

Re-evaluation of Regulatory Guidance Provided in Regulatory Guides 1.142 and 1.143

Stevenson and Associates

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Office of Nuclear Regulatory Research
Washington, DC 20555-0001**



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Re-evaluation of Regulatory Guidance Provided in Regulatory Guides 1.142 and 1.143

Manuscript Completed: July 1999
Date Published: August 1999

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NRC Job Code W6540**



Abstract

This report describes the technical basis and recommends changes in the Nuclear Regulatory Commission's (NRC's) criteria now used in the design of Nuclear Concrete Structures other than Containment and the design of Radioactive Waste Management Systems, Structures and Components. The authors judged that the best method by which the conclusions of this report can be implemented are through the issuance of revisions to existing Regulatory Guides. There have been several developments since the issuance of the last revision of Regulatory Guides 1.142 and 1.143 (1981 and 1979 respectively) which provides guidance for the design of these items. ACI 349, which presents the code requirements for nuclear safety related concrete structures, has had two major revisions since the last issue of the Regulatory Guides. New advanced reactor design concepts have been put forward that include unique design attributes that need to be considered. The Department of Energy (DOE) has been active in writing standards for the design of radioactive waste management systems. There have also been numerous changes to the Code of Federal Regulations since the issuance of Regulatory Guides 1.142 and 1.143. This report discusses these advancements and changes with respect to the scope of the Regulatory Guides and includes detailed recommended changes to Revision 1 of Regulatory Guide 1.142 and 1.143.

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1.0 Program Background And Overview

1.1 Introduction

The two Regulatory Guides (1.142, 1.143) considered in this review are somewhat diverse in their scope and focus. Regulatory Guide 1.142 ["Safety Related Concrete Structures For Nuclear Power Plants" (other than Reactor Vessels and Containments)] deals primarily with the design of Safety Related Nuclear Concrete Structures excluding concrete reactor vessels and concrete containments. This is accomplished by the application of American Concrete Institute (ACI) Standard ACI-349-76 (for design) and American Nuclear Society (ANSI) Standard N45.2.5 (for quality assurance requirements). Further Regulatory Guide 1.142 enumerates USNRC exceptions to these focal standards. Since issuance of Regulatory Guide 1.142 the ACI has published a 1985 Edition, a 1990 Edition with several erratas and addenda, and the latest release-ACI-349-97. Further, ANSI N45.2.5 is now essentially obsolete with quality assurance requirements for nuclear facility construction now found in ASME NQA-1. Per direction of the USNRC, ACI-349 Appendix B is being reviewed and evaluated under a separate research effort. Therefore, its consideration was specifically excluded from this review effort.

Regulatory Guide 1.143 ["Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light Water Reactors"] has a much more diverse scope than Regulatory Guide 1.142. It covers all aspects of the design of liquid, gaseous, and solid radioactive waste facilities and the associated Systems, Structures, and Components (SSC). It is important to note that the current Regulatory Guides 1.143 (Revision 1, October 1979), in Section B has several exclusions from its scope, specifically spent nuclear fuel and its associated storage facilities. In performing the Regulatory Guide 1.143 review these exclusions were applied. Therefore, this Regulatory Guide encompasses the design of buildings, structures, systems, and components and the associated range of Design and Construction Codes and Standards. The Regulatory Guide references obsolete and superseded American National Standards Institute (ANSI) and American Nuclear Society (ANS) radioactive waste facility design standards. ANS has recently issued ANS-55.1-92, ANS-55.4-93 and ANS-55.6-93 which are the current applicable industry consensus standards applicable to the overall design of radioactive waste

facilities. In addition, several other referenced standards such as ACI-318, American Institute of Steel Construction (AISC) Allowable Stress Design, etc. have been updated and modified since the issuance of this Regulatory Guide. Finally, the use of the OBE as the design basis creates further difficulties. The Standard (Advanced) Reactor Designs use only a SSE event as the design basis, consistent with the methodology in the recent revision to 10CFR100.23 and the addition of Appendix S to 10CFR50.

Due to the significant differences between the scope and focus of these two standards the reviews of the reference and background material were conducted as two separate activities.

1.2 Program Objective

The main objective of this program is to determine and evaluate the need for revisions to Regulatory Guides 1.142 and 1.143 with the goal of providing the basis for updating the subject Regulatory Guides.

1.3 Program and Report Overview

This section provides an overview of the process used in conducting the background reviews necessary to assemble sufficient data to determine the need for changes to Regulatory Guides 1.142 and 1.143. It also identified the major topics which were considered by these reviews.

1.3.1 Regulatory Guide 1.142 Review Effort

This section describes the general overall approach used in defining the suggested modifications to Regulatory Guide 1.142. The focus of the review is Standard ACI-349. However, other related and referenced standards as given in Table 1.3.1 were also considered in the review.

S&A reviewed all revisions, addenda and errata of ACI-349 from 1976 up to and including ACI-349-97, delineated all the changes that have occurred in that time frame. In addition S&A reviewed the current status of Quality Assurance in relation to ASME NQA-1 and ACI-349.

It was important to ensure that the Regulatory Guide when modified is consistent with the licensing basis of

the new Standard (Advanced) Reactor Designs. Understanding and delineating the Standard Reactor Design Basis will be accomplished in two steps. First the current Safety Analysis Reports (either Final SSAR's or PSAR's) for the three reactor designs shown in Table 1.3.2 were reviewed to determine the application, implementation methods and referenced Codes and Standards applied to the design of safety related concrete structures. The basis for selection of these three Reactor Designs is discussed in Section 2.2. Also, as identified in NUREG/CR-6358, these Standard (Advanced) Reactor Designs contain unique design features which may not be completely addressed in ACI-349. These features and a proposed method to address them was identified as part of the review of the Standard (Advanced) Reactor Design basis review. The results of these reviews were tabulated for use in developing the final proposed changes to the Regulatory Guides to address the issues related to Advanced Reactor designs.

As the modified Regulatory Guide may have applicability to existing operating commercial nuclear power facilities it was also important to insure the Regulatory Guide maintains consistency with the licensing basis of the more recent vintage operating Nuclear Power Plants. To insure that this consistency is maintained, S&A reviewed the licensing basis of ten operating Nuclear Power Plants to determine the criteria and methodology which was used for the design and construction of safety related concrete structures in the operating plants. This will be accomplished by reviewing the applicable sections of the final FSAR's and final SER's. The specific plants which were reviewed are given in Table 1.3.3. The basis for selecting these plants is discussed in Section 2.5.

Three additional areas which required special additional attention and review were:

- The Seismic Design Basis
- Malevolent Vehicle Design Basis
- BWR Suppression Pool Applied Dynamic Loads

The USNRC has issued revisions to 10CFR100.23 and 10CFR50, Appendix S which essentially states 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. This issue is discussed in greater depth in Section 2.3. This is the design basis of several of the Advanced Reactor Designs. Currently ACI-349 explicitly requires the consideration of the OBE as a design basis load, as does the licensing basis of most (if not all) operating

Nuclear Power Plants. Therefore, in formulating final proposed changes to the Regulatory Guide a methodology must be developed which addresses both the New Advanced Reactor Designs and the operating plant licensing basis.

The USNRC via 10CFR73 issued design and retrofit requirements for the operating plants to protect against the possibility of a Malevolent Vehicle Assault. These requirements were issued in 1994. Due to the recent issue of these requirements they are not currently considered in Regulatory Guide 1.142 nor are they discussed in ACI-349. Using the recent regulations, in conjunction with recent experience in design and analyzing safety related concrete structures for this loading, proposed criteria for inclusion of this requirement into the Regulatory Guide is developed.

A major aspect of the design of safety related concrete structures in recent vintage BWR's (5 and 6, ABWR, SBWR) is the need to design for suppression pool dynamic loads which result from Main Steam Safety/Relief Valve (SRV) operation and a postulated Loss of Coolant Accident (LOCA) arising from a postulated Main Steam and/or Recirculation Loop piping break. The current revision of the Regulatory Guide states that this item will be addressed on a case by case basis. S&A reviewed this issue to determine what, if any additional guidance is necessary either directly in the standard or by reference to other appropriate regulatory guidance.

1.3.2 Regulatory Guide 1.143 Review Effort

The review effort for Regulatory Guide 1.143 is more diverse than that of Regulatory Guide 1.142 because this guide encompasses the design of radioactive waste management facilities, and Systems, Structures and Components (SSC) in light water reactor Nuclear Power Plants. Therefore, it includes solid, liquid and gaseous waste management systems. The Regulatory Guide was issued in 1979 and since that time the baseline American Nuclear Society (ANS) standards for radioactive facilities and the reference design standards for the SSC in these facilities have been modified and updated significantly. Also, there has been increased understanding and corresponding regulation changes in relation to radiation exposure and monitoring. Finally, there have been technology advances in the criteria and the graded approach applied to waste management systems especially within the US Department of Energy (DOE). Therefore, to comprehensively update this Regulatory Guide

required an extensive review of the design basis criteria and all the associated facility, structures, systems and component design basis standards.

Given the extensive range of Regulatory Guide 1.143, S&A developed a comprehensive design basis criteria for radioactive waste management facilities. These criteria addressed such items as (1) radiation exposure limits to the facility worker, colocated site workers, and the general public, (2) the safety classification of the facility at large and the safety classification of SSC within the facility (using a graded approach), and (3) the appropriate design basis accident scenarios, and natural and external man induced phenomena hazards. These foundation criteria were then modified and evolved into final criteria by consideration of (1) the licensing basis of radioactive waste facilities for the Standard (Advanced) Reactor Designs and recent vintage operating Nuclear Power Plants, (2) the current USNRC criteria for radioactive waste facilities, and (3) the criteria and level of protection (safety) provided by the current industry consensus design and construction standards.

As with Regulatory Guide 1.142, the design criteria for Waste Management Facilities of the three advanced reactor designs listed in Table 1.3.2 were reviewed. This was done to insure the modifications suggested to Regulatory Guide 1.143 address any special issues associated with these advanced reactor designs.

As the modified Regulatory Guide may have applicability to existing operating commercial Nuclear Power Plants, it is also important to insure that the Regulatory Guide provide some consistency with the licensing basis of more recent operating Nuclear Power Plants. As with the review for ACI-349, to insure this occurs, S&A reviewed the licensing basis of the operating Nuclear Power Plants shown in Table 1.3.3 to determine the criteria and methodology which were used for the design and construction of radioactive waste management facilities in these operating plants. This was accomplished by reviewing the applicable sections of the final FSAR's and final SER's for these domestic operating Nuclear Power Plants.

One area which required special attention for radioactive waste facilities is the definition of the seismic design basis. Currently Regulatory Guide 1.143 uses only the OBE as the design basis earthquake for seismic design as discussed in Section 4.3.1. The USNRC has issued revisions to 10CFR100.23 and 10CFR50, Appendix S. In formulating the proposed changes to the Regulatory

Guide, a methodology was developed which addresses both the new Standard (Advanced) Reactor Designs and the operating Nuclear Power Plants licensing basis.

In addition to the seismic design basis issues, the need to consider other types of natural and external man induced hazards in the design of radioactive waste facilities was also considered. Specifically, such events as tornado, strong wind, hurricane, flood, accidental explosion, aircraft crash and malevolent vehicle assault were reviewed and considered. The final criteria tempered with licensing basis information discussed in the proceeding paragraphs were proposed for these items.

The development of foundation design basis criteria into final criteria was considered in industry consensus standards. ANS 55.1, 55.4 and 55.6 Standards listed in Table 1.3-4 were reviewed and considered for use in developing the final design basis criteria.

Once the design basis criteria were developed, the next step in the process was the identification of the industry consensus standards which are suggested for use in the design of the facility structures and SSC within the facility. A final list of applicable standards for Structures (concrete and steel), Systems and Components was developed.

The final step in the process was to determine the Quality Assurance requirements necessary for the design and construction of the radioactive waste management facilities. This review considered application of industry consensus Quality Assurance Standards such as ASME NQA-1, and the regulatory requirements of 10CFR50 Appendix B, 10CFR21, Regulatory Guides and NUREG-0800 (SRP Section 17). The results of this review were used to specify the quality assurance requirements for the design and construction of radioactive waste facilities and the associated SSC.

Table 1.3-1 - Other Standards Considered as Part of the Review for Reg. Guide 1.142	
Standard	Description
ACI-318 -95	Building Code Requirements for Structural Concrete
ANS 6.4 - 1997	Guides on the Nuclear Analysis and Design of Concrete Radiation Shielding for Nuclear Power Plants

Table 1.3-2 - Standard (Advanced) Reactor Designs which were Reviewed as Part of the Program	
NSSS Supplier/Vendor	Reactor Name/Model
Westinghouse (W)	AP 600
General Electric (GE)	Advanced Boiling Water Reactor (ABWR)
ABB/Combustion Engineering	System 80+ (Sys 80+)

Table 1.3-3 - Commercial Operating Plants which were Reviewed as Part of this Program	
Plant	NSSS
Beaver Valley Unit #1 and #2	W - 3 Loop (PWR)
Monticello	GE-Mark I (BWR-4)
Davis Besse	BW - (PWR)
Perry Unit #1	GE - Mark III (BWR-6)
Quad Cities	GE - Mark III (BWR-6)
Vogtle Unit #1 and #2	W-4 Loop (PWR)
Waterford	CE-(PWR)
Palisades	CE-(PWR)
Washington Nuclear Plant-2	GE-Mark II (BWR-5)

Table 1.3-4 - Industry Standards Considered as Part of the Review of Regulatory Guide 1.143	
Standard	Description
ANS 55.1-1992	Solid Radioactive Waste Processing Systems for Lightwater Cooled Reactor Plants
ANS 55.4-1993	Gaseous Radioactive Waste Processing Systems for Lightwater Reactor Plants
ANS 55.6-1993	Liquid Radioactive Waste Management Systems for Processing System for Lightwater Reactor Plants

1.4 Overview of Report Structure

This section discusses the general organizational structure of the report. Section 2.0 discusses general topics applicable to both the Regulatory Guide 1.142 and 1.143 reviews. Section 3.0 provides a discussion of the in-depth reviews conducted for Regulatory Guide 1.142 and Section 4.0 provides an in-depth discussion of the reviews conducted in regards to Regulatory Guide 1.143. Section 5.0 provides both a summary of the suggested modifications to Regulatory Guide 1.142 and 1.143 and also presents detailed suggested changes to the Regulatory Guides. Also, Section 5.0 discusses selected implementation methods for the changes. Section 6.0 presents a summary of the review and the program, and Section 7.0 presents references used in the review.

2.0 General Review of Aspects Common to Both Regulatory Guide 1.142 and 1.143

2.1 General

This section provides background and a general overview of aspects of the review which are common to both Regulatory Guide 1.142 and 1.143. The direct applicability and effect of these items on Regulatory Guide 1.142 and 1.143 will be discussed in depth in Sections 3.0 and 4.0.

2.2 Advanced Reactor Design Review

Adams and Stevenson in NUREG/CR-6358 (7.6-[1]) conducted extensive review of the Applicability of Industry Consensus Codes and Standards to Design Advanced Reactors. Specific reactor designs considered in this study were:

- Westinghouse Electric Corporation (W) - AP600 [AP600]
- ASEA Brown Boveri/Combustion Engineering (ABB/CE) - System 80⁺ [Sys. 80⁺]
- General Electric (GE) - Advanced Boiling Water Reactor [ABWR]
- General Electric (GE) - Simplified Boiling Water Reactor [SBWR]
- United States Department of Energy (DOE)/General Atomics (GA) - Modular High Temperature Gas Cooled Reactor [MHTGR]
- ASEA Brown Boveri (ABB) - Process Inherent Ultimate Safety Reactor [PIUS]
- United States Department of Energy (DOE)/General Electric (GE) - Power Reactor Innovative Small Module [PRISM]
- Atomic Energy of Canada Limited (AECL) - Candu-3U [CANDU-3U]

As part of that review, necessary changes to ACI-349 were identified to (a) address unique design features of the Advanced Reactor Designs and (b) to address deficiencies or limitations of the Code. As part of that effort however, very little direct review was conducted relative to review of standards which may be implemented in Regulatory Guide 1.143.

Since the issuance of NUREG/CR-6358, the licensing activity of several of the Advanced Reactor Designs has

been significantly reduced. Currently, the ABB/CE-Sys 80⁺ and GE/ABWR have received 10 CFR Part 52 licenses, and the W-AP600 design is still actively pursuing a 10 CFR Part 58 license. As of the writing of this report, there is no licensing activity for the remaining reactor designs.

Relative to Regulatory Guide 1.142, for the three active advanced reactor designs (GE-ABWR, ABB/CE Sys. 80⁺, W-AP600), the information provided in NUREG/CR-6358 was updated to reflect the final SSAR's (7.1-[2], 7.1-[4]) and the SER's (7.1-[3], 7.1-[5], 7.1-[6]) for the GE ABWR and ABB/CE Sys. 80⁺ and the latest revision of the SSAR (7.1-[1]) was reviewed for the Westinghouse AP600. For Regulatory Guide 1.143 an in depth review of the licensing commitments of the same three active advanced reactor designs was conducted.

2.3 Revised OBE Criteria

Recent changes (7.2-[10]) in 10CFR50 Appendix S and Regulatory Guide 1.165 (7.2-[11]) have resulted in a change in the licensing of advanced nuclear power facilities for seismic (earthquake) events. The major change which impacts both Regulatory Guide 1.142 and 1.143 is that the final regulation allows the value of the OBE to be set at (i) one-third or less of the SSE, where OBE requirements are satisfied without an explicit response or design analyses being performed, or (ii) a value greater than one-third of the SSE, where analysis and design are required. The applicant may voluntarily select an OBE value at some higher fraction of the SSE (above 1/3) to avoid plant shutdowns. However, if an applicant selects an OBE value at a fraction of the SSE higher than one-third, a suitable analysis shall be performed to demonstrate that the requirements associated with the OBE are satisfied.

The above criteria are only applicable to site applications received on or after the effective date of the final regulations which is December 11, 1996. Therefore, it is not applicable to existing nuclear power plant facilities. However, it is not clear that it would not be applicable to new facilities at existing nuclear power plant sites.

The first effect of this regulation is that Regulatory Guide 1.142 (ACI-349) must address at least three unique design

criteria for the seismic loads:

1. Design for five seismic events at an OBE level of 1/2 SSE and design for a single seismic event (Operating Plants).
2. Design for a seismic event having an OBE level greater than 1/3 SSE up to a level equal to the SSE and design for a single SSE event (New Applicants).
3. Set the OBE level at 1/3 SSE and design only for an SSE level. (New Applicants).

The second effect is that Regulatory Guide 1.143 currently only requires radwaste management design for an OBE event. Therefore a modified design criteria must be established for nuclear power plants which could be licensed after December 11, 1996.

2.4 Quality Assurance Requirements

The ANSI/ANS N45.2.5 Standard Reference in Regulatory Guide 1.142 is no longer supported by ANSI or ANS. Current Quality Assurance requirements are provided by 10CFR50 Appendix B. Recently (1995) the ASME has updated and revised NQA-1 (7.5- [9]). The revised Standard provides revised requirements for quality assurance and consolidates NQA-1 and NQA-2 into one document. Therefore, necessary quality assurance requirements need to be reviewed in reference to both Regulatory Guide 1.142 and 1.143 and the potential application of ASME NQA-1.

2.5 Commercial Operating Plant Review

As the modified Regulatory Guides may have applicability to existing operating commercial nuclear power facilities, it is also important to insure that these maintain general consistency with the licensing basis of operating commercial Nuclear Power Plants. To insure this occurs, S&A reviewed the licensing basis of the operating Nuclear Power Plants given in Table 1.3.3 both in regards to Regulatory Guide 1.142 and 1.143. This was accomplished by reviewing the applicable sections of the final FSAR's and final SER's. The plants were selected to provide a cross section of the operating plants both in terms of plant vintage and plant type. The

following mix was applied in selecting the operating plants:

GE-BWR-4 Mark I Containment Design
GE-BWR-5 Mark II Containment Design
GE-BWR-6 Mark III Containment Design
Westinghouse PWR (3 Loop Plant)
Westinghouse PWR (4 Loop Plant)
Combustion Engineering, PWR
Babcock and Wilcox, PWR

The selected mix provided a representative sampling of the licensing design basis for operating Nuclear Power Plants. This mix also insured that any unique features (such as BWR building filtered hydrodynamic loadings, LOCA, tornado, and other accident and natural and man induced phenomena hazard loadings) were adequately considered in the development of the suggested modifications to the Regulatory Guides. The results of this review will be summarized for use in determination of the suggested modification to the Regulatory Guide.

3.0 Regulatory Guide 1.142 Investigation

3.1 General

This section provides a detailed summary of the review efforts conducted for Regulatory Guide 1.142. The focus of these reviews was ACI-349 and its implementation in commercial operating nuclear power facilities and possible implementation in future Advanced Reactor Designs. Also discussed are regulatory considerations, Quality Assurance considerations, and Special Issues and Concerns.

3.2 Comparison of ACI-349 Editions

3.2.1 Overview of the Code

The ACI-349 Code provides the minimum requirements for the design and construction of nuclear safety related concrete structures and structural elements for nuclear power generating stations. Safety related structures and structural elements subject to this standard are those concrete structures which support, house or protect nuclear safety class systems or components or which are component parts of nuclear safety class systems.

Specifically excluded from this Code are those structures covered by "Code for Concrete Reactor Vessels and Containments," ASME Boiler and Pressure Vessel Code Section III, Division 2 and pertinent General Requirements (ACI-359).

3.2.2 Versions Compared

The version endorsed in the current issue of Regulatory Guide 1.142 is the 1976 version including the 1979 supplement except for Appendix B.

In early 1998, the ACI-349 committee issued an updated version of ACI-349, ACI-349-97. This revised version has significant changes in relation to the ACI-349-90 version. Due to the participation of Dr. Harstead on the ACI-349 Committees, S&A was aware of the format and content of the changes which were incorporated into the 1997 issue of ACI-349. Therefore, the initial reviews for ACI-349 were based on this knowledge of the impending changes. For completeness and continuity, a final review and comparison of ACI-349-90 to the published version

of ACI-349-97 was also conducted.

3.2.3 1976 to 1990 Comparison

Table 3.2-1 presents a detailed comparison of the ACI-349-76 to ACI-349-90. Three levels of changes were identified, (a) Editorial, (b) Technical, but which will not significantly alter the design basis or application of the code and (c) Technical, which could significantly alter a design basis and/or application of the code. These Items are identified as E for Editorial, TI, for Technical but not significant and TS for Technical Significant. If ACI-349-90 is invoked in Regulatory Guide 1.142 the technically significant changes would have little direct impact as they would be directly incorporated.

3.2.4 1990 to 1997 Comparison

Table 3.2-2 presents a detailed comparison of changes between ACI-349-90 and ACI-349-97 (7.5-[1]). The changes are discussed in summary form in the publication by Zalesiak, et. al., (7.7-[1]). These changes, were considered to determine if modification to Regulatory Guide 1.142 should be made to provide consistency with ACI-349-97. The major change is in the change in philosophy; namely, that ACI-349 will follow the latest edition of ACI-318, except where issues related to safety related concrete structures need to be addressed.

Table 3.2-1 - Detailed Comparison of ACI-349 (1976 to 1990)		
Section/ Subsection/ Article/ Subarticle/	Type of Change E = Editorial TI = Technical, not significant TS = Technical, significant	Description of the Change (Including Regulatory Significance)
Front Page	E	Same names, key words, text is slightly changed.
Content	E	Only slight changes made. The appendices A,B,C,D, and E contents are new.
1.3.1	E	Some changes in the text about responsibilities for inspection.
1.3.2	TI	The text is changed. The restriction for temperatures below 40°F or above 95° F is no longer stated.
2.1	E	Some general definitions have been deleted, eg., Engineer, Load, Sustained.
3.1 through 3.6.7.2	E	Slight change in the text and arrangement of paragraphs on admixtures.
3.7.1 through 3.7.4	E	Numbering of paragraphs changed, including new paragraphs on storage and identification of materials.
3.8.1 through 3.8.3	TI	Changes in the referenced standards are made. The impact of potential changes in the referenced standards are not reviewed as part of this effort, but should be reviewed prior to invoking the latest code editions in the SRP.
4.1 through 4.6	TS	Significant changes in the text on concrete quality namely, on field experience, trial mixture water-cement ratio, average strength reduction. (Added Sections 4.4 through 4.6)
4.7.1 through 4.7.4.4	E	The text is rearranged, including new paragraphs on evaluation and acceptance of concrete.
5.1.1	E	Rearrangement of the paragraphs on equipment preparation.
5.2	TI	Some new paragraphs added. No significant technical changes.
5.3	E	A few wording changes.
5.4	E	A few wording changes.
5.5, 5.6, and 5.7	E	A few wording changes.
6.1	E	A few paragraphs are reordered.
6.2	E	Some paragraphs were reworded.
6.3	E	Some paragraphs were reworded.

Table 3.2-1 - Detailed Comparison of ACI-349 (1976 to 1990) (Continued)		
Section/ Subsection/ Article/ Subarticle/	Type of Change E = Editorial TI = Technical, not significant TS = Technical, significant	Description of the Change (Including Regulatory Significance)
6.4	TI	The text is rewritten.
7.2	E	Slight changes in the text. No significance
7.5	TI	Additional text on placing reinforcement is added.
7.6	E	The text and paragraphs are rearranged. Primarily editorial.
7.7	TI	This is Section 7.14 in the old standard. A few minor changes were made.
7.7.3 through 7.7.6	TI	Some rewording of these paragraphs.
7.8	TI	Old section is 7.10. Paragraphs are rearranged. Some text is changed. Old Sections 7.10.3, 7.10.7, 7.10.5 have editorial and content reductions.
7.8.2	E	Section rearranged.
7.10	E	The text is rearranged.
	E	Sections 7.5, 7.6, 7.7 and 7.8 in the old standard are not in the same location and have been folded into other sections.
7.12	TS	New Section on minimum reinforcement.
8.1	TI	Last two paragraphs in the old standard are deleted.
	E	Old Section 8.3 was moved (on evaluation of elasticity modulus for concrete) to Section 8.5.
8.3	E	Title changed.
8.4	E	It is the old Section 8.6.
8.6 through 8.9	E	These Sections are obtained from the rearrangement of old Section 8.5.
9.0	TI	Few new notations are defined. None of significant impact.
9.1	TI	It is rearranged. New notations are provided for liquid pressures, F, and lateral earth pressure, H, was added to all load combinations.
9.3	TS	This section was included in Section 9.2 of the old standard. It refers to Design Strength, (including the reduction factors).

Table 3.2-1 - Detailed Comparison of ACI-349 (1976 to 1990) (Continued)		
Section/ Subsection/ Article/ Subarticle/	Type of Change E = Editorial TI = Technical, not significant TS = Technical, significant	Description of the Change (Including Regulatory Significance)
Table 9.5(a)	TS	This is a new table. It compares deflections based on unfactored load combinations, which were obtained by defining a new factor γ . The values of the displacement limits are different than those given in the old standard. (Replaces Table 9.5(b) of 349-76).
9.5.1.1	TS	First paragraph has a new addition at the end. The 2nd and 3rd paragraphs are new and refer to table 9.5(a).
9.5.1.4	TS	New text with some new requirements for deflections is introduced.
Table 9.5(b)	TS	The minimum thickness required for beams is changed. Replaces Table 9.5(a) of ACI-349-76. The values in the new table are larger than the previous values.
9.5.2.5	TS	The long-time deflection resulting from creep and shrinkage is determined using a factor defined differently than that given in the old standard. The text is also changed.
9.5.3.1 through 9.5.3.3	TS	Text is changed. Minimum thickness is different and is given by Table 9.5(c) rather than by the evaluations of Sections 9-6 to 9-8 from the old document.
9.5.3.4	TS	There are changes to text references for displacement calculations.
9.5.3.5	E	Text editorial changes.
9.5.4.1	TS	A new sentence is added referring to cracked sections.
9.5.5.3 and 9.5.6	TS	These are new Sections. Section 9.5.6 refers to Walls subjected to lateral loads which has to satisfy the same requirements.
10.1	E	The last sentence in the old standard is deleted.
10.2.1	E	An addition to the first paragraph is provided.
10.3.3	TS	There are some wording changes (referring to magnitude of the design load strength). New requirements are included for flexural and compressive design strength.
10.3.5	TS	Includes new requirements for the design axial load strength.
10.6.4 through 10.6.6	TS	New requirements on flexible reinforcement distribution are provided.
10.9.3	TS	The yield strength of spiral reinforcement is limited to 60,000 psi in ACI-349.

Table 3.2-1 - Detailed Comparison of ACI-349 (1976 to 1990) (Continued)

Section/ Subsection/ Article/ Subarticle/	Type of Change E = Editorial TI = Technical, not significant TS = Technical, significant	Description of the Change (Including Regulatory Significance)
10.11.4.1	E	The moment relation is written in small letters.
10.11.5.1	TS	The expression for magnified factored moment M_c is changed.
10.11.5.2	TI	The Euler's formula for critical axial compression force is deleted from the section in ACI-349-90.
10.11.5.4 and 10.11.5.5	TS	New subsection referring to the braced compression member has been added.
10.11.7	TS	New paragraph for bi-axial bending concerning moment magnification is added in ACI-349-90.
10.14.7.3	TI	New paragraph included but with no significant impact.
	E	The last subsection in the old document on walls is moved to another subsection.
11.0	E	Small changes in notations are made, none of which are of significance.
	TI	The old Section 11.1 on Design Loads is removed.
11.1.1.1 through 1.1.3	TS	The shear strength is expressed in force rather than stress. New text is provided including new requirements on shear strength.
11.2	TI	New subsection which is not technically significant is added.
11.3.1 through 11.3.2.3	TI	The text is rewritten in terms of the shear force rather than shear stress.
11.4.1 through 11.4.3	TI	The text is rearranged in terms of shear force rather than shear stresses, as in the old document.
11.5.1	E	This section has been moved. It provides acceptable types of shear reinforcement. In the old standard, it was 11.2.7.
11.5.2	TS	This is a new section on spacing limits for shear reinforcement.
11.5.4	TI	These are new paragraphs which are not technically significant.
11.5.6	TS	This section is rewritten. The shear force rather than the reinforcement area is evaluated in the new version.
11.6.1	TS	New requirements for torsional effects with shear for flanged sections are introduced.

Table 3.2-1 - Detailed Comparison of ACI-349 (1976 to 1990) (Continued)		
Section/ Subsection/ Article/ Subarticle/	Type of Change E = Editorial TI = Technical, not significant TS = Technical, significant	Description of the Change (Including Regulatory Significance)
11.6.1 through 11.6.4	TS	It includes several new paragraphs related to evaluation of torsional effects.
11.6.6.1	TS	A new expression for torsional moment strength is introduced. The new expression is significantly different than criteria in the old standard.
	TS	Some subsections of the old standard are deleted, (11.7.4 and 11.7.5). Expression (11-17) from the old standard corresponds to the equation (11-22) in the new one.
11.6.7.3 through 11.6.7.5	TS	These are changes in criteria for non-prestressed rectangular members.
11.6.8	E	These paragraphs are moved earlier in the standard (in the old version they are in Subsection 11.8.6).
11.6.9.1	TS	This section is the old section 11.8.2, and is different. A new expression for torsional moment strength is given.
11.6.9.2	TI	New addition which is not technically significant.
11.6.9.3	TS	Some changes in the expressions of longitudinal bar required area, (equation (11-25)).
11.6.9.4	TS	New requirement on the magnitude of the allowable torsional moment strength.
11.7.1 and 11.7.2	TI	New text paragraphs are included in these subsections. Shear friction was not included in ACI-349-76.
11.7.4	TS	New test and requirements on shear/friction design are introduced. New expression (11-27).
11.7.6	TI	New requirements are introduced on shear/friction reinforcement.
11.7.8 through 11.7.10	TS	New text including new requirements are introduced on shear/friction.
11.8.2 through 11.8.10	TI	The text is rearranged, the order of paragraphs is modified. Equations in terms of shear forces in ACI-349-90 rather than shear stresses.
11.9	TS	The entire section has significant changes which include new analytical formulas and new requirements. These are significantly different than the old ones.

Table 3.2-1 - Detailed Comparison of ACI-349 (1976 to 1990) (Continued)		
Section/ Subsection/ Article/ Subarticle/	Type of Change E = Editorial TI = Technical, not significant TS = Technical, significant	Description of the Change (Including Regulatory Significance)
11.10.1 through 11.10.8	E	The text is rearranged, the text paragraphs have been reordered. However, the requirements are unchanged.
10.9.1 through 11.10.9.5	TS	The text is slightly changed. In paragraph 11.10.9.5 the ratio $l_w/3$ has been changed from $l_w/5$ in the old document. Some old paragraphs, such as inspection, (11.15.7) are deleted in the new edition.
11.11.1.2	E	Small text changes.
11.11.2	TS	The entire section on design of slab/footing is drastically changed. New text, new analytical formulas, new requirements are introduced for shear force evaluation.
11.11.3 and 11.11.4	TS	New paragraphs and requirements on shear reinforcement are introduced. Significant differences in comparison with the old edition are noted.
11.11.5	TS	A new section on Openings in Slabs introduced.
11.12	TS	This section is around three times larger in the new document. It includes new requirements on moment transfer to columns from slabs. The analytical formulas for shear forces in nonprestressed slabs and at columns of two-way prestressed slabs are new.
12.1	TI	New subsection without significant impact.
12.2 and 12.3	TI	These paragraphs are rearranged but similar to the old 12.5 and 12.6 sections.
12.5 and 12.6	TS	This subsection on standard hooks in tension is significantly changed and references Appendix B of ACI-349.
12.7 through 12.8	TS	This section contains several new requirements on welded deformed wire fabric.
12.15.1 through 12.15.6	TS	These paragraphs contain new inclusions, new paragraphs, in comparison with the old sections 7.6.1 and 7.6.2.
12.16.1 through 12.16.4	TS	This section corresponds to the section 7.7.1 in the old standard. These are significant changes.
12.16.5	E	This section is similar to 7.7.2 in the 1976 code.
12.17	E	This section is similar to 7.10.3 through 7.10.5 in the 1976 code.
12.9.3	E	Only a few words changed in the text, compared with old paragraph 12.11.2.

Table 3.2-1 - Detailed Comparison of ACI-349 (1976 to 1990) (Continued)		
Section/ Subsection/ Article/ Subarticle/	Type of Change E = Editorial TI = Technical, not significant TS = Technical, significant	Description of the Change (Including Regulatory Significance)
12.11.3	E	The new text is included in subsection 12.11.3. This section is placed earlier, (being 12.2.3).
12.13.2.3	TS	New subsection on #5 Bar and D31 wire with web reinforcement.
12.14	E	This section is placed much earlier in the old version of the code.
12.14.3	TS	The text on Welded Splices has some changes in the first paragraph of 12.14.3.2
12.14.3.4	TS	New criteria on Mechanical Connections is included.
12.14.3.5 through 12.14.3.6	TS	New subsections on welding splice design criteria.
12.15	E	This section was placed earlier in the old standard (7.6).
12.18 and 12.19	TS	Significant changes in the text and new additions on Splices of Welded Deformed Wires and Welded Smooth Wire Fabric in tension are included.
13.1.2	TI	New subsection for slabs supported on columns or walls (obtained by changing the old subsection 13.1.6).
13.3.1.2 through 13.3.3.3	TS	New paragraphs and content changes are made. The subsection 13.3.3.2 is similar to the old section 11.12.2 which is placed earlier in the text. Changes are of significance.
13.4.1	TI	New subsection with no significant technical content.
13.4.8.3 and 13.4.8.4	TI	New paragraphs with no significant technical content.
13.6.1.7	TS	New subsection including a limitation on moment distribution for the Direct Design Method.
13.6.3.3	TI	It is a modified version of the old subsection 13.3.3.3; Static moment distribution is now given in a table not through an analytical expression.
13.6.3.5 and 13.6.3.6	TI	New subsections, introducing new requirements for edge beams are introduced.
13.6.5.3 and 13.6.6.1	E	These correspond to the old subsections 13.3.4.4 and 13.3.4.5. Some text changes are made, but no significant technical changes.

Table 3.2-1 - Detailed Comparison of ACI-349 (1976 to 1990) (Continued)		
Section/ Subsection/ Article/ Subarticle/	Type of Change E = Editorial TI = Technical, not significant TS = Technical, significant	Description of the Change (Including Regulatory Significance)
13.7.2.1 through 13.7.2.5	E	Small changes in comparison to the old standard.
13.7.4	TS	New subsections are provided in place of old section 13.4.7.5 (equation 13-5 in the old standard is deleted).
13.7.7.5	TI	New Subsection not technically significant.
14.1 and 14.2	TI	New text paragraphs are introduced on wall design.
14.3	TS	New subsection on minimum reinforcement in walls.
14.4 through 14.7	TS	This section includes new requirements for Walls. Equation 14.1 is changed for evaluation of design axial load.
15.1 through 15.3	E	Includes new text modifications to existing text but none of significance.
15.4	E	Only small editorial changes.
15.5 and 15.6	E	Only a few small editorial changes.
15.7 and 15.8	TS	This section is significantly changed. New requirements on force transfer are introduced.
15.9 and 15.10	TS	The text is significantly changed and includes the new requirements for sloped steeled fittings. Direct Design Method of Chapter 13 is excluded in ACI-349-90.
	TI	The old subsection 16.2.2 is deleted.
16.4.2	TI	New paragraph and requirements are provided for the design details for precast concrete cement.
17.5	E	Slight editorial changes in the text.
17.6.3	E	Text modifications and additions are editorial.
18.4.1	TS	The last paragraph is changed. This paragraph is close to the last paragraph of the old subsection 18.4.1. The other paragraphs have changes referring to permissible stress in concrete.
18.4.2	E	Similar requirements as in the old section 18.4.2. Some changes in the text are noted.
18.5.1	TS	Requirements on stresses in tendons are changed.

Table 3.2-1 - Detailed Comparison of ACI-349 (1976 to 1990) (Continued)		
Section/ Subsection/ Article/ Subarticle/	Type of Change E = Editorial TI = Technical, not significant TS = Technical, significant	Description of the Change (Including Regulatory Significance)
18.7	TS	Significant changes in this section, including new requirements and new analytical expressions on flexible strength.
18.9	TS	It includes significant additional text on Minimum Bonded Reinforcement.
18.10	TS	New text and new requirements are introduced. The expression for negative moment distribution is new.
18.11	E	The text is editorially changed.
18.12 and 18.13	TS	New text including several additions are made for design slab systems and tendon anchorage zones.
18.18.2 and 18.18.3	TS	New text is introduced on pretensioning measurement and force application in bulkheads and for long exposed pretensioning members.
18.19.4	TI	The paragraph is rewritten providing more direction on fatigue.
19.2 and 19.4	TS	The subsections are rewritten.
Appendix A thru E added	TS	New appendices added: A - Thermal Considerations B - Steel Embedments C - Special Provision for Impulsive and Impactive Effects D - SI metric equivalents of U.S. Customary Units E - MKS metric equivalents of U.S. Customary Units

Table 3.2-2 - Detailed Comparison of ACI-349 (1990 to 1997)		
Section/ Subsection/ Article/ Subarticle/	Type of Change E = Editorial TI = Technical, not significant TS = Technical, significant	Description of the Change (Including Regulatory Significance)
1.1	-	No change
1.2	TS	Also includes specifications. Requires additional specific information to be provided on drawings.
1.3	TS	Added additional inspection and reporting.
1.4	E	Editorial changes
1.5	-	No change
2.1	TS	Added some definitions. Reference for operating basis wind related to ASCE-7-93.
3.0	E	Editorial change. Deleted definition of d_b .
3.1	E	Editorial changes
3.2	-	No change
3.3	-	No change
3.4	E	Editorial changes
3.5	TS	Added requirements for low relaxation prestressing wire.
3.6	TS	Added requirements for admixtures to produce flowing concrete, and editorial changes.
3.7	E	Minor editorial changes
3.8	TS	Revised and updated cited standards
4.0	TI	Editorial change. Deleted definition of f_{ct} .
4.1	TS	Moved ACI 349-90 subsection 4.4 to 4.1. Added reference to ASTM C 150, ASTM C 595, ASTM C618 and ASTM C 1240.
4.2	TS	Moved ACI 349-90, Subsection 4.5.1 and 4.5.2 to 4.2.1 and 4.2.2. Changed requirements for special exposure conditions in Table 4.2.2 (Table 4.5.2 in ACI 349-90), added minimum f'_c requirements for special exposure conditions. In Table 4.2.2 (Table 4.5.2 in ACI 349-90) changed "watertight" to "low permeability".
4.3	TS	Moved ACI 349-90 Subsection 4.5.3 to 4.3. Added minimum f'_c requirements for sulfate exposure.

Table 3.2-2 - Detailed Comparison of ACI-349 (1990 to 1997) (Continued)		
Section/ Subsection/ Article/ Subarticle/	Type of Change E = Editorial TI = Technical, not significant TS = Technical, significant	Description of the Change (Including Regulatory Significance)
4.4	TS	Moved ACI 349-90 Subsection 4.5.4 and 4.5.5 to 4.4.1 and 4.4.2. Revised Table for maximum chloride content eliminating requirements for reinforced concrete that is not exposed to chloride in service. Testing procedures for chloride changed from AASHTO T260-841 to ASTM C1218.
5.0	TS	Added definitions for f'_c , f'_{cr} and s .
5.1	TI	Moved ACI 349-90 Subsection 4.1 to 5.1. Added reference to the Chapter 4 durability criteria.
5.2	E	Moved ACI 349-90 Subsection 4.2 to 5.2.
5.3	E	Moved ACI 349-90 Subsection 4.3 to 5.3.
5.4	E	Moved ACI 349-90 Subsection 4.4 to 5.4.
5.5	E	Moved ACI 349-90 Subsection 4.6 to 5.5.
5.6	TS	Moved ACI 349-90 Subsection 4.7 to 5.6. Revised requirement of curing and testing by including subsection 5.6.3 "field-cured specimens" and added option for load tests in Subsection 5.6.4.5, "if structural adequacy remains in doubt".
5.7	E	Moved ACI 349-90 Subsection 5.1 to 5.7.
5.8	E	Moved ACI 349-90 Subsection 5.2 to 5.8. Specified that the batch mixer should be approved by the Engineer.
5.9	E	Moved ACI 349-90 Subsection 5.3 to 5.9.
5.10	E	Moved ACI 349-90 Subsection 5.4 to 5.10.
5.11	TI	Moved ACI 349-90 Subsection 5.5 to 5.11. Minor changes in terminology of Sections 5.11.3.1 and 5.11.4.
5.12	E	Moved ACI 349-90 Subsection 5.6 to 5.12.
5.13	E	Moved ACI 349-90 Subsection 5.7 to 5.13.
6.1	E	Editorial changes
6.2	-	No change
6.3	E	Editorial changes
6.4	E	Editorial changes

**Table 3.2-2 - Detailed Comparison of ACI-349
(1990 to 1997) (Continued)**

Section/ Subsection/ Article/ Subarticle/	Type of Change E = Editorial TI = Technical, not significant TS = Technical, significant	Description of the Change (Including Regulatory Significance)
7.0	-	No change
7.1.	E	Editorial changes
7.2	E	Editorial changes
7.3	-	No change
7.4	TS	Added provisions for epoxy coated reinforcing bars
7.5	E	Editorial changes
7.6	E	Editorial changes
7.7	E	Editorial changes
7.8	E	Editorial changes
7.9	-	No change
7.10	E	Editorial changes
7.11	TS	Reduced lap requirement for forming closed stirrups.
7.12	-	No change
8.0	E	Editorial changes
8.1	-	No change
8.2	-	No change
8.3	TI	Added reference to Appendix B, and deleted "a more accurate method" when describing the frame analysis in Subsection 8.3.3.
8.4	-	No change
8.5	-	No change
8.6	E	Editorial changes
8.7	E	Editorial changes
8.8	E	Editorial changes
8.9	E	Editorial changes
8.10	-	No change

Table 3.2-2 - Detailed Comparison of ACI-349 (1990 to 1997) (Continued)		
Section/ Subsection/ Article/ Subarticle/	Type of Change E = Editorial TI = Technical, not significant TS = Technical, significant	Description of the Change (Including Regulatory Significance)
8.11	E	Editorial changes
8.12	E	Editorial changes
9.0	TI	Editorial changes. Change symbol for time dependent factor for sustained load and eliminated l_s , length of short span for slabs.
9.1	-	No change
9.2	E	Editorial changes
9.3	E	Editorial changes
9.4	-	No change
9.5	E	Change symbol for time dependent factor in Eq. (9-10) from T to ξ .
10.0	TI	Added definition for A_{sk} , clarified definition of A_z and M_{lb} and editorial changes.
10.1	-	No change
10.2	-	No change
10.3	-	No change
10.4	-	No change
10.5	E	Editorial changes
10.6	TS	Revised 10.6.7 on added longitudinal reinforcing.
10.7	E	Editorial changes. Corrected referenced subsections.
10.8	-	No change
10.9	-	No change
10.10	-	No change
10.11	TS	Changed moment factor for frames braced against side sway in equation 10-6 from 0 to 1.
10.12	E	Editorial changes
10.13	-	No change

Table 3.2-2 - Detailed Comparison of ACI-349 (1990 to 1997) (Continued)		
Section/ Subsection/ Article/ Subarticle/	Type of Change E = Editorial TI = Technical, not significant TS = Technical, significant	Description of the Change (Including Regulatory Significance)
10.14	E	Editorial changes
10.15	TS	The discussion for stepped supports has been removed and editorial changes.
11.0	TS	Added definition of b'_1 , b'_2 and β_p and editorial changes.
11.1	TI	Stress limits are placed on shear strength of high strength concrete.
11.2	-	No change
11.3	-	No change
11.4	E	Editorial changes
11.5	E	Editorial changes
11.6	E	Technical clarification is included in Paragraph 11.6.4, the rest of 11.6 has not been changed. (Note: ACI 318-95 completely changes torsion design)
11.7	-	No change
11.8	TS	A new subsection 11.8.3 has been added, requiring different design for continuous deep members. Other changes are editorial.
11.9	TI	Eliminated requirements for light weight concrete.
11.10	-	No changes
11.11	TI	A new section is added discussing the transfer of moments to columns. This section was 11.12 in ACI 349-90.
11.12	E	This Section was 11.11 in ACI 349-90. Editorial changes
12.0	TS	Added definition of A_{tr} , C , f_{yt} , K_{tr} , l_{db} , n , α , β and γ .
12.1	TS	Added "or a combination thereof" to Subsection 12.1.1. Added 12.1.2 limiting values of $\sqrt{f'_c}$ to 100 psi
12.2	TS	Significant changes to formulations for development of bars and wires in tension, and added factors for coatings and location.
12.3	E	Editorial changes.
12.4	TS	Added a subsection explaining how to use factors in Section 12.2 for bundled bars.

**Table 3.2-2 - Detailed Comparison of ACI-349
(1990 to 1997) (Continued)**

Section/ Subsection/ Article/ Subarticle/	Type of Change E = Editorial TI = Technical, not significant TS = Technical, significant	Description of the Change (Including Regulatory Significance)
12.5	TS	Added factor for development of epoxy-coated reinforcement and editorial changes.
12.6	-	No changes
12.7	TS	Significant changes to formulations for development of welded deformed wire fabric in tension.
12.8	E	Editorial changes
12.9	E	Editorial changes
12.10	TI	Added reference to Sections 12, 11.4 and 12.12.4 in Subsection 12.10.6.
12.11	TS	Added subsection 12.11.4 for development of positive moment reinforcement at simple supports and editorial changes.
12.12	TS	Added subsection 12.12.4 that requires that negative tension reinforcement shall be continuous over interior supports for deep flexural members.
12.13	TS	Significant changes to formulations for development of web reinforcement.
12.14	TS	Subsection 12.14.3.5 only allows exception to other Section 12.14 requirements for No. 5 bars and smaller.
12.15	TS	Deleted Class C splices and only allows exception to Section 12.14 requirements for No. 5 bars and smaller and editorial changes.
12.16	TS	Added new requirements for compression lap splices for f_y greater than 60,000 psi, moved provision for lap length in spirally reinforced compression members to Subsection 12.17.2.5 and editorial changes. Moved requirements for spirally reinforced concrete members to Subsection 12.17.2.5.
12.17	TS	Significant changes to formulations and requirements for splice requirements for columns.
12.18	TS	Reduced required length of splices for welded deformed wire from $1.7 l_d$ to $1.3 l_d$ and require splices with any plain wires to meet the requirements of Section 12.19.
12.19	E	Editorial Changes.
13.0	TI	Added definition for M_u , deleted definition for d , E_{cc} , I_c and K_{ec} .
13.1	TI	Clarified definition of length of slab at column supports.

**Table 3.2-2 - Detailed Comparison of ACI-349
(1990 to 1997) (Continued)**

Section/ Subsection/ Article/ Subarticle/	Type of Change E = Editorial TI = Technical, not significant TS = Technical, significant	Description of the Change (Including Regulatory Significance)
13.2	-	No change
13.3	TS	Changed location where fraction of the unbalanced moment should be considered and editorial changes.
13.4	TS	Added subsection 13.4.8.5 requirements for bottom bars at supports with Class A splices, and editorial changes. Figure 13.4.8 has no changes.
13.5	E	Editorial changes
13.6	TS	Equation 13-4 changed and editorial changes.
13.7	TS	Added subsection 13.7.2.4 for frames adjacent and parallel to an edge and editorial changes.
14.0	E	Editorial changes
14.1	TS	Added requirement for minimum horizontal reinforcement per Section 14.3.3
14.2	TS	Added reference to Section 15.8, on transfer of force to footing
14.3	TS	Changes in minimum reinforcing areas and spacing.
14.4	E	Editorial changes
14.5	E	Editorial changes
14.6-14.7	-	No change
15.0	TI	Deleted definitions for $\sqrt{f_c'}$ and ϕ
15.1	-	No change
15.2	-	No change
15.3	E	Editorial changes
15.4	-	No change
15.5	E	Editorial changes
15.6	E	Editorial changes
15.7	-	No change
15.8	E	Editorial changes

Table 3.2-2 - Detailed Comparison of ACI-349 (1990 to 1997) (Continued)		
Section/ Subsection/ Article/ Subarticle/	Type of Change E = Editorial TI = Technical, not significant TS = Technical, significant	Description of the Change (Including Regulatory Significance)
15.9-15.10	-	No change
16.1	-	No change
16.2	-	No change
16.3	-	No change
16.4	-	No change
16.5	-	No change
16.6	-	No change
17.0	TI	Added definitions for A_c , A_v , h , s , λ , and ρ_v
17.1	-	No change
17.2	-	No change
17.3	-	No change
17.4	E	Editorial changes
17.5	TS	Changed provisions for ties for horizontal shear.
17.6	E	Corrected referenced Section of ACI-349 to Section 12.3.
18.0	TS	Changed definition of γ_p , deleted definition of A_c and editorial changes.
18.1	TS	Added exception of Section 7.6.5 in Section 18.1.3.
18.2	-	No change
18.3	-	No change
18.4	-	No change
18.5	-	No change
18.6	-	No change
18.7	E	Editorial changes
18.8	-	No change
18.9	E	Editorial changes

**Table 3.2-2 - Detailed Comparison of ACI-349
(1990 to 1997) (Continued)**

Section/ Subsection/ Article/ Subarticle/	Type of Change E = Editorial TI = Technical, not significant TS = Technical, significant	Description of the Change (Including Regulatory Significance)
18.10	E	Editorial changes
18.11	-	No change
18.12	-	No change
18.13	-	No change
18.14	TS	Added Subsection 18.14.3 for corrosion protection of unbonded single strand tendons, (Referenced specification from the Post-Tensioning Institute, July 1993)
18.15	-	No change
18.16	E	Editorial changes
18.17	-	No change
18.18	TS	Differences in force requiring check raised from 5% to 7% for post tensioned construction in Section 18.18.1.
18.19	TS	Anchorage development requirement increased from 90% to 95% in Section 18.19.1.
19.1-19.5	-	No change
20.1-20.6	TS	Added chapter on strength evaluation of existing structures. This new chapter is based upon Chapter 20 of ACI-318-89 (Rev. 1992).
21.1-21.7	TS	This chapter on special provision for seismic design is modeled on ACI 318-89 (Rev. 1992). The following differences are noted: •References to lightweight concrete are eliminated. •Distinction between high and moderate seismic risk is eliminated. •Deleted linear and nonlinear response in 21.2.2 •Shear walls with height to length less than 2.0 are exempted in ACI-349-97 •Requirements for frames in regions of moderate seismic risk, 21.8, were eliminated.
Appendix A	-	No change
Appendix B		
B.0	TS	Added definition of A_b , A_c , C , f_b , K_c , P_a , P_y , and s .
B.1	-	No change including Figures B.1.1 and B.1.2.

**Table 3.2-2 - Detailed Comparison of ACI-349
(1990 to 1997) (Continued)**

Section/ Subsection/ Article/ Subarticle/	Type of Change E = Editorial TI = Technical, not significant TS = Technical, significant	Description of the Change (Including Regulatory Significance)
B.2	TI	Added definition of “strength, design” and “strength, nominal” and editorial changes.
B.3	TS	Subsection B.3.7 allows use of available confining force from tension anchors in combination with external loads to be utilized in determining the shear capacity of anchorages with shear lugs.
B.4	TS	Subsection B.4.5.1 limits the design bearing strength used for concrete or grout placed against shear lugs.
B.5	TS	Subsection B.5.1.2.2 changed substantially. ACI 349-1990 stated that shear lugs only in the compression zone may be considered effective, ACI-349-1997 allows consideration of these lugs but the evaluator should consider bearing strength, confinement and displacement compatibility. The spacing limitation on multiple shear lugs has been eliminated.
B.6	TS	Substantially changed the formulation and design requirements for embedment steel.
B.7	-	No change
B.8	-	No change
B.9	E	Editorial changes
B.10	-	No change
Appendix C C.0	E	Deleted definition for A_{sh} and P_s .
C.1	-	No change
C.2	-	No change
C.3	-	No change
C.4	-	No change
C.5	-	No change
C.6	-	No change
C.7	-	No change
Appendix D	TS	Made appendix consistent with new formulations in ACI 349-97 and editorial changes.

3.3 Related Codes and Standards

3.3.1 ACI-318

The only portion of ACI-318 which is considered for implementation is Section 11.6 for members subject to torsion and combined shear and torsion. ACI 313-95 includes more state of the art criteria. Chapter 21 on ductile detailing in ACI 318-95 has been included in ACI 349-97 and therefore does not require implementation. This item is discussed in Section 3.5.1.1.

3.3.2 ANS 6.4-1997

The Standard provides the criteria for the design of Concrete Radiation Shielding. ANS 6.4 does not alter the criteria or requirements of ACI-349. Further, it is the document applicable for the design of concrete shielding. Therefore, its reference in the current version of Regulatory Guide 1.142 is appropriate and no changes are necessary.

3.4 Quality Assurance Standards and Requirements

The ANSI/ANS N45.2.5 Standards referenced in the current revision of Regulatory Guide 1.142 are no longer supported by ANSI or ANS. However, they are still referenced as the licensing and design basis for many of the existing Nuclear Power Plants. Their text is consistent with 10CFR 50 Appendix B Quality Assurance requirements, hence they continue to be used. The current industry consensus standard applicable to Quality Assurance is ASME-NQA-1 and is also compatible with the requirement of 10CFR50 Appendix B. The applicability of the ASME-NQA-1 standard is reviewed in Section 3.4.1 Also, briefly discussed in Section 3.4.2 is the graded Quality Assurance approach.

3.4.1 ASME NQA-1

ANSI/ASME NQA-1 (7.5-[9]) standard follows the format of the 18 criteria listed in 10CFR50, Appendix B. The 1997 edition permits other quality assurance formats to be used, however, this edition has not been endorsed by the NRC. Regulatory Guide 1.28, "Quality Assurance Program Requirements (Design and Construction)" endorses ANSI/ASME NQA-1-1983 and describes a method acceptable to the NRC staff for complying with the provisions of 10CFR50, Appendix B. Licensees

should use their approved Quality Assurance programs as a model to address future site activities that require Quality Assurance programs.

3.4.2 Federal Regulation 830.120

This federal regulation [7.4-6] defines a simpler, graded approach to Quality Assurance. It is compatible with ASME NQA-1. The basic criteria are as follows:

Quality Assurance Criteria

-Management

- a) **Criterion 1 –Program.**
- b) **Criterion 2 –Personnel Training and Qualification.**
- c) **Criterion 3 — Quality Improvement.**
- d) **Criterion 4 – Documents and Records.**

-Performance

- a) **Criterion 5 – Work Processes.**
- b) **Criterion 6 – Design.**
- c) **Criterion 7–Procurement.**
- d) **Criterion 8–Inspection and Acceptance Testing.**

-Assessment

- a) **Criterion 9–Management Assessment.**
- b) **Criterion 10–Independent Assessment.**

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3.5 Advanced Reactor Design Commitments

As discussed briefly in Section 2.2, Adams and Stevenson in NUREG/CR-6358 conducted extensive review of the Applicability of Industry Consensus Codes and Standards to Design Advanced Reactors.

As part of that review, necessary changes to ACI-349 were identified to (1) address unique design features of the Advanced Reactor Designs and (2) to address deficiencies or limitations of the Code.

