Westinghouse Electric Company
Nuclear Power Plants
P.O. Box 355
Pittsburgh, Pennsylvania 15230-0355
USA

U.S. Nuclear Regulatory Commission
ATTENTION: Document Control Desk
Washington, D.C. 20555

Direct tel: 412-374-6206
Direct fax: 412-374-5005
e-mail: sisk1rb@westinghouse.com

Your ref: Docket Number 52-006
Our ref: DCP/NRC2159

June 13, 2008

Subject: AP1000 DCD Impact Document Submittal of APP-GW-GLE-007, Revision 0

Westinghouse is submitting Revision 0 of APP-GW-GLE-007, "ITAAC Changes." The purpose of this report is to identify changes to the AP1000 Design Control Document (DCD). These DCD changes were developed as a result of discussions during ITAAC closure workshops attended by NRC and Industry Representatives. These changes do not represent a change to the design of the AP1000. There are no changes to the configuration of structures, system, and components. Also, there is no change to the response of the AP1000 Design to anticipated transients of postulated accidents.

This report is submitted in support of the AP1000 Design Certification Amendment Application (Docket No. 52-006). The information provided in this report is generic and is expected to apply to all Combined Operating License (COL) applicants referencing the AP1000 Design Certification and the AP1000 Design Certification Amendment Application.

Questions or requests for additional information related to the content and preparation of this report should be directed to Westinghouse. Please send copies of such questions or requests to the prospective applicants for combined licenses referencing the AP1000 Design Certification. A representative for each applicant is included on the cc: list of this letter.

Very truly yours,

Robert Sisk, Manager
Licensing and Customer Interface
Regulatory Affairs and Standardization

/Enclosure

1. APP-GW-GLE-007, Revision 0, "ITAAC Changes"

D063
NRO

00398psa.doc

cc:	D. Jaffe	-	U.S. NRC	1E
	M. Miernicki	-	U.S. NRC	1E
	P. Buckberg	-	U.S. NRC	1E
	E. McKenna	-	U.S. NRC	1E
	R. Rasmussen	-	U.S. NRC	1E
	P. Ray	-	TVA	1E
	P. Hastings	-	Duke Power	1E
	R. Kitchen	-	Progress Energy	1E
	A. Monroe	-	SCANA	1E
	J. Wilkinson	-	Florida Power & Light	1E
	C. Pierce	-	Southern Company	1E
	G. Zinke	-	NuStart/Entergy	1E
	R. Grumbir	-	NuStart	1E
	L. Kass	-	NEI	1E
	E. Schmiech	-	Westinghouse	1E
	G. Adams	-	Westinghouse	1E

ENCLOSURE 1

APP-GW-GLE-007

Revision 0

"ITAAC Changes"

AP1000 DOCUMENT COVER SHEET

TDC:________ Permanent File:________

AP1000 DOCUMENT NO.	REVISION	PAGE	ASSIGNED TO	OPEN ITEMS (Y/N)
APP-GW-GLE-007	0	1 of 30	W-Sisk	

ALTERNATE DOCUMENT NUMBER: WORK BREAKDOWN #:

ORIGINATING ORGANIZATION: Westinghouse Electric

TITLE: **ITAAC Changes**

ATTACHMENTS: None	DCP #/REV. INCORPORATED IN THIS DOCUMENT REVISION:
CALCULATION/ANALYSIS REFERENCE: None	

ELECTRONIC FILENAME	ELECTRONIC FILE FORMAT	ELECTRONIC FILE DESCRIPTION
APP-GW-GLE-006.DOC	MSWord	

☒ **© 2008 WESTINGHOUSE ELECTRIC COMPANY LLC – WESTINGHOUSE NON-PROPRIETARY CLASS 3**

Class 3 Documents being transmitted to the NRC require the following two review signatures in lieu of a Form 36.

LEGAL REVIEW ~~T. J. White~~	SIGNATURE / DATE (If processing electronic approval select option) 6/12/08 6/13/2008
PATENT REVIEW D. Ekeroth	SIGNATURE / DATE 6/10/08

☐ **© 2008 WESTINGHOUSE ELECTRIC COMPANY LLC – WESTINGHOUSE PROPRIETARY CLASS 2**

This document is the property of and contains Proprietary Information owned by Westinghouse Electric Company LLC and/or its subcontractors and suppliers. It is transmitted to you in confidence and trust, and you agree to treat this document in strict accordance with the terms and conditions of the agreement under which it was provided to you.

☐ **© 2008 WESTINGHOUSE ELECTRIC COMPANY LLC and/or STONE & WEBSTER, INC. WESTINGHOUSE PROPRIETARY CLASS 2 and/or STONE & WEBSTER CONFIDENTIAL AND PROPRIETARY**

This document is the property of and contains Proprietary Information owned by Westinghouse Electric Company LLC and/or is the property of and contains Confidential and Proprietary Information owned by Stone & Webster, Inc. and/or their affiliates, subcontractors and suppliers. It is transmitted to you in confidence and trust, and you agree to treat this document in strict accordance with the terms and conditions of the agreement under which it was provided to you.

ORIGINATOR(S) D. A. Lindgren	SIGNATURE / DATE (If processing electronic approval select option) D. A. Lindgren 6/4/2008	
REVIEWER(S)	SIGNATURE / DATE	
	SIGNATURE / DATE	
	SIGNATURE / DATE	
VERIFIER(S) J. W. Willis	SIGNATURE / DATE 6/9/08	Verification Method: Independent Review Comparison to DCD-Rev16 & Drawings

Plant Applicability: ☒ All AP1000 plants except: HYG and SMG, HY1, HY2, SM1, SM2 6/13/08
☐ Only the following plants:

APPLICABILITY REVIEWER** J. A. Speer	SIGNATURE / DATE 6/10/08
RESPONSIBLE MANAGER* R. Sisk	SIGNATURE / DATE 6/13/08

* Approval of the responsible manager signifies that the document and all required reviews are complete, the appropriate proprietary class has been assigned, electronic file has been provided to the EDMS, and the document is released for use.

****** Electronically approved records are authenticated in the electronic document management system.***

Document Number: APP-GW-GLE-007 **Revision Number:** 0
Title: ITAAC Changes

Brief Description of the Impact (what is being changed and why):

Changes are made to Design Control Document Tier 1 information. These changes are a result of review of NRC Regulatory Issue Summary 2008-05 and information discovered during development of NEI 08-01. The changes include a revision of the definition of As-built, a revision to the ITAAC for the D-RAP, and splitting ITAAC in Section 3.3 into smaller pieces.

SRP Section Impacted:

The information in Tier 1 does not have guidance in the SRP. Certified Design Material is addressed by SRP 14.3. Building structures are addressed in SRP 3.8.3 and SRP 3.8.4. The D-RAP program is an operational program. Operational programs are addressed in SRP 13.4.

This evaluation is prepared to document the Design Control Document (DCD) change described above. The DCD change is a change from Tier 1 information of the AP1000 DCD, Revision 16. The changes identified in this document are intended to be included in a revision to the DCD and in the review of the Design Certification amendment or included as generic information in plant specific FSARs. Changes to Tier 1 information require review and approval by the NRC.

I. TECHNICAL DESCRIPTION

Four changes are included in this document.

1. The definition of as-built in Tier1, Section 1.1 of the AP1000 Design Control Document (DCD) is expanded to permit the performance of ITAAC activities at locations other than the final installed location.

