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: 412-374-6306
Direct fax: 412-374-5005
e-mail: sterdia@westinghouse.com

Your ref: Project Number 740
Our ref: DCP/NRC1952

July 27, 2007

Subject: AP1000 COL Standard Technical Report Submittal of APP-GW-GLN-122 (TR 122), Revision 0

Submittal of Proprietary and Non-Proprietary Technical Document Information, APP-GW-GLN-122, Revision 0 and APP-GW-GLN-122-NP, Revision 0, "Offsite and Control Room Dose Changes"

In support of Combined License application pre-application activities, Westinghouse is submitting Revision 0 of AP1000 Standard Combined License Technical Report Number 122. This report identifies and justifies standard changes to the AP1000 Design Control Document (DCD). The changes to the DCD identified in Technical Report 122 are included in the proposed amendment to the AP1000 Design Certification Rule (DCD Revision 16) with the exception of those editorial corrections discussed in Section III of the Technical Report. This report is submitted as part of the NuStart Bellefonte COL Project (NRC Project Number 740). The information included in this report is generic and is expected to apply to all COL applications referencing the AP1000 Design Certification.

The purpose for submittal of this report was explained in a March 8, 2006 letter from NuStart to the NRC.

Pursuant to 10 CFR 50.30(b), APP-GW-GLN-122, Revision 0 (Proprietary) and APP-GW-GLN-122-NP (Non-Proprietary), (Technical Report Number 122) are submitted as Enclosures 3 and 4 under the attached Oath of Affirmation.

It is expected that when the NRC review of Technical Report Number 122, the changes to the DCD identified in Technical Report 122 will be considered approved generically for COL applicants referencing the AP1000 Design Certification.

Also enclosed is one copy of the Application for Withholding, AW-07-2306 (non-proprietary) with Proprietary Information Notice, and one copy of the associated Affidavit (non-proprietary).

This submittal contains proprietary information of Westinghouse Electric Company, LLC. In conformance with the requirements of 10 CFR Section 2.390, as amended, of the Commission's regulations, we are enclosing with this submittal an Application for Withholding from Public

DO79
D063
NRO

Disclosure and an affidavit. The affidavit sets forth the basis on which the information identified as proprietary may be withheld from public disclosure by the Commission.

Correspondence with respect to the affidavit or Application for Withholding should reference AW-07-2306 and should be addressed to James A. Gresham, Manager, Regulatory Compliance and Plant Licensing, Westinghouse Electric Company, LLC, P.O. Box 355, Pittsburgh, Pennsylvania 15230-0355.

Questions or requests for additional information related to the content and preparation of these reports 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.

Westinghouse requests NRC to provide a schedule for review of this Technical Report within two weeks of its submittal.

Very truly yours,

D. F. Hatchings for

A. Sterdis, Manager
Licensing & Customer Interface
Regulatory Affairs and Standardization

/Attachment

1. "Oath of Affirmation," dated July 27, 2007

/Enclosures

1. AW-07-2306 "Application for Withholding Proprietary Information from Disclosure," dated July 27, 2007

2. AW-07-2306, Affidavit, Proprietary Information Notice, Copyright Notice dated July 27, 2007

3. APP-GW-GLN-122, Revision 0 (Proprietary), "Offsite and Control Room Dose Changes," Technical Report Number 122

4. APP-GW-GLN-122-NP, Revision 0 (Non-Proprietary), "Offsite and Control Room Dose Changes," Technical Report Number 122

cc:	D. Jaffe	-	U.S. NRC	1E	1A
	E. McKenna	-	U.S. NRC	1E	1A
	S. Adams	-	Westinghouse	1E	1A
	G. Curtis	-	TVA	1E	1A

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P. Hastings	-	Duke Power	1E	1A
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D. Lindgren	-	Westinghouse	1E	1A
A. Monroe	-	SCANA	1E	1A
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C. Pierce	-	Southern Company	1E	1A
E. Schmiech	-	Westinghouse	1E	1A
G. Zinke	-	NuStart/Entergy	1E	1A
C. Brockhoff	-	Westinghouse	1E	1A

OATH OF AFFIRMATION

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

In the Matter of:	)
NuStart Bellefonte COL Project	)
NRC Project Number 740	)

APPLICATION FOR REVIEW OF
"AP1000 GENERAL COMBINED LICENSE INFORMATION"
FOR COL APPLICATION PRE-APPLICATION REVIEW

W. E. Cummins, being duly sworn, states that he is Vice President, Regulatory Affairs & Standardization, for Westinghouse Electric Company; that he is authorized on the part of said company to sign and file with the Nuclear Regulatory Commission this document; that all statements made and matters set forth therein are true and correct to the best of his knowledge, information and belief.

W. E. Cummins
Vice President
Regulatory Affairs & Standardization

Subscribed and sworn to
before me this 27th day
of July 2007.

COMMONWEALTH OF PENNSYLVANIA
Notarial Seal
Patricia S. Aston, Notary Public
Murrysville Boro, Westmoreland County
My Commission Expires July 11, 2011
Member, Pennsylvania Association of Notaries

Notary

ENCLOSURE 1

AW-07-2306

APPLICATION FOR WITHHOLDING
PROPRIETARY INFORMATION FROM DISCLOSURE

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

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

Direct tel: 412-374-6306
Direct fax: 412-374-5005
e-mail: sterdia@westinghouse.com

Your ref: Project Number 740
Our ref: AW-07-2306

July 27, 2007

APPLICATION FOR WITHHOLDING PROPRIETARY INFORMATION FROM PUBLIC DISCLOSURE

Subject: Submittal of Proprietary and Non-Proprietary Technical Document Information, APP-GW-GLN-122, Revision 0 and APP-GW-GLN-122-NP, Revision 0, "Offsite and Control Room Dose Changes"

The Application for Withholding is submitted by Westinghouse Electric Company, LLC (Westinghouse), pursuant to the provisions of Paragraph (b) (1) of Section 2.390 of the Commission's regulations. It contains commercial strategic information proprietary to Westinghouse and customarily held in confidence.

The proprietary material for which withholding is being requested is identified in the proprietary version of the subject report. In conformance with 10 CFR Section 2.390, Affidavit AW-07-2306 accompanies this Application for Withholding, setting forth the basis on which the identified proprietary information may be withheld from public disclosure.

Accordingly, it is respectively requested that the subject information which is proprietary to Westinghouse be withheld from public disclosure in accordance with 10 CFR Section 2.390 of the Commission's regulations.

Correspondence with respect to this Application for Withholding or the accompanying affidavit should reference AW-07-2306 and should be addressed to James A. Gresham, Manager, Regulatory Compliance and Plant Licensing, Westinghouse Electric Company, LLC, P.O. Box, Pittsburgh, Pennsylvania, 15230-0355.

Very truly yours,

James W. Winters
Manager
Standardization and Configuration Management

cc: J. Thompson - U.S. NRC

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ENCLOSURE 2

Affidavit

AW-07-2306
July 27, 2007

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF ALLEGHENY:

Before me, the undersigned authority, personally appeared James W. Winters, who, being by me duly sworn according to law, deposes and says that he is authorized to execute this Affidavit on behalf of Westinghouse Electric Company LLC (Westinghouse), and that the averments of fact set forth in this Affidavit are true and correct to the best of his knowledge, information, and belief:

James W. Winters
Manager
Standardization and Configuration Management

Sworn to and subscribed
before me this 27th day
of July 2007.

COMMONWEALTH OF PENNSYLVANIA
Notarial Seal
Patricia S. Aston, Notary Public
Murrysville Boro, Westmoreland County
My Commission Expires July 11, 2011
Member, Pennsylvania Association of Notaries

Notary Public

(1) I am Manager, Standardization and Configuration Management, Westinghouse Electric Company, LLC (Westinghouse), and as such, I have been specifically delegated the function of reviewing the proprietary information sought to be withheld from public disclosure in connection with nuclear power plant licensing and rule making proceedings, and am authorized to apply for its withholding on behalf of Westinghouse.

