~~are pending until~~ **will be** confirm**ed** ~~ation by the staff, after~~ **based on** a detailed examination of the walls and floors of the excavations for the

seismic category 1 facilities and the applicant's geological map of these exposures; and an examination by the staff of the applicant's engineering solutions to mitigate any nontectonic geological hazard.

The information reviewed for the proposed nuclear power plant is discussed in Sections 2.5.1, 2.5.2, and 2.5.3.

The staff concluded that the site is acceptable from a geological and seismological standpoint and meets the requirements of (1) 10 CFR Part 50, Appendix A (General Design Criterion 2) and (2) 10 CFR Part 100, ~~Proposed~~ Section 100.23. This conclusion is based on the following:

1. The applicant has met the requirements of:

   a. Appendix A (General Design Criterion 2) of 10 CFR Part 50 with respect to protection against natural phenomena such as earthquakes, faulting, and collapse.

   b. ~~Proposed~~ Section 100.23 (Geologic and Seismic Siting Factors) to 10 CFR Part 100, with respect to obtaining the geologic and seismic information necessary to determine (1) site suitability and (2) the appropriate design of the plant. In complying with this regulation the applicant also meets the staff's guidance described in ~~Draft~~ Regulatory Guide ~~DG-1032~~1.165, "Identification and Characterization of Seismic Sources and Determination of Safe Shutdown Earthquake Ground Motion"; Regulatory Guide 1.132, "Site Investigations for Foundations of Nuclear Power Plants"; and Regulatory Guide 4.7, "General Site Suability Criteria for Nuclear Power Stations."

V. IMPLEMENTATION

The following is intended to provide guidance to applicants and licensees regarding the NRC staff's plans for using this SRP section.

Except in those cases in which the applicant proposes an acceptable alternative method for complying with specified portions of the Commission's regulations, the method described herein will be used by the staff in its evaluation of conformance with Commission regulations.

Implementation schedules for conformance to parts of the method discussed herein are contained in the referenced regulatory guides.

The provisions of this SRP section apply to reviews of construction permits (CP), operating licenses (OL), early site permits, and combined license (CP/OL) applications docketed pursuant to ~~the proposed~~ Section 100.23 ~~to~~ of 10 CFR Part 100.

## VI. REFERENCES


1. 10 CFR Part 50, Appendix A, General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena."

2. 10 CFR Part 100, ~~Proposed~~ Section 100.23, "Geologic and Seismic Siting Factors" ~~(59 FR 52255)~~.

3. US NRC, "Identification and Characterization of Seismic Sources and Determination of Safe Shutdown Earthquake Ground Motions," ~~Draft~~ Regulatory Guide ~~DG-1032~~1.165.

4. US NRC, Regulatory Guide 1.132, "Site Investigations for Foundations of Nuclear Power Plants."

5. US NRC, "General Site Suitability Criteria for Nuclear Power Stations," Regulatory Guide 4.7 ~~(Proposed Revision 2, DG-4004)~~.

6. US NRC, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants (LWR Edition)," Regulatory Guide 1.70.

7. US NRC, "Report of Siting Policy Task Force," NUREG-0625, August 1979.

8. 10 CFR Part 52, "Early Site Permits, Standard Design Certifications; and Combined Licenses for Nuclear Power Plants."

9. R.L. Bates and J. Jackson, editors, "Glossary of Geology," Second Edition, American Geological Institute, Falls Church, Virginia, 1980.

10. S.M. Colman, K. L. Pierce, and P. W. Birkeland, "Suggested Terminology for Quaternary Dating Methods," Quaternary Research, Volume 288, pp. 314-319, 1987.

11. J.B. Savy et al., "Eastern Seismic Hazard Characterization Update," Lawrence Livermore National Laboratory, UCRL-ID-115111, June 1993.

12. US NRC, "Revised Livermore Seismic Hazard Estimates for Sixty-Nine Nuclear Power Plant Sites East of the Rocky Mountains," NUREG-1488, April 1994.

13. Electric Power Research Institute, "Probabilistic Seismic Hazard Evaluation of Nuclear Power Plant Sites in the Central and Eastern United States," Volumes I through 10, NP-4726A, 1989.

14. Electric Power Research Institute, "Guidelines for Determining Design Basis Ground Motions," EPRI Report TR-102293, Vols. 1-4, May 1993.

15. A.L. Odom and R. D. Hatcher, Jr., "A Characterization of Faults in the Appalachian Foldbelt," U.S. Nuclear Regulatory Commission, NUREG/CR-1621, 1980.

16. G.V. Cohee (Chairman) et al., "Tectonic Map of the United States," U.S. Geological Survey and American Association of Petroleum Geologists 1962.

17. GeoRef Data Base, American Geological Institute, Falls Church, Virginia.

18. American Petroleum Institute data base, accessible through RECON system.

19. RECON/Energy Data base, Department of Energy.

# SRP SECTION 2.5.2, REVISION 3

# (VIBRATORY GROUND MOTION)

U.S. NUCLEAR REGULATORY COMMISSION
STANDARD REVIEW PLAN 2.5.2
VIBRATORY GROUND MOTION
~~SECOND PROPOSED~~ REVISION 3

~~February 1995~~
~~Contact: A.J. Murphy~~
~~(301)415-6010~~

REVIEW RESPONSIBILITIES

Primary - Civil Engineering and Geosciences Branch (ECGB)

Secondary - None

AREAS OF REVIEW

The Civil Engineering and Geosciences Branch review covers the seismological, ~~and~~ geological**, geophysical, and geotechnical** investigations carried out to ~~establish~~ **determine** ~~the acceleration for~~ the safe shutdown earthquake **ground motion** (SSE) ~~and the operating basis earthquake (OBE)~~ for the site. ~~The safe shutdown earthquake is that earthquake that is based upon an evaluation of the maximum earthquake potential considering the regional and local geology and seismology and specific characteristics of local subsurface material. It is that earthquake that produces the maximum vibratory ground motion for which safety-related structures, systems, and components are designed to remain functional. The operating basis earthquake is that earthquake that, considering the regional and local geology, seismology, and specific characteristics of local subsurface material, could reasonably be expected to affect the plant site during the operating life of the plant; it is that earthquake~~

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~~This standard review plan is being issued in draft form to involve the public in the early stages of its development. It has not received complete staff review and does not represent an official NRC staff position.~~

~~Public comments are being solicited on this draft standard review plan, which is part of a group of drafts of regulatory guides and standard review plan sections on meeting proposed amendments to the regulations on siting nuclear power plants (59 FR 52255). Comments should be accompanied by appropriate supporting data. Written comments may be submitted to the Rules Review and Directives Branch, DFIPS, Office of Administration, U.S. Nuclear Regulatory Commission, Washington, DC 20555. Copies of comments received may be examined at the NRC Public Document Room, 2120 L Street NW., Washington, DC. Comments will be most helpful if received by May 12, 1995.~~

~~Requests for single copies of this standard review plan (which may be reproduced) will be filled while supplies last. Requests should be in writing to the U.S. Nuclear Regulatory Commission, Washington, DC 20555, Attention: Office of Administration, Distribution and Mail Services Section.~~

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238

~~that produces the vibratory ground motion for which those features of the nuclear power plant necessary for continued operation without undue risk to the health and safety of the public are designed to remain function l.~~ The SSE represents the ~~potential for~~ design earthquake ground motion at the site and is the vibratory ground motion for which certain structures, systems, and components are designed to remain functional. The SSE is based upon a detailed evaluation of earthquake potential, taking into account regional and local geology, Quaternary tectonics, seismicity, and specific geotechnical characteristics of the site's subsurface material. The [illegible] is defined as the free-field horizontal and vertical ground response spectra at the plant site.

The principal regulation used by the staff in determining the scope and adequacy of the submitted seismologic and geologic information and attendant procedures and analyses is Section 100.23 of 10 CFR Part 100 (Ref. 1). Additional ~~guidance~~ information (regulations, regulatory guides, and reports) is provided to the staff through References 2 through ~~8~~ 9.

Guidance on seismological and geological investigations is ~~being developed~~ provided in ~~Draft~~ Regulatory Guide ~~DG-1032~~ 1.165, "Identification and Characterization of Seismic Sources and Determination of Safe Shutdown Earthquake Ground Motion." These investigations describe the seismicity of the site region and the correlation of earthquake activity with seismic sources. Seismic sources are identified and characterized, including the rates of occurrence of earthquakes associated with each seismic source. ~~All~~ Seismic sources that have any part within 320 km (200 miles) of the site must be identified. More distant sources that have a potential for earthquakes large enough to affect the site must also be identified. Seismic sources can be capable tectonic sources or seismogenic sources; a seismotectonic province is a type of seismogenic source.

Specific areas of review include seismicity (Subsection 2.5.2.1), geologic and tectonic characteristics of the site and region (Subsection 2.5.2.2), correlation of earthquake activity with ~~geologic structure or tectonic provinces~~ seismic sources (Subsection 2.5.2.3), ~~maximum earthquake potential~~ probabilistic seismic hazard analysis (PSHA) and controlling earthquakes (Subsection 2.5.2.4), seismic wave transmission characteristics of the site (Subsection 2.5.2.5), and safe shutdown earthquake ground motion (Subsection

2.5.2.6)~~, and operating basis earthquake (Subsection 2.5.2.7)~~.

The geotechnical engineering aspects of the site and the models and methods employed in the analysis of soil and foundation response to the ground motion environment are reviewed under SRP Section 2.5.4. The results of the geosciences review are used in SRP Sections 3.7.1 and 3.7.2.

II. <u>ACCEPTANCE CRITERIA</u>

The applicable regulations (Refs. 1, 2, and 3) and regulatory guides (Refs. 4, 5, 6, **and 9**) and basic acceptance criteria pertinent to the areas of this section of the Standard Review Plan are:

1. 10 CFR Part 100, "Reactor Site Criteria" (Ref. 3). This part describes **general** criteria that guide the evaluation of the suitability of proposed sites for nuclear power and testing reactors.

   **Proposed Section 100.23** ~~10 CFR Part 100~~, **"Geologic and Seismic Siting Factors,"** ~~Appendix A, "Seismic and Geologic Siting Criteria for Nuclear Power Plants." These criteria~~ describes the ~~kinds of~~ geologic and seismic information needed to determine site suitability and identify geologic and seismic factors required to be taken into account in the siting and design of nuclear power plants (Ref. 1).

2. 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants"; General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena" (Ref. 2). This criterion requires that safety-related portions of the structures, systems, and components important to safety ~~shall~~ be designed to withstand the effects of earthquakes, tsunamis, and seiches without loss of capability to perform their safety functions.

~~3. 10 CFR Part 100, "Reactor Site Criteria" (Ref. 3). This part describes criteria that guide the evaluation of the suitability of proposed sites for nuclear power and testing reactors.~~

~~4~~ 3. Regulatory Guide 1.132, "Site Investigations for Foundations of Nuclear Power Plants." This guide describes programs of site investigations related to geotechnical aspects that would normally meet the needs for evaluating the safety of the site from the standpoint of the performance of foundations under anticipated loading conditions, including earthquakes. It provides general guidance and recommendations for developing site-specific investigation programs as well as specific guidance for conducting subsurface investigations, including the spacing and depth of borings as well as sampling intervals (Ref. 4).

~~5~~ 4. Regulatory Guide 4.7 ~~(Proposed Revision 2, DG-4004)~~, "General Site Suitability Criteria for Nuclear Power Stations." This guide discusses the major site characteristics related to public health and safety ~~which~~ **that** the NRC staff considers in determining the suitability of sites for nuclear power stations (Ref. 5).

~~6~~ 5. Regulatory Guide 1.60, "Design Response Spectra for Seismic Design of Nuclear Power Plants." ~~This guide gives one method acceptable to the NRC staff for defining the response spectra corresponding to the expected maximum ground acceleration (Ref. 6). See also~~ **Smoothed response spectra are generally used for design purposes - for example, a standard spectral shape that has been used in the past is presented in Regulatory Guide 1.60 (Ref. 6). These smoothed spectra are still acceptable when the smoothed design spectra compare favorably with site-specific response spectra derived from the ground motion estimation procedures discussed** in Subsection 2.5.2.6.

6. **~~Draft~~ Regulatory Guide ~~DG-1032 (Ref. 9)~~1.165, "Identification and Characterization of Seismic Sources and Determination of Safe Shutdown Earthquake Ground Motion," describes acceptable methodologies for determining the controlling earthquakes and SSE ground motion for nuclear power plant sites. (Ref. 9)**

The **principal geologic and seismic consideration for site suitability and geologic and** ~~primary required investigations are described in 10 CFR Part 100, in Section IV(a) of Appendix A (Ref. 1) The acceptable procedures for determining the~~ seismic design bases are given ~~in Sections V(a) and Section~~

~~VI(a) of the appendix.~~ In ~~the proposed~~ Section 100.23 of 10 CFR Part 100. ~~Draft~~ Regulatory Guide ~~DG-1032~~ 1.165 (Ref. 9) ~~is being developed to~~ provides more detailed guidance on investigations. The seismic design bases are predicated on a reasonable, conservative determination of the SSE ~~and the OBE~~. ~~As defined in Section III of Appendix A (Ref. 1) to 10 CFR Part 100, the~~ The SSE ~~and OBE are~~ is 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 that 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.

2.5.2.1 Seismicity. ~~In~~ To meet~~ing~~ the requirements ~~of~~ ~~proposed~~ in Reference 1, this subsection 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 having Modified Mercalli Intensity (MMI) greater than or equal to IV or magnitude greater than or equal to 3.0 that have been reported ~~in all tectonic provinces~~ ~~for all seismic sources~~, any parts of which are within 320 km (200 miles) of the site. **Other large earthquakes outside of this area, but which would impact the SSE, should be reported.** A regional-scale map should be presented showing all listed earthquake epicenters and should be supplemented by a larger-scale map showing earthquake epicenters of ~~all known~~ events within 80 km (50 miles) of the site. The following information concerning each earthquake is required whenever it is available: epicenter coordinates, depth of focus, date, origin time, highest intensity, magnitude, seismic moment, source mechanism, source dimensions, distance from the site, and any strong-motion recordings (sources from which the information was obtained should be identified). All magnitude designations such as $m_b$, $M_L$, $M_s$, $M_w$ should be identified. **In the Central and Eastern United States, relatively little information is available on magnitudes for the larger historic earthquakes; hence, it may be appropriate to rely on intensity observations (descriptions of earthquake effects) or the dimensions of the area in which the event was felt to estimate magnitudes of historic events (e.g., Refs. ~~34 and 35~~ 10 and 11).** In addition, any reported earthquake-induced geologic failure, such as liquefaction **(including paleoseismic evidence of large prehistoric earthquakes)**, landsliding, landspreading, and lurching should be described completely, including the estimated level of strong motion that induced failure and the physical properties of the materials. The completeness of the earthquake history of the region is determined by comparison to published sources of information ~~(e.g., Refs. 9 through 13)~~. When conflicting descriptions of individual earthquakes are found in the published references, the staff should determine which is appropriate for licensing decisions.

<u>2.5.2.2 Geologic and Tectonic Characteristics of Site and Region</u>. ~~In meeting the requirements of References 1, 2, and 3, this subsection is accepted when all geologic structures within the region and tectonic activity seismic sources that are significant in determining the earthquake potential of the region are identified, or when an adequate investigation has been carried out to provide reasonable assurance that all significant tectonic structures seismic sources have been identified.~~ **For the CEUS sites, when the SSE is determined using the results of the LLNL or EPRI PSHA methodology and Regulatory Guide 1.165 (Ref.9), in meeting the requirements of References 1, 2, and 3, this subsection is acceptable when adequate information is provided to demonstrate: (1) that a thorough investigation has been conducted to identify seismic sources that could be significant in estimating the seismic hazard of the region if they exist; and (2) that existing sources (in the PSHA) are consistent with the results of site and regional investigations, or the sources have been updated in accordance with Appendix E of Regulatory Guide 1.165.**

**For sites where LLNL or EPRI methods and database have not been used, and it is necessary to identify and characterize seismic sources in meeting the requirements of References 1, 2,and 3, this subsection is acceptable when adequate information is provided to demonstrate that all seismic sources that are significant in determining the earthquake potential of the region are identified, or that an adequate investigation has been carried out to provide reasonable assurance that there are no unidentified significant seismic sources.**

Information presented in Section 2.5.1 of the applicant's safety analysis report (SAR) and information from other sources ~~(e.g., Refs. 9 and 14 through 18)~~ dealing with the current tectonic regime should be developed into a coherent, **well-documented** discussion to be used as the basis **for** characterizing the earthquake-generating potential of **seismic sources**. ~~the identified geologic structures~~ Specifically, each ~~tectonic province~~ **seismic source**, any part of which is within **320 km** (200 miles) of the site, must be identified. **In the CEUS the seismic sources will most likely be seismotectonic provinces.** The staff interprets **seismo**tectonic provinces to be regions of **assumed** uniform ~~earthquake potential (seismotectonic provinces)~~ **seismicity (same frequency of occurrence) distinct from the seismicity** of the **surrounding area**. The proposed **seismo**tectonic provinces may be based on seismicity studies, differences in geologic history, differences in the current tectonic regime, **or other tectonic considerations** ~~etc~~.

The staff considers that the most important factors for the determination of **seismic sources** ~~tectonic provinces~~ include both (1) development and characteristics of the current tectonic regime of the region that is most likely reflected ~~in the neotectonics (Post-Miocene or about 5~~ **in the Quaternary period (approximately the last 2** million years and younger geologic history) and (2) the pattern and level of historical seismicity. Those characteristics of geologic structure, tectonic history, present and past stress regimes, and seismicity that distinguish the various **seismic sources** ~~tectonic provinces~~ and the particular areas within those **sources** ~~provinces~~ where historical earthquakes have occurred should be described. Alternative regional tectonic models derived from available literature ~~sources, including previous SARs and NRC staff Safety Evaluation Reports (SERs),~~ should be discussed. The model that best conforms to the observed data is accepted. In addition, in those areas where there are capable ~~faults~~ **tectonic sources**, the results of **the additional** investigative requirements described in ~~10 CFR Part 100, Appendix A, Section IV(a)(8) (Ref. 1),~~ **SRP Section 2.5.1** must be presented. **The** discussion should be augmented by a regional-scale map showing the ~~tectonic provinces~~ **seismic sources**, earthquake epicenters, locations of geologic structures and other features that characterize the **seismic** sources. ~~, and the locations of any capable faults.~~

2.5.2.3 Correlation of Earthquake Activity with **Seismic Sources** ~~Geologic Structure or Tectonic Provinces~~. ~~In meeting~~ **To meet** the requirements **proposed in** ~~of~~ Reference 1, acceptance of this subsection is based on the development of the relationship between the history of earthquake activity and

~~the geologic structures or tectonic provinces~~ seismic sources of a region. For the CEUS sites, when the SSE is determined using LLNL or EPRI PSHA methodology and Regulatory Guide 1.165, in meeting the requirements of Reference 1, this subsection is acceptable when adequate information is provided to demonstrate: (1) that a thorough investigation has been conducted to assess the seismicity and identify seismic sources that could be significant in estimating the seismic hazard of the region if they exist; and (2) that existing sources (in the PSHA) are consistent with the results of site and regional investigations, or the sources have been updated in accordance with the Appendix E of Regulatory Guide 1.165.

For sites where LLNL or EPRI methods are not used, and it is necessary to identify and characterize seismic sources in meeting the requirements of Reference 1, this subsection is acceptable when adequate information is provided to demonstrate that all seismic sources that are significant in determining the earthquake potential of the region are identified, or that an adequate investigation has been carried out to provide reasonable assurance that there are no unidentified significant seismic sources.

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 tectonic province~~ seismic sources. Whenever an earthquake hypocenter or concentration of earthquake hypocenters can be reasonably correlated with geologic structures, the rationale for the association should be developed considering the characteristics of the geologic structure (including geologic and geophysical data, seismicity, and the tectonic history), and the regional tectonic model. The discussion should include identification of the methods used to locate the earthquake hypocenters, an estimation of their accuracy, and a detailed account that compares and contrasts the geologic structure involved in the earthquake activity with other areas within the ~~tectonic province~~ seismotectonic province. Particular attention should be given to determining the ~~capability~~ recency and level of activity of faults with which instrumentally located earthquake hypocenters ~~are~~ may be associated. ~~The presentation should be augmented by regional maps, all of the same scale, 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~~ seismic sources (those identified by the investigations) is based on the staff's independent review of the geologic and seismic information **presented by the applicant ~~and~~ available in the scientific literature.**

2.5.2.4 ~~Maximum Earthquake Potential~~ **Probabilistic Seismic Hazard Analysis (PSHA) and Controlling Earthquakes (CE)**. ~~In meeting the requirements of Reference 1, this subsection 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 that would produce the maximum vibratory ground motion at the site has been determined. The maximum credible earthquake is the largest earthquake that can reasonably be expected to occur on a geologic structure in the current tectonic regime. Geologic or seismological evidence may warrant a maximum earthquake larger than the maximum historic earthquake. Earthquakes associated with each geologic structure or tectonic province must be identified. Where an earthquake is associated with a geologic structure, the maximum credible earthquake that could occur on that structure should be evaluated, taking into account significant factors, for example, the type of the faulting, fault length, fault slip rate, rupture length, rupture area, moment, and earthquake history (e.g., Refs. 19 through 22).~~

~~In order to determine the maximum credible earthquake that could occur on those faults that 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 and geologic history of movement on the faults. Where the earthquakes are associated with a tectonic province, the largest historic earthquake within the province should be identified. Isoseismal maps should also be presented for the most significant earthquakes. The ground motion at the site should be evaluated assuming appropriate seismic energy transmission effects and assuming that the maximum earthquake associated with each geologic structure or with each tectonic province occurs at the point of closest approach of the structure or province to the site. (Further description is provided in Subsection 2.5.2.6.)~~

~~The earthquake(s) that would produce the most severe vibratory ground motion at the site should be defined. If different potential earthquakes would~~

~~produce the most severe ground motion in different frequency bands, these earthquakes should be specified. The description of the potential earthquake(s) is to include the maximum intensity or magnitude and the distance from the assumed location of the potential earthquake(s) to the site. The staff independently evaluates the site ground motion produced by the largest earthquake associated with each geologic structure or tectonic province.~~

~~Acceptance of the description of the potential that would produce the largest ground motion at the site is based on the staff's independent analysis.~~

For the CEUS sites relying on LLNL or EPRI methods and databases, the staff will review the applicant's probabilistic seismic hazard analysis, including the underlying assumptions and how the results of the site investigations ~~and findings of Sections 2.5.2.2 and 2.5.2.3~~ are used to update the existing sources in the probabilistic seismic hazard analysis, how they are used to develop additional sources, or how they are used to develop a new data base.