Since the issuance of NUREG/CR-6358, the licensing activity of several of the Advanced Reactor Designs has been significantly reduced. Currently, the ABB/CE-Sys 80⁺ and GE/ABWR have received 10 CFR Part 58 licenses, and the W-AP600 design is still actively pursuing a 10 CFR Part 58 license. As of the writing of this report, there is no licensing activity for the remaining reactor designs. For the three active advanced reactor designs (GE-ABWR, ABB/CE Sys. 80⁺, W-AP600), the information provided in NUREG/CR-6358 was updated to reflect the final SSAR's and the SER's for the GE ABWR and ABB/CE Sys. 80⁺ and the latest revision of the SSAR for the Westinghouse AP600. The results of these updated reviews are discussed in the following subsections.

3.5.1 ABB/CE Sys 80⁺

There are no significant changes which have occurred in the concrete structural area for the ABB/CE Sys. 80⁺ since the issuance of NUREG/CR-6358. Therefore, the data presented in this section directly extracted from NUREG/CR-6358.

Table 3.5-1 presents a summary of the unique aspects of the Sys. 80⁺ design as they apply to concrete structures. The following subsections discuss the major items in more detail.

3.5.1.1 Use of ACI-318, Chapter 21

ABB/CE committed to aspects of Chapter 21 of ACI 318-89, Revised 1992; however, the reference section to Chapter 21 is somewhat ambiguous. It is believed that this was done to take advantage of the enhanced ductile design features which are provided in Chapter 21 of ACI-318. These code features are now available in ACI-349-1997 and therefore reference to ACI 318 is no longer

necessary. This is an enhancement to the overall plant design. ACI-349-1997 includes a Chapter 21, Seismic Requirements. While the requirements for most reinforced and prestressed concrete components are similar to ACI-318, it should be noted that requirements for shear walls are different. The most significant difference being an exemption based upon height to length ratios of shear walls. Since shear walls are the most important components in nuclear power plants for resisting lateral loads, this exemption on shear walls may eliminate much of the seismic detailing requirements of ACI-318. Committee 349 decided that shear walls of low height to length ratios were not typical of ACI-318 structures and in any event would not develop concentrated compressive stresses which would lead to failures due to crushing of concrete. ACI-349-1997, Chapter 21 should be invoked including a stipulation for the shear wall exemption.

3.5.1.2 Concrete Missile Barrier

ABB/CE intends to use the modified Petry or Modified NDRC methodology to evaluate penetration potential for missile penetration in concrete. Methods suggested by the 1980 Specialty Conference for concrete and the ASCE Committee Report on Missile Impact and Penetration ((7.5-[8]) supplement the scope of the code. It is suggested missile barrier design should be included in the scope of Regulatory Guide 1.142 by modifying another extreme load case in ACI 349-1997.

3.5.2 GE - ABWR

As with the Sys 80+ for the GE-ABWR, there are no significant changes in concrete design basis and structures since the issuance of NUREG/CR-6358. Therefore, the results presented here are extracted directly from NUREG/CR-6358. Table 3.5-2 presents a summary of the unique aspects of the ABWR design as they apply to concrete structures. In addition, major items are discussed in detail in the following subsections.

3.5.2.1 Limited Use of Modular Construction

The RPV pedestal and shield wall structures are made up of two concentric steel shells joined by horizontal and vertical steel plate diaphragms. These steel structures are first set in place and then filled with concrete. Based on the information provided in the SSAR, it is stated that all loads are resisted by the integral action of the inner and outer steel shells. It is further stated that the concrete placed in the annulus between the inner and outer shells

acts to distribute loads between the steel shells and provides compressive load stability. The design of this concrete is per ACI-301 which is not a safety related design code nor does it address any possible composite action between the concrete and the steel members. Modifications should be made to ACI-301 to address the safety related aspects of this application. Further, the applicable steel and concrete design codes should be modified to insure that there is adequate concrete corrosion protection and adequate structure composite action.

Regulatory Guide 1.142 should provide directions on how to address this issue by incorporation and modification of the necessary ACI-301 design criteria.

3.5.2.2 Seismic Hydrodynamic Loadings

The ABWR uses a pressure suppression type of containment which includes a water suppression pool inside the reactor building. During an earthquake there will be significant seismic excitation of the reactor building which will cause excitation of the water in the suppression pool resulting in direct and building filtered hydrodynamic loadings being applied to the reactor building and internal structures. Analysis techniques for these types of loadings currently do not exist in ASCE 4-86 or in the SRP. In addition the design loads and load combinations for these loads are currently not specified in ACI -349. The suggested approach to address this issue in ACI-349 is discussed in Section 3.8.2.

3.5.2.3 Concrete Missile Barrier

In SAR 3.5.3.1.1, either the modified Petry formula or the TM5-855-1 formula may be used. In SAR Appendix 3H.1.5.7 only TM 5-855-1 was used. This should be clarified; however, the modified NDRC formulas have been generally used in past practice.

Missile shield design is somewhat unique to nuclear structures. Therefore, it is suggested that references supplementing ACI 349-1997 should be included in Regulatory Guide 1.142. It is suggested missile barrier design should be included in the scope of Regulatory Guide 1.142 by modifying another extreme load case in ACI 349-1997.

3.5.2.4 Concrete Codes

While only ACI-318 is listed in SAR 3.8.4.2.1 for reactor buildings, SAR Appendix 3H.1.4.11 lists ACI-318, ACI-

349 and ACI-359 for the reactor building and discusses no specific application of ACI-318. This leaves in doubt the implementation of the seismic detailing requirements as specified in Chapter 21, ACI-318.

This issue is a specific licensing issue for the GE-ABWR. It is not suggested that ACI-318 be cited in Regulatory Guide 1.142 as an acceptable replacement for ACI-349. ACI-359 is applicable only to concrete containment structures and is excluded from Regulatory Guide 1.142. Chapter 21 of ACI-318 has now been included with ACI 349-1997 as discussed in Section 3.5.1.1.

3.5.2.5 Concrete Containment and Reactor Building Interconnection

In a departure from past practice, the containment structure is structurally connected to the reactor building. The connections are made by the following:

- a) Six reinforced concrete slabs which monolithically connect the reactor building to the containment
- b) The containment top slab is integral with the fuel pool girders and the containment wall

In spite of the fact that the combined structure will respond to all loads, SAR Fig. 3H.1-2 indicates a division of code applicability; such that only the containment walls and basemat within the containment walls are covered by ACI-359/ASME III, Div.2. The remainder of the combined structure; ie, the reactor building, is presumably covered by ACI-349 and to some unknown extent by ACI-318. This situation is rather confused in that the basemat and reactor building participates with the containment and therefore, must be considered as adding strength and thus treated as containment. This item is discussed in greater depth in Section 3.8.6.

Therefore it is suggested that a jurisdictional criteria should be added to Regulatory Guide 1.142 to state that in such situations which clearly defines (a) jurisdictional boundaries for Regulatory Guide 1.142 (ACI-349) and (b) the methodology to address the interaction issues across such jurisdictional boundaries.

3.5.3 Westinghouse AP600

The SSAR for the AP 600 has gone through a significant number of revisions since the completion of NUREG/CR-6358. Therefore, the latest revision of the SSAR was reviewed in depth and combined to the results presented in NUREG/CR-6358. While the majority of the factors

and issues presented in NUREG/CR-6358 remained unchanged, there were a few significant changes which have occurred. Therefore, the results presented in Table 3.5-3 and the following subsections are based on the results of NUREG/CR-6358 updated to incorporate changes which have occurred through revisions of SSAR.

3.5.3.1 Modular Construction

The use of modular construction techniques are discussed throughout the AP600 SSAR, the major structural modules that will be implemented are "M" wall modules, "L" wall modules, floor modules and the use of prefabricated concrete reinforcing steel frames. The construction is reinforced concrete for the containment internal walls below 98'. Implementation of the "L" and "M" modules begins above the 98' elevations. However, as discussed below some of these modules extend below the 98' level to fulfill various design functions.

The "M" modules will be used in the construction of containment internal wall structures above elevations of 83'-98' depending on their location within the containment. These modules are prefabricated structural steel box sections with steel plates on each face stiffened by vertical diaphragms and horizontal angle stiffeners assembled from subunits which are constructed offsite. Wall modules are anchored to the concrete base by means of anchor bolts and dowels embedded in the concrete below el. 98'. After erection, the walls are filled with concrete. There is no reinforcing steel used in the concrete since, in general, the concrete will not be required to carry loads. In addition the concrete will not be designed to act compositely with the structural steel except for vertical compression loads.

"L" modules are permanent steel forms used for the containment internal base concrete structures below elevation 98'. The steel modules consist of steel plates (A36) reinforced with 2" x 2" angle stiffeners and 4" WT sections on the concrete side of the plate. The L wall modules are used in lieu of removable concrete forms.

The concrete floors in the containment interior above elevation 98' consist of steel tee-sections welded to horizontal steel plates. The steel plates are stiffened by angle stiffeners. Support is provided using deep girders whose webs pass through the horizontal plate. Reinforcing bars are placed above the top flange. After erection, concrete is poured on top of the horizontal plates embedding the upper section of the beams and reinforcing bars. The concrete will be designed to act compositely

with the tee sections for vertically downward (not upward) loads. However for upward loads the concrete together with angle plate stiffeners provides stability to the plates. The above also applies to the operating deck above the IRWST; however, the remainder of the operating floor consists of 12" concrete on "Q" decking supported by structural steel. Design of these slabs for composite action references the AISC N-690 standard provisions for fully encased beams, although the described configuration varies from that specified in AISC N-690. Due to the presence of the horizontal plate and angle stiffeners, the intent of the current provisions for fully encased beams is considered to be met, and a "better" bond between the steel and concrete than obtained by the standard configuration is claimed. Suggested changes to the AISC N-690 Standard to address this situation are provided in Section 4.

Floors located above the main control room and instrumentation and control rooms in the auxiliary building are designed as finned floor modules. The purpose of the finned floor modules is to provide a passive heat sink for each room. A finned floor is comprised of a 24 inch thick reinforced concrete slab poured over a stiffened steel plate ceiling. Composite action of the steel and concrete is developed using shear studs welded to the steel plate and embedded in the concrete. The horizontal steel plates are stiffened by welding 1/2" x 9" steel plate perpendicular to the ceiling plates. The steel fins project into the room and act as thermal fins to enhance the transfer of heat from the air to the concrete. Several modules cut to the room width are prefabricated in a shop. On site they are installed side by side perpendicular to the room length. Adjacent panels are made continuous by welding a flat bar along the interface of two panels.

Prefabricated, component rebar support frames will be used in the nuclear island basemat. The frames consist of a bi-direction top and bottom rebar mesh welded to the top and bottom of structural members. These frames are fabricated in sections off site, laid in place on site, and then concrete fills the forms covering the rebar frames. This concept may also be adopted in portions of the containment building interior and exterior concrete placement.

The subject industry codes and standards currently do not adequately address these construction techniques. Therefore the use of these designs will require code changes to Section Q1.11 of AISC N690 and probably to ACI-349 and possibly ACI-347. In addition some testing

may be required to verify the composite actions of these components. Therefore it is suggested the Regulatory Guide 1.142 identify this type of construction as an unique item and state it will be addressed on a case by case basis.

3.5.3.2 Passive Containment Cooling System (PCCS)

The Passive Containment Cooling System (PCCS) provides a mechanism to maintain the post accident containment pressure and temperature within design limits for at least 72 hours without external power and/or explicit operator actions to initiate or activate safety systems.. The containment is an ASME BPVC Section III, Division 1 Class MC (Subsection NE) steel cylindrical, elliptical head containment vessel. It is surrounded by cylindrical reinforced concrete shield building that results in an annular space between the containment vessel and the shield building. Initial post accident cooling is provided by natural circulation and free air convective heat transfer in this annular region. Should additional heat removal be required a containment wall cooling system is automatically initiated. The containment vessel has a weir structure attached to the top of the vessel which is used for water distribution during this portion of the cooling process. The top of the shield building has a annular shaped water storage tank. This tank discharges water into the weir system which allows it to flow over the containment vessel achieving the necessary cooling.

The PCCS will require changes to ASME-III Subsection NE. In addition changes may be required to ACI-349 to address the design of the concrete water storage tank. During an earthquake event hydrodynamic loads due to the water in the storage tank should be considered in the building design analysis and the generation of in-structure response spectra.

Regulatory Guide 1.142 should insure the load combinations consider hydrostatic pressure loads and should also provide a footnote to indicate that the design basis seismic loads should include the effects of hydrodynamic loads from the water stored in the tank. In addition other concerns such as water chemistry and it's long term effects should be discussed in Regulatory Guide 1.142.

3.5.3.3 Seismic Hydrodynamic Loadings

In the AP600 design the top of the reactor shield building contains an annular shaped storage tank which stores

350,000 gallons of water for potential use with the Passive Containment Cooling System. During an earthquake there will be significant seismic excitation of the tank which will cause excitation of the water within the tank resulting in direct and building filtered hydrodynamic loadings being applied to the shield building and to the balance of the nuclear island. It is suggested that this item be addressed in a manner similar to BWR hydrodynamic loads as discussed in Section 3.8.2.

3.5.3.4 Building Filtered IRWST Operation Hydrodynamic Loads

During an Automatic Depressurization Event (ADS) significant dynamic loads are imposed on the In-containment Reactor Water Storage Tank (IRWST). This input causes excitation (vibration) of the IRWST which in turn may excite the building structure housing the tank (the containment and containment internal structures). This could result in direct hydrodynamic loads on the containment structure and building and building filtered hydrodynamic loads on the balance of the nuclear island structures. This effect is similar to the hydrodynamic loadings in pressure suppression containments. While some test data is available, the magnitude of the loads are not yet defined and the overall effects of the significance of this load can not be quantified. It is suggested that this item be addressed in a manner similar to BWR hydrodynamic loads as discussed in Section 3.8.2.

3.5.3.5 Concrete Placement Issues

Specific concrete placement procedures for the "M" wall modules need to be developed for the following reasons:

- a) Height may require careful placement and consolidation of wet concrete (vibration)
- b) Vertical diaphragms and horizontal stiffeners may interfere with complete consolidation
- c) Access for concrete workers inside module may be restricted.

There may also be potential problems in concrete placement for the conical roof. Regulatory Guide 1.142 should address this issue by direct reference to the appropriate ACI concrete placement standards for such construction.

3.5.3.6 Concrete Codes

While the current SSAR does not reference ACI-318, it is the authors understanding that the detailing, placing, anchoring, and splicing of reinforcing bars will be in accordance with Chapter 21 of ACI-318. Chapter 21, Seismic Requirements is now included in ACI 349-1997. While the requirements for most reinforced and

prestressed concrete components are similar to ACI-318, it should be noted that requirements for shear walls are different. The most significant difference being an exemption based upon height to length ratios of shear walls. Since shear walls are the most important components in nuclear power plants for resisting lateral loads, this exemption on shear walls may eliminate much of the seismic detailing requirements of ACI-318. Committee 349 decided that shear walls of low height to length ratios were not typical of ACI-318 structures and in any event would not develop concentrated compressive stresses which would lead to failures due to crushing of concrete.

3.5.3.7 Concrete Missile Barrier Design

The use of the Modified NDRC is specified in SSAR 3.5.3. Appropriate standardized missile design criteria should be developed and incorporated into an appropriate industry standard. This standard can supplement the treatment given this subject in Regulatory Guide 1.142. It is suggested missile barrier design should be included in the scope of Regulatory Guide 1.142 by modifying another extreme load case in ACI 349-1997.

3.5.4 ACI Standards

In NUREG/CR-6358, the majority of the changes recommended to the ACI Standards are in ACI-349, which is the primary standard for the design of Nuclear Safety Related Reinforced Concrete Structures. The recommended changes of NUREG/CR-6358 are summarized in Table 3.5-4. Depending on the actual code language there could be additional changes required to several of the supporting ACI Standards but identification of possible subordinate changes is beyond the scope of this program. Several of the suggested changes have been introduced in to ACI-349-97. For those changes not yet incorporated in ACI-349, Regulatory Guide 1.142 should provide the necessary guidance to incorporate the needed requirements. The suggested changes to ACI 345 have been forwarded to the ACI Committee 349 for review and comment.

Table 3.5-1 Unique Features and Attributes of the ABB/CE System 80+							
Structure	Description	Unique Features and Attributes					
		Physical	Application	Design (Inc. Loads)	Analysis	Fabrication/ Construction	Testing
<u>Reactor Building</u> Shield Building	Cylindrical concrete structure with hemispheric dome enclosing containment	Part of nuclear island.	N/I ⁽¹⁾	1. Elimination of the OBE 2. Use of ACI-318, Chapter 21	N/I ⁽¹⁾	N/I ⁽¹⁾	N/I ⁽¹⁾
<u>Reactor Building</u> Internal Structures	Reinforced Concrete Over the Bottom of the Containment Shell (Note: The Containment Shell is Founded on the Basemat)	1. Structures not supporting Seismic Category I items are Seismic Category II	N/I ⁽¹⁾	1. Elimination of the OBE 2. Severe accident criteria 3. Use of AISC N690 4. Use of ACI-318, Chapter 21	1. Seismic II/I criteria and analysis methods	N/I ⁽¹⁾	N/I ⁽¹⁾
<u>Reactor Building</u> Subsphere	Portion of reactor building below 26.3m (91'-9") which is in direct contact with containment vessel.	1. Transition zone design criteria between the sphere and the subsphere 2. Compressible material is used.	N/I ⁽¹⁾	1. Elimination of the OBE 2. Interface design between containment vessel and basemat 3. Use of ACI-318, Chapter 21	N/I ⁽¹⁾	N/I ⁽¹⁾	N/I ⁽¹⁾
<u>Nuclear Annex</u> Control Area	Monolithic Attachment to Shield Building on Common Basemat	N/I ⁽¹⁾	N/I ⁽¹⁾	1. Elimination of the OBE 2. Use of ACI-318, Chapter 21	N/I ⁽¹⁾	N/I ⁽¹⁾	N/I ⁽¹⁾
<u>Nuclear Annex</u> Spent Fuel Pool	Steel lined pool Integral part of the Nuclear Annex	N/I ⁽¹⁾	N/I ⁽¹⁾	1. Elimination of the OBE 2. Use of ACI-318, Chapter 21	N/I ⁽¹⁾	N/I ⁽¹⁾	N/I ⁽¹⁾

Table 3.5-1 Unique Features and Attributes of the ABB/CE System 80+ (Continued)							
Structure	Description	Unique Features and Attributes					
		Physical	Application	Design (Inc. Loads)	Analysis	Fabrication/ Construction	Testing
<u>Nuclear Annex</u> Valve House Areas	Integral part of the nuclear annex	N/I ⁽¹⁾	N/I ⁽¹⁾	1. Elimination of the OBE 2. Use of ACI-318, Chapter 21	N/I ⁽¹⁾	N/I ⁽¹⁾	N/I ⁽¹⁾
<u>Nuclear Annex</u> Emer. Diesel Generator Areas	Integral part of the nuclear annex	N/I ⁽¹⁾	N/I ⁽¹⁾	1. Elimination of the OBE 2. Use of ACI-318, Chapter 21	N/I ⁽¹⁾	N/I ⁽¹⁾	N/I ⁽¹⁾
<u>Nuclear Annex</u> CVCS/Main Area	Integral part of the nuclear annex	N/I ⁽¹⁾	N/I ⁽¹⁾	1. Elimination of the OBE 2. Use of ACI-318, Chapter 21	N/I ⁽¹⁾	N/I ⁽¹⁾	N/I ⁽¹⁾
Station Service Water Pump Structure	Reinforced Concrete Structure	N/I ⁽¹⁾	N/I ⁽¹⁾	1. Elimination of the OBE 2. Use of ACI-318, Chapter 21	N/I ⁽¹⁾	N/I ⁽¹⁾	N/I ⁽¹⁾
Component Cooling Water Heat Exchanger Structure	Reinforced Concrete Structure	N/I ⁽¹⁾	N/I ⁽¹⁾	1. Elimination of the OBE 2. Use of ACI-318, Chapter 21	N/I ⁽¹⁾	N/I ⁽¹⁾	N/I ⁽¹⁾
Diesel Fuel Storage Structure	Reinforced Concrete Structure	N/I ⁽¹⁾	N/I ⁽¹⁾	1. Elimination of the OBE 2. Use of ACI-318, Chapter 21	N/I ⁽¹⁾	N/I ⁽¹⁾	N/I ⁽¹⁾
Foundations	Reinforced Concrete Structure	N/I ⁽¹⁾	N/I ⁽¹⁾	1. Elimination of OBE 2. Use of ACI-318, Chapter 21	N/I ⁽¹⁾	N/I ⁽¹⁾	N/I ⁽¹⁾

Table 3.5-1 Unique Features and Attributes of the ABB/CE System 80+ (Continued)							
Structure	Description	Unique Features and Attributes					
		Physical	Application	Design (Inc. Loads)	Analysis	Fabrication/ Construction	Testing
Missile Barrier	Concrete	N/I ⁽¹⁾	N/I ⁽¹⁾	N/I ⁽¹⁾	1. For concrete use modified Petry or modified NDRC methodology	N/I ⁽¹⁾	N/I ⁽¹⁾

Footnotes for Table 3.5-1:

⁽¹⁾ N/I = None Identified

Table 3.5-2 - Unique Features and Attributes of the GE ABWR							
Structure	Description	Unique Features and Attributes					
		Physical	Application	Design (Inc. Loads)	Analysis	Fabrication / Construction	Testing
Containment Vessel	Reinforced Concrete Structures	Structurally interconnected to reactor building. Current ACI codes do not cover this interaction.		1. Elimination of the OBE, Fatigue & Fatigue Ratchet Criteria for SSE for liner loadings 2. Seismic Hydrodynamic loadings 3. Hydrodynamic loadings	Combined analysis is required.	N/I ⁽¹⁾	N/I ⁽¹⁾
Diaphragm Floor	Reinforced Concrete Slab	Supported by reactor pedestal and containment wall	N/I ⁽¹⁾	1. Elimination of the OBE 2. Seismic Hydrodynamic loadings 3. Hydrodynamic loadings	ASME III BPVC, Div. 2 for intersection with containment wall	N/I ⁽¹⁾	N/I ⁽¹⁾
RPV Pedestal & Shield Wall	Pedestal consists of two concentric steel shells tied together. Annulus filled with concrete ----- Shield Wall, Two Cylindrical Steel Structures filled with Concrete	1. Composite Construction Steel and Concrete although only steel appears to be used for resisting forces and moments	N/I ⁽¹⁾	1. Elimination of the OBE 2. Composite Design Criteria 3. Use of ASIC N690 4. Seismic Hydrodynamic loadings 5. Hydrodynamic loadings 6. Use of ACI-301 for Concrete Design	N/I ⁽¹⁾	1. Appropriate Construction Procedures for Concrete for Pouring 2. Use of ACI-301 for Concrete Fabrication	N/I ⁽¹⁾

Table 3.5-2 - Unique Features and Attributes of the GE ABWR (Continued)							
Structure	Description	Unique Features and Attributes					
		Physical	Application	Design (Inc. Loads)	Analysis	Fabrication / Construction	Testing
Reactor Building	Similar to Current Layouts. Reinforced Concrete Structure except that it is structurally interconnected with reinforced concrete containment structure	N/I ⁽¹⁾	N/I ⁽¹⁾	1. Elimination of the OBE 2. Fatigue & Ratchet design criteria for SSE for components designed to NE 3. Seismic Hydrodynamic loadings 4. Hydrodynamic loadings	Both ACI-349 and ACI 318 are listed. No specifics on ACI-318 are provided	N/I ⁽¹⁾	N/I ⁽¹⁾
Containment Internal Concrete	Reinforced Concrete	N/I ⁽¹⁾	N/I ⁽¹⁾	1. Elimination of the OBE 2. Seismic Hydrodynamic loadings 3. Hydrodynamic loadings	N/I ⁽¹⁾	N/I ⁽¹⁾	N/I ⁽¹⁾

Footnotes for Table 3.5-2:

⁽¹⁾ N/I = None Identified

Table 3.5.3 - Unique Features and Attributes of the Westinghouse AP600						
Structure	Description	Unique Features and Attributes				
		Physical	Application	Design (Inc. Loads)	Analysis	Fabrication/ Construction
Nuclear Island Basemat (Foundations)	Reinforced Concrete Basemat Common to Containment Vessel, Shield Building and Auxiliary Building 1. Exterior and interior concrete walls in the Auxiliary Building frame into the reinforced concrete shield structure. Therefore, the basemat, the Auxiliary building, and the shield structure are in effect a monolithic structure, such that the entire structure functions as a combined basemat and superstructure. 2. Flexural moments are introduced into mat spanning between Auxiliary Building walls. This then incorporates the Deep Beam requirements. ACI Code should specifically address this situation.	1. Interaction Requirements between steel containment shell type vessel and reinforced concrete basemat 2. Mat is 6'-0" thick under the Auxiliary Building. This is shallow in comparison to similar structures (basemats) in the SYS 80* and the ABWR ($\approx 40\%$ less).	N/I ⁽¹⁾	1. Elimination of the OBE 2. Interaction with containment vessel 3. Building Filtered Hydrodynamic loads from Seismic Excitation of Shield Building Storage Tanks 4. Building Filtered IRWST Hydrodynamic Loads	1. Finite element analysis includes effect of walls and interior structures to resist mat bearing pressures.	N/I ⁽¹⁾
						N/I ⁽¹⁾

Table 3.5-3 - Unique Features and Attributes of the Westinghouse AP600 (Continued)

Structure	Description	Unique Features and Attributes				
		Physical	Application	Design (Inc. Loads)	Analysis	Fabrication/ Construction
<u>Nuclear Island</u> Containment Interior	Concrete at the base of steel containment. 1. Concrete is placed over the lower ellipsoidal head: forming cavities for the reactor vessel and refueling canal and the steam generators.	1. L modules 2. M modules 3. Floor modules	1. Use of AISC N690 for structural steel designs	1. Elimination of the OBE 2. Sandwich panels (modular construction) 3. Missile barrier capacity of sandwich panels 4. Building Filtered Hydro-dynamic loadings from Seismic Excitation of the Shield Building Storage Tank 5. Building Filtered IRWST Hydrodynamic Loads 6. Direct IRWST Hydrodynamic Loadings	1. Stress Criteria for sandwich panels 2. Potential for testing requirements to demonstrate adequacy of composite design approach	1. Modular construction 2. Concrete Placement
						N/I ⁽¹⁾

Table 3.5-3 - Unique Features and Attributes of the Westinghouse AP600 (Continued)

Structure	Description	Unique Features and Attributes					Testing
		Physical	Application	Design (Inc. Loads)	Analysis	Fabrication/ Construction	
<u>Nuclear Island</u> Shield Building	Cylindrical concrete supported on basemat with conical roof	1. Modular rebar design 2. Concrete storage water tanks on top of shield building	N/I ⁽¹⁾	1. Elimination of the OBE 2. Hydrodynamic loadings from Seismic Excitation of the Shield Building Storage Tanks 3. Building Filtered IRWST Hydrodynamic Loadings	1. Finite element analysis includes effect of walls and interior structures to resist mat bearing pressures.	1. Modular construction and poured in place 2. Potential problems for concrete placement for conical roof concrete.	N/I ⁽¹⁾
Missile Barriers	Concrete and Steel	N/I ⁽¹⁾	N/I ⁽¹⁾	N/I ⁽¹⁾	1. For steel use of BRL or Stanford Formulas 2. For reinforced concrete barrier the use of Modified NDRC formula.	N/I ⁽¹⁾	N/I ⁽¹⁾

Footnotes for Table 3.5-3:

(1) N/I = Not Investigated or Reviewed

Table 3.5-4 - Recommended Changes to ACI-349 From NUREG/CR-6358		
Needed Change	Reason or Basis for the Change	Discussion of the Change
Earthquake Design Basis	The change to Appendix S to 10CFR50 will eliminate the OBE from the design basis.	Changes should be made to eliminate or provide the option to eliminate the OBE from the design basis equations, load combinations and strength criteria.
Ductile Detailing	Improve the seismic resistance design of related reinforced concrete structures.	Add a chapter similar to Chapter 21 to ACI-318 to provide improved detailing of connections, etc., to improve the ductility and seismic capacity of nuclear safety related structures.
Earthquake Experience Related Changes	Modify specification to address observations from the behavior of structures subjected to recent strong motion for earthquakes.	The earthquake (SSE) design criteria should be evaluated for displacement story drift limiting acceptance criteria in addition to the current strength limiting criteria. This new criteria needs to be developed.

Table 3.5-4 - Recommended Changes to ACI-349 From NUREG/CR-6358 (Continued)		
Needed Change	Reason or Basis for the Change	Discussion of the Change
Composite or Modular Construction	Westinghouse AP600 and GE ABWR modular or composite construction items.	1. Changes should be made to address composite steel and concrete members being used in the subject reactor designs. Specific areas where the code lacks or provides insufficient requirements include composite concrete steel in-filled walls and members subjected to out-of-plane flexural, in-plane shear as well as vertical compressive loads, and the use of partially encased steel beams without mechanical shear connectors for composite design of slabs. Changes should be made in the areas of design when the concrete is a primary load carrying member as well as considerations (e.g. displacement limits, etc.) for protection of the concrete subjected to loads assumed taken primarily or solely by steel elements. Appropriate design and evaluation criteria should be provided along with required detailing to insure that the assumed composite actions occurs. Such changes should be co-ordinated with applicable sections to AISC N690. 2. For Modules prefabricated off site (at vendor facilities) prefabrication and transportation load evaluation criteria should be added to the code. 3. Concrete placement criteria and requirements for the prefabricated modules as being used in the Westinghouse AP600 should be provided by an Appendix to this standard. This appendix can directly provide the criteria or it can be provided by reference to other existing ACI standards. This appendix must provide necessary testing and quality assurance requirements.
Concrete Water Holding Facility Design Criteria	Several advanced reactors contain concrete water storage tanks and spent fuel pools	Enhance and specify guides for the design of concrete water storage facilities and spent fuel pools. This should include consideration of hydrodynamic loads and steel (stainless steel) liner design. This liner design should address leakage protection, concrete interaction, and any potential fatigue design requirements. This also includes possible modifications to Appendix C for impulse and impact loads.

Table 3.5-4 - Recommended Changes to ACI-349 From NUREG/CR-6358 (Continued)		
Needed Change	Reason or Basis for the Change	Discussion of the Change
New Commodity Loads	The use of large in plant water storage facilities in some advanced reactors and the deep embedment of the other facilities.	Add a new commodity load to the load definition which is different from dead and live loads and can be used to address these items. This is done partially in recognition that commodities respond dynamically at their frequencies and not the frequency of their supporting member.
Jurisdictional Boundaries	The ABWR Reactor Building is a structure monolithically attached to the containment. The containment is constructed to ASME BPVC, Division 2, Subsection CC (ACI-359) while the reactor building is designed to ACI-349.	Changes are required to address the code jurisdictional boundaries, inter-connection joints and applied loading, interface and load transfer. All construction interface boundaries and jurisdictional definition should be reviewed.
Man-made Hazard Criteria	Address man-made phenomenon hazards.	Loadings, load combinations and acceptance criteria should be provided for man-made phenomena hazards.

3.6 Regulatory Requirements and Issues

This section provides a summary of the reviews of Regulatory Guidance in and relative to Regulatory Guide 1.142. The effect on suggested changes to Regulatory Guide 1.142 is also discussed.

3.6.1 Review of Current Regulatory Positions in Regulatory Guide 1.142

This section reviews the 12 USNRC positions which established modifications or clarifications to ACI-349-76 in Regulatory Guide 1.142. It is suggested that positions similar to those currently provided positions 1, 4, 5 and 9, 10, 11, 12 should be stipulated with the suggested implementation of ACI-349-97. It is suggested the remaining positions should be modified or amended as discussed in the following subsections.

3.6.1.1 Load Factors for OBE (Position 6)

The current USNRC position on load factors for OBE is $1.9 E_o$ instead of the current ACI-349 position of $1.7 E_o$. Assuming the option to use OBE, it is recommended that an OBE factor be reduced to 1.7 as currently stated in ACI-349. The basis for this recommendation is as discussed in the following paragraph.

Using the damping values from Regulatory Guide 1.61 for reinforced concrete of 4% and 7% for OBE and SSE, respectively and ground response spectra of Regulatory Guide 1.60, one obtains amplification factors of 2.92 and 2.27 for OBE and SSE, respectively. If one takes OBE peak ground acceleration as $1/2$ of SSE, and apply factors of 1.7 and 1.0 for OBE and SSE, respectively, one obtains peak response of a single degree of freedom structure at 9 Hz:

$$\text{OBE: } (2.92)(1.7)/2 = 2.48$$

$$\text{SSE: } 2.27$$

Similarly at 2.5 Hz, the response values for OBE and SSE are 2.98 and 2.72, respectively. The indications are that even with a factor of 1.7 on OBE, the response would still exceed the SSE. If additional increases in allowables are considered, as well as plastic redistribution of stresses, the OBE set at $1/2$ SSE is still the controlling load case for earthquake. Similarly,

other OBE load factors should remain as provided in ACI-349.

3.6.1.2 Other Load Factors (Position 6)

Currently Regulatory Guide 1.142 uses modified (higher) load factors for certain pressure and temperature loads. These factors were based on the application of ACI-318 philosophy to ACI-349 at a time when ACI-349 was in its initial stages. It is suggested that the modifications to these load factors be maintained in Regulatory Guide 1.142.

3.6.1.3 Ductile Detailing (Position 3)

This item is discussed in depth in Section 3.5.1.1 and this position should be amended as discussed in that section.

3.6.1.4 Differential Settlement (Position 8)

ACI-349-97 includes settlement stresses in load combinations Number 4 thru 11. Therefore this position is no longer required as it is addressed in the Code.

3.6.1.5 Use of "F" Load for Hydrostatic Loadings (Position 7)

In ACI-349-97 new notation for liquid pressures, "F", was added and lateral earth pressure, H, was added to load combinations. It further requires that the F load be considered in the lateral earth pressure. Therefore this item (concern) is now addressed in the current Code and this position is no longer required.

3.6.1.6 Radiation Shielding (Position 2)

The current revision of ANS 6.40 is 1997 which was reviewed and found to be applicable. (See Section 3.4) Therefore this latest edition of the standard should be referenced in Regulatory Guide 1.142.

3.6.2 Issues From Advance Reactor Design Criteria

The main document reviewed for the Advanced Reactor design was SECY-93-087 (7.2-[1]). The only issue which has any significant effect on ACI-349 is revised seismic design criteria which are discussed in depth in Section 3.8.1.

3.6.3 SRP Sections

The SRP sections related to concrete structure designs are 3.8.4 and primarily reference Regulatory Guide 1.142 and ACI-349 for concrete structures. The load combinations and capacity criteria are similar to that given in ACI-349. The only difference is the use of different factors for the OBE load case as discussed in 3.6.1 above. Therefore, there are no significant items or changes associated with the SRP.

3.7 Summary of Operating Nuclear Power Plant Design Criteria

A study of the current licensing basis of a sampling of operating nuclear power plants was conducted. This was conducted by reviewing the Final Safety Analysis Reports and updated Final Safety Analysis Reports for the operating commercial nuclear power facilities listed in Table 3.7-1.

The design basis concrete design commitments for the sampled plants was ACI-318 not ACI-349. As previously discussed, the scope of this study excludes consideration of ACI-349, Appendix B. The actual loads, load combinations and load factors used in these designs are as given in the Safety Analysis Reports. The following subsections provide a discussion and comparison of the loads, load combinations and load factors used in licensing these plants to the loads, load combinations, and load factors given in ACI-349-90.

The overall assessment of the compliance of the criteria used to design these plants and ACI-349-1990 are given in Table 3.7-2. The general conclusion is that the facilities that reference ACI-318-71 as the licensing basis, used design criteria that are either consistent with or more conservative than ACI-349-1990.

3.7.1 Loading Definitions

The loadings in the individual plant FSAR's and ACI-349-1990 are defined as follows:

U	= required strength to resist factored loads or related internal moments and forces	B	= loads generated from postulated blast along transportation routes
D	= dead loads, or related internal moments and forces, including piping and equipment dead loads	B _i	= Buoyancy forces, (normal, flood, and extreme)
L	= live loads, or related internal moments and forces	E, E _o	= load effects of Operating Basis Earthquake (OBE), or related internal moments and forces, including OBE-induced piping and equipment reactions
L'	= Deadweight from equipment	E', E _{ss}	= load effects of safe shutdown earthquake (SSE), or related internal moments and forces, including SSE-induced piping and equipment reactions
		F	= lateral and vertical pressure of liquids or related internal moments and forces, may include loads from flood.
		F*	= (Washington) lateral and vertical pressure of liquid including seismic effects due to internal flooding.
		H, Q	= lateral earth pressure, or related internal moments and forces.
		Q*	= (Washington) lateral earth pressure including seismic effects due to dry or saturated soils.
		M	= Internal missile loads; includes jet impingement
		N	= loads generated by the probable maximum precipitation
		P _a	= differential pressure load, or related internal moments and forces, generated by a postulated pipe break.
		P _o	= Normal operating external pressure differential pressure loads resulting from pressure variation either inside or outside the containment.
		P'	= Negative or positive pressure acting only in the Reactor Building in conjunction with the design basis tornado or SSE
		Q	= (Waterford) Static and Dynamic loads from equipment or pipe accident.
		R _a	= piping and equipment reactions, or

related internal moments and forces, under thermal conditions generated by a postulated pipe break and including R_o

- R_o, A = piping and equipment reactions or related internal moments and forces, which occur under normal operating and shutdown conditions, excluding dead load and earthquake reactions
- S, S' = Earth loads at rest and during an earthquake respectively.
- T_a = internal moments and forces caused by temperature distributions within the concrete structure occurring as a result of accident conditions generated by a postulated pipe break and including T_o
- T_o = internal moments and forces caused by temperature distributions within the concrete structure occurring as a result of normal operating or shutdown conditions
- W = operating basis wind load (OBW), or related internal moments and forces
- W_o, W' = loads generated by the design basis tornado (DBT), or related internal moments and forces. These include loads due to tornado wind pressure, tornado created differential pressures, and tornado generated missiles
- Y_j = jet impingement load, or related internal moments and forces, on the structure generated by a postulated pipe break
- Y_m = missile impact load, or related internal moments and forces, on the structure generated by a postulated pipe break, such as pipe whip
- Y_r = loads, or related internal moments and forces, on the structure generated by the reaction of the broken pipe during a postulated break

3.7.2 Beaver Valley Unit 2

Beaver Valley Unit 2 is designed to the strength provisions of ACI-318-71. The accident loading combinations controlled the design of the structures.

The FSAR for Beaver Valley Unit 2 states that "these equations are equal to or more conservative than their counterparts in ACI-349 or SRP 3.8.3".

Table 3.7-3 provides a comparison between the FSAR load case provisions and ACI-349-1990. Load cases 1 to 10 include the Containment Internal Structures. These structures are not subject to fluid loadings, F , or lateral earth loads H . Other concrete structures are designed to the earthquake load cases (3, 4, 5, and 6). For these other structures, loads T_a , R_o , P_a , Y_r , Y_j and Y_m are zero. Load cases 11 to 15 include structures that may be subjected to wind, tornado wind and floods. As shown in the table, the FSAR is "equal" or more conservative depending on the proportion of each loading phenomena.

There is one load case in ACI-349-1990 not listed in Table 3.7.2. This load case is $U = 1.05D + 1.05F + 1.3L + 1.3H + 1.05T_o + 1.3R_o$. Normally this load case is enveloped by load case 10. Load case 10 also includes $1.3E_o$.

As indicated in Table 3.7-3, there are four load conditions in the FSAR that are more conservative and two load conditions that are less conservative than ACI-349-1990.

The Beaver Valley Unit 2 FSAR load combinations are "equal to or more conservative than their counterparts in ACI-349 or SRP 3.8.3" provided T and R are small and that the earthquake loadings govern the design of the structures as opposed to wind, tornado wind or flood. This is normally the case for heavy nuclear concrete structures. Since the load cases that normally control design of nuclear structures are more conservative in the Beaver Valley FSAR, it is concluded that the FSAR is more conservative than ACI-349-1990

3.7.3 Alvin W. Vogtle Unit 2

Alvin W. Vogtle Unit 2 (VEGP) is designed to the strength provisions of ACI-318-71 including the 1974 supplement. The FSAR for VEGP states that "the differences between ACI-349 and ACI-318 are minor except for the load combination equations which, in the case of VEGP, are in conformance with the Standard Review Plan. Thus, the design procedures and construction practices delineated in the FSAR ensure that the structure will withstand the specified design conditions, without impairment of structural integrity or the performance of required safety functions."

After review of the Vogtle Unit 2 FSAR load combinations, it is presumed that F (normal Fluid Load) and H (Earth Loads) are either zero for Vogtle structures or included with dead load and live load respectively. Table 3.7-4 presents a comparison of the load cases from the FSAR and the closest ACI-349-90 load case.

The differences between the Vogtle Unit 2 FSAR and ACI-349 load combinations are "minor" provided R is small, and that F (normal fluid load) and H (earth loads) are either zero or are included with dead load and live load respectively. The inclusion of these loads with live and dead load is a common practice for structures evaluated to ACI-318-71. As indicated in Table 3.7-4, there are five load cases where the FSAR is more conservative than ACI-349-1990. Therefore, it is concluded that the licensing basis for Vogtle Unit 2 is more conservative than ACI-349-1990.

3.7.4 Washington Public Steam Supply System Unit #2

Washington Public Steam Supply Systems Unit #2 (WNP-2) is designed to the strength provisions of ACI 318-71. The FSAR does not refer to ACI-349 and does not claim to be in compliance with this standard. However, the load cases included in the WNP-2 FSAR are very similar to the ACI-349 formulations as shown in Table 3.7-5. The equation numbers in Table 3.7-5 correspond to the load combination number in Table 3.8-15 of the WNP-2 FSAR. The loading F at WNP includes water pressure, vertical and horizontal, due to internal flooding of pump rooms in the basement of the reactor building only. Loads F* and Q* are internal flooding loads and earth pressure loads including seismic effects. Normal liquid load, F, as defined in ACI-349 is presumed to be included with dead load or equal to zero.

There are two load cases in ACI-349-1990 not listed in Table 3.7-4. These load cases are:

$$U = 1.4 D + 1.4 F + 1.7 L + 1.7 H + 1.7 W + 1.7 R_o$$

and

$$U = 1.05 D + 1.05 F + 1.3 L + 1.3 H + 1.05 T_o + 1.3 R_o$$

The first load case is normally enveloped by the tornado wind load or OBE. The second is normally enveloped by load case 2b and U8. As indicated in Table 3.7-5, the load cases in the FSAR are in the majority of cases

equivalent to ACI-349.

3.7.5 Waterford Unit #3

Waterford Unit 3 (WSES-3) is designed to the strength provision of ACI-318-63. The FSAR does not refer to ACI-349 and does not claim to be in compliance with this standard. However, the load cases included in the WSES-3 FSAR include load conditions for the extreme and severe environmental loadings included in the ACI-349 formulations. The main differences are that the normal dead load and live load factor in ACI-318-63 are 1.5 and 1.8 as opposed to 1.4 and 1.7 in ACI-318-71 and ACI-349-1990. The load factors for the severe earthquake (OBE) and wind are in general not as conservative as in ACI-349-1990. These differences are illustrated in the Table 3.7-6. The equation numbers shown in that table correspond to the load combination number in Table 3.8-39 of the WSES-3 FSAR. The FSAR load cases use different nomenclature in comparison to ACI-349.

As indicated in table 3.7-6, there are about the same number of load conditions in the FSAR that are conservative and unconservative in comparison to ACI-349-1990. However, several load conditions that normally control design are less conservative in the FSAR than ACI-349. Therefore, it is concluded that the design criteria for Waterford Unit #3 is less conservative than ACI-349-1990.

3.7.6 Perry Unit #1

Perry Unit 1 is designed to the strength provisions of ACI-318-71. However, the FSAR for Perry Unit 1 states that "ACI-318-71 which was used for concrete structures on the PNPP and ACI-349-76 are quite similar. For this reason, PNPP is, in general, in compliance with the requirements of ACI-349-76." Section 3.8.3.3.7 on the PNPP FSAR provides a summary of specific sections where ACI-349 differs from ACI-318 and how PNPP complies.

With regard to load factors the section states that the load combinations specified in the PNPP FSAR meet or exceed the requirements of ACI-349. "Only the load factor of 1.7 specified for R_o exceed the 1.3 factor used for the same term on PNPP. This is not a controlling case, however, because on PNPP thermal has been combined with R_o and because of the higher factor used on the steam relief valve reactions." Table 3.7-7 includes a summary of the PNPP load cases and the load cases from ACI-349-1990. As shown in Table 3.7-7 the

FSAR load cases at PNPP Unit 1 are in general conservative in comparison to ACI 349-1990.

3.7.7 Indian Point Unit #3

The Indian Point Unit 3 FSAR does not reference the ACI Code in Section 16.4, Details of Structural Design. The FSAR describes the method of design, stress analysis, critical stresses and locations.

The description in the FSAR suggests implies that strength design methods were used for reinforced concrete. The following paragraphs describe the design criteria and loading controlling design.

Containment Interior Structures

The .91m (3 ft.) thick fill slab controlling loads occur from piping support reactions due to postulated pipe breaks. Stresses in the reinforcement were limited to F_y . Maximum stresses occur immediately below the primary equipment supports.

The .91m (3 ft.) thick crane wall was designed for a 48kPa (7psi) differential pressure after a primary pipe break. Stresses in the reinforcing steel were limited to 0.9 of yield but stresses were much lower 89,632kPa (13 ksi) for hoop reinforcement and 21kPa (3psi) for vertical. Local areas in the wall subjected to jet forces were evaluated. The analysis determined that minor yield of reinforcement would occur in local areas. Acceptability of the crane wall was also demonstrated using strength yield line analysis. The wall has the capacity to resist jet load and differential pressure simultaneously. The wall was also evaluated for earthquake loadings, however, this did not govern design.

The 1.22m (4 ft) to 1.83m (6 ft) thick refueling canal was designed for the 48kPa (7psi) differential pressure. Stresses in the reinforcing steel were limited to .9 F_y . As in the case of the crane wall some local yielding of reinforcement would occur from jet loads and earthquake loadings did not govern design.

Due to its many openings the .61m (2 ft.) thick operating floor slab is designed as a series of beams. Principal loadings are DL and LL and pressure differential. The DL and LL were designed with Part IV-B of ACI-318-63. This is the ultimate strength provision in ACI-318-63 that conservatively uses a 1.5 and 1.8 load factor for dead and live load respectively. Stresses in the reinforcing steel for pressure differential

were limited to 0.9 F_y . The floor was also evaluated for jet forces and the reinforcement was within the 0.9 F_y stress limit.

The primary shield wall or containment is a 1.83m (6 ft.) thick circular reinforced concrete shell and is designed to sustain internal pressure and provide missile protection. The maximum reinforcing steel stress was limited to .95 times its strength. The containment was also evaluated for thermal conditions but these loads were small in comparison to pressure.

The containment structure, Primary Auxiliary Building (PAB), Control Building, Fuel Storage Building and Intake Structure were evaluated for tornado and earthquake loadings. The Spent Fuel Pit was also evaluated for thermal stresses. The FSAR does not elaborate with regard to the design equations for these evaluations.

It is concluded that due to the high limits on the reinforcement steel stress used by the evaluation for IP-3, the IP-3 criteria is less conservative than the ACI-349-1990 criteria.

3.7.8 Palisades

The Palisades concrete structures were originally designed to the strength provisions of ACI 318-63. Subsequently, additions to existing concrete structures have been designed and built to the requirements of ACI-318-71 and ACI-318-77 depending on the year of the addition. The FSAR does not refer to ACI-349 and does not claim to be in compliance with this standard. However, the loads included in the Palisades FSAR include load conditions for the extreme and severe environmental loadings included in the ACI-349 formulations. These loads include the 45m/s (100 mph) design wind loadings, 134m/s (300 mph) tornado wind, tornado and internal missile loadings, high and moderate energy pipe break loads and loadings from the OBE and SSE.

The main difference between the codes is that the normal dead load and live load factor in ACI-318-63 are 1.5 and 1.8 as opposed to 1.4 and 1.7 in ACI-318-71 and ACI-349-1990. The Palisades FSAR did not include the load combinations actually used and therefore, the degree of conservatism of the Palisades evaluations could not be addressed. However, the FSAR criteria is judged to be not as conservative as ACI-349 in that several loadings and load combinations are not normally considered for plants licensed to the

ACI-318-63 criteria as discussed in Section 3.7.5 for Waterford, Section 3.7.7 for Indian Point, Section 3.7.9 for Davis Besse and 3.7.10 for Quad Cities.

3.7.9 Davis-Besse

Davis-Besse is designed to the strength provisions of ACI-318-63. The FSAR does not refer to ACI-349 and does not claim to be in compliance with this standard. However, the load cases included in the Davis-Besse FSAR include load conditions for the extreme and severe environmental loadings included in the ACI-349 formulations. The main differences are that the normal dead load and live load factor in ACI-318-63 are 1.5 and 1.8 as opposed to 1.4 and 1.7 in ACI-318-71 and ACI-349-1990. The load factors for the severe earthquake (OBE) and wind are in general not as conservative as in ACI-349-1990. These differences are illustrated in Table 3.7-6. The FSAR load cases uses different nomenclature in comparison to ACI-349 as discussed in the table. As indicated in Table 3.7-6 there are several ACI-349 load cases that are not included in the FSAR load cases. Therefore, it is concluded that the licensing criteria for Davis-Besse is less conservative than ACI 349-1990.

3.7.10 Quad Cities

Quad Cities is in general designed to the working stress provisions of ACI-318-63. The load factors for the working stress loading are 1.0. The steel allowable stress is $0.5 F_y$ and the allowable stress for concrete is $0.45 f'_c$. Working stress design methods normally result in safe designs in comparison to strength methods. However, the degree of conservatism is dependent on several factors including the ratio of dead to live load and the percent reinforcement of the structural members.

The FSAR does not refer to ACI-349 and does not claim to be in compliance with this standard. However, the loads included in the Quad Cities FSAR include load conditions for the extreme and severe environmental loadings included in the ACI-349 formulations. These loads include jet force or pressure due to rupture of pipes, the OBE, SSE and 49m/s (110 mph) wind with gusts up to 89m/s (200 mph). Normal allowables are used for the OBE design case. Allowable loads for the wind case are $.667 F_y$ and $0.60 f'_c$. The FSAR allows use of the strength design of the structure for the SSE with Load Factor taken equal to 1.0.

Review of the stress analysis results indicate that the

Quad Cities structures have sufficient margin over the allowable loading that the structures are most likely adequate for the ACI-349-1990 load combinations. However, the FSAR criteria is not as conservative as ACI-349 in that several loadings and load combinations have not been considered.

3.7.11 Summary

Table 3.7-2 provides an overview assessment of the licensing basis criteria of these plants in comparison to ACI-349-90. This summary identifies the plants that have a similar, more conservative, or less conservative design basis in comparison to ACI-349-90. This provides some insight into the general level of conservatism (and margin) which exists in the current operating plants and provides insight into the affect on these plants should they decide to implement the updated Regulatory Guide 1.142 in their licensing basis. Also, design basis and load combination will be considered when developing the final recommendations for Regulatory Guide 1.142.

The general conclusion in Table 3.7-2 is that the facilities that reference ACI-318-71 as the licensing basis, used design criteria that are either consistent with or more conservative than ACI-349-1990. The facilities that reference ACI-318-63 as the licensing basis, used design criteria that are less conservative than ACI-349-1990. The primary reason for this distinction is that potential loadings and the appropriate load factors and combinations were not fully developed for the vintage of plants that referenced ACI-318-63. Newer vintage plants that referenced ACI-318-71 were licensed after the basis for the load factors and combinations were established. This basis has undergone relatively minor revision since the publication of ACI-318-71.

Table 3.7-1 - Plants Whose Concrete Design Basis Was Reviewed		
Plant	Owner	Plant Type Containment
Beaver Valley Unit 2	Duquesne Light Co.	W-PWR (3)
Alvin W. Vogtle Unit 3	Georgia Power	W-PWR (4)
Washington Public Steam Supply Stem Unit #2	Washington Public Power	GE-BWR-S (11)
Waterford Unit #3	Louisiana Power and Light Co.	BW-PWR
Perry Unit #1	Cleveland Electric Illuminating Co.	GE-BWR (6) III
Indian Point Unit #3	Consolidated Edison	W-PWR (4)
Palisades	Consumers Power	CE-PWR-600
Davis-Besse	Toledo Edison Electric Company	CE-PWR-900
Quad Cities	Commonwealth Edison	GE-BWR-4 (I)

Table 3.7-2 Overall Assessment of Existing Plant Licensing Basis In Comparison to ACI-349-90		
Plant	Primary Licensing Basis	General Assessment
Beaver Valley Unit 1 and 2	ACI-318-71 Strength	More Conservative
Alvin W. Vogtle Unit 1 and 2	ACI-318-71 Strength	More Conservative
Washington Public Steam Supply Stem Unit #2	ACI-318-71 Strength	Consistent With
Waterford Unit #3	ACI-318-63. Strength	Less Conservative
Perry Unit #1	ACI-318-71 Strength	More Conservative
Indian Point Unit #3	ACI-318-63 Strength	Consistent With
Palisades	ACI-318-63; Strength	Less Conservative
Davis-Besse	ACI-318-63; Strength	Less Conservative
Quad Cities	ACI-318-63; Working Stress	Less Conservative

Table 3.7-3 Beaver Valley Unit #2 Review			
EQN. No.	FSAR Load Combination Beaver Valley Unit 2	Corresponding ACI-349-1990 Load Combination	Discussion
1	$U=0.75 (1.4 D + 1.7 CL)$	-	This load case includes construction loadings. Although there is no explicit ACI-349 load case, Construction Loads (CL) need to be considered.
2	$U=1.4 D + 1.7L$	$U=1.4 D + 1.4 F + 1.7L + 1.7 H + 1.7 R_o$	FSAR equivalent to ACI-349 provided normal load R_o is small. F and H are zero.
3	$U=1.4D + 1.7L + 1.9 E$	$U=1.4 D + 1.4F + 1.7L + 1.7H + 1.7 E_o + 1.7 R_o$	Load Factor for 1/2 SSE (E) more conservative in FSAR. FSAR equivalent or more conservative than ACI-349 provided normal load R_o is small. F and H are zero.
4	$U=0.9 D + 1.4 E$	-	This load case includes a reduction in gravity load for members when gravity load reduces the overall stress for 1/2 SSE.
5	$U=1.0 D + 1.0 L + 1.0 E'$	$U=D + F + L + H + T_o + R_o + E_{ss}$	FSAR is equivalent to ACI-349 provided load T_o and R_o are small. F and H are zero.
6.	$U=0.9 D + 1.0 E'$	-	This load case includes a reduction in gravity load for members when gravity load reduces the overall stress for SSE.
7	$U=1.0 D + 1.0 L + 1.0 T_a + 1.0 R_a + 1.5 P_a$	$U=D + F + L + H + T_a + R_a + 1.25 P_a$	Load factor for P_a more conservative in the FSAR. F and H are zero. FSAR more conservative than ACI-349.
8	$U=1.0 D + 1.0 L + 1.0 T_a + 1.0 R_a + 1.25 P_a + 1.0 (Y_r + Y_j + Y_m) + 1.0 E'$	$U=D + F + L + H + T_a + R_a + 1.0 P_a + 1.0 (Y_r + Y_j + Y_m) + 1.0 E_{ss}$	Load factor for P_a more conservative in the FSAR. F and H are zero. FSAR more conservative than ACI-349.
9	$U=1.0 D + 1.0 L + 1.0 T_a + 1.0 R_a + 1.25 P_a + 1.0 (Y_r + Y_s + Y_m) + 1.25 E$	$U=D + F + L + H + T_a + R_a + 1.15 P_a + 1.0 (Y_r + Y_j + Y_m) + 1.15 E_o$	FSAR load factor for E and T_o more conservative and load factor for R_a is less conservative in the FSAR. F&H are zero. Normally, the FSAR is more conservative than ACI-349.

Table 3.7-3 Beaver Valley Unit #2 Review (Continued)			
EQN. No.	FSAR Load Combination Beaver Valley Unit 2	Corresponding ACI-349-1990 Load Combination	Discussion
10	$U=0.75 (1.4D + 1.7L + 1.9E + 1.7 T_o + 1.7 R_o)$	$U=1.05D + 1.05F + 1.3L + 1.3H + 1.3E_o + 1.05 T_o + 1.3 R_o$	Load factor for E and T_o more conservative and load factor for R_o less conservative in the FSAR than ACI-349. F and H are zero. Normally the FSAR is more conservative.
11	$U=0.75 (1.4D + 1.7L + 1.7W)$	$U=1.4D + 1.4F + 1.7L + 1.7 H + 1.7 W + 1.7 R_o$	FSAR less conservative than ACI-349. F (normal load), H, and R_o are zero.
12	$U=0.9D + 1.3W$	—	This load case includes a reduction in gravity load for members when gravity load reduces the overall stress for design wind.
13	$U=1.0D + 1.0L + 1.0 W_t$	$U=D + F + L + H + T_o + R_o + W_t$	FSAR essentially the same as ACI-349. F (normal load), H, T_o and R_o are zero.
14	$U=0.9D + 1.0 W_t$	—	This load case includes a reduction in gravity load for members when gravity load reduces the overall stress for tornado wind.
15	$U=1.0D + 1.0F$	$U=D + F + L + H + T_o + R_o + W_t$	F for the FSAR load case is for flood. To be consistent with ACI-349, live load, L, should be included with the load case. H, T_o and R_o are zero. W_t need not be included with the flood load. However, the FSAR load case is not equivalent to ACI-349.

