Westinghouse

Westinghouse Electric Company
Nuclear Plant Projects
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USA

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

Direct tel: 412-374-5355
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e-mail. corletmm@westinghouse.com

Your ref. Docket No. 52-006
Our ref: DCP/NRC1529

November 1, 2002

SUBJECT: Transmittal of Westinghouse Proprietary and Non-Proprietary Responses to U.S. Nuclear Regulatory Commission Requests for Additional Information on the AP1000 Application for Design Certification

This letter transmits the Westinghouse responses to NRC Requests for Additional Information (RAI) regarding our application for Design Certification of the AP1000 standard plant. The list of RAI responses that are transmitted with this letter is provided in Attachment 1. Attachments 2 and 3 to this letter provide the proprietary and non-proprietary responses to the NRC RAI.

The Westinghouse Electric Company Copyright Notice, Proprietary Information Notice, Application for Withholding, and Affidavit are also enclosed with this submittal letter as Enclosure 1. Attachment 2 contains Westinghouse proprietary information consisting of trade secrets, commercial information or financial information which we consider privileged or confidential pursuant to 10 CFR 2.790. Therefore, it is requested that the Westinghouse proprietary information attached hereto be handled on a confidential basis and be withheld from public disclosures. Attachment 3 contains no proprietary information.

This material is for your internal use only and may be used for the purpose for which it is submitted. It should not be otherwise used, disclosed, duplicated, or disseminated, in whole or in part, to any other person or organization outside the Commission, the Office of Nuclear Reactor Regulation, the Office of Nuclear Regulatory Research and the necessary subcontractors that have signed a proprietary non-disclosure agreement with Westinghouse without the express written approval of Westinghouse.

D063

November 1, 2002

Correspondence with respect to the application for withholding should reference AW-02-1566, and should be addressed to Hank A. Sepp, Manager of Regulatory and Licensing Engineering, Westinghouse Electric Company, P.O. Box 355, Pittsburgh, Pennsylvania, 15230-0355.

Please contact me at 412-374-5355 if you have any questions concerning this submittal.

Very truly yours,

Michael M. Corletti

M. M. Corletti
Passive Plant Projects & Development
AP600 & AP1000 Projects

/Enclosure

1. Westinghouse Electric Company Copyright Notice, Proprietary Information Notice, Application for Withholding, and Affidavit AW-02-1566

/Attachments

1. Table 1, "List of Westinghouse's Responses to RAIs Transmitted in DCP/NRC1529"
2. Westinghouse Proprietary Response to US Nuclear Regulatory Commission Requests for Additional Information dated November 2002
3. Westinghouse Non-Proprietary Response to US Nuclear Regulatory Commission Requests for Additional Information dated November 2002

DCP/NRC1529
Docket No. 52-006

November 1, 2002

Enclosure 1

Westinghouse Electric Company
Copyright Notice, Proprietary Information Notice, Application for Withholding, and Affidavit

DCP/NRC1529
Docket No. 52-006

November 1, 2002

Copyright Notice

DCP/NRC1529
Docket No. 52-006

November 1, 2002

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 2.790 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 2.790(b)(1).

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

November 1, 2002

AW-02-1566

Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555

ATTENTION: Mr. Lawrence Burkhart

APPLICATION FOR WITHHOLDING PROPRIETARY
INFORMATION FROM PUBLIC DISCLOSURE

SUBJECT: Transmittal of Westinghouse Proprietary Class 2 and Non-Proprietary Class 3 versions of Document: "AP1000 Design Certification Review – Responses to Requests for Additional Information"

Dear Mr. Burkhart:

The application for withholding is submitted by Westinghouse Electric Company, LLC ("Westinghouse") pursuant to the provisions of paragraph (b)(1) of Section 2.790 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 documents. In conformance with 10 CFR Section 2.790, Affidavit AW-02-1566 accompanies this application for withholding setting forth the basis on which the identified proprietary information may be withheld from public disclosure.

Accordingly, it is respectfully requested that the subject information which is proprietary to Westinghouse be withheld from public disclosure in accordance with 10 CFR Section 2.790 of the Commission's regulations.

Correspondence with respect to this application for withholding or the accompanying affidavit should reference AW-02-1566 and should be addressed to the undersigned.

Very truly yours,

Michael M. Corletti

M. M. Corletti
Passive Plant Projects & Development
AP600 & AP1000 Projects

/Enclosures

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AW-02-1566

COMMONWEALTH OF PENNSYLVANIA:

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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:

KAY E. GONGAWARE
COMMONWEALTH OF PENNSYLVANIA
NOTARY PUBLIC

James W. Winters, Manager
Passive Plant Projects & Development
Nuclear Plant Projects
Westinghouse Electric Company, LLC

Sworn to and subscribed
before me this 1st day
of November, 2002

Notary Public

Notarial Seal
Kay E. Gongaware, Notary Public
Monroeville Boro, Allegheny County
My Commission Expires Feb. 7, 2005
Member, Pennsylvania Association of Notaries

(1) I am Manager, Passive Plant Projects & Development, in the Nuclear Plant Projects Business Unit, of the 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 rulemaking proceedings, and am authorized to apply for its withholding on behalf of the Westinghouse Electric Company, LLC.

(2) I am making this Affidavit in conformance with the provisions of 10 CFR Section 2.790 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 the Westinghouse Electric Company, LLC 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 2.790 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 which 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 2.790, 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.

The proprietary information sought to be withheld in this submittal is that which is appropriately marked in Attachment 1 as Proprietary Class 2 in the Westinghouse document DCP/NRC1529 for submittal to the Commission: (1) "AP1000 Design Certification Review – Response to Requests for Additional Information."

This information is being transmitted by Westinghouse's letter and Application for Withholding Proprietary Information from Public Disclosure, being transmitted by Westinghouse Electric Company (W letter AW-02-1566) and to the Document Control Desk, Attention: Lawrence Burkhart, DIPM/NRLPO, MS O-4D9A.

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

(a) Provide documentation supporting determination of APP-GW-GL-700, "AP1000 Design Certification Document," analysis on a plant specific basis

(b) Provide the applicable engineering evaluation which establishes the Tier 2 requirements as identified in APP-GW-GL-700.

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 meeting NRC requirements for Licensing Documentation.

(b) Westinghouse can sell support and defense of AP1000 Design Certification.

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 methodologies 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 for performing and analyzing tests.

Further the deponent sayeth not.

Attachment 1

List of Westinghouse's Responses to RAIs Transmitted in DCP/NRC1529

Table 1
"List of Westinghouse's Responses to RAIs Transmitted in DCP/NRC1529"

210.028	410.012	440.160
210.042	410.015	440.164
251.004	410.021	440.169
251.005	420.008	440.170
251.006	420.015	440.171
251.007	420.016	440.172
251.008	420.017	480.002
251.009	420.018	480.003
251.010	420.020	480.005
251.022	420.022	480.006
261.013	420.031	480.007
280.004	420.043	480.009
280.005	440.035	640.001
280.006	440.053	720.006
280.008	440.054	720.015
280.009	440.063	720.027
280.010	440.072	720.028
280.011	440.097	720.031
410.001	440.098	720.032
410.002	440.129	720.059
410.003	440.151	720.062
410.004	440.158	720.077
410.008	440.159	720.081
410.011		

DCP/NRC1529
Docket No. 52-006

November 1, 2002

Attachment 3

"AP1000 Design Certification Review –
Response to Request for Additional Information"

RAI Number: 210.028

Question:

Reference, Volume 6, Table 3.9-5, Pg. 3.9-102:

The Level B Service loading combinations do not appear to include earthquake loading (see USNRC Standard Review Plan (SRP) 3.9.3, Appendix A, C.1.3.2). The Level C Service loading combinations do not appear to include design basis pipe break loading (see SRP 3.9.3, Appendix A, C.1.3.3). Please clarify.

Westinghouse Response:

The operating basis earthquake (OBE) has been eliminated as a design requirement for the AP1000 (see DCD section 3.7). AP1000 ASME Class 1, 2, and 3 components and structures are designed for one occurrence of the safe shutdown earthquake which is evaluated as a Service Level D condition for pressure boundary integrity (see DCD section 3.9.3.1.1). This is the same design basis as for the AP600 components and structures.

The design basis pipe break identified in the Standard Review Plan 3.9.3, Appendix A, Section C.1.3.3 is defined as a break in Class 1 branch lines that result in the loss of reactor coolant at a rate less than or equal to the capability of the reactor coolant makeup system. The resulting loads are considered secondary loads under Service Level C conditions. Per ASME Code Section NB-3224 (Figure NB-3224-1) evaluation of these secondary loads are not required for Level C Service Limits. Loss-of-coolant (LOCA) events are considered faulted events and are included in DCD Table 3.9-5 under Service Level D. The worst case LOCAs are considered as Level D events and envelope all the smaller LOCAs identified as emergency conditions under Level C.

AP1000 DCD Table 3.9-5 provides the same loading combinations as for the AP600 components and structures.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 210.042

Question:

Section 3.7.3.17 discusses time history broadening which generally involves performing three analyses to include normal, as well as contracted and expanded time scales to account for uncertainties. References to time scale variations of "+ or - 15 percent" and to stiffness variations of "+ or - 30 percent" should be corrected to indicate "+ and - 15 percent" and "+ and - 30 percent," respectively, since both variations must be analyzed. This subsection also states that when the results are shown to be acceptable based on comparison with test data, only one analysis may be performed using normal time. For what types of loadings and under what conditions would this option be used? Provide justification.

Westinghouse Response:

The reference to building stiffness variations and time scale variations will be corrected to + and – 30 percent and to + and – 15 percent respectively as requested. Additionally, since the seismic criteria does not provide for the alternate method utilizing test data, it will be removed.

Design Control Document (DCD) Revision:

The third paragraph of section 3.7.3.17 will be revised as follows:

For dynamic analysis, including seismic analysis at a hard rock site, three separate analyses are performed for each loading case to account for uncertainties. The three analyses correspond to three different time scales: normal time, time expanded by 15 percent, and time compressed by 15 percent. ~~Alternatively, when the results are shown to be acceptable based on comparison with test data, one time history analysis is performed using normal time.~~ For time history analysis of piping system models that include a dynamic model of the supporting concrete building either the building stiffness is varied by + and~~or~~ - 30 percent, or the time scale is shifted by + and~~or~~ - 15 percent. Alternately, when uniform enveloping time history analysis is performed, modeling uncertainties are accounted for by the spreading that is included in the broadened response spectra.

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 251.004

Question:

Due to the primary water stress corrosion cracking (PWSCC) of the V. C. Summer primary loop welds, the staff finds that the information that we have today is substantially different from the information that was available when we approved leak-before-break (LBB) applications for existing pressurized water reactor (PWR) systems which contain Inconel 82/182 materials. The following three questions are related to staff concerns regarding this recently discovered degradation mechanism as it applies to any LBB-candidate piping systems proposed in AP1000. (DCD Section 3.6.3)

A. Section 5.2.3 of the DCD indicates that the "use of nickel-chromium-iron alloy in the reactor coolant pressure boundary is limited to Alloy 690. Alloy 600 may be used in limited areas for welding or buttering. Where Alloy 600 is used, it is not in contact with the reactor coolant." However, in addition to the reactor coolant system (RCS) piping, there is LBB-candidate piping, for example the passive core cooling system, exposed to primary water under temperature and pressure conditions similar to those in the RCS. Discuss the susceptibility of these systems to PWSCC.

B. Provide test and plant operational data regarding the crack growth rate for Alloy 52/152 welds to be used in contact with reactor coolant in the proposed lines for which LBB will be applied and demonstrate that this material is not susceptible to PWSCC.

C. LBB is based, in part, upon the premise that LBB will only be applied to piping materials that are not susceptible to any known degradation mechanisms. Until sufficient information is acquired to ensure that Inconel 52/152 materials are essentially "PWSCC resistant" through the anticipated 60 year operational lifetime of an AP1000 facility, the staff believes that augmented inservice inspection of Inconel welds in LBB lines, including the use of inside-diameter (ID) eddy current on a periodic basis, is an essential element for approval of the AP1000 "design" to support application of LBB. To facilitate resolution of the PWSCC issue for AP1000, please provide an inspection plan that the combined licensee would be required to perform. This inspection plan should address additional inspection techniques (e.g., eddy current testing) to supplement ultrasonic testing (UT) so that tight flaws in piping welds similar to those detected in the V. C. Summer primary loop weld could be detected.

Westinghouse Response:

A. The following thirteen AP1000 piping systems are defined as Leak-Before-Break systems and identified in the appropriate figures in Appendix 3E of the DCD.

RCS	Reactor Coolant Loop (RCL)
RCS	1st, 2nd, 3rd Stage Automatic Depressurization System
RCS	Pressurizer Surge Line
RCS	4th Stage ADS (East)
RCS	4th Stage ADS (West)
RNS	Normal Residual Heat Removal Suction
PXS	Passive RHR Return
PXS	Direct Vessel Injection – A
PXS	Direct Vessel Injection – B
PXS	Core Make-Up (A)
PXS	Core Make-Up (B)
SGS	Main Steam Line A
SGS	Main Steam Line B

With the exception of the Main Steam lines, which are connected directly to the steam generators, all other LBB piping systems are connected to the Reactor Coolant primary system via the Reactor Coolant Loop, Reactor Pressure Vessel, or the Pressurizer.

Alloy 600 will not be used for any of the AP1000 LBB candidate piping systems.

B. Background and Experience - SCC Resistance of Alloys 52 and 152

INTRODUCTION

Alloy 52 is the filler metal used for the joining of Alloy 690 components by either the gas-tungsten arc welding (GTAW) or gas metal arc welding (GMAW) processes. The welding electrode used for the shielded metal arc welding (SMAW) process is Alloy 152. Both of these materials have compositions not differing greatly from the parent Alloy 690 material. Nominal compositions are provided in the following table.

Element	Alloy 690	Alloy 152	Alloy 52
	Base Metal	E-NiCrFe-7	ER-NiCrFe-7
	SB-167	SMAW	GTAW/GMAW
C	0.05 max	0.05 max	0.04 max
Mn	0.5 max	5.00 max	1.00 max
Fe	7.0 to 11.0	7.0 to 12.0	7.0 to 11.0
P	-	0.03 max	0.02 max
S	0.015 max	0.015 max	0.015 max
Si	0.5 max	0.75 max	0.5 max
Cu	0.5 max	-	0.3 max
Ni	58.0 min	Bal	Bal
Co	-	Incl. with Ni	Incl. with Ni
Al	-	0.50 max combined	1.10 max Al or 1.50 max combined
Ti	-		
Cr	27.0 to 31.0	28.0 to 31.5	28.0 to 31.5
Nb + Ta	-	1.0 to 2.5	0.10 max
Mo	-	0.50 max	0.50 max
Other elements	-	0.50 max	0.50 max

Essentially coincident with the introduction of Alloy 690, Alloys 52 and 152 have been used for all fusion welding applications as the material of choice for applications with Alloy 690. The following provides a summary of the experience with respect to these filler metals in service and in laboratory testing.

SERVICE EXPERIENCE

The majority of the operating plant experience with Alloy 690 and the weld metals Alloys 52 and 152 is associated with replacement steam generator (SG) programs beginning in approximately 1994 with the Delta 75 replacements for V. C. Summer. In addition to the exclusive use of Alloy 690 for the SG heat transfer tubing applications, the weld metals were used for a range of applications in which contact with primary reactor coolant was required. A brief summary of the weld metal applications, for Westinghouse-designed components, follows.