The staff will review the controlling earthquakes and associated ground motions at the site derived from the applicant's probabilistic hazard analysis to be sure that they are either consistent with the controlling earthquakes/ground motions used in licensing of (a) other licensed facilities at the site, (b) nearby plants, or (c) plants licensed in similar seismogenic regions, or the reasons they are not consistent are understood. For the CEUS, a comparison of the PSHA results can be made with the information included as Table 1, which is a very general representation based on technical information developed over the past two decades of licensing nuclear power plants.

The applicant's probabilistic analysis, including the derivation of controlling earthquakes, is considered acceptable if it follows the procedures ~~proposed~~ in ~~DG-1032~~ Regulatory Guide 1.165 and its Appendix C (Ref. 9). The incorporation of results of site investigations into the probabilistic analysis is considered acceptable if it follows the procedure outlined in Appendix E of ~~DG-1032~~ Regulatory Guide 1.165 and is consistent with the review findings of Sections 2.5.2.2 and 2.5.2.3.

For the sites not using LLNL or EPRI methods and databases, the staff will review the applicant's PSHA or other methods used to derive controlling earthquakes. The staff will particularly review the approaches used to address uncertainties. The staff will perform an independent evaluation of the earthquake potential associated with each seismic source that could affect the site. The staff will evaluate the applicant's controlling earthquakes based on historical and paleo-seismicity. In this evaluation, the controlling earthquakes for each source are at least as large as the maximum historic earthquake associated with the source.

TABLE 1

Controlling Earthquakes

| SEISMIC SOURCE | LLNL Magnitude | LLNL Distance (KM) | EPRI Magnitude | EPRI Distance (KM) |
|---|---|---|---|---|
| Northern New England | 5.6 - 5.7 | 15 | 5.7 - 5.8 | 18 |
| Piedmont - New England | 5.5 - 5.7 | 14 | 5.7 | 19 |
| Southern Valley and Ridge | 5.6 - 5.7 | 14 | 5.4 - 5.7 | 18, 19 |
| Atlantic Coastal Plain | 5.5 - 5.6 | 15-16 | 5.4 - 5.5 | 19, 21 |
| Gulf Coast | 5.3 | 16-18 | 5.3 | 23, 39 |
| Central Stable Region | 5.4 - 5.5 | 15-20 | 5.3 - 5.5 | 19, 20 21, 30 |
| Charleston | 7.5 Ms | Site-Specific | | |
| New Madr[redacted] | 8.5 Ms | Site-Specific | | |

2.5.2.5 Seismic Wave Transmission Characteristics of the Site. In the PSHA procedure described in ~~DG-1032~~ Regulatory Guide 1.165 (Ref. 9), the controlling earthquakes are determined for actual or hypothetical rock conditions. The site amplification studies are performed in a distinct separate step as a part of the determination of the SSE. In this section the applicant's site amplification studies are reviewed in conjunction with the

**geotechnical and structural** engineering reviews.

~~In meeting the requirements of Reference 1, this subsection is accepted when~~ **To be acceptable,** 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 **(Ref.12)**. The following material properties should be determined for each stratum under the site: **thickness,** seismic compressional and shear wave velocities, bulk densities, soil index properties and classification, shear modulus and damping variations with strain level, and water table elevation and its variation **(Ref. 13)**. In each case, methods used to determine the properties should be described in Subsection 2.5.4 of the SAR and cross-referenced in this subsection. ~~For the maximum earthquake determined in Subsection 2.5.2.4, the free-field ground motion (including significant frequencies) must be determined, and an analysis should be performed to determine the site effects on different seismic wave types in the significant frequency bands. If appropriate, the analysis should consider the effects of site conditions and material property variations upon wave propagation and frequency content.~~

~~The free-field ground motion (also referred to as control motion) should be defined to be on a ground surface and should be based on data obtained in the free field. Two cases are identified, depending on the soil characteristics at the site and subject to availability of appropriate recorded ground-motion data. When data are available, for example, for relatively uniform sites of soil or rock with smooth variation of properties with depth, the control point (location at which the control motion is applied) should be specified on the soil surface at the top of the finished grade. The free-field ground motion or control motion should be consistent with the properties of the soil profile. For sites composed of one or more thin soil layers overlying a competent material, or in case of insufficient recorded ground-motion data, the control point is specified on an outcrop or a hypothetical outcrop at a location on the top of the competent material. The control motion specified should be consistent with the properties of the competent material.~~

Where vertically propagating shear waves may produce the maximum ground motion, a one-dimensional equivalent-linear analysis (e.g., Ref. ~~23 or 24~~ **14 or 15**) or nonlinear analysis (e.g., Refs. ~~25, 26, and 27~~ **16, 17, or 18**) may be appropriate and is reviewed in conjunction with geotechnical and structural engineering. Where horizontally propagating shear waves, compressional waves, or surface waves may produce the maximum ground motion, other methods of analysis (e.g., Refs. ~~28 and 29~~ **19 and 20**) may be more appropriate. However, since some of the variables are not well defined and the techniques are still in the developmental stage, no generally agreed-upon procedures can be promulgated at this time. Hence, the staff must use discretion in reviewing any method of analysis. To ensure appropriateness, site response characteristics determined from analytical procedures should be compared with historical and instrumental earthquake data, when available.

2.5.2.6 Safe Shutdown Earthquake **Ground Motion**. ~~In meeting the requirements of Reference 1, this subsection is accepted when the vibratory ground motion specified for the SSE is described in terms of the free-field response spectrum and is at least as conservative as that which would result at the site from the maximum earthquake determined in Subsection 2.5.2.4, considering the 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.~~

**In this subsection, the staff reviews the applicant's procedure to determine the SSE, including the procedure used to derive spectral shape from the controlling earthquakes as described in Reference 9.**

**As a part of the review to judge the adequacy of the SSE proposed by the applicant, the staff performs an independent evaluation of ground motion estimates, as required.** ~~In these independent estimates, the staff may consider effects on ground motion from the controlling earthquakes discussed in Subsection 2.5.2.4 by assuming the controlling earthquake for each seismic source (geological structures or seismotectonic provinces) to be at its closest approach to the site.~~

~~The staff reviews the free-field response spectra of engineering significance (at appropriate damping values). Ground motion may vary for different foundation conditions at the site. When the site effects are significant, this~~

~~review is made in conjunction with the review of the design response spectra in Section 3.7.1 to ensure consistency with the free-field motion. The staff normally evaluates response spectra on a case-by-case basis. The staff considers compliance with the following conditions acceptable in the evaluation of the SSE. In all these procedures, the proposed free-field response spectra shall be considered acceptable if they equal or exceed the estimated 84th percentile ground-motion spectra from the maximum or controlling earthquake described in Subsection 2.5.2.4.~~

**The following procedures (in descending order of preference) should be used to develop the site-specific spectral shapes for controlling earthquakes. ~~The staff will also use t~~These procedures are also used to make ~~its independent~~ ground motion estimates when the probabilistic methods are not used. In the following procedures, 84th percentile response spectra are used for both spectral shape as well as ground motion estimates.**

~~The following steps summarize the staff review of the SSE.~~

1. Both horizontal and vertical component site-specific response spectra should be developed statistically from response spectra of recorded strong motion records that are selected to have similar source, propagation path, and recording site properties as the controlling earthquakes. It must be ensured that the recorded motions represent free-field conditions and are free of or corrected for any soil-structure interaction effects that may be present because of locations ~~and/~~or housing of recording instruments. Important source properties inc**lude** magnitude and, if possible, fault type~~,~~ and tectonic enviro**nment**. Propagation path properties include distance, depth, and attenuation. Relevant site properties include shear velocity profile and other factors that affect the amplitude of waves at different frequencies. A sufficiently large number of site-specific time-histories or response spectra or both should be used to obtain an adequate~~ly~~ broadband spectrum to encompass the uncertainties in these parameters. An 84th percentile response spectrum for the records should be presented for each damping value of interest. ~~and compared to the SSE free-field and design response spectrum~~ (e.g., Refs. ~~30, 31, 32, and 33~~ **21, 22, 23, and 24**). The staff considers direct estimates of spectral

ordinates preferable to scaling of spectra to peak accelerations. ~~In the Eastern United States, relatively little information is available on magnitudes for the larger historic earthquakes; hence, it may be appropriate to rely on intensity observations (descriptions of earthquake effects) to estimate magnitudes of historic events (e.g., Refs. 34 and 35).~~ If the data for site-specific response spectra were not obtained under geologic conditions similar to those at the site, corrections for site effects should be included in the development of the site-specific spectra.

2. Where a large enough ensemble of strong-motion records is not available, response spectra may be approximated by scaling that ensemble of strong-motion data that represent the best estimate of source, propagation path, and site properties (e.g., Ref. ~~36~~ 25). Sensitivity studies should show the effects of scaling.

3. If strong-motion records are not available, site-specific peak ground acceleration, velocity, and displacement (if necessary) should be determined for appropriate magnitude, distance, and foundation conditions. Then response spectra may be determined by scaling the acceleration, velocity, and displacement values by appropriate amplification factors (e.g., Ref. ~~37~~ 26). ~~Where only estimates of peak ground acceleration are available, it is acceptable to select a peak acceleration and use this peak acceleration as the high frequency asymptote to standardized response spectra such as described in Regulatory Guide 1.60 (Ref. 6) for both the horizontal and vertical components of motion with the appropriate amplification factors.~~ For each controlling earthquake, the peak ground motions should be determined using current relations between acceleration, velocity, and, if necessary, displacement, earthquake size (magnitude or intensity), and source distance. Peak ground motion should be determined from state-of-the-art relationships. Relationships between magnitude and ground motion are found, for example, in References 12 and 27. ~~Due to~~ Because of the limited data for high intensities greater than Modified Mercalli Intensity (MMI) VIII, the available empirical relationships between intensity and peak ground motion may not be suitable for determining the appropriate reference acceleration for seismic design.

4. ~~**Response**~~ Spectra developed by theoretical-empirical modeling of ground motion **may** be used to supplement site-specific spectra if the input parameters and the appropriateness of the model are thoroughly documented (e.g., Refs. ~~19, 44, 45, and 46~~ **12, 27, and 28**). Modeling is particularly useful for sites near ~~capable faults~~ ~~**tectonic**~~ seismic sources ~~or for deeper structures~~ that may experience ground motion that is different in terms of frequency content and wave type from ground motion caused by more distant earthquakes.

~~5. Probabilistic estimates of seismic hazard should be calculated (e.g., Refs. 41 and 47) and the underlying assumptions and associated uncertainties should be documented to assist in the staff's overall deterministic approach. The probabilistic studies should highlight which seismic sources are significant to the site. Uniform hazard spectra (spectra that have a uniform probability of exceedance over the frequency range of interest) showing uncertainty should be calculated for 0.01, 0.001, and 0.0001 annual probabilities of exceedance at the site. The probability of exceeding the SSE response spectra should also be estimated and comparison of results made with other probabilistic studies.~~

**The SSE ground motion response spectra proposed by the applicant are considered acceptable if they meet Regulatory Position 4 and Appendix F of Reference 9. If the independent staff estimates of ground motion are significantly different than those proposed by the applicant, the staff will review the reasons for differences and resolve them as appropriate.**

The time duration and number of cycles of strong ground motion are 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 Section 3.7.1. The time history is reviewed in this SRP section to confirm that it is compatible with the seismological and geological conditions in the site vicinity and with the accepted SSE model. At present, models for ~~deterministically~~ computing the time history of strong ground motion from a given source-site configuration ~~may be~~ **are** limited. ~~It is therefore acceptable to use an ensemble of ground-motion time histories from earthquakes with similar size, site-source characteristics, and spectral~~

~~characteristics or results of a statistical analysis of such an ensemble.~~ Total duration of the motion is acceptable when it is as conservative as values determined using current studies such as References ~~48, 49, 50, and 51~~ **29, 30, 31, and 32**.

**For evaluation of the liquefaction potential at the site, the time duration and number of cycles of strong ground motion are ~~more~~ critical parameters and require additional consideration. If the controlling earthquakes for the site have magnitudes of less than 6, the time history selected for the evaluation of liquefaction potential must have duration and number of strong motion cycles corresponding to at least an event of magnitude 6.**

~~2.5.2.7 Operating Basis Earthquake. In meeting the requirements of Reference 1, this subsection is acceptable when the vibratory ground motion for the OBE is described and the response spectrum (at appropriate damping values) at the site specified. Probability calculations (e.g., Refs. 41, 47, and 52) should be used to estimate the probability of exceeding the OBE during the operating life of the plant. The maximum vibratory ground motion of the OBE should be at least one-half the maximum vibratory ground motion of the SSE unless a lower OBE can be justified on the basis of probability calculations. It has been staff practice to accept the OBE if the return period is on the order of hundreds of years (e.g., Ref. 31).~~

## III. REVIEW PROCEDURES

Upon receiving the applicant's SAR, an acceptance review is conducted to determine compliance with the ~~**proposed**~~ investigative requirements of 10 CFR Part 100, **Section 100.23** ~~Appendix A~~ (Ref. 1). The reviewer also identifies any site-specific problems, the resolution of which could result in extended delays in completing the review.

After SAR acceptance and docketing, ~~those areas are identified where~~ **the reviewer identifies areas that need** additional information ~~is required~~ to **support the review of the applicant's seismic design** ~~determine the earthquake hazard~~. These are transmitted to the applicant as ~~draft~~ requests for additional information.

A site visit **may** be conducted, during which the reviewer inspects the geologic conditions at the site and the region around the site as shown in outcrops, borings, geophysical data, trenches, and those geologic conditions exposed during construction ~~if the review is for an operating license~~. The reviewer also discusses the 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. ~~Following the site visit, a revised set of requests for additional information, including any additional questions that may have been developed during the site visit, is formally transmitted to the applicant.~~

The reviewer evaluates the applicant's response to the questions, prepares requests for **any** additional ~~clarifying~~ information. and formulates positions that may agree or disagree with those of the applicant. These are formally transmitted to the applicant.

The Safety Analysis Report and amendments responding to the requests for additional information are reviewed to determine that the information presented by the applicant is acceptable according to the criteria described in Section II (Acceptance Criteria) above. Based on information supplied by the applicant **and information** obtained from site visits, ~~or from~~ staff consultants, or literature sources, the reviewer independently identifies **and evaluates** the relevant ~~seismotectonic provinces~~ **seismic sources**, **including their** ~~evaluates the~~ capability ~~of faults in the region~~, and determines the earthquake potential for each ~~province and each capable fault or tectonic structure~~ **using** procedures noted in Section II (Acceptance Criteria) above. The reviewer evaluates the vibratory ground motion that the ~~potential earthquakes~~ **controlling earthquakes** could produce at the site and ~~defines~~ **compares that ground motion to** the SSE **used for design.** ~~safe shutdown earthquake and operating basis earthquake~~.

## IV. EVALUATION FINDINGS

~~If the evaluation by the staff,~~ On completion of the review of the geologic and seismologic aspects of the plant site, **if the evaluation by the staff confirms that** the applicant has met the requirements or guidance of applicable portions of References 1 through 6 **and 9**, the conclusion in the SER states that the information provided and investigations performed support the applicant's conclusions regarding the seismic ~~integrity~~ **characterization** of the subject nuclear power plant site. In addition to the conclusion, this section of the SER includes **an evaluation of** (1) ~~definitions of tectonic provinces~~ **seismic sources**, (2) ~~evaluations of~~ the capability of geologic structures in the region, (3) ~~determinations of the SSE earthquake(s) and~~ **controlling earthquakes and associated** free-field response spectra ~~based on evaluation of the potential~~ ~~earthquakes~~, **(4) the SSE, and** (~~5~~ 4) **the** time history of strong ground motion~~, and (5) determinations of the OBE free-field response spectra~~. Staff reservations about any significant deficiency presented in the applicant's SAR are stated in sufficient detail to make clear the precise nature of the concern. **In addition, the staff will also note the results of its independent analyses, if performed, and discuss how these results were used in the safety evaluation.** The above evaluations ~~determinations or redeterminations~~ are made by the staff during ~~both~~ the construction permit (CP), ~~and~~ operating license (OL)**, combined license (COL), or early site permit phases of review as appropriate**.

OL **and cumbined license** applications are reviewed for any new information developed subsequent to the CP ~~safety evaluation report~~ SER **or the early site evaluation**. The review will also determine whether the ~~CP~~ recommendations **made following the CP or early site review** have been implemented.

A typical **combined license or** OL-stage summary finding for this section of the SER follows:

> In our review of the seismologic aspects of the plant site, we have considered pertinent information gathered since our initial seismologic review ~~which~~ **that** was made in conjunction with **an early site review** or the issuance of the Construction Permit. This new information includes data gained from both site and near-site investigations as well as from a review of recently published literature.
>
> As a result of our recent review of the seismologic information, we have

**determined** that our earlier conclusion regarding the safety of the plant from a seismological standpoint remains valid. These conclusions can be summarized as follows:

1. Seismologic information provided by the applicant and required by ~~Appendix A~~ **Section 100.23** ~~to~~ **of** 10 CFR Part 100 provides an adequate basis to establish that no ~~capable faults~~ **seismic sources** exist in the plant site area ~~which~~ **that** would cause earthquakes to be centered there.

2. The response spectrum proposed for the safe shutdown earthquake is the appropriate free-field response spectrum in conformance with ~~Appendix A~~ **Section 100.23 of** ~~to~~ 10 CFR Part 100.

The new information reviewed for the proposed nuclear power plant is discussed in Safety Evaluation Report Section 2.5.2.

The staff concludes that the site is acceptable from a seismological standpoint and meets the requirements of (1) 10 CFR Part 50, Appendix A (General Design Criterion 2), (2) 10 CFR Part 100, and (3) 10 CFR Part 100, ~~Appendix A~~ **Section 100.23**. This conclusion is based on the following:

1. The applicant has met the requirements of:

   a. 10 CFR Part 50, Appendix A, General Design Criterion 2 with respect to protection against natural phenomena such as faulting.

   b. 10 CFR Part 100, Reactor Site Criteria, with respect to the identification of geologic and seismic information used in determining the suitability of the site.

   c. 10 CFR Part 100, ~~Appendix A (Seismic and Geologic Siting Criteria for Nuclear Power Plants)~~ **Section 100.23** (Ref. 1) with respect to obtaining the geologic and seismic information necessary to determine (1) site suitability and

(2) the appropriate design of the plant. Guidance for complying with this regulation is contained in Regulatory Guide 1.132, "Site Investigations for Foundations of Nuclear Power Plants" (Ref. 4); **~~Draft~~ Regulatory Guide ~~DG-1032~~ 1.165, "Identification and Characterization of Seismic Sources and Safe Shutdown Earthquake Ground Motion" (Ref. 9); and** Regulatory Guide 4.7, "General Site Suitability Criteria for Nuclear Power Stations" ~~(Proposed Revision 2)~~ (Ref. 5)~~; and Regulatory Guide 1.60, "Design Response Spectra for Seismic Design of Nuclear Power Plants" (Ref. 6)~~.

V. IMPLEMENTATION

The following is intended to provide guidance to applicants and licensees regarding the NRC staff's plans for using this SRP section.

Except in those cases in which the applicant or licensee proposes an acceptable alternative method for complying with specific portions of the Commission's regulations, the methods described herein will be used by the staff in its evaluation of conformance with Commission regulations.

Implementation schedules for conformance to parts of the method discussed herein are contained in the referenced regulatory guides and NUREGs (Refs. 4 through ~~8~~ **9**).

The provi~~si~~**si**ons of this SRP section apply to reviews of construction permits (CP), operating licenses (OL), **early site permits,** ~~preliminary design approval (PDA), final design approval (FDA),~~ and combined license (CP/OL) applications docketed **pursuant to ~~the proposed~~ Section 100.23 to 10 CFR Part 100.** ~~after the date of issuance of this SRP section.~~

VI. REFERENCES


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2. 10 CFR Part 50, Appendix A, General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena."

3. 10 CFR Part 100, "Reactor Site Criteria."

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# SRP SECTION 2.5.3, REVISION 3

# (SURFACE FAULTING)

U.S.NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN 2.5.3 ~~February 1995~~

SURFACE FAULTING ~~Contact: A.J. Murphy~~

~~PROPOSED~~ REVISION 3 ~~(301)415-6010~~

## REVIEW RESPONSIBILITIES

Primary - Civil Engineering and Geosciences Branch (ECGB)

Secondary - None

## I. AREAS OF REVIEW

ECGB reviews information in the applicant's Safety Analysis Report (SAR) or Early Site Evaluation Report (ESR) that addresses the existence of a potential for surface deformation that could affect the site. The **technical** information presented in this section **of the SAR or ESR** results largely from detailed surface and subsurface geological, seismological, and geophysical investigations performed in the site subregion ~~(~~[40 km ~~or~~ (25 mi) from the site~~)~~], site vicinity ~~(~~[8 km ~~or~~ (5 mi) from the site~~)~~], and in the site area ~~(~~[within 1 km ~~or~~ (0.6 mi) of the site~~)~~]. The following specific subjects are addressed: the structural and stratigraphic conditions of the site subregion, site vicinity, and site area (subsection 2.5.3.1), any evidence of fault offset, including near-surface folding, uplift, or subsidence that reflects faulting at depth, or evidence demonstrating the absence of faulting within these areas (subsection 2.5.3.2), earthquakes associated with tectonic

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~~Public comments are being solicited on this draft standard review plan, which is part of a group of drafts of regulatory guides and standard review plan sections on meeting proposed amendments to the regulations on siting nuclear power plants (59 FR 52255). Comments should be accompanied by appropriate supporting data. Written comments may be submitted to the Rules Review and Directives Branch, DFIPS, Office of Administration, U.S. Nuclear Regulatory Commission, Washington, DC 20555. Copies of comments received may be examined at the NRC Public Document Room, 2120 L Street NW., Washington, DC. Comments will be most helpful if received by May 12, 1995.~~

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structures within these areas (subsection 2.5.3.3), determination of the age of most recent

movement on faults or other near-surface tectonic deformation (subsection 2.5.3.4), determination of structural relationships of site area faults to regional faults (subsection 2.5.3.5), identification and characterization of capable tectonic sources (subsection 2.5.3.6), zones of Quaternary deformation that require detailed fault investigations (subsection 2.5.3.7), and ~~results of studies in zones requiring~~ **the potential for surface tectonic deformation identified during the** detailed Quaternary faulting investigations (subsection 2.5.3.8).