Table 3.7-4 Alvin W. Vogtle Unit 2			
EQN. No.	FSAR Load Combination Vogtle-Unit 3	Corresponding ACI-349-1990 Load Combination	Discussion
1	$U=1.4D + 1.7L$	$U=1.4D + 1.4F + 1.7L + 1.7 H + 1.7 R_o$	FSAR case is equivalent to ACI-349 provided R_o is small. F and H are zero.
2	$U=1.4D + 1.7L + 1.7W$	$U=1.4D + 1.4F + 1.7L + 1.7H + 1.7 W + 1.7R_o$	FSAR case is equivalent to ACI-349 provided R_o is small. F and H are zero.
3	$U=1.4D + 1.7L + 1.9E$	$U=1.4D + 1.4F + 1.7L + 1.7H + 1.7E_o + 1.7R_o$	The load factor in the FSAR for OBE (E) is conservative. FSAR normally conservative in comparison to ACI-349 provided R_o is small. F and H are zero.
4	$U=1.05D + 1.275L + 1.275 T_o + 1.275 R_o$	$U=1.05 D + 1.05F + 1.3L + 1.3H + 1.05 T_o + 1.3 R_o$	Load factor in the FSAR for L and R_o slightly less than ACI-349 (negligible). Load factor for T_o is greater in the FSAR. FSAR normally conservative compared to ACI-349. F and H are zero.
5	$U=1.05D + 1.275L + 1.275 T_o + 1.275W + 1.275 R_o$	$U=1.05 D + 1.05F + 1.3L + 1.3H + 1.3W + 1.05 T_o + 1.3 R_o$	Load factor in the FSAR for L T_o , W and R_o slightly less than ACI-349 (negligible). FSAR equivalent to ACI-349. F and H are zero.
6.	$U=1.05D + 1.275L + 1.275 T_o + 1.425E + 1.275 R_o$	$U=1.05 D + 1.05F + 1.3L + 1.3H + 1.3E_o + 1.05 T_o + 1.3 R_o$	Load factor in the FSAR for L, T_o and R_o slightly less than ACI-349 (negligible). Load factor for T_o and the E is greater in the FSAR. FSAR normally conservative compared to ACI-349. F and H are zero.
7	$U=1.0D + 1.0L + 1.0 T_o + 1.0 E' + 1.0 R_o$	$U= D + F + L + H + T_o + R_o + E_{ss}$	FSAR case is equivalent to ACI-349. F and H are zero.
8	$U=1.0D + 1.0L + 1.0 T_o + 1.0 W_t + 1.0 R_o$	$U= D + F + L + H + T_o + R_o + W_t$	FSAR case is equivalent to ACI-349. F and H are zero.
9	$U=1.0D + 1.0L + 1.5 P_a + 1.0 T_a + 1.0 R_a$	$U= D + F + L + H + T_a + R_a + 1.25 P_a$	Load factor in the FSAR for P_a is greater than ACI-349. FSAR normally conservative compared to ACI-349. F and H are zero
10	$U=1.0 D + 1.0 L + 1.0 T_a + 1.0 R_a + 1.25 P_a + 1.0 (Y_r + Y_j + Y_m) + 1.25 E$	$U=D + F + L + H + T_a + R_a + 1.15 P_a + 1.0 (Y_r + Y_j + Y_m) + 1.15 E_o$	Load factor in FSAR for P_a and E is greater than ACI-349. FSAR case is conservative in comparison to ACI-349. F and H are zero.

Table 3.7-4 Alvin W. Vogtle Unit 2 (Continued)			
EQN. No.	FSAR Load Combination Vogtle-Unit 3	Corresponding ACI-349-1990 Load Combination	Discussion
11	$U=1.0 D + 1.0 L + 1.0 T_a + 1.0 R_a + 1.0 P_a + 1.0 (Y_r + Y_s + Y_m) + 1.0 E'$	$U=D + F + L + H + T_a + R_a + 1.0 P_a + 1.0 (Y_r + Y_j + Y_m) + 1.0 E_{SS}$	FSAR case is equivalent to ACI-349. F and H are zero.
12	$U=1.0 D + 1.0 L + 1.0 T_o + 1.0 R_o + 1.0 B$	—	Appropriate load case for considering blast loading, B.
13	$U=1.0 D + 1.0 L + 1.0 T_o + 1.0 R_o + 1.0 N$	—	Appropriate load case for considering maximum hypothetical precipitation load, N.

Table 3.7-5 Washington Public Power Steam Supply Systems Unit #2			
EQN. No.	FSAR Load Combination WNP-2	Corresponding ACI-349-1990 Load Combination	Discussion
U1	$U=1.4 D$	---	This load case includes construction loadings. Although there is no explicit ACI-349 load case, construction loads need to be considered.
U2	$U=.9 D + 1.3 L + .9 R_o + 1.3 T_o + 1.3 W$	---	This load case includes construction loadings. Although there is no explicit ACI-349 load case, construction loads need to be considered.
1	$U=1.4 D + 1.7 L + 1.7 P_o$	$U=1.4 D + 1.4 F + 1.7 L + 1.7 H + 1.7 R_o$	This load case enveloped by load case 1b.
1b	$U= 1.4 D + 1.7 L + 1.4 R_o + 1.4 T_o + 1.7 P_o$	$U=1.4 D + 1.4 F + 1.7 L + 1.7 H + 1.7 R_o$	F (ACI-349) presumed to be zero. Load factor for R_o in FSAR less than in ACI-349. T_o included in FSAR but not ACI-349. H included as Q in equation U3. P_o (FSAR) included as L in ACI-349. FSAR normally conservative for most structural members.
U3	$U=1.4 D + 1.7 L + 1.7 P_o + 1.4 F + 1.7 Q$	$U= 1.4 D + 1.4 F + 1.7 L + 1.7 H + 1.7 R_o$	This load case enveloped by load case 1b.
U4.	$U=1.4 D + 1.7 L + 1.4 T_o + 1.7 P_o + 1.4 F + 1.7 Q$	$U= 1.4 D + 1.4 F + 1.7 L + 1.7 H + 1.7 R_o$	F (FSAR) is internal flooding. $H(ACI-349)=Q(FSAR)$. R_o not included in FSAR. T_o included in this equation in the FSAR but not in ACI-349. FSAR equivalent to ACI-349 provided $T_o \geq R_o$.
U5	$U=0.9 D + 1.4 F + 1.7 Q$	----	This load case includes a reduction in gravity load for members where gravity load reduces member stress.
U6	$U=0.9 D + 1.4 T_o + 1.4 F + 1.7 Q$	----	This load case includes a reduction in gravity load for members where gravity load reduces member stress.

Table 3.7-5 Washington Public Power Steam Supply Systems Unit #2 (Continued)			
EQN. No.	FSAR Load Combination WNP-2	Corresponding ACI-349-1990 Load Combination	Discussion
2	$U = 1.4 D + 1.7 L + 1.7 P_o + 1.9 E$	$U = 1.4 D + 1.4 F + 1.7 L + 1.7 H + 1.7 R_o + 1.7 E_o$	P_o (FSAR) included as L in ACI-349. Load factor for E in FSAR greater than ACI-349. F is presumed to be zero or included with D. R_o not included in the FSAR load case. FSAR or equivalent to ACI-349 provided R_o is small.
2b	$U = 1.4 D + 1.4 L + 1.4 R_o + 1.4 T_o + 1.4 P_o + 1.4 E$	---	This FSAR equation a mix of ACI-349 combinations discussed in this table.
2b'	$U = 0.9 D + 1.4 E$	---	This load case includes a reduction in gravity load for members where gravity load reduces member stress.
3	$U = 1.1 D + 1.3 L + 1.3 P_o + 1.3 W$	$U = 1.05 D + 1.05 F + 1.3 L + 1.3 H + 1.3 R_o + 1.3 T_o + 1.3 W$	This load case enveloped by case 3b.
3b	$U = 1.1 D + 1.3 L + 1.3 P_o + 1.1 R_o + 1.1 T_o + 1.3 W$	$U = 1.05 D + 1.05 F + 1.3 L + 1.3 H + 1.3 R_o + 1.3 T_o + 1.3 W$	P_o (FSAR) = H (ACI-349). Load factor for D in FSAR conservative. F presumed to be included with D. Load factor in FSAR for R_o and T_o less than ACI-349. Load case slightly unconservative in comparison to ACI-349.
3b'	$U = 0.9 D + 1.3 W$	---	This load case includes a reduction in gravity load for members where gravity load reduces member stress.
U7	$U = 1.1 D + 1.3 L + 1.1 F + 1.3 Q + 1.3 P_o + 1.3 W$	$U = 1.05 D + 1.05 F + 1.3 L + 1.3 H + 1.3 W + 1.3 R_o + 1.3 T_o$	This load case enveloped by case U8.
U8	$U = 1.1 D + 1.3 L + 1.1 F + 1.3 Q + 1.3 P_o + 1.3 W + 1.1 T_o$	$U = 1.05 D + 1.05 F + 1.3 L + 1.3 H + 1.3 W + 1.3 T_o + 1.3 R_o$	Q (FSAR) = H (ACI-349). FSAR includes internal flood issue at WPPS. Load factor for D and F in FSAR conservative. FSAR equation does not include R_o . FSAR load factor less than ACI-349 for T_o . FSAR load case equivalent to ACI-349.

Table 3.7-5 Washington Public Power Steam Supply Systems Unit #2 (Continued)			
EQN. No.	FSAR Load Combination WNP-2	Corresponding ACI-349-1990 Load Combination	Discussion
U9	$U=1.4D + 1.7L + 1.7 P_o + 1.9 E + 1.4F^* + 1.4Q^*$	$U=1.4D + 1.4F + 1.7L + 1.7 H + 1.7 E_o + 1.7 R_o$	$Q(FSAR)= H(ACI-349)$. FSAR includes the internal flood issue at WPPS. FSAR load factor for E conservative. FSAR normally conservative provided $P_o \geq R_o$.
U10	$U=1.4D + 1.4L + 1.4 T_o + 1.4 P_o + 1.4 E + 1.4F^* + 1.4Q^*$	$U=1.4D + 1.4F + 1.7L + 1.7 H + 1.7 E_o + 1.7 R_o$	$Q(FSAR) = H(ACI-349)$. FSAR includes the internal flood issue at WPPS. Load factor for L and E in FSAR less conservative than ACI-349.
4	$U=1.0D + 1.0L + 1.0 R_o + 1.0 T_o + 1.0E'$	$U=D + F + L + H + T_o + R_o + E_{ss}$	F and H presumed to be zero or included with D and L. FSAR load case equivalent to ACI-349.
6	$U=1.0 D + 1.0 L + 1.5 P_a + T_a + R_t$	$U=D + F + L + H + T_a + R_a + 1.25 P_a$	Load factor for P_a in FSAR conservative. F and H are presumed to be zero or included with D and L. FSAR is normally conservative in comparison to ACI-349.
7	$U=1.0D + 1.0L + 1.25 P_a + 1.25 E + 1.0 T_a + 1.0 R_t + 1.0R_i$	$U=D + F + L + H + T_a + R_a + 1.15 P_a + 1.0 (Y_r + Y_j + Y_m) + 1.15 E_o$	FSAR load case R_t and R_i includes loads Y_r , Y_s , Y_m , and R_m . F and H are presumed to be zero or included with D and L. FSAR load factor for P_a and E_o is greater than ACI-349. FSAR normally conservative in comparison to ACI-349.
8	$U=1.0D + 1.0L + 1.0 P_a + 1.0 E' + 1.0 T_a + 1.0 R_t + 1.0 R_i$	$U=D + F + L + H + T_a + R_a + 1.0 P_a + 1.0 (Y_r + Y_j + Y_m) + 1.0 E_{ss}$	F and H are presumed to be zero or included with D and L. FSAR load case R_t and R_i includes loads Y_r , Y_j , Y_m , and R_m . FSAR equivalent to ACI-349.
U11	$U=1.0D + 1.0L + 1.0 R_o + 1.0 T_o + 1.0P' + 1.0E'$	$U=D + F + L + H + T_o + R_o + E_{ss}$	F and H are presumed to be zero or included with D and L. FSAR equivalent to ACI-349.
U11	$U=1.0D + 1.0L + 1.0 R_o + 1.0 T_o + 1.0P' + 1.0W'$	$U=D + F + L + H + T_o + R_o + W_t$	$P'(FSAR) = H(ACI-349)$. F is presumed to be zero or included with D. FSAR equivalent to ACI-349.

Table 3.7-6 Waterford Unit #3			
EQN. No.	FSAR Load Combination WSES-3	Corresponding ACI-349-1990 Load Combination	Discussion
1	$U = 1.5(D + L' + T) + 1.8(L + A) + 1.0 B$	$U = 1.4 D + 1.4 F + 1.7 L + 1.7 H + 1.7 R_o$	L' (FSAR) a portion of D (ACI-349). A (FSAR) = R_o (ACI-349), T (FSAR) = T_o (ACI-349), B (FSAR) = F (ACI-349). FSAR load case is conservative in comparison to ACI-349.
2i	$U = 1.25(D + L' + A + E + T) + 1.0 B'' + 1.0 S'$	$U = 1.4 D + 1.4 F + 1.7 L + 1.7 H + 1.7 E_o + 1.7 R_o$	S' (FSAR) = H (ACI-349) L', A, T, B - Defined in 1. Load factors in FSAR less conservative than ACI-349.
2ii	$U = 0.9(D + L') + 1.1(E + T) + 1.0 B'' + 1.0 S'$	---	L', B'', S' defined in 1 and 2. This load case enveloped by 2i.
3	$U = (1.0 \pm 0.10)(D + L' + A + T) + 1.0 (E' + B'')$	$U = D + F + L + H + T_o + R_o + E_{ss}$	L', A, B'' defined in 1 and 2. FSAR load case does not include L or F loads. FSAR load case does not meet ACI-349 requirements.
4i	$U = 1.25(D + L' + T + H_U + A) + 1.0(B'' + S)$	$U = 1.05 D + 1.05 F + 1.3 L + 1.3 H + 1.05 T_o + 1.3 R_o$	L', A, B'', S defined in 1 and 2. H_U (FSAR) = W (ACI-349). Load factors in FSAR less conservative than ACI-349.
4ii	$U = 0.9(D + L') + 1.1(H_U + T) + 1.0(B'' + S)$	---	L', B'', S defined in 1 and 2. H_U defined in 4i. Load case enveloped by 4i.
5	$U = (1.0 \pm 0.10)(D + L' + A + T) + 1.0 (B' + S)$	---	L', A, B'', S defined in 1 and 2. Load case enveloped by 6.
6	$U = (1.0 \pm 0.10)(D + L' + A + T) + 1.0 (W + B' + S) + 1.0 M$	$U = D + F + L + H + T_o + R_o + W_t$	L', A, B'', S defined in 1 and 2. M (FSAR) - Tornado missiles W (FSAR) = W_t (ACI-349) L not included in FSAR load case. Tornado missiles included in FSAR. Load case in FSAR equivalent to ACI-349.
7	$U = (1.0 \pm 0.10)(D + L' + A + S + T) + 1.25 Q + 1.0 B$	---	Q (FSAR) - pipe accident load, normally considered as done in load case 9b.
8a	$U = (1.0 \pm 0.05)(D + L' + T) + 1.5 LOCA + 1.25 A + 1.0 B$	----	L', A, B defined in 1 and 2. LOCA (FSAR) = T_a, P_a, R_a (ACI-349). Load case normally enveloped by 8b and 8c.

Table 3.7-6 Waterford Unit #3 (Continued)

EQN. No.	FSAR Load Combination WSES-3	Corresponding ACI-349-1990 Load Combination	Discussion
8b	$U = (1.0 \pm 0.05)(D + L' + T) + 1.25 \text{ LOCA} + 1.25 (E + A) + 1.0 B''$	$U = D + F + L + H + T_a + R_a + 1.15 P_a + 1.0 (Y_r + Y_j + Y_m) + 1.15 E_o$	L', A, B'' defined in 1 and 2. LOCA defined in 8a. FSAR load case conservative for E, A and LOCA. H and L not included in FSAR load case. Normally FSAR will be conservative.
8c	$U = (1.0 \pm 0.05)(D + L' + T) + A + 1.0(\text{LOCA} + E + B'')$	$U = D + F + L + H + T_a + R_a + 1.0 P_a + 1.0 (Y_r + Y_j + Y_m) + 1.0 E_{ss}$	L', A, B'' defined in 1 and 2. LOCA defined in 8a. FSAR equivalent to ACI-349.
8d	$U = (1.0 \pm 0.05)(D + L' + T) + 1.25(\text{LOCA} + H_u + A) + 1.0 B''$	$U = D + F + L + R_o + T_o + W_t$	L', A, B'' defined in 1 and 2. LOCA defined in 8a. H _u defined in 4i. FSAR normally conservative in comparison to ACI-349.
8e	$U = (1.0 \pm 0.05)(D + L' + T) + 1.25 \text{ IN(IP)} + 1.0B + 1.25 A$	---	IN (IP) Internal pressure for LOCA. Load case normally enveloped by 8b and 8c.
9a	$U = D + L + 1.5 P_a + T_a + R_a$	$U = D + F + L + H + T_a + R_a + 1.25 P_a$	Load factor for P _a conservative in comparison to ACI-349. H and F not included in FSAR load case. Therefore, FSAR normally not as conservative as ACI-349.
9b	$U = D + L + 1.25 P_a + T_a + R_a + 1.0 (Y_r + Y_j + Y_m) + 1.25 E$	$U = D + F + L + H + T_a + R_a + 1.15 P_a + 1.0 (Y_r + Y_j + Y_m) + 1.15 E_o$	Load factor in FSAR for P _a and E conservative. FSAR normally conservative in comparison to ACI-349.
9C	$U = D + L + 1.0 P_a + T_a + R_a + 1.0 (Y_r + Y_j + Y_m) + 1.0 E'$	$U = D + F + L + H + T_a + R_a + 1.0 P_a + 1.0 (Y_r + Y_j + Y_m) + 1.0 E_{ss}$	FSAR equivalent to ACI-349.
10	$U = (1.0 \pm 0.05)(D + L' + T) + 1.0AP + 1.0B$	----	AP (FSAR) load is a river accident load unique to WSES-3

Table 3.7-7 Perry Unit 1 (PNPP)			
EQN. No.	FSAR Load Combination Perry Unit 1 (PNPP)	Corresponding ACI-349-1990 Load Combination	Discussion
1	$U = 1.4D + 1.7L + 1.4G + 1.7H + 1.3T_o + 1.3R_o$	$U = 1.4D + 1.4F + 1.7L + 1.7H + 1.7R_o$ and $U = 1.05D + 1.05F + 1.3H + 1.05T_o + 1.3R_o$	$G(\text{FSAR}) = F(\text{ACI-349})$. FSAR includes T_o . FSAR normally conservative if $1.3T_o > .4R_o$.
2	$U = 1.4D + 1.7L + 1.4G + 1.7H + 1.7W + 1.3T_o + 1.3R_o$	$U = 1.4D + 1.4F + 1.7L + 1.7H + 1.7R_o$ and $U = 1.05D + 1.05F + 1.3L + 1.3H + 1.3W + 1.05T_o + 1.05R_o$	$G(\text{FSAR}) = F(\text{ACI-349})$. FSAR includes T_o . FSAR normally conservative if $1.3T_o > .4R_o$.
3	$U = 1.2D + 1.4G + 1.3H + 1.7W$	---	$G(\text{FSAR}) = F(\text{ACI-349})$. Load case enveloped by 2.
4	$U = 1.4D + 1.7L + 1.4G + 1.7H + 1.9F_{eqo} + 1.3T_o + 1.3R_o$	$U = 1.4D + 1.4F + 1.7L + 1.7H + 1.7E_o + 1.7R_o$ and $U = 1.05D + 1.05F + 1.3L + 1.3H + 1.3E_o + 1.05T_o + 1.05R_o$	$G(\text{FSAR}) = F(\text{ACI-349})$. $F_{eqo}(\text{FSAR}) = E_o(\text{ACI-349})$. FSAR includes T_o and is conservative for E_o . FSAR conservative if $1.3T_o + .2E_o > .4R_o$.
5	$U = 1.2D + 1.4G + 1.3H + 1.9F_{eqo}$	---	$G(\text{FSAR}) = F(\text{ACI-349})$. $F_{eqo}(\text{FSAR}) = E_o(\text{ACI-349})$. Load case enveloped by 4.
6	$U = 1.0D + 1.0L + 1.0G + 1.0W_t + 1.0Z + 1.0T_o + 1.0R_o$	$U = D + F + L + T_o + R_o + W_t$	$G(\text{FSAR}) = F(\text{ACI-349})$. FSAR conservative in comparison to ACI-349.
7	$U = 1.0D + 1.0L + 1.0G + 1.0H + F_{eqs} + 1.0T_o + 1.0R_o$	$U = D + F + L + T_o + R_o + E_{ss}$	$G(\text{FSAR}) = F(\text{ACI-349})$. $F_{eqs}(\text{FSAR}) = E_{ss}(\text{ACI-349})$. FSAR equivalent to ACI-349.
8	$U = 1.0D + 1.0L + 1.0G + 1.0H + 1.5P_a + 1.0T_a + 1.0R_a$	$U = D + F + L + H + T_a + R_a + 1.25P_a$	$G(\text{FSAR}) = F(\text{ACI-349})$. Load factor for P_a conservative in the FSAR. FSAR conservative in comparison to ACI-349.
9	$U = 1.0D + 1.0L + 1.0P_a + 1.0M + 1.0R_a$	---	FSAR load case includes internal missiles.
10	$U = 1.0D + 1.0L + 1.0G + 1.0H + 1.0T_o + 1.0R_o + 1.0P_r$	----	$G(\text{FSAR}) = F(\text{ACI-349})$. FSAR load case includes internal missiles.
11	$U = 1.0D + 1.0L + 1.0G + 1.0H + 1.25F_{eqo} + 1.25P_a + 1.0T_a + 1.0R_a + 1.0(Y_r + Y_j + Y_m)$	$U = D + F + L + H + T_a + R_a + 1.15P_a + 1.0(Y_r + Y_j + Y_m) + 1.15E_o$	$G(\text{FSAR}) = F(\text{ACI-349})$. $F_{eqo}(\text{FSAR}) = E_o(\text{ACI-349})$. Load factors for P_a and E_o conservative. FSAR conservative in comparison to ACI-349.

Table 3.7-7 Perry Unit 1 (PNPP) (Continued)			
EQN. No.	FSAR Load Combination Perry Unit 1 (PNPP)	Corresponding ACI-349-1990 Load Combination	Discussion
12	$U = 1.0 D + 1.0 L + 1.0 G + 1.0 H + 1.0 F_{eqs} + 1.0 P_a + 1.0 T_a + 1.0 R_a + 1.0 (Y_r + Y_j + Y_m)$	$U = D + F + L + H + T_a + R_a + 1.0 P_a + 1.0 (Y_r + Y_j + Y_m) + E_{ss}$	$G (FSAR) = F (ACI-349).$ $F_{eqs} (FSAR) = E_{ss} (ACI-349).$ FSAR equivalent to ACI-349.
13	$U = 1.0 D + 1.0 L + 1.0 W_t + 1.0 M$	---	FSAR load case includes tornado, wind and tornado missiles.

Table 3.7-8 Davis-Besse			
EQN. No.	FSAR Load Combination Davis-Besse	Corresponding ACI-349-1990 Load Combination	Discussion
1	$U = 1.5D + 1.8 L$	$U = 1.4D + 1.4F + 1.7L + 1.7H + 1.7R_o$	(FSAR) D&L factor conservative. However, the FSAR load case does not include several loadings included in ACI-349. In general, FSAR not conservative.
2	$U = 1.25(D + L + H_o E) + 1.0 T_o$	$U = 1.4D + 1.4F + 1.7L + 1.7H + 1.7 E_o + 1.7 R_o$ or $U = 1.05D + 1.05F + 1.3L + 1.3H + 1.3E_o + 1.05 T_o + 1.3 R_o$	$H_o(\text{FSAR}) = R_o$ (ACI-349). FSAR load case not conservative in comparison to ACI-349. All load factors are less than ACI-349. In general, FSAR not conservative.
3	$U = 1.25(D + L + H_o + W) + 1.0 T_o$	$U = 1.4D + 1.4F + 1.7L + 1.7H + 1.7W + 1.7R_o$ or $W = 1.05D + 1.05F + 1.3L + 1.3H + 1.3W + 1.05T_o + 1.3 R_o$	$H_o(\text{FSAR}) = R_o$ (ACI-349). FSAR load case not conservative in comparison to ACI-349. All load factors are less than ACI-349. In general, FSAR not conservative.
4	$U = .9D + 1.25(H_o + E) + 1.0 T_o$	—	$H_o(\text{FSAR}) = R_o$ (ACI-349). This load case includes a reduction in gravity load for members when gravity loads reduce the overall stress for earthquake.
5	$U = .9D + 1.25(H_o + W) + 1.0 T_o$	—	$H_o(\text{FSAR}) = R_o$ (ACI-349). This load case includes a reduction in gravity load for members when gravity loads reduce the overall stress for wind.
6*	$U = 1.4(D + L + E) + 1.0 T_o + 1.25 H_o$	$U = 1.05D + 1.05F + 1.3L + 1.3H + 1.3E_o + 1.05 T_o + 1.3R_o$	$H_o(\text{FSAR}) = R_o$ (ACI-349). FSAR load case conservative provided ACI loads F and H are included with D and L.
7*	$U = 0.9D + 1.25E + 1.0 T_o + 1.25 H_o$	—	$H_o(\text{FSAR}) = R_o$ (ACI-349). This load case includes a reduction in gravity load for members when gravity loads reduce the overall stress for earthquake.

Table 3.7-8 Davis-Besse (Continued)			
EQN. No.	FSAR Load Combination Davis-Besse	Corresponding ACI-349-1990 Load Combination	Discussion
8**	$U = 1.0 D + 1.0 L + 1.8 E + 1.0 T_o + 1.25 H_o$	$U = 1.4D + 1.4F + 1.7L + 1.7H + 1.7E_o + 1.7R_o$	$H_{oa} (FSAR) = R_o$ (ACI-349). Most load factors less in FSAR compared to ACI-349. In general, FSAR not conservative.
9	$U = 1.0 D + 1.0 L + 1.25E + 1.0 T_A + 1.0 H_A + 1.0R$	$U = D + F + L + H + T_a + R_a + 1.15 P_a + 1.0 (Y_r + Y_j + Y_m) + 1.15 E_o$	$R (FSAR) = Y_r + Y_j + Y_m$ (ACI-349). Load factor in FSAR conservative for E. However, the FSAR load case does not include several loadings included in ACI-349). In general, FSAR not conservative.
10	$U = 1.0D + 1.25E + 1.0 T_A + 1.0H_A + 1.0R$	----	$R (FSAR) = Y_r + Y_j + Y_m$ (ACI-349) This load case includes a reduction in gravity load for members when gravity loads reduce the overall stress for earthquake.
11	$U = 1.0D + 1.0L + 1.0E' + 1.0 T_o + 1.25 H_o + 1.0R$	$U = D + F + L + H + T_a + R_a + 1.0 P_a + 1.0 (Y_r + Y_j + Y_m) + 1.0 E_{ss}$	$R (FSAR) = Y_r + Y_j + Y_m$ (ACI-349). FSAR load case does not include several loadings in ACI 349. In general, FSAR not conservative.
12	$U = 1.0D + 1.0 L + 1.0E' + 1.0T_A + 1.0 H_A + 1.0R$	$U = D + F + L + H + T_a + R_a + 1.0 P_a + 1.0 (Y_r + Y_j + Y_m) + 1.0 E_{ss}$	$R (FSAR) = Y_r + Y_j + Y_m$ (ACI-349). FSAR load case does not include several loadings in ACI 349. In general, FSAR not conservative.
13	$U = 1.0 D + 1.0 L + 1.0 W' + 1.0 T_o + 1.25 H_o$	$U = D + F + L + H + T_o + R_o + W_t$	$H_o (FSAR) = R_o$ (ACI 349). FSAR equivalent provided ACI loads F and H are included with D and L.

* For Ductile moment concrete space frames, shear walls, and braced frames.

** For structural elements carrying mainly earthquake forces, such as equipment supports.

3.8 Special Issues and Concerns

3.8.1 Revised OBE Criteria

As overviewed in Section 2.2, recent changes in 10CFR50 Appendix S have resulted in a change in the licensing of nuclear power facilities for Seismic Events. The major change which impacts Regulatory Guide 1.142 is that the final regulation allows the value of the OBE to be set at (i) one-third or less of the SSE, where OBE requirements are satisfied without an explicit analysis being performed, or (ii) a value greater than one-third of the SSE, where design for the OBE is required. The applicant may voluntarily select an OBE value at a fraction of the SSE higher than one-third to avoid plant shutdowns in the event of a site earthquake. However, if an applicant selects an OBE value at a fraction of the SSE higher than one-third, a design analysis is required to demonstrate that the requirements associated with the OBE are satisfied.

The above criteria are only applicable to site applications received on or after the effective date of the final regulations which is December 11, 1996. Therefore, it is not applicable to existing nuclear power plant facilities. NUREG/CR-6358 also discusses this item and indicates changes should be made to address this regulatory change. However, at the time of issuance of NUREG/CR-6358, the final format of the regulation was unclear. Further, the changes suggested in NUREG/CR-6358 have yet to be implemented in ACI-349.

The most significant effect of this regulation is that Regulatory Guide 1.142 (ACI-349) must address at least three unique design criteria for the seismic loads:

1. Design for five seismic events at OBE level of 1/2 SSE and Design for single SSE seismic event (Operating Plants).
2. Design for a seismic event having an OBE level greater than 1/3 SSE up to a level equal to the SSE and design for a single SSE event (New Applicants).
3. Set the OBE level at 1/3 SSE and design only for an SSE level. (New Applicants).

Therefore, Regulatory Guide 1.142 must provide adequate discussion and modifications to the ACI-349 design equations and load combinations to address each of these three possible design issues.

3.8.2 BWR Hydrodynamic Loads

An important consideration for the design of BWR concrete structures (inside containment) is the effect of hydrodynamic loadings associated with the Design Basis Accidents (LOCA) and other plant transients such as safety relief valve discharge. The nature of the applied loads is highly dependent on the containment design but they can be grouped into two broad categories:

- (a) Direct Impact Loads
- (b) Building Filtered Vibratory Loads

Currently Regulatory Guide 1.142 is silent on these loads saying only that they will be considered on a case by case basis.

Appendix C of ACI-349 provides ductility factors for impulsive and impactive effect loads. These factors appear consistent with factors used in the design of ABWR SRV discharge and/or LOCA loadings.

Several issues make it difficult to define the applied LOCA and SRV loadings on a generic basis:

- (a) The loads are highly dependent on the location of the concrete structures within the concrete containments.
- (b) The load occurrence and time sequence is dependent on the SRV event or LOCA under consideration.
- (c) The type and nature of the loadings is dependent on the actual containment design and location of the suppression pool.
- (d) A significant amount of the data necessary to determine the load definition, load sequence, etc. is proprietary to the Nuclear Steam Supply System (NSSS) Vendors.

Considering these factors it is very difficult to provide a clear definition of the appropriate load combinations, load and ductility factors. This diversity is further witnessed in the review of the BWR plants summarized in Section 3.7.

The need to provide such guidance for the Advanced Reactor Designs was identified in Section 3.5 and

NUREG/CR-6358. However given the factors associated with load definition, guidance would be difficult to provide in a Regulatory Guide. Therefore it is suggested the Regulatory Guide require that the load combinations and acceptance criteria be developed consistent with Appendix C of ACI-349 and actual implementations will be reviewed on a case by case basis. Further, it is suggested that the ACI-349 committee continue to enhance and modify Appendix C of ACI-349 to more concisely address hydrodynamic loadings associated with pressure suppression containments.

3.8.3 Malevolent Vehicle

The USNRC in a regulation issued 10CFR Part 73 provided requirements for protection against malevolent use of vehicles at Nuclear Power Plants. In addition, the USNRC has issued Regulatory Guide 5.68 and NUREG/CR 6190 to provide guidance on barrier design, stand-off distance, and subsequent vehicle detonation blast effects on vital structures.

These blast detonation effects would only be applicable to outer most concrete structures and would not necessarily need to be considered concurrently with other extreme loads. Regulatory Guide 1.142 should be modified to (a) Identify when or if these loads must be considered and (b) provide acceptable load combination and load factors for the necessary evaluation. Load factors, combinations, and allowable stress criteria should be consistent with that provided in Appendix C of ACI-349. Blast loadings from malevolent details should be considered as W_1 in Load Combination 5 of ACI -349-97, Section 9.2.1.

3.8.4 Small Aircraft Crash

Based on the review by Adams and Stevenson in NUREG/CR-6358, the USNRC intends to accept a reduced tornado design criteria for the Advanced Light Water Reactor and Advanced Reactor Design namely:

134m/s (300 mph) versus 161m/s (360 mph) Wind Speed east of 105° longitude and 89m/s (200 mph) versus 107m/s (240 mph) or 134m/s (300 mph) west of 105° longitude.

Using DOE Standard 3014 for small airplane crash, the crash would typically govern for wind speeds under 111m/s (250 mph).

In current domestic United States operating nuclear power plants, the tornado design basis requirements have been used to establish structural requirements (such as minimum

concrete wall thicknesses) to protect nuclear plant safety-related SSC against effects not explicitly addressed in regulatory guidance (such as RG or the SRP). Specifically, the staff has routinely reviewed and evaluated aviation crashes (involving general aviation light aircraft), nearby explosions, and explosion debris or missiles, taking into account the tornado protection requirements. Depending on how the design basis is established for these man-made hazard phenomena, the staff's acceptance of these reduced tornado criteria may now require explicit consideration of some external impact hazards such as small airplane crash.

Therefore, suggested modification to the ACI-349 load combinations, load factors and capacity criteria should be made in Regulatory Guide 1.142 to adequately address design for small aircraft crash. These modification should consider criteria and guidance found in Appendix C of ACI-349. Impact loads from small aircraft crash should be considered as W_1 in Load Combination 5 of ACI 349-97, Section 9.2.1.

3.8.5 Accidental Explosion

Standard Review Plan SRP Section 2.2.1-2.2.2, "Identification of Potential Hazards in Site Vicinity" and SRP Section 2.2.3 "Evaluation of Potential Accidents (external to the plant)" identify the need to evaluate the nuclear power plant site for such phenomena. The evaluations include potential mal operation or accidental failure of high energy (i.e. rapidly rotating equipment, high pressure and explosion) Structures, Systems, or Components (SSC) in the vicinity of the nuclear power plant site.

Regulatory Guide 1.91, Rev. 1, provides specific guidance as to external accidental explosion loads and effects to be considered. In general, this R.G. establishes that nuclear power plant SSC designed to resist tornado wind loads of 107m/s (240 mph) or higher are capable of resisting an equivalent static blast load of 6.9kPa (1.0 psi). The stand off distance for various quantities of explosives (TNT equivalent) are defined in R.G. 1.91 as a function of the one psi static equivalent over pressure.

It is recommended that waste structures, systems and components which are currently designed to resist equal to or more than 0.5 SSE should be designed for accidental explosions as well.

3.8.6 Data from Experience Reviews

NUREG/CR-6358 identified that recent earthquake experience supports the need to consider relatively longer displacement and story drift limits as well as stress limits in structural members responding to earthquake ground motions. This is particularly true for facilities near (within 10km) the epicenter or fault rupture lines of thrust type faults from a damaging earthquake. Changes to ACI-349 were suggested to address this item but they have yet to be incorporated into the Code. Therefore guidance should be provided in Regulatory Guide 1.142 to address this item.

3.8.7 Monolithic Structures

Monolithic Structures refer to layouts of nuclear plants in which the concrete containment or shield building is hard connected to the surrounding concrete safety related structures. This raises two new issues.

First, for the case of a concrete containment connected to surrounding concrete structures, the clear distinction between ACI-349 and ASME III, Div. 2 and ACI-359 is confused. It is recommend that surrounding structures which structurally participate with the containment in resisting accident and seismic loads be required to meet the controlling provisions of both codes.

Second, for the case of either containment or shield structure, the structural response of shear walls, slabs, and foundations are dramatically altered. This is primarily due to the fact that seismic loads and possibly accident loads are transferred from the shield structure or containment to the concrete structures which are subjected to much greater loads and are required to respond differently than layouts which separate these types of structures.

This item was also discussed in the review of the Advanced Reactor Designs. As was suggested in that section, jurisdictional boundary guidance should be provided in Regulatory Guide 1.142.

4.0 Regulatory Guide 1.143 Investigation

4.1 General

4.1.1 Objective

The objective of this Section is to evaluate the structural-mechanical design guidance contained in Regulatory Guide 1.143 Revision 1 (RG 1.143) for radioactive waste management of selected Structures, Systems and Components, SSC, as applicable to commercial Nuclear Power Plants. In addition, suggested changes to RG 1.143 to update it to current practice and criteria will be developed.

4.1.2 Scope

This Section will include the review of the following USNRC principal guidance and industry consensus standards.

- (a) Regulatory Guide 1.143, Revision 1, "Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light Water-Cooled Nuclear Power Plants". (7.2-[5])
- (b) USNRC Standard Review Plan, Chapter 3.2
- (c) USNRC Standard Review Plan, Chapter 11, "Radioactive Waste Management". (7.2-[3])
- (d) USNRC Standard Review Plan, Chapter 15.7, "Radioactive Release (Accident) from a Subsystem or Component". (7.2-[3])
- (e) American Nuclear Society, "Solid Radioactive Waste Processing System for Light Water Cooled Reactor Plants", ANSI/ANS-55.1-1992. (7.5-[5])
- (f) American Nuclear Society, "Gaseous Radioactive Waste Processing Systems for Light Water Reactor Plants", ANSI/ANS-55.4-1993. (7.5-[6])
- (g) American Nuclear Society, "Liquid Radioactive Waste Processing System for Light Water Reactor Plants", ANSI/ANS-55.6-1993. (7.5-[7])

The boundaries and scope of Radioactive Waste Systems is currently defined in R.G. 1.143 Rev. 1 as "beginning at the

interface valve(s) 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 prior to shipment on or off-site to a licensed burial ground".

The steam generator blow-down system begins at, but does not include, the outermost containment isolation valve on the blow-down line. It terminates at the point of controlled discharge to the environment, at the point of interface with other liquid systems, or at the point of recycle back to the secondary systems. 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, RG 1.143, Rev. 1 does not apply to the reactor water cleanup system, the condensate cleanup system, the chemical and volume control system, the reactor coolant system and the boron recovery system. These systems are part of the reactor and power generation systems. Other systems to which RG 1.143 does not apply are auxiliary building equipment drain tanks, the sumps and floor drains provided for collecting liquid wastes, equipment used to prepare solid waste solidification agents, the building ventilation systems (heating, ventilating, and air conditioning), instrumentation and sampling systems beyond the first root valve, or a chemical fume hood exhaust system. However, these systems which are not designated by the guide as radwaste systems are often included in the radwaste management systems for individual plants. In addition, the RG 1.143 does not apply to the main condenser circulating or component cooling water systems, the spent fuel handling and storage systems, or the spent fuel pool water cleanup system.

4.2 Literature and Technology Study

In addition to the seven documents listed in Section 4.1.2, there are a number of documents which have been prepared after the publication of RG 1.143 Rev. 1 in 1979 which were considered in developing the suggested revision of RG 1.143. These documents include the following:

- (a) Letter from L.C. Dale, Duke Power Corp. to T.H. Novak, NRC Dated 29 March 1982.

- (b) NUREG/CR - 0619 - "Properties of Radioactive Wastes and Waste Containers".
- (c) NUREG/CR - 1005 Vol. 1 & 2 A: "Radioactive Waste Disposal Classification System".
- (d) NUREG/CR - 1758. "Safeguards for High-Level Waste Depositories".
- (e) NUREG/CR - 2333 Vol. 1-5; "Nuclear Waste Management Technical Support in Development of Nuclear Waste Form Criteria for the NRC".
- (f) NUREG/CR - 3168; "Technical Consideration for High Integrity Container for the Disposal of Radioactive Ion - Exchange Resin Waste".
- (g) NUREG/CR - 3444 Vol. 2 through 5.
- (h) NUREG/CR 4062; "Extended Storage of Low-Level Radioactive Wastes".
- (i) NUREG/CR 4509; "Waste Package Reliability".
- (j) NUREG 1418; "Characteristics of Level Radioactive Waste Disposed During 1987 through 1989"
- (k) NUREG/CP-0028, Vol. 3; "Proceedings on the Symposium on Low-Level Waste Disposal.
- (l) DOE Std. 1020-94, Ch. Notice 1-1996, Natural Phenomena Hazard Design and Evaluation Criteria for Department of Energy Facilities", U.S. Department of Energy, 1996. (7.4-[1])
- (m) NUREG/CR-6407; "Classification of Transportation and Packaging and Dry Spent Fuel Storage System Components According to Important to Safety, Idaho National Laboratory", February 1996.
- (n) Title 10 Code of Federal Regulation Part 71, "Packaging and Transportation of Radioactive Material", Federal Register, 27 May 1988. (7.4-[2])
- (o) Regulatory Guide 7.10, "Establishing Quality Assurance Programs for Packaging Used in Transport of Radioactive Material", Rev. 1 Nuclear Regulatory Commission, June 1986.
- (p) Regulatory Guide 7.11, "Fracture Toughness Criteria of Base Material for Ferritic Steel Shipping Cask Containment Vessels with a Maximum Wall Thickness of 4 Inches (0.1m)", Nuclear Regulatory

Commission, June 1991.

- (q) DOE Std. 3009-94, "Preparation Guide for U.S. Nuclear Department of Energy Non-Reactor Nuclear Facilities Safety Analysis Reports". (7.4-[3])
- (r) DOE Std. 1027-92, "Hazard Categorization and Accident Analysis Techniques for Compliance with DOE Order 5480.23", Nuclear Safety Analysis Reports. (7.4-[4])
- (s) Branch Technical Position, ETSB 11-5 "Postulated Radioactive Releases Due to a Waste Gas Systems Leak or Failure". (7.2-[6])
- (t) Branch Technical Position, "ETSB 11-3, "Design Guidance for Solid Radioactive Waste Management Systems Installed in Light Water-Cooled Nuclear Power Reactor Plants". (7.2-[7])

Currently, in RG 1.143 and/or in SRP Section 11.0, there is no numerical guidance provided which would permit users of RG 1.143 to decide what systems and components of waste processing and storage systems should be placed in seismically designed buildings (e.g. auxiliary building) and in a non-nuclear safety building (e.g. the radwaste storage building). In addition, there is no guidance currently given in RG 1.143 as to how natural phenomena hazards other than earthquake such as wind, tornado, flood etc. and man-induced hazards such as accidental or malevolent explosion, vehicle intrusion etc. are to be considered in design. However, potential for the development of an accidental buildup of a explosive mixture of hydrogen in the radwaste system is addressed in RG 1.143.

SRP Section 11.0 for liquid and gaseous waste gives requirements and references 10 CFR 20, 10 CFR 50 Appendix I which limit the amount of radioactive material release due to normal and anticipated transient operations and the resulting quantities and potential doses to the public associated with that release. However, there is no similar guidance given except for Branch Position Paper ETSB 11-5 for quantities and doses associated with design basis natural and man-induced hazard events and accidents or failure. ETSB 11-5 discusses radiation dose limits to the public for design basis waste gas system leak, failure or operator error. However, this dose has been steadily reduced from the guidance given in 10 CFR 100 of 25 REM to the public to the current 0.5 REM limit based on the realization that the potential for an accidental release of radioactive gas is not necessarily a very low probability event as classified in ETSB 11-5.

In compliance with SRP Section 11.0, the licensee reports

the isotope and waste quantities of radioactive materials and the effective release fractions which effectively define doses for normal operation to various areas of the plant and its boundaries and limit the isotope release to quantities given in 10CFR50 para. 20.1001-20.2402 Appendix B (7.2-[15]).

As discussed in some detail in Sections 4.3 for Advanced Reactor designs and in Section 4.4 for Operating Plant Waste Management Facilities, associated SSC are divided into two categories, Seismic (Safety) and Non-Seismic (Non-Nuclear Safety). Seismic class being applied to gaseous and liquid (up to spill height) containing structures and non-nuclear safety for all other radwaste structures, systems and components. There is no clear guidance provided in the current edition of RG 1.143 into which category waste SSC fall as a function of stored or processed isotope quantities and unmitigated dose to the public (or to collocated workers) under Normal and Design Basis and Accident conditions.

SRP Section 15.7 addresses Design Basis Accidents which may impact structural-mechanical design of SSC at LWR nuclear power plants, hence, could be included in a revised RG 1.143 to address these issues for completeness.

4.3 ARD Waste Management Facility Commitments

As discussed in Section 2.2, there are currently only 3 Advanced Reactor designs either licensed or actively pursuing a licensing basis. They are the General Electric (GE) Advanced Boiling Water Reactor (ABWR), ASEA Brown Boveri Combustion Engineering (ABB/CE) System 80+ and the Westinghouse (W) AP600. Therefore, only these three advanced Reactors are considered in this review.

Advanced Reactor Designs for General Electric, Combustion Engineering and Westinghouse are in general compliance with current editions of RG 1.143, SRP-11.0 and the American Nuclear Society Waste Processing Standards for solid, gaseous and liquid waste, ANS 55.1-1992, ANS 55.4-1993 and ANS 55.6-1993. Furthermore, they have provided the safety and design classification of their waste SSC as required in Section 3.2 and their accident analyses in Section 15.7.3 of their SSAR's. A summary of their radwaste commitments for the ARD's is given in Tables 4.3-1 to 4.3-6.

4.3.1 Advanced Building Water Reactor - General Electric Radwaste Systems

A detailed description of the ABWR Radwaste System can be found in Chapter 11 of the ABWR SSAR which discusses: 1) a general description, 2) the design basis, 3) system description and 4) releases for each type of radwaste system, liquid, gaseous and solid. In addition to meeting the requirements of Standard Review Plan for the SSAR and associated Regulatory Guides, the ABWR Radwaste System meets the criteria contained in the applicable ANS-55 series of standards.

4.3.1.1 Liquid Waste Management System

The ABWR Liquid Waste Management System collects, monitors, and treats liquid radioactive wastes for return to the primary system whenever practicable. The radwaste processing equipment is located in the non-nuclear safety Radwaste Building. Processed waste volumes discharged to the environs are expected to be small. Limits of any discharge is such that concentrations and quantities of radioactive material and other contaminants are in accordance with applicable local, state, and federal regulations.

All potentially radioactive liquid wastes are collected in sumps or drain tanks at various locations in the plant. These wastes are transferred to collection tanks in the radwaste facility.

Waste processing is done on a batch basis. Each batch is sampled as necessary in the collection tanks to determine concentrations of radioactivity and other contamination. Equipment drains and other low-contamination liquid wastes are treated by filtration and demineralization and are transferred to the condensate storage tank for reuse. Laundry drain wastes and other detergent wastes of low activity are treated by filtration, sampled and released via the liquid discharge pathway and demineralization and may be released from the plant on a batch basis. Protection against inadvertent release of liquid radioactive waste is provided by design redundancy, instrumentation for the detection and alarm of abnormal conditions, automatic isolation, and administrative controls.

Equipment is selected, arranged, and shielded to permit operation, inspection, and maintenance with minimum radiation exposure to personnel.

4.3.1.2 Gaseous Waste Management System

The objective of the ABWR Gaseous Waste Management

System is to process and control the release of gaseous radioactive effluents to the site environs so as to maintain the exposure of persons in unrestricted areas to radioactive gaseous effluents as low as reasonable achievable (10CFR50, Appendix I). This is accomplished while maintaining occupational exposure as low as reasonably achievable and without limiting plant operation or availability.

The off gas system provides for holdup and decay of radioactive gases in the off gas from the nuclear steam and reactor air ejector system and consists of process equipment along with monitoring instrumentation and control equipment.

4.3.1.3 Solid Waste Management System

The ABWR Solid Waste Management System provides for the safe handling, packaging, and short-term storage of radioactive solid and concentrated liquid wastes that are produced. Wet waste processed by this system is transferred to the solidification system, where it is solidified in containers. Dry active waste is surveyed and disposed of whenever possible via the provisions of 10CFR20.302 (a) (7.2-[12]). The remaining combustible waste is compacted. Incinerator ash is compacted solid waste and shipped in containers for off-site disposal.

4.3.1.4 Radwaste Building

The Radwaste Building (RWB) houses the Liquid and Solid Waste Management Systems. The Gaseous (offgas) Waste Management System is housed in the Turbine Building. Below grade the Radwaste Building is designed as Seismic Category I. Above grade it is defined as Non-Nuclear Safety.

The RWB is a reinforced concrete structure 60.4m by 41.2m and a height of 29.5m from the top of the basemat. The building consists of a below grade substructure consisting of walls (1.2m thick) and slabs of reinforced concrete forming a rigid box structure which serves as a container to hold radioactive waste in case of an accident. This substructure is located below grade to increase shielding capability and to maximize safety. It is supported on a separate foundation mat whose top is 13.5m below grade. In addition, a reinforced concrete superstructure 15.7m high extends above grade floor level and houses the balance of the radwaste equipment.

The RWB substructure houses the high and low conductivity tanks, clean-up phase separators, spent resin storage tanks, a concentrated waste storage tank, distillate tank and associated filters, and pumps for the radioactive

liquid and solid waste treatment systems.

Although the radwaste superstructure is not a Seismic Category I structure, its major structural concrete walls, slabs, columns and roof are designed to resist Seismic Category I loads.

A summary report for the radwaste building is in SSAR Section 3H.3. This report contains a description of the radwaste building, the loads, load combinations, reinforcement stresses, and concrete stresses at locations of interest. In addition, the report contains reinforcement details for the basemat, seismic walls, and floors.

4.3.2 Advanced Pressurized Water Reactor System 80+ - Combustion Engineering Radwaste Systems

A scope description of the System 80+ Radwaste Systems is found in Section 11 of the System 80+ SSAR.

4.3.2.1 Liquid Waste Management Systems

The design objectives of the Liquid Waste Management System (LWMS) is to protect the plant personnel, the general public, and the environment by providing a means to collect, segregate, store, process, sample, and monitor radioactive liquid waste. Each type of liquid waste is segregated to minimize the potential for mixing and contamination of non-radioactive flow streams. The processed liquid radioactive waste is sampled prior to release from Waste Monitor Tanks and radiation monitors are provided in the discharge line to provide for a controlled, monitored release. The concentration of the liquid effluent at the site discharge point released during normal operation, including anticipated operational occurrences, is below concentrations specified in 10 CFR 20, Appendix B of Sections 20.1001-20.2402, Table 2, Column 2 (7.2-[12]), and meet the As Low As Reasonably Achievable (ALARA) criteria of 10 CFR 50, Appendix I (7.2-[13]).

4.3.2.2 Gaseous Waste Management Systems

Effluents normally released to unrestricted areas must meet the limiting requirements of 10 CFR 20 (7.2-[12]) and meet the ALARA objectives of 10 CFR 50, Appendix I (7.2-[13]). The Gaseous Waste Management System (GWMS) continuously discharges effluent.

The GWMS is a non-nuclear safety related system. It has no reactor accident mitigation functions. The GWMS is designed in accordance with the guidance in ANSI/ANS

55.4, Regulatory Guide 1.143 and 1.140. This includes the following features:

1. The GWMS is designed to preclude a buildup of an explosive mixture of hydrogen and oxygen which could impact the operation of the plant.
2. The GWMS is designed with sufficient capacity and redundancy to accommodate an increase in demand during normal operation of the plant.

Accidental releases of radioactive materials from a single component of the GWMS must not result in off-site doses which exceed the guides of Branch Technical Position ESTB-11-5.

The GWMS is designed to preclude the buildup of an explosive mixture of hydrogen and oxygen in the system. Nevertheless, the charcoal vessels condenser cooler, piping, analyzer pressure boundary and valves within the GWMS are designed to withstand a hydrogen explosion (i.e., twenty times normal operating pressure) in accordance with ANSI Standard 55.4. One hydrogen and one oxygen gas analyzer is utilized to monitor H₂ and O₂ gas concentrations in the GWMS. Alarms are provided locally in the Nuclear Annex and in the Main Control Room to alarm on high oxygen concentration.

4.3.2.3 Solid Waste Management System

The Solid Waste Management System (SWMS) is designed to protect the plant personnel, the general public, and the environment by providing a means to collect, segregate, store, process, sample, and monitor solid waste. The SWMS processes both wet solid active waste and dry active waste for shipment to a licensed burial site.

Releases of radioactive materials to the environment are controlled and monitored in accordance with 10 CFR 50, Appendix A (General Design Criteria 60 and 64) (7.2-[14]).

The SWMS is designed so that liquids removed during the dewatering process of wet solid waste are routed back to the Liquid Waste Management System (LWMS) to be processed prior to release to the environment.

4.3.2.4 Radwaste Building

The Radwaste Building is intended to be located adjacent to the Nuclear Annex. Provisions are included to preclude an uncontrolled release of effluents to the environment due to a LWMS or SWMS failure. If site specific requirements preclude locating the Radwaste Building adjacent to the

Nuclear Annex, the consequences of a LWMS or SWMS failure are evaluated to demonstrate compliance with 10 CFR 20, Appendix B limits. This analysis demonstrates that the concentration of the liquid effluent at the potable water source, released from the Radwaste Building due to a LWMS or SWMS failure, is within 10 CFR 20, Appendix B limits.¹

Ventilation ensures controlled and monitored release of gaseous effluent from the Radwaste Building. The Radwaste Building is equipped with area and airborne radiation monitors to provide indication of a spill and to ensure that personnel exposures are maintained ALARA.

4.3.2.5 On-site Storage Facility

The Radwaste Building is located in close proximity to an Interim On-site Storage Facility to facilitate transport of packaged waste for interim storage prior to shipment to a licensed burial facility. Adequate space is provided for storage and processing of radwaste. This On-site Storage Facility is not further discussed in the SSAR and it can be assumed to be classified as non-nuclear safety.

4.3.3 Advanced Pressurized Water Reactor System AP-600 Westinghouse Radwaste Waste Systems

This section contains perhaps the most information concerning the radwaste management facilities including the structural-mechanical design of its various SSC than any of the other ARD's.

4.3.3.1 Liquid Waste Management Systems

The Westinghouse AP-600 Liquid Waste Management Systems (LWMS) include the systems that may be used to process disposal liquids containing radioactive material. These include the following:

- Steam generator blowdown processing system
- Radioactive waste drain system
- Liquid radwaste system

¹ It should be noted that a commitment has been made to keep released radioactive material quantities within 10CFR50 Paragraph 20.1001-20.2401 Appendix B limits resulting from a liquid or solid waste management failure. In making such an evaluation it is assumed that only Design Basis failures have occurred and all mitigation SSC and control procedures are available.

This discussion primarily addresses the liquid radwaste system. The liquid radwaste system is designed to control, collect, process, handle, store, and dispose of liquid radioactive waste generated as the result of normal operation, including anticipated operational occurrences.

The AP-600 liquid radwaste system serves no safety-related functions except for:

- Containment isolation.
- Draining the passive core cooling system compartments to the containment sump to prevent flooding of these compartments and possible immersion of safety-related components.
- Back flow prevention check valves in the drain lines from the chemical and volume control system compartment and the passive core cooling system compartments to the containment sump, which prevent cross flooding of these compartments. Each drain line has two check valves in series so that a single failure does not compromise the back flow prevention safety function.

The liquid radwaste system provides adequate capacity to meet the anticipated processing requirements of the plant. The projected flows of various liquid waste streams to the liquid radwaste system under normal conditions are identified in SSAR Table 11.2-1.

The liquid radwaste system design can accept equipment malfunctions without affecting the capability of the system to handle both anticipated liquid waste flows and possible surge load due to excessive leakage. SSAR Table 11.2-4 contains information on the surge capacity of individual tanks.

Portions of the liquid radwaste system may become unavailable as a result of the malfunctions listed in SSAR subsection 11.2.1.2.2.

Ample surge capacity of the system, provisions for using mobile processing equipment and the low load factor of the processing equipment permits the system to accommodate waste until failures can be repaired and normal plant operation resumed. In addition, the liquid radwaste system is designed to accommodate the anticipated operational occurrences described in subsection 11.2.1.2.3 of the SSAR.

(a) Controlled Release of Radioactivity

The liquid radwaste system provides the capability to reduce the amounts of radioactive nuclides released in the liquid wastes through the use of demineralization and time

delay for decay of short-lived nuclides.

The assumed equipment decontamination factors appear in SSAR Table 11.2-5. Estimates of the radioactive source terms and annual average flow rate that will be processed in the liquid radwaste system or discharged to the environment during normal operation appear in SSAR Table 11.2-1.

Before radioactive liquid waste is discharged, it is pumped to a monitor tank. A sample of the monitor tank contents is analyzed, and the results are recorded. In this way, a record is kept of planned releases of radioactive liquid waste.

The liquid waste is discharged from the monitor tank in a batch operation, and the discharge flow rate is restricted as necessary to maintain an acceptable concentration when diluted by the circulating water discharge flow. These provisions preclude uncontrolled releases of radioactivity.

In addition, the discharge line contains a radiation monitor with diverse methods of stopping the discharge. The first method closes an isolation valve in the discharge line, which prevents any further discharge from the liquid radwaste system. The valve automatically closes and an alarm is actuated if the activity in the discharge stream reaches the monitor setpoint. The second method stops the monitor tank pumps.