2. ITAAC in Section 3.3 that require completion of activities in the entire nuclear island are split up to permit closure based on smaller areas. The ITAAC are divided into containment, shield building radiologically controlled auxiliary building, non-radiologically controlled auxiliary building, and annex building as required. This will permit closure and NRC review of the closure of a portion of ITAAC earlier than if the closure included all of the activities collected in to one large package. Tables 3.3-1 and 3.3-3 are also reconfigured to match the division of the ITAAC. In most cases the information in Table 3.3-1 is moved from one portion of the table to another without changing the information. In a few cases where walls or floors extended from one portion of the Auxiliary Building to another the existing entry was altered and a new entry was added to split the floor or wall into two separate elements unique to the two portions of the Auxiliary Building. Floors and walls that provide the boundary between the radiologically controlled and non radiologically controlled portions of the Auxiliary Building are assigned to only one portion of the building. One element previously labeled as a Floor is corrected to Roof.

3. Additional identifying numbers are added to the ITAAC in Section 3.3 to provide unique numbers for discrete ITAAC. The addition of the numbers does not change the ITAAC.

4. The acceptance criteria for the D-RAP ITAAC in Section 3.7 are modified to permit evaluation if the actual reliability of a component is less than originally assumed. One option is to evaluate the net effect of changes in as-built component reliability. Another option is to evaluate the effect on the core melt frequency and large release frequency and determine that there is no adverse effect. An adverse effect on the core melt frequency and large release frequency due to change in component reliability would be results greater that the safety goals or an increase in probability results greater that ten percent of the value of the safety goals.

II. CHANGE JUSTIFICATION

1. The revised definition of As-built in DCD Tier 1 Section 1.1 is consistent with the definition provided in NEI 08-01. This definition revision reflects NRC comments on the revision initially suggested by the NEI. The revised definition permits measurement of properties but does not close the ITAAC until after a component is installed. Installation is considered complete when a structure system or component is in its final location and activities such as welding a component into a system or bolting a component to supports or embedments are complete. Construction testing does not have to be completed to consider installation completed. It is intended that the ITAAC closure letters for these activities would discuss the steps taken to minimize the potential to change the measured properties but would not require taking the measurements again.

   This revision is consistent with the observation in NRC Regulatory Information Summary (RIS) 08-05 that "Applicants that envision using modular construction should consider the impact of these activities in developing their ITAAC. For example, the inspection of as-built SSCs implies that "as-built" refers to the completion of construction at the final location at the plant site. However, if a module is fabricated at a remote location (e.g., a shipyard), the individual components within the module (e.g., a pipe support) would be in their final locations, even if the module had not yet been transported to the site. Such considerations will facilitate ITAAC inspections and timely closures."

2. Breaking up ITAAC that have a large number of acceptance criteria activities collected into a single ITAAC that covers the entire nuclear island will not reduce the number of activities required or change the acceptance criteria for the ITAAC. Dividing the ITAAAC and reconfiguring the referenced tables does not change the walls and floors that are inspected. Note that the term limited hazard area used in one of the ITAAC is defined in IEEE 384 and is not impacted by the division of the ITAAC.

   This revision is consistent with the observation in RIS 08-05 that "Applicants should consider breaking ITAAC with a large number of SSCs into areas of construction. For instance, if an ITAAC envelops construction from the basemat to the top of containment and applies to several different buildings, the large area and scope create difficulties not only in tracking the applicable SSCs, but also in connecting the interdependent nature (e.g., seismic) of the applicable structures. Breaking a large single ITAAC into several areas will make the ITAAC closure process more efficient."

3. Adding identifying numbers to ITAAC requirements separated by dashes will reduce confusion and clarify which ITAAC is cited when ITAAC text is not repeated. This revision is consistent with the observation in RIS 08-05 that "Applicants should consider using a consistent system to identify and

number individual ITAAC within their applications. While an alphanumeric system has generally been used in past submittals, in some cases dashes or separate paragraphs with no labels were used in the body of the text to specify separate ITAAC requirements. In other cases, the alphanumeric designations were not consistently aligned for the applicable ITAAC table requirements. Use of a standard and consistent ITAAC identification system will minimize confusion."

4. Revising the D-RAP ITTAC to permit the evaluation of the effect of a reduction in reliability assumed for components is consistent with intent of requiring use of the PRA. If the net effect of reliability changes is that there is no impact on overall results, then the change of reliability of a particular component is not of regulatory concern. The use of ten percent of the safety goals for core melt and large release probabilities to determine if a change is adverse is consistent with other regulatory actions and requests for information.

   This revision is also consistent with the observation in RIS 08-05 that "Applicants should ensure that design commitments and ITAAC are consistent. It is important for the language and details of the ITAAC to comport with the language of the design commitment." The phrase in the Design Commitment of "consistent with their risk analysis assumptions" is more flexible than the "is at least equal to the assumed reliability" in the Acceptance Criteria.

III. REGULATORY IMPACT

A. EVALUATION OF DEPARTURE FROM TIER 2 INFORMATION (Check correct response and provide justification for that determination under each response)

   10 CFR Part 52, Appendix D, Section VIII. B.5.a. provides that an applicant for a combined licensee who references the AP1000 design certification may depart from Tier 2 information, without prior NRC approval, if it does not require a license amendment under paragraph B.5.b. These questions are addressed here to provide an evaluation of the regulatory impact. Regardless of the answers to these questions these changes are being provided to the NRC for review and approval as part of the design certification amendment. All changes to Tier 1 require NRC review and approval. The questions below address the criteria of B.5.b.

1. Does the proposed departure result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the plant-specific DCD? ☐ YES ☒ NO

   The ITAAC changes do not alter the results of analyses for structures, systems, or components. The likelihood of a failure of a structure, system, or component is not increased by the ITAAC changes. The ITAAC changes do not alter accident precursors or the design function of structures, systems, or components.

2. Does the proposed departure result in more than a minimal increase in the likelihood of occurrence of a malfunction of a structure, system, or component (SSC) important to safety and previously evaluated in the plant-specific DCD? ☐ YES ☒ NO

   The ITAAC changes do not increase the likelihood of an occurrence of a malfunction of a structure, system, or component important to safety. The ITAAC changes do not alter the response of structures, systems, and components to transient conditions, postulated accident conditions, or other loading combinations.

3. Does the proposed departure Result in more than a minimal increase in the consequences of an accident previously evaluated in the plant-specific DCD? ☐ YES ☒ NO

The ITAAC changes do not alter the design function of structures, systems, or components or alter the response to an accident previously evaluated in the plant-specific DCD. The ITAAC changes do not alter the calculation of radiation releases for postulated accident conditions.

4. Does the proposed departure result in more than a minimal increase in the consequences of a malfunction of an SSC important to safety previously evaluated in the plant-specific DCD? ☐ YES ☒ NO

   The ITAAC changes do not alter the design function of structures, systems, or components or alter the response to an accident previously evaluated in the plant-specific DCD. The ITAAC changes do not alter the calculation of radiation releases for postulated accident conditions.

5. Does the proposed departure create a possibility for an accident of a different type than any evaluated previously in the plant-specific DCD? ☐ YES ☒ NO

   The ITAAC changes do not alter the design function of structures, systems, or components. The ITAAC changes do not add or modify accident precursors.

6. Does the proposed departure create a possibility for a malfunction of an SSC important to safety with a different result than any evaluated previously in the plant-specific DCD? ☐ YES ☒ NO

   The ITAAC changes do not alter operating conditions or design functions of SSCs important to safety. Therefore there is no new malfunction.

7. Does the proposed departure result in a design basis limit for a fission product barrier as described in the plant-specific DCD being exceeded or altered? ☐ YES ☒ NO

   The ITAAC changes do not alter the pressure boundary integrity design function of the reactor coolant system or other SSCs important to safety. The ITAAC changes will not adversely alter the results of the evaluation of pressure boundary integrity.

