Issue: Seismic Category ("SC") RW-IIa Radwaste ("RW") Building. Level of design information necessary for certification.

Safety Significance: The purpose of analyzing SC II buildings is for assuring that the buildings will not fail and adversely impact a SC I structure, system, or component (i.e., seismic interaction concerns). Completing the seismic and natural phenomena analysis is not necessary for assuring the safety of the standard design; rather, NRC guidance suggests including information in the DCD regarding location of non-seismic buildings, information on the design, description of analyses methods, and seismic design criteria that will be applied for these buildings (*see* below). Also, for the RW Building, RG 1.143 (Rev. 2, 2001) indicates that the "safety classifications" for the RW management facilities "were developed primarily for natural phenomena and man-induced hazard design" and that "the impact of these [radwaste management] systems on safety is limited." Finally, the SRP in effect at the time of ESBWR DC application submittal includes limited information regarding SC II buildings, and essentially no information in Chapter 3.8 that addresses FSAR content for non-SC I buildings, such as the RW building.

NRC Staff Position: In RAIs 3.8-79S03 and 3.8-80S03, the NRC requests that GEH include in the DCD additional description of seismic analyses and design of the RW building and provide the results of the seismic analysis.

Applicable NRC Guidance: NRC guidance in NUREG-0800, Section 3.7.2 (Rev. 2, 1989), suggests that the staff review the "design criteria to account for the seismic motion of non-Category I structures" and "procedures that are used to protect Category I structures from the structural failure of non-Category I structures, due to seismic effects." Regarding level of detail to include in the DCD, RG 1.70, Section 3.7.2.8 (Rev. 3, 1978) states:

> 3.7.2.8 Interaction of Non-Category I Structures with Seismic Category I Structures. Provide the design criteria used to account for the seismic motion of non-Category I structures or portions thereof in the seismic design of Seismic Category I structures or portions thereof. In addition, describe the design criteria that will be applied to ensure protection of Seismic Category I structures from the structural failure of non-Category I structures due to seismic effects.

Regarding the descriptive information for structures, RG 1.70, Section 1.2, suggests that the "general arrangement of major structures and equipment should be indicated by the use of plan and elevation drawings in sufficient number and detail to provide a reasonable understanding of the general layout of the plant." Other sections of RG 1.70 generally address safety-related structures. Specific to the RW building, RG 1.143 provides NRC guidance regarding the design, construction, installation, and testing of RW management facilities. Under the criteria in RG 1.143, the RW building is a "safety class" RW-IIa. In RG 1.143, earthquakes and other natural phenomena design criteria are listed as applicable to radwaste management safety class RW-IIa..

Applicant Position: As noted in the RAIs, GEH has provided certain design description and has indicated in the DCD that SC I methods and criteria will be the basis for seismically analyzing the RW buildings as part of completing the detailed design for the ESBWR. GEH clearly considers the RW building as part of the standard ESBWR design. GEH proposes, in accordance with NRC guidance in effect at the time of ESBWR DCA submittal and to address RAIs 3.8-79S03 and 3.8-80S03, to include additional description of the general design, seismic and natural phenomena analyses methods, and acceptance criteria to be used to ensure that the as-built RW building will conform to the criteria set forth in NRC guidance. GEH proposes to analyze the RW building to the full SC I requirements, which is conservative, and address the other phenomena at the time that the SC I analyses is performed as part of detailed design completion. In addition, GEH proposes to include two ITAAC in Tier 1 that will (1) require a report that concludes the seismic and natural phenomena analyses are consistent with the DCD and (2) require verification that the as-built RW building conforms to the results of the analyses (similar to ITAAC on SC I buildings).

Proposed Action: GEH has completed seismic analyses for all SC I buildings and included the results in the DCD, as per NRC guidance. For the RW-IIa building, the DCD will include information on the design and analysis criteria that will be used when the seismic and natural phenomena analyses are performed, consistent with NRC guidance. Although GEH has not completed the analyses, GEH proposes to add information in the DCD. GEH has discussed draft RAI responses with the NRC staff and is preparing the final responses that will show the additional information to be included in the DCD, consistent with NRC guidance. GEH requests NRC feedback on the approach for the RW building.

Pages from ESBWR DCD.

Table 1.9-20

NRC Standard Review Plans and Branch Technical Positions Applicability to ESBWR

SRP No.	SRP Title or BTP	Appl. Rev.	Issued Date	ESBWR Appli-cable?	Comments
	Appendix C to SPLB 3-1	3	Draft 04/1996	Yes	
3.6.2	Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping	2	Draft 04/1996	Yes	
	BTP EMEB-3-1	2	Draft 04/1996	Yes	
3.6.3	Leak-Before-Break Evaluation Procedures	0	03/1987	—	Not credited.
3.7.1	Seismic Design Parameters	2	08/1989	Yes	
	Appendix A	0	08/1989	Yes	
3.7.2	Seismic System Analysis	2	08/1989	Yes	
	Appendix A	0	08/1989	Yes	
3.7.3	Seismic Subsystem Analysis	2	08/1989	Yes	
3.7.4	Seismic Instrumentation	1	07/1981	Yes	
3.8.1	Concrete Containment	1	07/1981	Yes	
	Appendix	0	07/1981	Yes	
3.8.2	Steel Containment	1	07/1981	Yes	applies only to Drywell Head
3.8.3	Concrete and Steel Internal Structures of Steel or Concrete Containments	1	07/1981	Yes	
3.8.4	Other Seismic Category I Structures	1	07/1981	Yes	
	Appendix A	0	07/1981	Yes	
	Appendix B	0	07/1981	Yes	
	Appendix C	0	07/1981	Yes	
	Appendix D	0	07/1981	Yes	
3.8.5	Foundations	1	07/1981	Yes	
3.9.1	Special Topics for Mechanical Components	3	Draft 04/1996	Yes	

Table 1.9-21

NRC Regulatory Guides Applicability to ESBWR

RG No.	Regulatory Guide Title	Appl. Rev.	Issued Date	ESBWR Appli-cable?	Comments
1.137	Fuel-Oil Systems for Standby Diesel Generators	1	10/1979	No	No safety-related Diesel Generators for ESBWR. URD intent – see Table 1.9-21a
1.138	Laboratory Investigations of Soils and Rocks for Engineering Analysis and Design of Nuclear Power Plants	2	12/2003	—	COL
1.139	Guidance for Residual Heat Removal	0	05/1978	Yes	URD optimization – see Table 1.9-21a
1.140	Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Normal Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants	2	06/2001	Yes	
1.141	Containment Isolation Provisions for Fluid Systems	0	04/1978	Yes	
1.142	Safety-Related Concrete Structures for Nuclear Power Plants (Other than Reactor Vessels and Containments)	2	11/2001	Yes	
1.143	Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants	2	11/2001	Yes	
1.144	Auditing of Quality Assurance Programs for Nuclear Power Plants		Super-ceded		See Table 1.9-21b. Withdrawn 07/31/1991

NRC Guidance in effect at time of ESBWR DCD submittal.

REGULATORY GUIDE 1.70
REVISION 3

STANDARD FORMAT AND CONTENT OF SAFETY ANALYSIS REPORTS FOR NUCLEAR POWER PLANTS

LWR EDITION

NOVEMBER 1978

OFFICE OF STANDARDS DEVELOPMENT
U. S. NUCLEAR REGULATORY COMMISSION

1. INTRODUCTION AND GENERAL DESCRIPTION OF PLANT

The first chapter of the SAR should present an introduction to the report and a general description of the plant. This chapter should enable the reader to obtain a basic understanding of the overall facility without having to refer to the subsequent chapters. Review of the detailed chapters that follow can then be accomplished with better perspective and with recognition of the relative safety importance of each individual item to the overall plant design.

1.1 Introduction

This section should present briefly the principal aspects of the overall application, including the type of license requested, the number of plant units, a brief description of the proposed location of the plant, the type of the nuclear steam supply system and its designer, the type of containment structure and its designer, the core thermal power levels, both rated and design,* and the corresponding net electrical output for each thermal power level, the scheduled completion date, and the anticipated commercial operation date for each unit.

1.2 General Plant Description

This section should include a summary description of the principal characteristics of the site and a concise description of the plant. The plant description should include a brief discussion of the principal design criteria, operating characteristics, and safety considerations for the nuclear steam supply system; the engineered safety features and emergency systems; the instrumentation, control, and electrical systems; the power conversion system; the fuel handling and storage systems; the cooling water and other auxiliary systems; and the radioactive waste management system. The general arrangement of major structures and equipment should be indicated by the use of plan and elevation drawings in sufficient number and detail to provide a reasonable understanding of the general layout of the plant. Those features of the plant likely to be of special interest because of their relationship to safety should be identified. Such items as unusual site characteristics, solutions to particularly difficult engineering problems, and significant extrapolations in technology represented by the design should be highlighted.

* Rated power is defined as the power level at which the plant would be operated if licensed. Design power is defined as the highest power level that would be permitted by plant design and that is used in some safety evaluations.

2. SITE CHARACTERISTICS

This chapter of the SAR should provide information on the geological, seismological, hydrological, and meteorological characteristics of the site and vicinity, in conjunction with present and projected population distribution and land use and site activities and controls. The purpose is to indicate how these site characteristics have influenced plant design and operating criteria and to show the adequacy of the site characteristics from a safety viewpoint.

2.1 Geography and Demography

2.1.1 Site Location and Description

2.1.1.1 Specification of Location. The location of each reactor at the site should be specified by latitude and longitude to the nearest second and by Universal Transverse Mercator Coordinates (Zone Number, Northing, and Easting, as found on USGS topographical maps) to the nearest 100 meters. The State and county or other political subdivision in which the site is located should be identified, as well as the location of the site with respect to prominent natural and man-made features such as rivers and lakes.

2.1.1.2 Site* Area Map. A map of the site area of suitable scale (with explanatory text as necessary) should be included. It should clearly show the following:

1. The plant property lines. The area of plant property in acres should be stated.

2. Location of the site boundary. If the site boundary lines are the same as the plant property lines, this should be stated.

3. The location and orientation of principal plant structures within the site area. Principal structures should be identified as to function (e.g., reactor building, auxiliary building, turbine building).

4. The location of any industrial, commercial, institutional, recreational, or residential structures within the site area.

5. The boundary lines of the plant exclusion area (as defined in 10 CFR Part 100). If these boundary lines are the same as the plant property lines, this should be stated. The minimum distance from each reactor to the exclusion area boundary should be shown and specified.

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*"Site" means the contiguous real estate on which nuclear facilities are located and for which one or more licensees has the legal right to control access by individuals and to restrict land use for purposes of limiting the potential doses from radiation or radioactive material during normal operation of the facilities.

motion in determining the seismic response of structures, systems, and components follow the recommendations of Regulatory Guide 1.92, "Combining Modal Responses and Spatial Components in Seismic Response Analysis."

3.7.2.7 Combination of Modal Responses. When a response spectra method is used, a description of the procedure for combining modal responses (shears, moments, stresses, deflections, and accelerations) should be provided. Indicate the extent to which the recommendations of Regulatory Guide 1.92 are followed.