(2) I am making this Affidavit in conformance with the provisions of 10 CFR Section 2.390 of the Commission's regulations and in conjunction with the Westinghouse "Application for Withholding" accompanying this Affidavit.

(3) I have personal knowledge of the criteria and procedures utilized by Westinghouse in designating information as a trade secret, privileged or as confidential commercial or financial information.

(4) Pursuant to the provisions of paragraph (b)(4) of Section 2.390 of the Commission's regulations, the following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld.

(i) The information sought to be withheld from public disclosure is owned and has been held in confidence by Westinghouse.

(ii) The information is of a type customarily held in confidence by Westinghouse and not customarily disclosed to the public. Westinghouse has a rational basis for determining the types of information customarily held in confidence by it and, in that connection, utilizes a system to determine when and whether to hold certain types of information in confidence. The application of that system and the substance of that system constitutes Westinghouse policy and provides the rational basis required.

Under that system, information is held in confidence if it falls in one or more of several types, the release of which might result in the loss of an existing or potential competitive advantage, as follows:

(a) The information reveals the distinguishing aspects of a process (or component, structure, tool, method, etc.) where prevention of its use by any of Westinghouse's competitors without license from Westinghouse constitutes a competitive economic advantage over other companies.

(b) It consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), the application of which data secures a competitive economic advantage, e.g., by optimization or improved marketability.

(c) Its use by a competitor would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing a similar product.

(d) It reveals cost or price information, production capacities, budget levels, or commercial strategies of Westinghouse, its customers or suppliers.

(e) It reveals aspects of past, present, or future Westinghouse or customer funded development plans and programs of potential commercial value to Westinghouse.

(f) It contains patentable ideas, for which patent protection may be desirable.

There are sound policy reasons behind the Westinghouse system which include the following:

(a) The use of such information by Westinghouse gives Westinghouse a competitive advantage over its competitors. It is, therefore, withheld from disclosure to protect the Westinghouse competitive position.

(b) It is information that is marketable in many ways. The extent to which such information is available to competitors diminishes the Westinghouse ability to sell products and services involving the use of the information.

(c) Use by our competitor would put Westinghouse at a competitive disadvantage by reducing his expenditure of resources at our expense.

(d) Each component of proprietary information pertinent to a particular competitive advantage is potentially as valuable as the total competitive advantage. If competitors acquire components of proprietary information, any one component

may be the key to the entire puzzle, thereby depriving Westinghouse of a competitive advantage.

(e) Unrestricted disclosure would jeopardize the position of prominence of Westinghouse in the world market, and thereby give a market advantage to the competition of those countries.

(f) The Westinghouse capacity to invest corporate assets in research and development depends upon the success in obtaining and maintaining a competitive advantage.

(iii) The information is being transmitted to the Commission in confidence and, under the provisions of 10 CFR Section 2.390, it is to be received in confidence by the Commission.

(iv) The information sought to be protected is not available in public sources or available information has not been previously employed in the same original manner or method to the best of our knowledge and belief.

(v) The proprietary information sought to be withheld in this submittal is that which is appropriately marked in APP-GW-GLN-122, Rev. 0, "Offsite and Control Room Dose Changes" (Proprietary), dated July 2007, in support of Combined License application pre-application activities for the NuStart Bellefonte COL Project being transmitted by Westinghouse letter (DCP/NRC1952) and Application for Withholding Proprietary Information from Public Disclosure, to the Document Control Desk. The proprietary information as submitted by Westinghouse for the AP1000 NuStart Bellefonte plant is expected to be applicable in other licensee submittals in response to certain NRC requirements for justification of compliance of the safety system to regulations.

This information is part of that which will enable Westinghouse to:

(a) Manufacture and deliver products to utilities based on proprietary system designs.

(b) Advance the AP1000 Design and reduce the licensing risk for the application of the AP1000 Design Certification

(c) Determine compliance with regulations and standards

(d) Establish design requirements and specifications for the system.

Further this information has substantial commercial value as follows:

(a) Westinghouse plans to sell the use of similar information to its customers for purposes of plant construction and operation.

(b) Westinghouse can sell support and defense of safety systems based on the technology in the reports.

(c) The information requested to be withheld reveals the distinguishing aspects of an approach and schedule which was developed by Westinghouse.

Public disclosure of this proprietary information is likely to cause substantial harm to the competitive position of Westinghouse because it would enhance the ability of competitors to provide similar digital technology safety systems and licensing defense services for commercial power reactors without commensurate expenses. Also, public disclosure of the information would enable others to use the information to meet NRC requirements for licensing documentation without purchasing the right to use the information.

The development of the technology described in part by the information is the result of applying the results of many years of experience in an intensive Westinghouse effort and the expenditure of a considerable sum of money.

In order for competitors of Westinghouse to duplicate this information, similar technical programs would have to be performed and a significant manpower effort, having the requisite talent and experience, would have to be expended.

Further the deponent sayeth not.

PROPRIETARY INFORMATION NOTICE

Transmitted herewith are proprietary and/or non-proprietary versions of documents furnished to the NRC in connection with requests for generic and/or plant-specific review and approval.

In order to conform to the requirements of 10 CFR 2.390 of the Commission's regulations concerning the protection of proprietary information so submitted to the NRC, the information which is proprietary in the proprietary versions is contained within brackets, and where the proprietary information has been deleted in the non-proprietary versions, only the brackets remain (the information that was contained within the brackets in the proprietary versions having been deleted). The justification for claiming the information so designated as proprietary is indicated in both versions by means of lower case letters (a) through (f) located as a superscript immediately following the brackets enclosing each item of information being identified as proprietary or in the margin opposite such information. These lower case letters refer to the types of information Westinghouse customarily holds in confidence identified in Sections (4)(ii)(a) through (4)(ii)(f) of the affidavit accompanying this transmittal pursuant to 10 CFR 2.390(b)(1).

COPYRIGHT NOTICE

ENCLOSURE 4

Westinghouse Non-Proprietary

APP-GW-GLN-122-NP, Revision 0

"Offsite and Control Room Dose Changes"

Technical Report 122

F-AP-3.2-1 Rev. 0

AP1000 DOCUMENT COVER SHEET

TDC: _____ Permanent File: _____ APY: _____

RFS#: _____ RFS ITEM #: _____

AP1000 DOCUMENT NO.	REVISION NO.		ASSIGNED TO
APP-GW-GLN-122-NP	0	Page 1 of ~~27~~ 30	W-A. Sterdis

ALTERNATE DOCUMENT NUMBER: TR122 WORK BREAKDOWN #:

ORIGINATING ORGANIZATION: Westinghouse Electric Company

TITLE: **Offsite and Control Room Dose Changes**

ATTACHMENTS:	DCP #/REV. INCORPORATED IN THIS DOCUMENT REVISION: APP-GW-GEE-234 Rev. 1
CALCULATION/ANALYSIS REFERENCE:	

ELECTRONIC FILENAME	ELECTRONIC FILE FORMAT	ELECTRONIC FILE DESCRIPTION
APP-GW-GLN-122 Rev 0.doc	Microsoft Word	

(C) WESTINGHOUSE ELECTRIC COMPANY LLC – 2007

☒ WESTINGHOUSE CLASS 3 (NON PROPRIETARY)

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

LEGAL REVIEW	SIGNATURE/DATE
J. C. VALENTINE	J. C. Valentine 7/13/2007
PATENT REVIEW	SIGNATURE/DATE
Daniel J McDermott DJM 7/13/07	DJM 7/13/07

☐ 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.