8. Does the proposed departure result in a departure from a method of evaluation described in the plant-specific DCD used in establishing the design bases or in the safety analyses? ☐ YES ☒ NO

   The methodologies used to demonstrate closure of an ITAAC are defined in the ITAAC and in Tier 2 of the DCD. The ITAAC changes do not alter the methodologies used to demonstrate ITAAC closure.

B. IMPACT ON RESOLUTION OF A SEVERE ACCIDENT ISSUE

10 CFR Part 52, Appendix D, Section VIII. B.5.a. provides that an applicant for a combined licensee who references the AP1000 design certification may depart from Tier 2 information, without prior NRC approval, if it does not require a license amendment under paragraph B.5.c. The questions below address the criteria of B.5.c.

1. Does the proposed activity result in an impact to features that mitigate ex-vessel severe accidents. If the answer is Yes answer Questions 2 and 3 below. ☐ YES ☒ NO

   There is no change to the design or response of safety systems used to mitigate ex-vessel severe accidents due to the ITAAC changes.

2. Is there is a substantial increase in the probability of a ex-vessel severe accident such that a particular ex-vessel severe accident previously reviewed and determined to be not credible could become credible? ☐ YES ☐ NO ☒ N/A

3. Is there is a substantial increase in the consequences to the public of a particular ex-vessel severe accident previously reviewed? ☐ YES ☐ NO ☒ N/A

C. SECURITY ASSESSMENT

1. Does the proposed change have an adverse impact on the security assessment of the AP1000. ☐ YES ☒ NO

The ITAAC changes will not alter barriers or alarms that control access to protected areas of the plant. The ITAAC changes will not alter requirements for security personnel.

D. OTHER REGULATORY CRITERIA

10 CFR 50.47 requires that a design certification application must also contain:

> The proposed inspections, tests, analyses, and acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a facility that incorporates the design certification has been constructed and will be operated in conformity with the design certification, the provisions of the Act, and the Commission's rules and regulations;

NRC Regulatory Issue Summary 2008-05, "Lessons Learned To Improve Inspections, Tests, Analyses, And Acceptance Criteria Submittal" is intended to communicate to addressees the good practices for submitting inspections, tests, analyses, and acceptance criteria (ITAAC) as part of their applications for early site permits, standard design certifications, or combined licenses.

NEI 08-01, "Industry Guideline for the ITAAC Closure Process Under 10 CFR Part 52" is a document prepared to provide guidance for activities required for closure of ITAAC. This document has been prepared in close cooperation with the NRC and is expected to be endorsed by an NRC Regulatory Guide.

IV. DCD MARK-UP

Changes are shown with deletions shown with ~~strikeout~~ and additions <u>underlined</u>.

Revise the definition of As-built in Section 1.1 of DCD Tier 1 as follows:

As-built means the physical properties of a structure, system, or component following the completion of its installation or construction activities at its final location at the plant site. <u>Determination of physical properties of the as-built structure, system, or component may be based on measurements, inspections, or tests that occur prior to installation provided that subsequent fabrication, handling, installation, and testing does not alter the properties.</u>

Document Number: APP-GW-GLE-007 **Revision Number:** 0
Title: ITAAC Changes

Revise Table 3.3-3 as follows: The shaded row is moved but not otherwise changed.

Table 3.3-3
Class 1E Divisions in Nuclear Island Fire Areas

Fire Area Number	Class 1E Divisions			
	A	C	B	D
Auxiliary Building Radiologically Controlled				
1200 AF 01	Yes	Yes	–	–
1204 AF 01	Yes	–	–	–
Auxiliary Building Non-Radiologically Controlled				
1200 AF 03	–	–	Yes	Yes
1201 AF 02	–	–	Yes	–
1201 AF 03	–	–	–	Yes
1201 AF 04	–	–	Yes	Yes
1201 AF 05	–	–	Yes	Yes
1201 AF 06	–	–	Yes	Yes
1202 AF 03	–	Yes	–	–
1202 AF 04	Yes	–	–	–
1220 AF 01	–	–	Yes	Yes
1220 AF 02	–	–	–	Yes
1230 AF 01	Yes	Yes	–	–
1230 AF 02	–	–	Yes	Yes
1240 AF 01	Yes	Yes	–	–
1242 AF 02	Yes		–	

Note: Dash (–) indicates not applicable.

Document Number: APP-GW-GLE-007 **Revision Number:** 0
Title: ITAAC Changes

Revise Tier 1 Table 3.3-6 as follows:

Table 3.3-6
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The physical arrangement of the nuclear island structures and the annex building is as described in the Design Description of this Section 3.3 and Figures 3.3-1 through 3.3-14. The physical arrangement of the radwaste building, the turbine building, and the diesel generator building is as described in the Design Description of this Section 3.3.	An inspection of the nuclear island structures, the annex building, the radwaste building, the turbine building, and the diesel generator building will be performed.	The as-built nuclear island structures, the annex building, the radwaste building, the turbine building, and the diesel generator building conform with the physical arrangement as described in the Design Description of this Section 3.3 and Figures 3.3-1 through 3.3-14.
2.a) The nuclear island structures, including the critical sections listed in Table 3.3-7, are seismic Category I and are designed and constructed to withstand design basis loads as specified in the Design Description, without loss of structural integrity and the safety-related functions.	i) An inspection of the nuclear island structures will be performed. Deviations from the design due to as-built conditions will be analyzed for the design basis loads.	i) A. A report exists which reconciles deviations during construction and concludes that the as-built ~~nuclear island~~containment internal structures, including the critical sections, conform to the approved design and will withstand the design basis loads specified in the Design Description without loss of structural integrity or the safety-related functions.
		i) B. A report exists which reconciles deviations during construction and concludes that the as-built shield building structures, including the critical sections, conform to the approved design and will withstand the design basis loads specified in the Design Description without loss of structural integrity or the safety-related functions.
		i) C. A report exists which reconciles deviations during construction and concludes that the as-built structures in the non-radiologically controlled area of the auxiliary building, including the critical sections, conform to the approved design and will withstand the design basis loads specified in

Table 3.3-6
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
		the Design Description without loss of structural integrity or the safety-related functions.
		i) D. A report exists which reconciles deviations during construction and concludes that the as-built structures in the radiologically controlled area of the auxiliary building, including the critical sections, conform to the approved design and will withstand the design basis loads specified in the Design Description without loss of structural integrity or the safety-related functions.
	ii) An inspection of the as-built concrete thickness will be performed.	ii) A. A report exists that concludes that the containment internal structures as-built concrete thicknesses conform ~~with~~ to the building sections defined on Table 3.3-1.
		ii) B. A report exists that concludes that the as-built concrete thicknesses of the shield building sections conform to the building sections defined on Table 3.3-1.
		ii) C. A report exists that concludes that as-built concrete thicknesses of the non-radiologically controlled area of the auxiliary building sections conform to the building sections defined on Table 3.3-1.
		ii) D. A report exists that concludes that the as-built concrete thicknesses of the radiologically controlled area of the auxiliary building sections conform to the building sections defined on Table 3.3-1.
		ii) E. A report exists that concludes that the as-built concrete thicknesses of the Annex building

Table 3.3-6
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
		sections conform with the building sections defined on Table 3.3-1.
		ii) F. A report exists which concludes that the as-built concrete thicknesses of the turbine building sections conform to the building sections defined on Table 3.3-1.
2.b) Site grade level is located relative to floor elevation 100'-0" per Table 3.3-5.	Inspection of the as-built site grade will be conducted.	Site grade is consistent with design plant grade within the dimension defined on Table 3.3-5.
2.c) The containment and its penetrations are designed and constructed to ASME Code Section III, Class MC.	See Tier 1 Material, Subsection 2.2.1, Containment System.	See Tier 1 Material, Subsection 2.2.1, Containment System.