3.7.2.8 Interaction of Non-Category I Structures with Seismic Category I Structures. Provide the design criteria used to account for the seismic motion of non-Category I structures or portions thereof in the seismic design of Seismic Category I structures or portions thereof. In addition, describe the design criteria that will be applied to ensure protection of Seismic Category I structures from the structural failure of non-Category I structures due to seismic effects.

3.7.2.9 Effects of Parameter Variations on Floor Response Spectra. The procedures that will be used to consider the effects of expected variations of structural properties, dampings, soil properties, and soil/structure interaction on floor response spectra (e.g., peak width and period coordinates) and time histories should be described.

3.7.2.10 Use of Constant Vertical Static Factors. Where applicable, identify and justify the application of constant static factors as vertical response loads for the seismic design of Seismic Category I structures, systems, and components in lieu of a vertical seismic-system dynamic analysis method.

3.7.2.11 Method Used to Account for Torsional Effects. The method used to consider the torsional effects in the seismic analysis of the Seismic Category I structures should be described. Where applicable, discuss and justify the use of static factors or any other approximate method in lieu of a combined vertical, horizontal, and torsional system dynamic analysis to account for torsional accelerations in the seismic design of Seismic Category I structures.

3.7.2.12 Comparison of Responses (FSAR). For the operating license review where both modal response and time history methods are applied, the responses obtained from both methods at selected points in major Seismic Category I structures should be provided, together with a comparative discussion of the responses.

3.7.2.13 Methods for Seismic Analysis of Dams. A comprehensive description of the analytical methods and procedures that will be used for the seismic system analysis of Seismic Category I dams should be provided. The assumptions made, the boundary conditions used, and the procedures by which strain-dependent soil properties are incorporated in the analysis should be provided.

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3.7.2 SEISMIC SYSTEM ANALYSIS

REVIEW RESPONSIBILITIES

Primary - Structural and Geosciences Branch (ESGB)

Secondary - None

I. AREAS OF REVIEW

The following areas related to the seismic system analysis described in the applicant's safety analysis report (SAR) are reviewed.

1. Seismic Analysis Methods

For all Category I structures, systems, and components, the applicable seismic analysis methods (response spectra, single time history or multiple time histories, equivalent static load) are reviewed. The manner in which the dynamic system analysis method is performed, including the modeling of foundation torsion, rocking, and translation, is reviewed. The method chosen for selection of significant modes and an adequate number of masses or degrees of freedom is reviewed. The manner in which consideration is given in the seismic dynamic analysis to maximum relative displacements between supports is reviewed. In addition, other significant effects that are accounted for in the dynamic seismic analysis such as hydrodynamic effects and nonlinear response are reviewed. If tests or empirical methods are used in lieu of analysis for any Category I structure, the testing procedure, load levels, and acceptance basis are also reviewed. The SRP criteria generally deal with linear elastic analysis coupled with allowable stresses near elastic limits of the structures. However, for certain special cases (e.g., evaluation of as-built structures), the staff has accepted the concept of limited inelastic/nonlinear behavior when appropriate. The actual analysis, incorporating inelastic/nonlinear considerations, is reviewed on a case-by-case basis.

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8. Interaction of Non-Category I Structures with Category I Structures

The design criteria to account for the seismic motion of non-Category I structures or portions thereof in the seismic design of Category I structures or portions thereof are reviewed. The procedures that are used to protect Category I structures from the structural failure of non-Category I structures, due to seismic effects, are reviewed.

9. Effects of Parameter Variations on Floor Responses

The procedures that are used to consider the effects of the expected variations of structural properties, dampings, soil properties, and soil-structure interaction on the floor response spectra and time histories are reviewed.

10. Use of Equivalent Vertical Static Factors

Where applicable, justification for the use of equivalent static factors as vertical response loads for designing Category I structures, systems, and components in lieu of the use of a vertical seismic system dynamic analysis is reviewed.

11. Methods Used to Account for Torsional Effects

The method employed to consider torsional effects in the seismic analysis of Category I structures is reviewed. The review includes evaluation of the conservatism of any approximate methods to account for torsional accelerations in the seismic design of Category I structures.

12. Comparison of Responses

For the operating license review, where applicable, the comparison of seismic responses for major Category I structures using modal response spectrum and time history approaches is evaluated.

13. Analysis Procedure for Damping

The analysis procedure to account for the damping in different elements of the model of a coupled system is reviewed.

14. Determination of Category I Structure Overturning Moments

The description of the method and procedure used to determine design overturning moments for Category I structures is reviewed.

15. Interface Review

Review of geological and seismological information to establish the free-field ground motion is performed as described in SRP Sections 2.5.1 through 2.5.3. The geotechnical parameters and methods employed in the analysis of free-field soil media and soil properties are reviewed as described in SRP Section 2.5.4. The results of the reviews for the where R_k is the response for the k^{th} mode and N is the number of significant modes considered in the modal response combination.

When modes with closely spaced modal frequencies exist (two modes having frequencies within 10 percent of each other), the methods delineated in Reference 8 are acceptable. Use of other methods for considering closely spaced modes, such as those outlined in References 4 and 9 will be reviewed and accepted on a case-by-case basis. Acceptance criteria for the adequate consideration of high-frequency modes are provided in Appendix A to this SRP section.

8. Interaction of Non-Category I Structures with Category I Structures

To be acceptable, the interfaces between Category I and non-Category I structures and plant equipment must be designed for the dynamic loads and displacements produced by both the Category I and non-Category I structures and plant equipment. In addition, a statement indicating the fact that all non-Category I structures meet any one of the following requirements should be provided.

a. The collapse of any non-Category I structure will not cause the non-Category I structure to strike a seismic Category I structure or component.

b. The collapse of any non-Category I structure will not impair the integrity of seismic Category I structures or components.

c. The non-Category I structures will be analyzed and designed to prevent their failure under SSE conditions in a manner such that the margin of safety of these structures is equivalent to that of Category I structures.

9. Effects of Parameter Variations on Floor Response Spectra

Consideration should be given in the analysis to the effects on floor response spectra (e.g., peak width and period coordinates) of expected variations of structural properties, dampings, soil properties, and soil-structure interactions. The acceptance criteria for the consideration of the effects of parameter variations are provided in subsection II.5 of this SRP section.

10. Use of Equivalent Vertical Static Factors

The use of equivalent static load factors as vertical response loads for the seismic design of all Category I structures, systems, and components in lieu of the use of a vertical seismic system dynamic analysis is acceptable only if it can be justified that the structure is rigid in the vertical direction. The criterion for rigidity is that the lowest frequency in the vertical direction is more than 33 cps.

11. Methods Used to Account for Torsional Effects

An acceptable method of treating the torsional effects in the seismic analysis of Category I structures is to carry out a dynamic analysis that

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SECTION 3.8.4 OTHER SEISMIC CATEGORY I STRUCTURES

REVIEW RESPONSIBILITIES

Primary - Structural Engineering Branch (SEB)

Secondary - None

I. AREAS OF REVIEW

The following areas relating to all seismic Category I structures and other safety-related structures that may not be classified as seismic Category I, other than the containment and its interior structures, are reviewed:

1. Description of the Structures

The descriptive information including plans and sections of each structure, is reviewed to establish that sufficient information is provided to define the primary structural aspects and elements relied upon for the structure to perform the safety-related function. Also reviewed is the relationship between adjacent structures including the separation provided or structural ties, if any. Among the major plant structures that are reviewed, together with the descriptive information reviewed for each, are the following:

a. Containment Enclosure Building

The containment enclosure building, which may surround all or part of the primary concrete or steel containment structure, is primarily intended to reduce leakage during and after a loss-of-coolant (LOCA) from within the containment. Concrete enclosure buildings also protect the primary containment, which may be of steel or concrete, from outside hazards.

The enclosure building is usually either a concrete structure or a structural steel and metal siding building.

Where it is a concrete structure, it usually has the geometry of the containment and, as applicable, the descriptive information reviewed is

C. General Design Criterion 4 as it relates to safety-related structure being capable of withstanding the dynamic effects of equipment failures including missiles and blowdown loads associated with the loss of coolant accidents.

D. General Design Criterion 5 as it relates to sharing of structures important to safety unless it can be shown that such sharing will not significantly impair their validity to perform their safety functions.

E. Appendix B to 10 CFR Part 50 as it relates to the quality assurance criteria for nuclear power plants.

The Regulatory Guides and industry standards identified in item 2 of this subsection provides information, recommendations and guidance and in general describes a basis acceptable to the staff that may be used to implement the requirements of 10 CFR Part 50, §50.55a and GDC 1, 2, 4, 5 and Appendix B to 10 CFR Part 50. Also, specific acceptance criteria necessary to meet the relevant requirements of these regulations for the areas of review, described ~~in subsection I of this SRP section are as follows:~~

1. Description of the Structures

The descriptive information in the SAR is considered acceptable if it meets the minimum requirements set forth in Section 3.8.4.1 of the "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants" (Ref. 4).

Deficient areas of descriptive information are identified by the reviewer and a request for additional information is initiated at the application acceptance review. New or unique design features that are not specifically covered in the "Standard Format..." may require a more detailed review. The reviewer determines the additional information that may be required to accomplish a meaningful review of the structural aspects of such new or unique features.

2. Applicable Codes, Standards, and Specifications

The design, materials, fabrication, erection, inspection, testing, and surveillance, if any, of Category I structures are covered by codes, standards, and guides that are either applicable in their entirety or in portions thereof. A list of such documents is as follows:

Specification	Title
ACI 349	"Code Requirements for Nuclear Safety-Related Concrete Structures"
AISC	"Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings"
Regulatory Guides	
1.10	Mechanical (Caldweld) Splices in Reinforcing Bars of Category I Concrete Structures

The following pages show the addition of the guidance suggesting that the seismic design of structures whose continued function is not required but whose failure could adversely impact the safety function of a Category I structure, or result in incapacitating injury to control room occupants, is reviewed.

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3.7.2 SEISMIC SYSTEM ANALYSIS

REVIEW RESPONSIBILITIES

Primary - ~~Structural~~Civil Engineering and Geosciences Branch (~~ESGB~~ECGB[1])

Secondary - None

I. AREAS OF REVIEW

The following areas related to the seismic system analysis described in the applicant's safety analysis report (SAR) are reviewed.

1. Seismic Analysis Methods

For all Category I structures, systems, and components (SSCs)[2], the applicable seismic analysis methods (response spectra, single time history or multiple time histories, equivalent static load) are reviewed. The manner in which the dynamic system analysis method is performed, including the modeling of foundation torsion, rocking, and translation, is reviewed. The method chosen for selection of significant modes and an adequate number of masses or degrees of freedom is reviewed. The manner in which consideration is given in the seismic dynamic analysis to maximum relative displacements between supports is reviewed. In addition, other significant effects that are accounted for in the dynamic seismic analysis such as hydrodynamic effects and nonlinear response are reviewed. If tests or empirical methods are used in lieu of analysis for any Category I structure, the testing procedure, load levels, and acceptance basis are also reviewed. The SRP criteria generally deal with linear elastic analysis coupled with allowable stresses near elastic limits of the structures. However, for certain special cases (e.g., evaluation of as-built structures), the staff has accepted the concept of procedure for combining modal responses (shears, moments, stresses, deflections, and ~~accelerations) is reviewed, including that for modes~~ with closely spaced frequencies.