References 1 through ~~8~~7 (regulations and regulatory guides) provide guidance to the ECGB reviewers in evaluating potential nuclear power plant sites. The principal regulation that will be used by ECGB ~~in the future~~ to determine the scope and adequacy of the submitted geological, seismological, and geophysical information is ~~Proposed~~ Section 100.23, "Geologic and Seismic Siting Factors," 10 CFR Part 100 (Ref. 2). Specific guidance for implementing this ~~proposed~~ regulation can be found in ~~Draft~~ Regulatory Guide ~~DG-1032~~ **1.165**, "Identification and Characterization of Seismic Sources and Determination of Safe Shutdown Earthquake Ground Motion" (Ref. 3). Guidance regarding the geotechnical engineering aspects is found in Regulatory Guide 1.132, "Site Investigations for Foundations of Nuclear Power Plants" (Ref. 4). Additional guidance is provided to the ECGB reviewers through information published in the scientific literature. As the state of the art regarding the geosciences is advancing rapidly, it is the responsibility of the reviewers to stay abreast of changes by reviewing the current scientific literature on a regular basis and attending professional meetings.

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## II. ACCEPTANCE CRITERIA

ECGB acceptance criteria are based on meeting the requirements of the following regulations:

1. Appendix A, "General Design Criteria for Nuclear Power Plants", General Design Criterion 2 - "Design Bases for Protection Against Natural Phenomena, 10 CFR Part 50." This criterion requires that safety-related portions of the structures, systems, and components important to safety be designed to withstand the effects of earthquakes, tsunami, and seiches without loss of capability to perform their safety functions (Ref. 1).

2. 10 CFR Part 100 ~~Proposed~~ Section 100.23, "Geologic and Seismic Siting Factors." These ~~proposed~~ requirements describe the general nature of the geological, seismological, and geophysical data necessary to determine the site suitability (Ref. 2).

   The following regulatory guides provide information, recommendations, and guidance and in general describe bases acceptable to the staff for implementing the requirements of General Design Criterion 2, Part 100, and ~~Proposed~~ Section 100.23 of Part 100.

   a. ~~Draft~~ Regulatory Guide ~~DG-1032~~1.165, "Identification and Characterization of Seismic Sources and Determination of Safe Shutdown Earthquake Ground Motion." This ~~draft~~ guide and its appendices ~~are being developed to~~ describe geological, seismological, and geophysical investigations to determine site suitability; methods to identify and characterize potential seismic sources; acceptable methods to conduct probability seismic hazard analyses; and methods to determine the Safe Shutdown Earthquake ground motion (SSE) (Ref. 3).

   b. Regulatory Guide 1.132, "Site Investigations for Foundations of Nuclear Power Plants." This guide describes programs of site investigations related to geotechnical aspects that would normally meet the needs for evaluating the safety of the site from the standpoint of the performance of foundations and earthworks under anticipated loading conditions, including earthquakes. It provides general guidance and recommendations for developing site-specific investigation programs as well as specific guidance for conducting subsurface investigations such as borings, sampling, and geophysical explorations (Ref. 4).

c. Regulatory Guide 4.7, "General Site Suitability Criteria for Nuclear Power Stations." This guide discusses the major site characteristics related to public health and safety that the NRC staff considers in determining the suitability of sites for nuclear power stations (Ref. 5, also see Ref. 6).

The data and analyses presented in the SAR or ESR are acceptable if, as a minimum, they describe and document the information proposed to be required by Reference 2, show that the methods described in Reference 3 or comparable methods were employed, and conform to the format suggested in Reference 7. References 8 and 9 have been used by the staff in past licensing activities as relevant guides to judge whether or not all of the current pertinent references have been consulted. References 10 through 17 are also used by the staff.

Specific criteria necessary to meet the relevant requirements of the Commission regulations identified above are described in the following paragraphs. If the information that satisfies these criteria is presented in other sections of Chapter 2.5, it may be cross-referenced and not repeated in this section.

Subsection 2.5.3.1 Geological, Seismological, and Geophysical Investigations. In meeting the requirements of References 1 and 2 and the positions of References 3 and 4, this subsection is considered acceptable if the discussions of the Quaternary tectonics, structural geology, stratigraphy, geochronological methods used, paleoseismology, and geological 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. For coastal and inland sites near large bodies of water, similar detailed investigations are to be conducted, and the information is to be provided in the SAR or ESR regarding offshore geology and seismology as well as onshore. In some instances it may be possible to identify an onshore projection of the offshore fault or fold of concern, or a tectonic structure that is analogous to it at an onshore location. It is acceptable to the staff, along with other investigations of the specific feature, to investigate the more remote, accessible exposure to learn the nature of the potentially hazardous offshore or buried fault and apply it to the local structure (Refs. 3 and 18). Site and regional maps (Ref. 3) 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 have been included in the SAR or ESR.

Subsection 2.5.3.2 **Geological Evidence, or Absence of Evidence for Surface Deformation**. In meeting the requirements of References 1, 2, and 3, this subsection is acceptable if sufficient surface and subsurface information is provided and supported by detailed investigations, either to confirm the absence of surface tectonic deformation (i.e., faulting) or, if present, to demonstrate the age of its most recent displacement and ages of previous displacements. If tectonic deformation is present in the site vicinity, it must be defined as to geometry, amount and sense of displacement, recurrence rate, and age of latest movement. In addition to geological evidence that may indicate faulting, linear features interpreted from topographic maps, low and high altitude aerial photographs, satellite imagery, and other imagery should be documented and investigated. In order to expedite the review process, an identification list, index, and duplicates of the remote sensing data used in the linear features study should be provided to and reviewed by the staff. Evidence for the absence of tectonic deformation is obtained by the applicant conducting site surface (geological reconnaissance and mapping, etc.) and subsurface investigations (geophysical, core borings, trenching and logging, etc.) in such detail and areal extent to ensure that undetected offsets or other deformations are not likely to exist.

In the Central and Eastern United States (CEUS), except for the New Madrid Seismic Zone, the Meers fault, ~~and possibly the Harlan County fault of Nebraska~~ and the Cheraw fault of the Colorado piedmont, earthquake generating faults either do not extend to ground surface or there is insufficient overlying soil or rock of known or of a sufficient ag. to date those that do.

In tectonically active regions such as the Western United States (WUS), many capable tectonic sources are exposed at ground surface and can be characterized as to their seismic potential. However, in these regions many other capable tectonic sources are buried (blind faults), and may be expressed at the surface or near surface by folding, uplift, or subsidence (including faults related to subduction zones). Investigations in these regions should take these phenomena into account. The nature of geological, seismological, and geophysical investigations will vary in detail and extent according to the geological complexity of the specific site.

Subsection 2.5.3.3 Correlation of Earthquakes with Capable Tectonic Sources. In meeting the requirements of References 1 and 2, this subsection is acceptable if all historically reported earthquakes within 40 km (25 mi) of the site are evaluated with respect to hypocenter accuracy and source origin, and if all capable tectonic sources that could, based on their orientations, extend to ~~that trend~~ within 8 km (5 mi) of the site are evaluated with respect to their potential for causi g surface deformation. In conjunction with these discussions, a plot of the earthquake epicenters superimposed on a map showing the local capable tectonic sources should have been ~~shown~~ provided.

Subsection 2.5.3.4 Ages of Most Recent Deformations. In meeting the requirements of References 1 and 2, this subsection is acceptable when every fault, or fold associated with a blind fault, any part of which is within 8 km (5 mi) of the site, is investigated in sufficient detail using geological and geophysical techniques of sufficient sensitivity to demonstrate, or allow relatively accurate estimates of the age of most recent movement and identify geological evidence for previous displacements if it exists (Ref. 3). An evaluation of the sensitivity and resolution of the exploratory techniques used should be given.

Subsection 2.5.3.5 Relationship of Tectonic Structures in the Site Area to Regional Tectonic Structures. In meeting the requirements of References 1 and 2, this subsection is satisfied by a discussion of the structural and genetic relationship between site area faulting or other tectonic deformation and the regional tectonic framework. In regions of active tectonism it may be necessary to conduct detailed geological and geophysical investigations to assess possible structural relationships of site area faults to regional faults known to be seismically active.

Subsection 2.5.3.6 **Characterization of Capable Tectonic Sources.** In meeting the requirements of References 1 and 2, this subsection is acceptable when it has been demonstrated that the investigative techniques used have sufficient sensitivity to identify all potential capable tectonic sources such as faults, or folds associated with blind faults, within 8 km (5 mi) of the site and when the geometry, length, sense of movement, amount of total offset, amount of offset per event, age of latest and any previous displacements, and limits of the zone are given for each capable tectonic source. Investigations are to extend at least 8 km (5 mi) beyond all plant site~~s~~ **boundaries**, including those adjacent to large bodies of water such as oceans, rivers, and lakes.

Subsection 2.5.3.7 **Designation of Zones of Quaternary Deformation in the Site Region**. In meeting the requirements of Reference 2, this subsection is judged acceptable if the zone designated by the applicant as requiring detailed faulting investigation is of sufficient length and breadth to include all Quaternary deformation significant to the site (Ref. 3).

Subsection 2.5.3.8 **Potential for Surface Tectonic Deformation at the Site**. In meeting the requirements of References 1 and 2, this subsection must be presented by the applicant if the aforementioned investigations reveal that surface displacement must be taken into account. If there is a potential for tectonically induced surface displacement at the site, it would be prudent of the applicant to abandon the site. No commercial nuclear power plant has been constructed on a known capable fault (capable tectonic source) and it is an open question as to whether it is feasible to design for tectonic 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 (capable tectonic sources) as determined by the detailed faulting investigations. If in the future it becomes feasible 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 three-phase review procedure described in Section 2.5.1 should be applied to assessing the potential for surface faulting. The first phase consists of an acceptance review to determine the completeness of the ESR or SAR by comparing the contents with the Criteria described in Part II, Acceptance Criteria, of this section. The second phase consists of a detailed review of the applicant's data and other independently derived information, which may result in requests for additional information. The third phase is a final review to resolve open issues and prepare a Safety Evaluation Report (SER).

The staff review procedure involves an evaluation to determine that the applicant has performed adequate investigations to fulfill the general requirements of Reference 2. Acceptable methods are described in Reference 3. Consultants or advisors may be called on to assist the staff in reviewing this section of the ESR or SAR on a case-by-case basis. On request, the advisor or consultant provides expertise in numerous earth science disciplines and occasionally is able to provide first-hand knowledge of the site. A literature search is conducted independently by the staff concerning the regional and local geology and seismology. The staff also utilizes the expertise of the U.S. Geological Survey and other Federal agencies, State geological surveys, universities, and private industry to obtain additional, up-to-date geosciences information regarding Quaternary tectonics at the site.

~~The Proposed~~ Section 100.23 of 10 CFR Part 100 ~~would~~ requires that applicants investigate the potential for near-surface deformation, both tectonically induced and that induced by other phenomena (Ref. 2). The steps that applicants may follow in determining the presence and extent of deformation and whether near-surface deformation (if present) represents a hazard are in ~~Draft~~ Regulatory Guide ~~DG-1032~~ 1.165, Appendix D (Ref 3). The site vicinity ~~(~~[8 km ~~(~~5 mi~~)~~ from the site~~)~~] and site area ~~(~~[1 km ~~(~~0.6 mi~~)~~ from the site~~)~~] must be investigated by a combination of exploratory methods that should include borings, trenching, seismic profiling and other geophysical methods, geological mapping, and seismic instrumentation. The results of these explorations are cross-compared with other available data and evaluated by the staff. An important part of the staff's review effort is to compare the new information derived from these investigations or other sources with the specific data base used in the probabilistic seismic hazard analysis (PSHA) for the site (Ref. 3).

It has been the policy of the staff to encourage applicants to avoid areas that have a possibility for near-surface tectonic deformation. As the question of whether or not a surface tectonic deformation condition exists is ~~so~~ critical in determining site suitability, this consideration is usually addressed very early in the review. The exceptions are cases in which a previously unknown fault is revealed in excavations during construction or is discovered during the course of other investigations in the area. The staff should require early ~~on~~ in the review that it be notified by the applicant when the excavations for Seismic Category I structures are available for NRC inspection and when the detailed geological maps to be used by the staff while examining the excavations will be available. In addition, the staff should require that it be contacted immediately if a fault, not previously identified in the SAR or ESR, is found within 8 km (5 mi) of the plant.

10 CFR Part 52 describes an alternative licensing approach that may be used in lieu of ~~T~~the ~~previous~~ current two-step procedure of requiring applicants to obtain a Construction Permit, followed several years later after the plant design bases have been approved by the staff, by application for an Operating License.~~, has been provided with an alternative method, a combined licensing procedure, by 10 CFR Part 52.~~ This procedure, called combined licensing, could create a problem for the staff in that the Safety Evaluation Report (SER) will already have been written and the applicant ~~could~~ will ~~already~~ have a license before excavations are started.~~, and~~ Therefore, faults discovered for the first time in the excavations ~~that fall in the category described in the previous paragraph~~ will not have been evaluated by the staff~~ before time for the preparation of the Safety Evaluation Report (SER)~~
. ~~Therefore, It is imperative that~~ To alleviate this potential problem, ~~Section 2.5.3 of the SER~~ be there must be a commitment in the site specific portion of the SAR for a facility to: (1) notify the staff immediately if previously unknown geologic features that could represent a hazard to the plant are encountered in the excavation; (2) geologically map all excavations

for Category 1 structures, as a minimum; and (3) notify the staff when the excavations are open for examination and evaluation. ~~staff has carefully examined the walls and floors of the excavation r the plant and determined that there are no previously unidentified pote lly hazardous faults or other features beneath the proposed plant. When the staff is satisfied regarding this issue, the SER should be finalized as soon as possible.~~ ~~made conditional on the demonstrated absence of previously unknown potentially hazardous faults beneath the plant as determined by careful examination of the excavations by the staff as described in the previous paragraph.~~

When faults are identified in the site vicinity or site area, it must be demonstrated that the faults do not have the potential to ~~generate earthquakes at the site (seismogenic source) or~~ cause near-surface ground displacement (capable tectonic source) at the site. This is accomplished by determining the ages of the latest displacement on the faults, preferably by stratigraphic methods, that is, identifying strata or a stratum of datable soil or rock overlying the fault that is undeformed by the fault. Other methods include correlating the last faulting event with regional tectonic activity of known ancient age, geomorphic evidence of age, and determining the relationship between the time of the fault rupture event and the ages of marine or fluvial terraces. Geochronological methods are discussed in References 3 and 17. ~~Draft~~ Regulatory Guide ~~DG-1032~~ 1.165 (Ref. 3) provides brief descriptions and a list of references of state-of-the-art methods and their applications, which can be used to estimate the geochronological history of geological materials associated with faults or other features.

In cases such as are described in the ~~last~~ previous paragraph, the staff will carry out limited site observations and investigations of its own such as examinations of excavations. In some cases, the staff may select samples from shear zones or other materials for subsequent dating and analysis. In past investigations ~~activities~~ ~~Applicants~~ ~~usually~~ applicants have often excavated trenches in the areas where major facilities are to be located for in situ testing ~~and~~ to reduce the chance for surprises when the construction excavations are made.

<u>Subsection 2.5.3.1 Geological, Seismological, and Geophysical Investigations.</u> This subsection is evaluated by conducting an i pendent literature search

and cross-comparing the results with the information submitted in the SAR or ESR. 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 or consultants. If the applicant's conclusions and assumptions conflict with the literature, and the staff disagrees with the applicant's analysis and assumptions, additional investigative results to support those conclusions must be submitted to the staff for review.

Subsection 2.5.3.2 **Geological Evidence, or Absence of Evidence for Surface Deformation.** **This subsection** is evaluated by first determining through a literature search **and comparison with the applicant's data,** that all ~~known~~ evidences of tectonic deformation such as fault offset **identified in the literature** 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 possible displacements. If such evidence is found, additional investigations such as field mapping, geophysical investigations, borings, or trenching must be carried out to demonstrate that there is no offset or to define the characteristics of the fault if it does exist. It is important to distinguish between tectonically induced near-surface deformation and deformation caused by nontectonic phenomena such as growth faulting, collapse caused by the development of karst terrane, etc. (Ref. 3).

Subsection 2.5.3.3 **Correlation of Earthquakes with Capable Tectonic Sources** **This subsection** is reviewed in conjunction with the consideration of SRP Section 2.5.2. Historical earthquake data derived from the review of SRP 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 tectonic structures. This determination includes an evaluation of the hypocentral error estimates of the earthquakes. When available, the earthquake source mechanisms should be evaluated with respect to fault geometry. In addition, applicants and licensees are encouraged to evaluate the relationship of fault parameters to earthquake magnitude. These parameters may include, but are not limited to slip rate, recurrence intervals, length, rupture area, and fault type (Ref. 18).

Subsection 2.5.3.4 Ages of Most Recent Deformation **This subsection** is evaluated to determine whether the geochronological methodologies used by the applicant are based on accepted ~~geological~~ procedures. In some cases unusual or untested age-dating techniques may have been used. When such methods are employed, the staff will require documentation of the technique. The ~~resolution~~ **precision** of all age dating techniques **used in the applicant's analysis** should be carefully documented. The staff may require the services of **one or more** ~~a~~ consultants who haves expertise in the methods used.

Subsection 2.5.3.5 Relationship of Tectonic Structures in the Site Area to Regional Tectonic Structures **This Subsection** is evaluated by determining through a literature search that the applicant's evaluation of the regional tectonic framework is consistent with that of recognized experts whose reports appear in the **peer reviewed** published literature. The conclusions reached by the applicant should be based on sound geological principles and should explain the available geological and geophysical data. When special investigations are made to determine the structural relationship between faults that pass within 8 km (5 mi) of the site and regional faults, the resolution accuracy of the investigative techniques should be given.

Subsection 2.5.3.6 Characterization of Capable Tectonic Sources **This subsection** is evaluated to determine whether a sufficiently detailed investigation has been made by the applicant to define the specific characteristics of all potential capable tectonic sources **any part of which** is located within 8 km (5 mi) of the site. The ~~fault~~ **structural** ~~e's~~ characteristics that must be defined include length, orientation, geometry, and relationship of the fault or fold to regional structures; the nature, amount, and geological history of displacements along the fault; and the outer limits of the zone established by mapping the extent of Quaternary deformation in all directions. The staff must be satisfied that the investigations cover a large enough area and are in sufficient detail to demonstrate that there is little likelihood of near-surface deformation hazards associated with capable tectonic sources existing undetected near the site.

Subsection 2.5.3.7 Designation of Zones of Quaternary Deformation in the Site Region. The zone that ~~needs~~ **requires** detailed investigations is defined by the area characterized by Quaternary deformation in the site subregion (within a distance of 40 km or 25 miles of the site). The staff reviews the results of the applicant's investigation together with a review of 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 may be asked to review certain aspects of the investigative program. The results of the investigations are analyzed to determine whether the outer limits of the zone of Quaternary deformation investigation are appropriately conservative.

Subsection 2.5.3.8 **Potential for Surface Tectonic Deformation of the Site.** If the detailed ~~faulting~~ investigations for the proposed commercial nuclear power plant reveal that there is a potential for surface deformation at the site, the staff recommends that an alternative location for the proposed plant be considered. **It is not expected that nuclear power plants could be successfully designed for displacement in its foundation at the present time. However,** ~~I~~in the future, ~~when~~ **if** it ~~may~~ be**comes** feasible to design a commercial nuclear power plant ~~for~~ **to accommodate** displacements, substantial information would be required to support the design basis for surface ~~faulting~~ **deformation**.

While fulfilling the tasks of Subsections 2.5.3.1 through 2.5.3.8, it is important for the staff SAR or ESR reviewer to identify all significant new information, such as a seismic source or a new tectonic model that was not included in the site PSHA, and coordinate that information with the staff PSHA reviewer.

## IV. EVALUATION FINDINGS

If the evaluation by the staff, on completion of the review of the geological and seismological aspects of the plant site, confirms that the applicant has met the requirements of applicable portions of General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena," of Appendix A to 10 CFR Part 50; and ~~Proposed~~ **10 CFR Part 100,** Section 100.23, "Geologic and Seismic Siting Factors," the conclusion in the SER would state that the investigations performed, and the information and analyses provided, support the applicant's conclusions regarding the geologic and seismic suitability of the subject nuclear power plant site with respect to surface deformation potential. Staff reservations about any significant deficiency, either presented in the applicant's ESR or SAR, and identified by the staff, should be stated in sufficient detail to make clear the precise nature of the concern. The above determinations are made by the staff during the early site, construction permit, operating license, or combined license reviews.

The ESR or SAR is also reviewed for any significant new information derived by the site-specific geological, seismological, and geophysical investigations that had not been applied to the tectonic and ground motion models used in the PSHA. Appendix E of ~~Draft~~ Regulatory Guide ~~DG-1032~~ 1.165 (Ref. 3) discusses an acceptable method to address significant new information in the PSHA.

A typical finding for this section of the SER follows:

> In its review of the geological and seismological aspects of the plant site, the staff considered pertinent information gathered during the regional and site-specific geological, seismological, and geophysical investigations. The information includes data gathered from both site and near-site investigations and from an independent review of state-of-the-art, published literature and other sources by the staff.
>
> As a result of this review, the staff concludes that the geological, seismological, and geophysical investigations and information provided by the applicant in accordance with ~~the Proposed~~ Section 100.23 of 10 CFR Part 100 and ~~Draft~~ Regulatory Guide ~~DG-1032~~ 1.165 provide an adequate basis to establish that no capable tectonic sources exist in the plant site vicinity that would cause surface deformation or localize earthquakes there.

The information reviewed for the proposed nuclear power plant concerning the potential for near-surface tectonic deformation is summarized in Safety Evaluation Report Section 2.5.3.

The staff concludes that the site is suitable from the perspective of tectonic surface deformation and meets the requirements of: (1) 10 CFR Part 50, Appendix A (General Design Criterion 2), and (2) ~~the Proposed~~ Section 100.23 of 10 CFR Part 100. This conclusion is based on the following:

1. The applicant has met the requirements of:

a. 10 CFR Part 50, Appendix A (General Design Criterion 2) with respect to protection against natural phenomena such as faulting.

b. ~~The Proposed~~ Section 100.23 of 10 CFR Part 100 (Geologic and Seismic Siting Factors) with respect to obtaining the geological and seismological information necessary (1) to determine site suitability, (2) to determine the appropriate design of the plant, and (3) to ascertain that any new information derived from the site-specific investigations does not impact the SSE ground motions derived by a PSHA. In complying with this regulation, the applicant also meets the staff's guidance proposed in ~~Draft~~ Regulatory Guide ~~1032~~ **1.165**, ~~"Geologic and Seismic Siting Factors~~ **"Identification and Characterization of Seismic Sources and Determination of Safe Shutdown Earthquake Ground Motion**"; Regulatory Guide 1.132, "Site Investigations for Foundations of Nuclear Power Plants;" and Regulatory Guide 4.7, "General Site Suitability Criteria for Nuclear Power Plants."

## V. IMPLEMENTATION

The following is intended to provide guidance to applicants and licensees regarding the NRC staff's plans for using this SRP section.