To minimize the leakage from the liquid radwaste system, the system is of welded construction except where flanged connections are required to facilitate component maintenance or to allow connection of temporary or mobile equipment. Air-operated diaphragm pumps or pumps having mechanical seals are used. These pumps minimize system leakage thereby minimizing the release of radioactive gas that might be entrained in the leaking fluid to the building atmosphere.

Provisions are made to control spills of radioactive liquids due to tank overflows. SSAR Table 11.2-3 lists the provisions for tank level indication, alarms, and overflow disposition for liquid radwaste system tanks outside containment.

The liquid radwaste system is designed so that the annual average concentration limits established by 10 CFR 20 (Appendix B, table 2, column 2) for liquid releases are not exceeded during plant operation. SSAR subsection 11.2.3 describes the calculated releases of radioactive materials from the liquid radwaste system and other portions of the liquid waste management systems resulting from normal operation.

SSAR subsections 11.2.1.2.2 and 11.2.1.2.3 describe the capability of the liquid radwaste system to accommodate abnormal conditions for various equipment and other anticipated operation occurrences. During these anticipated occurrences, the effectiveness of the liquid radwaste system in controlling releases of radioactivity remains unaffected, so releases are limited as during normal operation. SSAR subsection 11.2.3 discusses the calculated releases of radioactive materials from the liquid radwaste system for abnormal situations.

The liquid radwaste system equipment design parameters are provided in SSAR Table 11.2-2.

The seismic design classification and safety classification for the liquid radwaste system components and structures are listed in Table 4.3-5. The components listed are located in the Seismic Category I Nuclear Island (Auxiliary Building).

(b) Use of Mobile and Temporary Equipment

The liquid radwaste system is designed to handle most liquid effluents and other anticipated events using installed equipment. However, for events occurring at a very low frequency or producing effluents not compatible with the installed equipment, temporary equipment may be brought into the radwaste building mobile treatment facility truck bays.

Connections are provided to and from various locations in the liquid radwaste system to these mobile equipment connections. This allows the mobile equipment to be used in series with installed equipment, as an alternate to it with the treated liquids returned to the liquid radwaste system, or an ultimate disposal point for liquids that are to be removed from the plant site for disposal elsewhere.

The use of temporary equipment is common practice in operating plants. The radwaste building truck bays and laydown space for mobile equipment, and the flexibility of numerous piping connections to the liquid radwaste system, allow the plant operator to incorporate mobile equipment in an integrated fashion.

Temporary equipment is also used to clean up the condensate storage tank if it becomes contaminated following steam generator tube leakage. This use of temporary equipment is similar to that just described, except that the equipment is used in the yard rather than in the radwaste building truck bays.

4.3.3.2 Gaseous Waste Management System

The AP600 Gaseous Radwaste System (GRS) is designed to perform the following major functions:

- Collect gaseous wastes that are radioactive or hydrogen bearing
- Process and discharge the waste gas, keeping off-site releases of radioactivity within acceptable limits.

The gaseous radwaste system is designed to receive hydrogen bearing and radioactive gases generated during process operation. The radioactive gas flowing into the gaseous radwaste system enters as trace contamination in a stream of hydrogen and nitrogen.

Given the various inputs to the gaseous radwaste system, with licensing basis assumptions for analysis and with normally operating gaseous radwaste system equipment available, the combined plant releases must be within the limits outlined in 10 CFR 20 and 10 CFR 50 Appendix I.

The gaseous radwaste system operates at low pressures, slightly above atmospheric pressure, thus limiting the potential for leakage. Manual valves are the type which eliminate the potential for stem leakage. The system is of welded construction to further limit leakage.

A number of features prevent wetting the activated carbon delay beds. These features include controls and alarms in the liquid radwaste system to prevent high degasifier separator water level, the gas cooler/vapor condenser moisture separator and traps in the gaseous radwaste system, and moisture monitors in the gaseous radwaste system. Additional protection is provided by the activated carbon guard bed, which removes residual moisture as well as iodine from the gas stream.

The AP600 design prevents the annual average concentration limits established by 10 CFR 20 (Appendix B, table 2, column 1) for gaseous releases from being exceeded due to the releases resulting during plant operation. SSAR Subsection 11.3.3 describes the calculated releases of radioactive materials from the gaseous radwaste system and other pathways during normal operation.

SSAR Subsection 11.3.3 also contains an evaluation which demonstrates that the doses to individuals, at or beyond the site boundary, resulting from the expected releases from the gaseous waste management systems are within numerical design objectives of Appendix I of 10 CFR 50.

To prevent the release of radioactive gases resulting from equipment failure or operator error, a radiation monitor is

located in the discharge line. This instrument provides an alarm signal at a high level setpoint to alert operators of rising radiation levels. The monitor is also interlaced with an isolation valve in the discharge line; the valve closes at a higher level setpoint.

Few operator actions are required during gaseous radwaste system operation since, once aligned for operation, the system operates automatically in response to the control signals from the instrumentation.

4.3.3.3 Solid Waste Management

The AP600 Solid Waste Management System (WSS) is designed to collect and accumulate spent ion exchange resins and deep bed filtration media, spent filter cartridges, dry active wastes, and mixed wastes generated as a result of normal plant operation, including anticipated operational occurrences. The system is located in the auxiliary and radwaste buildings. Processing and packaging of wastes are by mobile systems in the auxiliary building rail car bay and in the mobile systems facility part of the radwaste building. The packaged waste is stored in the auxiliary and radwaste buildings until it is shipped offsite to a licensed disposal facility.

The use of mobile systems for the processing functions permits the use of the latest technology and avoids the equipment obsolescence problems experienced with installed radwaste processing equipment. The most appropriate and efficient systems may be used as they become available.

This system does not handle large, radioactive waste materials such as core components or radioactive process wastes from the plant's secondary cycle. However, the volumes and activities of the secondary cycle wastes are provided in this section.

The solid waste management system performs no function related to the safe shutdown of the plant. The system's failure does not adversely affect any safety-related system or component; therefore, the system has no nuclear safety design basis.

There are no safety related systems located near heavy lifts associated with the solid waste management system. Therefore, a heavy loads analysis is not required.

The solid waste management system is designed to meet the following objectives:

- Provide for transfer and retention of spent radioactive ion exchange resins and deep bed

filtration media from the various ion exchangers and filters in the liquid waste processing, chemical and volume control, and spent fuel cooling systems.

- Provide the means to mix, sample, and transfer spent resin and filtration media to high integrity containers or liners for dewatering or solidification as required.
- Provide the means to change out, transport, sample, and accumulate filter cartridges from liquid systems in a manner that minimizes radiation exposure of personnel and spread of contamination.
- Provide the means to accumulate spent filters from the plant heating, ventilation, and air conditioning systems.
- Provide the means to segregate solid wastes (trash) by radioactivity level and to temporarily store the wastes.
- Provide the means to accumulate radioactive hazardous (mixed) wastes.
- Provide the means to segregate clean wastes originating in the radiologically controlled area (RCA)
- Provide the means to store packaged wastes for at least 6 months in the event of delay or disruption of offsite shipping.
- Provide the space and support services required for mobile processing systems that will reduce the volume of and package radioactive solid wastes for offsite shipment and disposal according to applicable regulations, including Department of Transportation regulation 49 CFR 173 and USNRC regulation 10 CFR 71.
- Provide the means to return liquid radwaste to the liquid radwaste system (WLS) for subsequent processing and monitored discharge.

The solid waste management system is designed according to NRC Regulatory Guide 1.143 to meet the requirements of General Design Criterion (GDC) 60 as discussed in SSAR Sections 1.9 and 3.1. The seismic design classifications of the radwaste building and system components are provided in SSAR Section 3.2.

Provisions are made in the auxiliary and radwaste building to use mobile radwaste processing systems for processing and packaging each waste stream including concentration and solidification of chemical wastes from the liquid waste management system, spent resin dewatering, spent filter cartridge encapsulation and dry active waste sorting and compaction.

The radio activities of influents to the solid waste management system are based on estimated radionuclide concentrations and volumes. These estimates are based on operating plant experience, adjusted for the size and design

differences of AP600. The solid waste management system airborne process effluents are released through the monitored plant vent as described as part of the 10 CFR 50, Appendix I, analysis presented in SSAR subsection 11.3.3.

The solid waste management system collects and stores radioactive wastes within shielding to maintain radiation exposure to plant operation and maintenance personnel ALARA according to General Design Criteria 60 as discussed in Section 3.1 and Regulatory Guide 8.8. Personnel exposures will be maintained well below the limits of 10 CFR 20. Design features incorporated to maintain exposures ALARA include remote and semi-remote operations, automatic resin transport line flushing, and shielding of components, piping and containers holding radioactive materials. Access to the solid waste storage areas is controlled, to minimize inadvertent personnel exposure, by suitable barriers such as heavy storage cask covers and locked or key-card-operated doors or gates.

The solid waste management system conforms with the design criteria of NRC Branch Technical Position ETSB 11-3. Suitable fire protection systems are provided as described in SSAR subsection 9.5.1.

Waste disposal containers are to be selected from available designs that meet the requirements of the DOT and NRC. The solid waste management system does not require source-specific waste containers. Waste containers must meet the regulatory requirements for radioactive waste transportation in 49 CFR 173 and for radioactive waste disposal in 10 CFR 61 as well as specific disposal facility requirements.

4.3.3.4 Radwaste Building

The AP-600 radwaste building includes facilities for segregated storage of various categories of waste prior to processing, for processing by mobile systems, and for storing processed waste in shipping and disposal containers. No safety-related equipment is located in the radwaste building. Dedicated floor areas and trailer parking space for mobile processing systems is provided for the following:

- Contaminated laundry shipping for off-site processing
- Dry waste processing and packaging
- Hazardous/mixed waste shipping for off-site processing
- Chemical waste treatment
- Empty waste container receiving and storage
- Storage and loading packaged wastes for shipment

The radwaste building also provides for temporary storage of other categories of plant wastes. It is a non-seismic structure designed in accordance with the Uniform Building Code. The liquid radwaste processing areas are designed to contain any liquid spills. These provisions include a raised perimeter and floor drains that lead to the liquid radwaste system waste holdup tanks. The foundation for the entire building is a reinforced concrete mat on grade.

**Table 4.3-1(a) SSAR Nuclear Design Criteria
Codes and Standards for the Plant General Electric ABWR**

Item	Nuclear Design Criteria	Class			Construction Codes	Comment
		Quality ⁽¹⁾	Seismic ⁽²⁾	Tornado	Safety ⁽³⁾	
I. Building Structure A) Radwaste Building. a. Substructure b. Super Structure ⁽⁴⁾		B D	I NNS	I ⁽⁶⁾ NNS	2 NNS	ACL-349 & ASCE 4 ACL-318 & UBC
II. Systems A) Liquid	10CFR50 - Appendix I	NI	NI	NI	NI	NI
B) Gaseous	10CFR20.302 (a)	NI	NI	NI	NI	NI
C) Solid	NI	NI	NI	NI	NI	NI
III. Distribution Systems A) Piping 1. Process ⁽⁵⁾	NI	NNS	NNS	NA	NNS	See Tables 4.3-1(b), 4.3-1(c), 4.3-1(d)
2. Fire	NI	NNS	NNS	NA	NNS	NI
3. Drain	NI	D	NNS	NA	NNS	NI
B) Piping/Tubing (Inst.)	NI	NNS	NNS	NNS	NNS	See Tables 4.3-1(b), 4.3-1(c), 4.3-1(d)
C) Cable Trays	NI	NNS	NNS	NNS	NNS	See Tables 4.3-1(b), 4.3-1(c), 4.3-1(d)
D) Ventilation	NI	NNS	NNS	NNS	NNS	See Tables 4.3-1(b), 4.3-1(c), 4.3-1(d)

**Table 4.3-1(a) SSAR Nuclear Design Criteria
Codes and Standards for the Plant General Electric ABWR (Continued)**

Item	Nuclear Design Criteria	Class				Construction Codes	Comment
		Quality ⁽¹⁾	Seismic ⁽²⁾	Tornado	Safety ⁽³⁾		
IV. Components							
A) Pumps	NI	NNS	NNS	NNS	NNS	NI	See Tables 4.3-1(b), 4.3-1(c), 4.3-1(d)
B) Tanks & Vessels	NI	NNS	NNS	NNS	NNS	NI	See Tables 4.3-1(b), 4.3-1(c), 4.3-1(d)
C) Valves Sump Back Flow Check Others	NI	C NNS	NNS NNS	NA NNS	NNS NNS	NI	See Tables 4.3-1(b), 4.3-1(c), 4.3-1(d)
D) Heat Exchanger	NI	NNS	NNS	NNS	NNS	NI	See Tables 4.3-1(b), 4.3-1(c), 4.3-1(d)
E) Switch Gear	NI	NNS	NNS	NNS	NNS	NI	See Tables 4.3-1(b), 4.3-1(c), 4.3-1(d)
F) MCC	NI	NNS	NNS	NNS	NNS	NI	See Tables 4.3-1(b), 4.3-1(c), 4.3-1(d)
G) Fans, Dampers	NI	NNS	NNS	NNS	NNS	NI	See Tables 4.3-1(b), 4.3-1(c), 4.3-1(d)

Footnotes for Table 4.3-1 (a):

1. For this and all other Tables, Quality Class A, B, C and D as defined in R.G.1.26.
2. Seismic designation given in SSAR Section 3.2 (Note: Seismic Category I = SSC designed for both OBE and SSE; Seismic Category II=SSC design for OBE only).
3. Safety designation given in SSAR Section 3.2. (Note: Safety Class is not defined by the USNRC. If Safety Class designation is given, it is either a local designation or Safety Class as defined ANS Stds. 51.1 or 52.1).
4. There is some confusion as to the seismic design basis of the Radwaste Building Superstructure. SSAR Section 3.8.4.1.3 states that although not a Seismic Category I Structure, its major structure is designed to resist Seismic Category I loads. SSAR Section 3.8.4.2.3 states that the UBC is used for design. These two seismic design requirements are not the same, so it is not clear how the super structure will be designed.
5. Process Piping forming part of a containment boundary will be designed to Safety Class 2; Quality Class B and Seismic Class I.
6. Tornado Class defined in Regulatory Guide 1.76:

NA=Not Applicable NNS=Non Nuclear Safety NI = No Information

Table 4.3-1 (b) SSC Classification for GE ABWR Summary						
Principal Component	Safety Class ⁽²⁾	Location ⁽³⁾	Quality Group Classification ⁽⁴⁾	Quality Assurance Requirement ⁽⁵⁾	Seismic Category ⁽⁶⁾	Notes
Radwaste System 1. Drain Piping including supports and valves-radioactive	N	C,H,SC,T,W,X	D	E	-	(7)
2. Piping including supports and valves forming part of containment boundary	2	SC	B	B	1	
3. Pressure vessels including supports	NNS	W	-	E	-	(7)
4. Atmospheric tanks including supports	NNS	SC,T,W	-	E	-	(7)
5. 0-103.42 kPaG Tanks and supports	NNS	W	-	E	-	(7)
6. Heat exchangers and supports	NNS	SC,W	-	E	-	(7)
7. Piping including supports and valves	NNS	SC,T,W	-	E	-	(7)
8. Other mechanical and electrical modules	NNS	ALL	D	E	-	(7)
9. ECCS equipment room sump backflow protection check valves	NNS	SC	C	B	1	
10. Control Building high water level sensors	3	X	-	B	1	
11. Electrical modules and cables with safety-related functions	3	SC,X,RZ	-	B	1	
Radioactive Waste Control Panels	NNS	W	-	E	-	(7)
Radwaste Building 1. Structural walls and slabs above grade level (see Subsection 3H.3.3)	NNS	W	-	E	-	
2. Radwaste Building Substructure	3	W	-	B	1	(7)

Footnotes to Table 4.3-1(b):

- (1) A module is an assembly of interconnected components which constitute an identifiable device or piece of equipment. For example, electrical modules include sensors, power supplies, signal processors, and mechanical modules include turbines, strainers, and orifices. Safety-related motor control centers, power centers, metal clad switchgear, and remote multi-plexing units in the Reactor Building are located outside the Secondary Containment in the emergency electric equipment rooms.
- (2) Safety-related structures, systems, and components of the ABWR Standard Plant are classified for design requirements as Safety Class 2, or Safety Class 3 in accordance with their nuclear safety importance. The notation NNS indicates non nuclear safety related.

The definitions of the safety classes are based on Section 3.3 of ANS Standard 52.1, and examples of their broad application are given. Because of specific design considerations, these general definitions are subject to interpretation and exceptions on a case by case basis.

- (3) Structures, systems, and components that are not SC-1, or -3 are non-nuclear safety-related (NNS).

RZ	=	Reactor Building Clean Zone (portion of the reactor building outside the Secondary Containment Zone)
SC	=	Secondary Containment portion of the reactor building
T	=	Turbine Building
W	=	Radwaste Building
X	=	Control Building

- (4) A,B,C,D, = Quality groups are defined in Regulatory Guide 1.26.
- (5) B = The quality assurance requirements of 10CFR50, Appendix B are applied in accordance with the quality assurance program described in Chapter 17 of the SSAR.

E = Elements of 10CFR50, Appendix B are generally applied, commensurate with the importance of the equipment's function.
- (6) I = The design requirements of Seismic Category I structures and equipment are applied as described in Sections 3.7 and 3.8 and 3.9 of the SSAR.
- (7) A Quality Assurance Program meeting the guidance of Regulatory Guide 1.143 is applied during design and construction.

Table 4.3-1(c) Minimum Design Requirements for an Assigned Safety Designation in GE ABWR				
		Minimum Design Requirement		
Safety Designation	Quality Group ⁽¹⁾	Seismic Category ⁽²⁾	Electrical Classification ⁽³⁾	Quality Assurance ⁽⁴⁾
SC-1	A	1	-	B
SC-2	B	1	-	B
SC-3	C	1	1E	B
NNS	(1)	(2)	-	-

Footnotes for Table 4.3.1-(c):

- (1) Quality group for SSC given in Table 4.3-1(a).
- (2) Seismic Categories are given in Table 4.3-1(a).
- (3) Safety-related electrical equipment and instrumentation are designed SC-3 and are designed to meet IEEE Class 1E (as well as Seismic Category I) design requirements.
- (4) Safety-related structures, systems and components meet the quality assurance requirements of 10CFR50, Appendix B, as described in SSAR Chapter 17.

Table 4.3-1(d) Quality Group Designations-Codes and Industry Standards For GE ABWR								
Quality Group Classification	ASME Section III Code Classes ⁽¹⁾	Pressure Vessels and Heat Exchangers ⁽¹⁾	Pipes, Valves and Pumps	Storage Tanks 0-15 psig	Storage Tanks Atmospheric	ASME Section III Component Parts	Core Support Structures	Primary Containment Boundary
A	1	NCA & NB TEMA C	NCA and NB	-	-	NCA and NF	-	-
B	2	NCA and NC TEMA C	NCA and NC	NCA and NC	NCA and NC	NCA and NF	-	-
	CC and MC CS	-	-	-	-	-	NG	-
C	3	NCA and ND TEMA C	NCA and ND	NCA and ND	NCA and ND	NCA and NF	-	-
D	-	- ASME Section VIII Div 1 - TEMA	Piping/Valves ANSI B31.1 Pumps⁽²⁾	API-620 ⁽³⁾	- API-650⁽³⁾ - AWWA-D100 - ANSI B96.1	-	-	-

Footnotes for Table 4.3-1(d):

- (1) Applicable Classes of the ASME Code Section III.
 (2) For pumps classified in Group D, ASME Code Section VIII, Division 1, shall be used as a guide in calculating the wall thickness for pressure-retaining parts and in sizing the cover bolting.
 (3) Tanks shall be designed to meet the intent of API, AWWA, and/or ASME B96.1 Standards as applicable.

Table 4.3-2 SSAR DESIGN CRITERIA
Applied Loads
General Electric ABWR

Item	Operating Conditions	Extreme Loads							Accident Loads	
		Seismic	Tornado	Wind	External Flood	Accidental Explosion	Malevolent Vehicle	Other		
I. Building Structure										
A) Radwaste Bldg.										
1) Substructure	NI	I	NNS	NA	Y ⁽²⁾	NI	NI	NI	NI	NI
2) Super Structure	NI	Y ⁽¹⁾	N	Y	N	NI	NI	NI	NI	NI
II. Systems										
A) Liquid	NI	NNS	NA	NA	NA	NA	NI	NI	NI	NI
B) Gaseous	NI	NNS	NA	NA	NA	NA	NI	NI	NI	NI
C) Solid	NI	NNS	NA	NA	NA	NA	NI	NI	NI	NI
III. Distribution Systems										
A) Piping	NI	NNS	NA	NA	NA	NA	NI	NI	NI	NI
1. Process ⁽³⁾										
2. Fire	NI	NNS	NA	NA	NA	NA	NI	NI	NI	NI
3. Inst. & Cont.	NI	NNS	NA	NA	NA	NA	NI	NI	NI	NI
B) Cable Trays	NI	NNS	NA	NA	NA	NI	NI	NI	NI	NI
C) Ventilation	NI	NNS	NA	NA	NA	NI	NI	NI	NI	NI
IV. Components										
A) Pumps	NI	NNS	NA	NA	NA	NI	NI	NI	NI	NI
B) Tanks & Vessels	NI	NNS	NA	NA	NA	NI	NI	NI	NI	NI
C) Valves	NI	N	NA	NA	NA	NI	NI	NI	NI	NI

Table 4.3-2 SSAR DESIGN CRITERIA
Applied Loads
General Electric ABWR (Continued)

Item	Operating Conditions	Extreme Loads						Accident Loads	
		Seismic	Tornado	Wind	External Flood	Accidental Explosion	Malevolent Vehicle		
D) Heat Exchanger	NI	N	NA	NA	NA	NI	NI	NI	NI
E) Switch Gear	NI	N	NA	NA	NA	NI	NI	NI	NI
F) MCC	NI	N	NA	NA	NA	NI	NI	NI	NI
G) Fans, Dampers	NI	N	NA	NA	NA	NI	NI	NI	NI
V. Misc Items									
A) Off Gas System	NI	I	NA	NA	NA	NI	NI	NI	NI

Footnotes for Table 4.3-2:

- (1) There is some confusion as to the seismic design basis of the Radwaste Building Super Structure. SSAR Section 3.8.4.1.3 states that although not a Seismic Category I structure, its major structures are designed to resist Seismic Category I loads. SSAR Section 3.8.4.2.3 states that the UBC will be used for design. These two seismic design requirements are not the same so it is not clear how the Super Structure will be designed.
- (2) Flood protection both internal or external is provided by located SSC above the flood level or enclosing Seismic Class 1, 2 or 3 items in concrete water tight vaults.
- (3) Process piping formerly part of a containment boundary will be designed to Safety Class 2, Quality Class B and Seismic Class 1.
Y = Yes; N = No; NNS = Non-Nuclear Safety; NA = Non Applicable; NI = No Information.

Table 4.3-3 - SSAR or Design Criteria Code and Standards and Nuclear Design Criteria ABB/CE System 80 ⁺						
Item	Nuclear Design Criteria	Class				Comment
		Quality	Seismic	Tornado	Safety	
I. Building Structure A) Radwaste Bldg. Liquid and Solid B) Nuclear Annex C) Gaseous Waste	NI	2 ⁽¹⁾	Cat I ⁽²⁾	Cat. I ⁽³⁾	2 ⁽¹⁾	Except for Commitment to Req. of RG 1.143, detailed Const. Codes not specified.
II. Systems A) Liquid	All: 10 CFR 20 App. B 10 CFR 50 App. I	NI	NI	NI	NI	Except for Commitment to Req. of RG 1.143, detailed Const. Codes are specified.
B) Gaseous	NUREG 0017	NI	NI	NI	NI	
C) Solid	ANSI N 13.1 R.G's 1.21, 4.15, 8.8	NI	NI	NI	NI	
III. Distribution Systems A) Piping 1. Process	NI	2	NNS	NNS	2	NI
2. Fire	NI	2	NNS	NNS	2	NI
B) Piping/Tubing (Inst.)	NI	2	NNS	NNS	2	NI
C) Cable Trays	NI	2	NNS	NNS	2	NI
E) Ventilation	NI	2	NNS	NNS	2	NI

Table 4.3-3 - SSAR or Design Criteria Code and Standards and Nuclear Design Criteria ABB/CE System 80* (Continued)						
Item	Nuclear Design Criteria	Class			Construction Codes	Comment
		Quality	Seismic	Tornado	Safety	
IV. Components						
A) Pumps	NI	2	NNS	NNS	2	NI
B) Tanks & Vessels	NI	2	NNS	NNS	2	NI
C) Valves ⁽⁵⁾	NI	2	NNS	NNS	2	NI
D) Heat Exchanger	NI	2	NNS	NNS	2	NI
E) Switch Gear	NI	2	NNS	NNS	2	NI
F) MCC	NI	2	NNS	NNS	2	NI
G) Fans, Dampers	NI	2	NNS	NNS	2	NI

Footnotes for Table 4.3-3:

1. Quality Class 2 A, B, C and D is related to Regulatory Guide 1.26 as defined in Section 3.2 of the SSER.
2. Seismic designation given in SSAR Section 3.2.
3. Tornado Class Defined in Regulatory Guide 1.76

NNS = Non Nuclear Safety NI = No Information

Table 4.3-4 SSAR DESIGN CRITERIA Applied Loads ABB/CE System 80 ⁺									
Item	Operating Conditions	Extreme Loads						Accident Loads	
		Seismic	Tornado	Wind	External Flood	Accidental Explosion	Malevolent Vehicle	Other	
I. Building Structure A) Radwaste Bldg. (Liquid and Solid Waste)	NI	Y	NI	Y	NI	NA	NI	NI	NI
B) Nuclear Annex (Gaseous Waste)	NI	Y	Y	Y	Y	NA	NI	NI	NI
II. Systems A) Liquid	NI	N	N	N	N	NA	NA	NI	NI
B) Gaseous	NI	N	N	N	N	NA	NA	NI	NI
C) Solid	NI	N	N	N	N	NA	NA	NI	NI
III. Distribution Systems A) Piping ⁽¹⁾									
1. Process	NI	N	N	N	N	NA	NA	NI	NI
2. Fire	NI	N	N	N	N	NA	NA	NI	NI
3. Inst. & Cont.	NI	N	N	N	N	NA	NA	NI	NI
B) Cable Trays	NI	N	N	N	N	N	N	NI	NI
C) Ventilation	NI	N	N	N	N	N	N	NI	NI
IV. Components A) Pumps	NI	N	N	N	N	N	N	NI	NI
B) Tanks & Vessels	NI	N	N	N	N	N	N	NI	NI
C) Valves ⁽¹⁾	NI	N	N	N	N	N	N	NI	NI

Table 4.3-4 SSAR DESIGN CRITERIA Applied Loads ABB/CE System 80+ (Continued)									
Item	Operating Conditions	Extreme Loads						Accident Loads	
		Seismic	Tornado	Wind	External Flood	Accidental Explosion	Malevolent Vehicle	Other	
D) Heat Exchanger	NI	N	N	N	N	N	N	NI	NI
E) Switch Gear	NI	N	N	N	N	N	N	NI	NI
F) MCC	NI	N	N	N	N	N	N	NI	NI
G) Fans, Dampers	NI	N	N	N	N	N	N	NI	NI

Footnotes for Table 4.3-4:

1. Piping and valves in lines which penetrate containments are designed for seismic loads.
N = No; Y = Yes; NA = Not Applicable; NI = No Information

Table 4.3-5(a) - SSAR Design Criteria Code and Standards and Nuclear Design Criteria W-AP600							
Item	Nuclear Design Criteria	Class				Construction Codes	Comments
		Quality	Seismic	Tornado	Safety		
I. Building Structure							
A. Auxiliary Bldg.	NI	C	C-I	I	I	ACI-349	
b. Radwaste Bldg.	NI	D	NNS	NNS	NNS	ACI-318	
II. Systems							
A) Liquid	NI	D	NNS	NNS	NNS	See Table 4.3-5(c)	See Table 4.3-5(b)
B) Gaseous	NI	D	NNS	NNS	NNS	See Table 4.3-5(c)	See Table 4.3-5(b)
C) Solid	NI	D	NNS	NNS	NNS	See Table 4.3-5(c)	See Table 4.3-5(b)
III. Distribution Systems							
A) Piping							
1. Process	NI	D	NNS	NNS	NNS	See Table 4.3-5(c)	See Table 4.3-5(b)
2. Fire	NI	D	NNS	NNS	NNS	See Table 4.3-5(c)	See Table 4.3-5(b)
B) Piping/Tubing (Inst.)	NI	D	NNS	NNS	NNS	See Table 4.3-5(c)	See Table 4.3-5(b)
C) Cable Trays	NI	D	NNS	NNS	NNS	See Table 4.3-5(c)	See Table 4.3-5(b)
E) Ventilation	NI	D	NNS	NNS	NNS	See Table 4.3-5(c)	See Table 4.3-5(b)
IV. Components							
A) Pumps	NI	D	NNS	NNS	NNS	See Table 4.3-5(c)	See Table 4.3-5(b)
B) Tanks & Vessels	NI	D	NNS	NNS	NNS	See Table 4.3-5(c)	See Table 4.3-5(b)
C) Valves	NI	D	NNS	NNS	NNS	See Table 4.3-5(c)	See Table 4.3-5(b)
D) Heat Exchanger	NI	D	NNS	NNS	NNS	See Table 4.3-5(c)	See Table 4.3-5(b)
E) Switch Gear	NI	D	NNS	NNS	NNS	See Table 4.3-5(c)	See Table 4.3-5(b)

Table 4.3-5(a) - SSAR Design Criteria Code and Standards and Nuclear Design Criteria W-AP600 (Continued)						
Item	Nuclear Design Criteria	Class			Construction Codes	Comments
		Quality	Seismic	Tornado		
F) MCC	NI	D	NNS	NNS	See Table 4.3-5(c)	See Table 4.3-5(b)
G) Fans, Dampers	NI	D	NNS	NNS	See Table 4.3-5(c)	See Table 4.3-5(b)

NNS = NON NUCLEAR SAFETY

NI = NO INFORMATION

Table 4.3-5(b) Comparison of Safety Classification Requirements for W-AP600								
AP600 Code Letter ⁽¹⁾	ANS Equipment Safety Class ⁽²⁾	RG 1.29 Seismic Design Requirements ⁽³⁾	ASME Code, Sec. III Class ⁽⁴⁾	IEEE Requirements	RG 1.26 NRC Quality Group ⁽⁵⁾	10 CFR 50 Appendix B ⁽⁶⁾	Inspection & Testing Requirements	Required Test & Maintenance
A	SC-1	I	1	NA	Group A	Yes	Yes ⁽⁷⁾	(8)
B	SC-2	I	2	NA	Group B	Yes	Yes ⁽⁷⁾	(8)
C	SC-3	I	3	1E	Group C	Yes	Yes ⁽⁷⁾	(8)
D	NNS ⁽²⁾	NA ⁽⁹⁾	NA ⁽¹⁰⁾	⁽¹⁰⁾	Group D	NO ⁽¹⁰⁾	Yes ⁽¹¹⁾	⁽¹¹⁾
Other	NNS ⁽²⁾	NA ⁽¹³⁾	NA	NA	NA	NA ⁽¹²⁾	NA	NA

Footnotes for Table 4.3-5(b):

1. A single letter equipment classification identifies the safety class, quality group, and other classifications for AP600. See the SSAR subsection 3.2.2 for definitions
2. AP600 safety classification is an adaptation of that defined in ANSI 51.1. The NNS defined in the ANSI 51.1 standard is divided into several AP600 equipment classifications namely, Classes D, E, F, L, P, R, and W.
3. See SSAR Subsection 3.2.1 for definition of seismic categories.
4. ASME Boiler and Pressure Vessel Code, Section III defines various classes of structures, systems, and components for nuclear power plants. It defines criteria and requirements based on the classification. It is not applicable for non-safety-related components.
5. The quality group classification corresponds to those provided in Regulatory Guide 1.26.
6. "Yes" means quality assurance program is required according to 10 CFR 50 Appendix B.
7. "No" means quality assurance program is not required according to 10 CFR 50 Appendix B.
8. Class A, B, and C, structures, systems, and components build to ASME Code, Section III are inspected to ASME Code, Section XI requirements. See the text for additional specification of requirements.
9. Class A, B, and C structures, systems, and components that are required to function to mitigate design basis accidents have some testing requirements included in the plant technical specifications. In addition to the requirements in the technical specifications, testing and maintenance requirements are included in an administratively controlled reliability assurance plan.
10. See SSAR Subsection 3.2.1 for cases when Seismic Category II requirements are applicable for Class D structures, systems, and components.
11. See the text for a discussion of the industry standards used in the construction of Class D structures, systems and components.
12. Class D structures, systems, and components have selected reliability assurance programs and procedures to provide availability when needed. These programs are administratively controlled programs and are not included in the technical specifications.
13. Normal industrial procedures are followed in procuring, designing, fabricating, and testing these non-safety-related structures, systems, and components.

NA - Not Applicable

TABLE 4.3-5(c) W-AP600 Classification of Mechanical and Fluid Systems, Components, and Equipment				
Description	AP600 Class	Seismic Category	Principal Construction Code	Comments
Radwaste Building Ventilation System (VRS)				
Shutoff, Isolation and Balancing Dampers	L	NNS	ANSI/AMCA-500	
Fire Damper	Note 1	NNS	UL-555	
Air Handling Units w/Filters	L	NNS	ASME N509	
Fans, Ductwork	L	NNS	ASME N509	
Balance of system components are Class E or Class L	NI	NI	NI	
Gaseous Radwaste System (WGS)				Location: Auxiliary Building
Gas Cooler	D	NNS	ASME VIII/TEMA	
Sample Pumps	D	NNS	Manufacturer Std.	
Guard and Delay Beds	D	II	ASME VIII	Design for 1/2 SSE
Moisture Separator	D	NNS	ASME VIII	
Valves Providing WGS	D	NNS	ANSI 16.34	
AP600 Equipment Class D Function	NI	NI	NI	
Liquid Radwaste System (WLS)				Location: Containment and Auxiliary Building
Heat Exchangers, WLS and CCS Side	D	NNS	ASME VIII/TEMA	
Pumps	D	NNS	Manufacturer Std.	
Tanks	D	NNS	ASME III without Code Stamp	
De-gasifier	D	NNS	ASME VIII	
Ion Exchangers	D	NNS	ASME VIII	
Filters	D	NNS	ASME VIII	

TABLE 4.3-5(c) W-AP600 Classification of Mechanical and Fluid Systems, Components, and Equipment (Continued)				
Description	AP600 Class	Seismic Category	Principal Construction Code	Comments
Valves Providing WLS AP600 Equipment Class D Function (local drain valves in Radwaste Building)	D	NNS	ANSI 16.34	
Radioactive Waste Drain System (WRS)				Location: Auxiliary Building
Pumps	D	NNS	Manufacturer Std.	
Valves Providing WRS AP600 Equipment Class D Function	D	NNS	ANSI 16.34	
Solid Radwaste System (WSS)				Location Auxiliary Building
Pumps	D	NNS	Manufacturer Std.	
Tanks	D	NNS	ASME VIII	
Filters	D	NNS	ASME VIII	
Valves Providing WSS AP600 Equipment Class D Function	D	NNS	ANSI 16.34	
Balance of System Components are Class E	NI	NI	NI	
Waste Water System (WWS)				
System Components are Class E	NI	NI	NI	Location: Various
On-site Standby Power System (ZOS)				Location: Diesel Generator Building
Diesel Generator Engines	D	NNS	Manufacturer Std.	
Diesel Generator Starting Units	D	NNS	NEMA MG 1	
Diesel Generator Radiators	D	NNS	CAGI	
Diesel Generator Silencers	D	NNS	API 661	
Valves Providing ZOS Diesel D Generator Engines AP600 Equipment Class D Function	D	NNS	ANSI 16.34	
Balance of System Components are Class E	NI	NI	NI	

Footnotes to Table 4.3-5(c):

1. Fire dampers are constructed to the requirements of UL-555 or UL-Ps if they are fire and smoke dampers and are located in Class D, Class L, and Class R ducts.

NI = NO INFORMATION

NNS = NON NUCLEAR SAFETY

Table 4.3-6 - SSAR or Design Criteria Applied Loads Westinghouse AP 600									
Item	Operating Conditions	Extreme Loads							Accident Loads
		Seismic	Tornado	Wind	Flood-External	Accidental Explosion	Malevolent Vehicle	Other	
I. Building Structure A. Auxiliary Bldg. B. Radwaste Bldg.	I NNS	I NNS	I NNS	I NNS	NI NNS	NI NNS	NI NNS	NI NI	NI NI
II. Systems A) Liquid B) Gaseous C) Solid		NNS NNS NNS	NA NA NA	NA NA NA	NA NA NA	NI NI NI	NI NI NI	NI NI NI	NI NI NI
III. Distribution Systems A) Piping 1. Process 2. Fire 3. Inst. & Cont.		NNS NNS NNS	NA NA NA	NA NA NA	NA NA NA	NI NI NI	NI NI NI	NI NI NI	NI NI NI
B) Cable Trays		NNS	NA	NA	NA	NI	NI	NI	NI
C) Ventilation		NNS	NA	NA	NA	NI	NI	NI	NI
IV. Components A) Pumps		NNS	NA	NA	NA	NI	NI	NI	NI
B) Tanks & Vessels		NNS	NA	NA	NA	NI	NI	NI	NI
C) Valves		NNS	NA	NA	NA	NI	NI	NI	NI
D) Heat Exchanger		NNS	NA	NA	NA	NI	NI	NI	NI

Table 4.3-6 - SSAR or Design Criteria Applied Loads Westinghouse AP 600 (Continued)									
Item	Operating Conditions	Extreme Loads						Accident Loads	
		Seismic	Tornado	Wind	Flood- External	Accidental Explosion	Malevolent Vehicle		
E) Switch Gear		NNS	NA	NA	NA	NI	NI	NI	NI
F) MCC		NNS	NA	NA	NA	NI	NI	NI	NI
G) Fans, Dampers		NNS	NA	NA	NA	NI	NI	NI	NI

Footnotes for Table 4.3-6:

NA = Not Applicable; NI = No Information; NNS = Non Nuclear Safety

4.4 Operating Plant Waste Management Facility Commitments

As part of the review effort, an in-depth review of the licensing basis commitments of five operating Commercial Nuclear Power facilities was conducted. This was done to insure that the existing operating plant commitments were given adequate consideration in the development of suggested changes to Regulatory Guide 1.143. The plants reviewed were:

- Perry Unit 1-BWR-6 (GE) Construction Permit 5/77
- Waterford Unit 3-PWR (CE) Construction Permit 11/74
- Beaver Valley 2-PWR (W) Construction Permit 5/74
- Washington Public Power Construction Permit
- Supply 2 - BWR-5 (GE) 3/73 Permit
- Davis Besse 1-PWR (B&W) Construction Permit 3/71

These plants included recent vintage commercial facilities from all the major reactor vendors and in addition considered both the GE BWR-5 MARK II and GE BWR-6 MARK III containment designs.

Initially other earlier plants were also selected for evaluation. However, insufficient information was presented in their SAR's to permit meaningful evaluation of these earlier plant waste management systems since in the earlier plants, emphasis was placed on reactor related systems with little information presented with regard to their waste management system structural-mechanical design.

It should be noted that all of the plants reviewed (indeed all operating nuclear power plants in the U.S.) had their initial design criteria established by the issue date of the construction permit which was well before the initial issuance of RG 1.143 in 1979. For this reason the applicability of RG 1.143 had to be negotiated on a case by case basis during the active construction period of each plant.

The structural-mechanical waste management criteria applicable to the five operating nuclear power plants

can be found in Tables 4.4-1 to 4.4-10. In Tables 4.4-3(a) to 4.4-3(x) are the detailed design criteria of the Waterford Unit 3 Waste Components as an example of the type of information contained in Section 11 of a current UFSAR.

4.4.1 Perry - 1 Waste Management Systems

The Waste Management Systems of the Perry 1 Plant provide the means for controlling, handling, storage and preparation for disposal of liquid, gaseous and solid wastes.

4.4.1.1 Liquid Waste Management Systems

The liquid radioactive waste (LRW) system is designed to collect and treat, for reuse or disposal, all radioactive (or potentially radioactive) liquid wastes produced in the plant. This is done in such a manner that, for all anticipated quantities of waste produced, the availability of the plant for power generation is not adversely affected.

The LRW system is designed to restrict releases of radioactive material to the environment and exposures to both operating personnel and the general public to ALARA in accordance with the guides given in Appendix I to 10 CFR 50.

The LRW system is designed in accordance with the following design criteria:

- a. For each reactor at the site, the estimated annual total quantity of radioactive material (excluding tritium) above background in the liquid effluents released to unrestricted areas is less than 5 curies.
- b. For the total radioactive liquid effluents of Units 1 and 2, the resultant normal design basis whole body dose to any individual offsite is less than 5 mrem/yr.
- c. Design and construction of all LRW system components satisfies or exceeds the intent of all applicable criteria set forth in Regulatory Guide 1.143 (as detailed in FSAR Section 1.8) and ANSI N197-1976.
- d. All LRW system components and the structure in which they are housed are designed and constructed in accordance with the codes, standards, seismic classifications, and safety classifications listed in FSAR Table 3.2-1.

- e. LRW components are located in areas of sufficient size and accessibility to facilitate efficient maintenance.

Audits are performed to determine the radiological consequences for design basis situations in which equipment malfunctions and/or operator errors are hypothesized during periods of operation at design basis fuel leakage. Design provisions are included to prevent the uncontrolled release of radioactive material to the environment as a result of any single equipment malfunction or operator error.

Redundancy of equipment for collecting and processing inputs to the liquid waste system is described in detail in Section 11.2.2 and summarized by Table 11.2-8 of the FSAR. Considerable excess capacity is built into the collecting and processing equipment of each subsystem to handle all anticipated normal and maximum input quantities. An evaluation of this capability is presented in FSAR Table 11.2-9 which shows that only when the suppression pool is drained for maintenance does the LRW system fall short of needed capacity. This occurrence is satisfactorily handled by reducing the rate at which the suppression pool is drained. This method adds at most, three days to the outage and results in no increase in radiation exposures to operating personnel or the general public.

Considerable detail as to the structural-mechanical design of the equipment and distribution systems relative to the liquid radwaste systems is found in FSAR Section 11.2. However, the type of information provided is not consistent. For example, FSAR Table 11.2-6 provides details on liquid radwaste pipe lines to include design code, material and wall schedule. Specific design data presented in FSAR Table 11.2-7 for liquid waste equipment does not mention design code or wall thickness but does include pressure and temperature design criteria.

4.4.1.2 Gaseous Waste Management Systems

The objective of the Perry Unit 1 Gaseous Waste Management system is to process and control the release of gaseous radioactive effluents to the site environs to maintain ALARA, the exposure of persons in unrestricted areas to radioactive gaseous effluents (Appendix I to 10 CFR 50). This is to be performed while maintaining occupation exposure ALARA and without limiting plant operation or availability.

The gaseous effluent treatment systems are designed to limit the dose to offsite persons from routine station releases to significantly less than the limits specified in 10 CFR 20.

In addition, the Off-Gas Treatment System limits the dose to offsite persons from a Control Rod Drop Accident to significantly less than the limits specified in 10 CFR 100.

The gaseous radwaste equipment is selected, arranged and shielded to maintain occupational exposure ALARA in accordance with Regulatory Guide 8.8, and 10 CFR 20.

The gaseous effluent treatment system is designed to the requirements of the General Design Criteria that follow

4.4.1.2.1 General Design Criterion 60

The system has sufficient capacity to reduce the offgas activity to permissible levels for release during normal operation, including anticipated operation occurrences, and to avoid any termination of releases or limitation of plant operation due to unfavorable site environmental conditions.

4.4.1.2.2 General Design Criterion 64

Continuous monitoring of activity levels in the system upstream of the delay line provides advance notice of any potentially significant increase in releases. Continuous monitoring of the system effluent, with automatic isolation at activity levels corresponding to administrative release limits and annunciation at lower levels, along with continuous monitoring of the plant vent release, provide assurance that activity releases to the environment will in all events be maintained within established limits.

- (a) **Leakage of Radioactive Gases**
Leakage of radioactive gases from the system is limited by welding piping connections where possible and using bellows stem seals or equivalent valving. The system operates at a maximum of 48kPa (7 psig) during startup and less than 13.8kPa (2 psig) during normal operation so that the differential pressure to cause leakage is small.
- (b) **Hydrogen Concentration**
Hydrogen concentration or gases from the air ejector is kept below the flammable limit by maintaining adequate process steam flow for dilution at all times. The steam flow rate is monitored and alarmed. Furthermore, hydrogen concentration is monitored by the redundant hydrogen analyzers prior to the holdup process. If the hydrogen concentration exceeds 2 percent, an alarm is set to annunciate in the control room.

- (c) **Field Run Piping**
No piping in this system is field routed. This includes major process piping, drain lines, steam lines, and sample lines.
- (d) **Single Failures and Operator Errors**
Design provisions are incorporated which preclude the uncontrolled release of radioactivity to the environment as a result of any single operator error or of any single equipment malfunction short of the catastrophic equipment failures described in FSAR Chapter 15.7.

Design precautions taken to prevent uncontrolled releases of activity include the following.

- (a) The system design minimizes ignition sources so that a hydrogen detonation is highly unlikely even in the event of a recombined failure.
- (b) Even though measures are taken to avoid a possible detonation, the system pressure boundary is designed to be detonation-resistant.
- (c) All discharge paths to the environment are monitored.
- (d) Dilution steam flow to the steam jet air ejector is monitored and alarmed, and valving is such that loss of dilution steam cannot occur without coincident loss of motive steam, so that the process gas is sufficiently diluted if it is flowing at all.
- (e) **Maintainability of Gaseous Radwaste System**

Design features which reduce or ease required maintenance include the following:

- (a) Redundant components for all active, in-process equipment pieces.
- (b) No rotating equipment in the process stream.
- (c) Rotating equipment is located in the system only where maintenance can be performed while the system is in operation

Design features which reduce leakage and releases of radioactive material include the following:

- (a) Extremely stringent leak rate requirements placed upon all equipment, piping and instruments, and enforced by system integrity

testing.

- (b) Use of welded joints wherever practicable
- (c) Specification of valve types with extremely low leak rate characteristics, i.e., bellows seal, double stem seal or equal.
- (d) Use of loop seals with automatic shutoff valves to prevent loss of liquid due to siphoning.
- (e) Specification of stringent seat leakage characteristics for valves and lines discharging to the environment through other systems.

4.4.1.2.3 Seismic Design

This Section presents the seismic design criteria used at the Perry Plant for Radwaste facilities and equipment.

(a) Equipment

Equipment and components used to collect, process or store gaseous radioactive waste are not designed as Seismic Category I.

(b) Buildings Housing Gaseous Radioactive Waste Processing Systems

That portion of the offgas system upstream of the gas dryer refilters is housed in the turbine building, which is a non-seismic, non-safety class, reinforced concrete structure. The remaining portion of this system is located in the offgas building, which is a Safety Class 3, Seismic Category I, reinforced concrete structure. A detailed discussion of the seismic design for this building is given in Section 3.7 of the FSAR.

4.4.1.2.4 Detonation Resistance

Even though the system is designed to be free of ignition sources, the process pressure boundary of the system is detonation resistant. The pressure vessels are designed to withstand 2413kPa (350 psig) static pressure, and piping and valving are designed to resist dynamic pressures encountered in long runs of piping at the design temperature. This analysis is covered in a proprietary report submitted to the NRC.

4.4.1.3 Solid Radioactive Waste Management System

The primary design objective of the solid radioactive

waste (SRW) system is to control, collect, handle, process, and package all wet and dry solid radioactive waste generated by the Perry Unit 1 as a result of normal operation, and to store these wastes until they are shipped to authorized receiving and storage areas offsite. This is done in such a manner that, for all anticipated quantities of waste produced, the availability of the power plant for power generation is not adversely affected.

Subsequent to the original design of the SRW system, modifications have been made to the condensate cleanup and liquid radwaste systems which could result in quantities or activities of solid radioactive waste which are different than those in the original design basis. The control, processing and packaging of the solid radioactive waste remains unchanged.

Packaging of solid radioactive material is accomplished in a manner which ensures that no radioactive material will be released to the environment during shipment of the waste to offsite burial or storage facilities. The SRW system is designed to limit exposures to both operating personnel and the general public to ALARA.

The following are the primary design criteria for the Solid Radiation Waste (SRW) System

- (a) The SRW system components, piping and the structure that houses the system are designed and fabricated in accordance with the codes, standards, seismic categories, and quality group classifications given in FSAR Table 3.2-1.
- (b) The SRW system design is in compliance with the guidance provided by Regulatory Guide 1.143 and Branch Technical Position ETSB 11-3.
- (c) All wet radioactive waste (filter backwash slurries, spent resins, evaporator bottoms) are dewatered or solidified with cement prior to shipment offsite. Packaging and transporting of radioactive wastes is performed in conformance with 10 CFR 71 and applicable ICC and DOT regulations.
- (d) The SRW system design and shielding provision ensure that (during all phases of processing, handling and shipment of radioactive waste) exposure to operating personnel and the general public is within the applicable limits of 10 CFR 20, 49 CFR 173 and ALARA in accordance with Regulatory Guide 8.8.
- (e) The SRW system design provides a means to verify the absence of free liquid in the containers in accordance with Branch Technical Position

ETSB 11-3.

- (f) The SRW system design, equipment sizing and equipment redundancy ensure that the maximum expected quantities of all radioactive waste inputs during any 30 day period can be solidified, and/or dewatered, packaged, and temporarily stored onsite without affecting plant availability.

The SRW system is designed to process evaporator concentrate, spent resin slurry, precoat-type filter backwash slurry, and traveling belt discharge cake. These waste streams are transferred from the LRW system collection tanks or traveling belt filters to one of the two redundant waste mixing/dewatering tanks on a batch basis. After this transfer, the fill isolation valve is closed and the fill line is backflushed to the tank from which the waste stream originated. The mixing/dewatering tanks can also be bypassed when processing waste from the LRW collection tanks. This bypass line ties directly in with a vendor's solidification/dewatering system.

Filling of the waste mixing/dewatering tank and subsequent processing of the waste are controlled automatically through a programmable logic controller in the SRW control panel. Using selector switches on this panel, the operator selects which waste storage tank to take waste from and which waste mixing tank to use.

After the tank filling step is completed, a positive displacement type waste feed pump transfers the waste slurry at a preset rate to a vendor's mobile solidification/dewatering system.

Once onsite processing of the waste is complete, the overhead bridge crane picks up the container and takes it either to a short term storage area or to a truck bay where it is loaded for transfer to an authorized receiving, reprocessing, or storage area.

The SRW system is designed to preclude the accidental release of radioactive waste into the solid waste packaging area due to component failure or system malfunctions. Instrumentation and controls monitor each phase of the packaging operation, serving to detect possible system malfunctions and terminate the packaging operation as required to prevent inadvertent releases of radioactive waste into the solid waste.

4.4.1.4 Waste System Design Basis Leaks or Failure

The following radioactive gas waste system components are evaluated for several failure mode conditions.

- (a) Main condenser gas treatment system failure.
- (b) Malfunction of main turbine gland sealing system.
- (c) Failure of air ejector lines.

4.4.1.4.1 Main Condenser Gas System Failure

Those events which could cause a gross failure in the off-gas treatment systems are:

- (a) A seismic occurrence exceeding the seismic capabilities of the equipment.
- (b) A hydrogen detonation which ruptures the system pressure boundary.
- (c) A fire in the filter assemblies.
- (d) Failure of adjacent equipment which could subsequently compromise offgas equipment.

The seismic event is considered to be the most probable and is the only conceivable event which could cause significant system damage.

The equipment and piping are designed to contain any hydrogen-oxygen detonation which has a reasonable probability of occurring. A detonation is not considered as a possible failure mode.

The decay heat on the filters is insignificant and cannot serve as an ignition source for the filters.

The system is isolated from other systems or components which could cause any serious interaction or failure.

4.4.1.4.2 Malfunction of Main Turbine Gland Sealing System

Postulated malfunctions of the turbine gland sealing system including the failure of the steam seal evaporator and its backup steam supply, failure of the steam packing exhaust fan and excessive pressure in the steam seal header are considered.

This event is categorized as a limiting fault.

4.4.1.4.3 Failure of Main Turbine Steam Air Ejector Lines

Those events which could cause a failure in the main turbine steam air ejector systems are:

- (a) Failure of the steam line to the air ejectors.
- (b) Failure of the air ejector suction line.
- (c) Failure of the air ejector discharge line to the offgas system.

In each of these postulated failures it is assumed that the worst case condition exists and that the failure is in a section of line common to both air ejectors so as to negate the use of the standby air ejector.

This event is categorized as a limiting fault.

4.4.1.4.4 Radioactive Liquid Waste System Failures (Release to Atmosphere)

The liquid radwaste treatment systems are classified as quality Group D. Radioactive releases considered include postulated rupture of radwaste tanks, equipment malfunction or small leaks in the system process lines that transport liquid radwaste.

4.4.1.4.5 Postulated Radioactive Releases Due to Liquid-Containing Tank Failures

It is considered highly improbable for significant cracks to develop in the Seismic Category I safety class buildings containing radioactive waste materials and equally improbable for a waste tank to fail. However, it is postulated that an unspecified event causes a failure of a tank in the radwaste building and subsequent failure of the radwaste building.

This failure is classified as a limiting fault.

4.4.2 Waterford Unit-3 Radioactive Waste Management Systems

The Waste Management Systems (BMS and WMS) of the Waterford Unit 3 provide the means for controlled handling, storage and disposal of liquid, gaseous and solid wastes. In addition, the EMS provides the mechanism for reconcentrating and recovering dissolved boron from the liquid effluent for reuse in the plant.

Liquid effluent from the Emergency Cooling System first passes through the purification filter in the Chemical Volume Control System. It is then processed in the BMS by successively passing through the flash tank, filters, ion exchangers and a boric acid concentrator. These operations remove the radioactive material and concentrate the boric acid for reuse or draining. All other radioactive liquid wastes are processed in the WMS for release to the environment or solidification and draining. All liquid wastes are sampled prior to release. The waste release rates are ALARA and within the guides and limits for waste release established by 10CFR20 and 10CFR30, Appendix I.

All solid wastes are stored in suitable containers for ultimate off-site disposal in accordance with applicable regulations.

Waste gases are either collected in the gas surge header or filtered and released to the atmosphere via the gas collection header, depending on expected activity level. High activity gases are collected in the gas surge header and compressed into gas decay tanks. The waste gas held in the gas decay tanks is released to the plant vent after sampling. The tank contents are released at rates well within the limits established by 10CFR20 and 10CFR50, Appendix I.

Additional radwaste handling capability has been provided for by installing means to allow the use of portable solidification and demineralization systems.

See FSAR Chapter 11 for detailed information.

4.4.2.1 Liquid Waste Management System

Radioactive liquid wastes which are discharged from the plant are first processed by the Waste Management System (WMS) or the Boron Management System (BMS).

The principal design objective of the WMS system is to protect plant personnel, the general public, and the environment by ensuring that all releases of radioactive materials, both in plant and to the environment are ALARA and within the requirements of 10CFR20 and Appendix I to 10CFR50.

The WMS and BMS are located in the Seismic Category I Reactor Auxiliary Building.

Tanks outside of containment are provided with level indication and alarms for high level conditions. The level alarms will alert operators when tanks are nearly full and transfer flow from filled tank to alternate tanks should proceed. For cubicles which contain tanks with significant radioactivity and thus require shielding, the floors in these cubicles are pitched to floor drains located at low points to facilitate floor drainage.

Collection tanks and tanks which receive processed waste are generally provided in pairs. The pairing of tanks allows one tank to be in the fill mode while the other tank is in the sampling, recirculation, process, or standby mode. An empty standby tank would normally be available for any filled tank. Thus, switching from one tank to another will prevent overflow of the tanks.

The monitor tanks (i.e., boric acid condensate tanks, waste condensate tanks, and laundry tanks) can be

sampled prior to discharge or prior to transferring to the outside water storage tanks (i.e., primary water storage tank or the discharge structure). If analysis indicates further processing is required, the water will be reprocessed prior to leaving the building. Therefore, the potential release of water by overflowing outside storage tanks would not result in uncontrolled release of radioactive materials. The outside storage tanks have level indication instrumentation which will annunciate under high level conditions.

In addition to the outside storage tanks discussed previously, equipment drain tank, spent resin tank, waste storage tank and concentrate storage tank do not have backup tanks. The equipment drain tank does not have an overflow nozzle and hence does not have the potential for overflow to the floor. Overflow of the concentrate storage tank and spent resin tank will go onto the floor of tanks' respective cubicles. Curbing prevents the spread of liquid from the cubicles. Provisions to prevent entry of the concentrates and resin into the floor drain system are necessary to prevent potential solidification within the drainage system.

Equipment in the BMS and WMS which requires shielding, is placed in separate cubicles. The separate cubicles are provided with appropriate shielding. This allows operation to continue by using alternate process routes while performing maintenance on a particular piece of equipment.

To reduce gaseous radioactive releases to the building atmosphere the following design features have been incorporated into the plant design:

- (a) venting of high radiation level equipment is provided and collected in the vent gas collection header
- (b) tanks are provided with water seals on overflow lines to prevent radioactive gases from escaping, and
- (c) diaphragm valves are used to provide protection against leakage.