<table>
<tr><th>Plant</th><th>Repl. efpy</th><th>Component</th><th>Material*</th><th>Application</th></tr>
<tr><td rowspan="3">V. C. Summer</td><td rowspan="3">7 years</td><td>SG nozzle welds</td><td rowspan="3">Alloy 52 and/or Alloy 152</td><td rowspan="2">Buttering over Alloy 82/Alloy 182 welds</td></tr>
<tr><td>Safe end-nozzle welds</td></tr>
<tr><td>Divider plate-channel head & stub runner</td><td>Final weld layer (in contact with RCS)</td></tr>
<tr><td>Kori 1</td><td>5 +</td><td rowspan="7">Tubesheet cladding
SG nozzle welds
Safe end-nozzle welds
Divider plate-tubesheet welds</td><td rowspan="7">Alloy 52 and/or Alloy 152</td><td rowspan="7">All buttering, cladding and welding operations</td></tr>
<tr><td>Shearon Harris</td><td>3 +</td></tr>
<tr><td>S. Texas 1</td><td>3</td></tr>
<tr><td>S. Texas 2</td><td>1 +</td></tr>
<tr><td>ANO-2</td><td>2</td></tr>
<tr><td>Farley 1</td><td>2</td></tr>
<tr><td>Farley 2</td><td>1 +</td></tr>
<tr><td rowspan="3">Kewaunee</td><td rowspan="3">~ 3</td><td>Tubesheet cladding</td><td rowspan="3">Alloy 52 and/or Alloy 152</td><td>All cladding</td></tr>
<tr><td>SG nozzle welds</td><td rowspan="2">Buttering and welding operations</td></tr>
<tr><td>Safe end-nozzle welds</td></tr>
<tr><td colspan="5">* - Nearly all procedures permit either Alloy 52 or Alloy 152 to be used</td></tr>
</table>

In addition to these Westinghouse units, similar experience has been accrued with replacement SGs in Europe and in Japan.

There have been no reported instances of environmental degradation of any kind for any of these applications; this includes both the Alloy 690 base metal and the Alloy 52 or Alloy 152 weld metals.

This experience is fully consistent with expectations from laboratory testing performed to support the qualification of these materials. This class of austenitic nickel-base alloys, containing greater than 27 wt. pct. chromium, has exhibited full resistance to primary water stress corrosion cracking (PWSCC), to the extent that they are generally regarded as immune to this form of environmental degradation.

This experience combined with the growing operating plant experience, provided the basis for the use of Alloys 52 and 152 for the recent primary loop nozzle repairs at V. C. Summer.

LABORATORY EXPERIENCE

For the reasons implied above, i.e., lack of experience with PWSCC of Alloy 690, relatively little testing for either crack initiation or crack propagation has been performed for either the base metal or the weld metals.

Psaila-Dombrowski et al. (Ref. 1) evaluated the SCC resistance of Alloy 152 welds in primary water environments using constant extension rate tests (CERT) at 343°C (650°F). Examination of the fracture surfaces indicated no environmentally-related degradation. All fracture occurred by ductile rupture.

Psaila-Dombrowski et al. (Ref. 2) performed a series of CERT tests on Alloys 52 and 152 weldments in simulated primary water at 343°C (650°F). After testing for periods up to 4122 hours, environmentally-related crack propagation was not observed.

These are the only published test results with which we are familiar.

REFERENCES

1. M. J. Psaila-Dombrowski et al., "Evaluation of Weld Metal 82 and Weld Metal 182 Stress Corrosion Cracking Susceptibility," *Proceedings, Seventh International Symposium on Environmental Degradation of Materials in Nuclear Power Systems – Water Reactors,* NACE Int'l. (1995) 81-91.

2. M. J. Psaila-Dombrowski et al., "Evaluation of Weld Metals 82, 152, 52 and Alloy 690 Stress Corrosion Cracking and Corrosion Fatigue Susceptibility," *Proceedings, Eighth International Symposium on Environmental Degradation of Materials in Nuclear Power Systems – Water Reactors,* ANS (1997) 412-421.

C. As explained in response to question b above, the proposed Inconel 52/152 weld material is the material of choice for the industry and for AP1000 and has better crack resistance than Inconel 82/182 materials. Augmented inservice inspection of Inconel 52/152 materials welds including the use of inside-diameter (ID) eddy current on a periodic basis has not been required for the operating plants and therefore, should not be required for the AP1000 applications.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 251.005

Question:

Provide crack morphology parameters, e.g., surface roughness, number of 45 degree and 90 degree turns, etc., that were used in generating the bounding analysis curves for LBB. To address the staff's concerns resulting from recent experience with stress corrosion cracking in Inconel and stainless steel materials in PWR environments, please provide a comparative study on the most biased line from the LBB candidates using the crack morphology parameters for transgranular stress corrosion cracking. Information regarding crack morphology parameters for various degradation mechanisms is available in NUREG/CR-6443, "Deterministic and Probabilistic Evaluations for Uncertainty in Pipe Fracture Parameters in Leak-Before-Break and In-Service Flaw Evaluations." Report the reduced margin on flaw size from this comparative study of the most biased line when the original bounding analysis curve (BAC) for this line is maintained. (DCD Appendix 3B)

Westinghouse Response:

In generating the bounding analysis curves, crack morphology parameters used are:

surface roughness = [ ] a,c,e;
number of 45 degree turns = 0;
number of 90 degree turns = 0;
crack shape = rectangle.

In order to avoid the concern of the Inconel 82/182 PWSCC issue for the AP1000, Inconel 52/152 will be used for all applicable locations of LBB piping systems.

Westinghouse believes that the Transgranular Stress Corrosion Cracking (TGSCC) mechanism is highly unlikely in the AP1000 LBB candidate piping systems. TGSCC has not been observed in PWR stainless steel piping. For TGSCC to occur, an aggressive species such as chlorides would also need to be present. Since these types of aggressive species will be controlled and kept at minimum levels in the AP1000 LBB candidate piping systems water environment, a much higher level of oxygen would be required to be present to provide the appropriate environment for SCC to develop. Since the oxygen levels are kept to near zero by the hydrogen overpressure, the AP1000 piping systems will not be susceptible to TGSCC.

The occurrence of TGSCC in CRDMs at Palisades and Ft. Calhoun (basically the same CRDM design) is not expected in AP1000 LBB candidate piping systems. The Palisades incidents occurred because the materials used were susceptible to TGSCC in the CRDM environment (elevated levels of dissolved oxygen, some level of chloride ions. The TGSCC cracking incidents at Palisades and Ft. Calhoun CRDM are unique to that geometry and do not apply to AP1000.

Since the AP1000 piping systems will not be susceptible to TGSCC, we do not believe a leak rate calculation based on the hypothetical assumption of TGSCC for the AP1000 LBB application is necessary.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 251.006

Question:

NUREG-1512, "Final Safety Evaluation Report Related to Certification of the AP600 Standard Design," September 1998, documents Westinghouse's actions in resolving open items with regard to the AP600 review. These actions include (1) fatigue crack growth analyses are performed for the Class 2 and 3 piping systems selected for LBB applications, and (2) thermal stratification loads are considered in three piping systems (pressurizer surge line, PRHR return line, and another line not identified) that Westinghouse identified to be susceptible to thermal stratification. Are these measures to be implemented on AP1000 also? If not, please provide justification. (DCD Section 3.6.3)

Westinghouse Response:

(1) For AP1000 ASME Code Class 2 and 3 piping systems for which LBB is demonstrated fatigue crack growth analyses are to be performed.

(2) The following AP600 piping systems are identified as being susceptible to thermal stratification affects:

- Cold leg piping in the loop with passive RHR (during long-term PRHR operation)
- Pressurizer surge line
- Automatic depressurization system stage 4 lines
- Normal residual heat removal suction line
- Passive residual heat removal return line

As part of the detailed piping design for the AP1000, Westinghouse will perform system reviews for the corresponding AP1000 piping systems similar to the calculations performed for AP600. Resulting thermal loadings will be included in the piping design analyses.
(Refer to RAI 210.049)

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 251.007

Question:

It was stated under 3.6.4.2 that "Combined Licensee applicants referencing the AP1000 certified design will complete the leak-before-break evaluation by comparing the results of the as-designed piping stress analysis with the bounding analysis curves (BAC) documented in Appendix 3B." Since piping satisfying all American Society of Mechanical Engineers (ASME) Code requirements on stresses could have a stress state that is outside the BAC for LBB, you need to establish a process to give the LBB BAC the same status as the ASME Code requirements on stresses to ensure a successful path for the design and construction of all LBB candidates proposed in the submittal. Please provide additional information addressing this issue. (DCD Section 3.6.4)

Westinghouse Response:

The use of Bounding Analysis Curves in the piping analyses for appropriate AP1000 Leak-Before-Break piping systems was discussed in detail with the NRC staff in the meetings held at the Westinghouse office on September 9th through the 11th. Criteria documents, including LBB bounding analysis curve calculations and High Energy Line Break (HELB) criteria, were also reviewed by the NRC staff at the meeting. In addition, several AP600 piping analysis calculations for LBB piping systems were reviewed by the NRC staff to confirm that the LBB criteria had been applied and documented as defined by the applicable criteria documents. This review confirms that a process exists for the evaluation of both LBB BAC and ASME Code requirements. (Refer to RAI 210.036)

It can be noted that for all thirteen AP1000 candidate Leak-Before-Break piping systems (LBB piping systems are indicated in part a of RAI 251.004 response), both ASME stress criteria and LBB stress criteria need to be satisfied as defined in the appropriate AP1000 Piping Analysis Criteria documents.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 251.008

Question:

Appendix 3B.3.1.3, 3B.3.1.4, and 3B.3.1.5

Using Figure 3B-12 as an example, provide flow stress and the ASME Code specified S_m value for the material. Flow stress can be defined as one half of the ultimate strength and yield strength, or $3S_m$ of a material. Justify your choice if your selection gives higher flow stress for the piping material. Provide the axial stress, bending stress, leakage flaw size, and critical flaw size for the normal stress state and the maximum stress state corresponding to the low normal stress case (Case 1). Provide similar information for the high normal stress case (Case 2) also. (DCD Appendix 3B)

Westinghouse Response:

For Figure 3B-12:

Material Type is SA 312 TP316LN. Flow stress (one half of the ultimate strength and the yield strength) = 40.70 ksi

ASME Code specified S_m value = 16.40 ksi at 610 °F.

The flow stress (40.70 ksi) used in the LBB BAC calculation is less than $3S_m$ (3x16.40 =49.20 ksi) of the material.

It can be noted that for all cases, the BAC flow stress values are less than $3S_m$ of the materials.

Low normal stress case (Case 1):

For the normal stress state:
Axial stress = 4.503 ksi, bending stress = 0 ksi, leakage flaw size = 5.808 inches.

For the maximum stress state:
Axial stress = 4.503 ksi, bending stress = 20.276 ksi, critical flaw size = 11.616 inches.

High normal stress case (Case 2):

For the normal stress state:
Axial stress = 4.503 ksi, bending stress = 7.234 ksi, leakage flaw size = 3.173 inches.

For the maximum stress state:
Axial stress = 4.503 ksi, bending stress = 36.218 ksi, critical flaw size = 6.345 inches.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 251.009

Question:

Appendix 3B.3.1.3, 3B.3.1.4, and 3B.3.1.5

The high normal stress case was determined using flow stress as the bending stress. In some figures, for instance Figure 3B-21, a normal stress of 30 ksi (thousand pounds-per-square inch) would correspond to more than two times the flow stress of the material. Even greater multiples of flow stress are expected for the maximum stress of 40 ksi. What is the meaning of the region to the right of Point "B (the point corresponding to Case 2)" for all BACs in terms of the piping design ASME Code criteria? For each BAC shown in Figures 3B-1 to 3B-21, construct a separate design curve based on the appropriate piping design ASME Code such that every point within the design curve would automatically satisfy all ASME Code requirements on piping stresses. If any of the design curves exceed its corresponding BAC by 25%, provide detailed piping stress and LBB analyses for that line to demonstrate that it is feasible to build a line according to a more restrictive piping design criteria considering the LBB BAC. This additional work needs to be performed for lines other than those lines that have been approved for LBB applications for operating plants involving essentially the same analysis parameters (pipe diameter, wall thickness, material properties, and loading conditions) and for the five exemplary lines studied in AP600. (DCD Appendix 3B)

Westinghouse Response:

For Figure 3B-21:

The high normal stress case (Case 2) point (Point "B" as shown in figure 3B-1)) was determined using flow stress of 40.70 ksi, as the total stress (not the bending stress alone). In Figure 3B-21, a normal stress of 30 ksi (thousand pounds-per-square inches) would not correspond to more than two times the flow stress (i.e. 2x40.70 ksi = 81.40 ksi) of the material. The maximum stress limit of 40.70 ksi is same as flow stress and can not be greater multiples of flow stress.

For Point "B" critical flaw size was obtained using a flow stress as the maximum stress. Corresponding normal stress was determined using a leakage flaw size which is equal to one half of the critical flaw size. The horizontal line towards right side from Point "B" at the top is represented by the flow stress (maximum stress for Point "B"). In a situation where a normal stress due to pipe stress analysis is greater than the normal stress corresponding to Point "B", the point can be plotted on the BAC. A stress point to the right of Point "B" will provide higher LBB margin since the leakage flaw size will be smaller with a higher normal stress.

Due to the difference in loading combinations and acceptance criteria between ASME piping qualification vs. LBB BAC, it is difficult to construct a design curve such as every point would automatically satisfy the ASME Code requirements and that's why for all thirteen AP1000 candidate Leak-Before-Break piping systems (LBB piping systems are indicated in part a of RAI 251.004 response), both ASME stress criteria and LBB stress criteria need to be satisfied as defined in the appropriate AP1000 Piping Analysis Criteria documents. The corresponding AP600 piping systems have all been evaluated for both ASME criteria and LBB criteria and found to be acceptable. The AP1000 criteria documents were reviewed by the NRC staff in meetings held at the Westinghouse office on September 9th through September 11 (Refer to RAI 210.036). As defined in Appendix 3B of the DCD, the maximum LBB stress (critical location) is defined by the following loading combination:

|Pressure| + |Deadweight| + |Thermal (100% Power or applicable stratification)| + |Safe Shutdown Earthquake|

Review of the ASME stress criteria and the Supplemental stress criteria as defined in Section 3.9 of the DCD shows that the combination of Thermal + Safe Shutdown Earthquake is not considered for piping stress analysis. Therefore, the construction of a design curve that compares the ASME stress allowable to the corresponding LBB BAC is not possible as the loading combinations for ASME vs. LBB are not the same.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

RAI Number: 251.010

Question:

Since it is unlikely that the relationship between the maximum stress and the normal stress shown in Figures 3B-1 to 3B-21 is linear, an intermediate point should be plotted on all these curves. Please provide additional information to address this issue. (DCD Appendix 3B)

Westinghouse Response:

The approach approved for AP600 was to determine bounding analysis curves for the candidate LBB piping systems. The bounding curves are a plot of the maximum stress over a range of normal stresses that would be observed in the candidate LBB piping systems. The COL applicant will perform the LBB analysis with as-built information and verify that the stress analysis results are within the bounding curve limits. The generation of bounding analysis curves is conservative with respect to the LBB analysis methods that have been approved. In NUREG-1512, the NRC discusses the approach approved for AP600.

In revised GDC 4, the NRC states, in part, that "dynamic effects associated with postulated pipe ruptures in nuclear power units may be excluded from the design basis when analyses reviewed and approved by the Commission demonstrate that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping." The analyses referred to in the revised GDC 4 should be based on such specific data as piping geometry, materials, and piping loads. The staff will review the LBB analyses for specific piping design before the applicant can exclude the dynamic effects from the design basis.

Applicants seeking design certification for ALWRs under 10 CFR Part 52 are allowed to incorporate preliminary stress analysis results, provided bounding limits (both upper and lower bound) are determined in order to establish assurance that adequate margins are available for leakage, loads, and flaw sizes. These bounding values and preliminary analyses can be verified when as-built and as-procured information becomes available during the COL phase.