8. Interaction of Non-Category I Structures with Category I Structures

The design criteria to account for the seismic motion of non-Category I structures or portions thereof in the seismic design of Category I structures or portions thereof are reviewed. **The seismic design of structures whose continued function is not required but whose failure could adversely impact the safety function of a Category I structure, or result in incapacitating injury to control room occupants, is reviewed.**[6] The procedures that are used to protect Category I structures from the structural failure of non-Category I structures, due to seismic effects, are reviewed.

9. Effects of Parameter Variations on Floor Responses

The procedures that are used to consider the effects of the expected variations of structural properties, dampings **values**[7], soil properties, and soil-structure interaction on the floor response spectra and time histories are reviewed.

10. Use of Equivalent Vertical Static Factors

Where applicable, justification for the use of equivalent static factors as vertical response loads for designing Category I ~~structures, systems, and components~~SSCs[8] in lieu of the use of a vertical seismic system dynamic analysis is reviewed.

11. Methods Used to Account for Torsional Effects

The method employed to consider torsional effects in the seismic analysis of Category I structures is reviewed. The review includes evaluation of the conservatism of any approximate methods to account for torsional accelerations in the seismic design of Category I structures.

12. Comparison of Responses

For the operating license review, where applicable, the comparison of seismic responses for major Category I structures using modal response spectrum and time history approaches is evaluated.

13. Analysis Procedure for Damping

The analysis procedure to account for the damping in different elements of the model of a coupled system is reviewed.

14. Determination of Category I Structure Overturning Moments

The description of the method and procedure used to determine design overturning moments for Category I structures is reviewed.

SRP Draft Section 3.7.2
Attachment A - Proposed Changes in Order of Occurrence

Item numbers in the following table correspond to superscript numbers in the redline/strikeout copy of the draft SRP section.

Item	Source	Description
1.	Editorial - Current PRB names and abbreviations.	Editorial change made to reflect current PRB name and responsibility for SRP section 3.7.2.
2.	Editorial.	Consistent with other SRP sections, added the acronym SSC for "structure system and component," to be used in lieu of the full expression.
3.	Editorial.	Substituted grammatically correct verb form to match subject of clause; the soil deposits are assumed to be symmetrical.
4.	Editorial.	Substituted the acronym SSC for "structure, system and component." This acronym was introduced at the beginning of this SRP section, as its use is consistent with other SRP sections.
5.	Editorial.	Substituted the acronym SSC for "structure, system and component." This acronym was introduced at the beginning of this SRP section, as its use is consistent with other SRP sections.
6.	**Integrated Impact 101.**	Revised paragraph I.8 of Areas of Review to clarify the applicability of seismic review to non-seismic Category I structures.
7.	Editorial.	For clarification, changed "dampings" to "damping values."
8.	Editorial.	Substituted the acronym SSC for "structure, system and component." This acronym was introduced at the beginning of this SRP section, as its use is consistent with other SRP sections.
9.	SRP-UDP format item, Reformat Areas of Review.	Added "Review Interfaces" heading to Areas of Review. Reformatted existing description of review interfaces in numbered format to describe how the ECGB reviews aspects of the seismic system analysis under other SRP Sections and how other branches support the review.
10.	SRP-UDP format item, Reformat Areas of Review.	Added standard leader sentence for new "Review Interfaces" subsection.
11.	SRP-UDP format item, Reformat Areas of Review.	Segregated the existing material into separate items, consistent with the SRP-UDP guidelines for the new Review Interfaces subsection.
12.	SRP-UDP format item, Reformat Areas of Review.	Segregated the existing material into separate items, consistent with the SRP-UDP guidelines for the new Review Interfaces subsection.

SRP Draft Section 3.7.2

Attachment B - Cross Reference of Integrated Impacts

Integrated Impact No.	Issue	SRP Subsections Affected
96	Provide technical rationale for guidance documents and for specific acceptance criteria.	No SRP change due to this ROC. Development of technical rationale is already provided for by SRP-UDP guidelines.
97	Revise Review Procedures to include inter-structure effects in soil structure interaction analyses.	No SRP change due to this ROC. Effect of adjacent buildings on building response is already addressed in III. Review Procedures, Item 4.
98	Consider qualified endorsement of industry standard ASCE 4-86 as a candidate for future work.	No SRP change due to this ROC. This item will be tracked with IPD-7.0 Form 3.7.2-1.
99	Revise specific criteria to allow, on a case by case basis, alternatives to peak broadening of floor response spectra.	No SRP change due to this ROC, as a definite and consistent staff position on peak shifting and peak clipping methodologies has not yet been solidified.
100	Provide a review procedure to consider effects of concrete cracking on seismic analysis of seismic Category I structures.	III. Review Procedures, Item 9. Effects of Parameter Variations on Floor Response Spectra.
101	Revise section to provide for seismic analysis of non-Category I structures, systems, and components.	I. Areas of Review, Item 15. Analysis of Non-Category I Structures; II. Acceptance Criteria, Item 15. Analysis of Non-Category I Structures: III. Review Procedures, Item 15. Analysis of Non-Category I Structures.
971	Revise Acceptance Criteria, Review Procedures and Evaluation Findings, applicable to evolutionary plants, for review of seismic system analysis.	III. Review Procedures VI. References
1138	Revise Acceptance Criteria, Review Procedures, and Evaluation Findings, as necessary, to incorporate the guidance of the proposed Draft Regulatory Guide DG-1015.	No SRP change due to this ROC, pending final approval of the Draft Regulatory Guide.
1222	Revise the SRP to incorporate the new and/or revised requirements from proposed rulemaking 59 FR 52255.	No SRP change due to this ROC, pending final approval of changes to 10 CFR 100, 10 CFR 50.34, 10 CFR 50.54, and of new Appendix S to 10 CFR 50.
1321	Modify the Areas of Review and Acceptance Criteria subsections to accept ASME Code Case N-411-1, as conditioned by Regulatory Guide 1.84, as a source of damping values.	II. Acceptance Criteria, Item 13. Analysis Procedure for Damping and VI. References, Items 4 and 11.

Regulatory Guide 1.143.

U.S. NUCLEAR REGULATORY COMMISSION Revision 2
November 2001

REGULATORY GUIDE

OFFICE OF NUCLEAR REGULATORY RESEARCH

REGULATORY GUIDE 1.143

(Draft was issued as DG-1100)

DESIGN GUIDANCE FOR RADIOACTIVE WASTE MANAGEMENT SYSTEMS, STRUCTURES, AND COMPONENTS INSTALLED IN LIGHT-WATER-COOLED NUCLEAR POWER PLANTS

A. INTRODUCTION

This regulatory guide has been revised to provide guidance to licensees and applicants on methods acceptable to the staff for complying with the NRC's regulations in the design, construction, installation, and testing the structures, systems, and components of radioactive waste management facilities in light-water-reactor nuclear power plants.

In 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," § 50.34, "Contents of Applications; Technical Information," requires that each application for a construction permit include a preliminary safety analysis report. Part of the information required is related to quality assurance and the preliminary design of the facility, including, among other things, the principal design criteria for the facility. Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR Part 50 establishes overall quality assurance requirements for structures, systems, and components important to safety. Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50 establishes minimum requirements for the principal design criteria for light-water-cooled nuclear power plants.

Criterion 1, "Quality Standards and Records," of Appendix A requires that structures, systems, and components important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance to safety of the safety function to be performed and that a quality

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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 NRC's regulations, techniques used by the staff in evaluating specific problems or postulated accidents, and data needed by the NRC staff in its review of applications for permits and licenses. Regulatory guides are not substitutes for regulations, and compliance with them is not required. Methods and solutions different from those set out in the guides will be acceptable if they provide a basis for the findings requisite to the issuance or continuance of a permit or license by the Commission.

This guide was issued after consideration of comments received from the public. Comments and suggestions for improvements in these guides are encouraged at all times, and guides will be revised, as appropriate, to accommodate comments and to reflect new information or experience. Written comments may be submitted to the Rules and Directives Branch, ADM, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001.

Regulatory guides are issued in ten broad divisions: 1, Power Reactors; 2, Research and Test Reactors; 3, Fuels and Materials Facilities; 4, Environmental and Siting; 5, Materials and Plant Protection; 6, Products; 7, Transportation; 8, Occupational Health; 9, Antitrust and Financial Review; and 10, General.

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assurance program be established and implemented in order to provide adequate assurance that these structures, systems, and components will satisfactorily perform their safety function.

Criterion 2, "Design Bases for Protection Against Natural Phenomena," of Appendix A requires, among other things, that structures, systems, and components important to safety be designed to withstand the effects of natural phenomena such as earthquakes, tornados, or flooding without loss of capability to perform their safety functions. The design bases for these structures, systems, and components are to reflect the importance of the safety functions to be performed. Appendix S, "Earthquake Engineering Criteria for Nuclear Power Plants," of 10 CFR Part 50 states general design requirements for the implementation of General Design Criterion 2. Criterion 60, "Control of Releases of Radioactive Materials to the Environment," of Appendix A requires that the nuclear power unit design include means to suitably control the release of radioactive materials in gaseous and liquid effluents and to handle radioactive solid waste produced during normal reactor operation, including anticipated operational occurrences. The release of radioactive materials from external man-induced events and design basis accidents must also be controlled.

This regulatory guide is being revised to provide design guidance acceptable to the NRC staff in regard to natural phenomena hazards, internal and external man-induced hazards, and quality group classification and quality assurance provisions for radioactive waste management systems, structures, and components.[1] Further, it describes provisions for mitigating design basis accidents and controlling releases of liquids containing radioactive materials, e.g., spills or tank overflows, from all plant systems outside reactor containment.

Licensees and applicants may propose means other than those specified by the provisions of the Regulatory Position of this guide for meeting applicable regulations. No new requirements are being imposed by this regulatory guide. Implementation of this guidance by licensees will be on a strictly voluntary basis.

The information collections contained in this regulatory guide are covered by the requirements in 10 CFR Part 50, which were approved by the Office Management and Budget, approval number 3150-0011. If a means used to impose an information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection.

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[1] Adams et al, "Re-evaluation of Regulatory Guidance Provided in Regulatory Guides 1.142 and 1.142," NUREG/CR-5733, August 1999. Copies are available at current rates from the U.S. Government Printing Office, P.O. Box 37082, Washington, DC 20402-9328 (telephone (202)512-1800); or from the National Technical Information Service by writing NTIS at 5285 Port Royal Road, Springfield, VA 22161; (telephone (703)487-4650; <http://www.ntis.gov/ordernow>. Copies are available for inspection or copying for a fee from the NRC Public Document Room at 11555 Rockville Pike, Rockville, MD; the PDR's mailing address is USNRC PDR, Washington, DC 20555; telephone (301)415-4737 or (800)397-4209; fax (301)415-3548; email is PDR@NRC.GOV.