Except in those cases in which the applicant/licensee proposes an acceptable alternative method for complying with specific portions of the Commission's regulations, the method described herein will be used by the staff in its evaluation of conformance with Commission regulations.

Implementation schedules for conformance to parts of the method discussed herein are contained in the referenced regulatory guides (Refs. 4, 5, 6, 7, and 8).

The provisions of this SRP section apply to reviews of construction permits (CP), operating licenses (OL), early site permits, and combined license

12. RECON/Energy data base, Department of Energy.

13. State geological maps and accompanying texts.

14. U.S. Geological Survey 7.5 and 15 minute topographic and geologic quadrangle maps.

15. Aerial photographs from Federal agencies such as the National Aeronautics and Space Administration, the U.S. Department of Agriculture, the U.S. Geological Survey, and the U.S. Forest Service.

16. Satellite imagery such as Landsat and Skylab.

17. P.J. Murphy, J. Briedis, and J. H. Pfeck, "Dating Techniques in Fault Investigations," pp. 153-168, in <u>Geology in the Siting of Nuclear Power Plants</u>, A.W. Hatheway and C.R. McClure, Jr., editors, "Reviews in Engineering Geology," Volume 4, Geological Society of America, 1979.

18. US NRC, "Safety Evaluation Report Related to the Operation of Diablo Canyon Nuclear Power Plant, Units 1 and 2," NUREG-0675, Supplement No. 34, June, 1991.

(CP/OL) applications docketed pursuant to the ~~proposed~~ Section 100.23 to 10 CFR Part 100.

## VI. REFERENCES


1. 10 CFR Part 50, Appendix A, General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena."

2. CFR Part 100, ~~Proposed~~ Section 100.23, "Geologic and Seismic Siting Factors," Federal Register, Volume 59, page 52255, October 17, 1994 (59 FR 52255).

3. US NRC, "Identification and Characterization of Seismic Sources and Determination of Safe Shutdown Earthquake Ground Motions," ~~Draft~~ Regulatory Guide ~~DG-1032~~1.165.

4. US NRC, "Site Investigations for Foundations of Nuclear Power Plants." Regulatory Guide 1.132.

5. US NRC, "General Site Suitability Criteria for Nuclear Power Stations." Regulatory Guide 4.7 ~~(Proposed Revision 2, DG-4004)~~.

6. US NRC, "Report of Siting Policy Task Force," NUREG-0625, August 1979.

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

8. American Petroleum Institute data base, accessible through RECON system.

9. GeoRef data base, American Geological Institute, Falls Church, Virginia.

10. R.L. Bates and J.A. Jacksons, editors, "Glossary of Geology," American Geological Institute, Falls Church, Virginia, 1980.

11. G.V. Cohee (Chairman) et al., "Tectonic Map of the United States," U.S. Geological Survey and American Association of Petroleum Geologists, 1962.

3

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
430TH MEETING
SEVERE ACCIDENT RESEARCH
APRIL 11, 1996
ROCKVILLE, MARYLAND

-TABLE OF CONTENTS-



Cognizant ACRS Member: M. Fontana

Cognizant ACRS Staff Engineer: N. Dudley

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
430TH MEETING
SEVERE ACCIDENT RESEARCH
APRIL 11, 1996
ROCKVILLE, MARYLAND

-AGENDA-

| | | Presentation Length | Time |
|---|---|---|---|
| I. | Introduction - Dr. Fontana | 5 min | 11:00-11:05 a.m. |
| II. | Staff Presentation -<br>Mr. Charles Ader, RES<br>Mr. Alen Rubin, RES<br>Mr. Charles Tinkler, RES | 80 min | 11:05-12:25 p.m. |
| III. | Closing Remarks - Dr. Fontana | 5 min | 12:25-12:30 p.m. |

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
430TH MEETING
SEVERE ACCIDENT RESEARCH
APRIL 11, 1996
ROCKVILLE, MARYLAND

-STATUS REPORT-

PURPOSE

The Committee will hear a presentation by the NRC staff regarding the Severe Accident Research Program.

BACKGROUND

On May 25, 1988, the NRC staff provided an integrated closure plan for severe accident issues in SECY-88-147. The staff has periodically updated the status of the closure plan in commission papers. The six main plan elements identified in SECY-88-147 and the disposition of the elements are provided below.

Individual Plant Examinations (IPE): The ACRS IPE Subcommittee is reviewing this element.

External Events: The ACRS IPE Subcommittee is reviewing this element.

Containment Performance Improvements: The staff considers this element closed based on the completion of containment performance improvement reports, which documented contractor analyses.

Improved Plant Operations: This element involves ongoing programs for which the Commission is provided updates through other mechanisms and is therefore not discussed in the severe accident status reports. Improved plant operations is an ongoing effort that is expected to continue after the severe accident issues are closed.

Severe Accident Research Program: It should be noted that "closure" does not imply that all research in the areas of severe accident phenomena will cease. The activities are designed to provide confirmation of previous judgements. Emergent issues will be considered on a case-by-case basis, and are not expected to bring into question the previous conclusions regarding closure.

Accident Management: The industry agreed to develop and implement severe accident procedures.

The ACRS last commented on the closure of severe accident issues in a report dated August 18, 1992 [**page 5**]. Some of the comments in the report on specific activities include the following:

- The experimental program on direct containment heating (DCH) is soundly based, and should resolve the issue.

- We do not believe that some important aspects of the hydrogen issue have received the attention they deserve.

- Debris coolability is still an open issue. It will probably not be resolved by existing or planned programs.

- The question of energy release associated with violent interaction of liquid corium and water is unresolved, and a resolution is not in sight. We recommend additional research in this area.

- Significant weaknesses have been identified by the peer reviewers of the MELCOR code. Decisions on the use of severe accident codes and on their required capability are needed before plans for further developments are made.

- We endorse, with the caveats noted, the core melt progression program.

The Severe Accidents Subcommittee heard briefing from the staff on the status of the severe accident research program on March 1, 1996. The minutes for the meeting begin on **page 14**. The staff issued NUREG-6109, "The Probability of Containment Failure by Direct Containment Heating in Surry," in May 1995. The NUREG resolved the DCH issue for Surry using a framework developed in evaluating the DCH issue for Zion. The staff plans to use the same framework to resolve the DCH issue for all large dry containments.

In the August 1992 ACRS report [**page 10**], the Committee commented on the severe accident codes. The comments included the following:

- the staff should decide how codes will be used in the regulatory process

- the staff should develop procedures to make it less likely that peer review of future codes will identify a significant number of problems

- the SCDAP/RELAP5 code could generate misinformation if severe accident models are based on bounding instead of best estimate assumptions

The Severe Accidents Subcommittee heard staff presentations on the NRC severe accident coded on April 8, 1996.

EXPECTED COMMITTEE ACTION

The Committee is expected to prepare a report regarding the severe accident research program.

August 18, 1992

The Honorable Ivan Selin
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Chairman Selin:

SUBJECT: SEVERE ACCIDENT RESEARCH PROGRAM PLAN

During the 387th and 388th meetings of the Advisory Committee on Reactor Safeguards, July 9-11 and August 6-8, 1992, we reviewed the Severe Accident Research Program (SARP) Plan that is being directed by the Office of Nuclear Regulatory Research (RES). This review followed meetings of our Severe Accidents Subcommittee on October 25 and 26, 1991, May 27, 1992, and June 25, 1992, at which this matter was discussed. We had the benefit of discussions with representatives of the NRC staff and of the documents referenced.

## GENERAL COMMENTS

First, we consider the updated SARP Plan, described in draft NUREG-1365, Revision 1, a noticeable improvement over what we have seen in the past. The document is well written. The goal of the overall program is said to be the reduction of the likelihood of early containment failure. Generally, the goals and objectives of individual projects are more clearly stated than we have seen in the past. Even so there are occasional ambiguities, and the organization needs improvement. For example, there is duplication as well as some inconsistency among the appendices and the main report. In addition, some project descriptions begin with statements that this is a very complex area, that large uncertainties exist in the understanding of severe accident phenomena, and that the proposed research will remove some of the uncertainty. There is no indication of how much uncertainty is likely to be removed by the proposed research, nor how much must be removed in order that the regulatory program proceed satisfactorily. The objectives of several projects are still described as an effort to "gain insights" without an indication of how much or what type of insight is required, or to achieve a "better understanding" of some phenomenon without an indication of where the existing understanding is deficient or of what will be contributed to the regulatory process by an increased understanding. We do observe that effort is now being made to identify the point at which the objectives of a project will have been achieved.

Second, we commend the staff for the extensive peer reviews that are now being required. The planning of research, the results of the research, and the conclusions drawn from the work are now being subjected to review. Our observations lead us to believe that as a result, the current research activities are making more efficient use of resources. Further review of the results and of their interpretation by those outside RES should produce conclusions that have greater general acceptance and are more broadly useful than has been the case in the past.

Third, we observe that those responsible for severe accident research labor under a significant handicap. As we have reported to you earlier, there has not yet been a decision as to how the severe accident issues are to be dealt with in the regulatory arena, either for evolutionary or advanced reactor designs. The Office of Research is thus in the position of a traveler with no road maps.

## COMMENTS ON SPECIFIC ACTIVITIES

### The Mark I Liner Failure Issue

RES reported to us that the Mark I liner issue is close to resolution based on the following developments:

- The report, NUREG/CR-5423, "The Probability of Liner Failure in a Mark I Containment," has been extensively reviewed and revised to take account of the reviewers' comments.

- The core-concrete interaction (CCI) issue has been resolved.

We agree that NUREG/CR-5423 provides a coherent treatment of early failure of the Mark I liner. We note that the effects of ex-vessel steam explosions, which might result if water is on the containment floor, were not treated. We also call attention to and agree with the observation of Dr. S. Hodge, Oak Ridge National Laboratory (ORNL), in his letter appended to the report, that the report concludes only that early failure is implausible. Later failure is not ruled out by the results of the report.

### Chemical Form of Iodine Released to Containment

We discussed work recently completed at ORNL (NUREG/CR-5752) on the chemical form of iodine expected to be released to containment. This work contributes to the formulation of the new source term, and should lead to a more reliable calculation of iodine released outside containment. It is not clear how these results will influence calculated risk of existing plants nor how the information will be used in the review of the individual plant examinations (IPEs) being performed. This should be investigated further.

## Direct Containment Heating

An experimental program expected to produce information that will provide a resolution of the direct containment heating (DCH) issue is now said to be on a solid technical base. A resolution is expected within about a year. The program was delayed because of questions about scaling. The recently issued severe accident scaling methodology (SASM) report, NUREG/CR-5809, provides the needed guidance. Experimental work at Sandia National Laboratories (SNL) has begun. Work at Argonne National Laboratory (ANL) is also under way. Early results indicate that a defensible case can be made for the loads on containment being well below the structural failure loads, at least for the large dry containments.

We note, however, that for many of the PWR PRAs, including two of those treated in NUREG-1150, containment bypass is the risk-dominant failure mode. Thus, it is expected that resolution of the DCH issue will not have a significant effect on the estimated risk or on the risk uncertainty for these plants. We are encouraged that useful guidance in this area has been provided by the severe accident scaling methodology.

## Hydrogen

We have some concerns about the conclusions concerning effects of hydrogen detonations on containments such as the steel shell proposed for the Westinghouse AP600. It appears that the NRC staff has not considered thin shell containments, nor have they gone beyond planar or spherical shocks. Some recent conversations that we have had with the members of the German RSK indicate that their investigations have convinced them that three dimensional calculations are required because of the shock interactions that will occur.

We are not satisfied that there has been adequate investigation of the following questions for containments generally:

- Where is the hydrogen in containment?
- How is appropriate igniter placement determined?
- How effective are igniters in removing hydrogen from mixtures of steam and other noncondensable gases?

- How effective are containment passive cooling systems as hydrogen concentrators?

- How likely is a detonation?

Core-Concrete Interaction

We agree with the report by Dr. D. Powers, SNL, that, in his view, the experimental work that has been completed is adequate for the validation of the models in the NRC severe accident codes that model core-concrete interaction. A major uncertainty in the results of calculations using current codes is the state of the molten material that exits the vessel. He considers the agreement between CORCON calculations and the German BETA Test to be very good.

Debris Coolability

This research is particularly important to an evaluation of the effects of molten corium on the containment loading for the new reactor designs currently being reviewed. A number of programs over the past several years, both in the U.S. and abroad, have investigated the cooling of molten corium on the containment floor covered by a layer of water. Data are sparse, and the issue of whether cooling will occur in actual containments under accident conditions is still open. How applicants will be required to demonstrate debris coolability in containments is also still not established. If it is to be done experimentally, additional research will be required. The small-scale Melt Attack and Debris Coolability Experiment (MACE) tests at ANL, scheduled for completion in FY 1993, are expected to provide additional information, but are unlikely to provide conclusive evidence of coolability of debris. Some additional experiments may be required after the results of the MACE tests are analyzed. The magnitude and scope of these should be determined by regulatory needs. Work on debris spreading, an important consideration in coolability, is planned for 1994.

Fuel-Coolant Interactions

The principal concern is whether explosive energy releases can occur when molten corium encounters coolant either in the vessel or after the corium has left the vessel. Despite a recognition of the problem almost two decades ago, no generally accepted method exists for calculating the conversion of thermal energy to mechanical energy in this situation. Currently there are several small programs in the U.S. being supported by the NRC, as well as a program in Europe in which the NRC is participating. It is questionable whether any of these will produce information that will resolve the issue. We recommend further research in this area.

## In-Vessel Core Melt Progression

The staff proposes relatively modest expenditures for core-melt progression research. The purpose of the work is said to be:

- the resolution of the question of whether to expect TMI-like blockage as a general behavior for BWRs, and

- to provide some technical basis for validation of blocked-pool models under development, and their predictions regarding the failure location of the crust and the melt relocation into the bottom head.

The above items, along with new models, may permit better estimates of the amount, superheat, metal content, and timing of melt relocation into and subsequent failure of the bottom head. This should provide a basis for better models for quantifying risk. If interpreted properly, the results may also provide guidance in the choice of accident management strategies, assist in the Safety Goal Policy implementation, and remove some of the uncertainty from cost/benefit analysis for backfit decisions.

We suggest, however, that the models that result from this work should be taken as representing only one possible severe accident progression. Future severe accidents, if they occur, may take as unexpected a course as those few that we have experienced. Thus predictions of their course and consequences with models based on limited past experience may be misleading. Analyses of the type reported by Dr. S. Levy (S. Levy, Inc.) in the SASM report could be useful for evaluating the uncertainties associated with such incomplete models.

We also believe that additional fundamental separate effects experiments are needed to better define the crusting behavior and the thermal hydraulics associated with molten pool conditions.

## Lower Head Failure Analysis

Lower head failure analysis (NUREG/CR-5642) of the TMI-2 vessel should be of considerable value if it can be shown that what happened there has general applicability. We suggest that further attention be given to:

- How typical is the TMI-2 accident, even for a PWR, and how well is it understood? For example, it was reported to us that SCDAP/RELAP5 still does not provide a good estimate of the lower head temperature rise.

- What are the uncertainties or the contributors to uncertainty in the results of the lower head failure analysis?

## Review of Severe Accident Codes

We were told that a program of peer review of the codes that RES expects the NRC staff to use over the next few years is under way. Dr. B. Boyack of Los Alamos National Laboratory (LANL) reported on a peer review of MELCOR that has been completed (LA-12240). After an extensive study of the code, the review group, chaired by Dr. Boyack, reported a significant number of deficiencies. It appears that the code should be used with considerable caution until these deficiencies have been corrected. It would also be desirable, before deciding on performance goals for the code, to decide how it is to be used in the regulatory process. We note that it is not being used in the formulation of the source term, which will replace the one that has been used as part of the siting rule (10 CFR Part 100). It is not clear whether the staff plans to use MELCOR in evaluation of IPE results. Such use appears undesirable until the code has been improved.

In light of the rather significant number of problems identified by the peer review, the RES staff should consider the development of procedures to make it less likely that so many problems would exist at such an advanced stage of a code's development.

We understand that a peer review of the SCDAP/RELAP5 code is under way. Since the results are not yet available, we choose not to comment generally on that code in this report. However, we are concerned that the modeling of parts of the severe accident sequence, which the code treats, are said to be based on bounding models rather than on best estimates. This could lead to generation of misinformation, especially if used in formulating accident management strategies, or in evaluating the results of Level 2 and Level 3 PRAs that may be submitted in response to the IPE program.

## Use of Risk Analysis in the Planning of Severe Accident Research

We are not convinced that enough attention is being given to the results of risk analysis in the planning of severe accident research. Both operating experience and analysis provide convincing evidence that severe accidents are low-probability and in many cases low-risk events. Further, as the industry accumulates additional experience, the risk should decrease. Indeed, there are some who would argue that the risk is already sufficiently low that additional research is unwarranted. We have not yet reached that conclusion. Nevertheless, we would like to see more evidence that the choice of research areas and the approach to the research is made with risk reduction as a principal focus.

The work at SNL described by Dr. F. Harper may be an effort in this direction. It is, however, at a very formative stage. The general approach, i.e., development of simplified event trees to approximate complex structures such as those found in NUREG-1150, might be a useful complement to engineering judgment in planning research or in making closure decisions on severe accident issues.

Whatever method is finally used, we believe that more attention should be given to the risk expected from an accident scenario before investments are made in its further elucidation.

Summary of Comments on Specific Activities

- We see no reason for further work on the Mark I early containment failure issue.

- The work on the chemical form of iodine released to containment provides important input to formulation of a new siting source term. The implication of the new information to risk of existing plants should be explored.

- The experimental program on DCH is soundly based, and should resolve the issue.

- We do not believe that some important aspects of the hydrogen issue have received the attention they deserve.

- Existing information is adequate to treat core-concrete interaction on a dry floor.

- Debris coolability is still an open issue. It will probably not be resolved by existing or planned programs.

- The question of energy release associated with violent interaction of liquid corium and water is unresolved, and a resolution is not in sight. We recommend additional research in this area.

- Significant weaknesses have been identified by the peer reviewers of the MELCOR code. Decisions on the use of severe accident codes and on their required capability are needed before plans for further developments are made.

- We endorse, with the caveats noted, the core melt progression program.

## CLOSING COMMENTS

The description of the Severe Accident Research Program Plan provided by draft NUREG-1365, Revision 1, is a significant improvement over previous reports that we have reviewed. The descriptions of the proposed research are generally clear and specific. The report defines a goal for the program, i.e., the exploration of phenomena that are expected to influence early containment failure.

We see a need for better communication among the various units working on parts of a larger problem. During the course of our review, we encountered several examples of lack of communication between the Accident Evaluation Branch and other branches engaged in closely related work. For example, we asked about the MACCS code, a key code in the evaluation of severe accident risk. The answer we got was that it was in another branch. Yet it is the MACCS code that eventually calculates risk, and unless its limitations and capabilities are well understood, information provided as input to the code may not be appropriate. We received a similar response when we asked about work on component heating due to natural convection of gases in a core damaging accident. But if either steam generator tubes or other upper reactor coolant system components are overheated to failure by this process, the course and consequences of the accident can be markedly affected.

Finally, lest this report seem overly negative, we emphasize that we concentrated our comments primarily on areas that were perceived to require further attention. We thank the NRC staff for the time and effort that was put into preparing for the many presentations that were part of this review. In general the presentations were well organized and well presented, and our questions were dealt with patiently and with good humor.

Dr. Thomas S. Kress did not participate in those Committee deliberations that would impact directly on his outside interests.

Sincerely,

David A. Ward
Chairman

<u>References</u>:

1. Memorandum dated April 22, 1992, from Brian W. Sheron, Office of Nùclear Regulatory Research, NRC, for R. F. Fraley, ACRS, Subject: Severe Accident Research Program Plan Update, attaching NUREG-1365, Revision 1, April 1992 (Draft Predecisional)
2. U. S. Nuclear Regulatory Commission, NUREG/CR-5423, "The Probability of Liner Failure in a Mark-I Containment," T. Theofanous, et al. (UCSB), August 1991, with Appendix K, Post-Workshop Summary Comments by the Experts, including "Recommendations for Additional Technical Work, Mark I Shell Survivability Issue," S. Hodge, November 12, 1990

3. U. S. Nuclear Regulatory Commission, NUREG/CR-5732, "Iodine Chemical Forms in LWR Severe Accidents" (Draft Report for Comment), E. Beahm, et al. (ORNL), July 1991
4. U. S. Nuclear Regulatory Commission, NUREG/CR-5809, "An Integrated Structure and Scaling Methodology for Severe Accident Technical Issue Resolution" (Draft Report for Comment), Technical Program Group, November 1991, with Appendix G, "Amount of Material Involved In DCH During a PWR Station Blackout Transient," S. Levy (S. Levy, Inc.)
5. U. S. Nuclear Regulatory Commission, NUREG-1150, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," Office of Nuclear Regulatory Research, December 1990
6. U. S. Nuclear Regulatory Commission, Draft NUREG/CR-5642, "Light Water Reactor Lower Head Failure Analysis," J. Rempe, et al. (EG&G), March 1992 (Draft Predecisional)
7. Los Alamos National Laboratory, LA-12240, "MELCOR Peer Review," B. Boyack, et al., March 1992
8. Verbal presentation by Dr. D. Powers (SNL) to the ACRS Severe Accidents Subcommittee, October 21, 1991
9. Verbal presentation by Dr. F. Harper (SNL) to the ACRS Severe Accidents Subcommittee, May 27, 1992
10. Letter dated April 24, 1990, from Carlyle Michelson, Chairman, ACRS, to Kenneth M. Carr, Chairman, NRC, Subject: Severe Accident Research Program

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
SUBCOMMITTEE MEETING MINUTES:
SEVERE ACCIDENTS
MARCH 1, 1996
ROCKVILLE, MARYLAND

## INTRODUCTION

The ACRS Subcommittee on Severe Accidents met on March 1, 1996, at 11545 Rockville Pike, Rockville, Maryland, in Room T-2 B3. The purpose of the meeting was to gather information on the status of the severe accident research program and on the progress of implementing severe accident management programs at operating nuclear reactors. The entire meeting was open to public attendance. Mr. Noel Dudley was the cognizant ACRS staff engineer for this meeting. The meeting was convened at 8:30 a.m. and adjourned at 3:05 p.m.

## ATTENDEES

### ACRS

M. Fontana, Chairman
G. Apostolakis, Member
I. Catton, Member
T. Kress, Member
W. Lindblad, Member
N. Dudley, ACRS staff

### INDUSTRY

D. Modeen, NEI

### NRC STAFF

C. Ader, RES
S. Basu, RES
A. Behbahani, RES
W. Hodges, RES
T. King, RES
R. Lee, RES
A. Malliakos, RES
R. Palla, NRR
A. Rubin, RES
C. Tinkler, RES

There were no written comments or requests for time to make oral statements received from members of the public. An attendance list of other members of the NRC staff and public is available in the ACRS office files. Public participation during this meeting was limited to the presentations by the above named industry representative.