Verification of the preliminary LBB analysis will be completed at the COL stage based on actual material properties and final, as-built piping analysis as part of ITAAC associated with 10 CFR Part 52 prior to fuel loading. The above staff position on LBB application is stated in SECY-93-087 and was approved by the Commission in its SRM dated July 21, 1993. A margin of 10 on leakage is required so that leakage from the postulated flaw size is assured of detection when the pipe is subjected to normal operational loads. A margin of $(2)^{0.5}$ (1.0 is acceptable if loads are combined by the absolute sum method) on loads is required to ensure that leakage-size flaws are stable at normal plus accident loads (e.g., SSE). A factor of 2 between the leakage-size flaw (postulated under normal loads) and the critical-size flaw (calculated under normal plus SSE loads) is required to ensure an adequate stability margin for the leakage-size flaw. The analysis must be performed for an entire pipe run from anchor to anchor. In addition, applicants seeking approval of LBB during the design certification phase for an ALWR will be required to perform LBB analyses to establish through-wall flaw sizes and flaw stability. For through-wall flaw sizes, a lower-bound, normal-operational stress limit must be established for dead weight, pressure, and thermal loadings. The mean or best-estimate stress strain curve should be used. For flaw stability, an upper-bound stress limit should be established for normal plus SSE loading. A lower-bound stress-strain curve for base metal should be used regardless of whether the weld or base metal is limiting. In addition, a lower-bound toughness (weld metal or base metal) will be used.

Westinghouse has maintained the conservatism in generating the bounding analysis curves for AP1000, and therefore the bounding analysis curve approach that was approved for the AP600 has been adopted for the AP1000. During AP600, Westinghouse performed calculations with intermediate points to assess the conservatism in the approach. The results shown in AP600 Appendix B Figure 3B-14 and Figure 3B-15 indicate that intermediate points are above the linear line connecting Point "A" and Point "B" and therefore the linear line approach was acceptable. We therefore believe it is not necessary to generate an intermediate point for AP1000.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 251.022

Question:

For the AP600 design, the response to RAI 252.110 indicated that the results of prototype tests and calculations were not yet completed with respect to the subject of flow-induced vibrations of the steam generators with special emphasis on fluid elastic vibration (AP600 DCD/Standard Safety Analysis Report (SSAR) Section 5.4.2.3.3). Please provide the results from the AP600 tests and calculations, if these are applicable to the AP1000 design. If the AP600 results are not applicable to the AP1000 design, please provide the results of the AP1000 prototype tests and calculations related to flow-induced vibrations of the tubes in different locations of the bundle. In addition, please discuss in more detail than in section 5.4.2.3.3, the criteria for establishing the instability threshold for ensuring that the fluid-elastic behavior does not contribute unacceptably to flow-induced vibration or alternating stresses. (Section 5.4.2)

Note: AP600 RAI 252.110 was issued by the NRC on October 1, 1992 (NUDOCS Accession No. 9210090123). Westinghouse provided its response to this RAI in a letter dated January 14, 1993 (NUDOCS Accession Nos. 9301250260).

Westinghouse Response:

The flow-induced vibration analysis for the AP1000 steam generator using final design information, including support configuration and tube bundle fluid flow rates, is not complete. However, evaluation of the tube bundle designs for the Delta-109 and Delta-75 steam generators have been performed. The Delta-109 tube bundle has a similar tube bundle configuration, including tube size and tube bundle diameter, as the AP1000 steam generator. The AP1000 tube bundle, however, has a greater (more favorable) tube pitch than that for the Delta-109 design. The tube bundle configuration for the AP1000 steam generator has the same tube pitch and similar tube bundle height as that for the Delta-75 steam generator. The effective tube bundle fluid flow rate for the AP1000 design is expected to be similar to that evaluated for the Delta-109 design. Any small increase in flow will be offset by the beneficial increase in stability constant associated with the greater tube pitch for the AP1000 design relative to the Delta-109 configuration.

Extensive testing and evaluation of tube bundle designs for Westinghouse steam generators have been performed. The analytical models used to evaluate tube vibration have been validated with a number of flow tests using various tube sizes and pitch geometries.

Two regions are of greatest interest in the evaluation of flow-induced vibration of steam generator tubes. The first of these is the inlet area at the bottom of the tube bundle where the water flowing down the annulus between the shell and wrapper turns and enters the tube bundle. The second area of interest is the U-bend region at the top of the tube bundle. In both of these regions the fluid is more or less in cross-flow over the tubes. A summary of the testing and analysis for each of these regions follows.

Cold flow vibration tests were conducted on a 49-tube test model in water with flow oriented 45 degrees to the square pattern to quantify tube response and qualify analytical models used to predict response to cross-flow in the inlet region. These tests provided measured tube vibration response to simulated inlet cross-flow from 10 to 200 percent of nominal full-power conditions and verified analytical predictions of frequencies and vibration amplitudes for this region of the steam generator. The absence of tube response to either of the potential vortex shedding or fluid-elastic mechanisms supported the design bases.

Subsequent cold-flow tests were conducted on a 15-degree sector of the lower tube bundle region of the Model F steam generator. A Model F steam generator has the same size tubes and tube pitch as the AP1000 steam generator, but with a square arrangement. This model had seven times more tubes and concentrated on flows in the 100 to 140 percent of nominal full-power range. Test series were done with tube arrays oriented for flow at 0 to 45 degrees through the square pattern. Tube vibration amplitudes, secondary fluid velocities, and dynamic forces at two support plates were measured for incremental flows ranging from 10 to 140 percent of nominal. Frequencies and vibration amplitudes were again consistent with analytical predictions. Tube dynamic characteristics at the support plates also indicated that the potential for tube degradation in this region is small at expected flow rates.

Correlations and empirical constants used in analyses of vortex shedding, fluid-elastic excitation, and turbulence had been derived based on years of laboratory testing at the Westinghouse Science and Technology Center. Vortex shedding is theoretically possible for the outermost tubes in the inlet cross-flow region. This had been demonstrated previously in carefully controlled laboratory tests when flow through the square array was staggered 45 degrees from the inline orientation. Two discrete Strouhal numbers that enveloped open literature predictions characterized tube response in the first and fifth tube rows in this orientation. No vortex shedding response was evident even in the carefully controlled laboratory environment for the inline flow configuration.

Periodic tube response characterized by a moving peak in the response spectrum as velocity increased could not be found in the 49-tube inlet cross-flow model for the full range of 10 to 200 percent of full-power nominal flow rates even when tested in the 45-degree, staggered-flow orientation. Tube response to vortex shedding, if present, was therefore too small to be observed over small, random turbulence effects. This is consistent with analyses based on expected fluctuating dynamic lift coefficients and correlation lengths characteristic of the steam generator flow distribution.

Both the potential for vortex shedding in carefully controlled tests, and its absence for full-size steam generator operating conditions are also applicable for triangular arrays (References 1 and 2). Thus, in addition to extensive Westinghouse experience with square arrays, it is generally noted that vortex shedding is only a potential design problem in the peripheral tube rows of large arrays in liquid flows (Reference 2). Conservative tube response calculations are typically made assuming that vortex shedding occurs in the inlet region even though such response is not expected.

Root mean square tube displacements from the 49-tube water flow tests were consistent with measurements from an operating plant (Reference 3) and with analytical predictions, which were made using empirical constants that envelope the magnitude of tube response to the random turbulent force spectrum typical of operating steam generators. These constants had been derived earlier by Westinghouse based on single cylinder data from Y. C. Fung (Reference 4). Appropriate constants for both peripheral and interior tubes were demonstrated to be conservative.

Fluid-elastic tube vibration did not occur in any of the 49-tube model tests. This is consistent with analyses based on threshold instability constants determined from previous laboratory tests. Instability constants are a function of the tube array pattern and spacing, and appropriate values have also been determined from similar laboratory tests for triangular configurations similar to the Delta-75. Conservative reference values for use with the gap pitch velocity (area ratio times upstream velocity) were defined for both arrays with a 1.42 pitch-to-diameter ratio. Results are consistent with data available from literature. For the pitch-to-diameter ratio of interest, triangular arrays are more stable, so that operating experience is conservative relative to fluid-elastic response. Tube vibrations in the straight-leg region are therefore known to be small and predictable using conventional approaches along with empirical constants determined from tests appropriate to the steam generator design configuration.

Boundary conditions for vibration analyses are obtained from qualified three-dimensional thermal-hydraulic codes such as ATHOS (Reference 5).

Effects of the wrapper inlet, tube support plate trifoil flow areas, annular flow area between plates and wrapper, and tube array geometry are included in the straight-leg portion of the overall model. Resulting velocity and density distributions are used to scale forcing functions and set boundary conditions for evaluation of the limiting vibration mechanisms. Vibration analyses are conducted using qualified finite element analytical models using approaches derived from extensive testing at the Westinghouse Science and Technology Center (References 6, 7, and 8).

Three potential secondary flow-excitation mechanisms are addressed: 1) vortex shedding, 2) turbulence, and 3) fluid-elastic excitation. The first is typically not a practical concern, except possibly for the outer few rows of tubes in the inlet region of steam generators for which nonuniform, two-phase turbulent flow exists throughout most of the tube bundle. Backup analyses are conducted for these outer-row tubes even though experimental results and field experience indicate that there is no vortex shedding response even here (probably due primarily to close tube spacing with high inlet turbulence and nonuniform velocities over the inlet spans). Strouhal numbers covering the range of values determined from carefully controlled laboratory experiments and external literature yield a range of potential synchronization frequencies. Correlation over the entire inlet flow span is then assumed, and tube response is calculated using an upper bound lift coefficient following methodology outlined by Connors (Reference 8). Calculated vibration amplitudes are typically less than the small calculated turbulence amplitudes, which are consistent with measured amplitudes in operating plants with years of operation without measurable tube wear. Resulting tube bending stresses are more than two orders of magnitude below ASME Code limits. Tube response to uncorrelated wake shedding in the bundle interior is covered by evaluation of random turbulence excitation.

Secondary flow turbulence throughout the bundle produces random tube displacements in a narrow frequency band that includes the natural frequency of the tube for the existing support configuration. Tube motions at locations that contact support plates are typically characterized by small amplitude displacements without liftoff, so that small fretting wear coefficients apply.

Fluid-elastic tube vibration is potentially more severe than either vortex shedding or turbulence because it is a self-excited mechanism: relatively large tube amplitudes can feed back proportionally large driving forces if an instability threshold is exceeded. Tube support spacing incorporated into the design of the tube support system provides tube response frequencies in such a way that the instability threshold is not exceeded for secondary fluid flow conditions. This is typically imposed by requiring that the calculated stability ratio (effective velocity/threshold velocity) be 0.75 or less. This approach provides margin against initiation of fluid-elastic vibration for tubes effectively supported at nominal locations.

Fluid-elastic instability analyses are performed for straight-leg tube spans using fluid flow conditions from qualified thermal-hydraulic analyses. The methodology follows that of Connors (Reference 8) using the appropriate threshold instability constant for the Delta-75 array. Typical stability ratios are much less than unity, indicating ample margins against initiation of fluid-elastic vibration in straight-leg tube spans.

Analyses are also performed for other postulated support conditions to demonstrate margin against instability even if dense corrosion products are postulated to form in the tube/support clearance. (This is a conservative assumption based on tests and operating experience with broached 405 SS supports.) In the limit this is assumed to result in the tube being "clamped" or "fixed" rather than "pinned" so that the positive damping of the tube is reduced, thereby possibly reducing the margin against initiation of fluid-elastic instability (frequency is simultaneously increased). Appropriate reduced damping values are used following the same analytical approach defined by Connors (Reference 8).

Results of vibration analyses are used both to assess satisfaction of tube stress limits and to demonstrate adequate margins against unacceptable wear. Typical tube responses to flow-induced excitation in the straight-leg region of feedring steam generators are benign, and vibrations have small effects on margins against tube stress and fatigue limits.

For the U-bend region at the top of the tube bundle, antivibration bars (AVBs) maintain tube-to-tube spacing, stiffen the tube bundle, and restrain vibration of the tubing U-bends above the top tube support plate. They are assembled in an advanced design configuration that was developed during a comprehensive program conducted over the past several years (Reference 9).

The advanced design configuration program was undertaken to eliminate the small percentage of tubes with moderate wall thinning in the U-bend region attributed to tube vibration and wear after 5 to 8 years of operation in some conventional operating steam generators with .875" OD x .050" T tubing and two sets of chromium-coated nickel-chromium-iron Alloy 600 AVBs. The mechanism leading to tube/AVB wear was established (fluid-elastic rattling within tube-to-AVB gaps), and design changes have been incorporated into the advanced configuration to provide enhanced margins against vibration-induced tube wear. These changes include a tighter tube-to-AVB fitup by both design and assembly, specifying the AVB material to be the same as the 405 stainless steel used in the steam generator straight leg tube supports, and increasing the width of the AVB consistent with tube dryout and leak-before-break constraints. More than thirty steam generators incorporating all of the advanced features have been fabricated and are now in service.

The basic objective of the advanced features is to avoid or minimize the consequences of fluid-elastic rattling (so called fluid-elastic vibration in the "support inactive mode," or "double-span behavior") between loosely fit AVBs in the U-bend region. Similar conclusions were reported by KWU relative to U-bend tube wear in a KWU steam generator configuration with relatively loose tube/support strip fitup. This mechanism is also the focus of studies described by typical literature from the United States (Reference 10), France (Reference 11), and England (Reference 12).

Extensive vibration testing has been conducted to refine conventional design approaches and to support development of advanced U-bend/AVB design configurations. Basic design information for tube vibration in prototypic steam-water flow was generated during Model Boiler 2 (MB-2) tests in a 0.01 power scale model of the Model F steam generator. There was no evidence of periodic vortex shedding in these tests, which also provided a threshold instability constant when tested with AVBs removed (tubes were fluid elastically stable with AVBs installed). Turbulent tube response characteristics in the U-bend region with AVBs in place were enveloped by calculations using the same force spectra scaling factors qualified for the straight-leg region.

A quarter scale 12-row x 8-column square array of aluminum U-bends was tested in the wind tunnel in the same fluid-elastic vibration regime as steam generators with steam-water flows. These tests provided additional information, especially in establishing threshold fluid-elastic instability constants for various support conditions in the U-bend region. Results were consistent with MB-2 values for loose support conditions but were lower for various tight fitup conditions. A lower bound value for U-bend analyses which is not the same as that for straight-leg response was derived from these tests. Tube/AVB dynamic interaction correlations were also established as a function of flow and fitup conditions.

Subsequently, a similar series of tests was conducted on a triangular array of the same size tubes in the same basic test rig, which has eight rows and nine columns of tubes configured to match the Delta-75 orientation in the U-bend. Results confirmed that this triangular array has an increased margin against instability in the U-bend region as it does in the straight-leg region.

Two series of wind tunnel experiments were conducted on cantilever tubes designed to simulate the response of curved U-bend tubes. A 7-row x 5-column array of full-size tubes mounted in such a way that orthogonal stiffnesses differed to match U-bend response provided two kinds of information. Basic fitup effects on tube response to both fluid-elastic and turbulent excitation were determined first. Contact with an AVB on one side of a gap with zero or very small preloads was effective in suppressing fluid-elastic tube response for low rates up to the maximum tested value that produced a stability ratio of four when tested without contact. Preloads on the order of 0.1 pound eliminated all impact/sliding motions with liftoff, which would otherwise result from both turbulence and fluid-elastic excitation for the tested diametral clearances. Both the threshold instability constant and turbulent tube response correlations were consistent with those derived from earlier tests. Then the test rig was modified and used to refine basic fluid-elastic driving force correlations for use in properly controlling mechanical shaker tests of full-size steam generator U-bends.