B. DISCUSSION

One aspect of nuclear power plant operation is the control and management of liquid, gaseous, and solid radioactive waste[2] (radwaste) generated as a byproduct of nuclear power. The purpose of this guide is to provide information and criteria that will provide reasonable assurance that components and structures used in the radioactive waste management and steam generator blowdown systems are designed, constructed, installed, and tested on a level commensurate with the need to protect the health and safety of the public and plant operating personnel. It sets forth minimum staff recommendations and is not intended to prohibit the implementation of more rigorous design considerations, codes, standards, or quality assurance measures.

ANSI/ANS Standards 55.1-1992, "Solid Radioactive Waste Processing System for Light Water Cooled Reactor Plants,"[3] 55.4-1993, "Gaseous Radioactive Waste Processing Systems for Light Water Plants,"[3] and 55.6-1993, "Liquid Radioactive Waste Processing Systems for Light Water Reactor Plants,"[3] have been reviewed for applicability to this guide. These ANSI/ANS Standards provide a wider range of guidance than that provided in Sections 11.2, "Liquid Waste Management System"; 11.3, "Gaseous Waste Management System"; and 11.4, "Solid Waste Management System," of Chapter 11, "Radioactive Waste Management," of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants."[4] As appropriate, guidance from the ANSI/ANS standards has been incorporated by reference.

For the purposes of this guide, the radwaste systems are considered to begin at the interface valves in each line from other systems provided for collecting wastes that may contain radioactive materials and to include related instrumentation and control systems. The radwaste system terminates at the point of controlled discharge to the environment, at the point of recycle to the primary or secondary water system storage tanks, or at the point of storage of packaged solid wastes.

The steam generator blowdown system begins at, but does not include, the outermost containment isolation valve on the blowdown line. It terminates at the point of controlled discharge to the environment, at that point of interface with other liquid systems, or at the point of recycle back to the secondary system. For design purposes, portions of radwaste systems that interface with other systems are considered to be in the system with more rigorous requirements.

Except as noted, this guide does not apply to the reactor water cleanup system, the condensate cleanup system, the chemical and volume control system, the reactor coolant and auxiliary building equipment drain tanks, the sumps and floor drains provided for collecting liquid wastes, the boron recovery system, equipment used to prepare solid waste solidification agents, the building ventilation systems (heating, ventilating, and air conditioning), instrumentation and

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[2] Radioactive waste, as used in this guide, means liquids, gases, or solids that contain radioactive materials that by design or operating practice will be processed prior to final disposition.

[3] Copies may be obtained from the American Nuclear Society, 555 North Kensington Avenue, La Grange Park, Illinois 60525.

[4] Copies of sections of NUREG 0800 are available by email to DISTRIBUTION@NRC.GOV or by fax to (301)415-2289.

sampling systems beyond the first root valve, or the chemical fume hood exhaust systems. In addition, this guide does not apply to the main condenser circulating or component cooling water systems, the spent fuel handing and storage systems, or the fuel pool water cleanup system.

The design and construction of radioactive waste management and steam generator blowdown systems should provide assurance that radiation exposures to operating personnel and to the general public are as low as is reasonably achievable. One aspect of this consideration is ensuring that these systems are designed to quality standards that enhance system reliability, operability, and availability. In developing this design guidance, the NRC staff has considered designs and concepts submitted in license applications and resulting operating system histories. It has also been guided by industry practices and the cost of design features, taking into account the potential impact on the health and safety of operating personnel and the general public.

C. REGULATORY POSITION

1. SYSTEMS HANDLING RADIOACTIVE MATERIALS IN LIQUIDS

1.1 Liquid Radwaste Treatment System

The liquid radwaste treatment system, including the steam generator blowdown system, downstream of the outer-most containment isolation valve should meet the following criteria.

1.1.1 The structures, systems, and components (SSCs) of the liquid radwaste treatment system should be designed and tested to requirements set forth in the codes and standards listed in Table 1 of this guide, supplemented by Regulatory Positions 1.1.2 and 1.1.3 of this guide.

1.1.2 Materials for pressure-retaining components, excluding HVAC duct and fire protection piping, should conform to the requirements of the specifications for materials listed in Section II of the ASME Boiler and Pressure Vessel Code,[5] except that malleable, wrought, or cast iron materials and plastic pipe should not be used. Materials should be compatible with the chemical, physical, and radioactive environment of specific applications during normal conditions and anticipated operational occurrences. Manufacturers' material certificates of compliance with material specifications such as those contained in the codes referenced in the materials column of Table 1 may be provided in lieu of certified material test reports.

1.1.3 Foundations and walls of structures that house the liquid radwaste system should be designed to the natural phenomena and internal and external man-induced hazards criteria described in Regulatory Position 6 of this guide to a height sufficient to contain the maximum liquid inventory expected to be in the building.

[5] Copies may be obtained from the American Society of Mechanical Engineers, United Engineering Center, 345 East 47th Street, New York, NY 10017.

1.2 SSCs Outside Containment that Contain Radioactive Liquids

All SSCs located outside the reactor containment that contain radioactive materials in liquid form should be classified as described in Regulatory Position 5 and designed in accordance with Regulatory Position 6. In addition, any such component should be designed to prevent uncontrolled releases of radioactive materials caused by spillage in buildings or from outdoor components. The following design features should be included for such components and should meet the criteria contained in Sections 5.2, 5.3, and 5.4 of ANSI/ANS 55.1-1992.

1.2.1. All tanks inside and outside the plant, including the condensate storage tanks, should have provisions to monitor liquid levels. Designated high-liquid-level conditions should actuate alarms both locally and in the control room.

1.2.2. All radwaste tanks, overflows, drains, and sample lines should be routed to the liquid radwaste treatment system. Retention by an intermediate sump or drain tank that is designed for handling radioactive materials and that has provisions for routing to the liquid radwaste system is acceptable.

1.2.3. Indoor radwaste tanks should have curbs or elevated thresholds with floor drains routed to the liquid radwaste treatment system. Retention by an intermediate sump or drain tank that is designed for handling radioactive materials and that has provisions for routing to the liquid radwaste system is acceptable.

1.2.4. The design should include provisions to prevent leakage from entering unmonitored and nonradioactive systems and ductwork in the area.

1.2.5. Outdoor tanks should have a dike or retention pond capable of preventing runoff in the event of a tank overflow and should have provisions for sampling collected liquids and routing them to the liquid radwaste treatment system.

2. GASEOUS RADWASTE SYSTEMS

For a BWR, the gaseous radwaste system includes the system provided for treatment of normal offgas releases from the main condenser vacuum system beginning at the point of discharge from the condenser air removal equipment; for a PWR the gaseous radwaste system includes the system provided for the treatment of gases stripped from the primary coolant.

2.1 The SSCs of the gaseous radwaste treatment system should be designed and tested to requirements set forth in the codes and standards listed in Table 1 supplemented by Regulatory Positions 2.2 and 2.3 of this guide.

2.2 Materials for pressure-retaining components, excluding HVAC duct and fire protection piping, should conform to the requirements of the specifications for materials listed in Section II of the ASME Boiler and Pressure Vessel Code, except that malleable, wrought, or cast iron materials and plastic pipe should not be used. Materials should be compatible with the chemical, physical, and radioactive environment of specific applications during normal conditions and anticipated operational occurrences. If the potential for an explosive mixture of hydrogen and oxygen exists, adequate provisions should be made to preclude buildup of explosive mixtures, or the system should be designed to withstand the effects of an explosion. Manufacturers' material certificates of compliance with material specifications such as those contained in the codes referenced in the materials column of Table 1 may be provided in lieu of certified materials test reports.

2.3 The portions of the gaseous radwaste treatment system that are intended to store or delay the release of gaseous radioactive waste, including portions of structures housing these systems, should be classified as described in Regulatory Position 5 and designed in accordance with Regulatory Position 6.

3. SOLID RADWASTE SYSTEM

The solid radwaste system consists of slurry waste collection and settling tanks, spent resin storage tanks, phase separators, and components and subsystems used to dewater or solidify radwastes prior to storage or offsite shipment.

3.1 The SSCs of the solid radwaste treatment system should be designed and tested to the requirements set forth in the codes and standards listed in Table 1 supplemented by Regulatory Positions 3.2 and 3.3 of this guide.

3.2 Materials for pressure-retaining components, excluding HVAC duct and fire protection piping, should conform to the requirements of the specifications for materials listed in Section II of the ASME Boiler and Pressure Vessel Code, except that malleable, wrought, or cast iron materials and plastic pipe should not be used. Materials should be compatible with the chemical, physical, and radioactive environment of specific applications during normal conditions and anticipated operational occurrences. Manufacturers' material certificates of compliance with material specifications such as those contained in the codes referenced in the materials column of Table 1 may be provided in lieu of certified materials test reports.

3.3 Foundations and adjacent walls of structures that house the solid radwaste system should be designed to the natural phenomena and internal and external man-induced hazards guidance given in Regulatory Position 6 of this guide to a height sufficient to contain the maximum liquid inventory expected to be in the building.

3.4 Equipment and components used to collect, process, or store solid radwastes need not be designed to the seismic guidance in Regulatory Position 6 of this guide.

4. ADDITIONAL DESIGN, CONSTRUCTION, AND TESTING

In addition to the requirements inherent in the codes and standards listed in Table 1, the following, as a minimum, should be applicable to SSCs listed in Regulatory Position 6 of this guide.

4.1 Radioactive waste management SSCs should be designed to control leakage and facilitate access, operation, inspection, testing, and maintenance in order to maintain radiation exposures to operating and maintenance personnel as low as is reasonably achievable. Regulatory Guide 8.8, "Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations Will Be As Low As Is Reasonably Achievable," provides guidance that is acceptable to the NRC staff on this subject.

4.2 The quality assurance provisions described in Regulatory Position 7 of this guide should be applied.

4.3 Pressure-retaining components of process systems should use welded construction to the maximum practicable extent. Process systems include the first root valve on sample and instrument lines. Flanged joints or suitable rapid-disconnect fittings should be used only where maintenance or operational requirements clearly indicate such construction is preferable. Screwed connections in which threads provide the only seal should not be used except for instrumentation and cast pump body drain and vent connections where welded connections are not suitable. Process lines should not be less than 3/4 inch (nominal). Screwed connections backed up by seal welding, mechanical joints, or socket welding may be used on lines 3/4 inches or larger but less than 2-1/2 inches. For lines 2-1/2 inches and above, pipe welds should be of the butt-joint type. Nonconsumable backing rings should not be used in lines carrying resins or other particulate material. All welding constituting the pressure boundary of pressure-retaining components should be performed in accordance with ASME Boiler and Pressure Vessel Code Section IX.

4.4 Piping systems should be hydrostatically tested in their entirety except (1) at atmospheric tanks where no isolation valves exist, (2) when such testing would damage equipment, and (3) when such testing could seriously interfere with other system or component testing. For (2) and (3), pneumatic testing should be performed. Pressure testing should be performed on as large a portion of the in-place systems as practicable. Testing of piping systems should be performed in accordance with applicable ASME or ANSI codes listed in Table 1.