ORIGINATOR	SIGNATURE/DATE
C. S. Brockhoff	7/11/07
REVIEWERS	SIGNATURE/DATE
	7/11/07 CB

VERIFIER	SIGNATURE/DATE	VERIFICATION METHOD
U. Bachrach (dose calculations)	7/11/07	Detailed Verification
P. L. Greco (VES and TS changes)	7/11/07	Page by page
AP1000 RESPONSIBLE MANAGER C. A. McGinnis	SIGNATURE*	APPROVAL DATE 7-12-07

* Approval of the responsible manager signifies that document is complete, all required reviews are complete, electronic file is attached and document is released for use.

APP-GW-GLN-122-NP
Revision 0

July 2007

Offsite and Control Room Dose Changes

Westinghouse Electric Company LLC
P.O. Box 355
Pittsburgh, PA 15230-0355

TABLE OF CONTENTS

Brief Description of the change (what is being changed and why):
Three changes are being implemented that reduce some of the calculated radiological doses offsite and in the main control room for design basis accidents. The changes allow the use of increased atmospheric dispersion factors (X/Q), which are appropriate based on current industry data with more realistic / updated radioactive material transport factors.

I. APPLICABILITY DETERMINATION

This evaluation is prepared to document that the change described above is a departure from Tier 2 information of the AP1000 Design Control Document (DCD) that may be included in plant specific FSARs without prior NRC approval.

A.	Does the proposed change include a change to:		
	1. Tier 1 of the AP1000 Design Control Document APP-GW-GL-700	☐ NO ☒ YES	(If YES prepare a report for NRC review of the changes)
	2. Tier 2* of the AP1000 Design Control Document, APP-GW-GL-700	☒ NO ☐ YES	(If YES prepare a report for NRC review of the changes)
	3. Technical Specification in Chapter 16 of the AP1000 Design Control Document, APP-GW-GL-700	☐ NO ☒ YES	(If YES prepare a report for NRC review of the changes)
B.	Does the proposed change involve:		
	1. Closure of a Combined License Information Item identified in the AP1000 Design Control Document, APP-GW-GL-700	☒ NO ☐ YES	(If YES prepare a COL item closure report for NRC review.)
	2. Completion of an ITAAC item identified in Tier 1 of the AP1000 Design Control Document, APP-GW-GL-700	☒ NO ☐ YES	(If YES prepare an ITAAC completion report for NRC review.)

☐ The questions above are answered no, therefore the departure from the DCD in a COL application does not require prior NRC review unless review is required by the criteria of 10 CFR Part 52 Appendix D Section VIII B.5.b. or B.5c

II. TECHNICAL DESCRIPTION AND JUSTIFICATION

INTRODUCTION

Three changes are being implemented that reduce some of the calculated radiological doses offsite and in the main control room for design basis accidents. The changes allow the use of increased atmospheric dispersion factors (X/Q), which are appropriate based on current industry data with more realistic / updated radioactive material transport factors.

Therefore, the offsite and main control room doses were evaluated to determine what design and/or analytical changes may be required to support the anticipated X/Q values.

Radiological dose calculations were re-evaluated for the low population zone, the site boundary, and the main control room, incorporating these three changes.

The first change is to direct the Main Control Room Emergency Habitability System (VES) discharge air flow into the entry vestibule to provide a continuous vestibule purge. This helps to reduce the radioactivity introduced into the main control room each time there is access to or from the main control room during a radiological accident. This design change also provides vent discharge openings between the vestibule and the outside hallway to provide a flowpath for the air passing from the main control room through the discharge dampers into the vestibule.

The second change is to increase the decay time in Technical Specification 3.9.7 from 24 hours to 48 hours, to provide increased radioactive decay of short-lived fission products prior to handling of irradiated fuel assemblies. This helps to reduce the source radioactivity levels for fuel handling accidents.

The third change is to revise the calculation of radioactivity released from containment leakage for the postulated loss-of-coolant accident with core melt to take credit for the reduction in the release of aerosols that occurs in the leakage through small containment cracks. [

]^{a,c} Based on a highly conservative calculation of the potential DF, the LOCA dose analysis conservatively assumes a DF of only 5.

These three changes allow some of the X/Q values shown in the DCD to be increased to accommodate sites with higher values than originally specified in DCD Chapters 2 and 15.

In accordance with WCAP-7211, the "[xx]" used in this report indicate information that is considered proprietary. The superscripts are defined as follows:

"a" Information that reveals the distinguishing aspects of a process or component, structure, tool, method, etc., and the prevention of its use by competitors provides a competitive economic advantage.

"c" Information, if used by a competitor, would reduce the competitor's expenditure of resources or improve the competitor's advantage in the design, manufacture, etc., of a similar product.

BACKGROUND

The following sections provide background information on each of the three changes to the plant design and safety analyses needed to support the increased site X/Q values, which were used in calculating new offsite and main control room doses for the design basis accidents.

Main Control Room Emergency Habitability System (VES) Purge Design Change

This system design change was implemented to reduce the potential doses to the operators in the main control room by reconfiguring the vent path out of the main control room when the main control room emergency habitability system (VES) purge is operating.

The VES discharge dampers originally discharged through the main control room wall directly to the atmosphere outside of the main control room envelope. This design change re-directs the damper discharge flow into the main control room vestibule, through the dampers that are now located adjacent to the access door into the main control room. The design change also adds openings from the vestibule to the hallway to allow free passage of the air coming from the main control room, adjacent to the vestibule access door into the hallway.

This configuration change uses the main control room purge air flow to provide a continuous purge of the vestibule area after the relatively clean air leaves the main control room and direct this air through the vestibule before it is discharged to the outside atmosphere.

Once plant personnel enter the vestibule from the outside prior to entering the main control room, they may be directed to wait for some specified period of time in the event of high radioactivity for the vestibule purge flow to reduce vestibule radioactivity levels. (Note that the design basis LOCA dose analysis assumes that, during the first 24 hours post-accident, any personnel entering the main control room would spend approximately 11 minutes in the entry vestibule to allow the purge to reduce the air concentration by a factor of five. This results in the reduction in unfiltered inleakage to the main control room from the 5 cfm associated with no vestibule purge to 0.531 cfm. With no credit for a waiting period in the vestibule, the effective unfiltered inleakage is 2.654 cfm.) The personnel may also be directed to use portable equipment to monitor airborne levels and wait for specified levels or activity reductions to be reached prior to opening the main control room access door. Depending on the access frequency and outside airborne radioactivity levels, these access requirements could potentially vary and relaxations could be made at lower outside radioactivity levels.

Therefore, the improvement to provide the vestibule purge, supplemented by main control room access procedures during a radioactivity release event, will help to reduce the radioactivity levels in the main control room and the control room dose for a plant accident with no required change in the VES storage tank capacity.

Fuel Handling Accident Decay Time Increase

Technical Specification 3.9.7 specifies the minimum radioactive decay time to ensure that the radiological consequences of a postulated fuel handling accident inside containment or in the fuel handling area inside the auxiliary building are consistent with the assumptions in DCD Chapter 15. This time provides for sufficient radioactive decay of short-lived fission products to limit the doses from the accident.

The minimum decay time was originally specified for AP1000 as 100 hours (which was consistent with the typical decay time value used in Technical Specifications for many years for operating plants). The AP1000 decay time was subsequently decreased to 24 hours to be consistent with updated AP1000 fuel handling accident dose analysis.

To support the desired increase in site boundary X/Q, the decay time assumed for the fuel handling accident is being increased to 48 hours. Technical Specification 3.9.7 is being revised to specify 48 hours for decay before fuel can be moved and thus will be consistent with the analysis.

This new decay time remains conservative relative to refueling outage activities such that it is not likely that this change has any impact on refueling outage critical path activities.