Mechanical excitation tests followed on full-size 0.687" OD x 0.040" T U-bends to characterize the wear-producing forces and motions at tube/AVB intersections. Parametric tests covered a range of fitup conditions subject to simulated out-of-plane fluid-elastic excitation, in-plane turbulence, and out-of-plane turbulence. Initial tests with four AVB intersections led to the fundamental conclusion that out-of-plane fluid-elastic vibration within tube/AVB gaps is the likely explanation for wear that had been observed in some operating steam generators. Subsequent tests with six AVB intersections simulated the excitation forces and fitup conditions characteristic of advanced design configurations. Wear-producing forces and motions were determined and recorded in the form of work rates for use in wear calculations (Reference 13). These work rates were verified by independent testing on the same full-size tube using a simulated negative damping feedback loop (Reference 14) in addition to the original effective sinusoidal force simulation. A semi-empirical wear calculation was developed (Reference 15) in which measured work rates from these tests are scaled to pertinent operating conditions using appropriate parameters from the thermal-hydraulic report and vibration analyses.

Thermal-hydraulic and tube vibration analyses follow the same general approach in the U-bend region as for the straight-leg region. Effects of the AVBs on flow distributions are obtained from qualified thermal-hydraulic models with explicit treatment of their size and location. There is no potential for flow peaking near the bend region of AVBs (involving small-radius tubes not supported by AVBs) in advanced design configurations as has been observed in some operating steam generators with conventional design and fabrication bases. (This is a consequence of explicit control of insertion depth during assembly.)

Basic vibration analyses employ the same qualified analytical models following a modal decomposition approach similar to that used for the straight-leg region. Threshold instability constants and scaling factors for random turbulence response are derived from results of testing summarized above. Analyses cover the range of possible fitup conditions determined by inspection of tube bundles during and after fabrication.

The evaluations performed for the Delta-109 and Delta-75 designs employed methods and criteria consistent with those summarized above. The results of the Delta-75 evaluations show that the calculated stability ratios (effective velocity/threshold velocity) for the expected conditions are a factor of two or greater below a limit of 1.0 which is associated with the instability threshold. The evaluations performed for the Delta-109 design demonstrate that the calculated stability ratios are a factor of 1.5 or greater below the limit. Similar results would be expected for the AP1000 tube bundle design.

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Design Control Document (DCD) Revision:

None

PRA Revision:

None

RAI Number 261.013

Question:

In the AP1000 DCD, Tier 2, Section 14.3, Table 14.3-2, "Design Basis Accident Analysis," page 14.3-30, states that "Nominal values for pertinent plant parameters used in accident analysis with 10% steam generator tube plugging - Reactor coolant flow (gpm) - 296,000 [as described in Section 15.03]." Table 5.1-3, "Thermal Hydraulic Parameters," states that the "Minimum Measured Flow (MMF), flow rate, gpm/loop, with 10% Tube Plugging" is 150,835 gallons-per-minute (gpm) or 301,670 gpm total reactor coolant flow. Table 15.0-3 lists the reactor coolant pump flow per loop with "Revised Thermal Design Procedure (RTDP) with 10% Steam Generator Tube Plugging" as 15.08 E+04 gpm which equals 301,600 gpm total reactor coolant flow. In DCD Chapter 16, "Technical Specifications," Section 3.4, "Reactor Coolant System," 3.4.1, "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits," LCO 3.4.1c, states "RCS total flow rate ≥[301,670] gpm." Based on these cross references to total reactor flow rates in the AP1000 design, the nominal total flow value with 10% steam generator tube plugging in Table 14.3-2 should be changed to 301,670 gpm.

Westinghouse Response:

The AP1000 DCD Table 14.3-2 provided in the current revision of the DCD contains the following:

Table 14.3-2 (Sheet 1 of 17)

DESIGN BASIS ACCIDENT ANALYSIS

Reference		**Design Feature**	**Value**
Table	5. 1-3	Minimum measured flow rate with 10% tube plugging (gpm/loop)	150,835

No additional changes are necessary.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

RAI Number: 280.004

Question:

Items 75 and 76 of Table 9.5.1-1 state that alternative or dedicated shutdown capability is not necessary. These statements are incorrect and conflict with Item 25 in the same table. As stated in NUREG 1512, Section 9.5.1.1.d, the staff concluded that the safety-related passive core cooling system (PXS) and passive containment cooling system (PCS) used to achieve and maintain safe shutdown following a fire in the AP600 are acceptable as an alternative/dedicated shutdown method for fire areas where the normal shutdown systems have not been protected in accordance with the guidance prescribed in the BTP. Please correct the discrepancy to be consistent with NUREG 1512 and Item 25 of Table 9.5.1-1 in the AP1000 DCD.

Westinghouse Response:

The DCD will be corrected as shown.

Design Control Document (DCD) Revision:

Correct spelling error in Item 25 of Table 9.5.1-1:

25. Alternative or dedicated shutdown capability should be provided where the protection of systems whose functions are required for safe shutdown is not provided by established fire suppression methods or by Position C.5.b.	C.1.d	AC	In Generic Letter (GL) 86-10, the staff stated its position that, for the purpose of analysis to Section III.G2 of Appendix R to 10 CFR Part 50 criteria, the safe shutdown capability is defined as one of the two normal safe shutdown trains. The safety-related PXS and PCS are used to achieve and maintain safe shutdown following a fire and are acceptable as an alternative/ dedicated shutdown method for fire areas where the normal shutudown systems have not been protected in accordance with the guidance prescribed in the BTP.

Revise Table 9.5.1-1, Item 75:

75. Provision of alternative or dedicated shutdown capability in certain fire areas.	C.5.b (3)	AC	**In Generic Letter (GL) 86-10, the staff stated its position that, for the purpose of analysis to Section III.G2 of Appendix R to 10 CFR Part 50 criteria, the safe shutdown capability is defined as one of the two normal safe shutdown trains. The safety-related PXS and PCS are used to achieve and maintain safe shutdown following a fire and are acceptable as an alternative/ dedicated shutdown method for fire areas where the normal shutdown systems have not been protected in accordance with the guidance prescribed in the BTP.**~~Safe shutdown systems are protected such that reliance on alternative or dedicated shutdown capability, as defined in 10 CFR 50 Appendix R, is not necessary.~~

Revise Table 9.5.1-1, Item 76:

76. Alternative or dedicated shutdown capability.	C.5.c	NC	**In Generic Letter (GL) 86-10, the staff stated its position that, for the purpose of analysis to Section III.G2 of Appendix R to 10 CFR Part 50 criteria, the safe shutdown capability is defined as one of the two normal safe shutdown trains. The safety-related PXS and**

PCS are used to achieve and maintain safe shutdown following a fire and are acceptable as an alternative/ dedicated shutdown method for fire areas where the normal shutdown systems have not been protected in accordance with the guidance prescribed in the BTP.~~Safe shutdown systems are protected such that reliance on alternative or dedicated shutdown capability, as defined in 10 CFR 50 Appendix R, is not necessary.~~

The criteria concerning cold shutdown capability deviates from the criteria applied to the evolutionary reactor designs, but is consistent with the criteria applicable to existing plants. To enhance the survivability of the normal safe shutdown and cold shutdown capability in the event of a fire, and to reduce the reliance on the infrequently utilized safety-related passive systems, automatic suppression is provided in those fire areas outside containment where a fire could damage the normal shutdown capability, or result in a spurious operation of equipment that could result in a venting of the RCS. This criterion does not ensure that the normal shutdown capability will be free of fire damage, or that the equipment necessary to achieve and maintain cold shutdown can be repaired within 72 hours.

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number. 280.005

Question:

Item 198 of Table 9.5.1-1 states that safety-related battery rooms are separated from associated electrical rooms of the same division by one-hour fire rated barriers. NUREG 1512 states that safety-related battery rooms in the AP600 are separated from each other and other plant areas by three-hour fire rated barriers consistent with the guidance specified in Position C.7.g of the BTP. Provide a technical justification supported by mathematical fire modeling for not providing three-hour fire barriers for the safety related battery rooms in the AP1000, or revise the DCD to be consistent with the BTP and NUREG 1512.

Westinghouse Response:

There has been no change in fire barrier rating from AP600 to AP1000. Item 198 of Table 9.5.1-1 of the AP600 DCD and the associated fire area drawing for AP600 show that there is a one-hour fire rated barrier between a battery room and the electrical room directly above it. In both AP600 and AP1000, for each electrical division, its battery room, its dc equipment room and its I&C / penetration room are all in the same fire area. This approach was not changed during the AP600 Design Certification review process and a search of the AP600 RAIs did not reveal any formal NRC question on this design approach. The wording in the AP600 DCD, the AP1000 DCD and NUREG-1512 that: "Safety-related battery rooms are separated from each other and other plant areas by three-hour fire rated barriers." is a true statement for separation among different fire areas of the plant.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number 280.006

Question:

Position C 5.c.(7) of the BTP states that the safe shutdown equipment and systems for each fire area be isolated from associated circuits such that hot shorts, open circuits, or shorts to ground will not prevent the operation of safe shutdown equipment and that a fire involving associated circuits will not prevent safe shutdown. Fires involving associated circuits may impact safe shutdown capability through loss of functions, flow diversions, blockage of flow paths, lost or misleading instrumentation, and loss of control. Consistent with this position, Section 9A.2.7.1 of the AP1000 DCD states that no postulated fire involving associated circuits will prevent safe shutdown; however, Section 9A.2.7.1 of the AP1000 DCD and Section 5.3.1.10 of WCAP-15871 states that only one worst case spurious actuation or signal results from a fire. These statements are not consistent with existing NRC guidance in (1) GL 86-10, "Implementation of Fire Protection Requirements," Question 5.3.1, that states that for consideration of spurious actuations all possible functional failure states must be evaluated, (2) Question 5.3.8 that states that simultaneous high impedance faults for all associated circuits located in the fire area be considered, or (3) Section 9.5.1.5.c of NUREG-1512, which considered the potential for multiple spurious actuations resulting from a fire in the review of the AP600. Additional clarification on the staff's position concerning circuit failures was provided to the nuclear industry in a March 11, 1997, letter to Mr. Ralph Beedle, Nuclear Energy Institute (NEI), from Mr. Samuel J. Collins, Director, Office of Nuclear Reactor Regulation. Section 9A.3.7.1 of the DCD addresses multiple spurious operations correctly in several systems but appears to be limited to high/low pressure interfaces. Please revise the AP1000 DCD and WCAP-15871 to be consistent with the staff's positions concerning circuit failures and spurious actuations.

Westinghouse Response:

AP1000 meets Position C.5.c. (7) of the BTP. Westinghouse agrees to revise DCD Section 9A.2.7.1 and WCAP-15871 as shown below.

DCD Section 9A.3.7.1 addresses both spurious actuations involving high-low pressure interfaces and others. Subsection 9A.3.7.1.1 addresses multiple spurious operations in several systems associated with high-low pressure interfaces. Subsection 9A.3.7.1.2 addresses principal spurious actuations not involving high/low pressure interfaces. Principle spurious actuations are those that could cause a breach in the reactor coolant boundary or defeat safety-related decay heat removal capability or cause an increase in shutdown reactivity of the reactor. In no case does the spurious actuation of equipment prevent safe shutdown. Revision of DCD Section 9A.3.7.1 is not necessary.

Design Control Document (DCD) Revision:

9A.2.7.1 Criteria and Assumptions

Spurious Actuation of Equipment

Fire-caused damage is assumed to be capable of resulting in the following types of circuit faults: hot shorts, open circuits, and shorts to ground. Spurious actuation of components caused by these circuit faults are evaluated. Components are assumed to be energized or de-energized by one or more of the above circuit faults. For example, air operated and solenoid operated valves are assumed to fail open or closed; pumps are assumed to fail running or not running; electrical distribution breakers could fail open or closed. For three-phase ac circuits, the probability of getting a hot short on all three phases in the proper sequence to cause spurious operation of a motor is considered sufficiently low as to not require evaluation, except for cases involving high-low pressure interfaces. For ungrounded dc circuits, if spurious operation could only occur as a result of two ungrounded hot shorts of the proper polarity, then no further evaluation is necessary, except for any cases involving a high-low pressure interface. Therefore, spurious operation of ac or dc motor operated valves as a result of power cable hot shorts is not assumed, except for cases involving a high-low pressure interface.

It is assumed that a fire results in the loss of all automatic function (signals and logic) from the circuits located in the fire area.~~, in conjunction with one worst case~~ In addition, spurious actuations or signals resulting from the fire are postulated one at a time (except for high/low pressure interfaces). The spurious actuations and signals that are evaluated are those that could cause a breach in the reactor coolant boundary or defeat safety-related decay heat removal capability or cause an increase in shutdown reactivity of the reactor.

Spurious actuation of the redundant valves in any one high-low pressure interface line are postulated if the circuits for those valves are located in the fire area.

PRA Revision:

None

WCAP Revision:

Section 5.3.1.10 of WCAP-15871 will be revised as follows:

> AC - The AP1000 fire hazards analysis assumes ~~a single worst case~~ spurious actuations regardless of cable failure mode except for valve motor operators. **The spurious actuations are postulated one at a time (except for high/low pressure interfaces). Spurious actuation of the redundant valves in any one high-low pressure interface line are postulated if the circuits for those valves are located in the fire area. The spurious actuations that are evaluated are those that could cause a breach in the reactor coolant boundary or defeat safety-related decay heat removal capability or cause an increase in shutdown reactivity of the reactor.**

RAI Number: 280.008

Question:

Section 57.4.1 of Chapter 57, "AP1000 Fire Risk Assessment" of the AP1000 Probabilistic Risk Assessment (PRA) (or the fire PRA) assumes the probability of a spurious signal impacting the automatic depressurization system (ADS) valves inside containment is an independent event. Section 5.3 of reference 10 (i.e., Circuit Analysis - Failure Mode and Likelihood Analysis issued by Sandia National Laboratory [SNL]) of the fire PRA states that the assumption that a given failure mode's conditional probability value is actually independent remains a questionable practice. Please provide a technical basis for the assumption that the probability of a spurious signal that has the potential to impact safe shutdown capability is independent.

Westinghouse Response:

For AP1000, an ADS valve spurious opening requires hot-shorts in multiple cables due to design. The Sandia National Laboratories (SNL) indicated that simultaneous hot shorts of multiple conductors in a single cable may not be independent, but did not comment on the dependence of hot shorts in multiple cables. Westinghouse developed a method on how the probability of spurious opening of an ADS valve may be calculated. One may assign a probability of a single hot-short and then multiply it by itself multiple times to model multiple hot shorts. Or a dependence model may be used among the hot-shorts to limit the total probability of multiple hot-shorts.

The following answers this interpretation of the question in the RAI. If this interpretation is not correct, then the answer is misdirected!

The AP1000 Fire PRA scenarios were systematically reviewed and those scenarios that modeled multiple hot shorts were identified. Table 280.008.1 is shown below and contains a listing of all fire areas where ADS actuations were shown to have occurred. Note that the products of individual hot-short probabilities, and their number of combinations modeled in the Fire PRA are shown in column 3 of the table. Column four in the table shows if these hot-shorts are associated with ADS Stage 4 or Stages 1,2 and 3, since there is a fundamental difference in the way the hot-shorts will affect these two types of ADS valves.

The fire areas used multiples of (.06) in the probability of spurious actuation. The multipliers fell into two categories: 1) spurious Stage 1, 2, or 3 ADS due to spurious actuation of both motor operated valves (MOVs) in the same line or 2) spurious actuation of Stage 4 squib valves due to spurious actuation then removal of the 'arm' circuit followed by spurious actuation of the 'fire' circuit. These two scenarios are different in nature and need to be dealt with individually.