4.5 Inspection and testing provisions should be incorporated to enable periodic evaluation of the operability and required functional performance of active components of the system.

5. CLASSIFICATION OF RADWASTE SYSTEMS FOR DESIGN PURPOSES

There are three safety classes, or classifications, for radwaste management facilities: RW-IIa (High Hazard), RW-IIb (Hazardous), and RW-IIc (Non-Safety).[1] RW-IIa is the most stringent class and RW-IIc is the least stringent. These classifications were developed primarily for natural phenomena and man-induced hazard design. The radiological release criteria (500 millirem at the unprotected area boundary and 5 rem to facility personnel within the protected boundary) was selected to be consistent with the criteria of 10 CFR Part 20, "Standards for Protection Against Radiation." This safety classification is applied to SSCs as follows.

5.1 For a given structure housing radwaste processing systems or components, if the total design basis unmitigated radiological release (considering the maximum inventory) at the boundary of the unprotected area is greater than 500 millirem per year or the maximum unmitigated exposure to site personnel within the protected area is greater than 5 rem per year, the external structures are classified as RW-IIa.

5.2 For a given structure housing radwaste processing systems or components, if the total design basis unmitigated radiological release (considering the maximum inventory) at the boundary of the unprotected area is less than 500 millirem per year and the maximum unmitigated exposure to site personnel within the protected area is less than 5 rem per year, the external structure is classified as RWE-IIb.

5.3 Any systems or components in a RW-IIa facility (see Regulatory Position 5.1) that store, process, or handle radioactive waste in excess of the A_1 quantities given in Appendix A, "Determination of A_1 and A_2," to 10 CFR Part 71, "Packaging and Transportation of Radioactive Material," are classified as RW-IIa. These systems or components that process radioactive waste in excess of the A_2 quantities but less than the A_1 quantities given in Appendix A to 10 CFR Part 71 are classified as RW-IIb. All other components are classified as RW-IIc. This classification may be modified for specific radwaste components.

5.4 Any systems or components in a RW-IIb structure (see Regulatory Position 5.2) that are used to store or process specified radioactive waste in excess of the A_1 quantities given in Appendix A to 10 CFR Part 71 are classified as RW-IIb. All other systems or components are classified as RW-IIc.

The unprotected area boundary mentioned in Regulatory Position 5.1 is shown in Figure 1. A flowchart of the Safety Classification Process is shown in Figure 2. The classifications discussed in Regulatory Positions 5.1 through 5.4 are not intended to apply to radwaste storage facilities if they do not contain any systems or components that exceed the quantities specified in Regulatory Positions 5.1 through 5.4.

6. NATURAL PHENOMENA AND MAN-INDUCED HAZARDS DESIGN FOR RADWASTE MANAGEMENT SYSTEMS AND STRUCTURES

6.1 General Design Criteria

Solid, liquid, and gaseous radwaste SSCs described in Regulatory Positions 1, 2, and 3 for natural phenomena and internal and external man-induced hazards should be evaluated as put forth in this position.

6.1.1. The natural phenomena and internal and external man-induced hazards demand definitions are as given in Table 2.

6.1.2. The natural phenomena and internal and external man-induced hazards design load combinations are as given in Table 3.

6.1.3. The natural phenomena and internal and external man-induced hazards should meet capacity criteria in Table 4.

6.1.4. The acceptability evaluation should be based on the requirements of the codes and standards given in Table 1, using the capacity criteria in Table 4.

6.2 Buildings Housing Radwaste Systems

6.2.1 Regardless of its safety classification, the foundation and walls up to the spill height of the building housing the radwaste systems should be designed to the criteria of Tables 1, 2, 3, and 4.

For classifications RW-IIb and RW-IIc, all SSCs should be designed at least for seismic base shear requirements of the Standard Uniform Building Code(UBC), 1997. The guidance of Volume 2 of the UBC 1997 and American Society of Civil Engineers ASCE 7-95, "Minimum Design Loads for Buildings and Other Structures," should be used as noted in Table 2 of this regulatory guide.

6.2.2. In lieu of the criteria and procedures referenced in this Regulatory Position 6, optional shield structures constructed around and supporting the radwaste systems may be erected to protect the radwaste systems from the effects of failure of the housing structure. If this option is adopted, Regulatory Position 6.2.1 need only be applied to the shield structures.

7. QUALITY ASSURANCE FOR RADWASTE MANAGEMENT SYSTEMS

Since the impact of these systems on safety is limited, the extent of control required by Appendix B to 10 CFR Part 50 is similarly limited. To ensure that systems will perform their intended functions, a quality assurance program sufficient to ensure that all design, construction, and testing provisions are met should be established and documented. A quality assurance program acceptable to the NRC staff is presented in ANSI/ANS-55.6-1993, "Liquid Radioactive Waste Processing System for Pressurized Water Reactor Plants."

Section 4.3, "Quality Assurance," of ANSI/ANS 55.6-1993 provides quality assurance guidance that is acceptable to the NRC staff for the system designer and procurer and for the system constructor. The design, procurement, fabrication, and construction activities should conform to the quality control provisions of the codes and standards specified in Table 1 of this guide. In addition, or when not covered by the referenced codes and standards, sufficient records should be maintained to furnish evidence that quality assurance measures are being implemented. The records should include results of reviews and inspections and should be identifiable and retrievable.

D. IMPLEMENTATION

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

Except in those cases in which the applicant proposes an acceptable alternative method for complying with the specified portions of the NRC's regulations, the method described in this guide reflecting public comments will be used in the evaluation of an applicant's design, construction, installation, and testing of radioactive waste management facilities, and in the evaluation of structures, systems, and components in light-water-cooled nuclear power plants. Current licensees may, at their option, comply with the guidance in this regulatory guide.

Table 1 - Codes and Standards for the Design of SSC in Radwaste Facilities[1]				
Component	**Design and Construction**	**Materials**	**Welding**	**Inspection and Testing**
Structures - Concrete	ACI-318 or ACI 349[2,3,4]	ACI-318 or ACI 349	ACI-318 or ACI 349	ACI-318 or ACI 349
Structures-Steel (Hot Rolled)	AISC-ASD or AISC LFRD or AISC N-690(S327) [2,4]	ASTM-A36	AWS-D1.1	AISC Standards and AWS Standards
Structures-Steel (Cold Formed)	AISI SG-673	ASTM-A500	AWS-D1.3, D9.1	AISC Standards and AWS Standards
Piping and Valves	ANSI/ASME B31.3 [5,6]	ASME-Sec. II[7]	ASME, Sec. IX	ANSI/ASME B31.3
Atmospheric Tanks	API-650	ASME Sec. II	ASME, Sec. IX	API-620
Tanks (0-15 psig)	API-620	ASME Sec. II	ASME, Sec. IX	API-650
Pressure Vessels and Tanks (>15 psig)	ASME BPVC Div. 1 or Div. 2	ASME Sec. II	ASME, Sec. IX	ASME Section VIII, Div. 1 or 2
Pumps	API-610; API-674; API-675; ASME BPVC Section VIII, Div. 1 or Div. 2	ASTM A571-84(1997) or ASME Sec. II	ASME, Sec. IX	ASME BPVC Code Section III, Class 3[8]
Heat Exchangers	TEMA STD, 8th Edition; ASME BPVC Section VIII Div. 1 or Div. 2	ASTM B359-98 or ASME Sec. II	ASME, Sec. IX	ASME Section VIII, Div. 1 or 2
HVAC Systems	SMACNA Stds. [6,9]	ASTM F856-97 ASTM C1290-00	AWS-D1.1, D1.3, D9.1	SMACNA Stds
Conduit and Cable Trays	NEMA TC2-1998 NEMA VE1-1998	ASTM B633-98, A123/A123M-01 NEMA TC2, VE1	AWS-D1.1, D1.3, D9.1	NEMA TC2-1998, NEMA VE1-1998
Fire Protection Systems	NFPA-13[6,10]; NFPA-14	ASTM-A795	AWS-D1.1, D1.3, D9.1, D10.9	NFPA-13
Flexible Hoses and Hose Connections for MRWP[11]	ANSI/ANS-40.37	ANSI/ANS-40.37	ANSI/ANS-40.37	ANSI/ANS-40.37

Footnotes for Table 1:

1 For a comprehensive lists of codes and standards referenced in Tables 1-4, see Appendix A to this regulatory guide.

2 Applicable to structure enclosing or supporting pressurized gaseous waste or liquid waste systems up to spill height. Also applicable to solid waste facility foundations slab and connected wall or column sections up to a height of 10 feet.

3 Appropriate load combinations and capacity criteria for component designs are specified in Table 3 of this regulatory guide.

4 Class RW-IIa Structures are to use ACI-349 and/or AISC N-690(S327) as applicable.

5 Class RW-IIa and RW-IIb Piping Systems are to be designed as category "M" systems.

6 Classes RW-IIa, RW-IIb, and RW-IIc are discussed in Regulatory Position 5 of this regulatory guide.

7 ASME BPVC Section II required for Pressure Retaining Components.

8 ASME Code Stamp, material traceability, and the quality assurance criteria of ASME BPVC, Section III, Div. 1, Article NCA are not required. Therefore, these components are not classified as ASME Code Class 3.

9 Class RW-IIa and RW-IIb HVAC systems are to use SMACNA "Seismic Restraint Manual Guides for Mechanical Systems."

10 Class RW-IIa and RW-IIb Fire Protection Systems are to be designed to NFPA-13, Section 4-14.4.3.

11 Flexible hoses should only be used in conjunction with Moblie Radwaste Processing Systems (MRWP).

Table 2 - Natural Phenomena and Internal/External Man-Induced Hazard Design Criteria for Safety Classification			
Loading	**Classification**		
	RW-IIa (High Hazard)	**RW-IIb (Hazardous)**	**RW-IIc (Non-Safety)**
Earthquake	OBE or 1/2 SSE	ASCE 7-95, Category III[1] or UBC 97, Category 2[2]	ASCE 7-95, Category II[1] UBC-97, Category 4[2]
Wind	ASCE 7-95, Category III[1]	ASCE 7-95, Category III[1]	ASCE 7-95, Category II[1]
Tornado	ANS 2.3 at a Probability of 1 x 10^{-5} /yr or three-fifths of Criteria in Regulatory Guide 1.76, Table 1.	Not Required	Not Required
Tornado Missile from SRP Section 3.5	A. 75 lbs, 3 in. nominal diameter sch. 40 pipe. Maximum velocity 0.4 x max. wind speed horizontal and 0.28 times max. wind speed vertical direction.[3] B. Automobile wt. 4000 lbs with frontal area of 20.0 sq. ft. traveling horizontally at 0.2 times maximum wind speed horizontally and 0.14 times maximum wind speed up to a height of 35 ft above grade.[4]	Not Required	Not Required
Flood	Regulatory Guide 1.59, one-half of the PMF.[5]	ASCE 7-95	ASCE 7-95
Precipitation [(6)] **(Rain, Snow)**	ANS 2.8 at probability of 1 x 10^{-3}/yr or Regulatory Guide 1.59, one-half precipitation specific for the PMF.[5]	ASCE 7-95, Category III[1]	ASCE 7-95, Category II[1]
Accidental Explosion Fixed Facility	To be evaluated on a case-by-case basis, plant-specific definition.	Not Required	Not Required
Accidental Explosion Transportation Vehicle	See Regulatory Guide 1.91.	Not Required	Not Required
Malevolent Vehicle Assault	Regulatory Guide 5.68 or plant-specific definition.	Not Required	Not Required
Small Aircraft Crash	Plant-specific definition	Not Required	Not Required

Footnotes for Table 2:

1 ASCE 7-95, Table 1-1.
2 UBC-97, Table 16-k.
3 Penetrating-type missile.
4 Impact-type missile.
5 PMF = Probable Maximum Flood.
4 Resistance to lightening strike should also be included in the design.