Aerosol Plugging in AP1000 Containment Leak Paths for the LOCA Dose Analysis

Background and Related Testing Discussion

There are a number of scientifically sound aerosol removal mechanisms that are assumed to operate in the AP1000 containment following a postulated LOCA with core melt. [

]a,c

Application to AP1000

[

]a,c

The following is a listing of different potential leak paths / characteristics for the AP1000:

Leak Path	Location Relative to Operating Deck	Geometry	Wet Inside	Wet Outside	Discharges to
Crack in pressure vessel weld	Above	1 rough slit 1.75" long	yes	yes	PCS air discharge to environment
Leak past equipment hatch gasket	Above or below	2 smooth short slits in series	yes	no	Aux bldg
Crack in penetration weld	Below	1 rough slit 1.75" long	yes	no	Aux bldg
Leak through cont. isolation valves	Below	2 smooth short slits in series	yes	no	Aux bldg
Leak through electrical penetration	Below	2 smooth short slits in series	Yes	no	Aux bldg

[

]a,c

AP1000 Aerosol Plugging and Removal Conclusions

For the anticipated AP1000 accident conditions, plugging is expected to occur with a significant DF. An aerosol removal DF of 5 was conservatively credited for the containment leakage pathway in the radiological consequences analysis for the loss-of-coolant accident.

[

]^{a,c}

Conclusions

The attached DCD markups reflect the increased X/Q values used for the dose re-evaluation.

The revised accident doses resulting from these three design and analysis changes for the four radiological accidents of interest are compared to the previously calculated values in the table below:

	Site Boundary		Low Population Zone		Control Room (VES operating)	
	Old	New	Old	New	Old	New
LOCA	24.6	11.5	23.8	14.2	4.77	2.96
FHA	5.6	5.2	3.5	2.6	4.5	3.1
REA	2.9	3.6	5.5	5.5	2.1	2.3
SLBOC	1.7	2.1	1.1	1.1	1.2	1.4

RECOMMENDATION

It is recommended that the X/Q values provided in the DCD be revised as indicated and that the calculated radiological safety analyses reflect the updated offsite and main control room dose values.

REFERENCES

None.

III. DCD MARK-UP

See the attached DCD markup pages in Attachment A.

The DCD dose calculations in DCD 6.4.4 were updated to identify a single bounding main control room accident dose for each event. For these events, the Nuclear Island Nonradioactive Ventilation System (VBS) is assumed to initially operate without credit for air filtration (which maximizes main control room radioactivity inputs) for these accidents until the VBS is isolated and the VES is actuated. The VES operation maintains acceptable main control room doses with or without credit for VBS operation.

The VBS operating cases in DCD 6.4.4 were removed since they assumed continuing VBS operation after VES actuation, which is not realistic based on actual VBS operation. The VBS normally operates in a mode that provides conditioned outside air to the main control room. If high radioactivity is detected in the VBS

main control room air supply duct, the VBS automatically shifts into a recirculation / supplemental air filtration mode that pressurizes the main control room. On a high-high radioactivity in the VBS main control room air supply duct or if ac power is unavailable, the main control room is isolated from the VBS and the VES is actuated to pressurize the main control room. Therefore, it is unnecessary to evaluate doses for the second set of VBS operating cases since VBS does not operate in parallel with VES.

Note that are five editorial errors in the DCD Revision 16 changes that do not match the changes identified in the DCD markup section for this technical report. Therefore, the following DCD Revision 16 pages should be corrected to match the associated markups in this report:

- Tier 1, Table 5.0-1, Atmospheric Dispersion Factors (LPZ Boundary X/Q value changes missed)
- Tier 2, Table 2-1, Atmospheric Dispersion Factors (LPZ Boundary X/Q value changes missed)
- Tier 2, Table 14.3-7 (Sheet 1 of 3), Section 2.3.4 Atmospheric Dispersion Factors (LPZ Boundary X/Q values incorrectly changed)

- Tier 2, page 6.4-5, Pressure Relief Damper (incorrect text in insert paragraph)

- Tier 2, Table 15.6.5-2 (Sheet 2 of 3) (previous effective unfiltered inleakage via ingress/egress value not deleted)

- Tier 2, Table 15.6.5-3 (incorrect value in change for airborne activity entering the main control room)

- Tier 2, Technical Specification 3.9.7 Bases, Applicable Safety Analyses (missed changing "100" to "48" hours in the second paragraph)

IV. REGULATORY IMPACT

A. FSER IMPACT

The AP1000 FSER information related to DCD Subsections 6.4.2, 6.4.4, 15.1.5, 15.3.3, 15.4.8, 15.6.2, 15.6.3, 15.6.5, and 16.7.4, and the X/Q values in Table 2-1, 15A-5, and 15A-6 may be impacted by these three changes and the resulting re-calculation of the design basis accident doses.

B. SCREENING QUESTIONS (Check correct response and provide justification for that determination under each response)

1. Does the proposed change involve a change to an SSC that adversely affects a DCD described design function? ☐ YES ☒ NO

2. Does the proposed change involve a change to a procedure that adversely affects how DCD described SSC design functions are performed or controlled? ☐ YES ☒ NO

3. Does the proposed activity involve revising or replacing an DCD described evaluation methodology that is used in establishing the design bases or used in the safety analyses? ☒ YES ☐ NO

4. Does the proposed activity involve a test or experiment not described in the DCD, where an SSC is utilized or controlled in a manner that is outside the reference bounds of the design for that SSC or is inconsistent with analyses or descriptions in the DCD? ☐ YES ☒ NO

C. 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. 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

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

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

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

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

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

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

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 answers to the evaluation questions above are "NO" and the proposed departure from Tier 2 does not require prior NRC review to be included in plant specific FSARs as provided in 10 CFR Part 52, Appendix D, Section VIII. B.5.b

☒ One or more of the answers to the evaluation questions above are "YES" and the proposed change requires NRC review.

D. 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 severe accidents. If the answer is Yes answer Questions 2 and 3 below. ☐ YES ☒ NO

2. Is there is a substantial increase in the probability of a severe accident such that a particular 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 severe accident previously reviewed? ☐ YES ☒ NO ☐ N/A

☒ The answers to the evaluation questions above are "NO" or are not applicable and the proposed departure from Tier 2 does not require prior NRC review to be included in plant specific FSARs as provided in 10 CFR Part 52, Appendix D, Section VIII. B.5.c

☐ One or more of the he answers to the evaluation questions above are "YES" and the proposed change requires NRC review.

E. SECURITY ASSESSMENT

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

Attachment A

DCD Markups

1. Tier 1, Table 5.0-1, Site Parameters, pgs 5.0-3 and 5.0-5

Atmospheric Dispersion Factors (X/Q)	
Site Boundary (0-2 hr)	$\leq$ ~~5.1~~1.0 x 10^{-3} sec/m^3
Site Boundary (annual average)	$\leq$ 2.0 x 10^{-5} sec/m^3
Low Population Zone Boundary	
0 - 8 hr	$\leq$ ~~1.2~~5.0 x 10^{-4} sec/m^3
8 - 24 hr	$\leq$ ~~1.6~~3.0 x 10^{-4} sec/m^3
24 - 96 hr	$\leq$ ~~1.0~~1.5 x 10^{-4} sec/m^3
96 - 720 hr	$\leq$ 8.0 x 10^{-5} sec/m^3