Table 3.3-6 (cont.)
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
2.d) The containment and its penetrations retain their pressure boundary integrity associated with the design pressure.	See Tier 1 Material, Subsection 2.2.1, Containment System.	See Tier 1 Material, Subsection 2.2.1, Containment System.
2.e) The containment and its penetrations maintain the containment leakage rate less than the maximum allowable leakage rate associated with the peak containment pressure for the design basis accident.	See Tier 1 Material, Subsection 2.2.1, Containment System.	See Tier 1 Material, Subsection 2.2.1, Containment System.
2.f) The key dimensions of nuclear island structures are defined on Table 3.3-5.	An inspection will be performed of the as-built configuration of the nuclear island structures.	A report exists and concludes that the key dimensions of the as-built nuclear island structures are consistent with the dimensions defined on Table 3.3-5.
2.g) The containment vessel greater than 7 feet above the operating deck provides a heat transfer surface. A free volume exists inside the containment shell above the operating deck.	The maximum containment vessel inside height from the operating deck is measured and the inner radius below the spring line is measured at two orthogonal radial directions at one elevation.	The containment vessel maximum inside height from the operating deck is 146'-7" (with tolerance of +12", -6"), and the inside diameter is 130 feet nominal (with tolerance of +12", -6").
2.h) The free volume in the containment allows for floodup to support long-term core cooling for postulated loss-of-coolant accidents.	An inspection will be performed of the as-built containment structures and equipment. The portions of the containment included in this inspection are the volumes that flood with a loss-of-coolant accident in passive core cooling system valve/equipment room B (11207). The in-containment refueling water storage tank volume is excluded from this inspection..	A report exists and concludes that the floodup volume of this portion of the containment is less than 73,500 ft^3 to an elevation of 108'.

Table 3.3-6 (cont.)
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
3. Walls and floors of the nuclear island structures as defined on Table 3.3-1 except for designed openings or penetrations provide shielding during normal operations.	Inspection of the as-built nuclear island structures wall and floor thicknesses will be performed.	A. A report exists and concludes that the shield walls and floors of the ~~nuclear island~~containment internal structures as defined on Table 3.3-1 except for designed openings or penetrations are consistent with the concrete wall thicknesses provided in Table 3.3-1.
		B. A report exists and concludes that the shield walls and floors of the shield building structures as defined on Table 3.3-1 except for designed openings or penetrations are consistent with the concrete wall thicknesses provided in Table 3.3-1.
		C. A report exists and concludes that the shield walls and floors of the non-radiologically controlled area of the auxiliary building as defined on Table 3.3-1 except for designed openings or penetrations are consistent with the concrete wall thicknesses provided in Table 3.3-1.
		D. A report exists and concludes that the shield walls and floors of the radiologically controlled area of the auxiliary building as defined on Table 3.3-1 except for designed openings or penetrations are consistent with the concrete wall thicknesses provided in Table 3.3-1.

Table 3.3-6 (cont.)
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
4.a) Walls and floors of the annex building as defined on Table 3.3-1 except for designed openings or penetrations provide shielding during normal operations.	Inspection of the as-built annex building wall and floor thicknesses will be performed.	A report exists and concludes that the shield walls and floors of the annex building as defined on Table 3.3-1 except for designed openings or penetrations are consistent with the minimum concrete wall thicknesses provided in Table 3.3-1.
4.b) Walls of the waste accumulation room in the radwaste building except for designed openings or penetrations provide shielding during normal operations.	Inspection of the as-built radwaste building wall thicknesses will be performed.	A report exists and concludes that the shield walls of the waste accumulation room in the radwaste building except for designed openings or penetrations are consistent with the minimum concrete wall thicknesses of 1'-4".
4.c) Walls of the packaged waste storage room in the radwaste building except for designed openings or penetrations provide shielding during normal operations.	Inspection of the as-built radwaste building wall thicknesses will be performed.	A report exists and concludes that the shield walls of the packaged waste storage room in the radwaste building except for the wall shared with the waste accumulation room and designed openings or penetrations are consistent with the minimum concrete wall thicknesses of 2'.
5.a) Exterior walls and the basemat of the nuclear island have a water barrier up to site grade.	An inspection of the as-built exterior walls and the basemat of the nuclear island up to floor elevation 100'-0", for application of water barrier will be performed during construction before the walls are poured.	A report exists that confirms that a water barrier exists on the nuclear island exterior walls up to site grade.
5.b) The boundaries between rooms identified in Table 3.3-2 of the auxiliary building are designed to prevent flooding of rooms that contain safety-related equipment.	An inspection of the auxiliary building rooms will be performed.	A report exists that confirms floors and walls as identified on Table 3.3-2 have provisions to prevent flooding between rooms up to the maximum flood levels for each room defined in Table 3.3-2.

Document Number: APP-GW-GLE-007 **Revision Number:** 0

Title: ITAAC Changes

Table 3.3-6 (cont.)
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
5.c) The boundaries between the following rooms, which contain safety-related equipment – PXS valve/accumulator room A (11205), PXS valve/accumulator room B (11207), and CVS room (11209) – are designed to prevent flooding between these rooms.	An inspection of the boundaries between the following rooms which contain safety-related equipment – PXS Valve/ Accumulator Room A (11205), PXS Valve/Accumulator Room B (11207), and CVS Room (11209) – will be performed.	A report exists that confirms that flooding of the PXS Valve/ Accumulator Room A (11205), and the PXS/Accumulator Room B (11207) is prevented to a maximum flood level as follows: PXS A 110'-2", PXS B 110'-1"; and of the CVS room (11209) to a maximum flood level of 110'-0".
6.a) The available room volumes of the radiologically controlled area of the auxiliary building between floor elevations 66'-6" and 82'-6" exceed the volume of the liquid radwaste storage tanks (WLS-MT-05A, MT-05B, MT-06A, MT-06B, MT-07A, MT-07B, MT-07C, MT-11).	An inspection will be performed of the as-built radiologically controlled area of the auxiliary building between floor elevations 66'-6" and 82'-6" to define volume.	A report exists and concludes that the as-built available room volumes of the radiologically controlled area of the auxiliary building between floor elevations 66'-6" and 82'-6" exceed the volume of the liquid radwaste storage tanks (WLS-MT-05A, MT-05B, MT-06A, MT-06B, MT-07A, MT-07B, MT-07C, MT-11).
6.b) The radwaste building package waste storage room has a volume greater than or equal to 1293 cubic feet.	An inspection of the radwaste building packaged waste storage room (50352) is performed.	The volume of the radwaste building packaged waste storage room (50352) is greater than or equal to 1293 cubic feet.
7.a) Class 1E electrical cables, communication cables associated with only one division, and raceways are identified according to applicable color-coded Class 1E divisions.	Inspections of the as-built Class 1E cables and raceways will be conducted.	A. Class 1E electrical cables, and communication cables inside containment associated with only one division, and raceways are identified by the appropriate color code.
		B. Class 1E electrical cables, and communication cables in the non-radiologically controlled area of the auxiliary building associated with only one division, and raceways are identified by the appropriate color code.