Table 3 - Design Load Combinations

System, Structure, Component (SSC)	Service Levels	SSC Safety Class		
		RW - IIa	**RW- IIb**	**RW- IIc**
External Structures (Concrete, Steel, Component Support Structures[1])	A (Normal)	$D + L + T_o$	$D + L + T_o$	$D + L + T_o$
	B (Severe; Upset)	$D + L + T_o$ $D + L + T_o + E_o$ $D + L + T_o + W + R$ $D + L + T_o + F$	$D + L + Tb$ $D + L + T_o + E'_o$ $D + L + T_o + W + R$ $D + L + T_o + F$	$D + L + Tb$ $D + L + T_o + E''_o$ $D + L + T_o + W + R$ $D + L + T_o + F$
External Conduits and Cable Trays	D (Abnormal Extreme; Faulted)	$D + L + T_o + W_t$ $D + L + T_o + V_m$ $D + L + T_o + A_c$ $D + L + T_a + A_D$ $D + L + T_a + A$	Not required	Not required
Internal Structures (Concrete, Steel Component Support Structures[1])	A (Normal)	$D + L + T_o$	$D + L + T_o$	$D + L + T_o$
	B (Severe, Upset)	$D + L + T_b$ $D + L + T_o + E_o$ $D + L + T_o + F$	$D + L + T_b$ $D + L + T_o + E_o$ $D + L + T_o + F$	$D + L + T_b$ $D + L + T_o + E_o$ $D + L + T_o + F$
Internal Conduit and Cable Trays	D (Abnormal Extreme; Faulted)	$D + L + T_a + A_D$ $D + L + T_a + A$	N/R	N/R
Pressure Retaining Components[2] (Piping, Valves, Pressure Vessels, Atmosphere, Tanks, 0-15 psig Tanks, Pumps Heat Exchangers) HVAC Systems Fire Protection Systems	A (Normal)	$P_d + D + D_m$ T_o	$P_d + D + D_M$ T_o	$P_d + D + D_m$ T_o
	B (Severe, Upset)	$P_o + D + D_m + E_o$ $P_o + D + D_m + W + R$ $P + D + D_m + F$ T_b	$P_o + D + D_m + E_o$ $P_o + D + D_m + W + R$ $P + D + D_m + F$ T_b	$P_o + D + D_m + E_o$ $P_o + D + D_m + W + R$ $P + D + D_m + F$ T_b
	D (Abnormal, Extreme Faulted)	$P + D + D_m + W_t$ $P_o + D + D_m + Y_m$ $P_o + D + D_m + A_c$ $P_a + D + D_m + A_D$ $P_a + D + D_m + A$	N/R	N/R

Nomenclature:

D	=	Dead Loads	W_t	=	Tornado Loads Including Missile Effects
L	=	Live loads	V_m	=	Malevolent Vehicle Assault Loads
T_o	=	Normal Operating Thermal Expansion Loads	A_c	=	Aircraft Crash Loads
T_b	=	Upset Thermal Expansion Loads	A_D	=	Design Basis Accident Loads
T_a	=	Accident Thermal Loads	A	=	Other Accident Loads
E_o	=	OBE or ½ SSE Seismic Loads	P_d	=	Design Pressure
E'_o	=	Seismic Loads per Table 2 For RW-IIb Components	P_b	=	Maximum Upset Pressure
E''_o	=	Seismic Loads per Table 2 For RW-IIc Components	P_o	=	Normal Operating Pressure
W	=	Wind Load	P_a	=	Applicable Accident Pressure
R	=	Precipitation Loads (Rain, Snow)	D_M	=	Design Mechanical Loads
F	=	Flood Loadings			

Footnotes:

1 Component support structures include supporting elements for piping, tanks, vessels pumps, heat exchangers, conduits, cable trays, HVAC systems, fire protection systems, etc.

2 For most pressure-retaining components, primary and secondary stresses are evaluated separately to separate criteria. The design code of record is the controlling document in the establishment of the primary and secondary stress combination and evaluation methods.

Table 4 - SSC Design Capacity Criteria				
Code or Standard	**Service Level**	**Capacity Criteria**		
		RW-IIa	**RW-IIb**	**RW-IIc**
ACI-349	A, B, D	Load Factors and Capacity Criteria per ACI-349 as modified by Regulatory Guide 1.142	N/A	N/A
ACI-318	A, B, D	Load Factors and capacity criteria per ACI-349 as modified by Regulatory Guide 1.142. All other design per ACI-318 criteria	Load factors and capacity criteria per ACI-349 as modified by Regulatory Guide 1.142. All other design per ACI-318 criteria.	Load factors and capacity criteria per ACI-349 as modified by Guide 1.142. All other design per ACI-318 criteria.
AISC-N690	A	Capacity criteria Table Q 1.5.7.1 for normal loads.	Capacity criteria Table Q 1.5.7.1 for normal loads.	Capacity criteria Table Q 1.5.7.1 for normal loads
	B	Capacity criteria 1.33 times that for Level A loads	Capacity criteria 1.33 times that for Level A loads	Capacity criteria 1.33 times that for Level A loads
	D	Capacity criteria per Table Q.1.5.7.1 for Abnormal Extreme Loads	N/R	N/R
AISC-ASD	A	Capacity Criteria per "Specification for Structural Steel Buildings Allowable Stress Design and Plastic Design, Part 5,"chapters A-M	Capacity Criteria per "Specification for Structural Steel Buildings Allowable Stress Design and Plastic Design, Part 5,"chapters A-M	Capacity Criteria per "Specification for Structural Steel Buildings Allowable Stress Design and Plastic Design, Part 5,"chapters A-M
	B	Capacity Criteria 1.33 times that for Level A loads.	Capacity Criteria 1.33 times that for level A loads.	Capacity Criteria 1.33 times that for level A loads.
	D	Capacity Criteria per "Specification for Structural Steel Buildings Allowable Stress Design and Plastic Design, Part 5,"chapters A and N	N/R	N/R

Table 4 - SSC Design Capacity Criteria (continued)				
Code or Standard	**Service Level**	**Capacity Criteria**		
		RW-IIa	**RW-IIb**	**RW-IIc**
AISC LRFD	A,B,D	Load factors and capacities per LRFD specifications for structural steel buildings.	Not required	Not required
AISI CFSDM	A	Capacity criteria per "Specification for the Design of Cold Formed Steel Structural Members"	Capacity criteria per "Specification for the Design of Cold Formed Steel Structural Members"	Capacity criteria per "Specification for the Design of Cold Formed Steel Structural Members"
	B	Capacity criteria 1.33 times Level A	Capacity criteria 1.33 times Level A	Capacity criteria 1.33 times Level A
	D	Capacity criteria 1.6 times Level A	Not required	Not required
ANSI/ASME B31.3	A	B31.3 Design Load Capacities	B31.3 Design Load Capacities	B31.3 Design Load Capacities
	B	B31.3 Occasional Load Capacities	B31.3 Occasional Load Capacities	B31.3 Occasional Load Capacities
	D	1.8 Times B31.3 Occasional Load Capacities	Not required	Not required
ASME BPVC, Section VIII, Div. 1 or Div. 2	A	ASME BPVC, Section VIII, Div. 1 or Div. 2 Design Capacities	ASME BPVC, Section VIII, Div. 1 or Div. 2 Design Capacities	ASME BPVC, Section VIII, Div. 1 or Div. 2 Design Capacities
	B	Capacity criteria 1.2 Times Level A criteria	Capacity criteria 1.2 Times Level A criteria	Capacity criteria 1.2 Times Level A criteria
	D	Capacity criteria 1.8 times Level A criteria	Not required	Not required
SMACNA Stds.(1)	A	SMACNA Design Criteria	SMACNA Design Criteria	SMACNA Design Criteria
	B	SMACNA Design Criteria	SMACNA Design Criteria	SMACNA Design Criteria
	D	1. Duct support members to meet capacity criteria for AISI SG-673 or AISC-ASD for Level D Loads. 2. Ducting stresses to be less than the material yield stress and limited to 2/3 critical buckling.	Not required	Not required
NFPA-13[1]	A	NFPA Design Criteria	NFPA Design Criteria	NFPA Design Criteria

Table 4 - SSC Design Capacity Criteria (continued)				
Code or Standard	**Service Level**	**Capacity Criteria**		
		RW-IIa	**RW-IIb**	**RW-IIc**
	B	NFPA Design Criteria for Earthquake and Wind Loads	NFPA Design Criteria for Earthquake and Wind Loads	NFPA Design Criteria for Earthquake and Wind Loads
	D	3. Support members to meet capacity criteria for AISI SG-673 or AISC-ASD for Level D Loads 4. Piping Stresses to meet the B31.3 Level D Capacity Criteria	N/R	N/R
ANSI/NEMA STDS (Cable Trays/Conduit)	A	ANSI/NEMA Design Criteria for Normal Loads	ANSI/NEMA Design Criteria for Normal Loads	ANSI/NEMA Design Criteria for Normal Loads
	B	ANSI/NEMA Design Criteria for Wind and Seismic Loads	ANSI/NEMA Design Criteria for Wind and Seismic Loads	ANSI/NEMA Design Criteria for Wind and Seismic Loads
	D	5. Support members to meet capacity criteria for AISI-CFSDM or AISC-ASD for Level D Load 6. Trays and members to meet the capacity criteria for AISI-CFSDM for Level D Loads	Not required	Not required
Pumps (API Series Stds)	A	For Design Criteria, API 610, API 674, API 675	For Design Criteria, API 610, API 674, API 675	For Design Criteria, API 610, API 674, API 675
	B	ASME QME-1 1997	ASME QME-1 1997	ASME QME-1 1997
	D	ASME QME-1 1997	Not required	Not required
Tanks	A	API-620, API-650	API-620, API-650	API-620, API-650
	B	Capacity Criteria per ASME-BPVC - Section III, NC-3800, NC-3900 for Level B loads. All other Design per API Criteria.	Capacity Criteria per ASME-BPVC - Section III, NC-3800, NC-3900 for Level B loads. All other Design per API Criteria.	Capacity Criteria per ASME-BPVC - Section III, NC-3800, NC-3900 for Level B loads. All other Design per API Criteria.