Table 5.0-1 (cont.)
Site Parameters

Control Room Atmospheric Dispersion Factors (χ/Q) for Accident Dose Analysis					
χ/Q (s/m^3) at HVAC Intake for the Identified Release Points[3]					
	Plant Vent or PCS Air Diffuser[3]	**Ground Level Containment Release Points[4]**	**PORV and Safety Valve Releases[5]**	**Steam Line Break Releases**	**Fuel Handling Area[4]**
0 - 2 hours	~~2.2~~3.0E-3	~~2.2~~5.0E-3	2.0E-3	2.4E-3	6.0E-3
2 - 8 hours	~~1.4~~2.5E-3	~~1.4~~4.5E-3	1.8E-3	2.0E-3	4.0E-3
8 - 24 hours	~~6~~1.0E-~~3~~	~~6~~2.0E-~~3~~	7.0E-3	7.5E-3	2.0E-3
1 - 4 days	~~4.5~~8.0E-4	~~4.5~~1.8E-~~3~~	5.0E-3	5.5E-3	1.5E-3
4 - 30 days	~~2.6~~.0E-4	~~3.6~~1.5E-~~3~~	4.5E-3	5.0E-3	1.0E-3
χ/Q (s/m^3) at Control Room Door for the Identified Release Points[1]					
0 - 2 hours	~~6.6~~1.0E-~~3~~	~~6.6~~1.0E-~~3~~	4.0E-3	4.0E-3	5.0E-3
2 - 8 hours	~~4.8~~7.5E-4	~~4.8~~7.5E-4	3.2E-3	3.2E-3	4.0E-3
8 - 24 hours	~~2.1~~3.5E-4	~~2.1~~3.5E-4	1.2E-3	1.2E-3	2.0E-3
1 - 4 days	~~1.5~~2.8E-4	~~1.5~~2.8E-4	1.0E-3	1.0E-3	1.5E-3
4 - 30 days	~~1.2~~2.5E-4	~~1.2~~2.5E-4	8.0E-4	8.0E-4	1.0E-3

Notes:

2. Tier 2, Table 2-1 (Sheets 2 and 3), Site Parameters, pgs 2-15 and 2-16

Table 2-1 (Sheet 2 of 3)

SITE PARAMETERS

Plant Grade Elevation	Less than plant elevation 100' except for portion at a higher elevation adjacent to the annex building
Precipitation	
Rain	19.4 in./hr (5.3 in./5 min)
Snow/Ice	75 pounds per square foot on ground with exposure factor of 1.0 and importance factors of 1.2 (safety) and 1.0 (non-safety)
Atmospheric Dispersion Values - χ/Q[1]	
Site boundary (0-2 hr)	~~≤ 6.3 x 10^{-4}~~ ≤ 1.0 x 10^{-3} sec/m^3
Site boundary (annual average)	≤ 2.0 x 10^{-5} sec/m^3
Low population zone boundary	
0 - 8 hr	~~≤ 3.2~~≤ 5.0 x 10^{-4} sec/m^3
8 - 24 hr	~~≤ 1.6~~≤ 3.0 x 10^{-4} sec/m^3
24 - 96 hr	~~≤ 1.0~~≤ 1.5 x 10^{-4} sec/m^3
96 - 720 hr	≤ 8.0 x 10^{-5} sec/m^3
Population Distribution	
Exclusion area (site)	0.5 mi

Table 2-1 (Sheet 3 of 3)

SITE PARAMETERS

Control Room Atmospheric Dispersion Factors (χ/Q) for Accident Dose Analysis

χ/Q (s/m^3) at HVAC Intake for the Identified Release Points[1]

	Plant Vent or PCS Air Diffuser[3]	Ground Level Containment Release Points[4]	PORV and Safety Valve Releases[5]	Steam Line Break Releases	Fuel Handling Area[6]
0 - 2 hours	~~3.2~~3.0E-3	~~3.2~~5.0E-3	2.0E-2	2.4E-2	6.0E-3
2 - 8 hours	~~1.4~~2.5E-3	~~1.4~~4.5E-3	1.8E-2	2.0E-2	4.0E-3
8 - 24 hours	~~6~~1.0E-~~3~~	~~6~~2.0E-~~3~~	7.0E-3	7.5E-3	2.0E-3
1 - 4 days	~~4.5~~8.0E-4	~~4.5~~1.8E-~~3~~	5.0E-3	5.5E-3	1.5E-3
4 - 30 days	~~2.6~~5.0E-4	~~3.6~~1.5E-~~3~~	4.5E-3	5.0E-3	1.0E-3

χ/Q (s/m^3) at Control Room Door for the Identified Release Points[2]

	Plant Vent or PCS Air Diffuser[3]	Ground Level Containment Release Points[4]	PORV and Safety Valve Releases[5]	Steam Line Break Releases	Fuel Handling Area[6]
0 - 2 hours	~~6.6~~1.0E-~~3~~	~~6.6~~1.0E-~~3~~	4.0E-3	4.0E-3	6.0E-3
2 - 8 hours	~~4.8~~7.5E-4	~~4.8~~7.5E-4	3.2E-3	3.2E-3	4.0E-3
8 - 24 hours	~~2.1~~3.5E-4	~~2.1~~3.5E-4	1.2E-3	1.2E-3	2.0E-3
1 - 4 days	~~1.5~~2.8E-4	~~1.5~~2.8E-4	1.0E-3	1.0E-3	1.5E-3
4 - 30 days	~~1.5~~2.5E-4	~~1.5~~2.5E-4	8.0E-4	8.0E-4	1.0E-3

3. Tier 2, DCD 6.4.2.3, pg 6.4-5

- Pressure Relief Damper

Pressure relief dampers are located downstream of the butterfly isolation valves, and are set to open on a differential pressure of at least 1/8-inch water gauge with respect to the surrounding areas. The differential pressure between the control room and the relief damper exhaust location is monitored to ensure that a positive pressure is maintained in the control room with respect to its surroundings.

The pressure relief dampers discharge through the MCR vestibule in order to reduce the amount of radioactivity that can be transported into the MCR when operators enter. Two vestibule discharge openings provide a purge flow path from the vestibule to the corridor.

4. Tier 2, DCD 6.4.2.4, pg 6.4-6

The main control room pressure boundary main entrance is designed with an airlock-type double-door vestibule, which is purged by the pressure relief damper discharge flow during main control room emergency habitability system operation. The emergency exit door (stairs to elevation 100') is normally closed, and remains closed under design basis source term conditions.

5. Tier 2, DCD 6.4.3.2, pg 6.4-7

If radiation levels in the main control room supply air duct exceed the "high-high" setpoint, the nuclear island nonradioactive ventilation system is isolated from the main control room pressure boundary by automatic closure of the isolation devices located in the nuclear island nonradioactive ventilation system ductwork. At the same time, the main control room emergency habitability system begins to deliver air from the emergency air storage tanks to the main control room by automatically opening the isolation valves located in the supply line. The relief damper isolation valves also open allowing the pressure relief dampers to function and discharge the damper flow to purge the vestibule.