The WMS and BMS provide means to control the discharge of liquid waste. The operator in the main control room discharges the waste from either the waste condensate, laundry, or boric acid condensate tanks through flow meters recorders and radiation monitors which automatically alarms and terminates discharge flow on high activity. Other features and procedures used to prevent inadvertent releases to the environment from the system include strict administrative procedures, operator training, and redundant discharge valves.

4.4.2.2 Gaseous Waste Management Systems

Radioactive gases are collected and processed through the following systems depending upon their origin:

- (a) Gaseous Waste Management System (GWMS)
- (b) Vent Gas Collection Header (VGCH)
- (c) Main Condenser Evacuation System (MCES)
- (d) Turbine Gland Sealing System (TGSS)
- (e) Building Ventilation Systems
- (f) Atmospheric Dump Valves

The principal design objectives of the GWMS are as follows:

- (a) To protect the plant personnel, the general public and the environment by insuring that gaseous releases of potentially radioactive materials both in plant and to the environs are as low as practicable in accordance with 10CFR20 and 10CFR50 Appendix I.
- (b) To provide a means to collect, store, sample and monitor potentially radioactive gaseous wastes generated during plant operations as specified in 10CFR50, Appendix A, General Design Criterion 60. These wastes may contain noble gases, airborne halogens and radioactive material in particulate form.

Implementation of 10CFR50, Appendix A, General Design Criterion 64 is discussed in FSAR Section 11.5.

The numerical design objectives for plant releases during normal operations, including anticipated operational occurrences, are based on 10 CFR50 Appendix I. The design objectives are:

- (a) The calculated annual air dose due to gamma radiation at or beyond the site boundary is not to exceed 10 millirads.
- (b) The calculated annual air dose due to beta radiation at or beyond the site boundary is not to exceed 20 millirads.
- (c) The calculated annual total quantity of all radioactive iodine and radioactive material in particulate form will not result in an annual dose or dose commitment to any organ of an individual in an unrestricted area from all pathways in excess of 15 mrem.

The GWMS is classified as non-seismic Category I and non-nuclear safety. However, the Gas Decay Tanks interconnecting piping, and components located

between the upstream and downstream valves used to isolate the GDT from the rest of the system are required by Regulatory Guide 1.143 to be designed to meet seismic criteria for an Operating Basis Earthquake (OBE). All major components and piping supports were designed to meet seismic Category I (SSE) criteria. Use of non-seismic supports may be considered on other than the GDT and associated equipment in accordance with plant procedures.

The GWMS is designed to process maximum expected surges to the gas surge header with minimum impact on operations. There are two waste gas compressors to process the gas, one of which is in a standby mode to handle surges to the system that the on-line compressor cannot handle. The flow rates of the compressors are such that one compressor has sufficient capacity to process influents at their normal expected flow rates with sufficient reserve capacity in the standby compressor to handle surges.

The GWMS provides a means to control the discharge of gaseous waste. The discharges are controlled by the operator in the main control room. The operator discharges the gas decay tanks through a flow meter and recorder, and a radiation monitor, which automatically terminates discharge flow on high activity. The release is controlled by a needle valve preceded by a pressure regulator to ensure constant flow rate. The procedure of sampling the gas decay tank prior release protects discharge. The procedure for sampling and monitoring also protects against radiation monitor malfunction since the sample prior to discharge will be representative of tank contents.

The GWMS is designed to prevent or preclude explosive mixtures rather than withstand an internal explosion. The gas analyzer package is provided to monitor hydrogen and three oxygen concentrations via various plant components where potentially explosive mixtures could develop. The gas analyzer operates continuously by monitoring through a programmed sequence of sample sources. The gas analyzer is also capable of monitoring a single sample source for as long as desired by manually overriding the sequence selector. Each sample source is purged, analyzed and recorded. Continuous recording of sample concentrations allows for the detection and observation of trends which may be developing. When the analysis indicates that the hydrogen or oxygen concentration of a sample exceeds a predetermined set point, an alarm is annunciated.

If automatic analyzer operation is interrupted, samples can be obtained from the grab sample port on the analyzer, and from local sample ports at selected

locations in the GWMS. The sample port on the analyzer will allow manual sample collection of a tank which has sufficient pressure to drive the gas to the analyzer, if the analyzer pump is inoperable. Local sample lines are also located on headers where samples can be taken manually. Samples taken can then be analyzed by a portable analyzer or taken to the radiochemistry lab. This will allow GWMS operation to continue until the analyzer is back in automatic operation.

In addition to the gas analyzer package, which can automatically take sequential sample points, a separate gas analyzer with alarm capabilities will be used to continuously monitor between the common discharge of the waste gas compressors and the gas decay tanks.

A continuous backup oxygen analyzer has been installed on the high pressure side of the waste gas compressors in addition to the cyclic automatic gas analyzer already in place. This additional analyzer will give better indication of oxygen levels in the GWMS.

4.4.2.3 Solid Waste Management System

Low-level solid radioactive wastes are processed, packaged and stored for subsequent shipment and off-site burial by the Solid Waste Management System (SWMS). Wastes include spent ion exchange resin, used filter cartridges and miscellaneous refuse.

The design bases of the SWMS are as follows:

- (a) Provide for receipt and interim storage of spent resins. The SWMS provides for transferring concentrates from boric acid concentrator packages to the SWMS.
- (b) Provide for holdup capacity for normally radioactive spent resin to allow decay of short-lived radionuclides.
- (c) Provide for receipt and water content adjustment (dewatering and/or dilution) of radioactive wastes such as spent resins.
- (d) Provide remote means for transferring used filter cartridges from their respective filters to the Solidification or Compaction area. Appropriate lifting grabs and remote handling tools will be used during filter transfer.
- (e) Provide means to compact Low Specific Activity (LSA) solid waste such as contaminated clothing, rags, paper, laboratory equipment and supply items.

- (f) Provide means for encapsulation and solidification of wastes in large volume disposable liners for subsequent shipment and off-site burial.
- (g) Prevent the release of significant quantities of radioactive materials. The overall exposure to the public and operating personnel will be kept within the requirements of 10CFR20 and 10CFR50.
- (h) Disposal and transportation of wastes from the SWMS, including the shipping containers, will satisfy the requirements of 10CFR20, 10CFR71, 10CFR61, and 49CFR.
- (i) The in-plant SWMS is housed in the Reactor Auxiliary Building which meets the seismic criteria of Regulatory Guide 1.143.
- (j) Components of the in-plant SWMS are classified "NNS" at Waterford 3 (Regulatory Guide 1.26 - Quality Group "D"). Further discussion of the conformance of the SWMS with Regulatory Guide 1.143 and Branch Technical Position ET5B 11-3 can be found in FSAR Table 11.4-10.
- (k) The SWMS meets the QA requirements of Regulatory Guide 1.143.

The SWMS is comprised of the in-plant solidification system, the vendor solidification system and dewatering system, the spent resin handling system, filter handling, and the dry active waste handling system.

The facilities buildings, and systems described in this subsection are designed and constructed such that handling waste containers and operating equipment of the SEMS can be done with the least possible personnel exposure in accordance with the ALARA criteria of Regulatory Guides 8.8 and 8.10. In addition, plant procedures specify particular practices for waste handling and monitoring for the various systems and storage areas that also adhere to the ALARA requirements of Regulatory Guides 8.8 and 8.10.

4.4.2.3.1 Active Waste Handling

The bulk dry waste material is collected in containers as it is generated in the RCA. The waste is surveyed for radiation prior to feeding into the box compactor or loaded for transportation to a licensed volume reduction facility. Plant procedures provide guides for monitoring the dry waste for materials that could cause chemical reactions or spontaneous combustion.

An on-site box compactor utilizing hydraulic pressure or a licensed volume reduction facility may be used to volume reduce radioactive waste such as contaminated clothing, rags, paper, low activity filters, activated charcoal and HEPA filters from plant ventilation systems, and miscellaneous contaminated material generated by maintenance and operations of the facility.

4.4.2.4 Solidification Building

The function of the Solidification Building (SB) is to provide shelter for the vendor's solidification equipment and to supply the necessary service requirements and waste delivery for this system. In addition to service provided to this facility (air, water, electric power), a 9 metric ton (10 ton) overhead crane is provided to handle the solidification equipment and waste containers.

The bridge crane has adequate travel over the interior area of the Solidification Building.

The waste solidification (and/or resin dewatering) operation has provisions for the use of shielded containers. Waste is supplied to the liner by flexible hoses connected to waste transfer lines routed inside the RAB out to the SB. Support equipment for the vendor system is mounted on skids. The SB will supply space for the trailer, the skid mounted equipment, and materials required by the contractor for solidification and dewatering (See FSAR Figure 11.4-2).

The SB is a pre-engineered, prefabricated sheet metal building. The building is located on the concrete truck platform on the west side of the RAB, between the RAB main access doors and the solidification area doors at elevation + 6.0m (21 ft).

The design of the SB satisfies the applicable requirements of Regulatory Guide 1.143, NUREG-0800, BTP-ETSB 11-3 and the applicable ACI and AISC Codes and Standards. The BTP, which specifies minimum requirements for portable solid waste systems, calls for equipment to be located on concrete pads with curbs and drainage provisions for containing radioactive spills. To manage potential spills, a sump is provided in the SB with provisions to return any collected liquids to the plant radwaste system.

4.4.2.5 Compactor Building

In order to provide capacity for projected peak volumes of dry compacted waste, Waterford 3 has installed a metal building to house a dry radioactive and non-compacted waste compaction machine and storage area

for the compacted dry waste metal boxes. This building is located inside the security fence between the water treatment building and the chiller building. The design of this facility complies with the intent of NUREG-0787.

The metal building meets the applicable requirements of Regulatory Guide 1.143, NUREG-0800, SRP 11.4, BTP-ETSB 11-3, Appendix 11.4A and the applicable ACI and AISC Codes and Standards. The metal building is a completely enclosed structure which includes a separate partitioned room housing the compactor machine. The compactor machine, as described in Section 11.4.7 contains its own hood, ventilation fan, and high efficiency filtration system which processes the displaced air during the compaction cycle. This filtered air is then discharged into the compactor room, thus minimizing the risk of airborne radioactive contamination. The compactor room is constructed with smooth, continuous internal surfaces to facilitate decontamination should it ever be required. The entire building is constructed on a reinforced concrete slab with curbing such that water is kept from entering the building.

The use of curbing to separate the areas within the interim dry compacted waste building was evaluated during the design and determined to be unnecessary for the following reasons:

- (a) The building was designed with special attention given to ensuring that hard driving rain would not penetrate to the interior of the structure.
- (b) The building slab has special curbing and is raised above grade to prevent groundwater from flowing into the dry storage area.
- (c) The metal box containers containing dry compacted waste stored in this building have their own individual pallets which raise the bottom of the containers approximately 2.39m (94 inches) off the floor.
- (d) The box containers have solid seams and are water tight up to the lid which is approximately 1.2m (4 ft) off the floor.

In summary, water intrusion should not occur due to the design features of the building.

The building is 16.8m (55 ft.) by 12.2m (40 ft) and has sufficient storage capacity to house eighteen (18) 1.22m x 1.22m x 1.83m (4 ft. x 4 ft. x 6 ft.) metal boxes of compacted waste. The boxes are handled with a

forklift.

4.4.2.6 Storage Facilities

The solidified waste storage facility is shown on the FSAR general arrangement drawing and is presently used for the vendor demineralizer system. This storage capacity has been replaced by the outdoor storage area.

Due to revised operating philosophy and a re-evaluation of radwaste generation rates, the outside interim storage area provides sufficient space for over two months of shielded storage for wastes which are normally stored before shipment. To provide for temporary storage of solid wastes prior to shipment, storage areas are designed by the Radiation Protection Group. One designated storage area that will be used to store wastes is located west of the Turbine Building and south of the Solidification Building. This area is inside the plant Controlled Area and can be accessed by trucks and mobile heavy lifting equipment.

Wet waste, i.e., solidified concentrates spent filter cartridges and waste resins, are processed in disposable and/or storage containers. Waste ready for shipment and requiring shielding can be stored in this interim solid waste storage area.

The decision whether or not to store waste, shielded or unshielded, is made by the Radiation Protection group in accordance with 49CFR, 10CFR and plant technical specifications. Shielding can be provided as necessary.

Plant procedures contain provisions for periodic radiological surveys of stored waste to ensure that radiation levels are below the limits of 10CFR20.

The concrete On-site Storage Container (OSC) design incorporates a high strength liner for additional leakage protection and provisions for sampling and collecting any liquid found in the OSC cavity. Plant procedures provide for sampling and analysis of any residual water found in OSC prior to shipment and prior to dispositioning the water. The OSC's have a tube extending to the bottom of the inner cavity of the OSC's which allows sampling and removal of water with a portable pump.

The radwaste storage area can accommodate up to eight OSC's without stacking containers. Should circumstances dictate, this area could be expanded and additional on-site storage containers procured in a short period of time to provide additional shielded storage capacity. Plant procedures provide requirements to not only survey for external radiation and contamination,

but also to periodically inspect waste containers to ensure that no container deterioration is occurring during temporary storage.

4.4.2.7 Low Level Radwaste Storage Facility

The Low Level Radwaste Storage Facility (LLRWSF) is located outside the plant's protected area. The facility is a 24.4m (80 ft.) wide by 42.7m (140 ft.) long by 15.2m (50 ft.) tall steel frame building with metal siding and designed to support a traveling crane. The size of the facility was established based upon the total number of waste containers, size of the concrete vault, and requirements for inspection aisles. The LLRWSF has the capacity to store sixty (60) Sea/Land containers and 32 HICs.

The facility will contain four concrete cubicles to store HICs. Each cubicle has the capacity to hold eight HICs, i.e. four stacked two high.

The LLRWSF will be equipped with a natural convection ventilation system consisting of a building ridge vent, two supply fans and louvers mounted in the building walls. The system is of sufficient capacity to moderate inside temperatures for worker comfort and safety. There is no freeze protection in the facility.

The LLRWSF will be provided with 480 volt electrical power to supply power for the interior and exterior lighting, the overhead traveling bridge crane, supply fans, and receptacles throughout the facility. Weatherproof distribution panels will be mounted to the exterior of the LLRWSF per the request of the Waterford 3 insurers, American Nuclear Insurers and Nuclear Mutual Limited. Lightning protection will also be provided and the building grounding will be tied into the plant grounding system.

The LLRWSF is considered a storage occupancy with a fire loading classification of low hazard (NFPA-101 Section 4-2.2.2) since it will be unoccupied except during the delivery of waste and all waste will be stored in noncombustible containers or within a concrete vault. In order to reduce the number of openings in the surrounding concrete wall, only one personnel door will be installed in the south side of the building along with one roll-up door. Per NFA 101, "Life Safety Code", Chapter 29-2.4.1, Exception No. 1, for low hazard storage occupancies, only a single means of egress is required from any story or section: therefore, one exit is acceptable. Furthermore, since the LLRWSF will be unoccupied, no potable water will be required.

4.4.2.8 Accident Analysis Releases Due to Liquid Containing Tank Failures

It would take any release of contaminated liquid about 1000 years to travel to the nearest boundary of the restricted area. It should be noted that there are no potable water intakes or drinking wells within the restricted area. Due to radioactive decay, the total radionuclide concentration at this point after about 1000 years was calculated to be less than 10^{-6} of the Maximum Permissible Concentration as defined in Column 2 of Table II of Appendix B to 10 CFR 20.

This analysis was based on a liquid waste system which operated on the principle of evaporation. The configuration presently used is based on the principle of demineralization. The demineralizer system was evaluated using very conservative values to determine a maximum saturation activity level (i.e. : system total curies) using FSAR Reactor Coolant Isotopic Inventories to determine a weighted average influent and finally determining a dilution factor representative of the LWM stream. The system spent resin (solid waste) was conservatively treated as a liquid waste.

4.4.3 Beaver Valley Unit 2 Radioactive Waste Management Systems

The Beaver Valley Power Station - Unit 2 (BVPS-2) design of radioactive waste management systems, structures, and components meets the intent of Regulatory Guide 1.143 with the following clarifications and alternatives:

4.4.3.1 Waste Handling Building

The waste handling building is located adjacent to, and west of, the turbine building. It is a four-story, plus basement, structure.

The waste handling building is supported on a reinforced concrete foundation mat. The roof and walls of the top two stories are steel framed with metal siding and roof deck. The remainder of the structure is reinforced concrete.

The waste handling building is designed to provide biological shielding where required. The waste handling building is Non-Seismic Category I. With the exceptions described in FSAR Sections 3.7B.2 (use of other than RG1.60 shaped site spectra), the seismic design requirements are in accordance with Regulatory Guide 1.143.

The gaseous waste storage tank (GWST) enclosure is

located north at the fuel building. It is an in-ground one story structure.

The GWST enclosure is supported on a reinforced concrete foundation mat. The walls, roof, and interior structures are also reinforced concrete.

The GWST enclosure is designed to provide biological shielding where required. The GWST enclosure is non-Seismic Category I. With the exceptions described in FSAR Section 3.7B.1 and 3.7B.2 regarding spectra, the seismic design requirements are in accordance with Regulatory Guide 1.143. (See above comment concerning R.G. 1.60).

Beaver Valley Power Station - Unit 2 uses the modal time-history technique to generate floor response spectra. Refer to FSAR Section 3.7B.2

Beaver Valley Power Station - Unit 2 uses ACI-318-71 for reinforced concrete construction. This was the code in effect at the time of design. The differences between this code and ACI-318-77 for this building are judged insignificant in the BVPS-2 FSAR.

4.4.3.2 Liquid Waste Management Systems

This section describes the capabilities of BVPS-2 to control, collect, process, handle, store, recycle, and dispose of liquid radioactive waste generated as a result of normal BVPS-2 operations, including anticipated operational occurrences. The liquid waste management system (LWMS) includes the radioactive liquid waste system and utilizes some of the components of the steam generator blowdown system (SGBS) for processing.

The LWMS is designed to meet the anticipated processing requirements of BVPS-2. Adequate storage capacity is provided to process liquid wastes during periods when major processing equipment may be down for maintenance and during periods of excessive waste generation.

The refueling water storage tank is not provided with a dike or retention pond. Tank overflow is directed to the liquid waste system via piping.

The refueling water storage tank is fabricated in accordance with the requirements of ASME BPVC Section III, Class 2. The atmospheric tank is tested full of water to ensure that there are no leaks. Tank overflow is directed to the liquid waste system via the safeguards building sump piping. The radionuclide concentrations of the refueling water storage tank liquid will be determined following each refueling.

The Beaver Valley Power Station - Unit 1 Turbine Building is connected to the BVPS-2 pipe trench, just north of the electrical cable tunnel, by a pipe trench. This pipe trench, 8 feet to 9 feet wide by approximately 6 feet deep, runs north from the BVPS-2 pipe trench and then west to the BVPS-1 turbine building.

The cross-tie pipe trenches are reinforced concrete subsurface structures with the top of the trench covers approximately level with the ground grade.

The cross -tie pipe trenches are non-seismic Category I. With the exceptions described in FSAR Section 3.7B.2, the seismic design requirements are in accordance with Regulatory Guide 1.143.

4.4.3.3 Gaseous Waste Management Systems

The radioactive GWD system continuously processes and monitors all waste gas streams at BVPS-1 and BVPS-2 prior to discharge to the atmosphere. The system provides decay time for the BVPS-2 Boron Recovery System (BRS) degasifier gaseous effluent (FSAR Section 9.3.4.6) and the off gas stream from the air ejector system (FSAR Section 10.4.2) for BVPS-1 and BVPS-2, as necessary. The provision for storing gases generated from either unit when going to cold shutdown is also provided. The system further provides for recycling the major portion of the hydrogen present in the degasifier overheads back to the Volume Control Tank (VCT).

The GWMS system and its associated ventilation systems are designed to meet the following criteria;

- (a) The systems are designed to meet the requirements of 10CFR20, and the dose design objectives specified in the Annex to Appendix I to 10CFR50, including provisions to treat gaseous radioactive wastes such that:
 - (1) The calculated annual total quantity of all radioactive material released from the site to the atmosphere does not result in an estimated annual external dose from gaseous effluents to any individual in unrestricted areas in excess of 5 mrem to the total body, or 15 mrem to the skin.
 - (2) The calculated annual total quantity of all radioactive iodine and radioactive material in particulate form, released from the site to the atmosphere, does not result in an estimated annual dose or dose commitment for any individual in an unrestricted area from all pathways of exposure in excess of 15 mRems

to any organ.

- (3) The instantaneous dose rates of radioactive materials in gaseous effluents released to an unrestricted area do not exceed the limits in the Offsite Dose Calculation Manual.
- (b) The radioactive GWMS system is designed to meet the anticipated processing requirements of BVPS-2. Adequate capacity is provided to process gaseous wastes during periods when major processing equipment may be down for maintenance (single failures), and during periods of excessive waste generation.
- (c) The GWMS system is classified non-nuclear (NNS) class and is designated QA Category II and III. The GWMS and structures housing GWMS meet the seismic requirements of Regulatory Guide 1.143 with the exceptions noted above.

The gaseous waste delay beds, gaseous waste surge tank, waste gas chiller, overhead gas compressors and piping are designed and fabricated to ASME III. The valves are designed and fabricated to ANSI B31.1. The gaseous waste storage tanks are designed and fabricated to ASME VIII, Division 1.

All equipment manufacturing codes equal or exceed those specified in Table 1 of Regulatory Guide 1.143.

The gaseous waste system piping within the gaseous waste storage tank vault structure is designated non-seismic and has been analyzed for dead load and thermal stress only.

Failure of the gaseous waste system piping within the gaseous waste storage tank vault will not adversely affect any other systems or components because the storage tank vault contains only gaseous waste system piping. Technical Specification 3.11.2.5 provides a limit on the quantity of radioactivity contained in each group of gaseous waste storage tanks. The basis of this requirement is to restrict the quantity of radioactivity so any release due to a postulated leak or failure will not exceed the requirements of NRC Branch Technical Position ETSB 11-5, Rev. 0, dated July 1981, of 0.5 Rem at the exclusion area boundary. This accident is described in FSAR Section 15.7.1.

The storage tank system is used to store all the gas generated by either BVPS-1 or BVPS-2 when going to a cold shutdown condition. The system uses seven tanks, each of which has individual solenoid operated isolation valves, a pressure indicator, a high pressure alarm, and overpressure requirements and subsequently

were seismically mounted in the gaseous waste storage tank vault structure. The seven tanks will normally all be in service at one time, but can be operated in smaller groupings.

The piping downstream of the overhead gas compressors is one-half inch.

The discharge piping for the overhead gas compressors is designed to not impose any unnecessary pressure loss in that portion of the gaseous waste system.

4.4.3.4 Solid Radwaste Management System

The solid radwaste management system is designed to collect, hold, process, solidify, package, handle, and temporarily store radioactive materials prior to their shipment offsite and ultimate disposal.

The solid radwaste system is designed in accordance with the following criteria:

- (a) The system design parameters are based on radionuclide concentrations and volumes consistent with reactor operating experience for similar designs, and with the source terms of FSAR Section 11.1.
- (b) All radwastes that are wet will be solidified or dewatered prior to their shipment offsite. A process control program and operating procedures are used to verify the absence of free liquid in the containers. Provisions exist to reprocess containers having free liquid in accordance with Branch Technical Position (BTP) ETSB 11-3 (USNRC 1981). Processing equipment is sized to handle the design condition inputs without the need to ship bulk liquids. Onsite waste storage facilities provide sufficient storage capacity to allow time for shipping delays. Storage capacity will exceed 30 days' accumulation at expected generation rates in accordance with BTP ETSB 11-3.
- (c) Solid waste containers, shipping casks, and methods of packaging meet applicable federal regulations (10CFR71 and 49CFR171 through 178). Wastes to be shipped to a licensed burial site are in accordance with applicable U.S. Nuclear Regulatory Commission, U.S. Department of Transportation (USDOT), and state regulations.
- (d) The solid radwaste system components and piping are designed in accordance with the criteria of BTP ETSB 11-3 and Regulatory Guide 1.143, except as noted in Section 1.8, BVPS-2 position on Regulatory Guide 1.143.
- (e) The system contains provisions to control leakage and to facilitate operations and maintenance in accordance with BTP ETSB 11-3 and Regulatory Guides 1.143 and 8.8.
- (f) The filling of containers, the solidification, and the storage of radioactive solid wastes conform to 10CFR20.106 and 10CFR50, Appendix A, General Design Criteria 60, 63 and 64. In addition, 10 CFR 50.34a requirements and Regulatory Guide 8.8 guides are met to ensure occupation radiation exposures are ALARA.
- (g) The waste storage facilities in the condensate polishing building are shielded to provide protection for operating personnel in accordance with the radiation protection design considerations in Section 12.1.2. Solidification agents are stored in accordance with BTP ETSB 11-3.
- (h) The system is not safety-related and is classified as non-nuclear safety class, and QA Category II.
- (i) The part of the foundation and adjacent walls of the condensate polishing building up to a height sufficient to contain the liquid inventory in the building is designed to the seismic criteria of paragraph 5, and to the quality assurance criteria of paragraph 6, of Regulatory Guide 1.143. The specific building design criteria are given in FSAR Section 3.8.
- (j) The solid waste system and the structures housing the solid waste system meet the seismic requirements of Regulatory Guide 1.143 with the exceptions noted above.
- (k) Those portions of the system that handle liquid radwaste meet the applicable design bases of FSAR Section 11.2.1.
- (l) The system is designed to process, handle, and store the waste types and quantities generated as a result of normal operation, including anticipated operation occurrences, as described in FSAR Section 11.4.2.

Materials handled as solid wastes include concentrated waste solutions from the BVPS-2 evaporators, spent resin from radioactive process demineralizers, and condensate polishing system radioactive powdered resin sludges.

4.4.3.5 Design Basis Accidents

(a) Design Basis Waste Gas System Failure

The waste gas system failure is classified as an American Nuclear Society (ANS) Condition III event, an incident which may occur during the lifetime of a plant.

The failure and subsequent release of radioactive gases from various points in the Gaseous Waste System (GWS) has been evaluated. The following locations were considered for evaluation: the failure of an inlet line to the waste gas charcoal delay beds and a rupture of one gaseous waste storage tank.

The inlet line rupture analysis assumes that the noble gas inventory released to the environment from the ruptured lines is based on the sum of the following: a 1-hour release to the environment of degasifier effluent and the release of a fraction of the radioactivity absorbed on the charcoal delay beds. The reaction of noble gases released from the charcoal delay beds are given in FSAR Table 15.7-2. The total activity released from the ruptured line plus the charcoal delay beds is listed in Table FSAR 15.7-3.

For the GWST rupture, it is assumed that 100 percent of the noble gases produced from the complete degasification of the primary coolant is contained in the seven GWSTs. Since the tanks are not isolated from each other, the rupture of one tank is assumed to cause the release of the contents of all seven tanks.

(b) Postulated Radioactive Releases Due to Liquid Containing Tank Failures

The postulated radioactive release due to liquid-containing tank failure is classified as an ANS Condition III event, an incident which may occur during the lifetime of a plant. A more detailed description of Design Basis Accident can be found in FSAR Section 15.7.

4.4.4 Washington Public Power Unit 2 Waste Management Systems

The Waste Management Systems of the Washington Public Power Unit 2 provides the means for controlling, handling, storage and preparation for disposal of liquid, gaseous and solid waste.

4.4.4.1 Liquid Waste Management System

The liquid waste management system is designed to collect, segregate, store and process potentially radioactive liquids generated during normal plant operation and anticipated operational occurrences. The design objective is to keep the radiation doses in unrestricted areas ALARA within the guides of Appendix I to 10CFR50, May 5, 1975. The design incorporates the objectives of maximum recycling and minimum release of radioactive liquids without limiting plant operations or availability.

The criteria considered in the design of this system include volume, radioactivity, operation exposure and required quality for recycle of the processed liquid.

The system is designed to treat process liquids with radioisotope concentrations associated with the design basis fuel leakage and produce a quality of water which allows its recycle for plant reuse. Water inventory will at times require the discharge of processed liquids to the environs in which case concentrations of radioisotopes in the effluent will be significantly less than the values specified in 10 CFR 20 and within the release limits established in the technical specifications. Radiation exposure to persons in unrestricted areas, resulting from liquid waste discharged during normal operation and anticipated operational occurrences is less than the guides specified in 10CFR50, Appendix I. The radwaste building can retain the normal operating capacity of all liquid radwaste tanks in the event of an accident. FSAR Table 11.2-10 lists the design features of the tanks in the Radwaste Building used to prevent uncontrolled releases due to spillage, and shows overflow alarms and drainage paths.

The liquid waste management system meets the objectives of Regulatory Guide 1.143. Tanks which hold radioactive liquid including the condensate storage tanks, are monitored for level and alarm primarily in the radwaste control room. The radwaste systems are operated from the radwaste control room; hence, additional local alarms are not required for radwaste tank levels.

All liquid waste management system tank overflows and drains are routed to the radwaste building equipment and floor drain sumps. Radioactive liquid samples are primarily routed to sampling sinks. Those samples, which are local, drain into floor drain trenches, equipment or floor drain funnels, and pump beds. The above sample receivers are routed to various radioactive sumps, all of which are processed by the liquid waste management system.

Indoor tanks which hold radioactive liquid are not enclosed by individual curbs or elevated thresholds. The portion of the radwaste building which houses radioactive liquid tanks is Seismic Category I. The radwaste building can retain the normal operating capacity of all liquid radwaste tanks in the event of an accident. Therefore, the intent of Regulatory Guide 1.143 is met on this item.

Radioactive liquid wastes are collected and segregated into three categories, high purity waste, low purity waste and chemical waste. Wastes thus classified are treated in subsystems designated as equipment drain, floor drain and chemical waste respectively.

(a) High Purity Waste

High purity wastes, treated in the equipment drain subsystem, have low conductivity and relatively high radioactivity concentrations. Radioactive material is removed from these wastes by filtration and ion exchange. Following treatment and batch sampling, the processed waste is normally returned to condensate storage for reuse in plant.

(b) Low Purity Waste

Low purity wastes, treated in the floor drain subsystem, have moderate conductivity and generally low radioactivity. As for high purity wastes, radioactive material is removed here by filtration and ion exchange. Following treatment and batch sampling, the processed waste is normally returned to condensate storage for reuse in the plant.

(c) Chemical Waste

The high conductivity and organic content of these wastes preclude normal treatment in the system demineralizers by ion exchange. Chemical wastes can be neutralized by adding an appropriate neutralizing agent and thorough mixing. Following neutralization, if necessary, these wastes are presently routed to a back wash tank or phase separator where unexpended ion-exchange capacity of the resins is used to remove contaminants from the chemical waste. After thorough mixing and a period of quiescence to allow the ion-exchange process to proceed, the excess liquid is decanted to the floor drain sample tank for further processing. The spent resins in the separators are then processed as described in FSAR Section 11.2.2.1.

The installed chemical waste concentrators are

currently not used. Their preoperational testing and eventual use has been deferred until a need is identified for their use. There are currently no plans to activate the concentrators.

All liquid radwaste process streams terminate in one of the sample or distillate tanks. Since the liquid waste management system is operated in a batch basis, this arrangement allows each treated batch to be sampled to assure that the treatment was effective. If the sample indicates that the processed liquid is still above acceptable radioactivity limits or substandard in purity, equipment is provided to either recycle the batch through the same treatment or through a subsystem providing a higher degree of treatment. If the sample indicates that the level of activity is within limits required to discharge and the water is in excess of inventory capacity, the processed liquid may be discharged. The actual release of effluents from any processed and sampled batch tank requires the opening of a key-locked valve in accordance with written operating procedures. All required information regarding the batch release must be documented. These procedures are established to prevent inadvertent release of liquids that have not been suitably processed and analyzed.

Control of the liquid waste management system is from a shielded radwaste control room and from shielded local operating galleries. Flanges are provided where required for maintenance in an otherwise all welded system. The cells and concrete rooms provide secondary enclosures which facilitate collection of spills or leaks from the system for processing. Vessels that may have their contents mixed with air, or that may have batches transferred into them by means of pneumatic transfer, are closed and vented to the radwaste building ventilation system. Piping and tubing 2 inches and under is field routed, but required to be in specified space envelopes for shielding and in-plant exposure consideration.

The liquid waste management system is designed to minimize the effects of equipment malfunction and operator error. After initiation by the operator, valve positioning, equipment startup, and system operation can be automatically performed by a process controller. Failure of any valve to properly align, stops the sequence and prevents a pump start. Once on line, processing continues until the feed volume is processed or the capacity of a piece of process occurs with valves returning to the shutdown position. During initial system startup or in the event of controller failure, the processes are manually controlled. System variables, such as tank levels, flow rates, pressures and

conductivity are indicated and alarmed in the radwaste control room. The discharge from the liquid waste subsystems to the blowdown line is normally isolated by a closed manual valve and a key-locked closed manual valve separated by a telltale drain. This path is further protected by air operated isolation valves, flow control valves, flow indication and radiation monitoring instrumentation. These design features are complimented by administrative procedures which prevent inadvertent radioactive liquid releases and releases of liquids which exceed the technical specification release limit.

4.4.4.2 Gaseous Waste Management Systems

The objective of the gaseous waste management system is to process and control the release of gaseous radioactive effluents to the site environs so as to maintain as low as reasonably achievable the exposure of persons in unrestricted areas to radioactive gaseous effluents (Appendix I to 10CFR50, May 5, 1975). This is to be accomplished while maintaining occupation exposure as low as reasonably achievable and without limiting plant operation or availability.

The gaseous waste management systems are designed to limit the dose to off-site persons from routine station releases to significantly less than the limits specified in 10CFR20 and to operate within the emission rate limits established in the technical specifications.

The annual average exposure at the site boundary during normal operation and anticipated operational occurrences from gaseous effluents does not exceed the dose objectives of Appendix I to 10CFR50. The radiation dose design basis for the treated off-gas is to delay the gas until the required fraction of the radionuclides has decayed and the daughter products are retained by the charcoal and the HEPA filters.

The gaseous radwaste equipment is selected, arranged and shielded to maintain occupational exposure ALARA. The design of the system was accomplished prior to the issuance of Regulatory Guide 8.8. However, the system incorporates substantially the guidance provided in this regulatory guide. The gaseous effluent treatment system is designed to the requirement of General Design Criteria as follows:

4.4.4.2.1 General Design Criteria 60

The system has sufficient capacity to reduce the off-gas activity to permissible levels for release during normal operation, including anticipated operational occurrences, and to alleviate any termination of releases

or limitation of plant operation due to unfavorable site environmental conditions.

4.4.4.2.2 General Design Criteria 64

Continuous monitoring of activity levels in the system upstream of the delay line provides advance notice of any potentially significant increase in releases. Continuous monitoring of the system effluent, with automatic isolation limits and annunciation at lower levels, along with continuous monitoring of the reactor building elevated release duct, radwaste building ventilation exhausts and turbine general building ventilation exhausts, provide assurance that activity releases to the environment will in all events be maintained within established limits.

A list of the off-gas system major equipment items which includes materials, process conditions, and number of units supplied, is provided in FSAR Table 11.3-2. Equipment and piping is designed and constructed in accordance with the requirements of the applicable codes as given in FSAR Tables 3.2-1 and 3.2-2.

The quality group classifications of the various systems are shown in FSAR Table 3.2-1. Seismic category, safety class, quality assurance requirements, and principal construction codes information is contained in FSAR Section 3.2. The system is designed to Quality Group Classification D+.

Equipment and components used to collect, process, or store gaseous radioactive waste are not designed as Seismic Category I. Conservative analyses similar to those presented in FSAR Reference 11.3-5 demonstrate that equipment failure will not result in offsite doses exceeding .05 Rem.

Design precautions take to prevent uncontrolled releases of radioactivity include the following:

- (a) The system design eliminates ignition sources to that a hydrogen detonation is highly unlikely even in the event of a recombined failure.
- (b) The system pressure boundary is detonation resistant, despite the measures taken to preclude a detonation.
- (c) All discharge paths to the environment are monitored, the normal effluent path by the process radiation monitoring system, equipment areas by the area radiation monitoring system.

- (d) Dilution steam flow to the steam jet-air ejector is monitored and alarmed. Valve control logic causes the air ejector suction valves to close on low steam flow.

4.4.4.3 Solid Waste Management System

The objective of the solid waste management system is to collect, monitor, process, and package these waste products in a suitable form for offsite shipment and burial. The system capacity is sufficient to prevent impact on plant operations and availability due to anticipated operational occurrences. The system is designed to minimize exposure to operating personnel and to prevent undue exposure to persons.

In designing the system to meet the stated objective, the following criteria were applied:

The system has the capacity to handle the volumes of waste from normal operations and anticipated operation occurrences.

The system is designed to process the quantity of waste and concentration of radionuclides generated while maintaining occupational exposure ALARA. This is done by controlling the pipe run locations for shielding and exposure considerations, placing the process equipment in shielded areas, by providing remote operating stations, and by performing dewatering operations in radiation shields if required. The Radwaste Building shielding is designed for the highest radioactivity source, reactor water cleanup ADB resins.

In keeping with the ALARA philosophy and Appendix I to 10CFR50, the solid waste management systems's contribution to offsite doses is minimized by venting the process equipment to a filtered ventilation exhaust system and by directing the ventilation air flow from areas of low airborne contamination to areas of higher airborne contamination.

The solid waste management system operations and procedures are designed to limit the dose to offsite persons from station operations to significantly less than the limits specified in 10CFR20. Water separated in processing is returned to the liquid waste management system for treatment as described in FSAR Sections 11.2.2.1.5.

The system can accommodate a variety of shipping container sizes and shapes with and without shields. Provisions are made for the detection and removal of loose surface contamination on the waste containers. The radiation levels of the waste containers are monitored so that provisions can be made to ensure that

shipping regulation radiation levels are not exceed. Compliance with applicable regulations, e.g., 10CFR Parts 61 and 71 and 40CFR is discussed in FSAR Sections 11.4.2.9 and 11.4.3.

The safety class, quality group classification, quality class and seismic category of radwaste systems are specified in FSAR Table 3.2-1. FSAR Table 11.4-1 lists capacities, design pressure, and design temperature of the major equipment.

The solid waste management system is not designed to Seismic Category I. It is located in the Seismic Category I portion of the radwaste building. The seismic classification of the radwaste building is discussed in FSAR Section 3.8.4.1.2.

A portable solid waste management system is now utilized by WNP-2.

To assure that acceptable waste forms are produced for disposal in compliance with the requirements in 10 CFR 61, the WNP-2 process control program provides for characterization of individual waste streams, classification of final waste products, and proper verification that waste dewatering and/or solidification has been successful. In addition, waste storage, handling, and transportation activities take place under the process control program to ensure compliance with all applicable regulations in 10CFR20, 10CFR61 and 10CFR71. The WNP-2 process control program incorporates the following elements:

- (a) Process Control Systems
- (b) Waste Characterization
- (c) Dewatering and Solidification Processes (wet wastes)
- (d) Control, Instrumentation and Sampling Program
- (e) Solidification Process
- (f) Maintenance and Calibration
- (g) Waste Processing System Capacity
- (h) Waste Storage Capacity
- (i) Compliance with ALARA
- (j) Unanticipated Wastes
- (k) Waste Classification
- (l) Waste Packaging and Shipping

4.4.4.4 Radwaste and Control Building

The radwaste and control building is located west of the reactor building and diesel generator building and south of the turbine generator building.

The radwaste and control building houses equipment and components for safety related and non-safety related systems, some of the radwaste related systems

are:

- (a) Off gas filter system
- (b) Radioactive waste condensate demineralization system
- (c) Radioactive waste system and chemical waste process
- (d) Fuel pool cooling and cleanup systems for reactor building and radwaste and control building
- (e) Closed cooling water system for reactor building and radwaste and control building

The radwaste and control building is rectangular in shape. It is approximately 209 feet by 160 feet in plan, and 105 feet in overall height measured from the top of the foundation mat at elevation 437'-0".

Part of the building is a cast-in-place reinforced concrete structure and part is a structural steel framed structure. The portions of the building that are reinforced concrete are arranged so as to form a completely enclosed box-like structure, and are, therefore, comprised of concrete walls on four exterior sides and of top and bottom concrete slabs. The portions of the building that are structural steel framed are made up of steel framed floors and exterior walls.

The concrete portion of the building is a diaphragm slab/shear wall type structure, supported on a foundation mat which is separated and apart from the mats under each of the other structures in the plant complex. The floor and roof slabs are continuous one-way and two-way slab and beam systems monolithically constructed.

The steel framed portions of the building consist of conventionally braced framed wall systems with vertical bracing in bays along each wall. The girts and roof purlins are covered with insulated metal siding and insulated metal roof decking, respectively. The steel framed floors are fabricated of structural steel beam working in composite action with poured concrete slabs.

Areas housing safety related systems, equipment and components, and areas required to provide radiation shielding protection are constructed of reinforced concrete. In all other areas, structural steel construction is used.

The areas which are constructed of reinforced concrete are:

- (a) The radwaste area, which encompasses the foundation mat and all interior and exterior walls and slabs from the top of the foundation mat to and

including the entire floor slab at elevation 467'-0".

- (b) The centrifuge area and the demineralizers area.
- (c) The control room tower which consists of the entire vertical portion of the building encompassing the area of the main control room.

The entire structure is dynamically analyzed to Seismic Category I criteria, except that the Seismic Category II portions are designed to high allowable stresses permitted by the Uniform Building Code.

4.4.4.5 Design Basis Accident

The liquid radwaste tanks are constructed to specified American Petroleum Institute (API) 620 and 650 construction codes and standards and to the Uniform Building Code seismic requirements. These tanks operate at atmospheric pressure and low temperatures. A positive action interlock system is provided to prevent inadvertent opening of a drain valve because of operator error. Accordingly, the possibility of a complete tank failure or drainage is considered small.

An unspecified event is postulated to cause the complete release of the average radioactivity inventory in the tank containing the largest quantities of significant radionuclides in the liquid radwaste system. The tank postulated to rupture, is one of the two decontamination solution concentrated waste tanks.

- (a) Frequency Classification

This accident is categorized as a limiting fault. The transient is classified as a nonlimiting event for both original and upgraded power conditions. Therefore, no further analysis has been performed.

- (b) Sequence of Events and System Operation
- (c) Sequence of Events

The sequence of events expected to occur is as follows:

	EVENT TIME MINUTES
Event begins - failure occurs	0
Area radiation alarms alert plant personnel	-1
Operator action begins	-10

- (d) Identification of Operator Actions

The operator would, upon receiving the alarms, alert personnel to evacuate affected areas of the

radwaste building and isolate the radwaste building ventilation system.

(e) System Operation

Failure of a concentrated waste tank does not require a shutdown nor does it impair a safe shutdown. The Failure, however, will lead to limited operation of the concentrated waste system using the remaining tank.

The liquid contents of this tank will also be contained by the building walls and an unlined, 18-inch, high concrete dike around the radwaste tank area. Floor drain sump pumps would receive a high water level alarm, activate automatically and remove the spilled liquid.

(f) Barrier Performance

This event does not involve any containment barrier integrity except the tank itself and the radwaste building. The dike and walls of the radwaste building surrounding the tanks are built to Seismic Category I criteria.

4.4.5 Davis Besse Unit 1 Radioactive Waste Management Systems

The Waste Management Systems of the Davis Besse Plant provide the means for controlled handling, storage and preparation for disposal of liquid, gaseous and solid wastes.

4.4.5.1 Liquid Waste Management System

The systems handling liquid wastes for the Davis Besse, Unit 1 Plant are designed such that the estimated releases of liquid effluents comply with the following requirements of 10CFR20 and 10CFR50:

- (a) The individual radionuclide concentrations in liquid effluents at the site boundary shall not exceed the limits for releases to unrestricted areas given in Appendix B of 10CFR20.
- (b) The releases of radioactivity from the station shall comply with the ALARA standard set forth in 10CFR50.

The major source of clean liquid waste is reactor coolant letdown resulting from boron dilution operations or from coolant expansion during reactor start-ups. Other sources include leakage, drainage, and relief flows from valves and equipment containing

reactor-grade liquid. The following information pertaining to quantities and concentrations was provided as part of the initial safety analysis or added as changes to the systems were made. Operating experience has demonstrated that the reactor coolant activity levels are much lower than the activity levels assumed in the radwaste system design.

The major sources of non-detergent liquid wastes are:

- Miscellaneous system leakage
- Drainage from area washdown
- Sluicing of demineralizer resins
- Sampling and laboratory operations
- Condensate polishing demineralizer backwash (if there is a significant primary-secondary leak)

Detergent liquid waste comes from the hot showers (used to decontaminate personnel), contaminated laundry liquor and drains in the laboratory. During station operation, dirty and aerated liquid wastes from various sources are collected in tanks as follows:

After any of the collection tanks has been filled and after recirculation of the waste has ensured thorough mixing, a sample is taken and analyzed. If this shows no significant contamination, the contents of the tank can be released from the station without further treatment. If processing is required, the contents of the collection tank are pumped to the Radwaste Demineralizer System where suspended impurities are filtered and soluble ionic impurities are removed by ion exchange. The Radwaste Demineralizer System uses either a contractor-supplied demineralization service or installed demineralizers. The waste then flows through the miscellaneous waste monitor tank filter to the miscellaneous waste monitor tank. When the tank is filled and after recirculation has ensured proper mixing, a sample is taken and analyzed. Based on the results of this sample, the waste is then either recycled through part or all of the system for additional processing, or diluted in a controlled manner and released from the station. For flexibility, bypasses are provided around all major components with the exception of the monitor tank. These bypasses are used if a particular piece of equipment is not available for service or if, based on the results of sampling, the operator decides it is not needed.

The gas inlets and/or vents on the miscellaneous waste drain, detergent waste drain, and miscellaneous waste evaporator storage tanks are controlled to reduce the turnover of gases in the tank's vapor spaces. As the evolution of gaseous iodine from liquid waste is an equilibrium process, this tends to keep iodine liberation

to a minimum. The miscellaneous waste monitor tank has a nitrogen feed arrangement that can be used as a purge to maintain a slight positive pressure.

The miscellaneous waste drain tank's nitrogen blanketing system has been disabled. This tank is equipped with a vacuum breaker to prevent the drawing of a vacuum when this tank is emptied.

All liquid waste being discharged to the environment passes two monitors (one required for normal system operation) that measure and record the radiation level and that control two downstream valves in the flow path. If either of these monitors detects excessive radiation, both downstream valves are closed and flow is terminated.

4.4.5.2 Gaseous Waste Management System

None of the components of the gaseous waste system is specifically designed to withstand a hydrogen explosion for the following reasons:

- (a) The condition under which a hydrogen explosion may take place will never arise. That condition is greater than 5 percent oxygen by volume in a hydrogen-rich atmosphere.
- (b) There is no source input to the waste gas surge tank which contains sufficient amounts of oxygen. The major waste sources originate in primary reactor coolant which has been calculated to contain much less than 1 percent oxygen (see FSAR Subsection 9.3.2).
- (c) An oxygen/hydrogen monitor is installed on the waste gas surge tank to provide an additional measure of safety against the unlikely event of oxygen/hydrogen buildup. The concentration of oxygen shall be limited to less than or equal to 2% by volume whenever the hydrogen concentration exceeds 4% by volume. An oxygen or hydrogen concentration above the specified limit will actuate a local and control room alarm.

Additional oxygen/hydrogen monitors are provided on each Clean Waste Receiver Tank.

- (d) Should oxygen somehow contaminate one of the input sources, the condition will be alarmed, and the operator will take steps to stop the introduction of oxygen.
- (e) The waste gas surge tank is always kept at a net positive pressure with a nitrogen blanketing system to preclude the leakage.

4.4.5.3 Solid Waste Management System

The waste solidification system originally installed by Protective Packaging, Inc. (PP) is no longer used to process and solidify radwaste. Likewise, the Chem-Nuclear Systems, Inc. (CNSI) solidification system subsequently used is also not being employed for solidification. The present practice is to eliminate the use of any solidification system in the interests of easier handling and lower occupational exposures. For similar reasons, evaporation of liquid radwaste is also no longer carried out. The following practices are being followed in handling the wastes described above:

- (a) Spent Bead Resins are transferred to a high-integrity container. This container consists of cross-linked polyethylene. The polyethylene container is contained within a shipping cask that is provided with dewatering capability. The resins are dewatered to the point where residual water is less than 1% of the container volume. When the container is full and dewatered, it is closed. The shipping cask is also closed and shipped to a burial site. Current regulations allow shipment of dewatered resins of high specific activity (greater than 1 uCi/cc) provided a high-integrity container is used. This process eliminates solidification costs.
- (b) Powdered Resins are transferred to a metal container as long as they have specific activities less than 1 uCi/cc. Otherwise these resins are shipped in a high integrity container, as described above. Resins shipped in metal containers are dewatered until residual water is less than 0.5% of the container volume.
- (c) Resin from Liquid Waste Treatment have specific activities less than 1 uCi/cc. These resins are produced by the on-site Demineralization Unit. This unit processes the liquid radwaste at a rate of 25 gal/min. The liquid effluent from the system is recirculated for four hours to produce a homogeneous fluid, from which a reliable representative sample is taken. When it is verified that the liquid is within permissible limits for release to the environment, it is released to Lake Erie. The spent resin from the demineralization unit are transferred to a High-Integrity Container (HIC) and handled as described in FSAR Section 11.5.3a.
- (d) Filter Cartridges are transferred to shielded containers for storage and placed in high integrity containers for shipment and burial.

- (e) Miscellaneous Paper, Clothing, and Other Compressible Wastes are transferred to the Waste Compactor, where the volume of the material is reduced by a factor of 5 to 10 depending on materials and the compactor used.

The Barrel Waste Compactor is designed to compress solid waste materials such as rags, paper and filters into a 55 gallon drum. A drum carrier is provided so the drum can be filled outside the compactor and rolled in for compaction. The compactor can only be operated with the door closed and the drum in position. The ram is driven down compacting the waste at a maximum force of nine tons, holding its position for approximately 30 seconds to allow the waste to fully yield under the load.

The box waste compactor is designed to compress solid waste materials such as rags, paper and filters. The empty box is placed into the compactor and waste is alternately added and compressed until the box is full. The ram compresses the waste at a maximum force of fifty tons.

The compactor has a metal cabinet with an access door that is coupled with an interlock switch that prevents operation of the unit if the door is not fully closed. The cabinet is designed for containment of the drum and structural support.

A HEPA (High Efficiency Particulate Air) filter and blower assembly provide a minimum filtration efficiency of 99.9% of 0.3 micron particles. This assembly prevents airborne particles from escaping to the compactor. The filter can be removed for compaction and disposal.

When a drum is full of compacted waste, a lid is clamped on and the drum is stored in the designated area until shipment to a burial site.

Laundry is cleaned and dried using the washers and dryers in the Wet Wash Facility which is located adjacent to the Auxiliary Building. Typical items dry cleaned are: shoe covers, gloves, coveralls, caps, hoods and lab coats.

4.4.5.4 Design Basis Accidents

Class 3 events are accidents of very low probability, but are postulated because the conservatively calculated potential offsite doses resulting from these accidents is significant. This will have a bearing on the design and performance of the station to ensure that fission product release to the station environment will not result in undue risk to the public health and safety.

These postulated accidents may require operation of engineered safety features. Potential offsite doses resulting from design basis accidents must be less than the guide values given in 10CFR100 (25 REM). The only Class 3 event affecting the radioactive waste systems is the waste gas decay tank rupture.

A waste gas tank is assumed to contain all noble gases in one reactor coolant volume at the end of the third cycle and the iodine from one reactor coolant volume after a DF of 10^5 for the degasifier. Operation with 1 percent defective fuel is assumed. In addition, the following assumptions are made:

- (a) The accident duration is two hours, that is, 99.9 percent of all airborne activity is released over two hours.
- (b) The Control Room Ventilation System is isolated upon receipt of a high radiation signal in the Auxiliary Building exhaust stack (station vent). Isolation request about 10 seconds (including 5 seconds for instrument response). The air delivery rate for the five seconds after the fan is shut down was calculated using the following assumptions:
 - One supply fan with its related return fan is operating.
 - The supply and return fans are de-energized simultaneously.

4.4.6 Summary and Conclusions

4.4.6.1 Summary

As a result of the review of Radwaste Management System for the three advance reactor designs and the five operating light water cooled reactors, it can be concluded that RG 1.143 Rev. 1 guides have generally been followed. The largest difference is in the design of structures housing the Radwaste Systems. In most instances, the buildings housing systems and components of pressurized [$>50.3\text{kPa}$ (7.3 psig)] gaseous waste and other components listed in Section 2.1.3 of RG 1.143, Rev. 1 and portions of buildings necessary to contain the postulated rupture of liquid waste systems have been located in buildings designed to Seismic Category I requirements (have used the SSE as the seismic design basis). They also have been identified as Non-Nuclear Safety. Regulatory Guide 1.143 Rev. 1 would advise these building structures be designed to OBE requirements.

Those portions of radwaste building structures housing

solid waste systems and components located above the spill or flood height of the liquid waste systems have in general been designed to conventional Uniform Building Code requirements and are defined as both non-seismic Category I and Non Nuclear Safety.

In general, no SAR criteria for both advanced and operational reactors for natural phenomena hazards other than seismic (wind, tornado, flood etc.) have been found.

In some instances, the RG 1.60 shaped seismic spectra as given in Section 5.1.1 of RG 1.143 Rev. 1, have not been used. Instead, site specific ground response spectra defined for the site have been used. It should also be noted that Section 5.2.1 of RG 1.143 Rev. 1 guidance recommends the use of the lower damping values given in RG 1.61 for the OBE. In the limit because of the increased seismic response resulting from reduced damping values as compared to the SSE and the lower allowable stresses permitted by RG 1.143 Rev. 1 for the OBE as compared to the SSE it is possible that the OBE would govern seismic design.

The significant radwaste system design criteria for eight nuclear power plants surveyed are summarized in Table 4.4-11.

4.4.6.2 Conclusions

While the general philosophy of Regulatory Guide 1.143 has been followed, in specific detailed implementation, it has in general not been followed. The OBE and seismic design requirements of Regulatory Guide 1.143 have been addressed by placing systems and components in pre-existing Seismic Category I facilities and using existing Seismic Category I design criteria. Therefore, direct application of Regulatory Guide 1.143 associated with the elimination of the OBE as a design requirement has in general not been done for seismic design. Therefore, changes in Regulatory Guide 1.143 will have little direct impact on the Advanced Reactors or commercial operating plants. However the data provided in Table 4.4-11 have been considered in developing the suggested changes.

**Table 4.4-1(a) -FSAR Design Criteria Code and Standards and Nuclear Design Criteria
Perry Unit #1**

Item	Nuclear Design Criteria	Class				Construction Codes	Comment
		Quality	Seismic	Tornado	Safety		
I. Building Structure Radwaste Building	NI	D	I ⁽¹⁾	I ⁽²⁾	3 ⁽²⁾	NI	
II. Systems ⁽⁴⁾⁽⁵⁾ A) Liquid	10 CFR 20	D	NNS	NNS	NNS	Table 4.4-1(b) is applicable to all Q.C. Systems and Components, SC for Quality Group D.	
B) Gaseous	10 CFR 20	D	II	NI	3	Table 4.4-1(c) is applicable for Electrical SC in Gaseous Waste Systems.	
C) Solid	10 CFR 20	D	NNS	NNS	NNS		
III. Distribution Systems A) Piping 1. Process	NI	NI	NI	NI	NI	NI	
2. Fire	NI	NI	NI	NI	NI	NI	
B) Piping/Tubing (Inst.)	NI	NI	NI	NI	NI	NI	
C) Cable Trays	NI	NI	NI	NI	NI	NI	
E) Ventilation	NI	NI	NI	NI	NI	NI	
IV. Components A) Pumps	NI	NI	NI	NI	NI	NI	
B) Tanks & Vessels	NI	NI	NI	NI	NI	NI	
C) Valves	NI	NI	NI	NI	NI	NI	
D) Heat Exchanger	NI	NI	NI	NI	NI	NI	
E) Switch Gear	NI	NI	NI	NI	NI	NI	
F) MCC	NI	NI	NI	NI	NI	NI	
G) Fans, Dampers	NI	NI	NI	NI	NI	NI	

Footnotes for Table 4.4-1(a):

- (1) Concrete portions of the Radwaste Building up to 40 ft. above grade is Seismic Cat. I. Above this elevation, the steel structure is NNS.
- (2) Table III for mechanical components and systems is applicable to Safety Class 3.
- (3) Both Concrete and Steel portions of the Building is Tornado Resistant Designed.
- (4) Systems and Components in the liquid and solid radwaste systems are constructed to Quality Group D Criteria as identified in Regulatory Guide 1.26 as indicated in Tables 4.4.1(b). They also meet the requirements of Table 1 of Regulatory Guide 1.143. The liquid and solid radwaste systems are housed in a Seismic Category I structure. Radionuclide concentrations from the liquid radwaste system are prevented from exceeding the limits of 10 CFR 20 at the nearest potable water supply. Therefore, the components in the liquid and solid radwaste systems are classified Non-Nuclear Safety Class (NNS) and require no seismic design. The quality assurance requirements of Regulatory Guide 1.143 are not applicable for these systems and components.
- (5) The Systems and Components of the Gaseous Radwaste System are also classified as Quality Group D, Safety Class 3 and are designed to meet OBE seismic loads as defined in RG1.143.