Table 280.008.1 Multiple Hot Shorts Modeled in AP1000 Fire PRA

Fire Area	Description	Multiplier	1=Stages 1, 2 & 3 2 = Stage 4
1200 AF 03	Corridors 100'&117'6"	4(.06)(.06)(.06)	2
1201 AF 02	Division B Batteries, DC Equipment Room I&C	4(.06)(.06) 3(.06)(.06)	1
1201 AF 03	Division D Batteries, DC Equipment, I&C Room	4(.06)(.06) 3(.06)(.06)	1
1201 AF 06	MSIV Compartment B	2(.06)(.06)	1
1202 AF 02	Northeast Elevator Shaft	2(.06)(.06)	1
1202 AF 03	Division C Batteries, DC Equipment, I&C, RCP Trip Switchgear, I&C, Penetration Room	4(.06)(.06) 3(.06)(.06)	1
1202 AF 04	Division A Electrical Equipment, Battery, I&C Rooms	2(.06)(.06)	1
1210 AF 01	Corridor, Spare Battery Room, Spare Room, Spare Battery Charger Room	2(.06)(.06)	1
1220 AF 01	Division B RCP Trip Switchgear, Spare Room/Corridor 82' - 6"	2(.06)(.06)	1
1230 AF 01	Corridor Division A, B, C, D and Remote Shutdown Workstation	3(.06)(.06)	1
1230 AF 02	Non-Class 1E Electrical Compartment – Penetration Room	3(.06)(.06)	1
1232 AF 01	Remote Shutdown Workstation	3(.06)(.06)	1
1240 AF 01	Non-Class 1E Electrical Compartment - 117'	4(.06)(.06)	1
1242 AF 02	Division A Penetration Area	4(.06)(.06) 3(.06)(.06)	1
1243 AF 01	Reactor Trip Switchgear I	4(.06)(.06)(.06)	2
1243 AF 02	Reactor Trip Switchgear II	4(.06)(.06)(.06)	2
4031 AF 05	Access Area, Access Corridor, Security Room 2, Corridor, Rest Room	2(.06)(.06)	1
1100 AF11300A	Maintenance Floor (SE Quadrant Access)	2(.06)(.06)	1
1100 AF11300B	Maintenance Floor (NNE Quadrant) and RCDT Access	3(.06)(.06) 2(.06)(.06)	1
1100 AF11303A	ADS Lower Valve Area	3(.06)(.06)	1
1100 AF11303B	ADS Upper Valve Area	3(.06)(.06)	1
1100 AF 11500	Operating Deck	3(.06)(.06)	1
1200 AF 12341	Middle Annulus	2(.06)(.06)	1

1. Spurious Opening of Stage 1, 2, or 3 Valves

This category addresses the opening of two series MOVs that are in the same ADS line. Each MOV has its own power cable. The cables for the two valves are routed in the same tray. Sandia National Laboratories in its letter report, "Circuit Analysis - Failure Mode and Likelihood Analysis," (Reference 280.008-1) to the NRC stated that a hot short of a second conductor in the same cable can not be considered to be independent; however, the report had no conclusion regarding separate cables. And although Sandia's report did not support the assumption of independence, it also did not indicate that independence is wrong.

Section 7.2.3.5 of EPRI report "Spurious Actuations of Electrical Circuits Due to Cable Fires, Results of an Expert Elicitation" (Reference 280.008-2) addressed the issue of correlation for multiple spurious actuations and concluded that "this issue is too complex and the test data too difficult to work with". However, the report added that, "Based on an evaluation of the evidence, the two [probability of spurious actuation given cable damage] values used in the fire PRA should be taken as independent events, provided that the phenomena really do occur in different conductors." In the case of spurious Stage 1, 2, or 3 ADS, the conductors are not only different, but they are also in different cables.

The use of shielded cable for the AP1000 Stage 1, 2, or 3 ADS MOVs is included in the design specifications. Reference 280.008-1 stated that for shielded cable, "cable-to-cable shorts...would be virtually eliminated". Therefore in lieu of having an actual quantified value, a value of .06 was used for the spurious actuation of Stage 1, 2 and/or 3 MOVs in the AP1000 Fire PRA. This value was considered to be a conservative estimate based on the Sandia study and Sandia's terminology of "virtually eliminated".

2. Spurious Actuation of the "Arm'" and "Fire" Circuits for a Stage 4 Squib Valve

A value of Qxx = (.06) x (.06) x (.06) = 2.2E-04 was used for spurious actuation of arm and fire circuits for a Stage 4 squib valve. The required sequence for this category is for spurious actuation of the 'arm' circuit, followed by removal of this actuation, followed within 2 minutes by a spurious actuation of the fire circuit. The time between the actuation and the removal of the arm signal is not critical. Because the arm and fire circuits use separate cables, there is independence between arm and fire circuits. The probability of the arm circuit being removed perhaps may be higher than (.06); however, the probability of the actuation of the fire circuit within a two-minute window is considerably lower than (.06). So the question comes down to whether the 2.2E-04 probability of a single ADS fourth stage line spuriously opening during a fire event is a realistic representation of the design precautions built into the ADS 4^{th} stage to minimize spurious opening. It is our contention that the probability of 2.2E-04 is a good representation of the spurious opening probability of a single ADS 4^{th} stage valve, given a fire event that involves the arm and fire circuits.

It was recognized that the AP1000 Fire PRA results were sensitive to the hot-short modeling and to the uncertainty in its probability. The sensitivity analysis given in Section 57.7.2 of the AP1000 Fire PRA shows that, if the hot-short probability is increased by a factor of 2, the CDF increases from its base case value of 5.61E-08/yr to 1.57E-07/yr. This indicates that the model is sensitive to the probabilities used for hot-shorts causing ADS actuation. Note that the value calculated above (Qxx) becomes 1.7E-03 in the sensitivity analysis.

Conclusion

As explained above, it is believed that the modeling and the quantification of the probabilities of spurious opening of ADS valves due to hot-short in a fire event in the AP1000 Fire PRA is a realistic representation of the design features already built into the AP1000 to minimize spurious opening of ADS lines from all causes, including the fire events. The multiple hot-short probabilities used (regardless of the way they are calculated) are reasonable since there are no single cables that contain multiple conductors that could spuriously open a single ADS line as a result of a conductor-to-conductor fault. Moreover, the favorable conditions mentioned in the Sandia Report (Reference 280.008-1) for minimizing the occurrence of multiple hot-shorts is met in the AP1000 design.

REFERENCES

280.008-1. "Circuit Analysis - Failure Mode and Likelihood Analysis," A letter report to the USNRC, Sandia National Laboratories, May 8, 2000.

280.008-2. "Spurious Actuations of Electrical Circuits Due to Cable Fires, Results of an Expert Elicitation," EPRI, May 2002.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number. 280.009

Question:

Section 57.4.5.3.2 of the fire PRA only considers the potential effects of smoke from a fire on personnel performance, no assessment of the potential impacts on plant equipment has been provided. NUREG/CR-6597, "Results and Insights on the Impact of Smoke on Digital Instrumentation and Control," published in January 2001, concluded that smoke has the potential to be a significant environmental stressor that can result in adverse consequences. Please provide an analysis on the potential risk impacts of smoke on plant equipment.

Westinghouse Response:

In the fire scenarios studied for the AP1000 Fire PRA, EPRI FIVE methodology (Reference 208.009-1) was used. When a fire was postulated in an area, all components in the area are assumed to be inoperable, whether this follows from the fire or smoke damage. Moreover, it was modeled that even qualified fire barriers may fail with a finite probability (0.01). In that case, the most consequential neighboring fire area was assumed to be affected; all components in this neighboring area were also assumed to be inoperable due to fire or smoke damage. No credit was taken for a component to survive the fire in an affected fire area. Thus, there is no need for further modeling of smoke damage in a fire area. The same applies to the neighboring area, as discussed above.

The conclusion in NUREG/CR-6597 regarding the consequence of smoke to have adverse consequences apply to realistic fire PRA studies where credit is taken for survival of equipment after a fire. If more realistic fire analysis were to be performed and credit were to be taken for survival of some equipment after the fire, then it would have made sense to investigate the effect of smoke damage on components in the fire area or in the most dominant neighboring area. Since this is not the case in the AP1000 Fire PRA, the potential smoke damage is already included in failure of the components, and further investigation is not needed.

The above apply both to the fire areas outside the containment and the fire zones considered in the containment. In regard to the propagation of smoke to more than one fire zone in the containment, it is assessed that:

i.) it is a residual effect (compared to the conservative assumption that both the fire zone and the most consequential neighboring area are assumed totally disabled)

ii.) the equipment in the containment is already designed for harsh environmental conditions for post LOCA scenarios and the effect of smoke would be no harsher that those failure modes already imposed post LOCA.

Thus, the scenarios modeled in the containment fire zones also capture the main contributors to plant risk.

An additional consideration is the following:

Section C.4.5 of NFPA 805, "Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants," (Reference 280.009-2) states "Smoke from a fire that starts in one zone can propagate to other zones and potentially damage additional equipment. Currently, fire PSAs do not treat the question of smoke propagation to other areas and their effect on component operability in a comprehensive manner. The extent to which the issue is addressed depends on the analyst, and if it is addressed, it is typically addressed qualitatively."

Design Control Document (DCD) Revision:

None

PRA Revision:

None

References:

280.009-1 Electric Power Research Institute (EPRI) Report, "Fire-Induced Vulnerability Evaluation Methodology (FIVE) Plant Screening Guide," Revision 1, September 1993.

280.009-2 NFPA 805, "Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants," 2001 Edition.

RAI Number: 280.010

Question:

Section 57.7.2 of the fire PRA evaluates the risk impact associated with the spurious actuation of ADS. No other spurious actuations have been addressed in the fire risk assessment. Fire-induced spurious actuations may impact safe shutdown capability through loss of system performance, flow diversions, blockage of flow paths, lost or misleading instrumentation, and loss of systems control. Provide a technical basis for excluding all other potential spurious actuations from the fire risk assessment, or provide an assessment of the risk impact for all potential spurious actuations that may prevent the operation or cause maloperation of systems needed to achieve and maintain safe shutdown for the AP1000.

Westinghouse Response:

The RAI was examined for two areas: spurious actuation of safety systems, and non-safety systems.

I. Potential Spurious Actuation of Safety Systems Due to Fire

Spurious actuations of the passive safety systems other than ADS has been considered in the design (see DCD 9A.3.7.1.2 below). Spurious ADS was identified as the only spurious actuation of a safety system that is of potential consequence (loss of reactor shutdown, coolant or cooling). The following excerpt is taken from the DCD:

"9A.3.7.1.2 Other Spurious Actuation

Principal spurious actuation not involving high-low pressure interfaces are discussed below.

Passive Core Cooling System Passive Residual Heat Removal Heat Exchanger Inlet Valve Actuation

One normally open valve is provided to isolate the inlet line to the passive residual heat removal heat exchanger. To preclude the spurious closing of the inlet valve as a result of a fire, the power to the valve is locked out during power operations. Thus, spurious closing of the passive core cooling passive residual heat removal heat exchanger inlet valve does not occur and the safe shutdown capability is not affected.

Passive Containment Cooling System Valve Actuation

Two valves in series isolate each of the three discharge flow paths from the passive containment cooling system storage tank. For purposes of system reliability, one valve in each flow path is normally open and the other is normally closed. Electrical division assignments are shown in Table 9A-2.

Spurious actuation of one of these valves is assumed to occur where a fire affects its electrical circuitry. Such a fire can occur in the main control room, an electrical equipment fire area, in the passive containment cooling system valve room, or in fire areas or fire zones through which the applicable electrical cables are routed.

Spurious actuation of one of these valves causes a passive containment cooling system flow path to be disabled or inadvertently opened, depending on which valve is affected. If a normally closed valve spuriously opens, passive containment cooling system water delivery from that flow path will be initiated which does not adversely affect the capability to achieve and maintain safe shutdown. If one of the normally open valves were spuriously closed to prevent passive containment cooling system water delivery through that flow path when called upon during the safe shutdown process, the redundant passive containment cooling system water delivery flow paths would be sufficient to achieve and maintain safe shutdown.

Containment Isolation Valve Actuation

Spurious actuation of a containment isolation valve is assumed to occur where a fire affects its electrical circuitry. Each containment penetration has redundant means of containment isolation.

Reactor Trip Switchgear

The reactor trip switchgear receives signals from each of the four Class 1E electrical divisions. The signals are de-energized to trip. Also, two out of four signals are required to trip. There are two redundant sets of trip switchgear in separate fire areas. There is no single spurious signal which could prevent the reactor from being tripped.

Reactor Coolant Pump Trip Switchgear

There are two redundant sets of reactor coolant pump trip switchgear in separate fire areas. One is controlled from division B; the other from division C. Thus, a spurious signal in either train will not prevent trip of the reactor coolant pumps."

II. Potential Spurious Actuation of Non-Safety Systems Due to Fire

Spurious actuation of the non-safety systems was not evaluated in the deterministic analysis because no spurious actuation of a non-safety system would defeat the passive safety systems.

Likewise, in the AP1000 Fire PRA, spurious actuation of the non-safety systems was not evaluated. However, most of the documented problems associated with spurious actuation defeating the ability of a shutdown system are spurious actuations associated with control room fires. Specifically, hot shorts in the wiring to the main control board defeating the ability to shut down from the remote shutdown location. For control room fires, analysis about soft controls indicated that spurious actuation could not happen for control room fires. For fires outside the control room, it was assumed that there was a total loss of the equipment in the fire area. It was then considered that there was no difference between a loss because of loss of function or because of spurious actuation.

Table 280.010-1 shows the non-safety-related systems credited in the AP1000 PRA. Each non-safety related system is examined to determine if its spurious actuation can fail another safety or non-safety system. First, by design, no spurious actuation of a non-safety system can fail a safety system. This leaves the other non-safety systems that can be affected by spurious actuation of a non-safety system. Table 280.010-1 provides the result of this investigation.

Note that it is possible that a non-safety system may be spuriously actuated by a fire event, and fail due to this premature actuation (such as normal RHR pumps burning out). However, this failure mode is already subsumed in the existing conservative modeling assumption that if a system or its actuation cables/cabinets are in a fire area, that system is considered failed.

Table 280.010-1 AP1000 Non Safety-Related Plant Support Systems

System	Description	Effect of Spurious Actuation During Fire
CAS	Compressed and Instrument Air	None
CCS	Component Cooling Water	None
DAS	Diverse Actuation	None
EDS	Non Class 1E DC and UPS	None
ECS	Main AC Power	None
FWS	Main Feedwater	None
PLS	Plant Control	None
RNS	Residual Heat Removal	None
SWS	Service Water	None
VWS	Central Chilled Water	None

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number 280.011

Question:

Section 57.8 of the fire PRA states that the Containment (Fire Area 1000AF 01) core damage frequency (CDF) is an important plant contributor to the plant fire CDF. Table 57-9 indicates that approximately 41 percent of the total fire-induced CDF is assigned to the containment. Please provide a mathematical fire model (for each of the fire zones inside the Containment/Shield Building where redundant safe shutdown components required following a fire have not been separated by complete fire barriers) that supports the statements in the AP1000 DCD that a fire will be confined to the zone of origin such that the redundant components will remain free of fire damage. This includes the following fire zones: 1100 AF 111204, 1100 AF 11206, 1100 AF 11207, 1100 AF 11208, 1100 AF 11300A, 1100 AF 11300B, 1100 AF 11301, 1100 AF 11302, and 1100 AF 11500. Guidance on the application of fire modeling to nuclear power plant fire hazard analysis is provided in Appendix C of NFPA 805.

Westinghouse Response:

The fire analysis presented in the AP1000 PRA uses a performance based approach consistent with the Electric Power Research Institute (EPRI) Report "Fire-Induced Vulnerability Evaluation Methodology (FIVE) Plant Screening Guide," Revision 1, September 1993. The fire analysis presented in the AP1000 DCD is a deterministic approach consistent with that used for AP600 and endorsed by NUREG-1512.