6. Tier 2, DCD 6.4.4, pgS 6.4-8 and 6.4-9

6.4.4 System Safety Evaluation

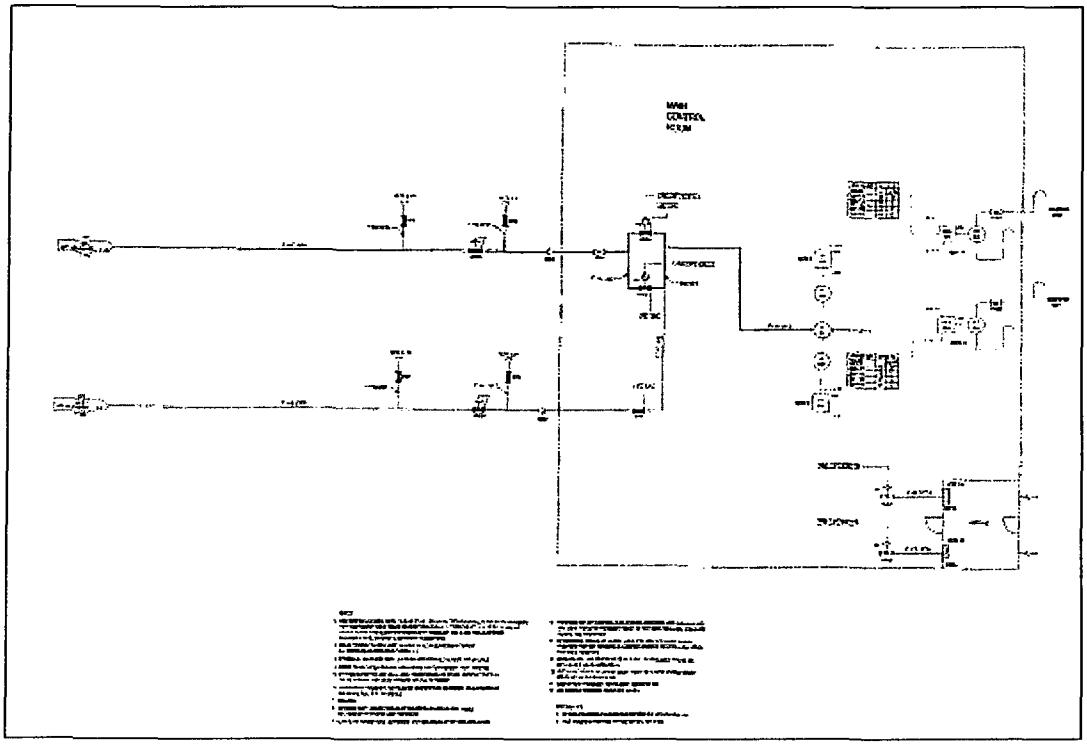
~~Doses to main control room personnel were calculated for both the situation in which the emergency habitability system (VES) is relied upon to limit the amount of activity the personnel are exposed to and the situation in which the nuclear island nonradioactive ventilation system (VBS) is available to pressurize the main control room with filtered air and provide recirculation cleanup. Doses were calculated for the following accidents~~In the event of an accident involving the release of radioactivity to the environment, the nuclear island nonradioactive ventilation system (VBS) is expected to switch from the normal operating mode to the supplemental air filtration mode to protect the main control room personnel. Although the VBS is a nonsafety related system, it is expected to be available to provide the necessary protection for realistic events. However, the accident doses were calculated based on operation of the safety-related emergency habitability system (VES), which is relied upon to limit the amount of activity the personnel are exposed to. Doses were determined for the following design basis accidents:

	~~VES Operating~~	~~VBS Operating~~
Large Break LOCA	~~1.8~~3.0 rem TEDE	~~1.5 rem TEDE~~

Fuel Handling Accident TEDE	~~4.5~~3.1 rem TEDE	~~2.4 rem~~
Steam Generator Tube Rupture		
(Pre-existing iodine spike)	4.8 rem TEDE	~~3.4 rem TEDE~~
(Accident-initiated iodine spike)	2.1 rem TEDE	~~1.8 rem TEDE~~
Steam Line Break		
(Pre-existing iodine spike)	3.4 rem TEDE	~~2.1 rem TEDE~~
(Accident-initiated iodine spike)	3.7 rem TEDE	~~1.9 rem TEDE~~
Rod Ejection Accident TEDE	~~2.1~~2.3 rem TEDE	~~1.3 rem~~
	~~VES Operating~~	~~VBS Operating~~
Locked Rotor Accident		
(Accident without feedwater available)	0.9 rem TEDE	~~0.9 rem TEDE~~
(Accident with feedwater available)	0.7 rem TEDE	~~1.6 rem TEDE~~
Small Line Break Outside Containment TEDE	~~1.2~~1.4 rem TEDE	~~0.3 rem~~

For all events the doses are within the dose acceptance limit of 5.0 rem TEDE. The details of analysis assumptions for modeling the doses to the main control room personnel are delineated in the LOCA dose analysis discussion in subsection 15.6.5.3.

7. Tier 2, DCD Figure 6.4-2 (Sheet 2 of 2), pg 6.4-22



8. Tier 2, DCD Table 14.3-7 (Sheet 1 of 3), pg 14.3-49

Table 14.3-7 (Sheet 1 of 3)

RADIOLOGICAL ANALYSIS

Reference	Design Feature	Value
Table 2-1	Plant elevation for maximum flood level (ft)	≤ 100
Section 2.3.4	Atmospheric dispersion factors - X/Q (sec/m^3)	
	- Site Boundary X/Q	
	0 - 2 hour time interval	≤ ~~8.0 x 10^{-4}~~ 1.0 x 10^{-3}
	- Low Population Zone Boundary X/Q	
	0 - 8 hours	≤ 5.0 x 10^{-4}
	8 - 24 hours	≤ 3.0 x 10^{-4}
	24 - 96 hours	≤ 1.5 x 10^{-5}
	96 - 720 hours	≤ 8.0 x 10^{-5}

9. Tier 2, DCD 15.1.5.4.6, pg 15.1-20

15.1.5.4.6 Doses

Using the assumptions from Table 15.1.5-1, the calculated total effective dose equivalent (TEDE) doses for the case with accident-initiated iodine spike are determined to be less than ~~0.8~~1.1 rem at the site boundary for the limiting 2-hour interval (0 to 2 hours) and 2.0 rem at the low population zone outer boundary. These doses are small fractions of the dose guideline of 25 rem TEDE identified in 10 CFR Part 50.34. A "small fraction" is defined, consistent with the Standard Review Plan, as being 10 percent or less. The TEDE doses for the case with pre-existing iodine spike are determined to be less than ~~0.8~~1.0 rem at the site boundary for the limiting 2-hour interval (0 to 2 hours) and 0.8 rem at the low population zone outer boundary. These doses are within the dose guidelines of 10 CFR Part 50.34.

10. Tier 2, DCD 15.3.3.3.6, pg 15.3-10

15.3.3.3.6 Doses

Using the assumptions from Table 15.3-3, the calculated total effective dose equivalent (TEDE) doses are determined to be less than ~~0.7~~0.8 rem at the exclusion area boundary for the limiting 2-hour interval (0 to 2 hours) and less than 0.4 rem at the low population zone outer boundary for the scenario in which there is no feedwater available to maintain water level in the steam generators. The doses for the scenario in which it is assumed that water level in the steam generators is maintained are ~~0.5~~0.6 rem at the exclusion area boundary for the limiting 2-hour interval of 7 to 8 hours and 0.8 rem at the low population zone outer boundary. These doses are a small fraction of the dose guideline of 25 rem TEDE identified in 10 CFR Part 50.34. A "small fraction" is identified as 10 percent or less consistent with the Standard Review Plan (Reference 4).

At the time the locked reactor coolant pump rotor event occurs, the potential exists for a coincident loss of spent fuel pool cooling with the result that the pool could reach boiling and a portion of the radioactive iodine in the spent fuel pool could be released to the environment. The loss of spent fuel pool cooling has been evaluated for a duration of 30 days. There is no contribution to the 2-hour site boundary dose because the pool boiling would not occur until after the first 2 hours. The 30-day contribution to the dose at the low population zone boundary is less than 0.01 rem TEDE, and when this is added to the dose calculated for the locked rotor event, the resulting total dose remains less than ~~0.5 rem TEDE~~the value reported above.

11. Tier 2, DCD 15.4.8.3.6, pg 15.4-38

15.4.8.3.6 Doses

Using the assumptions from Table 15.4-4, the calculated total effective dose equivalent (TEDE) doses are determined to be less than ~~2.9~~3.6 rem at the site boundary for the limiting 2-hour interval (0 to 2 hours) and less than 5.5 rem at the low population zone outer boundary. These doses are well within the dose guideline of 25 rem total effective dose equivalent identified in 10 CFR Part 50.34. The phrase "well within" is taken as being 25 percent or less.

At the time the rod ejection accident occurs, the potential exists for a coincident loss of spent fuel pool cooling with the result that the pool could reach boiling and a portion of the radioactive iodine in the spent fuel pool could be released to the environment. The loss of spent fuel pool cooling has been evaluated for a duration of 30 days. There is no contribution to the 2-hour site boundary dose because the pool boiling would not occur until after the first 2 hours. The 30-day contribution to the dose at the low population zone boundary is less than 0.01 rem TEDE, and when this is added to the dose calculated for the rod ejection accident, the resulting total dose remains less than ~~2 rem TEDE~~the value reported above.