DISCUSSION OF AGENDA ITEMS

Dr. Mario Fontana, the Subcommittee Chairman, convened the meeting at 8:30 a.m., and noted that the ACRS last reviewed severe accident issues in August 1992. He stated that the planned presentations would update the Subcommittee on the progress made by the staff in bringing severe accident issues to closure.

Staff Presentation Concerning The Severe Accident Research Program:

Mr. Charles Ader, RES, introduced the staff presentation by reviewing the background, present status, and future plans for the severe accident research program. The severe accident research program began after the Three Mile Island accident in 1979, and was formalized in SECY 88-147, "Integration Plan for Closure of Severe Accident Issues," dated May 25, 1988. At present, all six elements identified in SECY 88-147 have been closed or integrated into other ongoing NRC programs with the exception of the severe accident research program and severe accident management. Larger experimental programs are coming to a close and the issue of direct containment heating (DCH) is close to resolution.

The Subcommittee and the staff discussed the need for defining objective closure criteria for severe accident issues and the importance of maintaining expertise in the severe accident area.
Mr. Lindblad encouraged the staff to establish what deficiencies exist in the current understanding and to develop programs to extinguish the deficiencies. Mr. Wayne Hodges, RES, stated that there are few specific issues that require answers to understand phenomena and that most research is directed at maintaining a certain level of severe accident expertise. Dr. Fontana opined that it is important to maintain research capabilities.

**Direct Containment Heating**: Mr. Charles Tinkler, RES, presented the objectives, status, findings, and future plans for research on direct containment heating (DCH) being conducted at the following facilities:

- SURSEY 1/10 scale facility at Sandia National Laboratories
- Purdue University 1/10 scale separate effect test facility
- Containment Technology Test 1/6 scale facility at Sandia National Laboratories
- COREXIT 1/40 scale DCH integral test facility at Argonne National Laboratory

The staff has completed a peer review of NUREG/CR-6338, "Resolution of the Direct Containment Heating Issue For All Westinghouse Large Dry Containments or Subatmospheric Containments." The staff looked at both the top-down and bottom-up scaling issues associated with the test results. The staff determined that the major mitigator of DCH loads is the interception of debris along the core melt ejection trajectory paths. The staff varied initial test conditions including the pressure at which the reactor vessel fails, the hole size, and the amount of melt mass.

The staff is using similar analysis techniques to review other types of containments. Unlike the Westinghouse containment designs, the CE designs allow 60 to 70 percent of core melt to reach the dome. As a result, the staff undertook testing to identify additional mitigation mechanisms, such as the existence of water in the lower head. For ice condenser containments, the staff plans to consider the probability of unintentional depressurization. For boiling water reactor containments, the staff expects to address the DCH issue from a probabilistic standpoint.

The Subcommittee and the staff discussed the assumptions and sensitivities associated with the analytical model used to calculate the cumulative probability of core damage and the use of the ROAAM process. Mr. Tinkler explained that the model accounted for the existence of water, the amount of core ejected, metal fractions, coincident hydrogen burns, and conservation of melt mass and velocity. Dr. Apostolakis commented that based on the graph of cumulative probability of core damage to containment pressure, there is very little uncertainty associated with the calculated results. Mr. Hodges stated that the results indicate that the model is insensitive to the uncertainties.

**Fuel-Coolant Interaction Research Program**: Mr. Tinkler presented the objectives, status, findings, and future plans for research of fuel-coolant interactions being conducted at the following facilities:

- FARO quenching facility in Japan
- KROTOS steam explosion facility in Japan
- Fragment Chemical Augmentation facility at Argonne National Laboratory

The experts who attended the SERG-2 workshop held June 15-16, 1995, formed the consensus opinion that Alpha-Mode failure was resolved from the standpoint of risk significance. The workshop attendees agreed that additional work was required to assess fuel-coolant ex-vessel interactions.

The Subcommittee and the staff discussed reviewing the results of German experiments, the use of molten salts, the lack of understanding of phenomenological aspects of fuel-coolant interactions, the effect of different temperature water and melts, and the development of closure criteria. Mr. Tinkler stated that there is sufficient uncertainty associated with test results to presume that a triggering mechanism exits. Dr. Catton noted that General Electric films of underwater molten lava flows do not show steam explosions because of the easy crust formation on a low thermal diffusive material.

**Debris Coolability**: Mr. Tinkler presented the status and summarized the findings of debris coolability tests. The tests performed at Argonne National Laboratory under the Melt Attack and Coolability (MACE) Program demonstrated questionable coolability of molten core debris. Larger scale tests are planned. The staff inferred that an objective of the PHEBUS program is to collaborate with foreign experts in the development of a world wide understanding of debris coolability.

**Source Term**: Mr. Tinkler presented the status of using NUREG 1465, "Accident Source Terms for Light-Water Nuclear Power Plants," for the AP600 design and for operating nuclear power plants. The revised source terms take into consideration expanded delay time for release of radionuclides after onset of a severe accident, the chemical forms of radionuclides, and the cladding gas release interval.

**Lower Head Integrity**: Mr. Alan Rubin, RES, presented the current programs, accomplishments, and future plans for testing retention of molten core material in reactor pressure vessels. The four major areas of research are as follows:

- reactor pressure vessel external flooding experiments and analyses at Pennsylvania State University (Penn State)
- lower head creep fatigue failure experiments at Sandia National Laboratories
- review of in-vessel debris coolability experiments conducted by Fauske and Associates
- experimental and analytical investigations of retaining prototypic molten core material in the reactor pressure vessel lower head at the Russian Research Center, Kurchatov Institute (RASPLAV)

The Subcommittee and the staff discussed the consequences of flooding the cavity in operating plants and in advanced reactor designs, methods of heating test cylinders, global review approaches, use of finite element codes, and spacial heating of experimental vessels. In response to Subcommittee questions concerning the scaling analysis used at Penn State and the location of critical heat flux, the staff provided a report on lower head boiling experiments conducted at Penn State.

**Hydrogen Combustion Research**: Mr. Rubin presented the current programs, accomplishments, and future plans for the following:

- hydrogen combustion behavior and steam condensing environment experiments conducted at Sandia National Laboratories
- high temperature, high speed, combustion experiments conducted at Brookhaven National Laboratory
- low speed hydrogen combustion experiments conducted at the California University of Technology

- hot jet initiation, deflagration to detonation transition, and hydrogen igniters placement experiments conducted at the Russian Research Center, Kurchatov Institute

- passive autocatalytic recombiner experiments conducted at Sandia National Laboratories

The Subcommittee and the staff discussed the effect of time and contaminates on igniters, poisoning of passive recombiner catalysts, the likelihood of hydrogen detonation, and foreign research reports.

**Core Melt Progression**: Mr. Rubin presented information on the in-vessel progression tests conducted at Sandia National Laboratories and the associated analytical models. Dr. Catton requested a copy of an NRC report that contained a list of key in-vessel melt progression uncertainties.

**Future Activities**: Mr. Ader stated that the staff is continuing research in the areas of fuel-coolant interactions, lower head integrity, source term, and hydrogen combustion. The staff is maintaining and improving the SCDAP/RELAP5 detailed in-vessel code, the CONTAIN containment code, and the VICTORIA fission product code. The staff participates in international research efforts through the RASPLAV, FARO/KROTOS, ISPRA, and PHEBUS projects. The staff is maintaining a forum for exchanging information though the MCAP program, which is a MELCOR code assessment user's group.

Staff Presentation on Severe Accident Management:

Mr. Robert Palla, NRR, defined the objective of the Severe Accident Management Program as having licensees implement accident management plans, which would provide a framework, procedure, guidance, and training program for severe accidents. He explained the background for the Nuclear Energy Institute's submittal of a formal industry commitment to implement severe accident management. Vendors of pressurized water reactors have prepared Severe Accident Management Guidelines (SAMG) for use in technical support centers. General Electric is developing Emergency Response Guidelines (ERG) changes for use in main control rooms.

Mr. Palla provided details of the proposed temporary instruction, which the staff plans to use to verify implementation of licensee commitments. The staff plans to achieve closure on a plant-by-plant basis using industry and NRC developed evaluation guidance and methods. The NRC will maintain oversight of utility capabilities.

The Subcommittee and Mr. Palla discussed the different accident management strategies, communication between and the authority of main control room and the technical support center staffs, whether the temporary instruction is performance based, and the use of simulators for severe accident training. Dr. Catton recommended that the Subcommittee review the SAMG and the ERG during a future meeting.

Nuclear Energy Institute (NEI) Presentation on Severe Accident Management:

Mr. David Modeen, NEI, presented the formal industry position regarding the actions required for achieving closure, performing plant specific activities, and conducting regulatory oversight. He stated that severe accident management was an enhancement of present licensee capabilities and was not a new program. He requested that the staff provide clear delineation of NRC expectations and detailed plans for achieving closure on severe accident management issues.

The Subcommittee, Mr. Palla, and Mr. Modeen discussed quantifying the effect on core damage frequency of following severe accident management guidelines and procedures. They also discussed INPO involvement in developing training programs, and the use of self-evaluations and audits in lieu of NRC inspections.

Subcommittee Discussions:

The Subcommittee discussed proposed future Subcommittee meetings, preparing an ACRS report on the severe accident research program, and information requested from the staff. Dr. Fontana adjourned the meeting at 3:00 p.m.

SUBCOMMITTEE RECOMMENDATIONS

The Subcommittee recommended gathering additional information concerning severe accident codes prior to the April 1996, ACRS meeting, in support of preparing a Committee report on the severe accident research program. The Subcommittee recommended hearing additional information regarding the integration of SAMGs and ERGs during a future meeting.

FOLLOWUP ACTIONS

During the meeting the staff agreed to provide the Subcommittee copies of the following documents:

- Agenda to the CSARP meeting [received March 21, 1996]
- Reports on RASPLAV activities [received March 21, 1996]
- Penn State report "Steady-State Observations and Theoretical Modeling of Critical Heat Flux Phenomena On A Downward Facing Hemispherical Surface" [received March 1, 1996]
- Draft Report On Core Melt Procession Uncertainties [received March 21, 1996]

- Proceedings from the Northeast Utilities CSNI meeting [received March 12, 1996]

BACKGROUND MATERIAL PROVIDED THE SUBCOMMITTEE FOR THIS MEETING

- Excerpts from SECY-88-147, "Integrated Plan for Closure of Severe Accident Issues," dated May 25, 1988
- SECY-95-004, "Status of Implementation Plan for Closure of Severe Accident Issues, Status of the Individual Plant Examinations and Status of Severe Accident Research," dated January 4, 1995
- SECY-94-166, "Status of Implementation Plan for Closure of Severe Accident Issues, Status of the Individual Plant Examinations and Status of Severe Accident Research," dated June 17, 1994
- Letter dated August 18, 1992, from David Ward, Chairman ACRS, to Ivan Selin, Chairman NRC, Subject: Severe Accident Research Program Plan
- Letter dated April 24, 1990, from Carlyle Michelson, Chairman ACRS, to Kenneth Carr, Chairman NRC, Subject: Severe Accident Research Program Plan
- NUREG/CR-6109, "The Probability of Containment Failure by Direct Containment Heating in Surry," dated May 1995
- Report NEI 91-04 Revision 1, "Severe Accident Issue Closure Guidelines," by the Nuclear Energy Institute, dated December 1994
- NRC Draft Temporary Instruction 2515/XXX , "Licensee Implementation Of Accident Management," provided to ACRS February 12, 1996

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NOTE: Additional details of this meeting can be obtained from a transcript of this meeting available in the NRC Public Document Room, 2120 L Street, N.W., Washington D.C. 20006, (202) 634-3274, or can be purchased from Neal R. Gross and Company Incorporated, Court Reporters and Transcribers, 1323 Rhode Island Avenue, N.W., Washington, D.C. 20005, (202) 234-4433.

4

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
430TH MEETING
ROCKVILLE, MARYLAND
APRIL 11, 1996

GRADED QUALITY ASSURANCE PROGRAM

- TABLE OF CONTENTS -



Cognizant ACRS Member: C. Wylie

Cognizant ACRS Staff Engineer: M. El-Zeftawy

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
430TH MEETING
ROCKVILLE, MARYLAND
APRIL 11, 1996

# GRADED QUALITY ASSURANCE PROGRAM

- TENTATIVE AGENDA -

| | | Approx. Time |
|---|---|---|
| I. | Introductory Remarks, Subcommittee Chairman<br>C. Wylie (ACRS) | 5 mins. |
| II. | NRC Staff Presentation: S. Black, R. Gramm (NRR) | 65 mins. |
| | • Background<br>• GQA Overview<br>• Regulatory Basis<br>• Methodology<br>• NEI/GQA Initiative<br>• Volunteer Plants Implementation<br>• Interface and Future Staff Activities | |
| III. | Industry Presentation: | 15 mins. |
| | • Palo Verde, Mr. Carter Rogers<br>• Grand Gulf, Mr. Mike Meisner<br>• South Texas, Mr. Roy Rehkuler | |
| IV. | General Discussion | 5 mins. |

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
430TH MEETING
ROCKVILLE, MARYLAND
APRIL 11, 1996

GRADED QUALITY ASSURANCE PROGRAM

**- STATUS REPORT -**

**PURPOSE:**

To inform the Committee of the current status of the Graded Quality Assurance (GQA) initiative.

**BACKGROUND:**

In 1993, the Executive Director for Operations (EDO) established the Regulatory Review Group (RRG). The RRG reviewed the power reactor regulations and related processes and emphasized the potential application of performance-based regulations and the use of risk insights.

In the area of QA, the RRG determined that the existing regulations were performance-based and that 10 CFR Part 50, Appendix B contains provisions for the graded application of QA controls over activities affecting the quality of structures, systems, and components (SSCs) to an extent commensurate with their importance to safety.

Although both Appendix B and the associated industry standards allow a large degree of flexibility, the licensees and the NRC staff have been reluctant to make major changes in established QA practices.

The GQA initiative jointly undertaken by the industry and the NRC is intended to provide a safety benefit by allowing licensees and NRC to preferentially allocate resources to higher safety significant items, and to provide cost savings by reducing resources spent on lesser safety significant items (SECY-95-059 - Attachment 1).

The Nuclear Energy Institute (NEI) prepared a draft "Guideline for Implementing a Graded Approach to Quality" document dated June 1995 (Attachment 2). The NRC staff prepared a draft evaluation guide, "Development of Graded Quality Assurance Programs," dated January 1996 (Attachment 3). Licensees developing GQA programs will consider various methods to adjust their QA programs to accommodate their individual needs. Irrespective of a licensee's specific approach, the NRC envisions a GQA program to have four essential elements as follows:

- a process that determine the safety significance of SSCs in a reasonable and consistent manner

- the implementation of appropriate QA controls for SSCs, or groups of SSCs, according to safety function and safety significance

- an effective root-cause analysis and corrective action program

- a means for reassessing SSC safety significance and QA controls when new information becomes available

The NRC staff indicated that the safety significance classification of the SSCs can be based either on deterministic considerations or a combination of deterministic and probabilistic considerations. However, the draft NRR evaluation guide report is written in terms of the combined approach since it is the preferred method and has also been adopted by the volunteer plants developing GQA programs.

**FUTURE ACTION:**

This briefing is for information. However, the Committee may wish to propose a course of action to follow the GQA program's progress.

ATTACHMENT 1

POLICY ISSUE
(Information)

arch 10, 1995 SECY-95-059

FOR: The Commissioners

FROM: James M. Taylor
Executive Director for Operations

SUBJECT: DEVELOPMENT OF GRADED QUALITY ASSURANCE METHODOLOGY

PURPOSE:

To inform the Commission of the current status of the graded Quality Assurance (QA) initiative.

BACKGROUND:

On January 4, 1993, the Executive Director for Operations (EDO) established the Regulatory Review Group (RRG). The RRG conducted a disciplined review of power reactor regulations and related processes, placing special emphasis on the potential application of performance-based regulations and the use of risk insights.

In the area of QA, the RRG determined that the existing regulations were performance based and that 10 CFR Part 50, Appendix B contains provisions for the graded application of QA controls over activities affecting the quality of structures, systems, or components (SSCs) to an extent consistent with their importance to safety. However, the RRG noted that licensees had not availed themselves of the intended flexibility of 10 CFR Part 50, Appendix B. The RRG concluded that some of the implementing documents and guidance would need to be revised in order to implement Appendix B in a more performance-based and graded manner.

Contact:
Robert Latta, NRR
415-1023

NOTE: TO BE MADE PUBLICLY AVAILABLE IN 5 WORKING DAYS FROM THE DATE OF THIS PAPER.

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On December 16, 1993, members of the NRC staff met with representatives of the industry and the Nuclear Utilities Management and Resources Council (NUMARC) to initiate an effort to develop a consensus on a conceptual approach toward the graded application of 10 CFR Part 50, Appendix B, quality provisions. This meeting culminated with the general understanding that the current regulations allow adequate flexibility to accommodate the graded implementation of QA in a manner commensurate with the safety significance of SSCs. NUMARC proposed to develop a guidance document for implementing QA programs in a graded, performance-based manner. Specifically, the process envisioned would build upon the Maintenance Rule implementation guidance contained in NUMARC 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants." The Maintenance Rule implementation process, endorsed by Regulatory Guide 1.160, includes the use of an expert panel to evaluate both probabilistic risk assessment (PRA) and deterministic insights to gauge the relative safety significance of SSCs.

DISCUSSION:

Subsequent to the meeting on December 16, 1993, the staff developed a plan that identified project milestones for the graded QA project. A goal was established for implementing a pilot graded QA program in the fall of 1994. Attachment 1 contains the initial schedule that was developed by the staff.

The graded QA approach envisioned by the staff would separate SSCs into high-safety significant and low-safety significant categories with quality verification activities applied commensurate with the safety significance or safety functions of the SSCs. The scope of the SSCs covered under the proposed graded approach was considered equivalent to the Maintenance Rule (10 CFR 50.65) which includes both safety-related and non-safety-related SSCs.

The safety benefit derived from the proposed graded QA methodology would be that licensees and the NRC could apply the majority of the available quality verification and inspection resources to the equipment where the greatest safety benefit could be achieved.

A chronology of events for the graded QA project is depicted in Attachment 2. Following the public meeting on December 16, 1993, the staff conducted seven meetings of the Working Groups and three meetings of the Steering Groups with the Nuclear Energy Institute (NEI, previously NUMARC) and industry representatives. These meetings were open to the public. The primary objective of the Working Group meetings was to define a sufficiently detailed methodology for the graded application of quality provisions contained in 10 CFR Part 50, Appendix B, commensurate with the safety significance and function of SSCs. This methodology was envisioned to then be implemented in a pilot program environment. During the meetings of the Steering Groups, NRC and NEI representatives discussed conceptual aspects related to the positions and approaches developed by the NRC and NEI Working Groups.

Additionally, to facilitate the development of the graded QA methodology, the staff performed information gathering visits to three of the seven plants proposed for the pilot study. During these visits to North Anna (Virginia Power), Grand Gulf (Entergy), and Palo Verde (Arizona Pubic Service Company), the staff interacted with utility representatives to gain additional insights into the proposed graded QA implementation process. The four other plants proposed by NEI for participation in the pilot study to develop a graded QA methodology were Arkansas Nuclear One (Entergy), Byron (Commonwealth Edison), Crystal River (Florida Power Corporation) and Monticello (Northern States Power).

NEI provided the staff a copy of their draft "Guideline for Industry Pilot Project - Implementation of Graded Performance-Based Approach to Quality" on April 11, 1994. On April 20, 1994, the staff sent NEI comments on the draft guideline, emphasizing that the document needed additional details. During a meeting of the Steering Groups on May 12, 1994, NEI requested that the staff provide specific comments to their draft guideline. However, after further consideration, the NRC Steering Group noted that it would be more appropriate to ensure that the staff and NEI were in agreement on the fundamental concept for a graded QA program. The NRC goals and expectations for the graded QA project were conveyed to NEI on June 15, 1994 (see Attachment 3). In that letter, the staff identified the following four essential elements that a graded QA program should include:

(1) A process that, with high confidence, will identify the appropriate safety significance of all SSCs in a reasonable and consistent manner.

(2) An effective root-cause analysis and corrective action program.

(3) The determination of appropriate QA controls for individual SSCs, or groups of SSCs, based upon safety function and [safety] significance.

(4) A means of reassessing SSCs safety significance and QA related controls when new information becomes available.

Subsequently, NEI submitted a revision to its draft guidance document (see Attachment 4) on September 2, 1994. After the staff had evaluated the revised NEI guidance, the staff responded to NEI on October 14, 1994 (see Attachment 5). The staff continued to be concerned that the NEI revised draft document lacked specificity with respect to the four essential elements of graded QA that were envisioned by the staff.

The major issues identified by the staff with regard to the NEI guidance document included:

(1) There was insufficient detail regarding the expert panel composition and function, including the associated quality element assessment process.

(2) There was insufficient detail with respect to a corrective action program to ensure that sufficient information is maintained to conduct effective root-cause determinations and corrective action for low-safety significant SSCs. Also, an inappropriate threshold was established to initiate corrective action (e.g., a safety system functional failure).

(3) There was insufficient justification for the proposal to conduct performance monitoring in lieu of ensuring product quality.

(4) The guidance attempted to circumvent the applicable controls for changing QA commitments specified in 10 CFR 50.54(a).

Based on a commitment made to NEI during a meeting of the Steering Groups, the staff prepared detailed comments on NEI's guidance document and transmitted these to NEI on January 31, 1995 (see Attachment 6).

Because the NEI guidance document did not contain the requisite level of detail, the staff envisioned that two licensees would initiate a pilot project following the incorporation of fundamental staff concepts into the guidance document. The staff was prepared to actively monitor this phase of the pilot project implementation. After (1) gaining confidence in the NEI guidance document, based on the implementation results at the two lead pilot plants, and (2) supplementing the guidance document to reflect the initial lessons learned, the remainder of the pilot plants could then proceed with implementing the methodology. The staff planned to continue to actively monitor and verify the adequacy of the pilot program at the associated facilities. At the conclusion of the pilot phase, lessons learned from application of the guidance at all seven sites would be incorporated into the guidance document. The staff would then develop a draft regulatory guide endorsing the guidance and issue it for public comment.

By letter dated December 21, 1994 (see Attachment 7) NEI identified their future approach to achieve a performance-based graded QA regime. Specifically, NEI stated that it would actively encourage licensees to grade their quality programs using a blend of probabilistic and deterministic insights and that the graded QA pilot project would be deferred. Furthermore, NEI indicated their plans to forward rulemaking petitions to amend 10 CFR 50.54(a) and to enact a performance-based option to 10 CFR Part 50, Appendix B.