The FIVE methodology states that there is a low probability that a fire may occur in containment during operation. As a result, a quantitative mathematical fire model is not required. In addition, Appendix C of NFPA 805 does not require explicit mathematical modeling of fires if the FIVE methodology is used. As indicated in Attachment 57C, "Fire Area Event tree Defining Scenarios," of the AP1000 PRA, an appropriate probability of fire propagation to an adjacent fire zone in containment was included in the overall probabilistic analysis. The propagation frequencies assigned were consistent with the FIVE methodology, the physical arrangement of fire sources and fire barriers in containment, and the importance to safety of equipment in adjacent zones. As indicated in Table C.2.2(b) of NFPA 805, this technique provides an initial screen that leads to the use of PRA techniques with look up tables. The resulting probabilistic analysis leads to the conclusions of Chapter 57 of the AP1000 PRA.

As indicated in Appendix 9A of the AP1000 DCD, fire sources were identified in each fire zone and their position relative to zone boundaries were established. Then design features were identified which minimize the potential for fires to propagate from zone to zone. As a result of these specific design features, this deterministic analysis results in no propagation among zones within containment.

The AP1000 PRA states that the total fire CDF is small based on a probabilistic analysis and the AP1000 DCD states that no fire in a single zone in containment can prohibit safe shutdown of the plant. These statements are both valid within their own context.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 410.001

Question:

(DCD Sections 3.6.1 and 3.4.1) It is not clear to the staff what the design basis is for the protection against pipe breaks in non-seismically supported moderate energy lines for the AP1000. Item F of Section 3.6.1.1 of the DCD states that "[f]or systems not seismically analyzed for a safe shutdown earthquake, the safe shutdown earthquake is assumed to cause a pressure boundary failure." In Section 3.4.1.2.2, you identify in the second of six bullets that the (internal) flooding sources considered in the flooding analysis include, "moderate-energy piping (through wall cracks)." In the staff's AP600 FSER (NUREG-1512), Section 3.4.1.2, we listed the same 6 flood sources. However, in the FSER, the second bullet is identified as "moderate-energy (*breaks and* through-wall cracks)," which implies that the staff believed full breaks were also considered in the flooding analysis for non-seismically supported moderate-energy lines.

The staff identified this concern in AP600 RAI 410.403F during the AP600 review. Your response indicated that you believed that this was a change in guidance and provided a response that indicated the plant design could withstand the flooding effects from a full pipe break (double-ended rupture) in any non-seismically supported moderate-energy pipe. However, it is not apparent that changes were made to the DCD to indicate that ruptures are assumed (and analyzed for environmental effects, i.e., flooding) in non-seismically supported moderate-energy piping.

(The staff does not consider this a change in guidance as we have always required that plants be capable of a safe shutdown following a full break (as a result of a seismic event) in any non-seismically supported moderate-energy line coupled with a worst-case single active failure. Section B.3.d of BTP 3.6.1, attached to SRP Section 3.6.1, Revision 1, is intended to identify the above as the staff's position with respect to non-seismically supported moderate-energy piping.)

Please verify that you have analyzed the AP1000 design for the flooding effects associated with full pipe breaks in non-seismically supported moderate energy piping systems, concurrent with any single active failure, and that the plant can still achieve and maintain a safe shutdown condition. A complete analysis is not necessary for those areas of the plant where the effects of such a break are obviously bounded by other piping systems in those areas. You should also revise the DCD to reflect this as the design basis.

NOTE: AP600 RAI 410.403F was issued by the staff on December 17, 1997 (NUDOCS Accession No. 9802040013). Westinghouse provided its response on January 9, 1998 (NUDOCS Accession No. 9801150055).

Westinghouse Response:

The Westinghouse response to AP600 RAI 410.403F also applies to AP1000. The two paragraphs from that response that most pertain to this RAI are repeated below.

"Safety-related systems required for safe shutdown are not expected to be adversely affected by the dynamic effects of postulated pipe breaks in non-seismic, moderate-energy piping. By design, non-seismic piping is not routed near safety-related piping or equipment. If there is B31.1 piping whose continued function is not required, but whose failure or interaction could degrade the functioning of a safety class component to an unacceptable level, then this B31.1 piping is analyzed and designed for the SSE using the same methods as specified for seismic Category 1 piping. For example, non-safety-related piping connected to safety-related components is analyzed and designed for seismic loadings, because the piping model includes piping adjacent to the containment penetration area up to the first anchor.

The effect of moderate-energy line breaks on safety-related equipment inside containment and in compartments outside of containment that include high-energy lines are bounded by the effects of the high-energy lines. The compartments outside of containment that include safety-related components and moderate-energy lines, and do not contain high-energy lines are limited to a few rooms containing containment isolation valves in the auxiliary building and the PCS valve room located above the containment near the shield building roof. The PCS valve room does not include non-seismically analyzed, moderate-energy piping. The moderate-energy lines connected to the containment isolation valves are analyzed seismically from the penetration up to the anchor. The turbine building, annex building, and the radwaste building do not contain safety-related systems or components and are not evaluated."

Specifically for the AP1000 DCD, Section 3.6.1.1 F states: "For systems not seismically analyzed for a safe shutdown earthquake, the safe shutdown earthquake is assumed to cause a pressure boundary failure". This statement applies to both high energy and moderate energy piping.

Design Control Document (DCD) Revision:

Section 3.4.1.2.2 of the DCD will be clarified as follows:

"The flooding sources considered in the analysis consist of the following:
- High-energy piping (breaks and cracks)
- **Through-wall cracks in seismically-supported moderate energy piping**
- **Breaks and through-wall cracks in non-seismically-supported moderate energy piping**
- Pump mechanical seal failures
- Storage tank ruptures
- Actuation of fire suppression systems
- Flow from upper elevations and adjacent areas"

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 410.002

Question:

(DCD Section 3.6.1 and Appendix 3E) Appendix 3E of the DCD indicates that the hot water heating system (VYS) contains a limited amount of high energy piping in the auxiliary building (3-inch supply and return headers). You also state that "[t]here are no anchors or fittings on these lines in the nuclear island. Therefore, there are no postulated pipe breaks in these lines on the nuclear island." However, the VYS is identified as a Class E system (non-seismic) in Table 3.2-3 of the DCD. If the piping in this system is neither seismically analyzed nor seismically supported, an analysis must be performed to assess the effects of a postulated double-ended rupture of this piping (coupled with a single active failure) in areas with safe shutdown equipment and to assure that safe plant shutdown can still be achieved and maintained. A complete analysis is not necessary for areas where the effects would obviously be bounded by other pipe breaks in those areas. Please provide this analysis and revise the DCD as necessary.

Westinghouse Response:

There are 3" lines and 1" lines in the hot water heating system (VYS) on the nuclear island. There are no 3" lines in auxiliary building sub-compartments that include seismic category 1 systems or components (safe shutdown equipment). The VYS lines in the auxiliary building sub-compartments that include safety-related equipment are restricted to pipes 1 inch and smaller NPS. Pipe breaks are not postulated in piping runs of a nominal diameter equal to or less than one inch. This system ties into the central chilled water system (VWS) for containment heating during cold weather plant outages. Since this part of the system is activated only during shutdown events, it qualifies as a moderate energy system (system experiences high-energy conditions for less than two percent of the system operating time). Moderate energy systems are not evaluated for pipe failures at the occasional high-energy conditions.

Appendix 3E of the DCD has been revised as part of the process of addressing RAI 210.057 and now reads as follows:

> *"In addition to the high-energy pipe identified in the figures, the hot water heating system (VYS) includes a limited amount of high-energy piping in the auxiliary building. The subject piping is the 3 inch-diameter supply and return header piping for the heating coils in HVAC equipment in the auxiliary building. The hot water heating system lines in the auxiliary building sub-compartments that include seismic category 1 systems or components are restricted to pipe sizes less than or equal to 1 inch NPS. Therefore, there are no postulated pipe breaks in these lines on the nuclear island."*

Design Control Document (DCD) Revision:

None – Please see the response to RAI 210.057 for a related change to the DCD.

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 410.003

Question:

Paragraph III.5.d of Section 10.3 of the SRP states that the main steam isolation valves, shut-off valves in connected connecting piping, turbine stop valves, and bypass valves should be able to close against maximum steam flow. Verify that these valves are capable of being closed against maximum steam flow.

Westinghouse Response:

The main steam isolation valves, shut-off valves in connected connecting piping, turbine stop valves, and bypass valves are capable of being closed against maximum steam flow.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 410.004

Question:

(Section 10.4.1) Section 10.4.1.2.1 of the DCD states "[r]efer to Table 10.3.5-1 for permissible cooling water in-leakage and time of operation for maintaining the required condensate/feedwater quality." Where is this information found on Table 10.3.5-1?

Westinghouse Response:

Table 10.3.5-1 provides acceptable water chemistry limits but does not include "permissible cooling water and time of operation for maintaining the required condensate/feedwater quality." DCD subsection 10.4.1.2.1 will be revised to refer to DCD subsection 10.3.5.5 "Action Levels for Abnormal Conditions" as shown below in the "Design Control Document (DCD) Revision:" portion of this RAI response.

Design Control Document (DCD) Revision:

Revision (to 2nd to last paragraph of "10.4.1.2.1 System Operation":

The main condenser interfaces with secondary sampling system (SSS) to permit sampling of the condensate in the condenser hotwell. Also, grab sampling capability is provided for each condenser tubesheet. Should circulating water in-leakage occur, these provisions permit determination of which tube bundle has sustained the leakage. Steps may be taken to repair or plug the leaking tubes. This is performed by isolating the circulating water system from the affected water box. Plant power is reduced as necessary. This will temporarily reduce condenser capacity by approximately 50 percent. The water box is then drained and the affected tubes are either repaired or plugged. Refer to ~~Table 10.3.5-1 for permissible cooling water inleakage and time of operation~~ **subsection 10.3.5.5 for a discussion regarding action levels for abnormal secondary cycle chemistry conditions** ~~maintaining the required condensate/feedwater quality~~.

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 410.008

Question:

(DCD, Tier 2, Section 6.4 and Chapter 16 for TS 3.7.6) Chapters 6 and 16 state the following:

(1) Technical Specification Bases for SR 3.7.6.5: Verification of the initial air quality (in combination with other surveillances) ensures that breathable air is available for 11 main control room (MCR) occupants for at least 72 hours.

(2) Technical Specification Bases for SR 3.7.6.10: One VES air delivery flow path using the safety-related compressed air storage tanks, pressurizes the MCR envelope (MCRE) to at least a positive 1/8 inch water gauge pressure relative to the surrounding spaces at the required air addition flow rate of 65± 5 standard cubic feet-per-minute (scfm).

(3) Section 6.4.4 of the DCD: The VES maintains carbon dioxide (CO_2) concentration to less than 0.5 percent for up to 11 MCR occupants.

(4) Section 6.4.4 of the DCD: The VES nominally provides 65 scfm of ventilation air to the MCR from the compressed air storage tanks. Sixty scfm of ventilation flow is sufficient to pressurize the control room to at least positive 1/8-inch water gauge differential pressure with respect to the surrounding areas in addition to limiting the CO_2 concentration below ½ percent by volume for a maximum occupancy of 11 persons and to maintain air quality within the guidelines of Table 1 and Appendix C, Table C-1, of Reference 1 ([American Society of Heating, Refrigerating, and Air Conditioning Engineers] ASHRAE Standard 62-1989).

Provide detailed justification as to why the AP600 MCRE design is equally applicable to the AP1000 design regarding compliance with the requirements of GDC 19 with respect to maintaining the safety-related radiation protection, toxic protection, and cooling functions. Since the AP1000 plant thermal rating is substantially higher than that of the AP600 design thermal rating, your detailed rationale should provide a discussion that includes, but is not limited to, the number of MCRE occupants, VES system capacity and capability, system redundancy to meet single failure criteria, safety-related system, structures, and components, and breathing air quality to meet U.S. Occupational Safety and Health Administration (OSHA) and ASHRAE Standards.

Westinghouse Response:

The AP1000 main control room emergency habitability system (VES) design is identical to the AP600 VES design and therefore the AP1000 VES capacity, system redundancy to meet single failure criteria, and safety-related system, structures, and components are the same as those in the AP600 VES design.

The control room envelopes are also identical in both plant designs.

The AP1000 heat source loads are the same as AP600's because the I&C designs (and their associated heat loads) are essentially the same and are required to be below those assumed in the AP1000 VES design basis (which is identical to AP600 VES design basis).

The number of AP1000 MCRE occupants is the same as for AP600 because the plant control room and required number of operating and emergency personnel are identical in both plant designs.

There is a difference in the main control room doses due to the higher power rating of the AP1000, however, it does not require a change to the VES design. The main control room doses analyses for the AP1000 are presented in DCD subsection 15.6.5.3.5. In addition, please refer to the response to RAI 470.006 for a discussion regarding the AP1000 main control room doses.

The AP1000 VES design meets all required radiation protection, toxic protection, and cooling functions and provides MCRE occupants with breathing air quality that meets the guidelines of Table 1 and Appendix C, Table C-1, of the, "Ventilation for Acceptable Indoor Air Quality," ASHRAE Standard 62 - 1989.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

RAI Number: 410.011

Question:

(DCD, Tier 2, Section 9.4.1) You stated in response to AP600 RAI 410.240 that Table 9.4-1 would be revised to summarize the plant areas served by nuclear filtration systems with their associated design/testing standards, filtration efficiency, design air flow rates, humidity control, charcoal adsorber thickness and maximum in-leakage flow. You also stated that the AP600 standard safety analysis report (SSAR) (DCD, Tier 2) would be revised to add Table 9.4-2 which would identify the minimum instrumentation and controls for nuclear filtration systems (as per RG 1.140, "Design, Testing, and Maintenance for Normal Ventilation Exhaust System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants") based on American Society of Mechanical Engineers (ASME) Code N509 Table 4-2 criteria.

In the same AP600 RAI response, you also stated that the SSAR (AP600 DCD, Tier 2), Chapter 11 would be revised to reference Chapter 9, Table 9.4-1, to reflect gaseous radwaste management performance to state that "[i]n addition to the gaseous radwaste system release pathway, release of radioactive material to the environment occurs through the various building ventilation systems. These systems are described in Section 9.4 with a summary of system air flow rates and filter efficiencies provided in Table 9.4-1. The estimated annual release reported in Section 11.3.3 include contributions from the major building ventilation pathways." These statements were to be inserted before Subsection 11.3.1 on page 11.3-1.

However, it appears that Table 9.4-1 was not revised to include the above information for the health physics area, radwaste building and radiation chemistry laboratory and that Table 9.4-2 was eliminated. Additionally, it appears that Section 11.3.1 was not revised to reflect the above information concerning the gaseous radwaste system. Please revise AP1000 DCD, Table 9.4-1, add Table 9.4-2, and revise Section 11.3.1 to reflect your AP600 RAI response and the above information or provide justification for their exclusion.

NOTE: AP600 RAI 410.240 was issued by the staff on May 23, 1994 (NUDOCS Accession No. 9406230015). Westinghouse submitted its response on July 22, 1994 (NUDOCS Accession No. 9407270187).

Westinghouse Response:

Re: Table 9.4-1

At the time of the issuance of Westinghouse's response to RAI 410.240 for AP600, the HVAC systems serving the MCR/TSC and the containment, as well as the health physics area, the radwaste building and the radiation chemistry laboratory included HEPA filters and/or charcoal adsobers. The design of the filtration systems followed the guidance of RG 1.140. Later, the designs of the AP600 health physics area, the radwaste building and the radiation chemistry laboratory HVAC systems were modified and filtration systems are no longer required in these systems. The AP600 SSAR (DCD Tier 2) was revised accordingly and that SSAR revision appropriately removed the health physics area, the radwaste building and the radiation chemistry laboratory HVAC systems from Table 9.4-1. Therefore, the latest AP600 SSAR Table 9.4-1 includes all plant areas served by nuclear filtration systems with their associated design/testing standards, filtration efficiency, design air flow rates, humidity control, charcoal adsorber thickness and maximum in-leakage flow and no revision to the table is required. For the same reason, no revision to the AP1000 DCD Table 9.4-1 is required.