12. Tier 2, DCD 15.6.2.6, pg 15.6-6

15.6.2.6 Doses

Using the assumptions from Table 15.6.2-1, the calculated total effective dose equivalent (TEDE) doses are determined to be ~~< 1.7~~ 2.1 rem at the exclusion area boundary and < 1.1 rem at the low population zone outer boundary. These doses are a small fraction of the dose guideline of 25 rem TEDE identified in 10 CFR Part 50.34. The phrase "a small fraction" is taken as being ten percent or less.

At the time the accident occurs, there is the potential for a coincident loss of spent fuel pool cooling with the result that the pool could reach boiling and a portion of the radioactive iodine in the spent fuel pool could be released to the environment. The loss of spent fuel pool cooling has been evaluated for a duration of 30 days. There is no contribution to the 2-hour site boundary

because pool boiling would not occur until after 2 hours. The 30-day contribution to the dose at the low population zone boundary is less than 0.01 rem TEDE and, when this is added to the dose calculated for the small line break outside containment, the resulting total dose remains less than ~~1.3 rem TEDE~~ the value reported above.

13. Tier 2, DCD 15.6.3.3.6, pg 15.6-15

15.6.3.3.6 Doses

Using the assumptions from Table 15.6.3-3, the calculated TEDE doses for the case in which the iodine spike is assumed to be initiated by the accident are determined to be less than ~~0.9~~ 1.1 rem at the exclusion area boundary for the limiting 2-hour interval (0-2 hours) and less than 0.8 rem at the low population zone outer boundary. These doses are a small fraction of the dose guideline of 25 rem TEDE identified in 10 CFR Part 50.34. A "small fraction" is defined, consistent with the Standard Review Plan, as being ten percent or less.

For the case in which the SGTR is assumed to occur coincident with a pre-existing iodine spike, the TEDE doses are determined to be less than ~~1.8~~ 2.2 rem at the exclusion area boundary for the limiting 2-hour interval (0 to 2 hours) and less than 1.3 rem at the low population zone outer boundary. These doses are within the dose guideline of 25 rem TEDE identified in 10 CFR Part 50.34.

14. Tier 2, DCD 15.6.5.3.3, pg 15.6-21

15.6.5.3.3 Release Pathways

The release pathways are the containment purge line and containment leakage. The activity releases are assumed to be ground level releases.

During the initial part of the accident, before the containment is isolated, it is assumed that containment purge is in operation and that activity is released through this pathway until the purge valves are closed. No credit is taken for the filters in the purge exhaust line.

~~The majority of the releases due to the LOCA are the result of containment leakage. The containment is assumed to leak at its design leak rate for the first 24 hours and at half that rate for the remainder of the analysis period.~~

The majority of the releases due to the LOCA are the result of containment leakage, with credit for aerosol removal from containment cracks that form the leakage paths, as discussed in Reference 31. The containment is assumed to leak at its design leak rate for the first 24 hours and at half that rate for the remainder of the analysis period.

15. Tier 2, DCD 15.6.5.3.5, pg 15.6-22

The main control room is accessed by a vestibule entrance, which restricts the volume of contaminated air that can enter the main control room from ingress and egress. ~~The equivalent inflow of unfiltered air due to airborne ingress/egress has been determined to be 5.0 cfm.~~ The design of the emergency habitability system (VES) provides 65 cfm ±5 cfm to the control room and maintains it in a pressurized state. The path for the purge flow out of the main control room is through the vestibule entrance and this results in a dilution of the activity in the vestibule and a reduction in the amount of activity that might enter the main control room. Without this purge through the vestibule, the project unfiltered inleakage into the main control room is 5 cfm. However, the impact of the purge flow is to reduce the effective unfiltered inleakage rate to 2.654 cfm. Additionally, during the first 24 hours following the LOCA, personnel entering the control room will be required to wait inside the vestibule for a short period of time until the activity concentration is reduced by a factor of five or more. This reduces the effective unfiltered inleakage rate to 0.531 cfm. Conservatively, assuming a purge flow of only 55 cfm through the vestibule, the factor of five reduction in activity concentration would be achieved in less than 11 minutes.

16. Tier 2, DCD 15.6.5.3.8.1, pg 15.6-23

15.6.5.3.8.1 Offsite Doses

The doses calculated for the exclusion area boundary and the low population zone boundary are listed in Table 15.6.5-3. The doses are within the 10 CFR 50.34 dose guideline of 25 rem TEDE.

The reported exclusion area boundary doses are for the time period of 1.24 to 3.24 hours. This is the 2-hour interval that has the highest calculated doses. The dose that would be incurred over the first 2 hours of the accident is well below the reported dose.

17. Tier 2, DCD 15.6.6, pg 15.6-56

30. Chang, S. H. et al. "A study of critical heat flux for low flow of water in vertical round tubes under low pressure," Nuclear Engineering and Design, 132, 225-237, 1991.

31. Polestar, "Decontamination Factor Associated with Aerosol Removal," Revision 0, May 2007.

18. Tier 2, DCD Table 15.6.5-2 (Sheet 1 of 3), pg 15.6-63

Containment leakage release data	
– Containment volume (ft^3)	2.06 E+05
– Containment leak rate, 0-24 hr (% per day)	0.10
– Containment leak rate, > 24 hr (% per day)	0.05
– Aerosol removal efficiency due to impaction in the containment leakage path(s)/percent	80
– Elemental iodine deposition removal coefficient (hr^{-1})	1.7
– Decontamination factor limit for elemental iodine removal	200
– Removal coefficient for particulates (hr^{-1})	See Appendix 15B

19. Tier 2, DCD Table 15.6.5-2 (Sheet 2 of 3), pg 15.6-64

– Interval with operation of the emergency habitability system	
• Flow from compressed air bottles of the emergency habitability system (cfm)	60
• ~~Effective~~ Unfiltered inleakage via ingress/egress (cfm)	~~5.0~~
• 0 - 24 hr	0.531
• > 24 hr	2.654
• Recirculation flow (cfm)	Not applicable

20. Tier 2, DCD Table 15.6.5-3, pg 15.6-66

Table 15.6.5-3

RADIOLOGICAL CONSEQUENCES OF A LOSS-OF-COOLANT ACCIDENT WITH CORE MELT

	TEDE Dose (rem)
Exclusion zone boundary dose (1.42 - 3.42 hr)[1]	~~24.6~~11.5
Low population zone boundary dose (0 - 30 days)	~~23.6~~14.2
Main control room dose (emergency habitability system in operation)	
– Airborne activity entering the main control room	~~4.6~~2.8 rem
– Direct radiation from adjacent structures	0.15 rem
– Sky-shine	0.01 rem
– Total	~~4.77~~2.86 rem
Main control room dose (normal HVAC operating in the supplemental filtration mode)[2]	
– Airborne activity entering the main control room	~~4.38~~3.91 rem
– Direct radiation from adjacent structures	0.15 rem
– Sky-shine	0.01 rem
– Total	~~4.54~~4.07 rem

Note:
1. This is the 2-hour period having the highest dose.
2. Although the dose is reported for the case in which the normal HVAC operates in the supplemental filtration mode, for the design basis accident it is projected that the system would not maintain the activity in the main control room air low enough to avoid actuation of the emergency habitability system. A "realistic" event could be addressed without actuation of the emergency habitability system.

21. Tier 2, DCD 15.7.4.1.4, pg 15.7-3

15.7.4.1.4 Radiological Decay

The fission product decay time experienced prior to the fuel handling accident is at least ~~100~~48 hours.

22. Tier 2, DCD 15.7.4.4.8 and 15.7.4.5, pg 15.7-5

15.7.4.4.8 Time Available for Radioactive Decay

The dose analysis assumes that the fuel handling accident involves one of the first fuel assemblies handled. If it were one of the later fuel handling operations, there is additional decay and a reduction in the source term.

The dose evaluation was performed assuming ~~24~~48 hours decay~~, which bounds any credible refueling operation~~.