On February 9, 1995, the NRC responded to NEI's proposal for the future approach to a graded QA methodology. In this letter (see Attachment 8), the staff indicated its willingness to review any rulemaking petitions submitted by NEI relative to 10 CFR 50.54(a) or 10 CFR Part 50 Appendix B. The staff also stated its availability to conduct further discussions with NEI related to the concept of a performance-based Appendix B regulation that parallels the Maintenance Rule, 10 CFR 50.65. Relative to this performance-based approach, the staff stated that it continues to believe that the effectiveness of established QA programs cannot be gauged solely on the basis of plant performance monitoring, in that the implementation of QA program controls provides assurance that plant equipment will function reliably during routine operations and during design-basis events.

The staff indicated its willingness to continue a dialogue with the industry to address practical ways to improve the manner in which present QA controls are implemented. More importantly, the staff expressed its receptiveness to working with either individual volunteer licensees or groups of licensees to test and evaluate the implementation of graded QA concepts that would more efficiently focus resources on safety-significant aspects of plant activities and equipment.

Given that both the industry and the NRC recognize the potential benefits of implementing QA controls commensurate with safety significance, the staff agrees with NEI's recommendation to licensees to utilize both deterministic and probabilistic insights to grade their QA programs. However, if the industry moves forward with the implementation of a graded QA process on an individual licensee basis, the staff recommends that the individual licensees consider the staff's comments on the NEI graded QA guidance document. As indicated in the NRC's correspondence to NEI dated October 14, 1994, and January 31, 1995, the staff continues to have reservations regarding the completeness of NEI's guidance document. The staff considers that the NEI guidance document constitutes preliminary information applicable for use in a pilot plant environment. Accordingly, the staff comments on NEI's graded QA guidance should be considered by licensees if they utilize this information for revising their programs.

The staff recognizes that the industry's priorities have shifted since the inception of the graded QA initiative. Rather than expending resources to develop a practical methodology and demonstrate the process during a pilot program, the industry has chosen to expand its efforts into the rulemaking arena. Nonetheless, the staff considers that a significant safety benefit can be derived from continuing with the original concept of developing a workable guidance document. To that end, the staff has communicated with several licensees to ascertain their willingness to serve as a volunteer plant during the development of graded QA guidance. Although the NEI guidance document, as amended by the NRC's correspondence to NEI dated October 14, 1994, and January 31, 1995, could be used as appropriate, it is anticipated that each volunteer licensee will adjust the program to suit their individual needs and apply their efforts to different functional areas. The staff would then actively monitor the development and implementation of these volunteer plant programs. Lessons learned during these efforts would be integrated into a more generic guidance document that would then be made available to the rest of the industry. Thus, the guidance would be tested in a sequential manner to validate its completeness and workability. It is also anticipated that this pilot effort will facilitate the development of an integrated and dynamic risk management system in conjunction with NRC's PRA Implementation Plan. At present, several licensees have expressed an interest in participating in the volunteer plant process and the staff is optimistic that this effort will be initiated in the near future.

CONCLUSION:

The staff remains committed to working with individual licensees as practical approaches are developed for implementing a graded QA methodology. Individual licensees who plan on pursuing this approach are encouraged to interact with the staff as they develop their graded QA programs. The staff will continue to inspect and evaluate the implementation of graded QA programs as part of the NRC's normal safety oversight activities. It is expected that the industry and the staff will evaluate the lessons learned from developing a graded approach to QA and that these concepts will be factored into an integrated and dynamic risk management system. Additionally, as the process for implementing graded QA concepts evolves, the staff anticipates conducting periodic workshops to share individual licensee experiences with the rest of the industry. These efforts are intended to culminate in the NRC's endorsement of an industry developed guidance document or the development of a staff directive describing an acceptable graded QA methodology by the end of 1996. The staff will evaluate NEI's proposal for a performance-based QA program through rulemaking following its submittal, scheduled for this fall.

James M. Taylor
Executive Director
for Operations

Attachments:
1. Initial Graded QA Implementation Schedule
2. Chronology of Events for the Graded QA Initiative
3. Letter, J.L. Milhoan to W.H. Rasin, June 15, 1994
4. NEI Revised "Draft Pilot Project Guideline for Implementation of a Graded, Performance-Based Approach to Quality," September 1, 1994
5. Letter, J.L. Milhoan to T.E. Tipton, October 14, 1994
6. Letter, S.C. Black to R. Ng, January 31, 1995, Forwarding NRC Line-in/Line-out Version of NEI Revised Draft Guidance, September 1, 1994
7. Letter, T.E. Tipton to J.L.Milhoan, December 21, 1994
8. Letter, J.L. Milhoan to T.E. Tipton, February 9, 1995

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## GRADED QA IMPLEMENTATION SCHEDULE

| | | |
|---|---|---|
| 12/93 | - | Initial NUMARC Meeting on Graded QA |
| 1/94 | - | First Working Level Meeting |
| 3/94 | - | Visit to North Anna |
| 4/94 | - | Visit to Grand Gulf |
| 6/94 | - | Draft Methodology for Graded QA Implementation |
| 7/94 | - | Visit to Palo Verde |
| 9/94 | - | Implementation of Pilot Graded QA Program |
| 1/95 | - | Evaluate Pilot Program |
| 4/95 | - | Issue Draft Regulatory Guide |
| 6/95 | - | Evaluate Public Comments |
| 1/96 | - | Issue Final Regulatory Guide |

ATTACHMENT 1

# GRADED QA CHRONOLOGY

| | | |
|---|---|---|
| 12/16/93 | - | Initial NUMARC Meeting on Graded QA |
| 1/06/94 | - | First Meeting of Working Groups |
| 2/03/94 | - | Meeting of Working Groups |
| 2/17/94 | - | Meeting of Working Groups |
| 3/02/94 | - | Meeting of Working Groups |
| 3/08/94 | - | Visit to Virginia Power Headquarters |
| 3/18/94 | - | Visit to North Anna Plant |
| 3/23/94 | - | First Meeting of Steering Groups |
| 4/05/94 | - | Visit to Grand Gulf |
| 4/11/94 | - | Meeting of Working Groups |
| 4/11/94 | - | NEI Draft Document of Proposed Methodology for Graded QA Implementation Received by Staff |
| 4/20/94 | - | Staff Provided Comments to NEI Draft Guidance Document |
| 4/26/94 | - | Meeting of Working Groups |
| 5/12/94 | - | Second Meeting of Steering Groups |
| 6/15/94 | - | Staff Provided NEI with Specific Comments on their Draft Guidance Document and defined the Goals and Essential Elements associated with the Graded QA Process |
| 7/14/94 | - | Visit to Palo Verde |
| 7/21/94 | - | Third Meeting of Steering Groups |
| 9/02/94 | - | Staff Received Revised NEI Draft Guidance Document for Implementation of a Graded Performance-Based Approach to Quality |
| 9/14/94 | - | Meeting of Working Groups |
| 10/14/94 | - | Letter issued to NEI with staff comments pertaining to their revised guidance document for Implementation of Graded QA |

ATTACHMENT 2

12/21/94 - Received letter from NEI outlining future approach for graded QA

1/31/95 - Issued Detailed Staff Comments on NEI's Draft Guidance Document

2/09/95 - Issued Response to NEI's letter dated 12/21/94 describing NRC's plans for graded QA initiative

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

June 15, 1994

Mr. William Rasin
Nuclear Energy Institute
1776 Eye Street, N.W.
Suite 300
Washington D.C. 20006

Dear Mr. Rasin:

During the last meeting of the NRC and NEI Graded Quality Assurance (QA) Steering Groups on May 12, 1994, the staff committed to provide you with specific comments on your draft "Guideline for Industry Pilot Project — Implementation of Graded Performance-Based Approach to Quality." However, upon further consideration, the NRC Steering Group believes it would be more appropriate to first ensure that we are in agreement on the fundamental concept for a graded QA program. Therefore, the purpose of this letter is to provide the NRC Steering Group's views on some basic issues such that, if we are not in agreement, we can first address the conceptual differences and then move forward on this important initiative.

First, the purpose, or goal, for developing guidance for a graded QA program is to realize a savings from tailoring QA controls based on the safety significance of the structures, systems, and components (SSC) involved. Such an approach should be possible without any significant impact on plant safety, and would allow both the staff's and licensees' QA resources to be focused on the more safety-significant SSCs. We believe at this point that this fundamental goal may be accomplished by the development of guidance for the implementation of a graded Appendix B QA program. The pilot program should provide a meaningful evaluation of the guideline and allow us to determine if rulemaking may be necessary.

Secondly, the essential elements of any graded QA program should include the following: 1) a process that, with high confidence, will identify the appropriate safety significance of all SSCs in a reasonable and consistent manner, 2) an effective root-cause analysis and corrective action program for safety-significant SSCs, 3) the determination of appropriate QA controls for individual SSCs, or groups of SSCs, based upon safety function and significance; and 4) a means of reassessing SSC safety significance and QA related controls when new information becomes available. Each of these elements is discussed in further detail in the enclosure.

If there is agreement on the goal and elements of such a program, we can then proceed towards implementation of the program. We believe that the goal expressed above and the associated elements need to be specifically identified and discussed in NEI's guidance document. At that point we can provide specific comments on the guidance document.



ATTACHMENT 3

~~9406200070~~ 4pp.

Be assured that we continue to strongly support this effort. It is important that our regulatory requirements are effective and efficient and we are committed to working with you towards achieving that end.

The NRC Steering Group looks forward to your timely response to this letter. After receiving your response, we believe a meeting to discuss future activities, especially the upcoming pilot program, would be appropriate.

James L. Milhoan

James L. Milhoan
Deputy Executive Director
for N lear Reactor Regulation,
Regional Operations and Research

cc: Jack Skolds c/o NEI
William Bohlke c/o NEI

Enclosure:
As stated

Enclosure

## ESSENTIAL ELEMENTS OF A GRADED QUALITY ASSURANCE PROGRAM

1. Establishment of Risk Significance.

The nature and magnitude of an SSC's contribution to plant risk should determine the type and amount of QA controls and practices applied to that SSC. In our discussions, we have agreed that the process developed for implementing the maintenance rule, i.e., NUMARC 93-01, "Industry Guidelines for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," may offer a practical means for establishing the basis for a graded QA program. The philosophy applied in implementing the maintenance rule, i.e., the establishment of criteria or goals, performance or condition monitoring, and appropriate corrective actions when failures occur, should also be applicable to a graded QA program. For example, those SSCs that are determined to be risk significant within the scope of the maintenance rule are likely to be risk significant within the context of a graded QA program.

The expert panel described in NUMARC 93-01, and further expanded upon in NUMARC 93-02, "A Report on the Verification and Validation of NUMARC 93-01," would appear to have an extremely important role in both the maintenance rule and graded QA applications. The panel's role would be to establish the high, low and no safety-significant categories. In this regard, we understand that SSCs in the high safety-significant category would continue to reflect the Appendix B program as now constituted, and SSCs having no safety significance could be removed from the scope of Appendix B (in accordance with existing regulatory provisions). The real benefit from this activity is in tailoring the specific nature of needed QA controls for SSCs of low safety significance. The expert panel's role is critical to this determination and would be based upon:

a) The results of risk-significant determination methodologies, and

b) Deterministic factors associated with the nature and consequences of failure that could affect safety margins for SSCs such as passive pressure boundary components and standby safety systems.

The decision process needs to be described well enough such that it is likely to provide the classification results with reasonable confidence and consistency.

2. Effective Root Cause Analysis and Corrective Action Program.

NEI and the NRC have agreed that graded QA programs need to include an effective root cause analysis and corrective action program. This program would determine whether the failure of any SSC within a low-safety-significant category is acceptable in light of its specified safety significance and QA-related controls. Therefore, the NRC believes that the expert panel would need to pre-identify the necessary procurement, design, installation, and other records necessary to be able to conduct effective root cause analyses and corrective action determinations. In addition, an adequate corrective action program would require that the locations and applications of SSCs similar to that which failed should also be retrievable.

Consistent with the approach agreed upon for the implementation of the maintenance rule, failures of low safety-significant SSCs would have to be evaluated. If a second failure subsequently occurs that is determined to be QA-preventable and/or is due to inadequate earlier corrective actions, the need for augmented QA controls should be considered until corrective actions are shown to be effective.

3. Determination of Appropriate QA Controls.

Once the expert panel has determined the safety significance of an SSC to be low, the panel would then determine the specific nature and extent of the QA controls and practices to be applied to the SSC to, among other things, support an effective root cause analysis and corrective action program as discussed in Item 2 above. This determination would include the consideration of the safety function of the SSC and non-maintenance related factors such as design, procurement, fabrication, construction, installation, testing, and human factor issues. This "grading" of QA controls is critical in order to assure that the margin of safety continues to be adequate and yet unnecessary economic burdens are minimized.

4. Incorporation of New Information and Operational Experience.

The final element would be a mechanism for the timely reevaluation of an earlier safety-significance determination and related lessening of QA controls when new information is obtained. This information could be the result of a change to the plant's IPE, an equipment or system modification, adverse performance monitoring results, or operating experience from other plants. The NRC believes that a graded QA program should be a dynamic process and that the consideration of new information needs to be an inherent part of the program.

**DRAFT**
(9/1/94)

# DRAFT PILOT PROJECT GUIDELINE FOR IMPLEMENTATION OF A GRADED, PERFORMANCE-BASED APPROACH TO QUALITY

**NUCLEAR ENERGY INSTITUTE**

**SEPTEMBER 1994**

18

~~9409150058~~ 37pp.

ATTACHMENT 4

DRAFT

(9/1/94)

TABLE OF CONTENTS

**DRAFT**
(9/1/94)

## DRAFT GUIDELINE DOCUMENT PROCESS

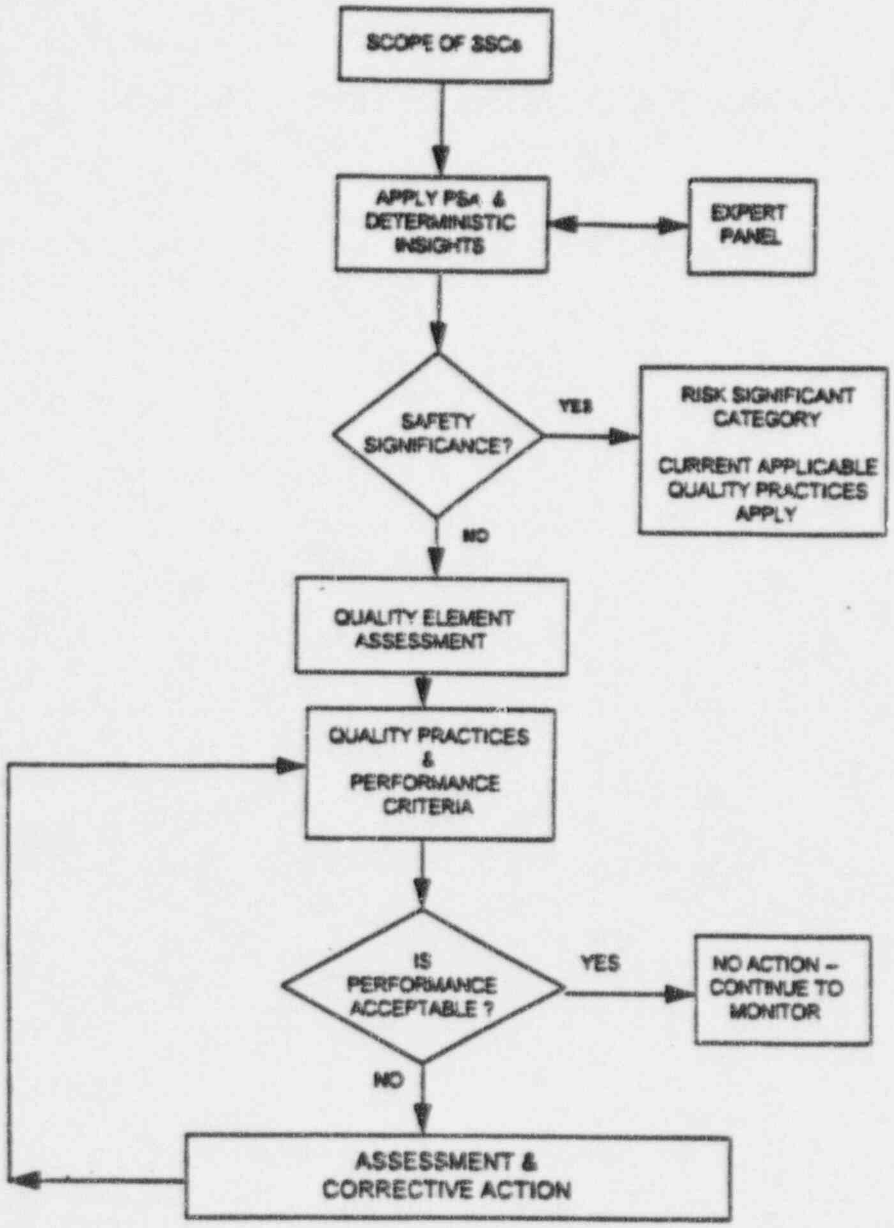


FIGURE 1

## EXECUTIVE SUMMARY

This guideline provides the general framework and guidance to support a pilot project for implementing a graded, performance-based approach to quality. On completion of the pilot project, the guidance will be amended as necessary to incorporate the lessons learned. A report will also be published that will provide examples, based on the pilot project experiences, to support an industrywide implementation. It is envisioned that other companies will base their decision to move to a graded, performance-based approach to quality on the final guidance document and the pilot project report, following NRC endorsement through a Regulatory Guide.

The pilot project builds on the momentum being established through the implementation of the Maintenance Rule towards a performance-based regulatory regime. This document should be read in conjunction with the guidance for implementing the Maintenance Rule, NUMARC 93-01, *Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants*. The continuing transition to a performance-based regime in the area of quality implementation moves the emphasis for quality from procedural programmatic compliance towards product quality and plant performance.

The implementation of a graded, performance-based quality regime is broader than the implementation of the Maintenance Rule. It involves assessments and evaluations of safety functional failures and deviations from the performance criteria in all applicable areas, not just those associated with a maintenance preventable function failure.

The objective of implementing a graded, performance-based approach to quality is to affect changes that will enable the regulations to be implemented in a more efficient and effective manner. These changes will enable the NRC staff and licensees to better identify and focus on safety significant issues, structures, systems, components (SSCs) and activities, taking advantage of modern quality concepts and programs for improving performance and productivity, while maintaining safety. It will improve the implementation of quality processes and practices, improve the effectiveness and efficiency in implementing the regulations, and could ultimately enhance public health and safety.

The requirements described in the language of 10 CFR 50, Appendix B, continue to be applicable to the pertinent SSCs and activities in a manner consistent with their safety significance.

The pilot project, consisting of seven facilities from six utilities, implements and assesses methodologies for a graded, performance-based approach to quality. In the time frame of the pilot project, the pilot licensees will be implementing the concept in a select number of functional work areas on a limited number of systems. Upon completion of the pilot project, the pilot licensees will continue to implement the performance-based concept in the applicable functional work areas, for the complete set of SSCs that are categorized as non-risk significant. In addition, once the pilot project is complete, the pilot licensees may implement a performance-based approach to quality in other functional work areas. The effectiveness of the quality program for the SSCs and their related activities that are categorized as non-risk significant is determined through a performance-based approach.

This guideline should not result in the development of an alternative quality assurance program. The intent is to refine and improve current quality implementation practices, building on industry experiences while taking advantage of improved technologies and advances in analytical techniques.

The project involves the following steps:

- Grade the SSCs based on safety significance, building on the concepts and practices established in implementing the Maintenance Rule and in developing the Individual Plant Examinations (IPE). In addition, structures and systems excluded from the scope of the Maintenance Rule, but included in the general regulatory scope because of other regulations, would also be included, if they are considered to be part of the pilot project. This approach involves combining probabilistic safety assessment (PSA) and deterministic evaluations in determining the importance to safety of the SSCs within the facility. There are two main groups, risk significant SSCs and non-risk significant SSCs. Both groups are a combination of safety related and non-safety related SSCs. There is a third group, SSCs that are not within the scope of the regulations and the scope of this document.

- As appropriate, each pilot licensee determines a select set of functional work areas and the applicable set of systems for validating the concept of a performance-based approach to implementing quality during the duration of the pilot project.

- For the SSCs in the risk significant group, current quality elements continue to apply.

- For the group of SSCs that are categorized as non-risk significant, existing regulatory commitments associated with quality assurance are replaced by a commitment to implement the requirements of 10 CFR 50, Appendix B, through the guidance given in this document. The regulatory vehicle for accomplishing this task is the 10 CFR 50.54(a) process. The applicable criteria of 10 CFR 50, Appendix B, as described in the language of the rule, continue to apply to pertinent SSCs to an extent commensurate with their safety significance, but specific commitments associated with quality assurance matters, including commitments to regulatory guides, are superseded by the general guidance contained in this document. A group of technically knowledgeable individuals within the facility determines which quality elements apply, as well as the depth of implementing the specific elements. Plant performance is monitored against predetermined licensee-established performance criteria to provide reasonable assurance that the safety functions will be fulfilled.

- A corrective action program, with its associated documentation, is implemented in a manner consistent with safety significance. When there is a safety function failure, or the performance or condition does not meet the performance criteria, appropriate evaluations are initiated and, if necessary, appropriate corrective action taken.

- Regulatory assessment of the effectiveness and adequacy of the quality processes is based on meeting the performance criteria.

As the project progresses, cost benefit and project evaluations are performed to determine whether the concept of a graded, performance-based quality regime should be applied to a broader spectru of company work activities as an industrywide evolution. Adjustments will be made as the pilot project progresses based on input from the pilot participants and feedback from NRC staff reviews.

Appendix A provides a list of definitions, terms and acronyms to clarify and assist in understanding the concepts and guidance given in this document.

**DRAFT**

(9/1/94)

## 1.0 INTRODUCTION

The originators of 10 CFR Part 50, Appendix B, drafted the regulation in a manner that permitted flexibility in its implementation. Further, as stated in the NRC Regulatory Review Group Report, and confirmed in the industry-NRC management interaction the language in the regulations allows for the quality program to be implemented to an extent that is consistent with the importance to safety. The regulations also require a determination to be made on the effectiveness of the program and allow significant latitude in the manner in which this is determined.

As a result of intense commercial competition, quality practices in non-nuclear industries reflect a general improvement over the quality practices currently employed in implementing 10 CFR 50, Appendix B. The general programmatic quality concepts in these industries are essentially the same, but the measure of a satisfactory quality program has advanced from programmatic procedural compliance to one that focuses on performance and product quality..

Some utilities have already begun the transition to a performance-based quality regime. This effort builds on these embryonic activities to implement and evaluate the generic progression to performance-based quality practices in areas that do not warrant inclusion in the risk significant category.