Re: Table 9.4-2
Table 9.4-2, "Minimum Instrumentation for Atmospheric Cleanup Systems" was added to the AP600 SSAR as stated in the response to AP600 RAI 410.240, however, it was removed in a later revision of the SSAR. As a result, the NRC issued AP600 RAI 410.353F, which requested Westinghouse to reinstate into the AP600 SSAR Table 9.4-2. In the response to AP600 RAI 410.353F, Westinghouse stated:

"SSAR subsection 9.4.7.5 provides a description of instrumentation associated with the containment filtration system. The description provided is consistent with the applicable items in Table 4-2 of ASME N509, Note that all instrumentation provided with the system can be monitored and alarmed in the main control room as described in Chapter 7. The specifics of display and alarm will be developed as part the human factors implementation process described in SSAR Chapter 18. Since the balance of the SSAR will be used as a guide for the level of detail to be implemented during this process, a statement will be added to subsection 9.4.7.5 which references system consistency with ASME N509."

Westinghouse then added to AP600 SSAR subsection 9.4.7.5, first paragraph, a new last sentence:

"Display and monitoring of system instrumentation is consistent with the requirements of Table 4-2 of ASME N509 (Reference 2)."

The aforementioned statement does appear in both the AP600 SSAR and the AP1000 DCD. Table 9.4-2 was not reinstated in the AP600 SSAR. For the same reasons, Table 9.4-2 is not required in the AP1000 DCD.

Re: Chapter 11
Clarification to the AP1000 RAI Number 410.011 question, second paragraph. The RAI question presently states, "In the same AP600 RAI response,...", in which the NRC is referring to Westinghouse's response to AP600 RAI 410.240. Westinghouse believes that the RAI question is intended to state, "In response to AP600 RAI 410.242,...", as there is no discussion of Chapter 11 in AP600 RAI 410.240. Also, as an additional clarification to the AP1000 RAI Number 410.011 question. The latter portion of the RAI question requests Westinghouse to **revise subsection "11.3.1"** of the DCD, while the earlier portion of the RAI question identifies the paragraph of concern to be "**before** Subsection 11.3.1 on page 11.3-1." Westinghouse

believes that the question is intended to focus on subsection 11.3 the subsection **before** subsection 11.3.1.

Westinghouse agrees to revise AP1000 DCD subsection 11.3 to be consistent with the response to AP600 RAI 410.242. Please refer to the AP1000 DCD revision section below.

Design Control Document (DCD) Revision:

At the end of the introduction to AP1000 DCD 11.3 (after the second bullet), add the following paragraph:

"In addition to the gaseous radwaste system release pathway, release of radioactive material to the environment occurs through the various building ventilation systems. These systems are described in Section 9.4 with a summary of system air flow rates and filter efficiencies provided in Table 9.4-1. The estimated annual release reported in Section 11.3.3 include contributions from the major building ventilation pathways."

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 410.012

Question:

(DCD Tier 2, Section 9.4.1) The NRC staff's RAI 410.371F, Item 2 for AP600 Tier 2 Material requested that Westinghouse "[s]tate in the text of the SSAR that the VES flow capacity conforms to: (1) the MCRE flow design "Table 1, and Appendix C Table C-1" of ASHRAE Standard 62-1989, "Ventilation for Acceptable Indoor Air Quality" and (2) 1993 ASHRAE Handbook, "Fundamentals SI Edition," Chapter 23.2, "Ventilation and Indoor Air Quality," since these references provide the appropriate guidelines for maintaining the carbon dioxide concentration limits below one-half percent by volume for a maximum occupancy of eleven persons inside the MCRE." Westinghouse responded with the markups for AP600 SSAR (DCD Tier 2) Sections 6.4 and 9.4. However it appears that the AP1000 DCD Tier 2 Section 9.4.1.2.1.1 was not revised to include the proposed statement that "[t]he high air flow rates provided by the VBS system operation precludes the CO_2 concentration in the MCR exceeding the 0.05 % limit." Please revise AP1000 DCD Tier 2 Section 9.4.1.2.1.1 (2nd sentence in the 3rd paragraph) accordingly or provide your rationale for its exclusion.

NOTE: AP600 RAI 410.371F was issued by the staff on December 8, 1997 (NUDOCS Accession No. 9712120340). Westinghouse submitted its response on December 29, 1997 (NUDOCS Accession No. 9801140150).

Westinghouse Response:

For clarification, Westinghouse notes that the CO_2 limit is one-half percent, as identified in the first part of the RAI question. In a later part of the RAI question, a value of "0.05 %" appears and is considered to be a typographical error. I.e. "0.05 %" should be "0.5 %".

In revision 20 of the AP600 SSAR, Westinghouse added to subsection 9.4.1.2.1.1, the NRC identified statement pertaining to the CO_2 concentration limit of one-half percent. In revision 21 of the AP600 SSAR, that same statement was moved from subsection 9.4.1.2.1.1 to the seventh bullet of subsection 9.4.1.1.2 "Power Generation Basis, Main Control Room/Technical Support Center Areas" consistent with Westinghouse's revised response to AP600 RAI 410.371F Revision 1. That bullet also appears in the AP1000 DCD and is repeated here for information.

"

- Maintains the main control room/technical support center carbon dioxide levels below 0.5 percent concentration and the air quality within the guidelines of Table 1 and Appendix C, Table C-1 of Reference 32."

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 410.015

Question:

(DCD Tier 2, Sections 6.4 and 9.4) During the AP600 design certification review, the NRC staff requested (in AP600 RAI 410.415F) that Westinghouse provide clarification regarding the location (i.e., located in the MCR, local, or both) of the system instrumentation (such as pressure indications and high differential pressure alarms for the system filters and unit coolers, airflow indication and alarms to monitor operation of the supply and exhaust fans, etc.), for the HVAC systems (VBS, VXS, VAS, VCS, VFS, VRS, VTS, VHS, and VZS).

You stated in your response to the above AP600 RAI that: "The A600 has a plant-wide network that provides pre-processed plant data to those locations where the information is required. Because of the rapid changes that are taking place in the digital computer and graphic display technology employed in a modern human system interface, design certification of the AP600 focuses upon the process used to design and implement human system interfaces for the AP600 rather than on the details of the implementation. As a result, SSAR Chapter 18 describes the processes used to provide human factors engineering in the design of the AP600. The specifics of display and alarm will be developed as part of the human factors implementation process. In general, variables discussed in this question are expected to be available both in the control room and at remote "data port" locations throughout the plant with the use of a portable data display device."

Please clarify in the AP1000 DCD system descriptions (including VES, VBS, VXS, VAS, VCS, VFS, VRS, VTS, VHS, and VZS) where the instrumentation information is provided; is the information provided locally, or in the main control room (MCR), or is it provided in both places? Provide the rationale for its exclusion.

NOTE: AP600 RAI 410.415F was issued by the staff on December 18, 1997 (NUDOCS Accession No. 9802030132). Westinghouse submitted its responses on January 1 and January 27, 1997 (NUDOCS Accession Nos. 9801130177 and 9802050165).

Westinghouse Response:

The response to the AP600 RAI 410.415F remains valid for the AP1000. I.e. "The specifics of display and alarm will be developed as part of the human factors implementation process. In general, variables discussed in this question are expected to be available both in the control room and at remote 'data port' locations throughout the plant with the use of a portable data display device." As the human factors implementation process has yet to be implemented, Westinghouse proposes no changes to the AP1000 DCD at this time.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 410.021

Question:

Provide appropriate verifications for the concerns identified below for the subject AP1000 HVAC systems and/or identify where these verification discussions are provided in the AP1000 DCD:

A. In response to the AP600 RAI 410.241, a summary of ventilation flows and corresponding ambient pressure data were provided for the AP600 design for the VAS, VBS, VHS and VRS. Are these data directly applicable to the AP1000. Please provide such data for these systems for the AP1000.

B. In response to the AP600 RAI 410.245.c, VZS was defined as a defense-in-depth system and conformed to the staff's identified criteria for non-safety-related systems requiring regulatory controls, i.e., RTNSS systems. Verify that this information is equally applicable to the AP1000 VZS design (is this system similarly classified a RTNSS system as it was in the AP600?).

C. AP600 DCD Tier 2 Section 9.4.7.1.2 states that VFS provides filtration of exhaust air from the fuel handling area, auxiliary or annex building to maintain these areas at a "slight negative pressure" with respect to the adjacent areas. In response to the AP600 RAI 410.345F, "slight negative pressure" is defined as a "nominal set point value of (negative) 0.15-inch water gauge (WG) of the differential pressure)." Verify that this information is equally applicable to AP 1000 VFS design.

NOTE: AP600 RAI 410.241 410.245 were issued by the NRC staff on May 23, 1994 (NUDOCS Accession No. 9406230033). AP600 RAI 410.345 was issued by the NRC staff on December 9, 1997 (NUDOCS Accession No. 9801260106). Westinghouse provided its response to RAI 410.241 on July 15, 1994 (NUDOCS Accession No. 9407250275); to RAI 410.245 on May 7, 1997 (NUDOCS Accession No. 9705280194); and to RAI 410.345 on December 12, 1997 (NUDOCS Accession No. 9712230390).

Westinghouse Response:

A. The data supplied in response to AP600 RAI 410.241 were correct for the AP600 HVAC designs circa 1994. However, a number of revisions to those AP600 systems occurred during the licensing process time period and prior to AP600 receiving its final design approval (FDA). Those revisions were reflected in later revisions for the AP600 SSAR and DCD. Westinghouse is providing in response to this RAI an updated summary of ventilation flows and corresponding ambient pressure data applicable to the both the AP600 and AP1000 designs for the radiological controlled area ventilation system (VAS), the nuclear island nonradioactive ventilation system (VBS), the health physics and hot machine shop

HVAC system (VHS) and the and radwaste building HVAC system (VRS). Note that the VAS no longer includes a separate radiation chemistry laboratory subsystem. The radiation chemistry laboratory is now served by the auxiliary/annex building subsystem of VAS. (The AP600 DCD and the AP1000 DCD currently reflect this design.)

System	Nominal Outside Supply Airflow (cfm)	Nominal Exhaust Airflow (cfm)	Ambient Pressure
Fuel Handling Area - VAS	17,300	19,000	Negative
Auxiliary/Annex Buildings -VAS	33,000	36,000	Negative
MCR/TSC - VBS	1,350	650	Positive
Health Physics and Hot Machine Shop - VHS	12,750	14,000	Negative
Radwaste Building -VRS	16,200	18,000	Negative

B. The AP1000 diesel generator building heating and ventilation system (VZS) is identical to and classified the same as the AP600 VZS.

Thus, as noted in Westinghouse's response to AP600 RAI 410.245.c.:

- Item (5)
"As a defense-in-depth system, the diesel generator building ventilation system is classified as an AP600 Class D system. As discussed in SSAR subsection 3.2.2.6, this classification invokes industrial quality assurance and industry design standards."

- Item (7)
"The diesel generator building ventilation system is nonsafety-related and therefore not included in Technical Specification LCO and surveillance requirements. The diesel generator function is identified in Reference 410.245-1 as an RTNSS important function at reduced inventory conditions during shutdown operations. This reference also provides no short-term availability recommendations for the equipment used to support these functions."

  Note: "Reference 410.245 -1" is "WCAP-13856, AP600 Implementation of the Regulatory Treatment of Nonsafety-Related Systems Process. September 1993."

C. Like the AP600, the AP1000 containment air filtration system (VFS) provides filtration of exhaust air from the fuel handling area, auxiliary or annex building to maintain these areas at a slight negative pressure with respect to the adjacent areas. The "slight negative pressure" is defined as a nominal set point value of negative 0.15-inch water gauge (WG) of the differential pressure.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

RAI Number: 420.008

Question:

420.8 (DCD Figure 7.1-2)

Describe the "GATEWAY" design and its interface with the Protection and Safety Monitoring System.

Westinghouse Response:

The purpose of the Protection and Safety Monitoring Gateway is to interface the safety PMS to the non-safety real-time data network that supports the remainder of the instrumentation and control system. The Gateway has two subsystems. One is the safety subsystem that interfaces to the Plant Protection Subsystem, the Engineered Safety Features Coincidence Logic and the Qualified Data Processing Subsystem. The other is the non-safety subsystem that interfaces to the real-time data network. The two subsystems are connected by a fiber optic link that provides electrical isolation.

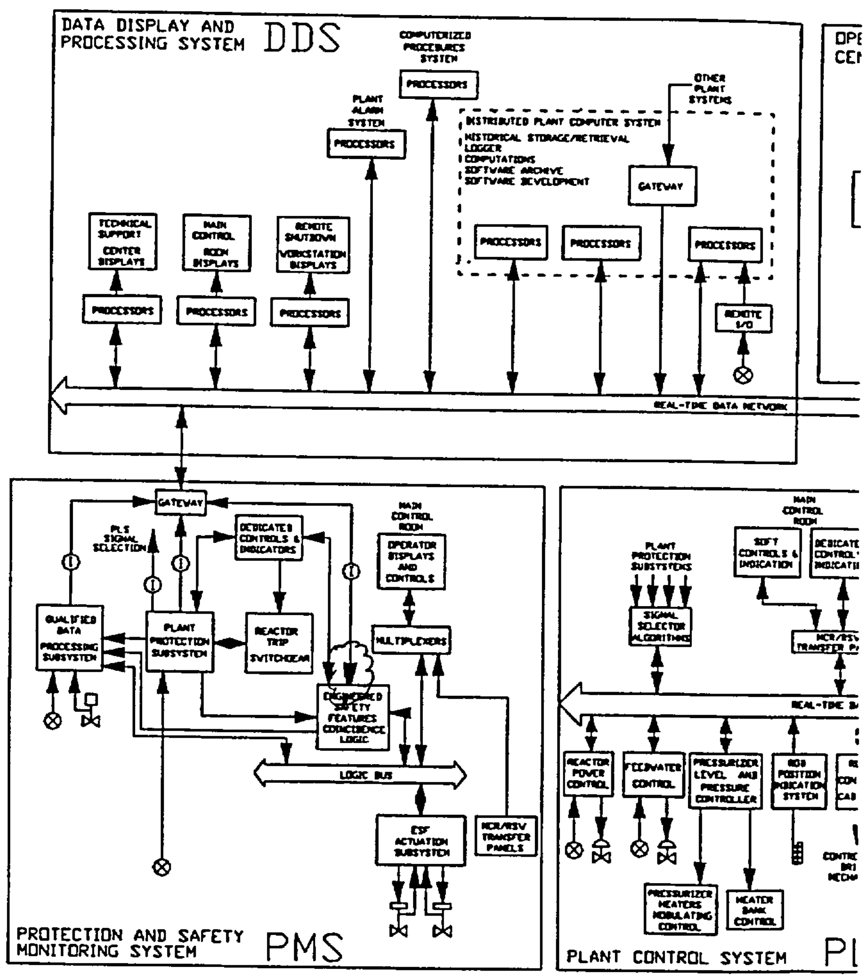
The primary flow of information between the two Gateway subsystems is from the safety subsystem to the non-safety subsystem. This information is a combination of plant process parameter values and equipment status information. The information that flows from the non-safety subsystem to the safety subsystem is limited to the following:

- The safety and non-safety subsystems exchange periodic interface signals that the communication controllers at each end of the link use to ensure that the link is functioning properly. These signals are used only by the communication controllers and are not propagated to the rest of the safety system. There is no application function in the safety system that uses this information.
- The main control room and the remote shutdown workstation operator consoles are non-safety. The soft control inputs to the PMS from these locations are provided from the non-safety subsystem to the safety subsystem of the Gateway.

Note that there is an error in DCD Figure 7.1-1. The gateway needs to communicate to the Engineered Safety Features Coincidence Logic (as well as from the ESF Coincidence Logic) to accomplish the second function listed above. DCD Figure 7.1-2 (Revision 1) shows this link correctly.