Table 4 - SSC Design Capacity Criteria (continued)				
Code or Standard	**Service Level**	**Capacity Criteria**		
		RW-IIa	**RW-IIb**	**RW-IIc**
	D	Capacity Criteria per ASME-BPVC - Section III, NC-3800, NC-3900 Level D Loads. All other Design per API Criteria.	Not required	Not required

Footnotes for Table 4:

1 For Level A and B Loads, the Design Criteria is primarily a "design by rule" approach versus a specific analysis criteria.

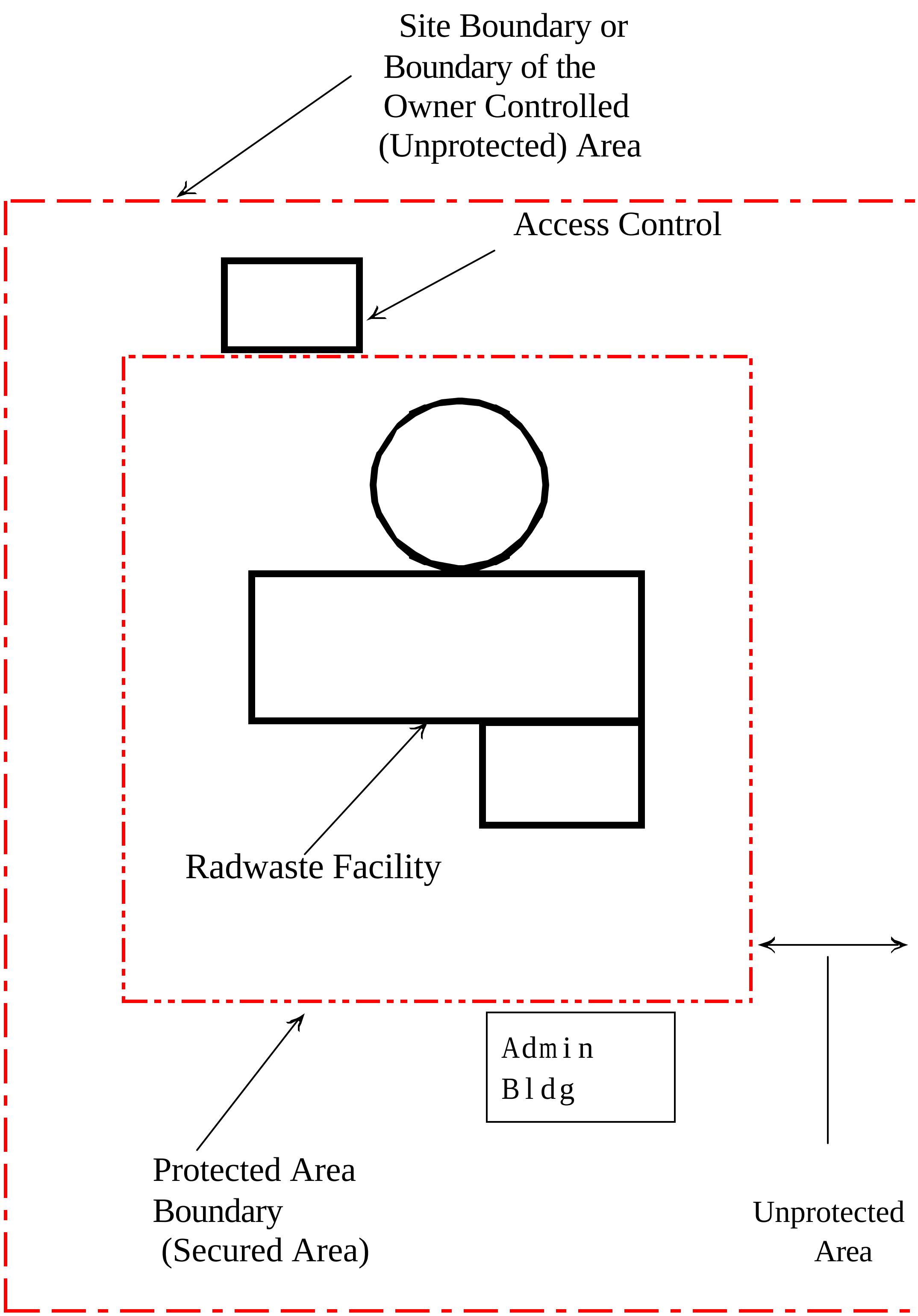

Figure 1 - Informational Schematic Describing
Protected and Unprotected Areas

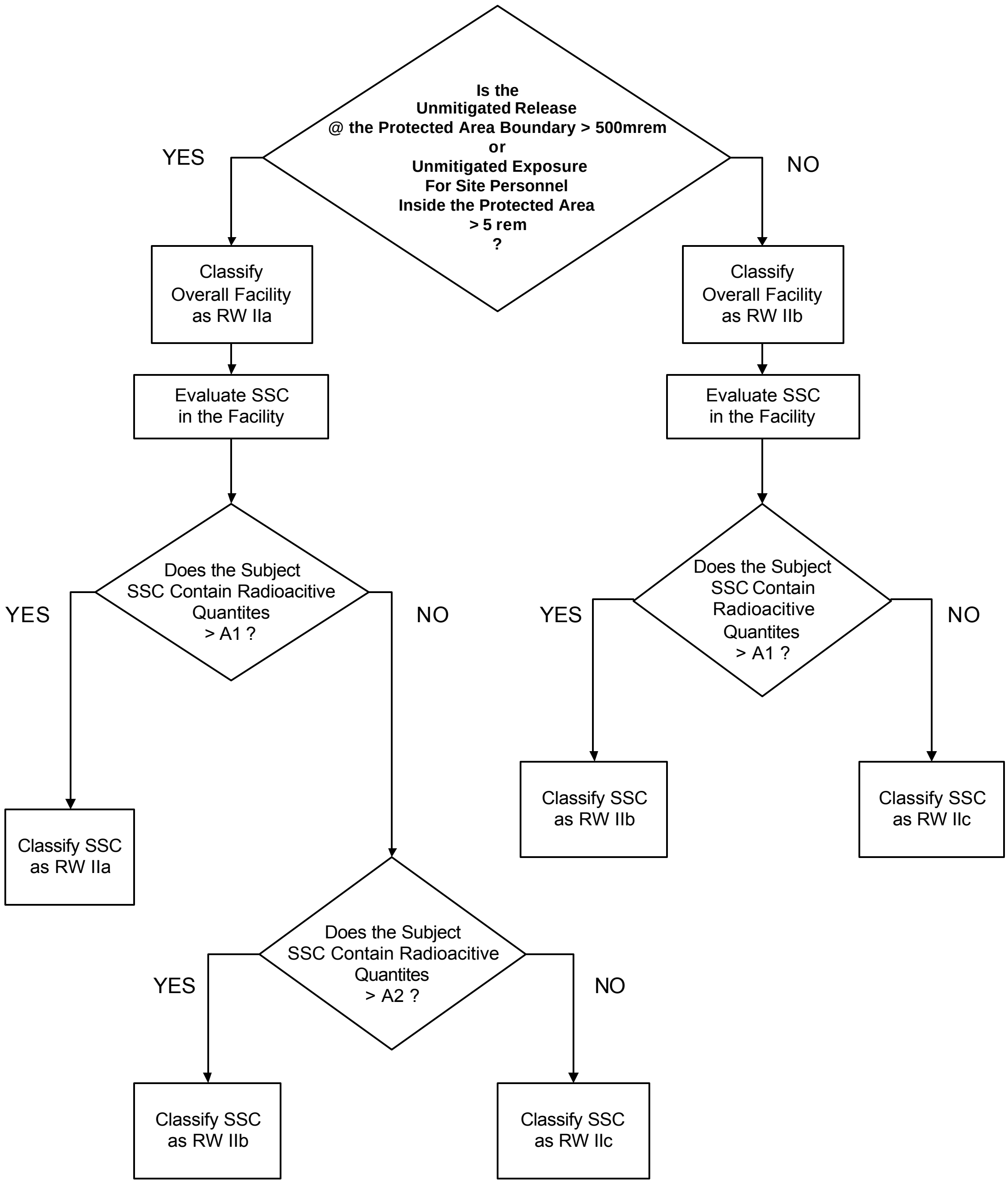

Figure 2 - Flowchart of Safety Classification Process

Appendix A

INDUSTRY CODES AND STANDARDS

American Concrete Institute, ACI-318, "Building Code Requirements for Reinforced Concrete" (ACI 318-89, Revised 1999), 1999.

American Concrete Institute, ACI-349, "Code Requirements for Nuclear Safety Related Concrete Structures," 1997.

American Institute of Steel Construction, N690 (S327), "Nuclear Facilities, Steel Safety-Related Structures For Design and Fabrication," 1984.

American Institute of Steel Construction, "Manual of Steel Construction Load and Resistance Factor Design," Volumes I and II, 2nd Edition, 1994.

American Institute of Steel Construction, "Specifications for Structural Steel Buildings, Manual of Steel Construction," 2nd Edition, 1995.

American Institute of Steel Construction, "Specifications for Structural Steel Buildings, Allowable Stress Design and Plastic Design, Manual of Steel Construction," 9th Edition, 1993.

American Iron and Steel Institute, SG-673, "Specification for the Design of Cold-Formed Steel Structural Members," August 1986 with December 1989 Addendum.

American Nuclear Society, "Standard for Estimating Tornado and Extreme Wind Characteristics at Nuclear Power Sites, ANSI/ANS 2.3-1983.

American Nuclear Society, "Determining Design Basis Flooding at Power Reactor Sites, ANSI/ANS 2.8-1992.

American Nuclear Society, "Mobile Radioactive Waste Processing Systems," ANSI/ANS 40.37-1993.

American Nuclear Society, "Solid Radioactive Waste Processing System for Light-Water-Cooled Reactor Plants," ANSI/ANS-55.1-1992.

American Nuclear Society, "Gaseous Radioactive Waste Processing Systems for Light Water Reactor Plants," ANSI/ANS-55.4-1993.

American Nuclear Society, "Liquid Radioactive Waste Processing System for Light Water Reactor Plants," ANSI/ANS 55.6-1993.

American Petroleum Institute, 610, "Centrifugal Pumps for Petroleum, Heavy Duty Chemical, and Gas Industry Services," 1995.

American Petroleum Institute, 620, “Design and Construction of Large, Welded, Low-Pressure Storage Tanks, 1990.

American Petroleum Institute, 650, “Welded Steel Tanks for Oil Storage,” 1998.

American Petroleum Institute, 674, “Positive Displacement Pumps-Reciprocating,” 1995.

American Petroleum Institute, 675, “Positive Displacement Pumps-Controlled Volume,” 1994.

American Society of Civil Engineers, 7-95, "Minimum Design Loads for Buildings and Other Structures," 1995.

American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, Section II, “Material Specification,” 1999.

American Society of Mechanical Engineers, Boildr and Pressure Vessel Code, Section III, “Rules for Construction of Nuclear Power Plant Components, Division 1, Subsection ND Class 3 Components, July 1998 with July 1999 Addenda.