Table 3.3-6 (cont.) Inspections, Tests, Analyses, and Acceptance Criteria		
Design Commitment	**Inspections, Tests, Analyses**	**Acceptance Criteria**
		C. Class 1E electrical cables, and communication cables in the radiologically controlled area of the auxiliary building associated with only one division, and raceways are identified by the appropriate color code.
7.b) Class 1E divisional electrical cables and communication cables associated with only one division are routed in their respective divisional raceways.	Inspections of the as-built Class 1E divisional cables and raceways will be conducted.	A. Class 1E electrical cables and communication cables inside containment associated with only one division are routed in raceways assigned to the same division. There are no other safety division electrical cables in a raceway assigned to a different division.
		B. Class 1E electrical cables and communication cables in the non-radiologically controlled area of the auxiliary building associated with only one division are routed in raceways assigned to the same division. There are no other safety division electrical cables in a raceway assigned to a different division.
		C. Class 1E electrical cables and communication cables in the radiologically controlled area of the auxiliary building associated with only one division are routed in raceways assigned to the same division. There are no other safety division electrical cables in a raceway assigned to a different division.
7.c) Separation is maintained between Class 1E divisions in accordance with the fire areas as identified in Table 3.3-3.	i) Inspections of the as-built Class 1E division electrical cables, communication cables associated with only one division, and raceways located in the fire areas identified in Table 3.3-3 will be conducted.	i) A. Results of the inspection will confirm that the separation between Class 1E divisions in the non-radiologically controlled area of the auxiliary building is consistent with Table 3.3-3.

Table 3.3-6 (cont.)
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
		i) B. Results of the inspection will confirm that the separation between Class 1E divisions in the radiologically controlled area of the auxiliary building is consistent with Table 3.3-3.
	ii) Inspections of the as-built fire barriers between the fire areas identified in Table 3.3-3 will be conducted.	ii) A. Results of the inspection will confirm that fire barriers exist between Class 1E divisions inside the non radiologically consistent with the fire areas identified in Table 3.3-3.
		ii) B. Results of the inspection will confirm that fire barriers exist between Class 1E divisions inside the radiologically controlled area of the auxiliary building consistent with the fire areas identified in Table 3.3-3.
7.d) Physical separation is maintained between Class 1E divisions and between Class 1E divisions and non-Class 1E cables.	Inspections of the as-built Class 1E raceways will be performed to confirm that the separation between Class 1E raceways of different divisions and between Class 1E raceways and non-Class 1E raceways is consistent with the following:	Results of the inspection will confirm that the separation between Class 1E raceways of different divisions and between Class 1E raceways and non-Class 1E raceways is consistent with the followings:
	01) ——Within the main control room and remote shutdown room, the minimum vertical separation is 3 inches and the minimum horizontal separation is 1 inch.	01) ——Within the main control room and remote shutdown room, the vertical separation is 3 inches or more and the horizontal separation is 1 inch or more.
	02) Within other plant areas (limited hazard areas), the minimum separation is defined by one of the following:	02) A. Within other plant areas inside containment (limited hazard areas), the separation meets one of the following:
	1) The minimum vertical separation is 5 feet and the minimum horizontal separation is 3 feet.	1) The vertical separation is 5 feet or more and the horizontal separation is 3 feet or more except.

Table 3.3-6 (cont.)
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
	2) The minimum vertical separation is 12 inches and the minimum horizontal separation is 6 inches for raceways containing only instrumentation and control and low-voltage power cables <2/0 AWG.	2) The minimum vertical separation is 12 inches and the minimum horizontal separation is 6 inches for raceways containing only instrumentation and control and low-voltage power cables <2/0 AWG.
	3) For configurations that involve exclusively limited energy content cables (instrumentation and control), the minimum vertical separation is 3 inches and the minimum horizontal separation is 1 inch.	3) For configurations that involve exclusively limited energy content cables (instrumentation and control), the minimum vertical separation is 3 inches and the minimum horizontal separation is 1 inch.
	4) For configurations involving an enclosed raceway and an open raceway, the minimum vertical separation is 1 inch if the enclosed raceway is below the open raceway.	4) For configurations that involve an enclosed raceway and an open raceway, the minimum vertical separation is 1 inch if the enclosed raceway is below the raceway.
	5) For configuration involving enclosed raceways, the minimum separation is 1 inch in both horizontal and vertical directions.	5) For configurations that involve enclosed raceways, the minimum vertical and horizontal separation is 1 inch.
		02) B. Within other plant areas inside the non-radiologically controlled area of the auxiliary building (limited hazard areas), the separation meets one of the following:
		1) The vertical separation is 5 feet or more and the horizontal separation is 3 feet or more except.

Document Number: APP-GW-GLE-007 **Revision Number:** 0

Title: ITAAC Changes

Table 3.3-6 (cont.)
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
		2) The minimum vertical separation is 12 inches and the minimum horizontal separation is 6 inches for raceways containing only instrumentation and control and low-voltage power cables <2/0 AWG.
		3) For configurations that involve exclusively limited energy content cables (instrumentation and control), the minimum vertical separation is 3 inches and the minimum horizontal separation is 1 inch.
		4) For configurations that involve an enclosed raceway and an open raceway, the minimum vertical separation is 1 inch if the enclosed raceway is below the raceway.
		5) For configurations that involve enclosed raceways, the minimum vertical and horizontal separation is 1 inch.
		02) C. Within other plant areas inside the radiologically controlled area of the auxiliary building (limited hazard areas), the separation meets one of the following:
		1) The vertical separation is 5 feet or more and the horizontal separation is 3 feet or more except.

Table 3.3-6 (cont.) Inspections, Tests, Analyses, and Acceptance Criteria		
Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
		2) The minimum vertical separation is 12 inches and the minimum horizontal separation is 6 inches for raceways containing only instrumentation and control and low-voltage power cables ≤2/0 AWG.
		3) For configurations that involve exclusively limited energy content cables (instrumentation and control), the minimum vertical separation is 3 inches and the minimum horizontal separation is 1 inch.
		4) For configurations that involve an enclosed raceway and an open raceway, the minimum vertical separation is 1 inch if the enclosed raceway is below the raceway.
		5) For configurations that involve enclosed raceways, the minimum vertical and horizontal separation is 1 inch.
	03) Where minimum separation distances are not maintained, the circuits are run in enclosed raceways or barriers are provided.	03) A. Where minimum separation distances are not met inside containment, the circuits are run in enclosed raceways or barriers are provided.
		03) B. Where minimum separation distances are not met inside the non-radiologically controlled area of the auxiliary building, the circuits are run in enclosed raceways or barriers are provided.
		03) C. Where minimum separation distances are not met inside the radiologically controlled area of the auxiliary building, the circuits are run in enclosed raceways or barriers are provided.

Table 3.3-6 (cont.)
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
	04) Separation distances less than those specified above and not run in enclosed raceways or provided with barriers are based on analysis	04) A. For areas inside containment A report exists and concludes that separation distances less than those specified above and not provided with enclosed raceways or barriers have been analyzed.
		04) B. For areas inside the non-radiologically controlled area of the auxiliary building a report exists and concludes that separation distances less than those specified above and not provided with enclosed raceways or barriers have been analyzed.
		04) C. For areas inside the radiologically controlled area of the auxiliary building a report exists and concludes that separation distances less than those specified above and not provided with enclosed raceways or barriers have been analyzed.
	05) Non-Class 1E wiring that is not separated from Class 1E or associated wiring by the minimum separation distance or by a barrier or analyzed is considered as associated circuits and subject to Class 1E requirements.	05) A. For areas inside containment ~~N~~non-Class 1E wiring that is not separated from Class 1E or associated wiring by the minimum separation distance or by a barrier or analyzed is treated as Class 1E wiring.
		05) B. For areas inside the non-radiologically controlled area of the auxiliary building non-Class 1E wiring that is not separated from Class 1E or associated wiring by the minimum separation distance or by a barrier or analyzed is treated as Class 1E wiring.