TABLE 4.4-1(b) Quality Groups and Applicable Construction Standards for Perry Unit 1 Plant				
Components	Group A	Group B	Group C	Group D
Pressure Vessels	ASME; Section III; Class I	ASME; Section III; Class 2	ASME; Section III; Class 3	ASME; Section VIII; Division 1
Piping	ASME; Section III; Class I	ASME; Section III; Class 2	ASME; Section III; Class 3	ANSI; B31.1.0 Power Piping
Pumps	ASME; Section III; Class I	ASME; Section III; Class 2	ASME; Section III; Class 3	ASME; Section VIII; Division 1; (See Note 1)
Valves	ASME; Section III; Class I	ASME; Section III; Class 2	ASME; Section III; Class 3	ANSI B31.1
Atm. Storage Tanks	Not Applicable	ASME; Section III; Class 2	ASME; Section III; Class 3	API-650, AWWA D 100 or ANSI B96.1 or equivalent
0-15 psig Storage Tanks	Not Applicable	ASME; Section III; Class 2	ASME; Section III; Class 3	API 620 or equivalent
Heat Exchangers	ASME; Section III; Class 1 and TEMA-C	ASME; Section III; Class 2 and TEMA-C2	ASME; Section III; Class 3 and TEMA-C	ASME; Section VIII, Division 1 and TEMA-C

Footnotes for Table 4.4-1(b):

1. For pumps classified Group D and operating above 150 psi or 212°F ASME, Section VIII, Div. 1 is used as a guide in calculating the wall thickness for pressure retaining parts and in sizing the cover bolting. For pumps operating below 150 psi and 212°F, manufacturers' standard pump for the intended service is used.

TABLE 4.4-1.(c) Applicable Construction Standards for Electrical Systems and Components for Perry Unit #1 Plant												
PROTECTION			CLASS 1E									
Components	Modules	Sensors	Systems ⁽¹⁾	Cables	Connectors	Switch Gear	Trans-formers	Diesel	Systems	Motors	Valve Actuators	Valve Penetra-tions
IEEE-323	IEEE-323 IEEE-344	IEEE-323 IEEE-344	IEEE-279	NDR	NDR	IEEE-344	IEEE-344	IEEE-387	IEEE-308	IEEE-323 IEEE-334 IEEE-344	IEEE-323 IEEE-344	NDR

Footnotes for Table 4.4-1(c):

1. IEEE-279 shall apply only to those Safety Class 2 or 3 systems and components which actuate reactor trip or, in the event of a serious reactor accident, actuate engineered safety features.

NDR - Design requirements have not been developed for this condition of design by the applicable code.

FSAR, SSAR OR SER DESIGN CRITERIA Table 4.4-2 Applied Loads Plant Perry Unit #1										
Item	Operating Conditions	Extreme Loads						Accident Loads		
		Seismic	Tornado	Wind	External Flood	Accidental Explosion	Malevolent Vehicle	Other		
I. Building Structure A) Radwaste Bldg.	NI	I ⁽¹⁾	I ⁽³⁾	I ⁽⁴⁾	I ⁽⁵⁾	NA ⁽⁶⁾	NI	NI	NI	
II. Systems A) Liquid	NI	NNS	NNS	NNS	NNS	NNS	NNS	NI	NI	
B) Gaseous	NI	II ⁽²⁾	NA	NA	NA	NA	NA	NI	NI	
C) Solid	NI	NNS	NNS	NNS	NNS	NNS	NNS	NI	NI	
III. Distribution Systems A) Piping 1. Process	NI	NI	NI	NI	NI	NI	NI	NI	NI	
2. Fire	NI	NI	NI	NI	NI	NI	NI	NI	NI	
3. Inst. & Cont.	NI	NI	NI	NI	NI	NI	NI	NI	NI	
B) Cable Trays	NI	NI	NI	NI	NI	NI	NI	NI	NI	
C) Ventilation	NI	NI	NI	NI	NI	NI	NI	NI	NI	
IV. Components A) Pumps	NI	NI	NI	NI	NI	NI	NI	NI	NI	
B) Tanks & Vessels	NI	NI	NI	NI	NI	NI	NI	NI	NI	
C) Valves	NI	NI	NI	NI	NI	NI	NI	NI	NI	
D) Heat Exchanger	NI	NI	NI	NI	NI	NI	NI	NI	NI	
E) Switch Gear	NI	NI	NI	NI	NI	NI	NI	NI	NI	
F) MCC	NI	NI	NI	NI	NI	NI	NI	NI	NI	

FSAR, SSAR OR SER DESIGN CRITERIA									
Table 4.4-2 Applied Loads Plant Perry Unit #1 (Continued)									
Item	Operating Conditions	Extreme Loads						Accident Loads	
		Seismic	Tornado	Wind	External Flood	Accidental Explosion	Malevolent Vehicle		
G) Fans, Dampers	NI	NI	NI	NI	NI	NI	NI	NI	NI

Footnotes for Table 4.4-2:

- (1) The SSE seismic loads are applicable to Seismic Category I PGA=0.15g.
- (2) The OBE seismic loads are applicable to Seismic Category II PGA=0.075g.
- (3) Both the concrete and NNS steel portion of the Radwaste Building are tornado resistant designed.
- (4) Design basis wind is 90 mph.
- (5) Grade of structure is located 45 ft. above highest lake level and 12 ft above maximum storm surge.
- (6) Sources of accidental explosives are far enough away so as not to pose a significant threat [maximum pressure load = 1.52kPa (.22 psi)] which is at the threshold of glass breaking and should cause no failure of standard doors or partitions.

NA = Not Applicable; NI = No Information; NNS = Non Nuclear Safety

Table 4.4-3(a) FSAR Design Criteria Code and Standards and Nuclear Design Criteria Plant Waterford							
Item	Nuclear Design Criteria	Class				Construction Codes	
		Quality	Seismic	Tornado	Safety	System	Component
I. Building Structure A) Auxiliary Bldg.	SRP Section 3.8.4	NA ⁽¹⁾	I	I	NA ⁽¹⁾	NI	ASCE Paper 3269 applicable to Wind and Tornado Design ACI-318-63 AISC-7th Edition
II. Systems A) Liquid	10CFR50 - Appendix I	D	NNS ⁽²⁾	NNS ⁽²⁾	NNS	NI	Seismic Category I applied only to: Gas Surge Tank, Waste Gas Tank, Gas Decay Tank
B) Gaseous	10CFR20 - All	C	I	I ⁽²⁾	3	NI	
C) Solid	NI	D	NNS ⁽²⁾	NNS ⁽²⁾	NNS	NI	
III. Distribution Systems A) Piping 1. Process	NI	NI	NI	NI	NI	NI	NA
2. Fire	NI	NI	NI	NI	NI	NI	NA
B) Piping/Tubing (Inst.)	NI	NI	NI	NI	NI	NI	NA
C) Cable Trays	NI	NI	NI	NI	NI	NI	NA
D) Ventilation	NI	NI	NI	NI	NI	NI	NI

Table 4.4-3(a) FSAR Design Criteria Code and Standards and Nuclear Design Criteria Plant Waterford (Cont.)								
Item	Nuclear Design Criteria	Class				Construction Codes		
		Quality	Seismic	Tornado	Safety	System	Component	
IV. Components								
A) Pumps	NI	NI	NI	NI	NI	NI	NI	
B) Tanks & Vessels	NI	NI	NI	NI	NI	NI	NI	
C) Valves	NI	NI	NI	NI	NI	NI	NI	
D) Heat Exchanger	NI	NI	NI	NI	NI	NI	NI	
E) Switch Gear	NI	NI	NI	NI	NI	NI	NI	
F) MCC	NI	NI	NI	NI	NI	NI	NI	
G) Fans, Dampers	NI	NI	NI	NI	NI	NI	NI	
V. Misc Items								
A) Compression; Waste Gas	NI	C	I	I ⁽²⁾	3	NI	ASME BPVC Section III-1971	
B) Ion Exchangers	NI	NNS	NNS	NNS	NNS	NI	ASME BPVC- Section III-1968/Sum 1970	

Footnotes for a Table 4.4-3(a):

- (1) Safety and Quality Classifications are applicable only to systems and components and not to structures.
- (2) Located in a building which is Seismic Category I and tornado protected.
- (3) Except for piping which is not isolated from Seismic Category I and Safety Class 3 components.
- (4) Specific Component Design Criteria can be found in Tables 4.4-3.(b) to 4.4-3.(x).

NI = No Information NNS = Non Nuclear Safety

Table 4.4-3(b) Waterford WMS Equipment Description-Liquid Waste System				
TANKS	WASTE	WASTE CONDENSATE	LAUNDRY	WASTE STORAGE
Quantity	2	2	2	1
Type (cylindrical)	Horizontal	Horizontal	Horizontal	Vertical
Internal volume (useful), m ³	13.63m ³ (3600 gal.)	56.78m ³ (15000 gal.)	13.63m ³ (3600 gal.)	143.85m ³ (38000 gal.)
Design Pressure	ATM	ATM	ATM	ATM
Normal Operating Pressure	ATM	ATM	ATM	ATM
Design Temperature, K	366 K (200 F)	394 K (250 F)	366 K (200 F)	322 K (120 F)
Normal Operating Temperature, K	322 K (120 F)	322 K (120 F)	322 K (120 F)	322 K (120 F)
Code	None ¹	None ¹	None ¹	ASME VIII. 1983 Edition-Winter Addenda
Material	SA-240	SA-240	SA-240	SA-240
	Type 304	Type 304	Type 304	Type 304

Footnotes for Table 4.4-3(b):

1. The tank is designed and fabricated to ASME VIII Division I but is not stamped.

Table 4.4-3(c) Waterford WMS Equipment Description-Liquid Waste System Laundry		
FILTERS WASTE		
Quantity	1	1
Type Element	Resin Bonded glass fiber and polyester	fiber and polyester
Particle Retention	20 micron absolute	20 micron absolute
Design Pressure, kPa	1034 kPa (150 psig)	1034 kPa (150 psig)
Normal Operating Pressure, kPa	414 kPa (60 psig)	414 kPa (60 psig)
Design Temperature, K	366 K (200 F)	366 K (200 F)
Normal Operating Temperature, K	322 K (120 F)	322 K (120 F)
Design Flow, m ³ /s	0.00315 m ³ /s (50 gpm)	0.00315 m ³ /s (50 gpm)
Normal Operating Flow, m ³ /s	0.00126 m ³ /s (20 gpm)	0.00315 m ³ /s (50 gpm)
Code for Vessel	ASME VIII 1968	ASME VIII. 1968
Material	Edition - Winter 1970 Addenda Type 316 Stainless Steel	Edition - Winter 1970 Addenda Type 316 Stainless Steel

Table 4.4-3(d) Waterford WMS Equipment Description - Liquid Waste System Oil Removal Filter	
Quantity	1
Type	Pleated Paper Cartridge
Design Pressure, kPa	1034 kPa (150 psig)
Normal Operating Pressure, kPa	414 kPa (60 psig)
Design Temperature, K	366 K (200 F)
Normal Operating Temperature, K	322 K (120 F)
Design Flow, m ³ /s	0.00315 m ³ /s (50 gpm)
Normal Operating Flow, m ³ /s	0.00126 m ³ /s (20 gpm)
Code for Vessel	ASME VIII. 1968 Edition Winter 1970 Addenda
Material	Austenitic Stainless Steel

**Table 4.4-3(e) Waterford WMS Equipment Description-Liquid Waste System
Waste Condensate Ion Exchanger**

Quantity	1
Type	Flushable deep resin bed
Design Pressure, kPa	1034 kPa (150 psig)
Normal Operating Pressure, kPa	414 kPa (60 psig)
Design Temperature, K	394 K (250 F)
Normal Operating Temperature, K	322 K (120 F)
Design Flow, m ³ /s	0.00315 m ³ /s (50 gpm)
Normal Operating Flow, m ³ /s	0.00126 m ³ /s (20 gpm)
Resin Volume (total), m ³	1.019 m ³ (36 ft ³)
Resin Volume (useful), m ³	0.906 m ³ (32 ft ³)
Code for vessel	ASME VIII. 1968 Edition, Summer 1970 Addenda
Material	Austenitic Stainless Steel
Resin	H-OH form

Table 4.4-3(f) Waterford WMS Equipment Description - Liquid Waste System Portable Waste Demineralizer	
Quantity	1
Design DF	10 ⁴
Design Pressure, kPa	1034 kPa (150 psig)
Design Temperature, K	355 K (180 F)
Design Flow, m ³ /s	0.00315 m ³ /s (50 gpm)
Normal Operating Flow, m ³ /s	0.00114 m ³ /s (18 gpm)
Electric Load	480V 30A 120V 30A 120V 15A
Material	Austenitic Stainless Steel
Code	ASME VIII "U" Stamp and ANSI B31.1. 1967 Edition

Table 4.4-3(g) Waterford WMS Equipment Description				
Pumps	Waste	Waste Condensate	Laundry	Waste Storage
Quantity	2	2	2	
Type	Centrifugal	Centrifugal	Centrifugal	Centrifugal
Design Pressure, kPa	1034 kPa (150 psig)	1034 kPa (150 psig)	1034 kPa (150 psig)	1034 kPa (150 psig)
Design Temperature, K	366 K (200 F)	366 K (200 F)	366 K (200 F)	366 K (200 F)
Normal Operating Temperature, K	322 K (120 F)	322 K (120 F)	322 K (120 F)	322 K (120 F)
Capacity, Rated, m ³ /s	0.00315 m ³ /s (50 gpm)	0.00315 m ³ /s (50 gpm)	0.00315 m ³ /s (50 gpm)	0.00379 m ³ /s (60 gpm)
Rated head, m	44.20 m (145 ft.)	44.20 m (145 ft.)	44.20 m (145 ft.)	38.10 m (125 ft.)
NPSH, Available, m	4.72 m (15.5 ft.)	7.13 m (23.4 ft.)	7.32 m (24 ft.)	1.52 m (5 ft.)
Motor Horsepower, W	5593 W (7.5 hp)	5593 W (7.5 hp)	5593 W (7.5 hp)	5593 W (7.5 hp)
Wetted Materials	Stainless Steel	Stainless Steel	Stainless Steel	Stainless Steel

Table 4.4-3(h) Waterford WMS Equipment Description - Liquid Waste System Reactor Drain Tank	
Quantity	1
Type	Horizontal, cylindrical
Internal Volume (useful), m ³	6.06 m ³ (1600)
Design Pressure, kPa	172 kPa (25 psig)Internal/ 103 kPa (15 psig)External
Design Temperature, K	394 K (250 F)
Normal Operating Pressure, kPa	13.8 kPa (2.0 psig)
Normal Operating Temperature, K	322 K (120 F)
Code	ASME VIII, Div. 1 1968 Edition, Summer 1970 Addenda
Material	SA-240, Type 304

Table 4.4-3(i) Waterford WMS Equipment Description - Liquid Waste System Equipment Drain Tank	
Quantity	1
Type	Horizontal, cylindrical
Internal Volume (useful), m ³	15.14m ³ (4,000 gal.)
Design Pressure, kPa	68.9 kPa (10 psig)
Design Temperature, K	366 K (200 F)
Normal Operating Pressure, kPa	20.7 kPa (3.0 psig)
Normal Operating Temperature, K	322 K (120 F)
Code	ASME VIII, Div. 1 1968 Edition, Summer 1970 Addenda
Material	SA-240, Type 304

Table 4.4-3(j) Waterford BMS Equipment Description - Liquid Waste System Equipment Flash Tank	
Quantity	1
Type	Vertical, cylindrical
Internal Volume (useful), m ³	1.51m ³ (400 gal.)
Design Pressure, kPa	517 kPa (75 psig)
Design Temperature, K	394 K (250 F)
Normal Operating Pressure, kPa	3.45 kPa (0.5 psig)
Normal Operating Temperature, K	322 K (120 F)
Code	ASME VIII, Class 3, 1968 Edition, Summer 1970 Addenda
Material	SA-240, Type 304

Table 4.4-3(k) Waterford BMS Equipment Description - Liquid Waste System Equipment Holdup Tank	
Quantity	41
Type	Vertical, cylindrical
Internal Volume (useful), m ³	181.32 m ³ (47,900 gal.)
Design Pressure, kPa	68.9 kPa (10 psig) Internal/13.8 kPa (2 psig) External
Design Temperature, K	389 K (240 F)
Normal Operating Pressure, kPa	3.45 kPa (0.5 psig) to 10.34 kPa (1.5 psig)
Normal Operating Temperature, K	322 K (120 F)
Code	ASME III, Class 3, 1968 Edition, Summer 1970 Addenda
Material	SA-240, Type 304

Table 4.4-3(l) Waterford BMS Equipment Description - Liquid Waste System Boric Acid Condensate Tanks	
Quantity	4
Type	Vertical, cylindrical
Internal Volume (useful), m ³	65.11 m ³ (17,200 gal.)
Design Pressure, kPa	Atmospheric
Design Temperature, K	394 K (250 F)
Normal Operating Pressure, kPa	Atmospheric
Normal Operating Temperature, K	322 K (120 F)
Code	ASME VIII, Div. 1, 1968 Edition, Summer 1970 Addenda
Material	SA-240, Type 304

Table 4.4-3(m) Waterford BMS Equipment Description - Liquid Waste System Reactor Drain Tank and Equipment Drain Tank Pumps	
Quantity	1 (each)
Type	Centrifugal
Design Pressure, kPa	1034 kPa (150 psig)
Design Temperature, K	366 K (200 F)
Normal Operating Temperature, K	322 K (120 F)
Capacity, rate, m ³ /s	0.00315 m ³ /s (50 gpm)
Rated Head, m	44.20 m (145 ft.)
NPSH Available, m	13.01 m (42.7 ft.)
Motor Horsepower, W	5593 W (7.5 hp)
Wetted Materials	Austenitic Stainless Steel
Code	ASME III, Class 3, 1971 Edition, Summer 1972 Addenda

Table 4.4-3(n) Waterford BMS Equipment Description - Liquid Waste System Flash Tank Pumps	
Quantity	2
Type	Centrifugal
Design Pressure, kPa	1034 kPa (150 psig)
Design Temperature, K	366 K (200 F)
Normal Operating Temperature, K	322 K (120 F)
Capacity, rate, m ³ /s	0.00946 m ³ /s (150 gpm)
Rated Head, m	18.29 m (60 ft.)
NPSH Available, m	7.35 m (24.1 ft.)
Motor Horsepower, W	5593 W (7.5 hp)
Wetted Materials	Austenitic Stainless Steel
Code	ASME III, Class 3, 1971 Edition, Summer 1972 Addenda

Table 4.4-3(o) Waterford BMS Equipment Description - Liquid Waste System Holdup Drain and Recirculation Pumps	
Quantity	1 drain, 1 recirc/drain 1 recirc.
Type	Centrifugal
Design Pressure, kPa	1034 kPa (150 psig)
Design Temperature, K	366 K (200 F)
Normal Operating Temperature, K	322 K (120 F)
Capacity, rate, m ³ /s	0.00315 m ³ /s (50 gpm)
Rated Head, m	44.20 m (145 ft.)
NPSH Available, m	11.06 m (36.3 ft.)(drain), 4.11 m (13.5 ft.)(recirc)
Motor Horsepower, W	5593 W (7.5 hp)
Wetted Materials	Austenitic Stainless Steel
Code	None

Table 4.4-3(p) Waterford BMS Equipment Description - Liquid Waste System Boric Acid Condensate Pumps	
Quantity	2
Type	Centrifugal
Design Pressure, kPa	1034 kPa (150 psig)
Design Temperature, K	366 K (200 F)
Normal Operating Temperature, K	322 K (120 F)
Capacity, rate, m ³ /s	0.00315 m ³ /s (50 gpm)
Rated Head, m	44.20 m (145 ft.)
NPSH Available, m	7.62 m (25 ft.)
Motor Horsepower, W	5593 W (7.5 hp)
Wetted Materials	Austenitic Stainless Steel
Code	None

Table 4.4-3(q) Waterford BMS Equipment Description - Liquid Waste System Pre-Concentrator Ion Exchanger	
Quantity	2
Type	Flushable
Design Pressure, kPa	1034 kPa (150 psig)
Design Temperature, K	394 K (250 F)
Normal Operating Pressure, kPa	414 kPa (60 psig)
Normal Operating Temperature, K	322 K (120 F)
Resin Volume (total), m ³	1.019 m ³ (36 ft ³)
Resin Volume (useful) required, m ³	0.963 m ³ (34 ft ³)
Design Flow, m ³ /s	0.00631 m ³ /s (100 gpm)
Normal Flow, m ³ /s	0.00126 m ³ /s (20 gpm)
Code for Vessel	ASME VIII, Div. 1, 1968 Edition, Summer 1970 Addenda
Material	Austenitic Stainless Steel
Resin	Mixed Bed

**Table 4.4-3(r) Waterford BMS Equipment Description - Liquid Waste System
Boric Acid Condensate Ion Exchangers**

Quantity	2
Type	Flushable
Design Pressure, kPa	1034 kPa (150 psig)
Design Temperature, K	394 K (250 F)
Normal Operating Pressure, kPa	414 kPa (60 psig)
Normal Operating Temperature, K	333 K (140 F)
Resin Volume (total), m ³	1.019 m ³ (36 ft ³)
Resin Volume (useful) required, m ³	0.963 m ³ (34 ft ³)
Design Flow, m ³ /s	0.00315 m ³ /s (50 gpm)
Normal Flow, m ³ /s	0.00126 m ³ /s (20 gpm)
Code for Vessel	ASME VIII, Div. 1, 1968 Edition, Summer 1970 Addenda
Material	Austenitic Stainless Steel
Resin	Anion Bed

Table 4.4-3(s) Waterford BMS Equipment Description - Liquid Waste System Boric Acid Concentrator and Boric Acid Waste Concentrator	
Quantity	1 of each
Design DF (bottoms/distillate)	10 ⁴ (minimum)
Design Pressure (process), kPa	207 kPa (30 psig)
Design Temperature (process), K	408 K (275 F)
Normal Operating Pressure (process), kPa	41 kPa (6 psig)
Normal Operating Temperature (process), K	275 K (36 F)
Resin Volume (useful) required, m ³	1.98 m ³ (70 ft. ³) - 6.51 m ³ (230 ft. ³)
Design Flow, (process), m ³ /s	0.00315 m ³ /s (50 gpm)
Normal Operating Flow, m ³ /s	0.00126 m ³ /s (20 gpm)
Steam Required at 50 psig, kg/s	1.6884 kg/s (13,400 lb/hr)
Cooling Water Flow, m ³ /s	0.03312 m ³ /s (525 gpm)
Code for Vessel	ASME VIII, Div. 1, 1971 Edition, Winter 1971 Addenda/B31.1, 1967 Edition
Material	Type 304SS
Electrical Load	4880V 30A /30 120V 30A 120V 15A

Table 4.4-3(t) Waterford BMS Equipment Description - Liquid Waste System Pre-Concentrator Filters	
Quantity	2
Type Element	Resin bonded glass fiber and polyester
Removal Rating, Absolute (Beta Method)	20 microns
Normal Operating Pressure, kPa	517 kPa (75 psig)
Design Pressure, kPa	1034 kPa (150 psig)
Design Temperature, K	389 K (240 F)
Normal Operating Temperature, K	339 K (150 F)
Design Flow, m ³ /s	0.00631 m ³ /s (100 gpm)
Normal Flow, m ³ /s	0.00126 m ³ /s (20 gpm)
Code for Vessel	ASME VIII, Div. 1, 1968 Edition, Winter 1970 Addenda
Material	Austenitic Stainless Steel

Table 4.4-3(u) Waterford GWMS Equipment Description - Gaseous Waste System (Tanks)		
Tanks	Gas Decay Tanks	Gas Surge Tank
Quantity	3	1
Type (cylindrical)	vertical	vertical
Internal Volume, m ³	16.99 m ³ (600 ft. ³) each	0.57 m ³ (20 ft. ³)
Design Pressure, kPa	2620 kPa (380 psig)	414 kPa (60 psig)
Normal Operating Pressure, kPa	0-2379 kPa (0-345 psig)	34.5 kPa (5 psig)
Design Temperature, K	366 K (200 F)	366 K (200 F)
Normal Operating Temperature, K	316 K (110 F)	311 K (100 F)
Code	ASME VIII. Division 1 (Built to ASME III-3. 1971 Edition. Summer 1972 Addenda)	ASME VIII. Division 1 (Built to ASME III-3. 1971 Edition. Summer 1972 Addenda)
Material	Carbon Steel	Carbon Steel

Table 4.4-3(v) Waterford GWMS Equipment Description - Gaseous Waste System (Waste Gas Compressor And Cooler)	
Quantity	2
Type	Positive displacement
Design Inlet Temperature, K	339 K (150 F)
Design Inlet Pressure, kPa	1034 kPa (150 psig)
Design Discharge Pressure, kPa	2758 kPa (400 psig)
Maximum Operating Discharge Pressure, kPa	2379 kPa (345 psig)
Maximum After Cooler Outlet Temperature, K	316 K (110 F)
Rated Flow SCEM	2
Code	Manufacturer Standards (Built to ASME III-3, 1971 Edition, Winter 1971 Addenda)
Motor Horsepower, W	5593 W (7.5 hp)
Wetted Materials	Austenitic Stainless Steel
Code	ASME III, Class 3, 1971 Edition, Summer 1972 Addenda

Table 4.4-3(w) Waterford GWMS Equipment Description - Gaseous Waste System (Gas Analyzer)	
Quantity	1
Type	Automatic Hydrogen/oxygen analyzer
Measurement Range: hydrogen oxygen	0-5%. 0-50%. 0-100%. by Volume 0-5%. 0-10%. 0-25%. by Volume
Sample Inlet Pressure, kPa	6.89 kPa (1 psig) to 345 kPa (50 psig)
Sample Discharge Pressure, kPa	34.47 kPa (5psig) - 103 kPa (15 psig)
Code	none

Table 4.4-3(x) Waterford Solid Waste Management System Component Summary Data								
TANKS	QUANTITY	CODE	DESIGN CAPACITY	DESIGN PRESSURE, kPa	DESIGN TEMP., K	NORMAL OPERATING PRESSURE, kPa	NORMAL OPERATING TEMP., K	MATERIAL
Spent Resin Tank	1	ASME Section VIII Div. 1 (1973)	12.11 m ³ (3200 gal.)	ATM	366 K (200 F)	ATM	311 K (100 F)	SA 240 Type 304
Dewatering Tank(1)	1	ASME Section VIII Div. 1 (1973)	2.83 m ³ (100 ft ³)	ATM	339 K (150 F)	ATM	333 K (140 F)	ASTM A-240
Concentrate Storage Tank	1	ASME Section VIII Div. 1 (1973)	11.36 m ³ (3000 gal.)	ATM	339 K (150 F)	ATM	333 K (140 F)	ASTM A-240
Cement Silo(1)	1	None	14.16 m ³ (500 ft ³)	ATM	333 K (140 F)	ATM	225-322 K (0-120 F)	ASTM A-36
Sodium Silicate Tank	1	None	5.66 m ³ (200 ft ³)	ATM	333 K (140 F)	ATM	225-322 K (0-120 F)	ASTM A-36

Table 4.4-3(x) Waterford Solid Waste Management System Component Summary Data										
PUMPS	QUANTITY	STANDARDS	TYPE	DESIGN PRESSURE, kPa	DESIGN TEMP., K	DESIGN FLOW, m ³ /s	MATERIAL	SEAL TYPE	MOTOR HP, W	MOTOR VOLTAGE PHASE/HZ
Concentrate Storage Tank Metering Pump	1	Manufacturer's Standard	Progressing Cavity	689 kPa (100 psig)	355 K (180 F)	4.73 x 10 ⁻⁴ m ³ /s (7.5 gpm)	SS 316	Mech.	1491 W (2 hp)	460/3/60
Dewatering Tank Metering Pump ⁽¹⁾	1	Manufacturer's Standard	Progressing Cavity	689 kPa (100 psig)	355 K (180 F)	4.73 x 10 ⁻⁴ m ³ /s (7.5 gpm)	SS 316	Mech.	1491 W (2 hp)	460/3/60
Process Mixing Pump ⁽¹⁾	1	Manufacturer's Standard	Progressing Cavity	689 kPa (100 psig)	355 K (180 F)	6.31 x 10 ⁻⁴ m ³ /s (10 gpm)	SS 316	Mech.	1491 W (2 hp)	460/3/60
Sodium Silicate Pump ⁽¹⁾	1	Manufacturer's Standard	Progressing Cavity	689 kPa (100 psig)	355 K (180 F)	4.73 x 10 ⁻⁵ m ³ /s (0.75 gpm)	C1-Body Buna N-Stator	Mech.	746 W (1 hp)	460/3/60
Spent Resin Transfer Pump	1	Manufacturer's Standard	Progressing Cavity	689 kPa (100 psig)	355 K (180 F)	1.45 x 10 ⁻² m ³ /s (230 gpm)	SS 316	Mech.	14914 W (20 hp)	460/3/60
Spent Resin Dewatering Pump	1	Manufacturer's Standard	Horizontal Centrifugal	689 kPa (100 psig)	355 K (180 F)	3.15 x 10 ⁻³ m ³ /s (50 gpm)	SS 316	Mech.	3729 W (5 hp)	460/3/60

Footnotes for Table 4.4-3(x):

1. This equipment is not being used and has been abandoned in place.

Table 4.4-4 - FSAR DESIGN CRITERIA Applied Loads Plant Waterford										
Item	Operating Conditions	Extreme Loads						Accident Loads		
		Seismic	Tornado	Wind	External Flood	Accidental Explosion	Malevolent Vehicle	Other		
I. Building Structure A) Auxiliary Bldg.	NI	I	I	I ⁽¹⁾	I ⁽²⁾	NI	NI	NI	NI	NI
II. Systems A) Liquid	NI	NI	NI	NI	NI	NI	NI	NI	NI	NI
B) Gaseous	NI	NI	NI	NI	NI	NI	NI	NI	NI	NI
C) Solid	NI	NI	NI	NI	NI	NI	NI	NI	NI	NI
III. Distribution Systems A) Piping 1. Process	NI	NI	NI	NI	NI	NI	NI	NI	NI	NI
2. Fire	NI	NI	NI	NI	NI	NI	NI	NI	NI	NI
3. Inst. & Cont.	NI	NI	NI	NI	NI	NI	NI	NI	NI	NI
B) Cable Trays	NI	NI	NI	NI	NI	NI	NI	NI	NI	NI
C) Ventilation	NI	NI	NI	NI	NI	NI	NI	NI	NI	NI
IV. Components A) Pumps	NI	NI	NI	NI	NI	NI	NI	NI	NI	NI
B) Tanks & Vessels	NI	NI	NI	NI	NI	NI	NI	NI	NI	NI
C) Valves	NI	NI	NI	NI	NI	NI	NI	NI	NI	NI
D) Heat Exchanger	NI	NI	NI	NI	NI	NI	NI	NI	NI	NI
E) Switch Gear	NI	NI	NI	NI	NI	NI	NI	NI	NI	NI
F) MCC	NI	NI	NI	NI	NI	NI	NI	NI	NI	NI

Table 4.4-4 - FSAR DESIGN CRITERIA Applied Loads Plant Waterford (Continued)									
Item	Operating Conditions	Extreme Loads						Accident Loads	
		Seismic	Tornado	Wind	External Flood	Accidental Explosion	Malevolent Vehicle	Other	
G) Fans, Dampers	NI	NI	NI	NI	NI	NI	NI	NI	NI

Footnotes for Table 4.4-4:

- (1) Wind design is for 89.4 m/s (200 mph) in accordance with ASCE Paper 3269.
- (2) The building structure is flood protected for a height of 19.8m (65 feet) above the basemat.

NI = No Information

Table 4.4-5 - FSAR DESIGN CRITERIA Code and Standards and Nuclear Design Criteria Beaver Valley Unit #2							
Item	Nuclear Design Criteria	Class				Construction Codes	
		Quality	Seismic	Tornado	Safety	System	Component
I. Building Structure							
A) Waste Handling	NI	I ⁽¹⁾	II ⁽³⁾	NNS	NNS	NI	ACI-318, AISC BLDG.
B) Auxiliary	NI	I ⁽¹⁾	I	I ⁽²⁾	I	NI	ACI-318, AISC BLDG.
C) Turbine Hall	NI	NNS	NNS	NNS	NNS	NI	NI
II. Systems							
A) Liquid ⁽⁵⁾	NI	II ⁽⁴⁾	NNS	NNS	NNS	RG 1.143 ANSI N199 NUREG-0017	NI
B) Gaseous	NI	II & III ⁽⁴⁾	II	NA	NNS		NI
C) Solid	NI	NNS	NNS	NNS	NNS		NI
III. Distribution Systems							
A) Piping	NI	NI	NI	NI	NI	USAS B31.1-67	NI
1. Process							
2. Fire	NI	NI	NI	NI	NI		NI
B) Piping/Tubing (Inst.)	NI	NI	NI	NI	NI		NI
C) Cable Trays	NI	NI	NI	NI	NI		NI
D) Ventilation							
A) Waste Handling	NI	III ⁽⁴⁾	NNS	NNS	NNS	NI	NI
B) Auxiliary	NI	I ⁽¹⁾	II ⁽³⁾	(2)	NNS	NI	NI

Table 4.4-5 - FSAR DESIGN CRITERIA Code and Standards and Nuclear Design Criteria Beaver Valley Unit #2 (Continued)							
Item	Nuclear Design Criteria	Class				Construction Codes	
		Quality	Seismic	Tornado	Safety	System	Component
IV. Components							
A) Pumps	NI	NI	NI	NI	NI	NI	
B) Tanks & Vessels	NI	NI	NI	NI	NI	BPVC VIII Div. I Certain Tanks & Vessels	
C) Valves	NI	NI	NI	NI	NI	NI	
D) Heat Exchanger	NI	NI	NI	NI	NI	NI	
E) Switch Gear	NI	NI	NI	NI	NI	NI	
F) MCC	NI	NI	NI	NI	NI	NI	
G) Fans, Dampers, Exhaust, Others	NI	NI	NI	NI	NI	SMACNA, AMCA	

Footnotes for Table 4.4-5:

- (1) This Beaver Valley Quality Designation is equivalent to RG 1.26 Quality Class A.
- (2) Tornado resistant design is provided below E1. 773.
- (3) Seismic Category II is equivalent to OBE seismic input with normal allowable acceptance criteria.
- (4) These Beaver Valley Quality designations are equivalent to RG 1.26 Quality Classes B and C respectively.
- (5) The steam generator flash tank, blowdown deminimizer and heat exchanger are located in the Turbine Hall.

Table 4.4-6 - FSAR DESIGN CRITERIA Applied Loads Beaver Valley Unit #2										
Item	Operating Conditions	Extreme Loads							Accident Loads	
		Seismic	Tornado	Wind	Flood-External	Accidental Explosion	Malevolent Vehicle	Other		
I. Building Structure										
A) Waste Handling	NI	II ⁽¹⁾	NNS	NNS	NNS ⁽³⁾	NA ⁽⁴⁾	NA ⁽⁴⁾	NI	NI	NI
B) Auxiliary	NI	I	I ⁽²⁾	I	I ⁽³⁾	NA ⁽⁴⁾	NA ⁽⁴⁾	NI	NI	NI
C) Turbine Hall	NI	NNS	NNS	NNS	NNS	NNS	NNS	NI	NI	NI
II. Systems ⁽⁵⁾										
A) Liquid	NI	NNS	NNS	NA	NA	NA ⁽⁴⁾	NA ⁽⁵⁾	NI	NA	NA
B) Gaseous	NI	II	NA	NA	NA	NA	NA	NI	Control of Hydrogen Release	NI
C) Solid	NI	NNS	NNS	NA	NA	NA	NA	NI	NA	NA
III. Distribution Systems										
A) Piping										
1. Process	NI	NI	NI	NI	NI	NI	NI	NI	NI	NI
2. Fire	NI	NI	NI	NI	NI	NI	NI	NI	NI	NI
3. Inst. & Cont.	NI	NI	NI	NI	NI	NI	NI	NI	NI	NI
B) Cable Trays	NI	NI	NI	NI	NI	NI	NI	NI	NI	NI
C) Ventilation	NI	NI	NI	NI	NI	NI	NI	NI	NI	NI
IV. Components										
A) Pumps	NI	NI	NI	NI	NI	NI	NI	NI	NI	NI
B) Tanks & Vessels	NI	NI	NI	NI	NI	NI	NI	NI	NI	NI
C) Valves	NI	NI	NI	NI	NI	NI	NI	NI	NI	NI

Table 4.4-6 - FSAR DESIGN CRITERIA Applied Loads Beaver Valley Unit #2 (Continued)									
Item	Operating Conditions	Extreme Loads						Accident Loads	
		Seismic	Tornado	Wind	Flood-External	Accidental Explosion	Malevolent Vehicle		
D) Heat Exchanger	NI	NI	NI	NI	NI	NI	NI	NI	NI
E) Switch Gear	NI	NI	NI	NI	NI	NI	NI	NI	NI
F) MCC	NI	NI	NI	NI	NI	NI	NI	NI	NI
G) Fans, Dampers	NI	NI	NI	NI	NI	NI	NI	NI	NI

Footnotes for Table 4.4-6:

- (1) Seismic levels applicable to seismic Category II are defined by the OBE.
- (2) Portions of the Auxiliary Building below El. 773 are tornado resistant designed in accordance with RG 1.76.
- (3) SSC of the plant are located above flood level.
- (4) Safety Class I structures are evaluated in compliance with RG 1.91.
- (5) Safety Class I structures are evaluated in compliance with 10CFR73.

Table 4.4-7(a) - FSAR Design Criteria Code and Standards and Nuclear Design Criteria WPPSS Unit #2							
Item	Nuclear Design Criteria	Class				Construction Codes	Comments
		Quality	Seismic	Tornado	Safety		
I. Building Structure ⁽¹⁾							
A) Radwaste Bldg.- Concrete	NI	B	I	I	III	ACI-318	
B) Radwaste Bldg.- Steel	NI	D	II	NNS	G	AISC - Bldg. Spec.	
II. Systems							
A) Liquid	Appendix I to 10CFR50 -1975 and 10CFR20 FSAR Sections: 3.1.2.6.3.1.3, 11	D ⁺	NNS	NNS	G	NI	
B) Gaseous		D ⁺	II	NA	G	NI	
C) Solid		D ⁺	NNS	NNS	G	NI	
III. Distribution Systems							
A) Piping	NI	NI	NI	NI	NI	B31.1.0-1967-Process	
1. Process							
2. Fire	NI	NI	NI	NI	NI	NI	
B) Piping/Tubing (Inst.)	NI	NI	NI	NI	NI	NI	
C) Cable Trays	NI	NI	NI	NI	NI	NI	
E) Ventilation	NI	NI	NI	NI	NI	SMACNA	

Table 4.4-7(a) - FSAR Design Criteria Code and Standards and Nuclear Design Criteria WPPSS Unit #2 (Continued)							
Item	Nuclear Design Criteria	Class				Construction Codes	Comments
		Quality	Seismic	Tornado	Safety		
IV. Components							
A) Pumps	NI	NI		NI	NI	For Oper. Pressure >1034kPa (150 psi) and 373 K (212°F) Section VIII will be used as guide to evaluations of press. retaining parts.	
B) Tanks & Vessels	NI	NI		NI	NI	ASME Section VIII, Div.1 API-620-Tanks API-650-Tanks AWWA-D100 Tanks ANSI B96.1	
C) Valves	NI	NI		NI	NI	B31.1.0 - 1967	
D) Heat Exchanger	NI	NI		NI	NI	ASME Section VIII, Div. 1 TEMAC	
E) Switch Gear	NI	NI		NI	NI	NI	
F) MCC	NI	NI		NI	NI	NI	
G) Fans, Dampers	NI	NI		NI	NI	SMACNA	

Table 4.4-7(b)¹ Radwaste and Control Building Quality Class and Design Bases Criteria WPSS Unit #2						
	Material of Construction	Quality Group	Designed to Withstand the Effects of The Following Postulated Environmental Disturbances			
			Seismic Category ³	Design Basis Wind	Design Basis Tornado	Tornado Generated Missiles
Those portions of the Radwaste and control building that house systems or components necessary for safe shutdown of the reactor.	Reinforced Concrete	D+	I	Yes	Yes	Yes
Those portions of the Radwaste building that house equipment containing significant quantities of radioactive material.	Reinforced Concrete	D+	I	Yes.	Yes.	Yes.
Other portions	Structural Steel	D	II	Yes	No. ²	No. ²

Footnotes for Table 4.4-7(b):

1. This Table applies to FSAR Section 3.8.4.
2. "Other Portions" denote those portions of the Radwaste and Control Building which are not required to withstand the effects of the design basis tornado and tornado generated missiles.
3. Safety Classification in accordance with FSAR Section 3.2.

Table 4.4-8 - FSAR Design Criteria Applied Loads WPPSS Unit #2										
Item	Operating Conditions	Extreme Loads						Accident Loads		
		Seismic	Tornado	Wind	Flood- External	Accidental Explosion	Malevolent Vehicle	Other		
I. Building Structure A) Radwaste Bldg. 1. Concrete 2. Steel		I ¹ II ²	I ³ NNS	I ⁴ NNS ⁴	I ⁵ NI	I ⁶ NI	NI NI	NI NI	NI NI	
II. Systems A) Liquid		NNS	NNS	NNS	NA	NI	NI	NI	NI	
B) Gaseous		II	NA	NA	NA	NI	NI	NI	NI	
C) Solid		NNS	NNS	NNS	NA	NI	NI	NI	NI	
III. Distribution Systems A) Piping 1. Process 2. Fire 3. Inst. & Cont.	NI	NI	NI	NI	NI	NI	NI	NI	NI	
B) Cable Trays	NI	NI	NI	NI	NI	NI	NI	NI	NI	
C) Ventilation	NI	NI	NI	NI	NI	NI	NI	NI	NI	
IV. Components A) Pumps	NI	NI	NI	NI	NI	NI	NI	NI	NI	
B) Tanks & Vessels	NI	NI	NI	NI	NI	NI	NI	NI	NI	
C) Valves	NI	NI	NI	NI	NI	NI	NI	NI	NI	
D) Heat Exchanger	NI	NI	NI	NI	NI	NI	NI	NI	NI	
E) Switch Gear	NI	NI	NI	NI	NI	NI	NI	NI	NI	

Table 4.4-8 - FSAR Design Criteria Applied Loads WPPSS Unit #2 (Continued)									
Item	Operating Conditions	Extreme Loads						Accident Loads	
		Seismic	Tornado	Wind	Flood- External	Accidental Explosion	Malevolent Vehicle		
F) MCC	NI	NI	NI	NI	NI	NI	NI	NI	NI
G) Fans, Dampers	NI	NI	NI	NI	NI	NI	NI	NI	NI

Footnotes for Table 4.4-8:

1. Designed for SSE PGA of 0.25g.
2. Designed for OBE PGA of 0.125g with AISC increased allowable stresses of 1.33.
3. Tornado loadings as defined in S.R.P. 3.3.3.5 and RG 1.76.
4. Wind Load is defined as 100 mph in accordance with ASCE Paper 3269
5. The maximum flood level is defined as El. 431' which is approximately 10' below grade.
6. Accidental explosion load defined as 20,000 lbs of dynamite 510 ft. from R.B. which is considerable less than what is required by RG 1.91.

Table 4.4-9 - FSAR DESIGN CRITERIA Code and Standards and Nuclear Design Criteria Davis Besse - Unit #1						
Item	Nuclear Design Criteria	Class				Construction Codes
		Quality	Seismic	Tornado	Safety	
I. Building Structure	NI	NI	NI	NI	NI	NI
II. Systems A) Liquid B) Gaseous C) Solid	10 CFR 20, 10 CFR 50	NI	NI	NI	NI	NI
III. Distribution Systems A) Piping 1. Process 2. Fire	NI	NI	NI	NI	NI	NI
B) Piping/Tubing (Inst.)	NI	NI	NI	NI	NI	NI
C) Cable Trays	NI	NI	NI	NI	NI	NI
E) Ventilation	NI	NI	NI	NI	NI	NI
IV. Components						
A) Pumps	NI	NI	NI	NI	NI	NI
B) Tanks & Vessels	NI	NI	NI	NI	NI	NI
C) Valves	NI	NI	NI	NI	NI	NI
D) Heat Exchanger	NI	NI	NI	NI	NI	NI
E) Switch Gear	NI	NI	NI	NI	NI	NI
F) MCC	NI	NI	NI	NI	NI	NI
G) Fans, Dampers	NI	NI	NI	NI	NI	NI

Table 4.4-10 - FSAR Design Criteria Applied Loads Davis Besse Unit # 1

Item	Operating Conditions	Extreme Loads							Accident Loads
		Seismic	Tornado	Wind	Flood-External	Accidental Explosion	Malevolent Vehicle	Other	
I. Building Structure	NI	NI	NI	NI	NI	NI	NI	NI	Assumed Waste Gas Tank Rupture. Tank is assumed to contain gaseous release from reactor coolant with 1% defective fuel consequences. See Note 1.
II. Systems A) Liquid B) Gaseous C) Solid	NI	NI	NI	NI	NI	NI	NI	NI	NI
III. Distribution Systems A) Piping 1. Process 2. Fire 3. Inst. & Cont.	NI	NI	NI	NI	NI	NI	NI	NI	NI
B) Cable Trays	NI	NI	NI	NI	NI	NI	NI	NI	NI
C) Ventilation	NI	NI	NI	NI	NI	NI	NI	NI	NI
IV. Components A) Pumps	NI	NI	NI	NI	NI	NI	NI	NI	NI
B) Tanks & Vessels	NI	NI	NI	NI	NI	NI	NI	NI	NI
C) Valves	NI	NI	NI	NI	NI	NI	NI	NI	NI
D) Heat Exchanger	NI	NI	NI	NI	NI	NI	NI	NI	NI
E) Switch Gear	NI	NI	NI	NI	NI	NI	NI	NI	NI

Table 4.4-10 - FSAR Design Criteria Applied Loads Davis Besse Unit # 1 (Continued)									
Item	Operating Conditions	Extreme Loads						Accident Loads	
		Seismic	Tornado	Wind	Flood-External	Accidental Explosion	Malevolent Vehicle	Other	
F) MCC	NI	NI	NI	NI	NI	NI	NI	NI	NI
G) Fans, Dampers	NI	NI	NI	NI	NI	NI	NI	NI	NI

Footnotes for Table 4.4-10:

The rupture of a waste gas decay tank would release the entire contents of the tank to the auxiliary building atmosphere. The Auxiliary Building is ventilated and discharged to the station vent. In the analysis, however, the activity is assumed to be released from the waste gas decay tank to the atmosphere over a two-hour time period.

Table 4.4-11 Comparison of Radwaste Design Criteria												
Facility	Structures											
	Liquid				Gaseous				Solid			
	1 Safety Class	2 Quality Class	3 Seismic	4 Explosion	1 Safety Class	2 Quality Class	3 Seismic	4 Explosion	1 Safety Class	2 Quality Class	3 Seismic	4 Explosion
Advanced Reactors												
GE-ABWR												
a. Flooded able Portions	SC-3	C ⁽¹⁾	I-3 ⁽¹⁾	NI	NI	NI	NI	NI	NI	NI	NI	NI
b. Other portions	NNS	D	NNS	NI	NI	NI	NI	NI	NI	NI	NI	NI
c. All	NI	NI	NI	NI	SC-3	C	I	NI	NNS	D	NNS	NI
ABB-CE 80 ⁺												
a. Flooded able Portions	SC-3	C ⁽¹⁾	I-3 ⁽¹⁾	NI	NI	NI	NI	NI	NI	NI	NI	NI
b. Other portions	NNS	D	NNS	NI	NI	NI	NI	NI	NI	NI	NI	NI
c. All	NI	NI	NI	NI	SC-3	C	I	NI	NNS	D	NNS	NI
Westinghouse AP-600												
a. Flooded Able Portions	SC-3	C ⁽¹⁾	I-3 ⁽¹⁾	NI	NI	NI	NI	NI	NI	NI	NI	NI
b. Other portions	NNS	D	NNS	NI	NI	NI	NI	NI	NI	NI	NI	NI
c. All	NI	NI	NI	NI	SC-3	C	I	NI	NNS	D	NNS	NI

Table 4.4-11 Comparison of Radwaste Design Criteria (Continued)

Facility	Systems and Components											
	Liquid				Gaseous				Solid			
	1 Safety Class	2 Quality Class	3 Seismic	4 Explosion	1 Safety Class	2 Quality Class	3 Seismic	4 Explosion	1 Safety Class	2 Quality Class	3 Seismic	4 Explosion
Advanced Reactors												
GE-ABWR												
Drain Piping	NNS	D	NNS	N	NI	NI	NI	NI	NI	NI	NI	NI
EECS Check Valves	NNS	C	I	N	NI	NI	NI	NI	NI	NI	NI	NI
Containment Boundary	2	B	I	N	NI	NI	NI	NI	NI	NI	NI	NI
Piping	NNS	D	NNS	N	NNS	D	NNS	N	NI	NI	NI	NI
Others	NNS	D	NNS	N	NNS	D	NNS	N	NNS	D	NNS	N
ABB-CE 80⁺												
Charcoal Cond. Piping	NI	NI	NI	NI	NNS	NI	NNS	Y	NI	NI	NI	NI
and Cooler												
Gas Analyzer and Asso.	NI	NI	NI	NI	NNS	NI	NNS	Y	NI	NI	NI	NI
Pipe												
Other	NNS	NI	NNS	NI	NI	NI	NI	NI	NNS	NI	NNS	N
Westinghouse AP-600												
Containment Isol. Syst.	NNS	D	I	N	NI	NI	NI	NI	NI	NI	NI	NI
Drain of Core Cooling	NNS	D	I	N	NI	NI	NI	NI	NI	NI	NI	NI
System												
ECCS and CVCS,	NNS	D	I	N	NI	NI	NI	NI	NI	NI	NI	NI
Check Valves	NI	NI	NI	NI	2	D	II ⁽¹⁾	N	NI	NI	NI	NI
Guard and Decay												
Beds	NNS	D	NNS	N	NNS	D	NNS	N	NNS	D	NNS	N
Other												

Table 4.4-11 Comparison of Radwaste Design Criteria (Continued)

Structures												
Facility	Liquid				Gaseous				Solid			
	1 Safety Class	2 Quality Class	3 Seismic	4 Explosion	1 Safety Class	2 Quality Class	3 Seismic	4 Explosion	1 Safety Class	2 Quality Class	3 Seismic	4 Explosion
Operating Reactors												
Perry-1 GE BWR-6	NI	NI	NI	NI	NNS	D	NNS	N	NI	NI	NI	NI
Turbine Bldg.	NI	NI	NI	NI	SC-3	D	I	N	NI	NI	NI	NI
Off Gas Bldg.												
Waterford - 3												
CE-PWR												
Auxiliary Bldg.	SC-3	B	I	N	SC-3	B	I	N	SC-3	D	I	N
Solidification Bldg.	NNS	D	NNS	N	NNS	D	NNS	N	NNS	D	D	N
Compactor Bldg.	NNS	D	NNS	N	NNS	D	NNS	N	NNS	D	D	N
Low Level Rad. Stor. Bldg.	NNS	D	NNS	N	NI	NI	NI	NI	NNS	D	NNS	N
Beaver Valley-2												
W-PWR												
Waste Handling Bldg.	NNS	D	NNS	N	NI	NI	NI	NI	NNS	D	NNS	N
GWST Enclosure	NI	NI	NI	NI	NNS	D	NNS	N	NI	NI	NI	NI
WWPS-2 GE BWR												
Radwaste Bldg Encl.	SC-3	B	I	N	SC-3	B	I	N	SC-3	B	I	N
Liquid Waste												
Radwaste Bldg. Superstructure	NNS	D	I ⁽²⁾	N	NNS	D	I ⁽²⁾	N	NNS	D	I ⁽²⁾	N
Davis Besse B&W-PWR	NI	NI	NI	NI	NI	NI	NI	NI	NI	NI	NI	NI

Table 4.4-11 Comparison of Radwaste Design Criteria (Continued)

Facility	Systems and Components											
	Liquid				Gaseous				Solid			
	1 Safety Class	2 Quality Class	3 Seismic	4 Explosion	1 Safety Class	2 Quality Class	3 Seismic	4 Explosion	1 Safety Class	2 Quality Class	3 Seismic	4 Explosion
Operating Reactors												
Perry-1 GE BWR-6 Mark III All	NNS	D	NNS	NI	NNS	D	NNS	Y	NNS	D	NNS	NI
Waterford - 3 CE-PWR Selected Part of Gaseous Radwaste System Other	NI NNS	NI D	NI NNS	NI N	NNS NNS	D D	I NNS	N N	NI NNS	NI D	NI NNS	NI NI
Beaver Valley-2 W- PWR Selected Part of Gaseous Radwaste System Other	NNS NI	D NI	NNS NI	N NI	NI NNS	NI D	NI NNS	NI N	NNS NI	D NI	NNS NI	N NI
WWPS-2 GE BWR 5 Mark II All	NNS	D	NNS	NI	NNS	D	NNS	N	NNS	D	NNS	NI
Davis Besse B&W-PWR All	NNS	D	NNS	NI	NNS	D	NNS	N	NNS	D	NNS	NI
Column Designations: 1. Safety Class (ANS 50 Criteria) 2. Quality Class (R.G. 1.29) 3. Seismic Class (SRP Section 3.2) 4. Hydrogen Explosion												
Notes: 1. Up to liquid system rupture flood height. 2. Designed to seismic Category I requirements 3. Designed to 0.5 SSE N = no; Y = yes, NA = Not Applicable, NI = No Information, NNS = Non Nuclear Safety												

4.5 Current Regulatory Guidance

Current regulatory guidance for the structural-mechanical design and construction of radioactive waste systems located at LWR plants are concentrated in the following Safety Analysis Report Sections and Regulatory Guides.

4.5.1 Safety Analysis Report Sections

- 1.2 General Plant Description
- *• 2.2.1-2.2.2 Identification of Potential Hazards in Site Vicinity
- 3.2.1 Seismic Classification
- 3.2.2 System Quality Group Classification
- *• 3.3.1 Wind Loadings
- *• 3.3.2 Tornado Loadings
- *• 3.4.1 Flood Protection
- *• 3.4.2 (Flood) Analysis Procedures
- *• 3.5.1.3 Turbine Missiles
- *• 3.5.1.4 Missiles generated by Natural Phenomena
- *• 3.5.1.5 Site Proximity Missiles (Except Aircraft)
- *• 3.5.1.6 Aircraft Hazards
- *• 3.5.2 Structures, Systems and Components to be Protected from Externally Generated Missiles
- *• 3.5.3 Barrier Design Procedures
- *• 3.6.1 Plant Design for Protection Against Postulated Piping Failures in Fluid Systems Outside Containment
- *• 3.6.2 Determination of Rupture Locations and Dynamic Effects Associated with Postulated Rupture of Piping
- 3.7 Seismic Design
- 3.8.4 Other Seismic Category I Structures
- 3.9.2 Dynamic Testing and Analysis of Systems, Components and Equipment
- 3.9.3 ASME Code Class 1, 2 and 3 Component and Component Supports
- *• 3.9.6 In-service Testing of Pumps and Valves
- *• 3.10 Seismic and Dynamic Qualifications of Mechanical and Electrical Equipment
- *• 3.11 Environmental Qualification of Mechanical and Electrical Equipment
- 11.1 Source Terms
- 11.2 Liquid Waste Management Systems
- 11.3 Gaseous Waste Management Systems
- 11.4 Solid Waste Management Systems
- 11.5 Process and Effluent Radiological Monitoring Instrumentation and Sampling Systems
- 15.7.3 Postulated Radioactive Release to Liquid Containing Tank Failure

4.5.2 Regulatory Guides

- RG 1.24 "Assumptions Used for Evaluating the Potential Radiological Consequences of a Pressurized Water Reactor Radioactive Gas Storage Tank Failure"
- RG 1.26 "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants"
- RG 1.29 "Seismic Design Classification"
- *• RG 1.59 "Design Basis Floods for Nuclear Power Plants" (Errata published 7/30/80)
- RG 1.60 "Design Response Spectra for Seismic Design of Nuclear Power Plants"
- RG 1.61 "Damping Values for Seismic Design of Nuclear Power Plants"
- *• RG 1.76 "Design Basis Tornado for Nuclear Power Plants"
- *• RG 1.91 "Evaluations of Explosions Postulated to Occur on Transportation Routes Near Nuclear Power Plants"
- RG 1.92 "Combining Modal Responses and Spatial Components in Seismic Response Analysis"
- RG 1.98 Assumptions Used for Evaluating the Potential Radioactive Consequences of a Radioactive Off-gas System Failure in a Boiling Water Reactor
- RG 1.100 "Seismic Qualification of Electric and Mechanical Equipment for Nuclear Power Plants"
- *• RG 1.102 "Flood Protection for Nuclear Power Plants"
- *• RG 1.115 "Protection Against Low-Trajectory Turbine Missiles"
- *• RG 1.117 "Tornado Design Classification"
- RG 1.122 "Development of Floor Design Response Spectra for Seismic Design of Floor-Supported Equipment or Components"
- RG 1.143 "Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants"

The structural-mechanical guidance documents marked with a "*" currently are not being applied to radwaste storage and processing SSC. The scope of RG 1.143 addresses only the seismic natural hazard phenomena and the accidental generation of hydrogen in the gaseous waste system. With the suggested inclusion of other natural hazards (wind, tornado and flood), man-induced accident hazards these other USNRC guidance documents would require consideration and referenced in Regulatory Guide 1.143.

4.6 Special Issues and Concerns

4.6.1 OBE/SSE Seismic Issues

The revised seismic design criteria as discussed in Section 2.3 of this report as currently applied to the Advanced Reactor Designs has effectively eliminated the OBE as a design basis seismic event. (OBE less than 1/3 SSE) For this reason, the reference to the OBE in Section 5, "Seismic Design for Radwaste Management Systems and Structures Housing Radwaste Management Systems" in the current edition of RG 1.143 Rev. 1 should be revised.