The implementation of a graded, performance-based approach to quality enables licensees to focus on the more safety significant issues and concerns. At present, equal priority is often apportioned to issues irrespective of their safety significance, increasing the potential for not resolving a significant safety issue in a time frame commensurate with its significance. Grading SSCs and the activities associated with implementing quality through the use of improved and accepted analytical and assessment techniques enables licensees to more sharply focus on issues of safety significance. Such a restructuring program, if coupled with more effective and efficient quality implementation practices, improves the effectiveness and efficiencies of implementing the regulations that could ultimately enhance the protection of public health and safety.

## 1.1 Pilot Project Background

In June 1992, the chairman of the NRC, addressing the NUMARC[1] Board of Directors, suggested that the industry should identify areas where there are significant expenditure of resources associated with marginal safety benefit. In November 1992, the NRC announced, through the *Federal Register* (57 Fed. Reg. 55156), that they believed further studies should be undertaken in a number of areas to assess the benefits of a performance-based approach to implementing the regulations, including those associated with quality assurance. The studies would indicate what level of resources could be apportioned to more effectively manage safety significant issues and improve the effectiveness of the regulation.

In January 1993, following the industry response to the chairman's suggestion, the NRC formed the Regulatory Review Group. This group was tasked with conducting a review of power reactor regulations and related processes, programs, and practices, with special attention placed on the feasibility of implementation of performance-based concepts. The Regulatory Review Group published their report in August 1993 and concluded that an industry-phased implementation of specific measures could be a practical way to improve quality practices based on a graded, performance-based concept, as currently permitted by the regulations. Such an approach could ultimately result in an improved public health and safety environment.

Traditionally, SSCs in a facility have been categorized into a number of lists, such as non-safety related, Q-lists, and augmented quality. In some cases, because of ease of implementation or specific regulatory commitments, the full extent and requirements of Appendix B, as defined in the numerous regulatory guides, have been applied to SSCs regardless of their safety significance.

Today, with over thirty years of operating experience, with the advances in technologies and improvements in analytical techniques, such as PSAs, additional analysis can be performed to provide further insights into identifying safety significant SSCs that would be categorized as risk significant. Application of such techniques to a

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[1] The Nuclear Management and Resources Council (NUMARC) is the predecessor industry organization to the Nuclear Energy Institute that addresses generic regulatory and technical issues.

facility's list of SSCs can then become a basis for a graded, performance-based approach to implementing quality.

The foundation for a graded, performance-based approach to quality has been established by two recent industry activities: the IPE project, defined by NRC Generic Letter 88-20, and the implementation of the Maintenance Rule (10 CFR 50.65) through the industry guidance document NUMARC 93-01, *Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants*, which has been endorsed by NRC Regulatory Guide 1.160. As such, this document references NUMARC 93-01.

In December 1993, NRC management proposed an approach for implementing quality based on the concepts established in the implementation practices for the Maintenance Rule. Apart from optimizing the utilization of resources, such an approach to implementing quality would assist in focusing both NRC and industry attention and resources on the SSCs that are categorized as risk significant[2].

The Maintenance Rule has established the concept and foundation for performance-based regulation. It requires licensees to monitor the performance or condition of SSCs against licensee established performance criteria, in a manner sufficient to provide reasonable confidence that a defined set of SSCs are capable of fulfilling their intended safety functions. The criteria are established commensurate with safety and, where practical, take into account industrywide operating experience. When the performance or condition of a structure, system or component (SSC) does not meet established criteria, appropriate evaluations are initiated, and if necessary, appropriate corrective action is taken.

Within the scope of this document, the SSCs are divided into two main groups: (1) risk significant and (2) non-risk significant. There is an additional category of components that are not within the regulatory scope that are subjected to quality practices, as determined by licensee management, but are not part of the scope of this document. The two main groups are a blend of the current component classifications, safety related and non-safety related SSCs. Figure 4-1 gives a graphical representation of the grading

[2] Risk significant and non-risk significant are used in this document as general categorization terms. Appendix A of this guideline document provides additional detail.

structure. Quality elements are applied in a manner and to an extent commensurate with safety significance, as permitted by the regulations.

In view of the time frame of the pilot project, the majority of the pilot project licensees will only validate the process for a select set of functional work areas in a limited set of systems. Each licensee will determine the optimum functional work areas and systems based on cost benefit considerations. However, it should be noted that the pilot licensees may have a longer term objective of addressing a broader range of functional work areas for a larger number of systems, and may use the pilot project as the first phase of such an undertaking.

It should be noted that individual licensee business plans may place additional emphasis on specific quality elements for the SSCs that are not in the risk significant category because of their importance (economic or operational aspects). The necessity for the additional emphasis on quality in this area is outside the scope of this document and is based on business, not safety, considerations.

The transition to a full performance-based quality regime is a three part project. This document reflects the guidance for the first part, a pilot project to implement and assess a performance-based quality regime for SSCs that are not categorized as risk significant. The pilot project validates the concept of a graded, performance-based approach to quality. It forms the foundation for the long term objective, the application of graded, performance-based approach to quality for the complete spectrum of SSCs and associated activities.

The second part involves NRC review and endorsement of the approach described in this guidance, amended as necessary to reflect lessons learned from the pilot projects. NRC endorsement would provide the regulatory acknowledgment of the option for other licensees to implement a graded, performance-based approach to quality, for SSCs and their associated activities that do not warrant categorization as risk significant.

The third part, a longer term issue, is to provide the option to adopt a performance-based quality regime for the complete facility. If justified, such an approach would further reduce the complexity of assessing quality, while improving regulatory effectiveness and stability, and enhancing public safety.

## 2.0 PURPOSE

This document provides an acceptable approach for a pilot project on implementing a graded, performance based approach to meeting the quality criteria described in 10 CFR 50, Appendix B, and includes:

- The use of PSA and deterministic insights, consistent with NUMARC 93-01, to refine and restructure a Q-list based on the safety significance of plant SSCs; and

- Applying appropriate quality elements, including Appendix B criteria, to the restructured Q-list in a select set of functional work areas for a set of systems.[3]

This guidance provides an approach for implementing graded, performance-based quality measures and is intended for the voluntary use of nuclear power plant licensees. It does not preclude the use of other approaches to implement graded, performance-based quality.

## 3.0 RESTRUCTURING THE Q-LIST

This section provides guidance for a process to refine a licensee's Q-list. It begins with the SSCs that are within the general scope of the Maintenance Rule and ends with a list of SSCs that are categorized by their safety significance. The restructured Q-list will have at least two main categories of SSCs, risk significant and non-risk significant. Optional guidance is provided on refining the scope of SSCs that reside in the two main categories. Optional guidance is also provided on grading the Q-list into more than two categories.

### 3.1 Defining the Scope of SSCs

The SSCs defined by the Maintenance Rule, 10 CFR 50.65, and the SSCs encompassed by other regulations and licensing commitments, provides the scope and the starting point for determining which plant SSCs will reside on the restructured Q-list. It includes safety related and non-safety related SSCs as required by the Maintenance Rule.

[3] See Appendix A, Definitions, Terms and Acronyms for an explanation of the terms, Q-list and restructured Q-list

*Extract from 10 CFR 50.65:*

1. *Safety-related SSCs that are relied upon to remain functional during and following design basis events to ensure the integrity of the reactor coolant pressure boundary, the capability to shut down the reactor and maintain it in a safe shutdown condition, and the capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposure comparable to the 10 CFR Part 100 guidelines.*

2. *Non-safety related SSCs:*

   - *That are relied upon to mitigate accidents or transients or are used in plant emergency operating procedures (EOPs);*

   - *Whose failure could prevent safety-related SSCs from fulfilling their safety-related function: or*

   - *Whose failure could cause a reactor scram or actuation of a safety-related system.*

### 3.2 Selection of Plant SSCs

NUMARC 93-01, Section 8.2.1, *Selection of Plant SSCs*, provides guidance on determining the SSCs that are within the scope of the Maintenance Rule. These same SSCs will reside on a restructured Q-list.

It is expected that most of the SSCs that comprise current licensee Q-lists will be determined to be within the scope of the Maintenance Rule. If some of these SSCs are not within the Maintenance Rule scope, they should be evaluated as a separate category of SSCs to determine whether there is a basis for including them within the scope.

### 3.3 Establishing the Safety Significance of SSCs

NUMARC 93-01, Section 9.3.1, *Establishing Risk Criteria*, provides guidance on establishing the safety significance of SSCs. The guidance involves a blend of both probabilistic and deterministic methods to appropriately identify the safety significance of

SSCs. This approach is intended to capitalize on the insights gained from plant-specific PSAs as well as the operating experience and expertise of plant personnel. For example, when probabilistic methods are used to determine the safety significance of SSCs, a panel of individuals experienced with the plant PSA, operations and maintenance should be employed to supplement the probabilistic results with their own expertise. The expert panel should compensate for limitations associated with applying PSA methods to establish the safety significance of SSCs.

Licensee management would determine the specific composition and experience for each of the positions on the expert panel. Additional information and guidance is contained in NUREG/CR 5424, *Eliciting and Analyzing Expert Judgment;* NUREG/CR 4962, *Methods for Elicitation and Use of Expert Opinion in Risk Assessment:* NUREG/CR 5695, *A Process for Risk Focused Maintenance;* and in NUMARC 93-02, *A Report on the Verification and Validation of NUMARC 93-01*. A licensee may wish to use the expert panel delineated in NUMARC 93-01 for grading SSCs for implementing the Maintenance Rule to fulfill this function.

There is one important difference to note here regarding the guidance provided in NUMARC 93-01 to establish the safety significance of SSCs. In using PSA importance measures[4] to gain insights into the safety significance of SSCs, pertinent failure modes, whether they are maintenance related or not, should be considered. The guidance in NUMARC 93-01 appropriately excludes consideration of failure modes that are not maintenance preventable. Given that quality measures are applied to a spectrum of functional areas other than maintenance, it is necessary to consider the pertinent failure modes in those areas and related activities that would impact the safety function.

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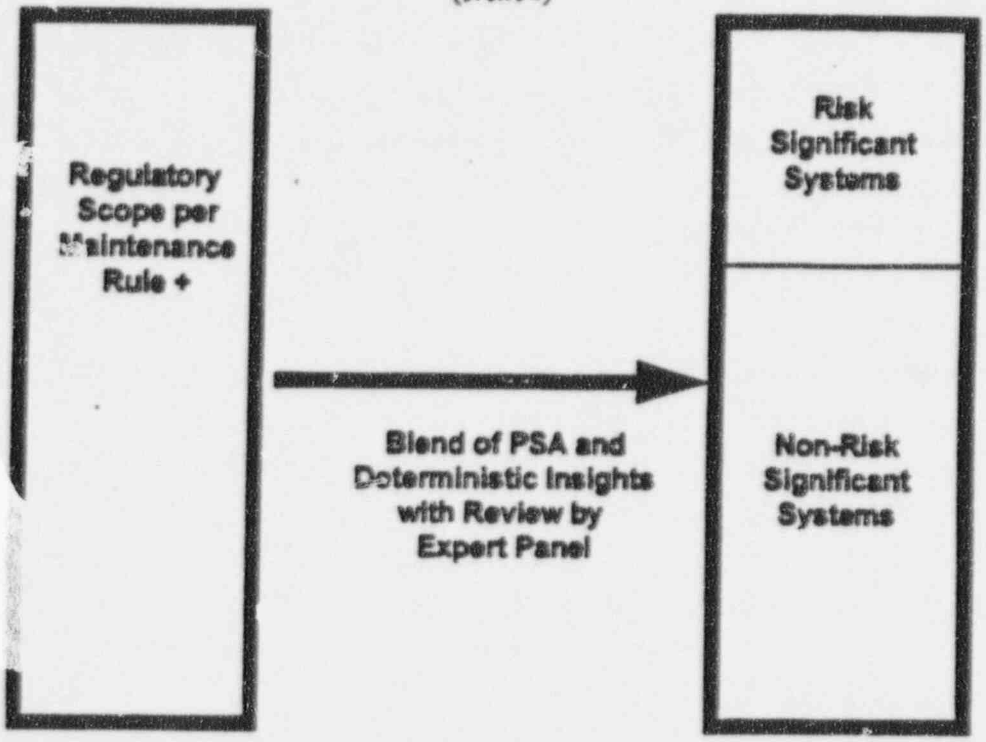
[4] Importance measures are those defined in NUMARC 93-01, risk reduction worth, risk achievement worth, and core damage frequency contribution.




+Reg. Scope = Maint. Rule Scope + Other SSCs from Other Regulations and Licensing Commitments

FIGURE 3-1

The methods described above to identify the safety significance of SSCs were employed by several utilities during the development of NUMARC 93-01. The results of those efforts were published in NUMARC 93-02, *A Report on the Verification and Validation of NUMARC 93-01, Draft Revision 2A, Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants.* The results indicated that the licensees generally identified the safety significance of SSCs within the scope of the Maintenance Rule at the system level. This step serves as the first cut at categorizing the systems based on the safety significance of the systems. Figure 3-1 illustrates the concepts discussed thus far in Section 3.2 and this section.

## 3.4 Initial Categorization at the Component Level

The next step involves a component level categorization. This section discusses the initial breakdown of the systems into two main categories on the restructured Q-list, risk significant components and non-risk significant components.

Within the systems identified as risk significant, all components in these systems should be initially categorized as risk significant components. If components are in systems that are not included within the risk significant category, these components would not be initially placed in the risk significant category. Figure 3-2 illustrates this concept. One could use this initial categorization as the foundation for implementing graded quality practices. However, it should be noted that there may be several components within a system that are categorized as risk significant that are not necessary for the system to support its safety function. In such cases, these components should be categorized as non-risk significant. The next section addresses this topic.

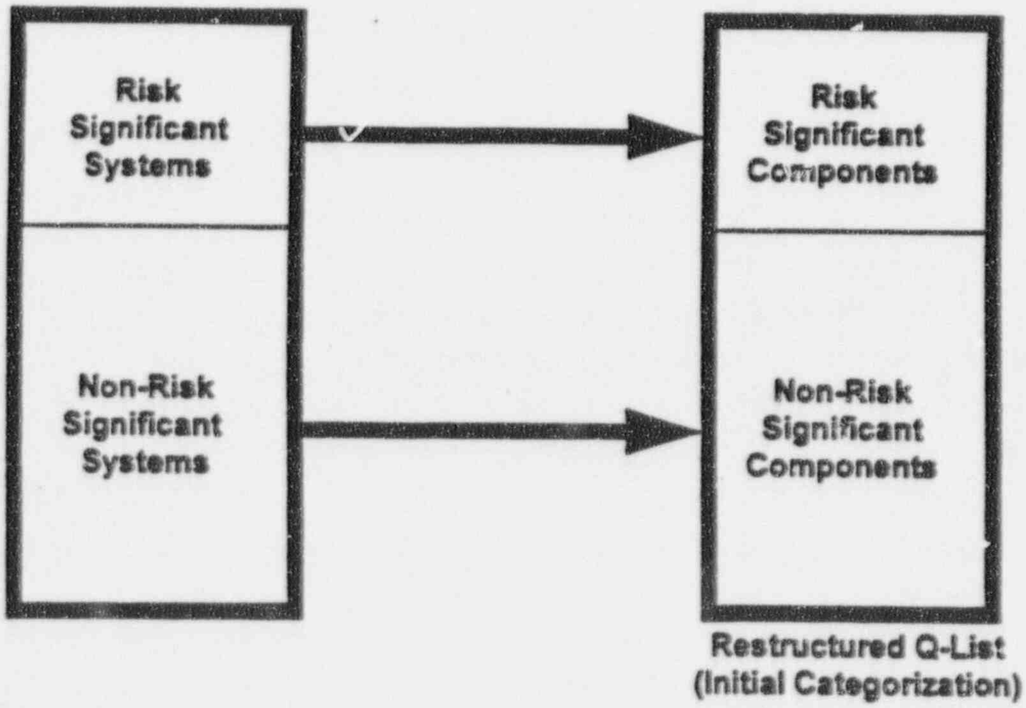



FIGURE 3-2

### 3.5 Reviewing the Scope of Components Categorized as Risk Significant

The section provides optional guidance that further refines the scope of the components that reside in the two categories. Up to this point, components have been categorized based upon the safety significance of the system of which they are a part. The review at this stage is intended to provide greater focus on truly safety significant components by determining the component's contribution to the system's safety significance.

The primary factor to consider in this review is the safety function of the system. If the component is necessary for the performance of the system safety function, it should remain in the risk significant category. If the component is not necessary for the performance of the system safety function, it should be categorized as non-risk significant.

There are a number of acceptable methods to determine a component's contribution to system safety function. In general, these methods consist of deterministic, or a blend of probabilistic and deterministic approaches to evaluate component safety significance.

A deterministic approach may be used to identify system safety function through a review of documents such as system descriptions or design basis documents. The role of an individual component in achieving or supporting achievement of that system's safety function could then be determined by knowledgeable personnel in assessing the ability to achieve the system safety function in light of component failure(s).

A probabilistic approach may be used to determine component safety significance through application of the plant IPE. Since the system has already been categorized as risk significant, it would be necessary to perform additional assessments of the IPE model to identify those components of the system which are modeled in the IPE. Because components in the IPE models are often "super-components" which represent two or more components of a system, it may be necessary to identify the multiple components which constitute a modeled IPE component. Having done this, those components which do not affect the system's safety function should not be included in the risk-significant category.

The licensee can then employ a final step to determine component safety significance. So far, the only criterion applied to components is to determine if the component has been modeled in the IPE. Those components which have been modeled, and therefore remain in the risk significant category, can now be assessed to determine their contribution to plant risk. Components whose failure is determined to have no risk significance or negligible risk significance should not be included in the risk significant category. The result is subject to a review by the expert panel.

By employing deterministic, or a blend of deterministic and probabilistic approaches to identify and retain those components that are relied upon to carry out the system safety function, the scope of components in the risk significant category would become more focused. Figure 3-3 illustrates this concept.

It is recognized that after completing this review, the two groups discussed in this section may contain a mixture of safety related and non-safety related components.

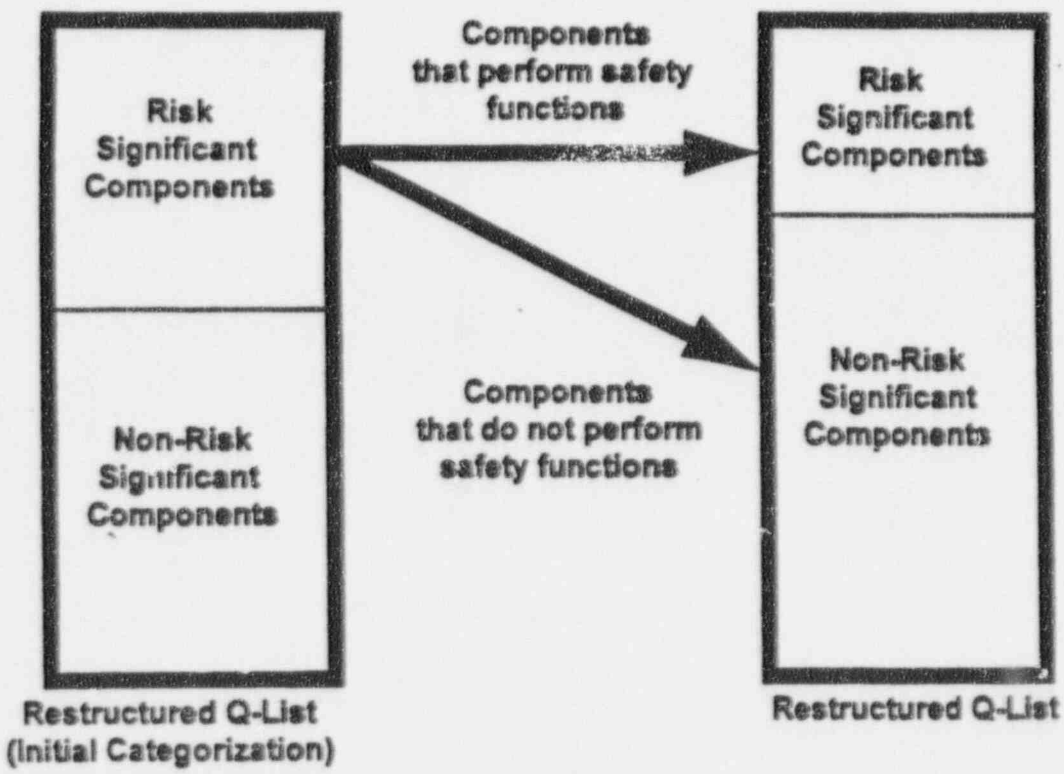



FIGURE 3-3

### 3.6 Additional Categorization of Components Categorized as Risk Significant

This section provides optional guidance on further subdividing the risk significant component category of the restructured Q-list into additional categories. It should be emphasized that a decision to proceed with implementation of this optional guidance would be based on cost benefit determination, and would vary from licensee to licensee. This decision should be based on whether the savings from implementing a graded, performance-based approach over the life of the facility outweigh the cost of further categorizing the SSCs.