Table 3.3-6 (cont.)
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
		05) C. For areas inside the radiologically controlled area of the auxiliary building non-Class 1E wiring that is not separated from Class 1E or associated wiring by the minimum separation distance or by a barrier or analyzed is treated as Class 1E wiring.
7.e) Class 1E communication cables which interconnect two divisions are routed and separated such that the Protection and Safety Monitoring System voting logic is not defeated by the loss of any single raceway or fire area.	Inspections of the as-built Class 1E communication cables will be conducted.	Class 1E communication cables which interconnect two divisions are routed and separated such that the Protection and Safety Monitoring System voting logic is not defeated by the loss of any single raceway or fire area.
8. Equipment labeled as essential targets in Table 3.3-4 and located in rooms identified in Table 3.3-4 are protected from the dynamic effects of postulated pipe breaks.	An inspection will be performed of the as-built high energy pipe break pipe whip restraints features for systems located in rooms identified in Table 3.3-4.	An as-built Pipe Rupture Hazard Analysis Report exists and concludes that equipment labeled as essential targets in Table 3.3-4 and located in rooms identified in Table 3.3-4 can withstand the effects of postulated pipe rupture without loss of required safety function.
9. The reactor cavity sump has a minimum concrete thickness as shown in Table 3.3-5 between the bottom of the sump and the steel containment.	An inspection of the as-built containment building internal structures will be performed.	A report exists and concludes that the reactor cavity sump has a minimum concrete thickness as shown on Table 3.3-5 between the bottom of the sump and the steel containment.

Table 3.3-6 (cont.)
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
10. The shield building roof and PCS storage tank support and retain the PCS water sources. The PCS storage tank has a stainless steel liner which provides a barrier on the inside surfaces of the tank. Leak chase channels are provided on the tank boundary liner welds.	i) A test will be performed to measure the leakage from the PCS storage tank based on measuring the water flow out of the leak chase collection system. ii) An inspection of the PCS storage tank exterior tank boundary and shield building tension ring will be performed before and after filling of the PCS storage tank to the overflow level. The vertical elevation of the shield building roof will be measured at a location at the outer radius of the roof (tension ring) and at a location on the same azimuth at the outer radius of the PCS water storage tank before and after filling the PCS storage tank.	i) A report exists and concludes that total water flow from the leak chase collection system does not exceed 10 gal/hr. ii) A report exists and concludes that there is no visible water leakage from the PCS storage tank and that inspection and measurement of the structure before and after filling of the tank shows structural behavior under normal loads to be acceptable.
11. Deleted		
12. The extended turbine generator axis intersects the shield building.	An inspection of the as-built turbine generator will be performed.	The extended axis of the turbine generator intersects the shield building.
13. Separation is provided between the structural elements of the turbine, annex and radwaste buildings and the nuclear island structure. This separation permits horizontal motion of the buildings in the safe shutdown earthquake without impact between structural elements of the buildings.	An inspection of the separation of the nuclear island from the annex, radwaste and turbine building structures will be performed. The inspection will verify the specified horizontal clearance between structural elements of the adjacent buildings, consisting of the reinforced concrete walls and slabs, structural steel columns and floor beams.	The minimum horizontal clearance above floor elevation 100′-0″ between the structural elements of the annex and radwaste buildings and the nuclear island is 4 inches. The minimum horizontal clearance above floor elevation 100′-0″ between the structural elements of the turbine building and the nuclear island is 12 inches.
14. The walls, doors, ceiling, and floors in the main control room, central alarm station, and secondary alarm station are bullet-resistant to a level 4 round.	Type test, analysis, or a combination of type test and analysis will be performed for the walls, doors, ceilings, and floors in the main control room, central alarm station, and secondary alarm station.	A report exists and concludes that the walls, doors, ceilings, and floors in the main control room, central alarm station, and secondary alarm station are bullet-resistant to a level 4 round.

Table 3.3-6 (cont.) Inspections, Tests, Analyses, and Acceptance Criteria		
Design Commitment	**Inspections, Tests, Analyses**	**Acceptance Criteria**
15. Central alarm station and main control room are vital areas.	An inspection of the as-built central alarm station and main control room will be performed.	Access to the central alarm station and main control room is through an activated intrusion alarm system and at least two security hardened barriers.
16. Security power supply system for alarm annunciator equipment and non-portable communications equipment is located within a vital area.	An inspection of the as-built location of the security power supply for alarm annunciator equipment and non-portable communications equipment will be performed.	Access to the security power supply for alarm annunciator equipment and non-portable communications equipment is through an activated intrusion alarm system and at least two security hardened barriers.
17. Vital areas are locked and alarmed with active intrusion detection systems that annunciate in the central and secondary alarm stations upon intrusion into a vital area.	An inspection of the as-built vital areas, and central and secondary alarm stations are performed.	Vital areas are locked and alarmed with active intrusion detection systems that annunciate in the central and secondary alarm stations upon intrusion into a vital area.
18. The locks used for the protection of the vital areas are manipulative-resistant.	Type test, analysis, or a combination of type test and analysis will be performed for the locks used in the protection of the vital areas.	A report exists and concludes that the locks used for the protection of the vital areas are manipulative-resistant.

Revise Table 3.3-1 as shown in the following:

Note: Shaded rows are moved from original position but not otherwise changed,

Table 3.3-1 (cont.)
Definition of Wall Thicknesses for Nuclear Island Buildings and Annex Building[(1)]

Wall or Section Description	Column Lines	Floor Elevation or Elevation Range	Concrete Thickness[(2)(3)]	Applicable Radiation Shielding Wall (Yes/No)
East wall of east steam generator compartment	Not Applicable	From 94'-0" to 153'-0"	2'-6"	Yes
North wall of east steam generator compartment	Not Applicable	From 87'-6" to 153'-0"	2'-6"	Yes
Shield Building				
Shield Building Cylinder	Not Applicable	From 100'-0" to 251'-6" From 256'-9" to 266'-6"	3'-0" 4'-6"	Yes Yes
Tension Ring	Not Applicable	From 266'-6" to 271'	3'-0"	Yes
Conical Roof	Not Applicable	From 271'-0" to 293'-9"	3'-0" (including 1/2 inch thick steel plate liner on each face)	Yes
PCS Tank External Cylindrical Wall	Not Applicable	From 293'-9" to 328'-9"	2'-0"	Yes
PCS Tank Internal Cylindrical Wall	Not Applicable	From 309'-4" to 329'-0"	1'-6"	Yes
PCS Tank Roof	Not Applicable	328'-9" (Lowest) 329'-0" (Highest)	1'-3"	No
Nuclear Island Basemat	Below shield building	From 60'-6" to containment vessel or 82'-6"	6'-0" to 22'-0" (varies)	No
Auxiliary Building Walls/Floors <u>Radiologically Controlled</u>				
Column Line 1 wall	From I to N	From 66'-6" to 100'-0"	3'-0"	No
Column Line 1 wall	From I to 5'-6" east of L-2	From 100'-0" to 180'-0"	2'-3"	Yes
Column Line 1 wall	From 5'-6" east of L-2 to N	From 100'-0" to 125'-0"	3'-0"	Yes
Column Line 1 wall	From 5'-6" east of L-2 to N	From 125'-0" to 180'-0"	2'-3"	Yes
Column Line 2 wall	From I to K-2	From 66'-6" to 135'-3"	2'-6"	Yes
Column Line 2 wall	From K-2 to L-2	From 66'-6" to 135'-3"	5'-0"	Yes
Column Line 2 wall	From L-2 to N	From 98'-1" to 135'-3"	2'-6"	Yes
Column Line 2 wall	From I to J-1	From 135'-3" to 153'-0"	2'-0"	Yes
Column Line 3 wall	From J-1 to J-2	From 66'-6" to 82'-6"	2'-6"	Yes
Column Line 3 wall	From J-1 to J-2	From 100'-0" to 135'-3"	2'-6"	Yes
Column Line 3 wall	From J-2 to K-2	From 66'-6" to 135'-3"	2'-6"	Yes