To address this item we suggest that if a high level seismic design is required the following criteria should apply:

- (a) If there is no explicit plant design for an OBE then the facility design basis should be 1/2 SSE.
- (b) If there is an explicit plant design for an OBE then the facility design basis should be the OBE.

This recommendation is made to address the revised seismic criteria of 10CFR50 Appendix S. The actual suggested seismic design criteria for a given facility will be further modified based on the following discussion and considerations presented in Section 4.7. This suggested revised criteria will address damping values and allowable stress criteria for various Radwaste SSC.

Very few waste facility SSC have been designed using the OBE level seismic design criteria. In most applications, gaseous waste system components identified as requiring seismic design in Regulatory Guide 1.143 and most liquid waste systems have been housed in Seismic Category I structures. Postulated failure of these systems (but not the structures housing them) due to an earthquake would not result in radiological consequences which would exceed current regulatory guides. Solid waste systems are normally housed in a non Seismic Category I structure. When discussed, the seismic design of these waste systems not categorized as Seismic Category I are usually designed in accordance with the Uniform Building Code. The Uniform Building Code contains different design requirements for hazardous SSC (typically multiplication factor which range from 1.0 to 1.5) than it does for ordinary SSC, and this difference is not identified in existing plant specific criteria. It should also be understood that all U.S. regional building codes (including the Uniform Building Code) are scheduled

to be replaced by the International Building Code, IBC, by the year 2002. The draft provisions of the IBC are considerably different and more complex than those used in the current edition of the UBC. It should also be noted that the UBC or IBC are not consensus standards while ASCE 7-95 which covers the same seismic design criteria is a consensus standard.

It is suggested that the use of a 1.33 increase in allowable stress and lower damping values associated with the OBE seismic input should also be modified if an OBE design is used. The SSE input typically use a multiplication factor of 1.6 increase in allowable stress values and higher damping values or an equivalent reduction in load factors. DOE in its design of waste facilities has used a graded approach from DOE Standard 1020. This approach places such facilities in Performance Category 3 with a significantly reduced seismic input (mean 10^{-4} /yr peak ground acceleration and spectral shape reduced to a 5×10^{-4} /yr pga mean spectral shape) with additional reductions due to an inelastic energy absorption factor and a 1.6 increase in allowable stress. The use of the current OBE damping values and stress limits could for many SSC result in an OBE controlled design.

It should also be noted that RG 1.143 Rev. 1 specifies what SSC should be designed for a modified reduced SSE RG 1.60 shaped spectral input. This includes such items as the structure having the liquid systems up to the spill height and gaseous waste structure and SSC as defined in RG 1.143 Rev. 1 Regulatory Position Section 5. However, it is silent as to what seismic requirements should be applied to solid waste SSC and liquid waste SSC above the potential spill height.

Finally, the use of RG 1.60 shaped spectra is specifically referenced in RG 1.143 Regulatory Position 5. In recent nuclear power plant applications, it has been fairly common practice to define site specific spectra at nuclear power plant sites which do not specify the RG 1.60 spectra. In addition, current seismic criteria for Advanced Reactor Design as discussed in RG 1.165 endorses the use of site specific spectra for which RG 1.60 is not applicable. Therefore, modifications to the specification of the spectra shape should be provided to address the possible use of site specific spectra.

4.6.2 Other Natural Hazard Design

The current RG 1.143 Rev. 1 does not address design criteria for natural phenomena hazards other than seismic. Needed is some consideration of extreme wind, tornado, flood (including local precipitation) and possibly volcanic action. Again, DOE Standard 1020 addresses these other natural phenomena hazards for

waste facilities identified as Performance Category 3 but with significantly less stringent design criteria as compared to the design of reactor safety related systems.

4.6.3 Man Induced Hazard Events

Nuclear power plants (reactor safety) are routinely evaluated for accidental fixed and transportation vehicle explosions (SRP Section 2.0 and RG 1.91). In addition, NPP are also evaluated for malevolent vehicle effects such as close vehicle proximity explosion and intrusion. It should also be noted that current tornado design of nuclear facilities (240-360 mph wind, pressure drop and associated missiles) can in general be shown to envelop the design requirements of small aircraft crash. However, if there is no tornado design or a reduced tornado design requirement, this may no longer be the case. In general, small aircraft crash cannot be considered a statistically defined event and should receive some consideration as a design basis event independent of probabilities.

4.6.4 The Use of the Graded Approach

In general, the radiological consequences from the design basis failure of nuclear power plant radwaste systems should result in much less effects to the public and colocated workers as compared to a reactor related accident. This is due primarily to the fact that there is normally no high energy driving force for the radwaste as compared to a reactor system failure. Secondly, the specific activity and quantity of most of the radwaste systems are several orders of magnitude less than the reactor systems. For these reasons, we suggest developing safety criteria for radwaste SSC considerably less stringent or rigorous than that used for large power reactors safety related systems consistent with the approach taken by the U.S. Department of Energy. This is further discussed in Section 4.7.

4.7 General Discussion of Suggested Modifications of Criteria for Radioactive Waste Facility Design

4.7.1 ANS Waste Standards

There are three relatively recently published ANS Standards dealing in detail with radioactive liquid, gaseous, and solid waste: ANSI/ANS 55.6-1993 "Liquid Radioactive Waste Processing System for Light Water Reactor Plants", ANSI/ANS 55.4-1993 "Gaseous Radioactive Waste Processing Systems for

Light-Water Cooled Reactor Plants" and ANSI/ANS 55.1-1992 "Solid Waste Processing System for light Water Cooled Reactor Plants" respectively. The content and organization of the liquid and gaseous standards are very similar, however the solid waste standard organization is somewhat different. In general the ANS radioactive waste standards are consistent with and considerably more detailed than the current requirements of R.G. 1.143 Rev. 1. The following subsection provides overviews of each of these standards

S&A has conducted a detailed review of the three current ANS Standards relative to radioactive waste management for possible incorporation into the text of the revised R.G. 1.143-Rev. 1, 1979. In review of the text of ANSI/ANS 55.1-1992, ANSI/ANS 55.4-1993 and ANSI/ANS 55.6-1993, it is noted that the scopes of these standards are much broader than that covered by the existing Regulatory Guide 1.143 since they also encompass much of the scope contained in Standard Review Plan, Section 11.0. It is not intended to increase the original scope of R.G. 1.143 except as might be indicated to be in compliance with the referenced Design Criteria of 10 CFR 50 Appendix A or as already indicated in the Regulatory Guide. Therefore, this review of the ANSI/ANS 55.1, 55.4 and 55.6 standards is limited to the scope covered by the existing R.G. 1.143-Rev. 1, 1979.

4.7.1.1 Overview of ANS 55.1-1992 (Solid Wastes)

This standard provides guidance on the design, construction, and performance standards for solid waste processing systems in light water cooled reactors. Radiation release requirements are required to be consistent with ALARA (without direct reference to 10CFR50 Appendix I) and 10 CFR Part 20. The only natural or man induced hazards considered are seismic (earthquake) design for the structure housing the solid waste systems and components. The solid radwaste equipment and systems are not required to be designed to withstand the effects of earthquake events. This appears to mean that no earthquake design resistance is provided, even that associated with normal commercial systems and components in accordance with applicable Building Codes. The foundations and adjacent walls of structures housing the solid radwaste processing equipment shall be designed to withstand the effects of an OBE. The standard also provides a set of suggested equipment codes and standards, however, the list deals primarily with pressure boundary equipment and is silent on electrical, instrumentation, structural, concrete, structural steel, and HVAC standards.

4.7.1.2 Overview of ANS-55.4-1993 (Gaseous Wastes)

This standard sets forth the minimum design, construction and performance requirements for gaseous radioactive waste processing systems in Light Water Reactors. Radiation release design requirements are specified to be 10 CFR20 and a small fraction (undefined) of 10 CFR Part 100. Exposure to operating personnel shall be monitored and ALARA is in accordance with 10CFR50, Appendix I.

These design criteria are multiple faceted and somewhat open to interpretation. These requirements are summarized as follows:

- (a) For equipment that is used to store gaseous waste material for decay the equipment supports and the building housing the equipment shall be designed for an OBE.
- (b) For radioactive waste gas process systems which operates at greater than 1.5 atm absolute [155kPa (22.5 psia)], all components shall be designed for an OBE. (This includes supports and the building in which the system/component is housed).
- (c) For radioactive waste gas process systems which operate at less than or equal to 1.5 atm absolute [155kPa (22.5 psia)] and retain gases on charcoal absorbers, only the equipment supports and building which houses the equipment need to be designed for an OBE.

No specific criteria is provided for systems which operate at less than or equal to 1.5 atm absolute [155kPa (22.5 psia)] which do not retain gases on charcoal absorbers. It could be assumed that (a) above applies, but clearly additional guidance is required.

For potential hydrogen/oxygen explosion, the standard requires that if the potential for an explosive mixture of hydrogen and oxygen exists, the gaseous waste handling and treatment system shall be designed either to withstand the effects of a hydrogen explosion or be provided with two independent gas analyzers to detect the presence of flammable mixtures and to annunciate both locally and in the main control room.

Catalytic recombined outlet piping shall not be interconnected with those piping regions which by design, contain explosive mixtures.

The standard also provides a set of suggested equipment codes and standards, but the list deals primarily with pressure boundary equipment and is

silent on electrical standards, structural standards, concrete standards etc.

4.7.1.3 Overview of ANS 55-6 - 1993 (Liquid)

The standard provides guidance on the design construction and performance standards for Liquid Radioactive Waste management facilities in light water cooled reactors. Radiation release requirements are to be per 10 CFR20 and operator exposure also shall be maintained ALARA. (The standard references 10CFR20 versus 10CFR50 Appendix I for ALARA requirements). The only natural phenomena and man made hazard requirements are seismic (earthquake) and this design requirement is limited to the structure containing the radioactive liquid inventory.

The Liquid Radwaste Processing Systems are not required to be designed to withstand the effects of a seismic event. The buildings housing the liquid Radwaste processing systems shall be designed to contain the liquid inventory of the building in the event of an OBE event.

The standard also provides a set of suggested equipment codes and standards but deals primarily with pressure boundary equipment and is silent on electrical standards, structural standards, concrete standards, etc.

4.7.1.4 Seismic Design Considerations

It is suggested that all radioactive waste SSC should be designed at least for local building code seismic requirements. In the absence of such local criteria it is recommended that the requirements of Standards UBC-97 or ASCE-7-95 be used. Furthermore, the SSC processing or storing small quantities of radioactive waste should be identified as hazardous and be required as a minimum to meet design requirements of hazardous SSC as defined in the conventional facility design codes. This typically requires multiplication factors which range from 1.0 to 1.5 applied to ordinary UBC-97 seismic design load requirements and restricts the type of construction that can be used with ASCE-7-95.

4.7.1.5 Other Hazards

It should also be noted that the current ANS Waste Standards are equally silent on natural phenomena hazards design criteria relative to wind, tornado, flood, lightning and volcano and man-induced external hazards such as accidental explosion, airplane crash, and malevolent vehicle intrusion and explosion. It is recommended that the RG 1.143 be expanded to cover generic natural phenomena such as wind, tornado and flood, and man-induced external hazards such as

malevolent vehicle intrusion, explosion and small aircraft crash. Also, accidental vehicle, or an adjacent facility explosion should be considered in the design.

4.7.1.6 Codes and Standards

Additional specification of radwaste facilities design standards to augment the information provided in the ANS is also required. In addition, modification of capacity acceptance criteria may be required for the specified standards to address natural phenomena and external man-induced hazards. Section 4.8 provides a more detailed discussion of applicable codes and standards.

4.7.1.7 Conclusion

Except for the items identified above for natural phenomena and external man-induced hazards, it is suggested that the requirements of ANS 55.1, 55.4 and 55.6 could be incorporated into a revised RG 1.143 by reference which could permit a significant reduction to the volume of the current text of RG 1.143.

4.7.2 U. S. Department of Energy Criteria and Approach

The U.S. Department of Energy DOE has developed a graded approach to design or evaluation of their nuclear facilities for natural phenomena hazards. DOE facilities include nuclear material processing and storage facilities for a variety of nuclides and forms as well as nuclear waste storage facilities and nuclear waste processing facilities. The total quantity of DOE nuclear waste in its current inventory far exceeds the total quantity generated by the civilian nuclear power program to date.

In DOE's graded approach they have developed three facility hazard classes, High, Moderate and Low or 1, 2 or 3 and within these facilities they specify two safety related classes of Structures, Systems or Components, SSC, Safety and Safety Significant plus a "non" safety classification. Associated with these Safety Related SSC classifications are five Performance or Design Classes. The fifth classification or PC-0 classification is applicable to general use facilities which have no significant hazards nor human occupancy issues associated with them. Therefore, it will not be discussed further here. The relationships between Hazard, Safety Related, Safety Significant and Performance Classifications are shown in Table 4.7-1. A further breakdown of design requirements for Natural Hazard Phenomena as a function of Performance Category is given in Table 4.7-2.

The definition of these various Performance Categories with different hazard design requirements based on probabilities is what is referred to as graded approach to design.

There are a number of Natural Phenomena hazards that have not been addressed by DOE Standards and these include:

- Lighting
- Storm Surge (directly)
- Tsunami and Seiche
- Wave Action
- Volcano
- Slope Stability

These items are addressed on a site and/or facility specific basis. DOE has not developed standards for design against man-induced external hazards other than to identify a requirement to consider them in design as given in DOE Order 420.1. A list of some potential external man-induced external hazards are as follows:

- Aircraft Impact (both large and small)
- High Energy Equipment Failure
- Industrial or Military Facility Explosion or Toxic Material Release
- Pipeline Explosion or Toxic Material Release
- Retaining Structure Failure
- Surface Vehicle Accidental Explosion
- Malevolent Vehicle Intrusion and Explosion

SSC found in commercial nuclear power plants having a seismic design requirement defined in RG 1.143, are similar to SSC which would be designed as DOE Performance Category 3. In general, there is a reasonable correlation between the current RG 1.143, criteria and that imposed by DOE Standard 1020 for the design of waste management SSC. This suggests that the DOE requirements for wind, tornado and flood associated with PC-3 also could be considered for application to commercial nuclear power waste management SSC. DOE, in Standard 1020 permits the use of inelastic energy absorption factors for seismic design of PC-3 structures. The factors used by DOE in Std 1020 are shown in Table 4.7-3. The DOE Std 1020 also suggests that an inelastic energy absorption factor, F_u greater than 1.0 could be used for ductile systems and components, as well as structures but does not provide any specific values.

DOE permits a limited amount of inelastic demand adjustment in the form of F_u , which is an inelastic energy demand coefficient divided into the elastically computed seismic stresses in structural elements. The DOE approach is consistent with but considerably more conservative than seismic building code requirements

of UBC-97 for structures designed for occupant life safety considerations. The UBC-97 R_w factors which range between 5 and 12 are shown in Table 4.7-4.

Reduction in elastically computed seismic load and load effects have not been used historically with USNRC accepted seismic design criteria. This makes the current USNRC seismic design criteria as given in RG 1.143 considerably more conservative than that used by DOE for similar facilities.

Considering these factors, it is suggested that if it is necessary that SSC in commercial nuclear facilities be designed for natural phenomena and man-made hazards that a criteria similar to DOE 1020 Performance categories be applied for Earthquake, Wind, Tornado, Flood and Precipitation. Further, a design criteria should also be specified for malevolent vehicle assault and small aircraft crash. The application of a specific performance category will be discussed in Section 4.7.3 and Section 5.0.

4.7.3 Threshold Quantities of Radioactive Waste

The primary goal of any Radwaste Facility design and/or operational criteria to (a) control and confine radioactive waste material, (b) limit public exposure to radiation to within USNRC established limits, and (c) to apply ALARA to exposure received by workers. This applies not only to normal facility operations, but also to the radiation exposure resulting to the public or to workers during or after a design basis natural phenomena or man induced hazard or design basis accident event. The section discusses release and exposure levels which it is suggested should be considered in establishing the facility design requirements.

4.7.3.1 SRP Section 11 Criteria

While threshold quantities of radioactive wastes and their forms are discussed in the SRP Section 11, there does not appear to be a clear design criteria distinction between so called high hazard and hazardous waste.

Normal operational radiological release rates for Waste Management facilities to individuals in the unrestricted area are provided in directly in SRP Section 11 and by reference to 10CFR20, Appendix B, Table 2. Seismic design criteria are provided via reference to Regulatory Guide 1.143.

4.7.3.2 10 CFR20 Criteria

10CFR20 Subpart D provides guidance on the average annual release rates of gaseous and liquid effluents for

radioactive waste management facilities at the boundary of the unrestricted area. The rate is exclusive of the disposal of radioactive material into sanitary sewers which is established in Section 10CFR20 Subpart K. The effluent release rate (air and water) and the release to sewers are specified in Appendix B, Table 2 and Table 3 respectively. Further, Subpart D sets the limits of the release rate (TEDE) to the general public at 500 millirem annually. 10CFR20 Subpart D also limits on-site personnel radiation exposure (TEDE) to 5 rem annually.

4.7.3.3 10CFR71 Considerations

While this section is directed toward containment of packaging used in the transportation of radioactive materials, it does provide guidance on threshold quantities of radioactive waste for various types of packaging to include low specific activity LSA I, LSA II and LSA III as well as type A and type B packaging. Specifically, it provides limits on radioactive quantity levels permitted for various types of transportation packages. It is provided in terms of specific isotope or mixed isotope of special form radioactive material limits A_1 and unrestricted form limits A_2 given in Table A-1 of Appendix A to 10CFR71. Specific examples of the A_1 and A_2 lower bound quantity values for a Type B containment in values are shown in Table 4.7.5.

Special form radioactive material A_1 , means radioactive material that satisfies the following conditions:

- a) It is either a single solid piece or is contained in a sealed capsule that can be opened only by destroying the capsule;
- b) The piece or capsule has at least one dimension not less than 5 mm (0.2 in); and
- c) It satisfies the additional requirements of 10CFR71.

4.7.3.4 Branch Technical Position ETSB 11-5, Rev. 0, July 1981

This branch position paper currently gives some historical prospective as well as current guidance as to the acceptable accident level radiation limits to workers and the public.

As stated in the branch technical position paper, BTP ETSB 11-5 the basic criterion for reactor accidents, including waste gas system failures, is that off-site doses shall not exceed 25 rem to the whole body. However, that criterion is meant to be used for siting and should not be used as a release limit and is predicated on the assumption that the probability of occurrence is very small. At least since 1972, it has been recognized that the probability of an accidental

release from the waste gas system is relatively high and that lower dose criteria are appropriate.

The BTP further states that waste gas systems characteristics differ between plants whose systems are designed to withstand the effects of a hydrogen explosion and earthquakes and those systems not so designed. Gross failure of the system is considered much less likely if the system is designed to withstand explosions and earthquakes. Accordingly, higher dose criteria have been considered appropriate for evaluating gross failure of these fortified systems. Initially, a 5 rem criterion was used, but more recently the value has been 2.5 rem. For systems not designed to withstand explosions and earthquakes, the criterion has been 0.5 rem.

This dichotomy had led to a problem in that system malfunctions appear to be the controlling failure mode and resistance to explosions and earthquakes provide no protection against operator error and system malfunction. No system malfunction type failures have been designated as representative. However, it appears that an event, such as valve misalignment or over-pressure could give a release approximating that from the rupture of a tank or pipe. Therefore, it was considered that for future safety evaluations the waste gas system failures analyzed could be limited to tank or pipe ruptures but that the dose criterion in every case should be 0.5 rem at the boundary of the unrestricted area.

4.7.3.5 DOE Facility Requirements

DOE-STD 1027 provides threshold quantities of radioactive isotope quantities which are used for facility hazard classifications. Quantities are provided for Level 1, 2, and 3 facilities, however, Level 1 is primarily for Class A Power Reactors. For comparison to the 10CFR70 limits, the Level 2 and Level 3 quantities are also shown in Table 4.4-5.

4.7.3.6 Suggested Approach

It is suggested that a graded approach similar to that used by DOE be provided in Regulatory Guide 1.143. This approach would correlate radiation exposure potential to a specific level of natural phenomena and man made hazard design and provide for a graded approach for such design.

4.7.4 Effect on Operating Plants and ARD's

For the Advanced Reactor Designs and the late vintage operating plants, SSC with significant potential for radiation release to the environment, are located in

Seismic Category I Structures. Therefore, the application of a graded approach would not directly affect these plants. In fact, their design would be conservative with respect to this proposed RG 1.143 approach.

The siting criteria of 10 CFR 100 Appendix A is referenced as the basis for this limit.

Table 4.7-1 DOE Criteria Relationship Between Facility Classification and Structures, System and Component, SSC, Safety-Related and Performance Classifications		
Facility Hazard⁽¹⁾ Classification	Safety Related⁽¹⁾ SSC Classification	SSC Performance⁽²⁾ or Design Classification
High or 1	Safety Safety Significant Non-Safety	PC-4 PC-3 PC-1 or 2
Moderate or 2	Safety Safety Significant Non-Safety	PC-3 PC-2 PC-1 or 2
Low or 3	Safety Safety Significant Non-Safety	PC-2 PC-1 or 2 PC-1

Footnotes to Table 4.7-1:

- (1) Based on DOE Stds 3009 and 1027, References (7.4 [3]) and (7.4 [4]).
(2) Based on DOE Stds 1020 and 1021, References (7.4 [1]) and (7.4 [5]).

Table 4.7-2 Summary of External Natural Hazard Design Basis Event Probabilities as Defined in DOE Standard 1020					
Event	SSC Performance Category				
	1	2	3	4	
EARTHQUAKE ⁽¹⁾	Hazard Annual Probability of Exceedence New Facility	1x10 ⁻³	5x10 ⁻⁴	1x10 ⁻⁴	
	Hazard Annual Probability of Exceedence	2x10 ⁻²	1x10 ⁻³	1x10 ⁻⁴	
	Importance Factor	1.0	1.0	1.0	
	Missile Criteria ⁽⁵⁾	NA	See Note 1	See Note 2	
TORNADO ⁽²⁾	Hazard Annual Prob. of Exceedence	NA	2x10 ⁻⁵	2x10 ⁻⁶	
	Importance Factor	NA	1.0	1.0	
	Associated Pressure Drop	NA	1.915kPa (40 lb/ft ²) @ .96kPa/sec (20 psf/sec)	5.98kPa (125 psf) @ 2.4kPa/sec (50 psf/sec)	
	Missile Criteria ⁽⁶⁾	NA	See Note 3	See Note 4	
FLOOD ⁽¹⁰⁾	Mean Hazard Annual Prob. of Exceedence New Facility	2x10 ⁻³	5x10 ⁻⁴	1x10 ⁻⁴	1x10 ⁻⁵
PRECIPITATION	Rain and Snow Surface Run-off	2x10 ⁻³	5x10 ⁻⁴	1x10 ⁻⁴	1x10 ⁻⁵
	Roof Loads	2x10 ⁻³	5x10 ⁻⁴	1x10 ⁻⁴	1x10 ⁻⁵
	Site Drainage	4x10 ⁻³	4x10 ⁻³	4x10 ⁻³	4x10 ⁻³

Footnotes for Table 4.7-2:

- (1) 2 x 4 Timber plank 6.8 Kg (15 lbs) at 22.4 m/sec (50 mi./hr.) (horizontal) maximum height above grade is 9.14 meters (30 feet).
- (2) 2 x 4 Timber plank 6.8 Kg (15 lbs) at 22.4 m/sec (50 mi./hr.) (horizontal) maximum height above grade is 15.24 meters (50 feet).
- (3) a. 2 x 4 Timber plank 6.8 Kg (15 lbs) at 44.7 m/sec (100 mi./hr.) (horizontal).
- b. 2 x 4 Timber plank 6.8 Kg (15 lbs) at 31.3 m/sec (70 mi./hr.) (vertical).
- c. 3 inch nominal diameter Sch 40 steel pipe 34 Kg (75 lbs) at 22.4 m/sec (50 mi./hr.) (horizontal).
- d. 3 inch nominal diameter Sch 40 steel pipe 34 Kg (75 lbs) at 15.6 m/sec (35 mi./hr.) (vertical).

Footnotes in Table 4.7-2 (continued)

- (4) a. 2 x 4 Timber plank 6.8 Kg (15 lbs) at 67 m/sec (150 mi./hr.) (horizontal).
b. 2 x 4 Timber plank 6.8 Kg (15 lbs) at 44.7 m/sec (100 mi./hr.) (vertical).
c. 3 inch diameter Sch 40 steel pipe 34 Kg (75 lbs) at 33.5 m/sec (75 mi./hr.) (horizontal) maximum height above grade 30.5 m (100 ft).
d. 3 inch diameter Sch 40 steel pipe 34 Kg (75 lbs) at 22.4 m/sec (50 mi./hr.) (vertical).
- (5) It is suggested that the timber plank missile be deleted and replaced by the 3 inch pipe defined in Notes 3c and 3d, since timber plank disintegrates when it impacts reinforced concrete.
- (6) It is suggested that the timber plank missile be deleted and replaced by an impact type missile: Automobile with frontal area of 1.86 m² (20 square feet) weighing 1814 Kg (4000 lbs) traveling at 15.65 m/sec (35 mi./hr.) within 10.67 m (35 ft) of grade (horizontal).
- (7) Minimum earthquake load for PC-4 defined as 0.1g pga. For PC-3 minimum earthquake load defined as 0.06g pga and UBC-97 or ASCE 7-95 for PC-2 and PC-1.
- (8) Wind loads as defined in ASCE Std. 7-95 for PC-2 and PC-1.
- (9) It is recommended that a minimum tornado Fujita Scale-2 [70.2 m/sec (157 mi./hr.)] be defined for PC-4 and Fujita Scale-1 [50.1 m/sec (112 mi./hr.)] be defined for PC-3.
- (10) See ANSI/ANS Standard 2.6 for guidance.

Table 4.7-3 Inelastic Energy Absorption Factors, F_u , From DOE-1020	
Structural System	F_u
MOMENT RESISTING FRAME SYSTEMS-Beams	
Steel Special Moment Resisting Frame (SMRF)	3.0
Concrete SMRF	2.75
Concrete Intermediate Moment Frame (IMRF)	1.5
Steel Ordinary Moment Resting Frame	1.5
Concrete Ordinary Moment Resisting Frame	1.25
SHEAR WALLS	
Concrete or Masonry Walls	
In-plane Flexure	1.75
In-plane Shear	1.5
Out-of-plane Flexure	1.75
Out-of plane Shear	1.0
Plywood Walls	1.75
Dual System, Concrete with SMRF	2.5
Dual System, Concrete with Concrete IMRF	2.0
Dual System, Masonry with SMRF	1.5
Dual System, Masonry with Concrete IMRF	1.4
STEEL ECCENTRIC BRACED FRAMES (EBF)	
Beams and Diagonal Braces	2.75 3.0
CONCENTRIC BRACED FRAMES	
Steel Beams	2.0
Steel Diagonal Braces	1.75
Concrete Beams	1.75
Concrete Diagonal Braces	1.5
Wood Trusses	1.75
Beams and Diagonal Braces, Dual Systems	
Steel with Steel SMRF	2.75
Concrete with Concrete SMRF	2.0
Concrete with Concrete IMRF	1.4
METAL LIQUID STORAGE TANKS	
Moment and Shear Capacity	1.25
Hoop Capacity	1.5

Footnotes for Table 4.7-3:

1. Values herein assume good seismic detailing practice per UBC-94 along with reasonably uniform inelastic behavior. Otherwise, lower values should be used.
2. F_u for columns for all structural systems is 1.5 for flexure and 1.0 for axial compression and shear. For columns subjected to combined axial compression and bending, interaction formulas shall be used.
3. Connections for steel concentric braced frames should be designed for at least the lesser of:
 - The tensile strength of the bracing.
 - The force in the brace corresponding to F_u of unity.
 - The maximum force that can be transferred to the brace by the structural system.

Footnotes for Table 4.7-3 (continued):

4. Connections for steel moment frames and eccentric braced frames and connections for concrete, masonry, and wood structural systems should follow the VEC-94 provisions utilizing the prescribed seismic loads from these criteria and the strength of the connecting members. In general, connections should develop the strength of the connecting members or be designed for member forces corresponding to F_u of unity, whichever is less.
5. F_u for chevron, V, and K bracing is 2.5. K bracing requires special consideration for any building if Z is 0.25g or more.

Table 4.7-4 UBC-94 Code Reduction Coefficients, R_w	
Structural System (Terminology is identical to the UBC)	R_w
MOMENT RESISTING FRAME SYSTEMS - Beams	
Steel Special Moment Resisting Frame (SMRF)	12
Concrete SMRF	12
Concrete Intermediate Moment Frame (IMRF)	8
Steel Ordinary Moment Resting Frame	6
Concrete Ordinary Moment Resisting Frame	5
SHEAR WALLS	
Concrete or Masonry Walls	8(6)
Plywood Walls	9(6)
Dual System, Concrete with SMRF	12
Dual System, Concrete with Concrete IMRF	9
Dual System, Masonry with SMRF	8
Dual System, Masonry with Concrete IMRF	7
STEEL ECCENTRIC BRACED FRAMES (EBF)	
Beams and Diagonal Braces	10
Beams and Diagonal Braces, Dual System with Steel SMRF	12
CONCENTRIC BRACED FRAMES	
Steel Beams	8(6)
Steel Diagonal Braces	8(6)
Concrete Beams	8(4)
Concrete Diagonal Braces	8(4)
Wood Trusses	8(4)
Beams and Diagonal Braces, Dual Systems	
Steel with Steel SMRF	10
Concrete with Concrete SMRF	9
Concrete with Concrete IMRF	6

Footnotes for Table 4.7-4:

Values herein assume good seismic detailing practice per the UBC along with reasonable uniform inelastic behavior. Otherwise lower values should be used.

Values in parentheses apply to bearing wall systems or systems in which bracing carries gravity loads.

Table 4.7-5 Comparison of Selected Isotope Quantities Between 10 CFR 72 Appendix A and DOE Std. 1027.				
	Curies⁽¹⁾ 10 CFR 71 Appendix A		Curies⁽²⁾ DOE Std. 1027 Hazard Categories	
Isotope	A₁	A₂	Moderate 2	Low 3
Americium-241	8	0.008	55	0.05
Carbon-14	1000	60	1.4x10 ⁶	420
Cesium-137	30	10	8.9x10 ⁴	60
Cobalt-60	7	7	1.9x10 ⁵	280
Hydrogen-3 (Tritium)	--	--	3x10 ⁵	1000
Iodine-131	40	10	1800	0.92
Plutonium-239	2	0.002	56	0.5
Thorium-228	6	0.008	22	1.0
Strontium-90	10	0.4	2.2x10 ⁴	16
Uranium-235	100	0.2	240	4.2
Uranium-238	Unlimited	Unlimited	240	4.2
Mixed Fission Prod.	10	0.4	2.5x10 ⁶	---
Only Beta-or Gamma-existing nuclides known to be present	5	0.5	---	---
Alpha - emitting nuclides known to be present or no relevant data available	2.7	5.4 x 10 ⁻⁴	---	---

Footnotes for Table 4.7-5:

- (1) Quantities shown for 10CFR71 Appendix A are for a single container or package.
(2) Quantities shown for DOE Standard 1027 are for facility categorization.

4.7.5 Safety Significance of Material State of Waste

Waste is typically divided into its three major states, gaseous (vapors), liquids and solids with different design guidance applicable to each of the three states. Gaseous waste is normally considered more mobile than the other two waste forms. For this reason, the systems and components which store or house gaseous waste are constructed to more stringent hazard design criteria than those associated with other waste forms. In general, when waste design pressures exceed about 7.5 psig, all the systems and components which house, process or store such gaseous waste should be designed to the more stringent loading requirements of Regulatory Position 5 and the proposed new Table 2 of R.G. 1.143 using the standard listed in Table 1 of R.G. 1.143.

Liquid waste storage systems are less mobile than gaseous waste hence are not required to be designed to the more stringent requirements of regulatory position and the proposed new Table 2 of R.G. 1.143 using the standards listed in Table 1 of R.G. 1.143. Only the structure housing liquid waste processing or storage up to a height sufficient to contain the maximum liquid inventory are expected to meet the more stringent requirements of Regulatory Position 5 and Table 2 of R.G. 1.143.

Only the foundations and the adjacent walls of structures that house solid waste radwaste systems and components up to the maximum height of the solid waste should be constructed to the more stringent loading criteria given in Regulatory Position 5 and Table 2 of R.G. 1.143 using the standards listed in Table 1 of R.G. 1.143.

Other portions of the gaseous, liquids and solid waste systems, components and structures should be constructed using the criteria listed in Table 1 of R.G. 1.143 for loads as defined in applicable National Building Codes for the applicable type of occupancy.

4.7.6 Graded Approach Applicable to Quantities of Waste Being Processed or Stored

In 10 CFR 71 applicable to transportation packaging the USNRC has established five types of packaging as a function of the radiological inventory and the size and robustness of the packaging or encapsulation. These are identified as Type B, Type A, Low Specific Activity, LSA-I, LSA-II-LSA-III.

4.8 Applicability of Industry Standards

4.8.1 Specification of Standards

Table 4.8-1 provides a summary of the suggested standards which should be specified in RG 1.143 for the design and fabrication and/or construction of SSC in Radwaste Facilities. The current Table 1 given in Regulatory Guides 1.143 only provides Code and Standard specification for a limited number of component equipment classes. Table 4.8.1 is a more encompassing Table which addresses the design of all SSC in Radwaste Facilities. Section 4.8.1 provides a detailed discussion of the basis for the specification of each of the Codes and Standards given in Table 4.8-1. Section 4.8.2 discusses the design basis load combinations, and Section 4.8.3 discusses those acceptable capacity criteria.

4.8.2 Basis of a Selection of Industry Codes and Standards

This section provides a discussion of the basis for the selection of the Codes and Standards given in Table 4.8-1. If a specified code and/or standard is different from one specified in the current Regulatory Guide, then the basis for these changes are discussed in detail.

4.8.2.1 Structures - Concrete

Currently, Regulatory Guide 1.143 specifies the use of ACI-318 for Concrete Design. This standard lacks the expanded criteria and load combinations for the extreme events to which high hazards concrete structures would be subjected. Therefore, for concrete components in those portions of facilities which are critical to plant safety, the ACI-349 (nuclear concrete code) is proposed.

4.8.2.2 Structures-Steel (Hot Rolled)

Currently, Regulatory Guide 1.143 Specifies the use of the AISC. Steel Construction Manual 7th Edition. The proposed codes and standards include the latest editions of the AISC and provides the additional option to use the AISC Load and Resistance Factor Design (LRFD) Criteria. Both these standards, however, lack the expanded criteria and load combinations for the extreme events to which high hazard facilities would be subjected. Therefore, for steel structures in those portions of facilities which are critical to plant safety, the ANSI-N690 (Nuclear Steel Structures) Standard is proposed.

4.8.2.3 Structures - Steel (Cold Formed)

Currently, Regulatory Guide 1.143 provides no criteria for the design of Cold Formed Steel. Therefore, the AISI-Cold Formed Steel Design Manual (CFSDM), which is the most common industry standard for cold formed steel design is proposed.

4.8.2.4 Piping and Valves

Currently, Regulatory Guide 1.143 specifies the use of the B31.1 for piping design. B31.1 is primarily for high temperature, high pressure steam power piping design. B31.3 focuses on chemical process system design. This later focus is more consistent with the function of the piping systems in radioactive waste processing facilities and therefore it is proposed.

4.8.2.5 Atmospheric Tanks

It is suggested the design of these tanks should be the requirements of API-650 which is consistent with the current version of Regulatory Guide 1.143.

4.8.2.6 Tanks (0-15 psig)

It is suggested the design of these tanks should be the requirements of API-620 which is consistent with the current version of Regulatory Guide 1.143.

4.8.2.7 Pressure Vessels and Tanks (> 15 psig)

It is suggested the design of these tanks should be to the requirements of ASME Boiler and Pressure Vessel Code (BPVC) Section VIII Division 1 or 2 which is consistent with the current version of Regulatory Guide 1.143.

4.8.2.8 Pumps

Currently, Regulatory Guide 1.143 only specifies "manufacturer's standards" for pump design. The appropriate API Pump Design Standards are suggested to provide more definitive design guidance. The ASME BPVC Section VIII is also referenced as it is specified for pressure retaining components in the API Standards.

4.8.2.9 Heat Exchangers

The standards currently specified in Regulatory Guide 1.143 are maintained without change.

4.8.2.10 HVAC, Conduit, Cable Trays, Fire Protection Systems

Currently, Regulatory Guide 1.143 provides no design guidance for these items. Therefore, the Industry Standards most accepted for these designs are proposed. Further, for high hazardous and hazardous HVAC, Cable Tray and Fire Protection piping, the seismic design criteria of the appropriate standards is also proposed.

4.8.3 Load Combinations and Capacities

A significant number of the standards provided in Table 4.8-1 are commercial standards. These standards typically do not provide design criteria, load combinations and capacity criteria for the extraordinary natural phenomena and manmade hazard loadings considered in the design of Nuclear Facilities. Therefore, it is proposed that a table be added to Regulatory Guide 1.143 which specifies the applicable load combinations for evaluation of Nuclear Waste Processing Facilities.

In developing the load combination, three service levels were defined:

- Level A (Normal)
- Level B (Severe or Upset)
- Level D (Abnormal Extreme, or Faulted).

Associated with these service levels are capacity criteria which will be discussed in the following section (Section 4.8.2). This capacity criteria will provide a certain level of protection for the SSC under review. Load combinations are then formulated such that for a given loading situation an appropriate level of protection is provided. Secondly, the load combinations determine which levels must be considered simultaneously and which can be considered independently.

The load combinations were developed from the investigations and studies conducted and summarized in the previous sections of this report. The major considerations were: a) the criteria which has been used for the design of commercially operating waste management facilities at operating nuclear power plants, b) the licensing commitments of the Advanced Reactor Designs and c) the criteria currently under development with in the DOE.

The load combinations are divided into two basic groups: a) Structures and other non pressure retaining items and b) pressure retaining components.

Structures and other non pressure retaining components are further subdivided into internal and external categories. The later subdivision was made to differentiate between structures subjected to uniquely external hazard loadings such as wind, tornado, etc., and those structures not subjected to such loadings.

For pressure retaining components, the concept that primary and secondary stresses can be evaluated separately is applied. Therefore, the secondary stress loading due to thermal expansion are evaluated separately from applied primary loadings. Further, since a single application of secondary stresses can not cause a failure, the thermal expansion loads are not considered for Level D (Faulted) loadings.

The resulting load combinations are given in Table 4.8-2. The load combinations are provided for each of three component safety classes. The basis and definition of these safety classes is provided in Section 5.2.2.2.

Finally, no criteria is provided for the evaluation of water hammer type loadings. It is the authors' belief that one does not design piping for water hammer. System design and operating procedures should be used to avoid such events. The after the fact evaluation of water hammer events should be conducted to an operability criteria and not to the design criteria of the facility.

4.8.4 Capacity Criteria

As briefly discussed in the previous section, the Service Levels are associated with a SSC capacity criteria which provides a certain level of protection to the SSC. This protection level is as discussed below:

Level A Service: Level A Service Limits are those sets of limits which must be satisfied for all Level A Service loadings identified in Table 4.8-2. Meeting these limits insures the SSC will perform its specified service function.

Level B Service: Level B Service Limits are those sets of limits which must be satisfied for all Level B Service loadings identified in Table 4.8-2. Meeting these limits insures the SSC can withstand the applied loadings without damage requiring repair.

Level D Service: Level D Service Limits are those sets of limits which must be satisfied for all Level D Service loadings identified in Table 4.8-2. These limits permit gross general deformations

with some consequential loss of dimensional stability and damage requiring repair, which may require removal of the component or component support from service.

As also discussed in Section 4.8-2 many of the standards selected for use in the design of SSC in Radwaste Facilities are commercial standards. These standards typically do not provide design criteria for the extraordinary natural phenomena and manmade hazard considered in the design of Nuclear Waste processing Facilities. Therefore, the proposed capacity criteria was developed from:

- 1) reviewing "Nuclear Versions" similar to the Code or Standard selected
- 2) reviewing capacity criteria used in the design of existing Radioactive Waste Facilities,
- 3) the criteria capacity specified for the design of Radwaste Facilities in the Advanced Reactor Designs and
- 4) Capacity Criteria currently employed by the DOE in design of Waste Processing Facilities.

The criteria of the latter three items have been reviewed and discussed in previous sections of this report. The resulting capacity criteria is provided in Table 4.8-3.

Table 4.8-1 Suggested Standards for the Design of SSC in Radwaste Facilities				
Component	Design and Construction	Materials	Welding	Inspection and Testing
Structures - Concrete	ACI-318 or ACI 349 ⁽³⁾	ACI-318 or ACI 349	ACI-318 or ACI 349	ACI-318 or ACI 349
Structures-Steel (Hot Rolled)	AISC-ASD or AISC LRFD or AISC N-690 ⁽⁵⁾	ASTM	AWS-D1.1	AISC Standards and AWS Standards
Structures-Steel (Cold Formed)	AISI-CFSDM	ASTM	AWS-D1.3, D9.1	AISC Standards and AWS Standards
Piping & Valves	ANSI/ASME B31.3 ⁽¹⁾	ASME-Sec. II	ASME, Sec. IX	ANSI/ASME B31.3
Atmospheric Tanks	API-650	ASME Sec. II	ASME, Sec. IX	API-650
Tanks [0-103kPa (0-15 psig)]	API-620	ASME Sec. II	ASME, Sec. IX	API-620
Pressure Vessels and Tanks [$>103\text{kPa}$ (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 or ASME Sec. II	ASME, Sec. IX	ASME BPVC Code Section III, Div. 1, Subsection ND
Heat Exchangers	TEMA STD, 7th Edition; ASME BPVC Section VIII Div. 1 or Div. 2	ASTM or ASME Sec. II	ASME, Sec. IX	ASME BPVC Section VIII, Div. 1 or 2
HVAC Systems	SMACNA Stds. ⁽²⁾	ASTM	AWS-D1.1, D1.3, D9.1	SMACNA Stds
Conduit and Cable Trays	IEEE/NEMA Standards	ASTM/NEMA	AWS-D1.1, D1.3, D9.1,	IEEE/NEMA Stds.
Fire Protection Systems	NPFA-13 ⁽³⁾ ; NFPA-14	ASTM	AWS-D1.1, D1.3, D9.1, D10.9	NFPA-13/NFPA

Footnotes for Table 4.8-1:

- (1) Class RW-IIa and RW-IIb, Piping Systems shall be designed as category "M" systems.
- (2) Class RW-IIa and RW-IIb HVAC systems shall use SMACNA "Seismic Restraint Manual Guides for Mechanical Systems".
- (3) Class RW-IIa and RW-IIb Fire Protection Systems shall be designed to NFPA Section 4-14.4.3.
- (4) The classes RW-IIa, RW-IIb, RW-IIc are discussed in Section 5.0

Footnotes for Table 4.8-1 (continued):

- (5) Applicable to structures enclosing or supporting pressurized gaseous waste or liquid waste systems up to the spill height. Also applicable to solid waste facility foundations slab and connected wall or column sections up to a height of 3.048 meters (10.0 feet).
- ASME BPVC Section II required for Pressure Retaining Components.
- Class RW-II/A Structures shall use ACI-349 and/or AISC N-690 as applicable.
- Appropriate load combinations and capacity criteria for component design is as specified in Table 4.8-2.
- 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.

Table 4.8-2 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⁾) External Conduits and Cable Trays	A (Normal)	D + L + To	D + L + To	D + L + To
	B (Severe; Upset)	D + L + Tb D + L + To + E _o D + L + To + W + R D + L + To + F	D + L + Tb D + L + To + E' _o D + L + To + W + R D + L + To + F	D + L + Tb D + L + To + E'' _o D + L + To + W + R D + L + To + F
	D (Abnormal Extreme; Faulted)	D + L + To + W _t D + L + To + V _m D + L + To + A _c D + L + T _a + A _D D + L + T _a + A	N/R	N/R
	A (Normal)	D + L + To	D + L + To	D + L + To
Internal Structures (Concrete, Steel, Component Support Structures ⁽¹⁾) Internal Conduit and Cable Trays	B (Severe, Upset)	D + L + Tb D + L + T _o + E _o D + L + T _o + F	D + L + Tb D + L + T _o + E _o D + L + T _o + F	D + L + Tb D + L + T _o + E _o D + L + T _o + F
	D (Abnormal Extreme; Faulted)	D + L + T _a + A _D D + L + T _a + A	N/R	N/R
	A (Normal)	P _D + D + D _m T _o	P _d + D + D _M T _o	P _D + D + D _m T _o
Pressure Retaining Components ⁽²⁾ (Piping, Valves, Pressure Vessels, Atmosphere, Tanks, 0-103KpA (0-15 psig)) Tanks, Pumps Heat Exchangers) HVAC Systems Fire Protection Systems	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

Table 4.8-2 Design Load Combinations (Continued)

System, Structure, Component (SSC)	Service Levels	SSC Safety Class		
		RW - IIa	RW - IIb	RW - IIc
	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
 L = Live loads
 T_o = Normal Operating Thermal Expansion Loads
 T_b = Upset Thermal Expansion Loads
 T_a = Accident Thermal Loads
 E_o = OBE or 1/2 SSE Seismic Loads
 E'_o = Seismic Loads per Table 5.2 - 1 For RW-IIb Components
 E''_o = Seismic Loads per Table 5.2 - 1 For RW-IIc Components
 W = Wind Load Including Missile Effects
 R = Precipitation Loads (Rain, Snow)
 F = Flood Loadings
 W_t = Tornado Loads Including Missile Effects
 V_m = Malevolent Vehicle Assault Loads
 A_c = Aircraft Crash Loads
 A_D = Design Basis Accident Loads
 A = Other Accident Loads
 P_D = Design Pressure
 P_b = Maximum Upset Pressure
 P_o = Normal Operating Pressure
 P_a = Applicable Accident Pressure
 D_M = Design Mechanical Loads

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 4.8-3 - 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 Stress Design and Plastic Design, Part 1"	Capacity Criteria per "Specification for Structural Steel Buildings Stress Design and Plastic Design, Part 1"	Capacity Criteria per "Specification for Structural Steel Buildings Stress Design and Plastic Design, Part 1"
	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 Stress Design and Plastic Design, Part 2".	N/R	N/R

Table 4.8-3 - SSC Design Capacity Criteria (Continued)				
Code or Standard	Service Level	Capacity Criteria		
		RW-IIa	RW-IIb	RW-IIc
AISC-LFRD	A, B, D	Load Factors and Capacities per "LFRD" Specification for Structural Steel Buildings	N/A	N/A
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.	N/R	N/R
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	2.4 Times B31.3 Design Load Capacities	N/R	N/R
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	1.2 Times Level A capacity criteria	1.2 Times Level A capacity criteria	1.2 Times Level A capacity criteria
	D	1.8 times Level A capacity criteria	N/R	N/R
SMACNA Stds. ⁽¹⁾	A	SMACNA Design Criteria	SMACNA Design Criteria	SMACNA Design Criteria
	B	SMACNA Design Criteria	SMACNA Design Criteria	SMACNA Design Criteria

Table 4.8-3 - SSC Design Capacity Criteria (Continued)				
Code or Standard	Service Level	Capacity Criteria		
		RW-IIa	RW-IIb	RW-IIc
	D	- Duct support members shall meet capacity criteria for AISI-CFSDM or AISC-ASD for Level D Loads - Duct stresses shall be less than the material yield stress and shall be limited to 2/3 critical buckling	N/R	N/R
NFPA-13 ⁽¹⁾	A	NFPA Design Criteria	NFPA Design Criteria	NFPA Design Criteria
	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	- Support members shall meet capacity criteria for AISI-CFSDM or AISC-ASD for Level D Loads - Piping Stresses shall meet the B31.3 Level D Capacity Criteria	N/R	N/R
IEEE/NEMA STDS (Cable Trays/Conduit)	A	IEEE/NEMA Design Criteria for Normal Loads	IEEE/NEMA Design Criteria for Normal Loads	IEEE/NEMA Design Criteria for Normal Loads
	B	IEEE/NEMA Design Criteria for Wind and Seismic Loads	IEEE/NEMA Design Criteria for Wind and Seismic Loads	IEEE/NEMA Design Criteria for Wind and Seismic Loads
	D	- Support members shall meet capacity criteria for AISI-CFSDM or AISC-ASD for Level D Loads - Trays and members shall meet the capacity criteria for AISI-CFSDM for Level D Loads	N/R	N/R
Pumps (API Series STDS)	A	Applicable API Standards for Design Criteria	Applicable API Standards for Design Criteria	Applicable API Standards for Design Criteria
	B	ASME QME-1 [7.5-12]	ASME QME-1 [7.5-12]	ASME QME-1 [7.5-12]
	D	ASME QME-1 [7.5-12]	N/R	N/R

Table 4.8-3 - SSC Design Capacity Criteria (Continued)				
Code or Standard	Service Level	Capacity Criteria		
		RW-IIa	RW-IIb	RW-IIc
APL-620/650 (Tanks)	A	API Design Capacity Criteria	API Design Capacity Criteria	API Design Capacity Criteria
	Level B	Capacity Criteria per ASME-BPVC - Section III, ND-3800, ND-3900 for Level B loads ⁽²⁾ . All other Design per API Criteria.	Capacity Criteria per ASME-BPVC - Section III, ND-3800, ND-3900 for Level B loads ⁽²⁾ . All other Design per API Criteria.	Capacity Criteria per ASME-BPVC - Section III, ND-3800, ND-3900 for Level B loads ⁽²⁾ . All other Design per API Criteria.
	Level D	Capacity Criteria per ASME-BPVC - Section III, NC-3800, NC-3900 Level D Loads ⁽²⁾ . All other Design per API Criteria.	N/R	N/R

Footnotes: Table 4.8-3:

- (1) For Level A and B Loads, the Design Criteria is primarily a "design by rule" approach versus a specific analysis criteria.
- (2) 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.

4.9 Quality Assurance Standards and Requirements

The discussion on Quality Assurance provided for Regulatory Guide 1.142 in Section 3.4 is also applicable to Regulatory Guide 1.143.

5.0 Investigative Study Summary and Recommendation

5.1 General

The results of the reviews conducted in Section 3.0 and 4.0 indicate there is a need to make modifications to the current revisions of Regulatory Guides 1.142 and 1.143.

Section 5.2 provides a general summary of the suggested changes to Regulatory Guides 1.142 and 1.143 while Section 5.3 provides a discussion of the detailed modifications suggested for implementation of the changes suggested in Section 5.2.

5.2 General Summary of Suggested Changes

This section provides a general summary of the suggested changes to Regulatory Guides 1.142 and 1.143. These changes are presented individually for each Regulatory Guide.

5.2.1 Regulatory Guide 1.142

This section provides a summary of the changes which are suggested for Regulatory Guide 1.142. The general approach will be to implement ACI-349 1997 Edition and then stipulate modifications and/or additions to the basic code text which will be required.

- (a) Position 1 in the current Regulatory Guide should remain unchanged. This position includes case-by-case reviews by the USNRC staff of the pressure resisting portions of Mark III and Mark II containments.
- (b) Position 2 in the current Regulatory Guide should be revised to reflect that specific guidance for radiation shields may be obtained from ANSI/ANS 6.4-1997. This is the current ANSI standard for radiation shielding. Reference to withdrawn standard ANSI N101.6 should be deleted.
- (c) Position 3 in the current Regulatory Guide should be deleted. ACI 349-97 now includes special provisions for seismic design in Chapter 21. These new provisions eliminate the need to reference ACI 318 for ductile detailing for earthquake loads.
- (d) Position 4 in the current Regulatory Guide should remain unchanged. This position provides an acceptable method for qualifying inspectors that is not included in ACI 349-97.
- (e) Position 5 in the current Regulatory Guide should remain unchanged. This position increases the frequency of testing for compression tests from that specified in ACI 349-97.
- (f) Position 6a in the current Regulatory Guide should remain essentially unchanged. This same concern exists for temperature loading, T_o , in ACI 349-97. The factor in the discussion to the Regulatory Position will be changed from 1.7 to 1.3, since the 1.3 load factor is what was intended for the load combinations that include this loading.
- (g) Position 6b in the current Regulatory Guide should remain unchanged. This same concern exists for differential pressure loading, P_a , in ACI 349-97.
- (h) Position 6c in the current Regulatory Guide should remain unchanged for loading P_a . This same concern exists for temperature loading, P_a , in ACI 349-97. However, the increases in load factors for E_o in position 6c and 6d should be deleted. The specification of the SSE and OBE in 10CFR50 Appendix S in combination with the load combinations in ACI 349-97 are acceptable for evaluating the earthquake hazard.
- (i) Position 7 in the current Regulatory Guide should be deleted. The definitions of lateral and vertical soil pressure in ACI 349-97 is now consistent with the USNRC position as stated in Regulatory Guide 1.142, Rev. 1, October 1981.
- (j) Position 8 in the current Regulatory Guide should be deleted. ACI 349-97 considers differential settlement consistent with the USNRC position as stated in Regulatory Guide 1.142, Rev. 1, October 1981.
- (k) Position 9 in the current Regulatory Guide should remain unchanged. This same need for the USNRC to evaluate pool dynamic loads on

a case-by-case basis exists with the issuance of ACI 349-97.

- (l) Position 10 in the current Regulatory Guide should remain essentially unchanged. The same restrictions for ductility should remain in place for impact and impulse loads as are in place in Regulatory Guide 1.142, Rev. 1, October 1981. An editorial revision should be made that includes pool dynamic loads with other impact and impulse loads.
- (m) Position 11 in the current Regulatory Guide should remain unchanged. The same restriction for impulsive loading associated with aircraft impact, turbine missiles and explosion should remain in place as are in Regulatory Guide 1.142, Rev. 1, October 1981.
- (n) Position 12 in the current Regulatory Guide should remain unchanged. The endorsement of the criteria of Appendix A should remain in place.
- (o) A new position should be included that defines how to consider loads due to malevolent vehicle assault, aircraft crash and accidental explosion. It is proposed that the pressure pulse from these loads should be taken as W_t . Missiles from these loads should be considered the same as internal missiles.
- (p) A new position should be included that states that hydrodynamic loads associated with earthquake loads should be considered with the same load factor as the earthquake load.
- (q) A new position should be included to clarify the definition of deep flexural members. There are conflicting definitions in ACI 349-97.
- (r) A new position should be included to clarify that when several buildings share seismic resistance; the definitions in ACI 349-97 may not be adequate for their design.
- (s) A new position should be included that identifies modular construction as a unique item which will be reviewed on a case-by-case basis.
- (t) A new position should be included that invokes the requirements of ACI 349-97 and ACI 359-92 for slabs and walls which frame

into concrete containments and participate in resisting accident and seismic loads.

- (u) A new position invoking the requirements of ACI 318-95 Section 11.6 instead of ACI 349-97 Section 11.6 for members subject to torsion and combined shear and torsion should be included. ACI 318-95 includes more state of the art criteria for this loading.
- (v) The latest revision of ANS 6.4 (1997) should be referenced in the standard. In addition, all references to the ANSI/ASME N45.XX.XX series standard should be deleted and replaced by ASME NQA-1

5.2.2 Regulatory Guide 1.143

In the introduction to R.G. 1.143-Rev. 1, 1979 three General Design Criteria as defined in 10 CFR 50 "General Design Criteria, GDC, for Nuclear Power Plant" Appendix A are identified. These are:

Criterion 1	"Quality Standards and Records"
Criterion 2	"Design Basis for Protection Against Natural Phenomena"
Criterion 60	"Control of Release of Radioactive Materials to the Environment"

The existing R.G. 1.143 Rev. 1, 1979 furnishes design guidance acceptable to the NRC staff relating to seismic and quality group classification and quality assurance provisions for radioactive waste management systems, structures 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. It also provides some guidance concerning the buildup of explosive mixtures of hydrogen and oxygen in gaseous radwaste treatment systems.

Although also not addressed in the applicable ANS 55 series of standards, in keeping with GDC2, this proposed revision of R.G. 1.143-Rev. 1, 1979 expands the scope of the existing guide to include natural phenomena hazards other than seismic. These would include extreme wind, tornado and flooding from both external sources such as rivers, streams, lakes and other bodies of water as well as local precipitation. Natural phenomena hazards from volcanic action and lighting might also be considered, but these phenomena are so

site dependent and limited that they are not specifically considered in this generic regulatory guide.

The Purpose of R.G. 1.143 Rev. 1, 1979 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.

The existing revision of the R.G. 1.143 Rev. 1, 1979 guide makes reference to the possible use of portions of the ANS Standards related to radioactive waste. However, since the scope of these ANS 55 Standards provides a much wider scope than R.G. 1.143 only specific applicable sections of these standards are referenced where appropriate in this proposed revision of R.G. 1.143. However, it would also be considered appropriate to consider by reference other current sections of the ANS 55 Standards not referenced by R.G. 1.143 in any future Standard Review Plan revision of Chapter 11.

The proposed revision of R.G. 1.143 Rev. 1, 1979 continues to consider the radwaste system to begin at the interface valve(s) 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 storage tanks, or at the point of storage of packaged solid wastes prior to shipment off-site to a licensed burial ground.

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 the point of interface with other liquid systems, or at the point of recycle back to the secondary systems. 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 sampling systems beyond the first floor 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, or the spent fuel handling and storage systems, or the spent 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 ALARA. 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.