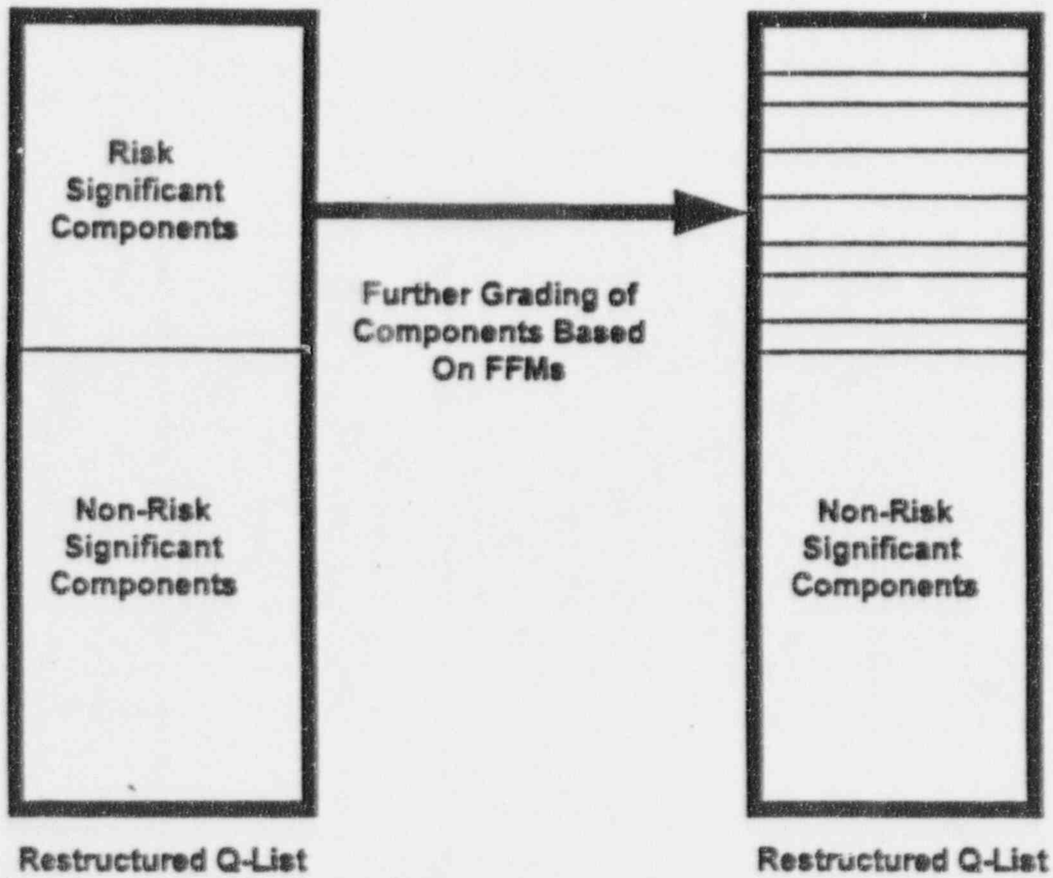


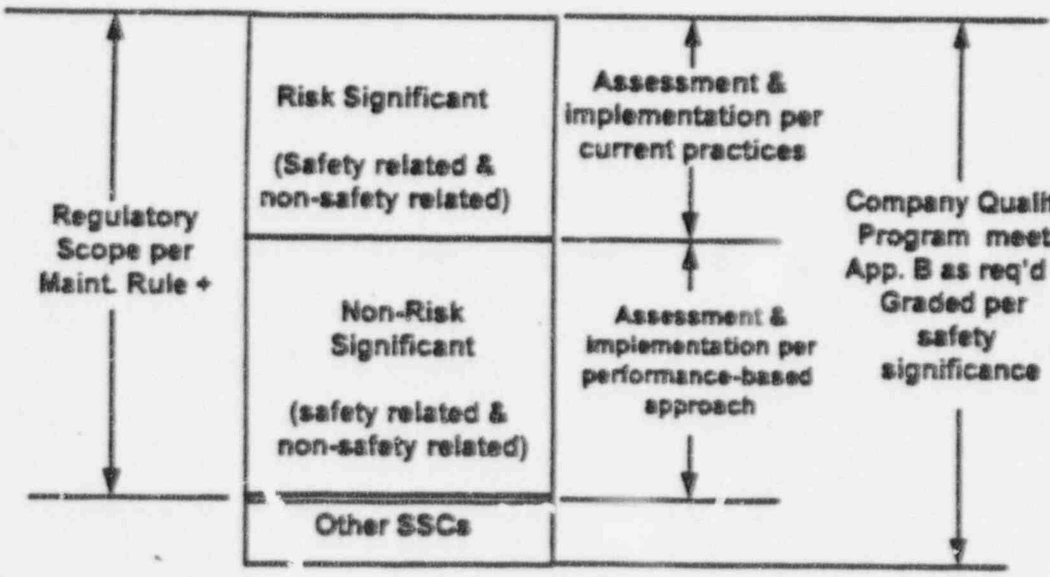

FIGURE 3-4

The additional categorization is primarily based on identifying the functional failure modes (FFM) of the components residing in the risk significant category. This approach would require additional use of the plant-specific PSA to identify the risk significant failure mode(s) of the component. Components would then be categorized based on the significance of their functional failure modes. Appropriate quality elements would then be applied to address the particular failure mode of that category of components. Figure 3-4 illustrates this concept.

## 4.0 APPLYING GRADED, PERFORMANCE-BASED QUALITY ELEMENTS

Figure 4-1 gives a graphical representation of the general process being described in this document.

GRADED APPROACH TO IMPLEMENTING QUALITY




+Reg. Scope = Maint. Rule Scope + Other SSCs from Other Regs.

FIGURE 4 - 1

This project will result in a number of adjustments to current quality elements that are applied to SSCs and their associated activities. The main area of adjustment is in the category of SSCs that are currently safety related and are categorized as non-risk significant. In addition, for SSCs that are currently non-safety related, yet are categorized as risk significant, a review of their current quality elements, their past performance histories, and safety significance may indicate additional adjustments to the current quality controls for these SSCs. The types of quality adjustments are described later in this guideline.

## 4.1 Administrative Guidance

Each company will make its own determination on the functional work areas for implementing a graded, performance-based approach to quality, as well as the degree and depth of implementing those processes.

The list of functional work areas described below is a provisional list, and there may be other functional work areas that may benefit from adopting a graded approach. The list is not in order of priority, or ranked in order of anticipated cost benefits.

- Procurement
- Warehouse receipt inspection
- QC inspections
- Maintenance
- Design process, including verification and change process
- Configuration control
- Records and documentation
- Material control and traceability
- Audits/assessments
- Independent verification
- Procedure development
- Work control processes (bolting, cable pulling....)
- Surveillances, including ISI/IST within code allowables
- Oversight process
- Testing

### 4.1.1 Quality Commitments

Changes in the quality program associated with 10 CFR 50, Appendix B, commitments are implemented through the 10 CFR 50.54(a) process.

Changes to quality program commitments for safety related SSCs that are categorized as risk significant are considered on a case-by-case basis, and are implemented through the normal commitment change process.

For safety related SSCs that are categorized as non-risk significant, the quality commitments are changed through the existing Section 50.54(a) process, by substituting the following commitments for the current set of quality commitments:

a) Implement the requirements of 10 CFR 50, Appendix B, through a graded, performance-based approach as described in this document;

b) Monitor and meet a predetermined set of licensee established performance criteria (predominantly plant, system and/or train level criteria); and

c) Implement corrective action and assessment processes to resolve deficiencies and deviations, and monitor performance. The corrective action element is implemented in a manner commensurate with the safety significance of the deficiency or deviation, and meets the requirements of Criterion XVI of 10 CFR 50, Appendix B.

The purpose of quality assurance commitments, described in the NRC approved program description, is to assist in assuring that 10 CFR 50, Appendix B, is implemented properly. The purpose of 10 CFR 50, Appendix B, is to provide adequate assurance that pertinent SSCs are capable of performing their safety functions. Furthermore, 10 CFR 50, Appendix B, permits quality controls to be applied in a manner consistent with the importance to safety. The Maintenance Rule (10 CFR 50.65) requires licensees to establish performance criteria, sufficient to provide reasonable assurance that applicable SSCs are capable of fulfilling their intended function. Therefore, the quality commitments described above are equivalent to the current set of commitments and can replace the existing quality commitments described in the NRC approved quality program description for those SSCs categorized as non-risk significant.

The changes described above substitute an alternative and equivalent set of commitments to achieve the same objective as existing regulatory commitments. The intent of the new commitments is the same as the current commitments, to provide adequate assurance that the safety functions will be fulfilled. As such, there is no reduction in commitment. The objective of providing reasonable assurance that the pertinent set of SSCs are capable of performing their intended safety function is accomplished through a different methodology (e.g., a graded, performance-based approach). Such a change is permitted under the requirements of 10 CFR 50.54(a).

The process described in this document establishes performance criteria that provide reasonable assurance that the pertinent SSCs will perform their safety function. The effectiveness and adequacy of the quality programs and processes is demonstrated through meeting these performance criteria.

As necessary, the pilot licensee submits a change to the quality program commitments under Section 50.54(a). The change is submitted to docket the change in commitments that have been generically approved, through the NRC endorsement of this document. This submittal should include, for informational purposes, the functional work areas that are being reviewed as part of the pilot project.

### 4.1.2 Quality Element Assessment

The specific details for adjusting current quality elements, including the extent of refinements to the implementation procedures, practices and instructions, are determined by each licensee.

A quality review group, consisting of technically knowledgeable, multi-disciplined licensee personnel, determines the applicability and depth of implementation of the appropriate quality elements that are being adjusted. Licensee management determines the experience requirements for each of the positions. The guidance in this document related to the quality review group can be supplemented by the information in NUREG/CR 5424, *Eliciting and Analyzing Expert Judgment* and NUREG/CR 5695, *A Process for Risk Focused Maintenance*. In addition, a licensee may wish to make one department responsible for this multi-disciplined review function. The group should also seek advice and input from personnel involved in implementing the Maintenance Rule, if not already included in the group's complement.

In some instances, the group could be the same as the expert panel described in Section 3, but the task associated with the assessment, determination and adjustment of quality elements might be undertaken in a different time frame to that of categorizing the facility SSCs.

The quality review group should take into account a number of factors when developing quality elements or assessing the need to adjust the programs, controls, procedures or instructions associated with SSCs, or the functional activity, or the quality elements under review. Some of the factors are described below (not in order of priority) and include:

- Existing non-safety related and safety related work practices and instructions
- Past plant performance history (equipment performance)
- Design specifications and conditions, including environmental (EQ and seismic)
- Training, professional development and certification programs
- Experience of the work force and use of outside contractors/consultants
- Health and safety implications
- Impact on plant performance
- Support of safety functions
- Testing and evaluation options

The quality review group determines which quality elements apply that provide adequate assurance that the performance criteria will be met and the safety function fulfilled.

When the quality review group determines that quality elements can be amended, it documents the basis for the new quality elements. It is not anticipated that additional documentation would be required, beyond documenting the basis for the new quality elements and practices.

## 4.2 SSCs Categorized as Risk Significant

### 4.2.1 Risk Significant, Safety Related SSCs

For those SSCs that have been classified as risk significant and are on the original Q-list (i.e., safety-related SSCs), the current 10 CFR 50, Appendix B, quality program is applied in a manner commensurate with safety significance. The regulatory commitments described or referenced in the FSAR still apply to these SSCs.

The quality elements, including the work procedures and instructions for the SSCs in this category, should not be impacted by this project, unless the licensee wishes to take the opportunity to further refine and improve the quality regime for implementing Appendix B for this group of SSCs and related activities. A licensee may wish to take the opportunity to grade or assess grading of quality elements for this group of SSCs based on safety significance of the SSC or the activity. The methodology for grading the quality elements for the SSCs and their associated activities categorized as risk significant may vary from licensee to licensee. In general, the grading is based on the concepts described in Sections 3.6 and 4.1.

It should be noted that changes to regulatory commitments that are applicable to SSCs or quality activities for this group are considered on a case by case basis through the commitment change process.

### 4.2.2 Risk Significant, Non-Safety Related SSCs

For those non-safety related SSCs that have been categorized as risk significant, an assessment is performed to determine whether the correct quality elements are being applied to those SSCs in view of their safety significance.

Past performance operating profiles, input from plant personnel, and current quality practices provide input into the quality review group's determination on the necessity to adjust the applicable quality elements (including implementing pertinent segments of 10 CFR 50, Appendix B, criteria, if appropriate). These quality elements may include graded elements of 10 CFR 50, Appendix B, criteria, as described in the language of the rule, but not necessarily as described in the associated regulatory guides. These adjustments are made through the quality review group to assure that the level of detail of implementation is commensurate with the safety significance of the SSC or activity.

If appropriate, graded Appendix B requirements for this group of SSCs (risk significant, non-safety related) and their associated activities are applied in a **forward-looking manner**. For example, spare parts and installed equipment are not qualified or dedicated as safety related SSCs. The applicable quality requirements for future modifications or replacements would be implemented to support the SSC safety function as determined by the quality review group.

### 4.3 SSCs Categorized as Non-Risk Significant (Safety Related and Non-Safety Related SSCs)

For the group of SSCs that are categorized as non-risk significant, the assessment of the effectiveness of the quality program is made through performance-based concepts, by monitoring performance and condition against a set of predetermined performance criteria. Performance criteria are established in a manner similar to that developed for the implementation of the Maintenance Rule. Section 4.4 provides additional information.

The quality review group determines the quality elements that will be employed for this group of SSCs to assure that the performance criteria are met. Section 4.1 and Appendix A provides general guidance.

Each licensee determines the applicability of the quality elements for this category. In some cases, existing quality elements are consistent and applicable. In other cases, a licensee may determine that refinements and adjustments to the quality elements and processes are beneficial.

The basis for any new quality elements is documented. In many instances, a licensee's existing documentation is sufficient to provide the basis for a new set of quality elements and for performing cause determinations should a deficiency occur.

For this group of SSCs, the current regulatory quality assurance commitments and ancillary programs are replaced with the general guidance described in Section 4.1 of this document.

For the safety related SSCs that are categorized as non-risk significant, Criterion XVI of 10 CFR 50, Appendix B, Corrective Action, is applied in a manner commensurate with safety significance. It should be noted that other criteria of 10 CFR 50, Appendix B, might be applicable dependent on the determinations of the quality review group based on the safety significance of the SSC or activity under consideration. The depth of implementation and specific details are determined by the quality review group and are not necessarily those described in current commitments to regulatory guides or industry standards.

The effectiveness and adequacy of the quality programs is demonstrated through meeting the performance criteria.

## 4.4 Performance Criteria

### 4.4.1 SSCs Categorized as Risk Significant

For the pilot project, the assessment of quality program effectiveness for the risk significant group is defined by current practices.

### 4.4.2. SSCs Categorized as Non-Risk Significant

Performance criteria are derived in a manner similar to that described in NUMARC 93-01, Section 9, *Establishing Risk and Performance Criteria/Goal Setting and Monitoring*. It should be noted that in the case of performance-based quality, the performance criteria and evaluations take into account the pertinent failures that could impact the safety function, not just those associated with maintenance preventable functional failures.

In conjunction with the expert panel for categorizing components, the quality review group determines the performance criteria. These determinations are based on design basis information, past plant performance data, and if applicable, PSA and other reliability studies. The applicable quality processes for these SSCs provide reasonable assurance that the performance criteria, and hence the safety function, will be met.

In general, for a graded, performance-based approach to quality, the performance criteria for those SSCs that are categorized as non-risk significant will be plant level. The performance criteria should be similar and possibly identical to the performance criteria developed for implementing the Maintenance Rule, (e.g., unplanned safety system actuations, unplanned scrams per 7,000 hours critical, unplanned capability loss factor or plant level safety functional failures).

For safety related SSCs that are categorized as non-risk significant and are standby systems, the effectiveness of the applicable quality elements would be based on meeting system level performance criteria. The performance criteria for these systems are similar to that prescribed in the guidance for implementing the Maintenance Rule, NUMARC 93-01, (i.e., reliability, availability, condition monitoring, or system safety function failures).

Performance criteria could be a single value or range of values, such as the industrywide plant performance indicators. Plant performance indicators are already established and could be the basis of plant level performance criteria. Performance criteria should reflect specific SSC performance histories.

The establishment of performance criteria for standby systems should include a review of surveillance tests, actual demands and safety functional failures. Consideration should be given to using industrywide operating experience, if applicable and appropriate. In addition, the licensee's specific PSA and other reliability studies can be used as input into the determination for establishing performance criteria.

For inherently reliable SSCs that are part of a system that are categorized as non-risk significant, plant level performance criteria would not be the appropriate measure of the applicable quality program effectiveness. For these inherently reliable SSCs, such as raceways, pressure vessels, jet shields, etc., that are part of non-risk significant systems, specific component reliability may not be a practical measure of performance. In these instances, performance criteria would be associated with the ability of the pertinent system to perform its safety function i.e., system level performance criteria.

Guidance on the action to be taken in the event of failing to meet the performance criteria is given in Section 4.6.2 and in NUMARC 93-01, Sections 9.3 and 9.4.

### 4.4.3 Evaluation of Performance Criteria

Each licensee should develop a process for evaluating plant data and for determining the specific performance criteria associated with this program. It is anticipated that such a process could be the similar to that used for implementing the Maintenance Rule. However, the implementation of a graded, performance-based quality regime is broader than the implementation of the Maintenance Rule. It involves evaluations of plant and system performance in all applicable areas including those associated with maintenance activities. These evaluations are documented.

The quality review group in conjunction with the expert panel for categorizing the components into risk significant and non-risk significant categories, conducts a review of the performance criteria and the process to be used in evaluating the performance criteria.

The periodicity of these performance criteria evaluations should be consistent with that established for implementing the Maintenance Rule (see NUMARC 93-01, Sections 9.3, 9.4 and 10.2). In addition, the licensee may determine, at its discretion, to undertake additional evaluations of whether the performance criteria are being sustained based on specific events, recommendations in oversight reports, or general trending reports.

### 4.4.4 Changing Performance Criteria

The licensee may make changes to the performance criteria based on new information, deviations, defects or as a result of quality assessments. The basis for changing performance criteria should be documented. NUMARC 93-01, Sections 9.3 and 9.4, provides additional detail.

## 4.5 Quality Assessments

Assessments, including self assessments, should be performed to provide adequate assurance that the quality program is effective. The type, frequency and degree of specificity of these assessments should be determined by the safety significance and performance history of the SSCs or work activity. The aim is to assess and assist line departments and line quality functions.

Assessments may be in the form of audits, examinations, inspections, monitoring, surveillances, tests or verifications. Safety significance and performance history should determine the degree of management, technical and independent oversight. This oversight can take many forms, from line supervisory teams, to reviews that interact directly with senior executives responsible for facility operations.

Licensee management is responsible for the quality assessment program. Personnel performing assessments should be qualified through training, work experience or certification.

### 4.6 Corrective Action Program

A corrective action program is a central element of a quality program. It is through this process that appropriate licensee actions are initiated, and deficiencies and deviations, such as safety functional failures and performance criteria deviations are evaluated and resolved.

Corrective action programs may take many forms, from informal (e.g., rough defect logs, front line technician/supervisor/foreman interactions) to formal (e.g., divisional and departmental interactions, with detailed engineering evaluations). Each has its benefits and disadvantages. The important considerations are that the programs are structured in a manner that provide reasonable assurance that deviations and deficiencies are resolved.

The licensee will determine the extent and nature of corrective actions and will base its assessments on the safety significance of the SSC. The objective is to take the appropriate action to provide reasonable assurance that the performance criteria would be met, and the safety function fulfilled.

Corrective actions that have safety significance are documented. For significant safety deficiencies and deviations from the performance criteria, satisfactory accomplishment of the corrective action shall be confirmed.

#### 4.6.1. SSCs Categorized as Risk Significant

The corrective action program for the deficiencies and deviations in the risk significant, safety related group of SSCs is the same as for current practices.

Deficiencies and deviations associated with the risk significant, non-safety-related SSCs and their related activities, shall be documented and appropriate corrective action taken. If deficiencies and deviations result in a safety system functional failure, the cause determination is reviewed by the appropriate level of management.

#### 4.6.2 SSCs Categorized as Non-Risk Significant

The same type of practices apply for resolving deficiencies in this group as for the risk significant group. The degree of documentation is consistent with appropriate licensee administrative procedures.

Failure to meet a performance criterion indicates a possible condition adverse to quality. It requires prompt action, a cause determination, corrective action, and appropriate managerial involvement commensurate with the safety significance of the performance criterion deviation. If applicable and appropriate, the cause determination includes an assessment on whether the performance criterion should be modified, or whether the quality elements, e.g., procurement or design control, need to be adjusted to provide adequate assurance that the performance criteria will be met and the safety function fulfilled.

Deviations from the performance criteria, the cause determination, the corrective action, and the basis for any changes to the performance criteria or quality elements are documented.

Deficiencies and deviations associated with SSCs and their related activities in the non-risk significant category that result in a safety system functional failure are documented. Appropriate corrective action is initiated and the appropriate level of management confirms resolution. If a safety system functional failure is caused by a repetitive deficiency or deviation, and even if the performance criteria have been met, management shall review the cause determination, including the assessment of the current quality elements and the pertinent performance criteria.

**DRAFT**
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## APPENDIX A

## DEFINITIONS, TERMS AND ACRONYMS

The following definitions and terms are provided to assist in understanding the terms used in the document.

**Assessments:** A collective term covering reviews, monitoring, tests, surveillances, inspections, audits or examinations.

**Deviation:** A departure from a specified requirement or performance criterion.

**Industrywide operating experience**: Information included in NRC, industry, and vendor equipment information that are applicable and available to the nuclear industry with the intent of minimizing adverse plant conditions or situations through shared experiences.

**Performance monitoring:** Continuous or periodic tests, inspections, measurement or trending of the performance or physical characteristics of a SSC to indicate current or future performance and potential failures.

**Performance criteria:** The term is used in the same manner as used in NUMARC 93-01. Unless otherwise stated, the term performance criteria, or predetermined performance criteria relate to plant level, or in specific instances, system and/or train level performance criteria. As used in this document, the term does **not** relate to specific component quality criteria, or criteria associated with manufacturing or procedural activities, as used in some other publications.

**Performance-based approach**: An approach that focuses on the end results, not the process, that directly contribute to safe and reliable plant operation. Meeting predetermined goals, limits or performance criteria based upon the design basis safety function, operating experience and pertinent reliability studies. A licensee is allowed the flexibility to determine how to achieve the results and adjust quality elements.

**Q-List:** The list of SSCs required by Criterion II of 10 CFR 50, Appendix B.

**Restructured Q-List:** List of SSCs that are part of the overall regulatory scope as defined by the scope of the Maintenance Rule and other NRC regulations.

**QA Topical Report:** A general report and description of the licensee's Appendix B quality program and the accompanying standards. It represents the quality assurance licensing commitments associated with implementing 10 CFR 50, Appendix B.

**Quality elements:** The quality attributes, controls, criteria, processes, or practices, associated with the safety function of a structure, system, component or activity.

**A repetitive safety functional failure:** A safety functional failure that is determined to have been caused by an identical deficiency within the last two years.

**Safety functional failure:** A failure of a component that results in a determination that the safety function of a structure or system cannot be performed.

**Risk significant SSCs:** This term equates to the term risk significant SSCs used in NUMARC 93-01. It relates to those SSCs that are significant contributors to safety and risk as determined through a blend of PSA and deterministic assessments.

**Non-Risk significant SSCs:** The term non-risk significant is a categorization term and implies low safety or risk significance. It equates to the term used in NUMARC 93-01.

**Standby system or train:** A system or train that is not normally operating and only performs its intended safety function when initiated by either an automatic or manual demand signal.

## List of Acronyms

| | |
|---|---|
| CFR | Code of Federal Regulations |
| EOPs | Emergency Operating Procedures |
| FFM | Functional Failure Modes |
| FSAR | Final Safety Analysis Report |
| IPE | Individual Plant Examination |
| ISI | Inservice Inspection |
| IST | Inservice Testing |
| NRC | Nuclear Regulatory Commission |
| NEI | Nuclear Energy Institute |
| NUMARC | Nuclear Management and Resources Council |
| PSA | Probabilistic Safety Assessment |

# ATTACHMENT 2