American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, Section VIII, “Pressure Vessels,” 1999.

American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, Section VIII, “Rules for Construction of Pressure Vessel, Division 1,” July 1998 with July 1999 Addenda.

American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, Section VIII, “Rules for Construction of Pressure Vessel, Division 2, Alternative Rules,” July 1998 with July 1999 Addenda.

American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, Section IX, “Welding and Brazing Qualification,” 1999.

American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, B31.3, “Process Piping,” 1999.

American Society of Mechanical Engineers, QME-1-1997, “Qualification of Active Mechanical Equipment Used in Nuclear Power Plants,” December 31, 1997.

American Society for Testing & Materials, A36-00, “Standard Specification for Carbon Structural Steel,” 2000.

American Society for Testing & Materials, A123/A123M-01, “Standard Specification for Zinc (Hot-Dip Galvanized) Coatings on Iron and Steel Products,” 2001.

American Society for Testing & Materials, A500-99, “Standard Specification for Cold-Formed Welded and Seamless Carbon Steel Structural Tubing in Rounds and Shapes,” 1999.

American Society for Testing & Materials, A571-84 (1997, “Standard Specification for Austenitic Ductile Iron Castings for Pressure-Containing Parts Suitable for Low-Temperature Service,” 1997.

American Society for Testing & Materials, A795-97, “Standard Specification for Black and Hot-Dipped Zinc-Coated Welded and Seamless Steel Pipe for Fire Protection Use,” 1997.

American Society for Testing & Materials, B359-98, “Standard Specification for Copper and Copper-Alloy Seamless Condenser and Heat Exchanger Tubes With Integral Fins,” 1998.

American Society for Testing & Materials, B633-98, “Standard Specification for Electrodeposited Coatings of Zinc on Iron and Steel,” 1998.

American Society for Testing & Materials, C1290-00, “Standard Specification for Flexible Fibrous Glass Blanket Insulation Used to Externally Insulate HVAC Ducts,” 2000.

American Society for Testing & Materials, F856-97, “Standard Practice for Mechanical Symbols, Shipboard Heating, Ventilation, and Air Conditioning (HVAC),” 1997.

American Welding Society, D1.1, “Structural Welding Code-Steel,” 17th Edition, 2000.

American Welding Society, D1.3, “Structural Welding Code-Sheet Steel,” 1998.

American Welding Society, D9.1, “Sheet Metal Welding Code,” 1990.

American Welding Society, D10.9, “Specification for Qualification of Welding Procedures and Welders for Piping and Tubing,” 1980.

International Conference of Building Officials, “Uniform Building Code,” 1997.

National Electrical Manufacturers Association, Publication Number TC2, “Electrical Polyvinyl Chloride(PVC) Tubing and Conduit,” 1998.

National Electrical Manufacturers Association, Publication Number VE1, “Metal Cable Tray Systems,” 1996.

National Fire Protection Association, NFPA 13, “Installation of Sprinkler Systems,” 1999.

National Fire Protection Association, NFPA 14, “Standard for the Installation of Standpipe Fire Protection, Private Hydrant, and Hose Systems,” 2000.

Sheet Metal and Air Conditioners Contractor National Association, “Seismic Restraint Manual Guides for Mechanical Systems,” 2nd Edition, 1998.

Standard Uniform Building Code, International Conference of Building Officials, 1997.

Tubular Exchanger Manufacturers Association, "Standards of the Tubular Exchanger Manufacturers Association, Eighth Edition," 2000.

The Codes and Standards are available from:

American Concrete Institute (ACI), Box 19150, Redford Station, Detroit, MI 48219.

American Institute of Steel Construction (AISC), One E. Wacker Drive, Suite 3100, Chicago, IL 60601-2001.

American Iron and Steel Institute (AISI),1101 17th Street, NW, Washington, DC 20036.

American Nuclear Society (ANS), 555 N. Kensington Avenue, La Grange Park, IL 60525.

American Petroleum Institute (API), 1220 L Street, NW, Washington, DC 20005.

American Society of Mechanical Engineers (ASME), 345 East 47th Street, New York, NY 10017.

American Society for Testing & Materials (ASTM), 100 Barr Harbor Drive, West Conshohocken, PA 19428-2959.

American Welding Society (AWS), 550 NW LeJeune Road, Miami, FL 33126.

International Conference of Building Officials, 5360 Workman Mill Road, Whittier, CA 90601-2798. (www.icbo.org)

National Electrical Manufacturers Association (NEMA), 1300 N. 17th Street, Rosslyn, VA 22209.

National Fire Protection Association (NFPA), Inc., Battery March Park, Quincy, MA 02269.

Sheet Metal and Air Conditioners Contractor National Association (SMACNA), 4201 Lafayette Center Drive, Chantilly, VA 20153-1230.

Tubular Exchanger Manufacturers Association (TEMA), 25 N. Broadway, Tarrytown, NY 10591.

REGULATORY ANALYSIS

1. STATEMENT OF PROBLEM

Revision 1 of Regulatory Guide 1.143, “Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants,” was issued in October 1979. This guide provided design guidance acceptable to the NRC staff related to seismic and quality group classification and quality assurance provisions for radioactive waste management structures, systems, and components. Further, it describes provisions for controlling releases of liquids containing radioactive materials, e.g., spills or tank overflows, from all plant systems outside reactor containment. Regulatory Guide 1.143 encompassed the design of buildings, structures, systems, and components and referred to several design and construction codes and standards, such as American National Standards Institute (ANSI) N197-1976, ANSI N199-1976, American Nuclear Society (ANS) ANS 55.1-1979, ANS 55.4-1979, American Concrete Institute ACI-318-1977, and American Institute of Steel Construction AISC-1969.

These references are now obsolete or have been superseded by newer ANSI and ANS radioactive waste facility design standards. ANS has since issued ANS-55.1-92, ANS-55.4-93, and ANS-55.6-93, which are the industry consensus standards currently applicable to the overall design of radioactive waste facilities. In addition, several other referenced codes such as “Building Code and Commentary,” ACI-318-77; or “Specification for the Design, Fabrication and Erection of Structural Steel for Buildings,” AISC-1969, have been updated and modified since Revision 1 of Regulatory Guide 1.143 was issued. Also, there has been increased understanding of, and corresponding changes in relation to, radiation exposure and monitoring and quality assurance needs for the design and construction of radioactive waste facilities and the associated systems, structures, and components.

The Operating Basis Earthquake (OBE), as was used in Revision 1 of Regulatory Guide 1.143 as the design basis, creates further difficulties. In 1997, the NRC staff revised 10 CFR 100.23 and added Appendix S to 10 CFR Part 50 that essentially state that, if the review level earthquake (OBE) is defined as less than 1/3 of the safe-shutdown earthquake (SSE), no explicit design analysis for the OBE level earthquake will be required. In other words, the revised criteria have effectively eliminated the OBE as a design basis seismic event. In recent staff licensing actions, the Standard (Advanced) Reactor Designs used only a SSE event as the design basis, consistent with the methodology in the recent revision of 10 CFR 100.23 and the addition of Appendix S to 10 CFR Part 50. Thus, Revision 1 of Regulatory Guide 1.143 was almost not usable for standard reactor designs.

The staff maintains that recommendations based on the latest editions of the design and construction Standards and Codes mentioned above and references to current quality assurance standards and NRC regulations provide a means to achieve better evaluation of radioactive waste management systems, structures, and components installed in light water-cooled nuclear power plants.

2. OBJECTIVE

The objective of the regulatory action is to update NRC guidance on the design, construction, and quality assurance of radioactive waste management systems, structures, and components installed in light-water-cooled nuclear power plants.

3. ALTERNATIVES AND CONSEQUENCES OF PROPOSED ACTION

3.1 Alternative 1 - Do Not Revise Regulatory Guide 1.143

If Regulatory Guide 1.143 were not revised, licensees would continue to rely on the current version of Regulatory Guide 1.143 with references from the late 1960s and mid-1970s. The staff acknowledges that many licensees who are presently involved in the design of radioactive waste management systems, structures, and components installed in light-water-cooled nuclear power plants, as a matter of practice, already rely on more recent editions of ANSI and ANS radioactive waste facility design standards and ACI and AISC codes.

3.2 Alternative 2 - Update Regulatory Guide 1.143

The NRC staff has identified the following consequences associated with adopting Alternative 2.

3.2.1 Licensees will use the latest consensus standards available, thereby improving design, evaluation, and quality assurance of radioactive waste management systems, structures, and components. The staff views the latest standards as improved because they incorporate the latest technology and knowledge on the subject.

3.2.2 Regulatory efficiency will be improved by reducing uncertainty as to what is acceptable and by encouraging consistency in the design, evaluation, and quality assurance of radioactive waste management systems, structures, and components. The benefits to both the NRC and industry will be to the extent this occurs. An updated regulatory guide would facilitate NRC review because licensee submittals should be more predictable and consistent analytically. Similarly, licensee's adherence to the latest consensus standards should benefit licensees by reducing the likelihood for follow-up questions and possible revisions to licensees' plans.

3.2.3 An updated regulatory guide could result in cost savings for both the NRC and industry. From the NRC's perspective, relative to the baseline, NRC will incur one-time incremental costs to develop the regulatory guide for public comment and to finalize the regulatory guide. However, the NRC should also realize cost savings associated with the review of licensee submittals. In the staff's view, the continuous and on-going cost savings associated with these reviews should more than off-set this one-time cost.

On balance, it is expected that industry would realize a net savings, as their one-time incremental cost to review and comment on a revised regulatory guide would be more than compensated for by the efficiencies (e.g., reduced follow-up questions and revisions) associated with each licensee submittal.

3.2.4 The use of industry consensus standards that are already being used by licensees would enhance the continued use of the guidance contained in ANS-55.1-92, ANS-55.4-93, and ANS-55.6-93, thereby avoiding costs related to a "new" agency-prepared standard. This approach would also comply with the Commission's directive that standards developed by consensus bodies be utilized per Public Law 104-113, "National Technology and Transfer Act of 1995."

4. CONCLUSION

Based on this regulatory analysis, it is recommended that the NRC revise Regulatory Guide 1.143. The staff concludes that the proposed action will reduce unnecessary burden on both the NRC and its licensees, and it will result in an improved process for the design, evaluation, and quality assurance of radioactive waste management systems, structures, and components. Furthermore, the staff sees no adverse effects associated with a revision to Regulatory Guide 1.143.

BACKFIT ANALYSIS

The regulatory guide does not require a backfit analysis as described in 10 CFR 50.109(c) because it does not impose a new or amended provision in the NRC's rules or a regulatory staff position interpreting the NRC's rules that is either new or different from a previous applicable staff position. In addition, this regulatory guide does not require the modification or addition to systems, structures, components, or design of a facility or the procedures or organization required to design, construct, or operate a facility. Rather, a licensee or applicant may select a preferred method for achieving compliance with a license or the rules or the orders of the Commission as described in 10 CFR 50.109(a)(7). This regulatory guide provides an opportunity to use industry-developed standards if that is the method preferred by the licensee or applicant.