Table 3.3-1 (cont.)
Definition of Wall Thicknesses for Nuclear Island Buildings and Annex Building[(1)]

Wall or Section Description	Column Lines	Floor Elevation or Elevation Range	Concrete Thickness[(2)(3)]	Applicable Radiation Shielding Wall (Yes/No)
Column Line 3 wall	From K-2 to L-2	From 66'-6" to 92'-8 1/2"	2'-6"	Yes
Column Line 4 wall	From I to J-1	From 66'-6" to 153'-0"	2'-6"	Yes
Column Line 4 wall	From J-1 to J-2	From 66'-6" to 92'-6"	2'-6"	Yes
Column Line 4 wall	From J-1 to J-2	From 107'-2" to 135'-3"	2'-6"	Yes
Column Line 4 wall	From J-2 to K-2	From 66'-6" to 135'-3"	2'-6"	Yes
Column Line 4 wall	From I to intersection with shield building wall	From 135'-3" to 180'-0"	2'-0"	Yes
Column Line 5 wall	From I to shield building; with opening east of J-1 (below 107'-2" floor).	From 66'-6" to 160'-6"	2'-0"	Yes
Column Line 7.1 wall	From I to 8' east of J-1	From 66'-6" to 82'-6"	2'-0"	Yes
Column Line 7.2 wall	From I to 5'-6"east of J-1	From 66'-6" to 100'-0"	2'-0"	Yes
Column Line I wall	From 1 to ~~11~~7.3	From 66'-6" to 100'-0"	3'-0"	No
Column Line I wall	From 1 to 4	From 100'-0" to 180'-0"	2'-0"	Yes
Column Line I wall	From 4 to ~~7.3~~5	From 100'-0" to 160'-6"	2'-0"	No
Column Line J-1 wall	From 1 to 2	From 82'-6" to 100'-0"	2'-0"	Yes
Column Line J-1 wall	From 2 to 4	From 66'-6" to 135'-3"	2'-6"	Yes
Column Line J-1 wall	From 2 to 4	From 135'-3" to 153'-0"	2'-0"	Yes
Column Line J-1 wall	From 4 to shield building	From 66'-6" to 107'-2"	2'-0"	Yes
Column Line J-2 wall	From 2 to 4	From 66'-6" to 135'-3"	2'-6"	Yes
Column Line J-2 wall	From 4 to intersection with shield building wall	From 66'-6" to 135'-3"	2'-0"	Yes
Column Line K-2 wall	From 2 to 4	From 66'-6" to 135'-3"	4'-9"	Yes
Column Line L-2 wall	From 2 to 4	From 66'-6" to 135'-3"	4'-0"	Yes
Column Line N wall	From 1 to 2	From 66'-6" to 100'-0"	3'-0"	No
Column Line N wall	From 1 to 12'-9" north of 1	From 100'-0" to 125'-0"	3'-9"	No
Column Line N wall	From 1 to 12'-9" north of 1	From 125'-0" to 135'-0"	2'-0"	No
Column Line N wall	From 12'-9" north of 1 to 2	From 100'-0" to 118'-2 1/2"	3'-0"	No
Column Line N wall	From 12'-9" north of 1 to 2	From 118'-2 1/2" to 135'-3"	2'-0"	No

Table 3.3-1 (cont.)
Definition of Wall Thicknesses for Nuclear Island Buildings and Annex Building[(1)]

Wall or Section Description	Column Lines	Floor Elevation or Elevation Range	Concrete Thickness[(2)(3)]	Applicable Radiation Shielding Wall (Yes/No)
Column Line N wall	From 1 to 2	From 118'-2 1/2" to 135'-3"	2'-0"	Yes
Column Line N wall	From 2 to 4	From 66'-6" to 98'-1"	3'-0"	No
Column Line N wall	From 2 to 4	From 98'-1" to 135'-3"	5'-6"	Yes
Column Line N wall	From 1 to 4	From 135'-3" to 180'-0"	2'-0"	Yes
Labyrinth Wall between Col. Line 3 and 4 and J-1 to 7'-3" from J-2	Not Applicable	From 82'-6" to 92'-6"	2'-6"	Yes
N-S Shield Wall (low wall)	Between K-2 and L-2 extending from column line 1 north	From 100'-0" to 107'-2"	2'-6"	Yes
N-S Shield Wall	Between K-2 and L-2 extending from column line 1 north	From 100'-0" to 125'-0"	2'-3"	Yes
E-W Shield Wall	Between 1 and 2 extending from column line N east	From 100'-0" to 125'-0"	2'-9"	Yes
Auxiliary Area Basemat	From 1-7.3 and I-N, excluding shield building	From 60'-6" to 66'-6"	6'-0"	No
Floor	From 1 to 2 and I to N	82'-6"	2'-0"	Yes
Floor	From 2 to 4 and J-1 to J-2	82'-6"	2'-0"	Yes
Floor	From 4 to 5 and J-1 to J-2	82'-6"	0'-9"	Yes
Pipe Chase Floor	From 2 to 5 and J-1 to J-2	92'-6"	2'-0"	Yes
Floor	From 2 to 3 and J-2 to K-2	90'-3"	3'-0"	Yes
Floor	From 3 to 4 and J-2 to K-2	92'-6"	2'-0"	Yes
Floor	From 4 to 7.3 and I to J-1	82'-6"	2'-0"	Yes
Floor	From 1 to 2 and I to N	100'-0"	3'-0"	Yes
Floor	From 2 to 4 and K-2 to L-2	92'-8 1/2"	3'-2 1/2"	Yes
Floor	From I to J-2 and 4 to intersecting vertical wall before column line 5	107'-2"	2'-0"	Yes
Floor	From I to shield building wall and from intersecting vertical wall before column line 5 to column line 5	105'-0"	0'-9"	Yes
Floor	From 1 to 10'-0" north of 1 and L-2 to N	125'-0"	3'-0"	Yes

Table 3.3-1 (cont.)
Definition of Wall Thicknesses for Nuclear Island Buildings and Annex Building(1)