15.7.4.5 Offsite Doses

Using the assumptions from Table 15.7-1, the calculated doses from the initial releases are determined to be 5.6~~2~~ rem TEDE at the site boundary and ~~3.52~~.6 rem TEDE at the low population zone outer boundary. These doses are well within the dose guideline of 25 rem TEDE identified in 10 CFR Part 50.34. The phrase "well within" is taken as meaning 25 percent or less.

23. Tier 2, DCD Table 15.7.1, pg 15.7-6

Table 15.7-1

ASSUMPTIONS USED TO DETERMINE
FUEL HANDLING ACCIDENT RADIOLOGICAL CONSEQUENCES

Source term assumptions	
– Core power (MWt)	3468
– Decay time (hr)	~~24~~48
Core source term after ~~24~~48 hours decay (Ci)	
I-130	~~0.550~~2.49 E+05
I-131	~~8.051~~8.26 E+07
I-132	~~1.14~~9.27 E+07~~8~~
I-133	~~0.130~~4.11 E+07
I-135	~~1.400~~1.21 E+06~~7~~
Kr-85m	~~6.505~~1.59 E+04~~5~~
Kr-85	1.05~~6~~ E+05
Kr-88	~~3.08~~5.81 E+02~~5~~
Xe-131m	1.05~~6~~ E+06
Xe-133m	~~5.21~~4.37 E+06
Xe-133	~~3.85~~1.69 E+08
Xe-135m	~~2.40~~1.94 E+05~~6~~
Xe-135	~~4.73~~1.08 E+07

24. Tier 2, DCD Table 15A-5, pg 15A-14

Table 15A-5

OFFSITE ATMOSPHERIC DISPERSION FACTORS (χ/Q) FOR ACCIDENT DOSE ANALYSIS

Site boundary χ/Q (s/m^3)	
0 – 2 hours[1]	~~5.1~~1.0x10^{-3}
Low population zone χ/Q (s/m^3)	
0 – 8 hours	~~3.2~~5.0x10^{-4}
8 – 24 hours	~~1.6~~3.0x10^{-4}
24 – 96 hours	~~1.0~~1.5x10^{-4}
96 – 720 hours	8.0x10^{-5}

25. Tier 2, DCD Table 15A-6, pg 15A-15

Table 15A-6

CONTROL ROOM ATMOSPHERIC DISPERSION FACTORS (χ/Q) FOR ACCIDENT DOSE ANALYSIS

χ/Q (s/m^3) at HVAC Intake for the Identified Release Points[1]					
	Plant Vent or PCS Air Diffuser[3]	Ground Level Containment Release Points[4]	PORV and Safety Valve Releases[5]	Steam Line Break Releases	Fuel Handling Area[6]
0 - 2 hours	~~2.2~~3.0E-3	~~2.2~~6.0E-3	2.0E-2	2.4E-2	6.0E-3
2 - 8 hours	~~1.4~~2.5E-3	~~1.4~~4.5E-3	1.8E-2	2.0E-2	4.0E-3
8 - 24 hours	~~6~~1.0E-3~~4~~	~~6~~2.0E-3~~4~~	7.0E-3	7.5E-3	2.0E-3
1 - 4 days	~~1.5~~8.0E-4	~~1.5~~1.8E-3~~4~~	5.0E-3	5.5E-3	1.5E-3
4 - 30 days	~~1.6~~5.0E-4	~~1.6~~1.5E-3~~4~~	4.5E-3	5.0E-3	1.0E-3
χ/Q (s/m^3) at Control Room Door for the Identified Release Points[2]					
	Plant Vent or PCS Air Diffuser[3]	Ground Level Containment Release Points[4]	PORV and Safety Valve Releases[5]	Steam Line Break Releases	Fuel Handling Area[6]
0 - 2 hours	~~6.6~~1.0E-3~~4~~	~~6.6~~1.0E-3~~4~~	4.0E-3	4.0E-3	6.0E-3
2 - 8 hours	~~4.8~~7.5E-4	~~4.8~~7.5E-4	3.2E-3	3.2E-3	4.0E-3
8 - 24 hours	~~2.1~~3.5E-4	~~2.1~~3.5E-4	1.2E-3	1.2E-3	2.0E-3
1 - 4 days	~~1.5~~2.8E-4	~~1.5~~2.8E-4	1.0E-3	1.0E-3	1.5E-3
4 - 30 days	~~1.3~~2.5E-4	~~1.3~~2.5E-4	8.0E-4	8.0E-4	1.0E-3

26. Tier 2, DCD Technical Specification 3.9.7

3.9 REFUELING OPERATIONS

3.9.7 Decay Time

LCO 3.9.7 The reactor shall be subcritical for ≥ ~~48~~100 hours.

APPLICABILITY: During movement of irradiated fuel in the reactor pressure vessel.

ACTIONS

- NOTE -

LCO 3.0.8 is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Reactor subcritical < ~~48~~100 hours.	A.1 Suspend all operations involving movement of irradiated fuel in the reactor pressure vessel.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.7.1	Verify that the reactor has been subcritical for ≥ ~~100~~48 hours by verification of the date and time of subcriticality.	Prior to movement of irradiated fuel in the reactor vessel

27. Tier 2, DCD Technical Specification 3.9.7 Bases, pgs B 3.9.7-1

B 3.9.7 Decay Time

BASES

BACKGROUND	The movement of irradiated fuel assemblies within containment or in the fuel handling area inside the auxiliary building requires allowing at least ~~48~~100 hours for radioactive decay time before fuel assembly handling can be initiated. During fuel handling, this ensures that sufficient radioactive decay has occurred in the event of a fuel handling accident (Refs. 1 and 2). Sufficient radioactive decay of short-lived fission products would have occurred to limit offsite doses from the accident to within the values reported in Chapter 15.
APPLICABLE SAFETY ANALYSES	During movement of irradiated fuel assemblies, the radioactivity decay time is an initial condition design parameter in the analysis of a fuel-handling accident inside containment or in the fuel handling area inside the auxiliary building, as postulated by Regulatory Guide 1.183 (Ref. 1). The fuel handling accident analysis inside containment or in the fuel handling area inside the auxiliary building is described in Reference 2. This analysis assumes a minimum radioactive decay time of ~~48~~100 hours. Radioactive decay time satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).
LCO	A minimum radioactive decay time of ~~48~~100 hours is required to ensure that the radiological consequences of a postulated fuel handling accident inside containment or in the fuel handling area inside the auxiliary building are within the values calculated in Reference 2.

28. Tier 2, DCD Technical Specification 3.9.7 Bases, pgs B 3.9.7-2

BASES

ACTIONS

LCO 3.0.8 is applicable while in MODE 5 or 6. Since movement of irradiated fuel assemblies with less than ~~100~~ 48 hours of decay time can occur in MODE 6 after removing the reactor vessel head following the reactor shutdown, the ACTIONS have been modified by a Note stating that LCO 3.0.8 is not applicable. If moving irradiated fuel assemblies while in MODE 6, the fuel movement is independent of shutdown reactor operations since the reactor is already shutdown. Entering LCO 3.0.8 while in MODE 6 would not specify any action.

A.1

With a decay time of less than ~~100~~ 48 hours, all operations involving movement of irradiated fuel assemblies within containment or in the fuel handling area inside the auxiliary building shall be suspended immediately to ensure that a fuel handling accident cannot occur.

The suspension of fuel movement shall not preclude completion of movement to safe position.

SURVEILLANCE REQUIREMENTS

SR 3.9.7.1

Verification that the reactor has been subcritical for at least ~~100~~ 48 hours prior to movement of irradiated fuel in the reactor pressure vessel to the refueling cavity in containment or to the fuel handling area inside the auxiliary building ensures that the design basis for the analysis of the postulated fuel handling accident during refueling operations is met. Specifying radioactive decay time limits the consequences of damaged fuel rods that are postulated to result from a fuel handling accident (Ref. 2).