5.2.2.1 General Modifications

This section provides suggested changes which are applicable to all three of the above standards:

- (a) A comprehensive Table (SSC Codes and Standards) should be developed and provided in RG 1.143 which encompasses all SSC associated with the design and construction of solid, liquid and gaseous radioactive waste processing facilities. This table should supersede Table 1 in the existing Regulatory Guide.
- (b) Unmitigated Radiological Release is defined as the radioactive release that would occur at the site boundary, considering the total facility inventory of radioactive materials, without an easement or lessening of the release by any means.
- (c) It is recommended that structures (buildings) housing the Radwaste systems and/or component be designed for natural phenomena and external man-induced hazard loadings. Suggested natural phenomena and external man-induced hazards demand criteria are provided in Table 5.2-1.
- (d) In addition to the hazards given in Table 5.2-1 it is recommended other external man-induced hazards

such as accidental explosions, and possible processing accidents (both radioactive and toxic chemicals) be considered in design of these facilities.

- (e) It is suggested that the capacity criteria of the codes and standards discussed in Section 4.8 be selectively modified to address the design of Classification RW-IIa and RW-IIb SSC for the natural phenomena and external man-induced hazard loads given in Table 5.2-1. It is recommended that Table 5.2.1 be added to the revised R.G.1.142 text as Table 2. In addition, a load combination Table for design purposes as shown in Table 4.8-2 should be added to R.G.1.143 as Table 3.
- (f) The seismic design input should be specified to be consistent with seismic design input at the plant site.

A detailed discussion of these changes and their basis is provided in Section 5.3

5.2.2.2 Classification of Waste Hazard

The following provides a suggested classification (Safety) of SSC for radwaste facilities. Three Safety Classes are established; RW-IIa (High Hazard), RW-IIb, (Hazardous) and RW-IIc (Non-Safety), where RW-IIa is the most stringent class and RW-IIc is the least stringent. These classifications are developed primarily for natural phenomena and external man-induced hazard design. This safety classification is applied to SSC as follows:

- (1) For a given structure housing radwaste processing systems or components, if the total design basis unmitigated release (considering the maximum inventory) at the boundary of the unprotected area is greater than 500 milli-rem per year, or the maximum unmitigated exposure to site personnel within the protected area is greater than 5.0 rem per year, then the external structures shall be classified as RW-IIa.
- (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 then the external structure shall be classified as RW IIb.

- (3) Any systems or components in a RW-IIa facility {as defined in (1)} which store process or handle radioactive waste in excess of the A_1 quantities given in 10 CFR71, Appendix A, shall be classified as RW-IIa. These systems or components which process radioactive waste in excess of the A_2 quantities but less than the A_1 quantities given in 10CFR71, Appendix A, shall be classified as RW-IIb. All other components shall be classified as RW-IIc. This classification and the associated design criteria may be modified for specific radioactive components as discussed in the subsections) 5.2.2.3-5.2.2.5) which follow.
- (4) Any systems or components in a RW-IIb structure {as defined in [2]} which store or process specified radioactive waste in excess of the A_1 quantities given in 10CFR71 Appendix A shall be classified as RW-IIb. All other systems components shall be classified as RW-IIc.

The intent of the unprotected area boundary is shown in Figure 5.2-1. A flowchart of the Safety Classification Process is shown in Figure 5.2-2.

The radiological release criteria (500 mrem at the unprotected area boundary and 5 rem to facility personnel within the protected boundary) was selected to be consistent with the criteria of 10CFR20. This criteria is discussed in depth in Section 4.7.3.2.

5.2.2.3 Suggested Modifications for Gaseous Waste Management Facilities

The following additional changes (suggested by ANS 55.6-1993) for the liquid radwaste treatment system, including the steam generator blowdown system, downstream of the outermost containment isolation valve:

- (a) materials for pressure-retaining components 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 may be provided in lieu of certified material test reports.

- (b) All tanks located outside reactor containment and containing radioactive materials in liquids should be designed to prevent uncontrolled releases of radioactive materials due to spillage in buildings or from outdoor tanks.
- (c) All tank overflows, drains, and sample lines should be routed to the liquid radwaste treatment system.
- (d) Indoor tanks should have curbs or elevated thresholds with floor drains routed to the liquid radwaste treatment system.
- (e) The design should include provisions to prevent leakage from entering unmonitored and nonradioactive systems and ductwork in the area.
- (f) 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.

5.2.2.4 Suggested Modifications for Liquid Waste Management Facilities

The following is a suggested modification based on ANS 55.4-1993:

Materials for pressure-retaining components 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 may be provide in lieu of certified material test reports.

5.2.2.5 Suggested Modifications for Solid Waste Management Facilities

The following is a suggested modifications based on ANS55.1-1993:

Materials for pressure-retaining components 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 occurrence. Manufacturers' material certificates of compliance may be provided in lieu of certified materials test reports.

Table 5.2-1 Natural Phenomena and 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 ⁽⁵⁾ or UBC 97 Category 2	ASCE 7-95 Category II ⁽⁵⁾ , UBC-97 Category 4
Wind	ASCE, 7-95 Category III ⁽⁵⁾	ASCE 7-95, Category III ⁽⁵⁾	ASCE 7-95, Category II ⁽⁵⁾
Tornado	ANS-2.3-at a Probability of 1×10^{-5} /yr or three fifths (3/5) of Regulatory Guide 1.76 Table 1 Criteria	ASCE 7-95, Category III ⁽⁵⁾	ASCE 7-95, Category II ⁽⁵⁾
Tornado Missile from SRP Section 3.5	a. 34 kg (75 lbs), 76.2 mm (3 in.) nominal diameter sch. 40 pipe. maximum velocity 0.4x max. wind speed horizontal and 0.28 times max. wind speed vertical direction. ⁽²⁾ b. Automobile wt. 1814 kg (4000 lbs) with frontal area of 1.858 m ² (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 10.668 m (35 ft) above grade ⁽³⁾ .	Not Required	Not Required
Flood	Regulatory Guide 1.59 one half (1/2) of the PMF ⁽¹⁾ .	ASCE 7-95, Category III ⁽⁵⁾	ASCE 7-95, Category II ⁽⁵⁾
Precipitation ⁽⁴⁾ (Rain, Snow)	ANS 2.6 at Probability of 1×10^{-3} /yr or Regulatory Guide 1.59, one-half (1/2) precipitation specific for the PMF ⁽¹⁾	ASCE 7-95 Category III ⁽⁵⁾	ASCE 7-95, Category II ⁽⁵⁾
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 USNRC Reg. Guide 1.91	Not Required	Not Required
Malevolent Vehicle Assault	USNRC Reg. Guide 5.4 or Plant Specific Definition	Not Required	Not Required
Small Aircraft Crash	Plant Specific Definition	Not Required	Not Required

Footnotes for Table 5.2-1:

- (1) PMF = Probable Maximum Flood.
- (2) Penetrating type missile.
- (3) Impact type missile.
- (4) Resistance to lightening strike should also be included in the design.
- (5) The definition of the ASCE 7-95 classes is provided in Table 5.2-2.

Table 5.2-2 ASCE 7-95 Classification of Buildings and Other Structures for Wind, Snow, and Earthquake Loads (Reproduced from Table 1-1 in ASCE 7-95)

Nature of Occupancy	Category
Buildings and other structures that represent a low hazard to human life in the event of failure including, but not limited to: • Agricultural facilities • Certain temporary facilities • Minor storage facilities All buildings and other structures except those listed in Categories I, III, and IV	I
All other buildings and facilities not discussed in Categories I, III and IV	II
Buildings and other structures that represent a substantial hazard to human life in the event of failure including, but not limited to: • Buildings and other structures where more than 300 people congregate in one area • Buildings and other structures with elementary school, secondary school, or day-care facilities with capacity greater than 250 • buildings and other structures with a capacity greater than 500 for colleges or adult education facilities • Jails and detention facilities • Power generating stations and other public utility facilities not included in Category IV • Buildings and other structures containing sufficient quantities of toxic or explosive substances to be dangerous to the public if released.	III
Buildings and other structures designated as essential facilities including, but not limited to: • Hospitals and other health-care facilities having surgery or emergency treatment facilities. • Fire, rescue and police stations and emergency vehicle garages. • Designated earthquake, hurricane, or other emergency shelters. • Communications centers and other facilities required for emergency response. • Power generating stations and other public utility facilities required in an emergency. • Buildings and other structures having critical national defense functions.	IV

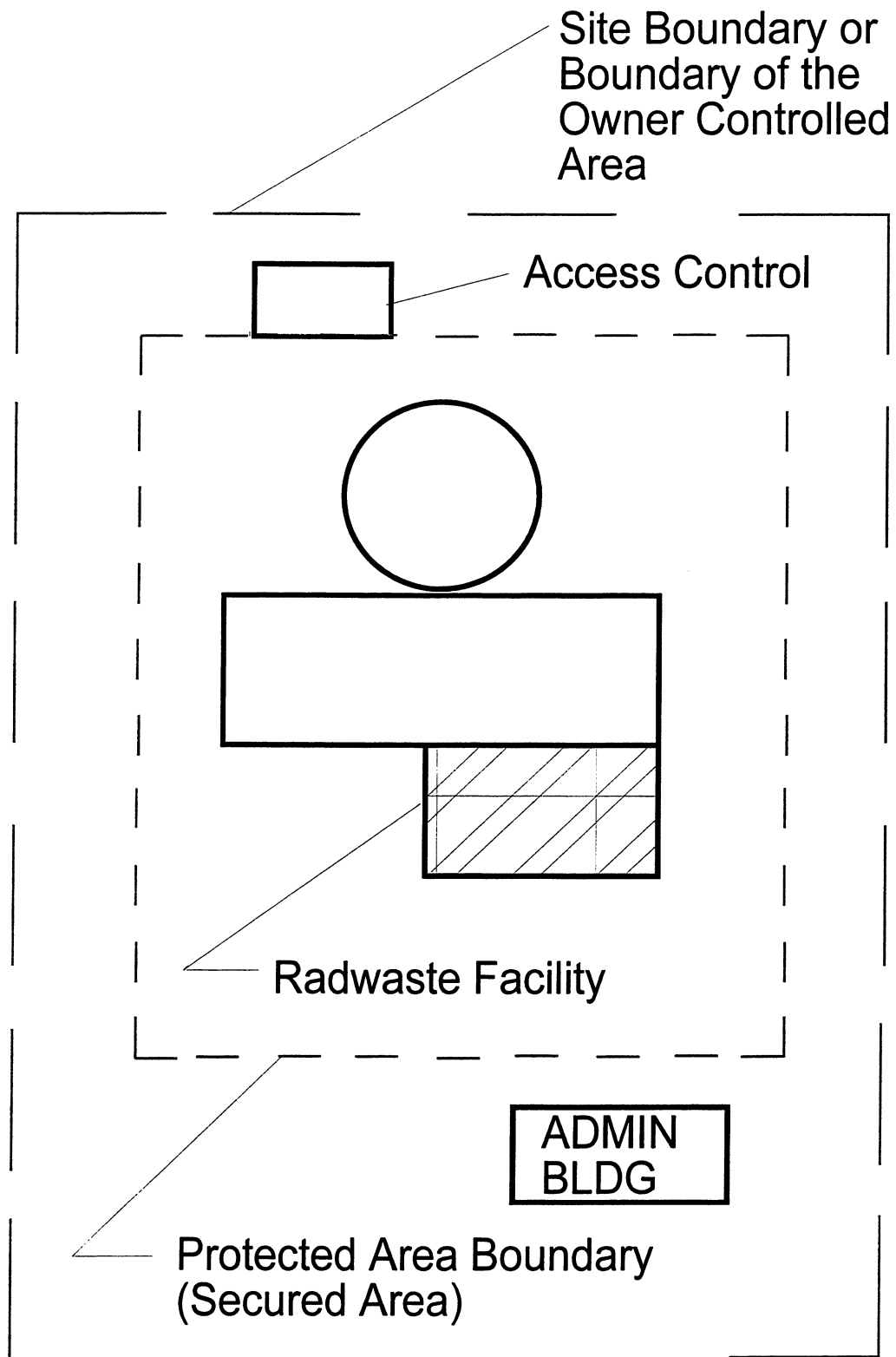


Figure 5.2-1 Schematic of Protected Area Boundary

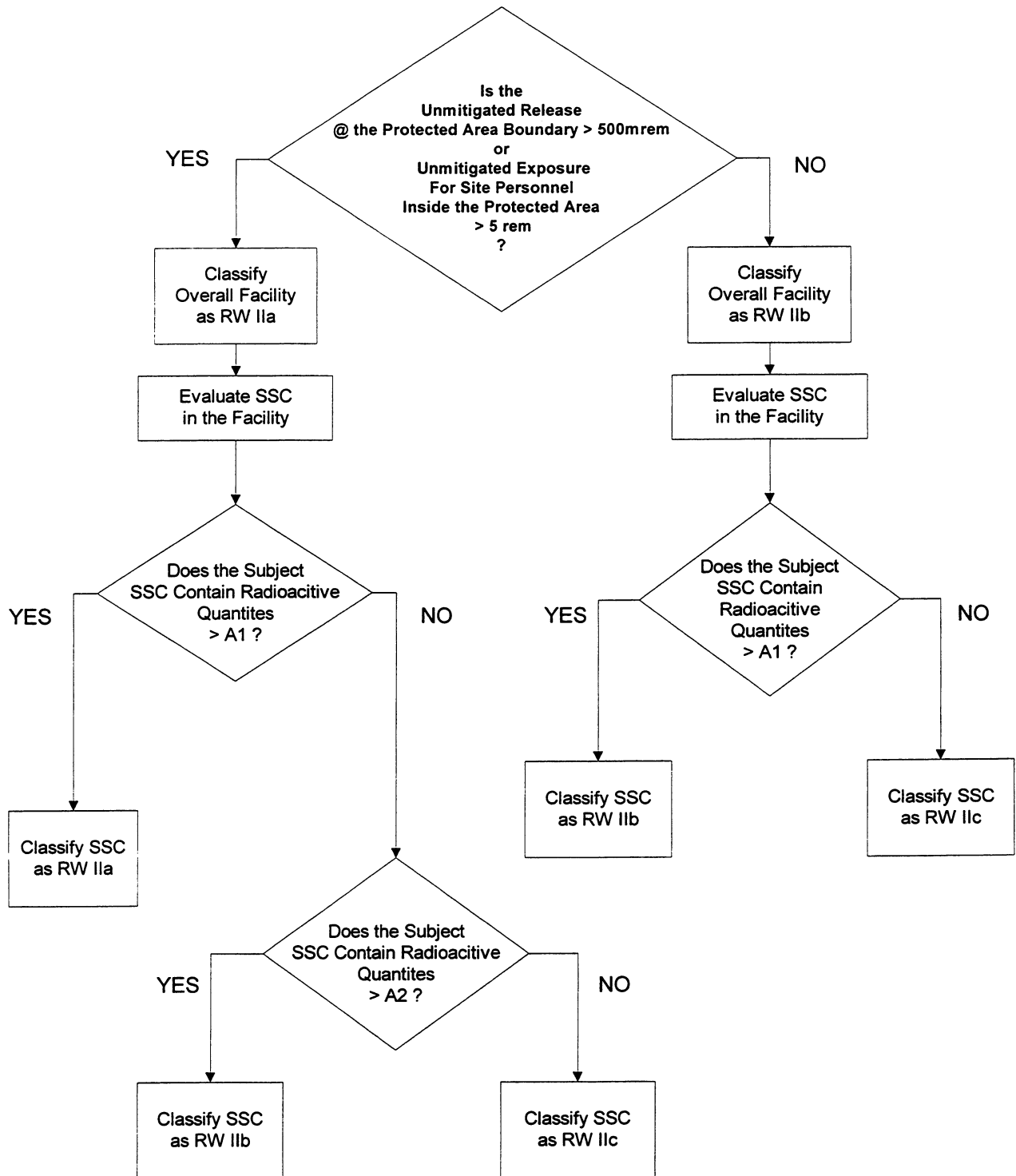


Figure 5.2-2 - Safety Classification Flowchart

5.3 Specific Suggested Changes to Regulatory Guides

5.3.1 Regulatory Guide 1.142

This section provides the details and specific recommendations on the suggested changes to Regulatory Guide 1.142

(a) ITEM 1

Replace all references to ANSI/ACI-349-76 with ACI-349-97. Update other code references to the current (1997) editions.

(b) ITEM 2

In the general discussion, the following note should be added:

There are sections of ACI 318-1995 that contain more current criteria on certain aspects of design. When this is the case, ACI 318-1995 is specified for use.

(c) ITEM 3

Replace Position 3 with the following:

Position 3. In complex structural systems where seismic resistance is shared by several buildings, the definitions of ACI 349-97 may not be adequate for nuclear safety related structures. The position alerts the designer to consider whether or not structural components are acting as parts of flexural frames, etc.

(d) ITEM 4

Delete the last paragraph of Position 6.

(e) ITEM 5

Replace Position 7 with the following:

Position 7. Loads due to malevolent vehicle assault, aircraft impact and accidental explosion shall be considered with the same load factors as tornado wind W_t .

(f) ITEM 6

Replace Position 8 with the following:

Position 8. Hydrodynamic loads associated with seismic loads shall be considered with the same load

factors of the seismic load. Other hydrodynamic loads shall be taken as Y_j .

(g) ITEM 7

Add the following four positions:

Position 13. In order to provide consistency and to ensure that the Code requirements for shear are properly applied the definition for deep flexural members should be as provided in 11.8 of the Code.

Position 14. This position is intended as a guide until the Code addresses composite construction.

Position 15. This position increases the requirements for slabs and walls that frame into concrete containments and participate in resisting accident and seismic loads.

Position 16. This position invokes the more current state of the art requirements contained in ACI 318-95 for members that are subject to torsion and combined shear and torsion.

(h) ITEM 8

Replace Regulatory Position 3 with the following:

3. Where structural components, normally defined as walls, slabs, and foundations, actually exhibit a structural response consistent with the response of structural frames, such components shall conform to the requirements of ACI 349-97 Chapters 10, 11, and 21 in addition to Chapters 13, 14 and 15 as appropriate.

(i) ITEM 9

Delete Item (d) of Regulatory Position 6.

(j) ITEM 10

Replace Regulatory Position 7 with the following:

7. Loads due to malevolent vehicle assault, aircraft impact, and accidental explosion shall be taken as W_t in the load combination (5).

(k) ITEM 11

Replace Regulatory Position 8 with the following:

8. Hydrodynamic loads associated with seismic loads (i.e. the impulsive and sloshing loads for fluids in tanks) shall be considered as E_{ss} in load case (4) and

(8), and E_o in load case (2), (7) and (10). All other hydrodynamic loads shall be taken as Y_j , in load combination (7) & (8).

(l) ITEM 12

Add the following four USNRC Positions:

13. The Code has inconsistent definitions of deep flexural members in 10.7.1 and 11.8.1. In order to be consistent the definition in 11.8.1 should be used.

14. The design of composite members used in modular construction shall conform to the intent of Code provisions of 10.14 and Chapter 17. Until the code contains more specific requirements for modular construction, future designs shall be evaluated on a case-by case basis.

15. Slabs and walls, which frame into concrete containments and will participate in resisting accident and seismic loads, shall meet the requirements of both ACI 349 and ACI 359.

16. Members that are subject to torsion and combined shear and torsion should be evaluated to the requirements of ACI 318-95, Section 11.6 instead of the requirements of ACI 349-97, Section 11.6.

(m) ITEM 13

Delete all References to ANSI N45.xx and replace with:

ASME NQA-1-1997 Edition, "Quality Assurance Requirements for Nuclear Facility Applications."

(n) ITEM 14

Update the address of the ACI and ASME as follows:

1. Copies may be obtained from the American Concrete Institute, P.O. Box 9094, Farmington Hills, Michigan 48333

2. Copies may be obtained from the American Society of Mechanical Engineers, Three Park Avenue, New York, NY 10016-5990

(o) ITEM 15

Modify Reference to ANSI/ANS 6.4 as follows:

ANSI/ANS 6.4-1997, "Guides on the Nuclear Analysis and Design of Concrete Radiation Shielding for Nuclear Power Plants."

5.3.2 Regulatory Guide 1.143

This section provides more details and specific recommendations on the changes suggested to Regulatory Guide 1.143.

(a) ITEM 1

1. Add "extreme winds, tornados, flooding, etc.", and "external man-induced and design basis accident" as cases to be evaluated.

2. Change "seismic" throughout the document to "Natural Phenomena and internal and external man made hazards".

3. Change "Systems" through out the document to "System Structures and Components".

(b) ITEM 2

Replace the second paragraph of "B. Discussion" with the following:

ANSI/ANS Standards 55.1-1992, "Solid Radioactive Waste Processing System for Light Water Cooled Reactor Plants", 55.4-1993, "Gaseous Radioactive Waste Processing Systems for Light Water Plants", and 55.6-1993, "Liquid Radioactive Waste Processing Systems for Light Water Reactor Plants" have been reviewed for potential application to this guide. The ANSI/ANS Standards provide a broader scope of guidance, including that provided in NUREG-0800, Section 11.0, plus that contained in this Regulatory Guide. Where appropriate guidance from the ANSI/ANS Standards have been incorporated by reference.

(c) ITEM 3

Delete Regulatory Position 1.1.4.

(d) ITEM 4

Replace Regulatory Position 1.2 with the following:

1.2 All SSC located outside of the reactor containment and containing radioactive materials in liquid form should be classified as described in Regulatory Position 5 and designed to the criteria put forth in Regulatory Position 6. In addition, any such component should be designed to prevent uncontrolled releases of radioactive materials due to spillage in building or from outdoor components. The following design features should be included for

such components and should meet the criteria contained in Section 5.2, 5.3 and 5.4 of ANSI/ANS 55.1-1191.

(e) ITEM 5

Replace Regulatory Position 2.1.3 with the following:

2.1.3 Those 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 to the criteria of Regulatory Position 6.

(f) ITEM 6

Add the following new Regulatory Position 5:

5.0 CLASSIFICATION OF WASTE SYSTEMS FOR DESIGN PURPOSES

The following provides the classification (Safety) of SSC for radwaste facilities. Three Safety Classes are established; RW-IIa (High Hazard), RW-IIb (Hazardous) and RW-IIc (Non-Safety), where RW-IIa is the most stringent class and RW-IIc is the least stringent. These classifications are developed primarily for natural phenomena and man-induced hazard design. This safety classification is applied to SSC as follows:

- (a) For a given structure housing radwaste processing systems or components, if the total design basis unmitigated 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 5.0 rem per year, then the external structures shall be classified as RW-IIa.
- (b) 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, then the external structure shall be classified as RE-IIb.
- (c) Any systems or components in a RW-IIa facility {as defined in (a)} which store, process, or handle

radioactive waste in excess of the A_1 quantities given in 10 CFR71 Appendix A shall be classified as RW-IIa. These systems or components which process radioactive waste in excess of the A_2 quantities but less than the A_1 quantities given in 10CRF71 Appendix A shall be classified as RW-IIb. All other components shall be classified as RW-IIc. This classification may be modified for specific radwaste components as discussed in the subsections which follows.

- (d) Any systems or components in a RW-IIb structure {as defined in (b)} which store or process specified radioactive waste in excess of the A_1 quantities given in 10CRF71 Appendix A shall be classified in RW-IIb. All other systems components shall be classified as RW-IIc.

The intent of the unprotected area boundary is shown in Figure 1. A flowchart of the Safety Classification Process is shown in Figure 2.

(g) ITEM 7

Renumber Regulatory Position 5 as Regulatory Position 6.

(h) ITEM 8

Delete the old Positions 5.1 and 5.2 and replace with the following:

6.1 General Design Criteria - Evaluation of Solid, Liquid, and Gaseous radwaste systems, structures and components described in Regulatory Positions 1, 2, and 3 for natural phenomena and internal and external man-induced hazards shall be as put forth in this position.

- (a) The natural phenomena and internal and external man-induced hazards demand definition is as given in Table 2
- (b) The natural phenomena and internal and external man-induced hazards design load combinations are as given in Table 3
- (c) The natural phenomena and internal and external man-induced hazards shall meet capacity criteria is as defined in Table 4

The acceptability evaluation shall 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 shall be classified RW-IIa and shall be designed to the criteria of Tables 1, 2, 3, and 4.

(i) ITEM 9

Renumber old Position 5.3 as "6.2.2".

(j) ITEM 10

Renumber old Position 6 as Position 7.

(k) ITEM 11

Delete references and old references to this NUREG/CR.

(l) ITEM 12

Replace Table 1 with the Table 1 on the following page.

(m) ITEM 13

Add new Tables 2, 3, and 4 shown on the pages which follow Table 1.

(n) ITEM 14

Add new Figures 1 and 2 shown on the pages which follow Table 4.

Table 1 - Codes and Standards for the Design of SSC in Radwaste Facilities				
Component	Design and Construction	Materials	Welding	Inspection and Testing
Structures - Concrete	ACI-318 or ACI 349 ⁽⁵⁾	ACI-318 or ACI 349	ACI-318 or ACI 349	ACI-318 or ACI 349
Structures-Steel (Hot Rolled)	AISC-ASD or AISC LRFD or AISC N-690 ⁽⁵⁾	ASTM	AWS-D1.1	AISC Standards and AWS Standards
Structures-Steel (Cold Formed)	AISI-CFSDM	ASTM	AWS-D1.3, D9.1	AISC Standards and AWS Standards
Piping & Valves	ANSI/ASME B31.3 ⁽¹⁾	ASME-Sec. II	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 or ASME Sec. II	ASME, Sec. IX	ASME BPVC Code Section III, Class 3 ⁽⁹⁾
Heat Exchangers	TEMA STD, 7th Edition; ASME BPVC Section VIII Div. 1 or Div. 2	ASTM or ASME Sec. II	ASME, Sec. IX	ASME Section VIII, Div. 1 or 2
HVAC Systems	SMACNA Stds. ⁽²⁾	ASTM	AWS-D1.1, D1.3, D9.1	SMACNA Stds
Conduit and Cable Trays	IEEE/NEMA Standards	ASTM/NEMA	AWS-D1.1, D1.3, D9.1,	IEEE/NEMA Stds.
Fire Protection Systems	NPFA-13 ⁽³⁾ ; NFPA-14	ASTM	AWS-D1.1, D1.3, D9.1, D10.9	NFPA-13/NFPA

Footnotes for Table 1:

- (1) Class RW-IIa and RW-IIb, Piping Systems shall be designed as category "M" systems.
- (2) Class RW-IIa and RW-IIb HVAC systems shall use SMACNA "Seismic Restraint Manual Guides for Mechanical Systems".
- (3) Class RW-IIa and RW-IIb Fire Protection Systems shall be designed to NFPA Section 4-14.4.3.
- (4) The classes RW-IIa, RW-IIb, RW-IIc are discussed in Section 5.0
- (5) 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 10.0 feet.
- (6) ASME BPVC Section II required for Pressure Retaining Components.
- (7) Class RW-IIa Structures shall use ACI-349 and/or AISC N-690 as applicable.
- (8) Appropriate load combinations and capacity criteria for component design is as specified in Table 4.8-2.
- (9) 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.

Table 2 - Natural Phenomena and External Man - Induce 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 or UBC 97 Category 2	ASCE 7-95 Category II, UBC-97 Category 4
Wind	ASCE, 7-95 Category III ⁽⁵⁾	ASCE 7-95, Category III	ASCE 7-95, Category II
Tornado	ANS-2.3-at a Probability of 1×10^{-5} /yr or three fifths (3/5) of Regulatory Guide 1.76 Table 1 Criteria	ASCE 7-95, Category III	ASCE 7-95, Category II
Tornado Missile from SRP Section 3.5	A. 75 lbs, 3 in. nominal diameter sch. 40 pipe. maximum velocity 0.4x max. wind speed horizontal and 0.28 times max. wind speed vertical direction. ⁽²⁾ 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 ⁽³⁾ .	Not Required	Not Required
Flood	Regulatory Guide 1.59 one half (1/2) of the PMF ⁽¹⁾ .	ASCE 7-95, Category III	ASCE 7-95, Category II
Precipitation ⁽⁴⁾ (Rain, Snow)	ANS 2.6 at Probability of 1×10^{-3} /yr or Regulatory Guide 1.59, one-half (1/2) precipitation specific for the PMF ⁽¹⁾	ASCE 7-95 Category III	ASCE 7-95, Category II
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 USNRC Reg. Guide 1.91	Not Required	Not Required
Malevolent Vehicle Assault	Reg. 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) PMF = Probable Maximum Flood.
- (2) Penetrating type missile.
- (3) Impact type missile.
- (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)	$D + L + T_b$ $D + L + T_o + E_o$ $D + L + T_o + W + R$ $D + L + T_o + F$	$D + L + T_b$ $D + L + T_o + E'_o$ $D + L + T_o + W + R$ $D + L + T_o + F$	$D + L + T_b$ $D + L + T_o + E''_o$ $D + L + T_o + W + R$ $D + L + T_o + F$
	D (Abnormal Extreme; Faulted)	$D + L + T_o + W_t$ $D + L + T_o + V_m$ $D + L + T_o + A_c$ $D + L + T_a + A_D$ $D + L + T_a + A$	N/R	N/R
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)	$D + L + T_b$ $D + L + T_o + E_o$ $D + L + T_o + F$	$D + L + T_b$ $D + L + T_o + E_o$ $D + L + T_o + F$	$D + L + T_b$ $D + L + T_o + E_o$ $D + L + T_o + F$
	D (Abnormal Extreme; Faulted)	$D + L + T_a + A_D$ $D + L + T_a + A$	N/R	N/R
Internal Conduit and Cable Trays	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
L	=	Live loads
T_o	=	Normal Operating Thermal Expansion Loads
T_b	=	Upset Thermal Expansion Loads
T_a	=	Accident Thermal Loads
E_o	=	OBE or ½ SSE Seismic Loads
E'_o	=	Seismic Loads per Table 5.2 -1 For RW-IIb Components
E''_o	=	Seismic Loads per Table 5.2 -1 For RW-IIc Components

W	=	Wind Load Including Missile Effects
R	=	Precipitation Loads (Rain, Snow)
F	=	Flood Loadings
W_t	=	Tornado Loads Including Missile Effects
V_m	=	Malevolent Vehicle Assault Loads
A_c	=	Aircraft Crash Loads
A_D	=	Design Basis Accident Loads
A	=	Other Accident Loads
P_D	=	Design Pressure
P_b	=	Maximum Upset Pressure
P_o	=	Normal Operating Pressure
P_a	=	Applicable Accident Pressure
D_M	=	Design Mechanical Loads

Footnotes:

(1) Components 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 1.143	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 Stress Design and Plastic Design, Part 1"	Capacity Criteria per "Specification for Structural Steel Buildings Stress Design and Plastic Design, Part 1"	Capacity Criteria per "Specification for Structural Steel Buildings Stress Design and Plastic Design, Part 1"
	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 Stress Design and Plastic Design, Part 2".	N/R	N/R

Table 4 - SSC Design Capacity Criteria (continued)				
Code or Standard	Service Level	Capacity Criteria		
		RW-IIa	RW-IIb	RW-IIc
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.	N/R	N/R
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	N/R	N/R
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	N/R	N/R
SMACNA Stds. ⁽¹⁾	A	SMACNA Design Criteria	SMACNA Design Criteria	SMACNA Design Criteria
	B	SMACNA Design Criteria	SMACNA Design Criteria	SMACNA Design Criteria

Table 4 - SSC Design Capacity Criteria (continued)

Code or Standard	Service Level	Capacity Criteria		
		RW-IIa	RW-IIb	RW-IIc
	D	(1) Duct support members shall meet capacity criteria for AISI-DFSDM or AISC-ASD for Level D Loads (2) Ducting stresses shall be less than the material yield stress and shall be limited to 2/3 critical buckling	N/R	N/R
NFPA-13 ⁽¹⁾	A	NFPA Design Criteria	NFPA Design Criteria	NFPA Design Criteria
	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) Supports members shall meet capacity criteria for AISI-CFSDM or AISC-ASD for Level D Loads (4) Piping Stresses shall meet the B31.3 Level D Capacity Criteria	N/R	N/R
IEEE/NEMA STDS (Cable Trays/Conduit)	A	IEEE/NEMA Design Criteria for Normal Loads	IEEE/NEMA Design Criteria for Normal Loads	IEEE/NEMA Design Criteria for Normal Loads
	B	IEEE/NEMA Design Criteria for Wind and Seismic Loads	IEEE/NEMA Design Criteria for Wind and Seismic Loads	IEEE/NEMA Design Criteria for Wind and Seismic Loads

Table 4 - SSC Design Capacity Criteria (continued)				
Code or Standard	Service Level	Capacity Criteria		
		RW-IIa	RW-IIb	RW-IIc
	D	(5) Support members shall meet capacity criteria for AISI-CFSDM or AISC-ASD for Level D Load (6) Trays and members shall meet the capacity criteria for AISI-CFSDM for Level D Loads	N/R	N/R
Pumps (API Series STDS)	A	Applicable API Standards for Design Criteria	Applicable API Standards for Design Criteria	Applicable API Standards for Design Criteria
	B	ASME QME-1 [xx]	ASME QME-1 {xx}	ASME QME-1 [xx]
	D	ASME QME-1 [xx]	N/R	N/R
API-620/650 (Tanks)	A	API Design Capacity Criteria	API Design Capacity Criteria	API Design Capacity Criteria
	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.
	D	Capacity Criteria per ASME-BPVC - Section III, NC-3800, NC-3900 Level D Loads. All other Design per API Criteria.	N/R	N/R

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.

N/R = Not Required

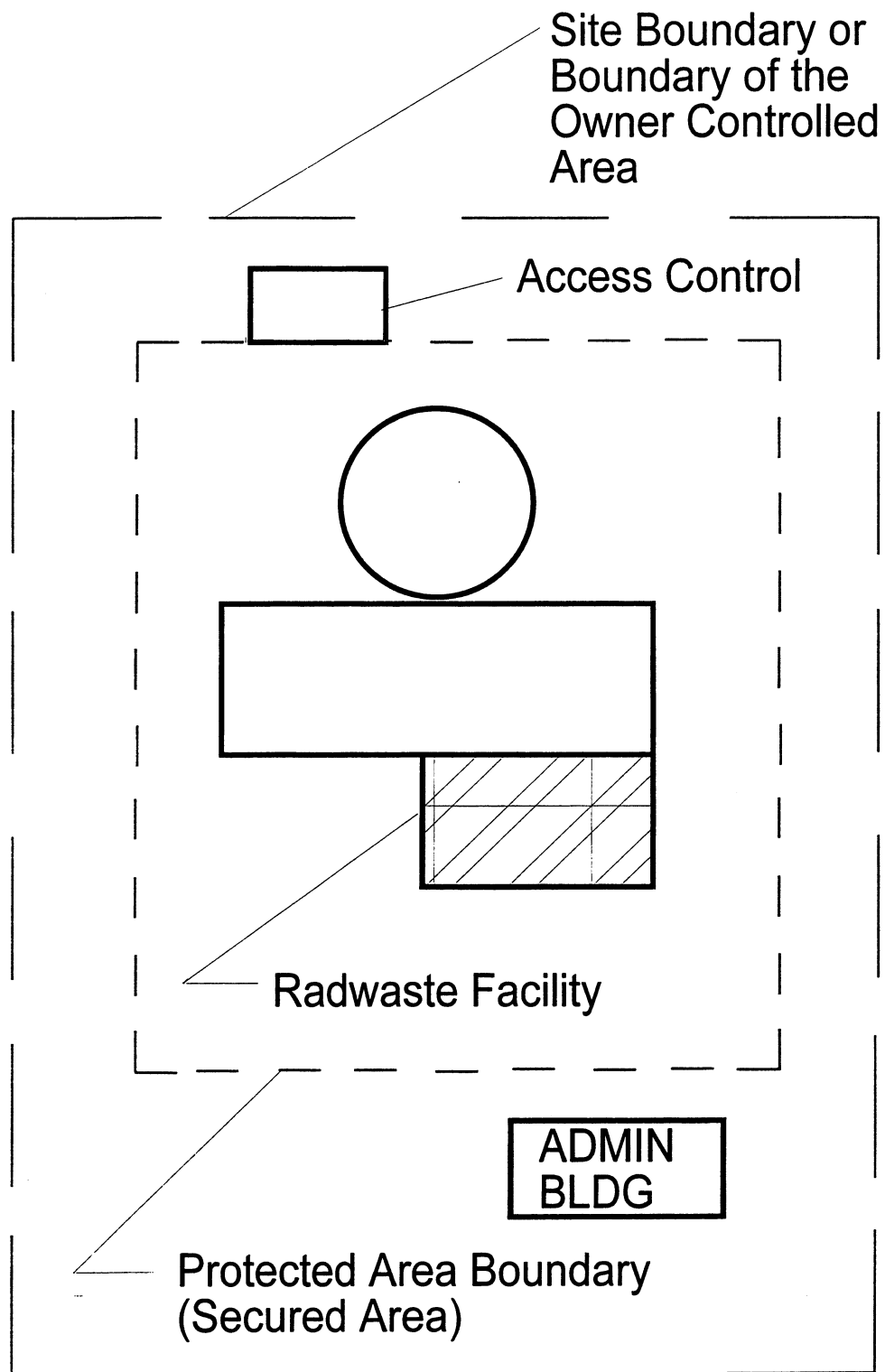


Figure 1 - Schematic of Definition of the Protected Area Boundary

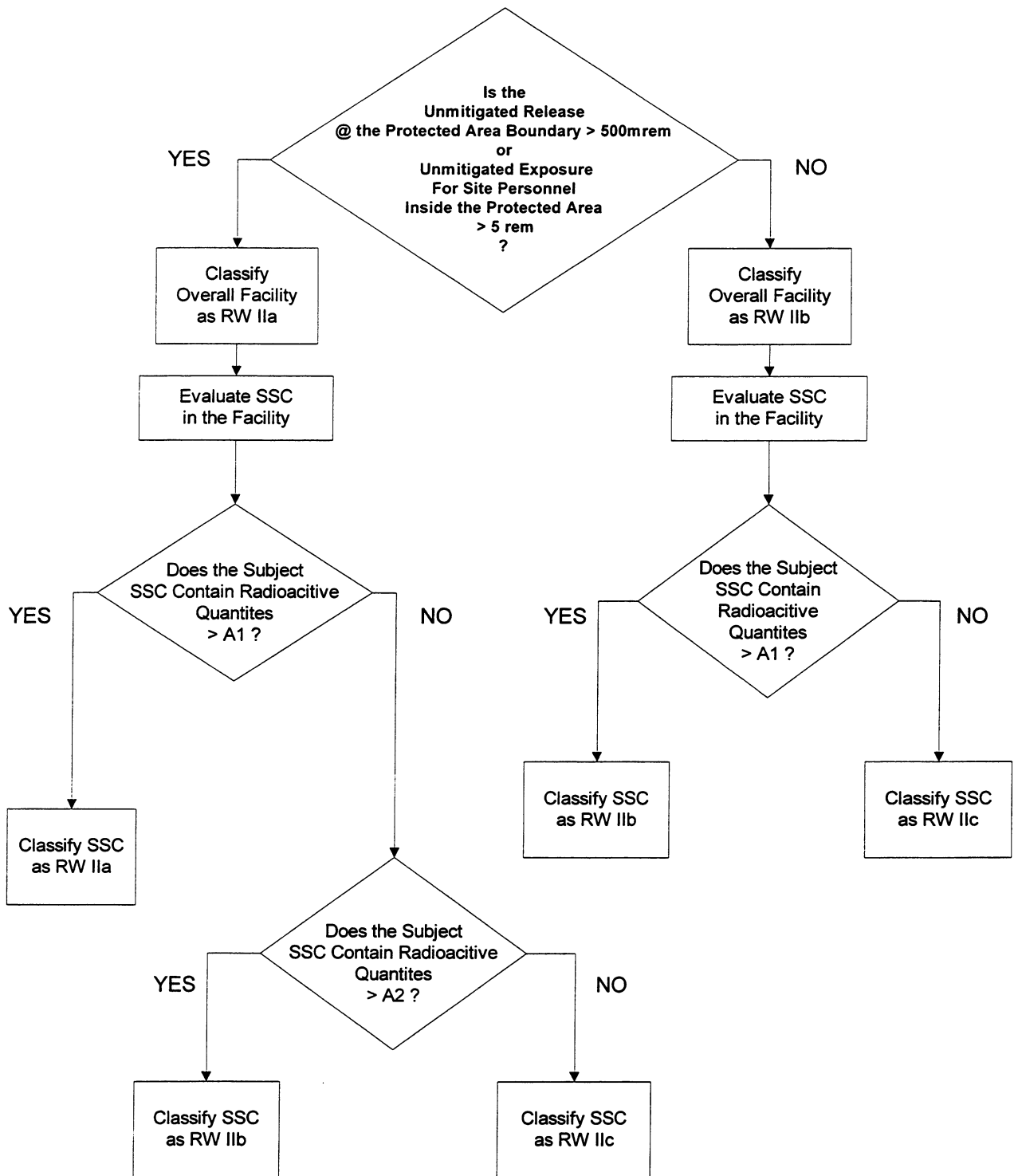


Figure 2 - Flowchart of Safety Classification Process

6.0 Summary and Closure

6.1 Recommendation for Implementation of the Suggested Changes

Stevenson and Associates considered several possible methods by which the conclusions of this report could be implemented. Among the methods considered were:

- (a) via the issuance of new Federal Regulations
- (b) incorporation into NUREG-0800 (the Standard Review Plan)
- (c) issuance of revisions to the existing Regulatory Guides (1.142 and 1.143)

After considering the various possibilities it was decided to recommend that these changes be implemented as revisions to the existing Regulatory Guides. The basis for this recommendation is discussed in the following paragraphs.

The recommended changes provide a significant amount of detailed design and evaluation requirements and criteria. Many of these requirements require interpretation and judgement both on the part of the user and the Regulatory oversight organization. Considering these factors, it is judged that it would be difficult to implement these requirements as Federal Regulations. Significant additional work and investigation, greatly beyond the scope of this program would be required to draft comprehensive Regulations to implement the recommendations of this report.

The update of the Standard Review Plan (NUREG-0800) is being conducted as an ongoing effort by the USNRC under a separate program. The results of this program could be considered by that program for incorporation into NUREG-0800. The authors believe that until that program is completed an appropriate incorporation of the results of this program would be very difficult. Also, the time frame for issuance of the updated Standard Review Plan is not clearly defined at this time. The authors felt by issuing the results of this program as Regulatory Guides would allow industry to immediately begin to use the criteria developed via this program. This would provide immediate benefits from this research program.

Traditionally the information investigated under this program has been published as Regulatory Guides. It was believed that the user community would be more familiar with and better able to apply the results of this program if the same format was applied.

6.2 Summary of the Results of the Regulatory Guide 1.142 Study

The current revision of Regulatory Guide 1.142 is based on the 1976 Edition of the ACI-349 Concrete Code. The current edition of the ACI-349 code is the 1997 Edition. Detailed comparisons were conducted between each revision of the code to clearly identify the changes which have occurred in the code. Secondly, the commitments on design of Concrete Structures in the currently licensed Advance Light Water Reactors were reviewed and summarized. Finally, a comparison of the current licensing commitments for concrete design of several operating reactors to the requirements of ACI-349-1997 were made. Other issues were also investigated related to the design of concrete structures including ductile detailing, special applied loadings (Hydrodynamic, Malevolent Vehicle Assault, etc.), related codes and standards and various regulatory issues. The results of these studies were reviewed in conjunction with the requirements currently in Regulatory Guide 1.142 and suggested revisions to Regulatory Guide proposed. The purpose of the proposed changes is to make Regulatory Guide 1.143 consistent with ACI-349-1997 while minimizing any impact these proposed changes any have on the operating reactors and the recently licensing Advanced Light Water Reactors. Significant effort was made to insure consistency between the proposed Regulatory Guide 1.143, ACI-349-1997 and the licensing commitment of the Advanced Light Water Reactors.

6.3 Summary of the Results of the Regulatory Guide 1.143 Study.

Since the issuance of Regulatory Guide 1.143, advances in radioactive waste management and treatment have occurred. These advances have been in the technical area and in the understanding of the risk and requirements for radioactive waste management. Stevenson and Associates conducted a series of investigations in several areas to evaluate the changes needed to Regulatory Guide 1.143 to provide a current comprehensive document on the Structural Design of Radioactive Waste Management facilities. These studies included significant consideration of the expansive effort in the design of such facilities which has been conducted by the DOE in this arena. The results of these studies are suggested changes to Regulatory Guide 1.143 which provides a comprehensive

document on the Safety Classification, hazards design criteria and overall structural design requirements for Gas, Liquid, and Solid Radioactive Waste Management Facilities.

6.4 Closure

The recommended changes in Regulatory Guide 1.142 and 1.143 provided in this report respectively consider the applicable advancements made since their last revision (1981 and 1979 respectively). The recommendations consider the two major revisions made to ACI 349 since the last issue of the Regulatory Guides, the recently advanced reactor design concepts, and the standards development effort for the design of radioactive waste management systems performed by the Department of Energy (DOE). The recommendations also considered the numerous changes to the Code of Federal Regulations since the issuance of Regulatory Guide 1.142 and 1.143.

7.0 References

Provided in this section is the complete listing of all reference documents reviewed as part of this program. These references are segmented by Advanced Reactor Design, USNRC, Operating Commercial Reactor Reference Documents, DOE Related Reference Documents and miscellaneous References. This was done to allow for the easy addition of reference documents as they became available during the course of the program.

7.1 Advanced Reactor Design Reference Documents

- 7.1-[1] Westinghouse Electric Corporation, "AP600 Standard Safety Analysis Report," W Document #DEACO3905F18495, Vols. 1-9, Proprietary Vols. 1-3, Revision 14, March 31, 1995. Portions are proprietary information. Not publicly available.
- 7.1-[2] ASEA Brown Boveri/Combustion Engineering, "Combustion Engineering Standard Safety Analysis Report," Chapters 1-6, App. 7A, Chapters 9-11, Chapter 15, Chapter 19, Amendment W, June 17, 1994. Portions are proprietary information. Not publicly available.
- 7.1-[3] United States Nuclear Regulatory Commission, "Draft Safety Evaluation Report Related to the Design Certification of CE System 80*," NUREG-1462, September 1992.
- 7.1-[4] General Electric Corporation, "Advanced Boiling Water Reactor - Standard Safety Analysis Report," GE Document #23A6100, Chapters 1-3, App. 3A, Chapters 4-9, App. 9A, Chapters 10-20, Revision 5, Amendment 34, May 1994. Portions are proprietary information. Not publicly available.
- 7.1-[5] United States Nuclear Regulatory Commission, "Preliminary Safety Evaluation Report Related to the Certification of the Advanced Boiling Water Reactor," NUREG-1469, October, 1992.
- 7.1-[6] United States Nuclear Regulatory Commission, "Final Safety Evaluation Report Related to the Certification of the Advanced

Boiling Water Reactor," NUREG-1503, July 1994.

7.2 USNRC Reference Documents and U.S. Regulations

- 7.2-[1] United States Nuclear Regulatory Commission, "Policy, Technical and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor Designs," Letter No. SECY-93-087, April 2, 1993.
- 7.2-[2] United States Nuclear Regulatory Commission, "NRC Briefing to Commission, SECY-93-087, Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light Water Reactors", May 14, 1993.
- 7.2-[3] United States Nuclear Regulatory Commission, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants", NUREG-0800, Revision 3, April 1989.
- 7.2-[4] Regulatory Guide 1.142, "Safety Related Concrete Structures for Nuclear Power Plants", Revision 1, October 1981.
- 7.2-[5] Regulatory Guide 1.143, "Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light Water Cooled Nuclear Power Plant", Revision 1, October 1979.
- 7.2-[6] Branch Technical Position, ETSB 11-5, "Postulated Radioactive Releases Due to a Waste Gas Systems Leak or Failure".
- 7.2-[7] Branch Technical Position, "ETSB 11-3, Design Guidance for Solid Radioactive Waste Management Systems Installed in Light Water Cooled Nuclear Power Reactor Plants".
- 7.2-[8] Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and radioactive-, Waste- Containing

Components of Nuclear Power Plants",
Revision 3, February 1976.

- 7.2-[9]** Regulatory Guide 1.76, "Design Basis Tornado for Nuclear Power Plants", Revision 0, April 1974.
- 7.2-[10]** 10CFR50 Appendix S, "Earthquake Engineering Criteria for Nuclear Power Plants", 1/1/97.
- 7.2-[11]** Regulatory Guide 1.165, "Identification and Characterization of Seismic Sources and Determination of Safe Shutdown Earthquake Ground Motion", Revision 0, March 1997.
- 7.2-[12]** 10CFR20, "Standards for Protection Against Radiation", 1/1/97.
- 7.2-[13]** 10CFR 50 Appendix I, "Numerical Guides for Design Objectives and Limiting Conditions for Operation to meet the Criterion, 'As Low as is Reasonably Achievable' for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents", 1/1/97.
- 7.2-[14]** 10 CFR50 Appendix A, "General Design Criteria for Nuclear Power Plants", 1/1/97.
- 7.2-[15]** 10CFR50 Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants", 1/1/97.
- 7.2-[16]** Regulatory Guide 1.91, "Evaluations of Explosions Postulated to Occur On Transportation Routes Near Nuclear Power Plants," Rev. 1, Feb. 1978.
- 7.2-[17]** Regulatory Guide 5.68, "Protection Against Malevolent Vehicles at Nuclear Power Plants," August 1994.
- 7.2-[18]** NUREG/CR-0098, "Development of Criteria for Seismic Review of Selected Nuclear Power Plants," Prepared for Nuclear Regulatory Commission, May 1978.

7.3 Operating Commercial Reactor Reference Documents

- 7.3-[1]** Beaver Valley Unit #1 and #2, Updated Safety Analysis Report.
- 7.3-[2]** Vogtle Unit #1 and #2, Updated Safety Analysis Report.
- 7.3-[3]** Washington Nuclear Power Plant, Updated Safety Analysis Report.
- 7.3-[4]** Waterford Nuclear Power Plant, Updated Safety Analysis Report.
- 7.3-[5]** Perry Nuclear Power Plant, Updated Safety Analysis Report.
- 7.3-[6]** Indian Point Plant, Unit #3, Updated Safety Analysis Report.
- 7.3-[7]** Palisades Nuclear Power Plant, Updated Safety Analysis Report.
- 7.3-[8]** Davis-Besse Nuclear Power Plant, Updated Safety Analysis Report.
- 7.3-[9]** Quad Cities Nuclear Power Plant, Updated Safety Analysis Report.

7.4 DOE Related Reference Documents

- 7.4-[1]** DOE Std. 1020-94, Ch. Notice 1-1996, Natural Phenomena Hazard Design and Evaluation Criteria for Department of Energy Facilities, U.S. Department of Energy, 1996.
- 7.4-[2]** Title 10 Code of Federal Regulation Part 71, "Packaging and Transportation of Radioactive Material", Federal Register, 27 May 1988.
- 7.4-[3]** DOE Std. 3009-94, "Preparation Guide for U.S. Nuclear Department of Energy Non-Reactor Nuclear Facilities Safety Analysis Reports", U.S. Department of Energy, July 1994.
- 7.4-[4]** DOE Std. 1027-92, "Hazard Categorization and Accident Analysis Techniques for

Compliance with DOE Order 5480.23",
Nuclear Safety Analysis Reports.

- 7.4-[5] DOE Std. 1021-94, "Natural Phenomena Hazards Performance Categorization Criteria for Structures, Systems and Components", Change Notice #1, January 1996.

- 7.4-[6] Code of Federal Regulation 10CFR830.120

7.5 Industry Consensus Codes and Standards

- 7.5-[1] American Concrete Institute, "ACI-349 Code Requirements for Nuclear Safety Related Concrete Structures", 1997

- 7.5-[2] American Concrete Institute, ACI-318 "Building Code Requirements for Reinforced Concrete (ACI 318-89) (Revised 1992)", 1992.

- 7.5-[3] American Concrete Institute, ACI-301-89, "Specifications for Structural Concrete for Buildings (ACI 301-89)", 1989.

- 7.5-[4] American Society of Civil Engineers, 7-95, "Minimum Design Loads for Buildings and Other Structures", 1995.

- 7.5-[5] American Nuclear Society "Solid Waste Processing System for Light Water Cooled Reactor Plants," ANSI/ANS-55.1-1992.

- 7.5-[6] American Nuclear Society, "Gaseous Radioactive Waste Processing Systems for Light-Water Cooled Reactor Plants," ANSI/ANS-55.4-1993.

- 7.5-[7] American Nuclear Society, "Liquid Radioactive Waste Processing System for Light Water Cooled Reactor Plants," ANSI/ANS 55.6-1993.

- 7.5-[8] ASCE Committee Report on Impact and Impulse 1980 Specialty Conference.

- 7.5-[9] American Society of Mechanical Engineers, NQA-1-1997 Edition, Quality Assurance Requirements for Nuclear Facility Applications", ASME NQ-1-1997.

- 7.5-[10] AISC/ANSI, "Specifications for Structural Steel Buildings Allowable Stress Design and Plastic Design, Manual of Steel Construction," 9th Edition, 1993.

- 7.5-[11] AISC/ANSI, "Specifications for Structural Steel Buildings, Manual of Steel Construction", 2nd Edition, 1995.

- 7.5-[12] ASME QME-1-1997, "Qualification of Active Mechanical Equipment Used in Nuclear Power Plants", December 31, 1997.

7.6 Miscellaneous NUREG/CR Reference Documents

- 7.6-[1] NUREG/CR-6358, "Assessment of United States Industry Structural Codes and Standards for Application to Advanced Nuclear Power Reactors", Adams and Stevenson, Vol. 1 and 2 October 1995.

- 7.6-[2] NUREG/CR-0098, "Development of Criteria for Seismic Review of Selected Nuclear Power Plants," Prepared for Nuclear Regulatory Commission, May 1978.

7.7 Miscellaneous References

- 7.7-[1] Zlesiak, Summary of Code Changes.

BIBLIOGRAPHIC DATA SHEET

(See instructions on the reverse)

1. REPORT NUMBER
(Assigned by NRC, Add Vol., Supp., Rev.,
and Addendum Numbers, if any.)

NUREG/CR-5733

2. TITLE AND SUBTITLE

Re-evaluation of Regulatory Guidance Provided in
Regulatory Guides 1.142 and 1.143

3. DATE REPORT PUBLISHED

MONTH	YEAR
August	1999

4. FIN OR GRANT NUMBER

W6540

5. AUTHOR(S)

T.M. Adams, J.D. Stevenson, G.G. Thomas, G.A. Harstead

6. TYPE OF REPORT

Technical

7. PERIOD COVERED (Inclusive Dates)

1/3/97 - 4/2/99

8. PERFORMING ORGANIZATION - NAME AND ADDRESS (If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)

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Cleveland, OH 44125

9. SPONSORING ORGANIZATION - NAME AND ADDRESS (If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)

Division of Engineering Technology
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

10. SUPPLEMENTARY NOTES

H. Graves III, NRC Project Manager

11. ABSTRACT (200 words or less)

This report recommends changes in the Nuclear Regulatory Commission's (NRC) criteria now used in the design of Nuclear Concrete Structures other than Containment and the design of radioactive Waste Management Systems, Structures and Components. The authors judged that the best method by which the conclusions of this report can be implemented are through the issuance of revisions to existing Regulatory Guidelines. There have been several developments since the issuance of the last revisions of Regulatory Guidelines 1.142 and 1.143 (1981 and 1979 respectively) which provides guidance for the design of these items. ACI 349, which presents the code requirements for nuclear safety related concrete structures, has had two major revisions since the last issue of the Regulatory Guidelines. New advanced reactor design concepts have been put forward that include unique attributes that need to be considered. The Department of Energy (DOE) has been active in writing standards for the design of radioactive waste management systems. There have also been numerous changes to the Code of Federal Regulations since the issuance of Regulatory Guidelines 1.142 and 1.143. This report discussed these advancements and changes with respect to the scope of the Regulatory Guidelines and includes detailed recommended changes to Revision 1 of Regulatory Guidelines 1.142 and 1.143.

12. KEY WORDS/DESCRIPTORS (List words or phrases that will assist researchers in locating the report.)

Regulatory Guide 1.142
Regulatory Guide 1.143
Concrete Design
ACI-349
Concrete Structures
Nuclear Waste Management
ANS 55.1
ANS 55.4
ANS 55.6Nuclear Waste Processing
Nuclear Waste Management Systems
Nuclear Waste Management Structures
Nuclear Waste Management Components

13. AVAILABILITY STATEMENT

unlimited

14. SECURITY CLASSIFICATION

(This Page)

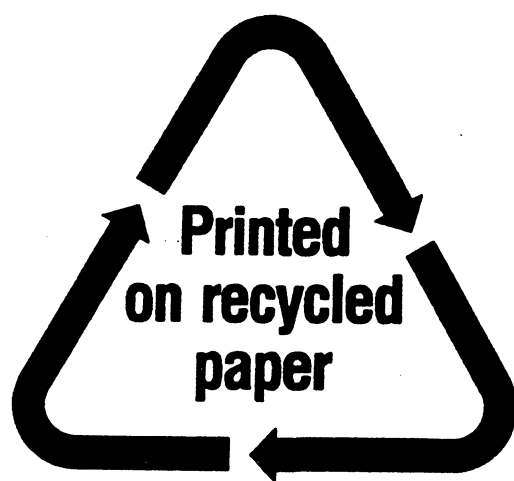
unclassified

(This Report)

unclassified

15. NUMBER OF PAGES

16. PRICE



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