Wall or Section Description	Column Lines	Floor Elevation or Elevation Range	Concrete Thickness(2)(3)	Applicable Radiation Shielding Wall (Yes/No)
Floor	From 10'-0" north of 1 to 2 and L-2 to N	118'-2 1/2"	2'-0"	Yes
Floor	From 3 to 4 and J-2 to K-2	117'-6"	2'-0"	Yes
Floor	From 2 to 4 and I to J-1	153'-0"	1'-1 1/2"	Yes
~~Floor~~Roof	From 1 to 4 and I to N	180'-0"	1'-3"	Yes
Floor	From 4 to short of column line 5 and from I to intersection with shield building wall	135'-5"	0'-9"	Yes
Floor	From short of column line 5 to column line 5 and from I to intersection with shield building wall	133'-0"	0'-9"	Yes
Auxiliary Building Walls/Floors Non Radiologically Controlled				
Column Line 11 wall	From I to Q	From 66'-6" to 100'-0"	3'-0"	No
Column Line 11 wall	From I to Q	From 100'-0" to 117'-6"	2'-0"	Yes
Column Line 11 wall	From I to L	From 117'-6" to 153'-0"	2'-0"	Yes
Column Line 11 wall	From L to M	From 117'-6" to 135'-3"	4'-0"	Yes
Column Line 11 wall	From M to P	From 117'-6" to 135'-3"	2'-0"	Yes
Column Line 11 wall	From P to Q	From 117'-6" to 135'-3"	4'-0"	Yes
Column Line 11 wall	From L to Q	From 135'-3" to 153'-0"	2'-0"	Yes
Column Line 7.3 wall	From I to shield building	From 66'-6" to 100'-0"	3'-0"	Yes
Column Line 7.3 wall	From I to shield building	From 100'-0" to 160'-6"	2'-0"	No
Column Line I wall	From 7.3 to 11	From 66'-6" to 100'-0"	3'-0"	No
Column Line I wall	From 7.3 to 11	From 100'-0" to 153'-0"	2'-0"	No
Column Line I wall	From 5 to 7.3	From 100'-0" to 160'-6"	2'-0"	No
Column Line J wall	From 7.3 to 11	From 66'-6" to 117'-6"	2'-0"	No
Column Line K wall	From 7.3 to 11	From 60'-6" to 135'-3"	2'-0"	Yes
Column Line L wall	From shield building wall to 11	From 60'-6" to 153'-0"	2'-0"	Yes
Column Line M wall	From shield building wall to 11	From 66'-6" to 153'-0"	2'-0"	Yes
Column Line P wall	From shield building wall to 11	From 66'-6" to 153'-0"	2'-0"	Yes

Table 3.3-1 (cont.)
Definition of Wall Thicknesses for Nuclear Island Buildings and Annex Building(1)

Wall or Section Description	Column Lines	Floor Elevation or Elevation Range	Concrete Thickness(2)(3)	Applicable Radiation Shielding Wall (Yes/No)
Column Line Q wall	From shield building wall to 11	From 66'-6" to 100'-0"	3'-0"	No
Column Line Q wall	From shield building wall to 11	From 100'-0" to 153'-0"	2'-0"	Yes
Column Line 9.2 wall	From I to J and K to L	From 117'-6" to 135'-3"	2'-0"	Yes
Labyrinth Wall between Column Line 7.3 and 9.2 and J to K	J to K	From 117'-6" to 135'-3"	2'-0"	Yes
Auxiliary Area Basemat	From ~~1~~7.3-11 and I-Q, excluding shield building	From 60'-6" to 66'-6"	6'-0"	No
Floor	From 5 to 7.3 and I to shield building wall	100'-0"	2'-0"	Yes
Floor	From K to L and shield building wall to column line 10	100'-0"	0'-9"	Yes
Main Control Room Floor	From 9.2 to 11 and I to L	117'-6"	2'-0"	Yes
Floor	Bounded by shield bldg, 7.3, J, 9.2 and L	117'-6"	2'-0"	Yes
Floor	From 9.2 to 11 and L to Q	117'-6"	2'-0"	Yes
Floor	From 5 to 7.3 and from I to intersection with shield building wall	135'-3"	0'-9"	Yes
Annex Building				
Column line 2 wall	From E to H	From 107'-2" to 135'-3"	19 3/4"	Yes
Column line 4 wall	From E to H	From 107'-2" to 162'-6" & 166'-0"	2'-0"	Yes
N-S Shield Wall between E and F	From 2 to 4	From 107'-2" to 135'-3"	1'-0"	Yes
Column line 4.1 wall	From E to H	From 107'-2" to 135'-3"	2'-0"	Yes
E-W Labyrinth Wall between column line 7.1 and 7.8 and G to H	Not Applicable	From 100'-0" to 112'-0"	2'-0"	
N-S Labyrinth Wall between column line 7.8 and 9 and G to H	Not Applicable	From 100'-0" to 112'-0"	2'-0"	
E-W Labyrinth Wall between column line 7.1 and 7.8 and G to H	Not Applicable	From 100'-0" to 112'-0"	2'-0"	Yes
N-S Shield Wall on Column line. F	From 4.1 North	From 100'-0" to 117'-6"	1'-0"	Yes
Column Line 9 wall	From E to connecting wall between G and H	From 107'-2" to 117'-6"	2'-0"	Yes

Table 3.3-1 (cont.)
Definition of Wall Thicknesses for Nuclear Island Buildings and Annex Building[1]

Wall or Section Description	Column Lines	Floor Elevation or Elevation Range	Concrete Thickness[2][3]	Applicable Radiation Shielding Wall (Yes/No)
Column Line E wall	From 9 to 13	From 100'-0" to 135'-3"	2'-0"	Yes
Column Line 13 wall	From E to I.1	From 100'-0" to 135'-3"	2'-0"	Yes
Column Line I.1 wall	From 11.09 to 13	From 100'-0" to 135'-3"	2'-0"	Yes
Corridor Wall between G and H	From 9 to 13	From 100'-0" to 135'-3"	1'-6"	Yes
Column Line 9 wall	From I to H	From 117'-6" to 158'-0"	2'-0"	Yes
Floor	2 to 4 from shield wall between E and F to column line H	135'-3"	0'-6"	Yes
Floor	From 4 to 4.1 and E to H	135'-3"	1'-0"	Yes
Floor	From 9 to 13 and E to I.1	117'-6"	0'-6"	Yes
Floor	From 9 to 13 and E to I.1	135'-3"	0'-8"	Yes
Containment Filtration Rm A (North Wall)	Between column line E to H	From 135'-3" to 158'-0"	1'-0"	Yes
Containment Filtration Rm A (East wall)	Between column line E to F	From 135'-3" to 158'-0"	1'-0"	Yes
Containment Filtration Rm A (West wall)	Between column line G to H	From 135'-3" to 158'-0"	1'-0"	Yes
Containment Filtration Rm A (Floor)	Between column line E to H	135'-3"	1'-0"	Yes
Containment Filtration Rm B (Floor)	Between column line E to H	146'-3"	0'-6"	Yes
Containment Filtration Rm B (West wall)	Between column line G to H	From 146'-3" to 158'-0"	1'-0"	Yes
Turbine Building				
Wall between I.2 and I.1	From 11.05 to 11.2	From 100'-0" to 161'-0"	2'-0"	No
Column Line 11.2 Wall	From 1'-2" off I.1 to 2'-4" past R	From 100'-0" to 161'-0"	2'-0"	No
Wall 2'-4" past R	From 11 to 11.2	From 100'-0" to 161'-0"	2'-0"	No
Wall 11	From 11-0" off Q to 2'-4" past R	From 100'-0" to 161'-0"	2'-0"	No

Revise Table 3.7-3 in Tier 1 as follows:

Table 3.7-3
Inspections, Tests, Analyses and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The D-RAP provides reasonable assurance that the design of risk-significant SSCs is consistent with their risk analysis assumptions.	Inspection will be performed for the existence of a report which establishes the estimated reliability of as-built risk-significant SSCs.	A report exists and concludes that the estimated reliability of each as-built component identified in Table 3.7-1 is at least equal to the assumed reliability and that industry experience including operations, maintenance, and monitoring activities were assessed in estimating the reliability of these SSCs. For an as-built component with reliability less than the assumed reliability an evaluation shall show that the net effect of as-built component reliabilities does not reduce the overall reliability. Or, an evaluation shall show that there is not a significant adverse effect on the core melt frequency or the large release frequency in the PRA applicable to the plant.