Design Control Document (DCD) Revision:

Revise DCD Figure 7.1-1 as shown.



PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number. 420.015

Question:

(DCD 7.1.2.12) Describe the design features of the graphic displays on the workstation and the wall panel. Discuss the interface between the workstation display, the wall panel display, and the qualified display processing system (QDPS).

Westinghouse Response:

The systems which present graphic displays on the workstation and the wall panel (i.e., the non-safety related alarms and displays) for the AP1000 are described in DCD Section 7.1.1. The workstation displays can be shown on any of the available Data Display and Processing System (DDS) video display units at the workstation consoles in the Main Control Room (MCR) or the remote shutdown room. As for the AP600, Chapter 10, Section 4.4 of the ALWR Utility Requirements Document contains requirements applicable to the workstation and the wall panel displays that will be met by the AP1000 display system. The DDS display design features include the following:

- A continuously available overview display on the MCR Wall Panel Information System which includes spatially dedicated alarms, parameter values, and status information, and is available on demand at any workstation.
- Functional displays that dynamically depict critical functions of the plant such as "reactivity control".
- Physical displays that dynamically depict the states of plant parameters and equipment, similar to piping & instrument diagrams or "mimics" but using live data.
- Computerized procedure displays that combine procedure text, dynamic plant data, and mechanisms for tracking and prompting.
- Point detail displays that dynamically depict sensor and component data.
- Alarm support displays that permit queries of the alarm system such as chronological listings, alarm trigger logic, group sorts, available messages, etc.
- Soft control displays that depict virtual control devices, along with the status of components and the values of parameters being controlled.

The workstation display devices, the wall panel display devices, and the nonsafety-related soft control display devices, are all part of the DDS, and are implemented on a common commercial product platform. The DDS displays employ a window-oriented graphical user interface that is native to this product platform. The operator uses a pointing device (e.g. mouse, trackball, or touchscreen) to select display elements for operations. The sensor inputs to the DDS are obtained either from the nonsafety-related Plant Control Systems (PLS) or the safety-related Protection and Safety Monitoring Systems (PMS). The current design has four DDS video display units for each operator position and four video display units for the SRO. The exact number of video display units in the control room, the exact number of display pages in the

DDS, the scheme for display navigation, and the details of specific operations will be determined as part of the human factors engineering program described in DCD Chapter 18.

The interface between the DDS and the PMS is provided by four separate, channelized gateway devices. The gateway devices are part of the safety-related PMS architecture, and provide qualified electrical isolation of the software-based safety and non-safety systems for display and control.

The display devices of the Qualified Display Processing System (QDPS) are on the dedicated safety panel in the main control room. The QPDS is the part of PMS that presents the minimum information necessary to achieve and maintain a safe shutdown condition, and to support post-accident monitoring required by Reg. Guide 1.97. The sensor inputs to the QDPS are obtained from either the Qualified Data Processing Subsystem, the Plant Protection Subsystem, or the ESF Actuation Subsystem (all of which are part of PMS). The QPDS display devices and associated sensors are supported by qualified DC power supplies that assure 72 hours of continuous emergency operation without site AC power. There is no interface between QPDS and other displays.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 420.016

Question:

(DCD 7.1.2.12) Describe the design features of the AP1000 alarm system such as alarm setpoint determination, alarm display, alarm message queues, and all the software and hardware to support the alarm systems. Discuss the ITAAC for the alarm management system. Describe how the operating procedures are implemented (computer-based procedures and alarm responses).

Westinghouse Response:

DCD subsection 7.1.1 describes the AP1000 I&C architecture. The advanced alarm system is provided by the nonsafety-related Data Display and Processing System (DDS). Typical alarm system inputs include values of process parameter and component status variables. These data values are received from process equipment via real-time data network (see DCD Figure 7.1-1). Computed data points may also be used. Nominally, the DDS accommodates up to 200,000 data points in real-time.

The DDS has an extensive alarm feature set that is native to its commercial product platform. Analog points can be programmed for multiple alarm setpoints (both fixed and incremental) in multiple modes. Digital points can be programmed to alarm for any state or state transition. Alarm acknowledgment may be configured as manual or automatic. Alarm behaviors can be tuned through software-based time delays, cutouts, deadband adjustments, and input filtering. Alarms can be grouped, prioritized, or logically related. User-defined alarms can be established on any monitored variable. Users can sort existing alarms by various categories or dimensions (systems, priorities, time, status, etc.).

Alarm setpoints are determined primarily by system designers. Alarm display characteristics (i.e., coding and organization) are determined primarily by Human System Interface (HSI) designers. These aspects of the AP1000 detailed design are not yet determined, but will be developed and proved acceptable through execution of the design process described in DCD Chapter 18, including its Combined License applicant items. The ITAAC assuring completion is DCD Tier 1 Section 3.2. (the human factors engineering ITAAC).

Alarm displays will be an integral element of the DDS workstation displays (see Response to RAI 420.015). This will include dedicated alarm display pages, distributed alarms (i.e., as associated with individual parameters shown on any given display page), and spatially dedicated alarms on the Wall Panel Information System.

The implementation of alarm response procedures is not dependent on alarm system design. Alarm response procedures will be provided in hardcopy form, which may be augmented by making them available on the computer-based procedure resource. Alarm response procedures are not utilized during emergency operations.

As with AP600, the Man-Machine Interface Systems requirements described in the EPRI ALWR URD, Volume II, Chapter 10, Section 4.3 (Alarms) will be used as design criteria for the AP1000 advanced alarm system. The acceptability of the detailed alarm system design will be demonstrated as part of the verification and validation program performed for the integrated control room HSI systems per the Human Factors Engineering Inspections, Tests, Analyses, and Acceptance Criteria (DCD Table 3.2-1).

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 420.017

Question:

420.17 (DCD 7.2.1.1.3)

Describe the reactor coolant hot leg and cold leg temperature measurement arrangement from the sensor to the plant protection system. Describe how the average coolant temperature (Tavg), Delta T, Overtemperature Delta T, and Overpower Delta T set point software is developed in the Common Q system.

Westinghouse Response:

Fast-response resistance-temperature detector (RTD) installations are provided in the hot leg and cold legs of each reactor coolant loop. These consist of detectors with rapid temperature response characteristics installed in special thin-wall-design thermowells. These devices generate input signals to the plant protection subsystem. There are six detectors in each hot leg and two detectors in each cold leg. The hot leg detectors are arranged in four groups of three fast-response RTDs that are spaced 120 degrees around the loop pipe. Each division in the protection and safety monitoring system uses one cold-leg detector and one group of three hot-leg detectors for the ΔT calculation. This arrangement produces a 2-out-of-4 logic on a plant-wide basis. The RTDs are terminated in the Level 1 plant protection subsystem (PPS) and subjected to signal conditioning and data acquisition functions, including engineering unit conversion.

The spatial dependency of the three hot leg RTDs is compensated for with manually-input biases. Given the three filtered hot leg RTD measurements, T_{hotj}, j = 1, 2, & 3.

Biased hot leg temperatures are calculated using:

$$\overline{T_{hotj}} = T_{hotj} - P_B S_j^{\circ} \qquad (1)$$

where:

$\overline{T_{hotj}}$ = biased hot leg temperature, °F
T_{hotj} = measured and filtered hot leg RTD temperature, °F
S_j° = manually input bias that corrects individual hot leg RTD measured value to the loop hot leg average, °F
P_B = correction factor for power level. See equation 3 for calculation of P_B. (Value of P_B from previous cycle is used.)

Loop average hot leg temperature is calculated using:

$$T_{hotavg} = \sum \overline{T_{hot_j}} \; / \; 3 \qquad (2)$$

The correction factor is calculated:

$$P_B = (T_{hotavg} - T_{cold}) \; / \; \Delta T^\circ \qquad (3)$$

where:

T_{cold} = cold leg temperature, °F
ΔT° = reference full power ΔT, °F

T_{avg} and ΔT are calculated as:

$$T_{avg} = (T_{hotavg} + T_{cold}) \; / \; 2 \qquad (4)$$

$$\Delta T = (T_{hotavg} - T_{cold}) \qquad (5)$$

The overtemperature ΔT setpoint and overpower ΔT setpoints are calculated using the conditioned input signals in the PPS using the process described in DCD subsection 7.2.1.1.3. The constants and time constants in the overtemperature ΔT and overpower ΔT setpoint equations are defined in Table 3.3.1-1 of DCD Chapter 16 (Technical Specifications).

The calculated setpoints are then compared in a bistable module to the actual RCS loop ΔT. If the actual RCS ΔT exceeds the calculated setpoint, the output of the corresponding bistable module becomes a logical 1. Otherwise, the bistable output remains a logical 0.

The bistable outputs are transmitted to the Level 2 RPS processor in which the output from the four divisions of overtemperature ΔT and overpower ΔT are combined in a two-out-of-four logic. If two or more of the divisions have a bistable output of logical 1, a trip signal is sent to the reactor trip switchgear.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

RAI Number: 420.018

Question:

420.18 (DCD 7.1.2.10)

Describe the inspection, tests, analyses, and associated acceptance criteria for the isolation devices to be used in the AP1000 I&C system design.

Westinghouse Response:

Both contact (digital) and analog electrical isolation devices may be used in the AP1000 I&C design. Type tests, analyses, or a combination of type tests and analyses will be performed to ensure the isolation devices meet the criteria of IEEE-384.

Modules are available for both contact and analog purposes that have designs compatible with the electrical isolation requirements of safety systems. These modules will be tested to demonstrate their isolation capability prior to such use in a safety application and the results of such testing will be documented. Any characteristics of the modules critical to their isolation function will be included in the procedures used to dedicate them for safety service.

Testing of the modules to demonstrate their ability to isolate electrical faults will be performed over a range of conditions (i.e., voltages) that bound the conditions that the modules might be called upon to isolate. Testing of the modules will ensure that faults are prevented from potentially damaging any equipment on the safety side of the module and that any damage to the module itself is limited so that it could not pose a threat to the integrity of the safety system that it is protecting. This testing is covered by DCD Tier 1 (ITAAC) 2.5.2, item 7.a.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number 420.020

Question:

420.20 (DCD Figures 7.1-2 and 7.1-10)

Describe the multiplexer configuration in the AP1000 design, where the multiplexer will be used in the protection and the control systems, how many are used, and how to maintain channel separation? Describe the inspection, tests, analyses, and associated acceptance criteria for the multiplexer devices to be used in the AP1000 I&C system design.

Westinghouse Response:

There are four main control room multiplexers, one for each safety division.

The main control room has historically used multiplexed communication of the very large amount of data that has to be communicated between the control room and the plant protection systems, thereby greatly reducing the amount of electrical cable required and associated congestion. There is little hardwired information shared between the main control room and the PMS in the AP1000 design so this data concentration function is not as important as it has been previously for earlier I&C system architectures. The principal purpose of the multiplexer in the AP1000 design is to facilitate the main control room to remote shutdown workstation transfer to be made in the event of a need for the operators to leave the main control room. The multiplexer, therefore, serves as the point of electrical and functional isolation of the limited number of hardwired control and indication devices connected between the main control room and the PMS cabinets. DCD Figures 7.1-1 and 7.1-2 will be revised as shown to clarify the multiplexer configuration.

The multiplexers use the same platform as the PMS to accept inputs from the consoles in the main control room. I/O modules interface with the discrete devices on the consoles with an isolated fiber optic output provided directly to the PMS logic bus.

The multiplexers are considered to be a functional part of the MCR Safety-related displays and MCR safety-related controls listed in DCD Tier 1 (ITAAC) Table 2.5.2-1. The multiplexers are thus required to meet the qualification requirements of ITAAC 2.5.2, items 2, 3, and 4 and the separation requirements of ITAAC 2.5.2, item 5a. The functionality of the multiplexers will be tested as part of ITAAC 2.5.2, items 8a, 8b, and 8c.

Design Control Document (DCD) Revision:

Revise Figures 7.1-1 and 7.1-2 as shown to reflect that the manual reactor trip switches are the only dedicated controls and indicators hardwired into the PMS.

Figure 7.1-1

7.1 Instrumentation and Controls

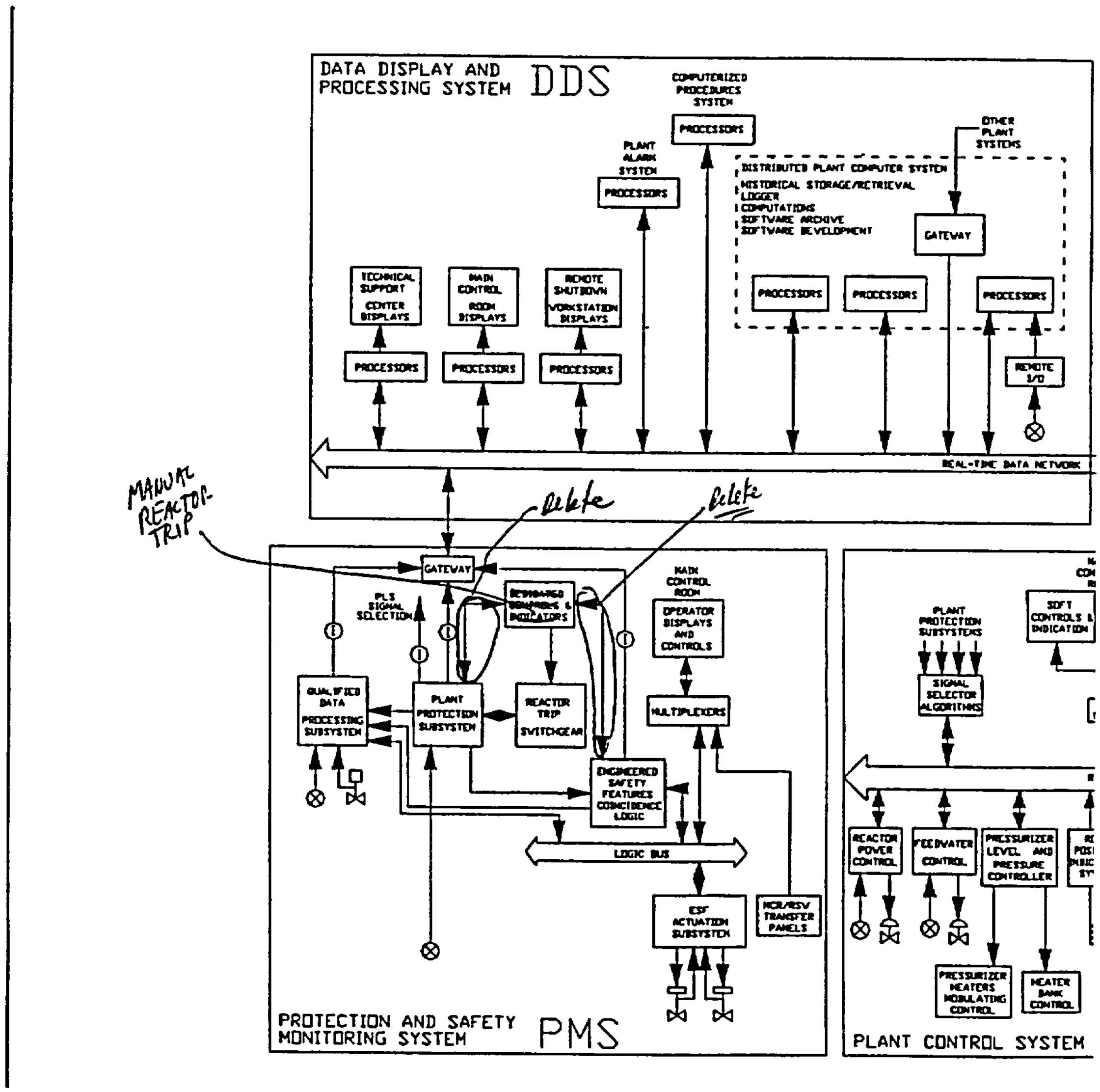


Figure 7.1-2

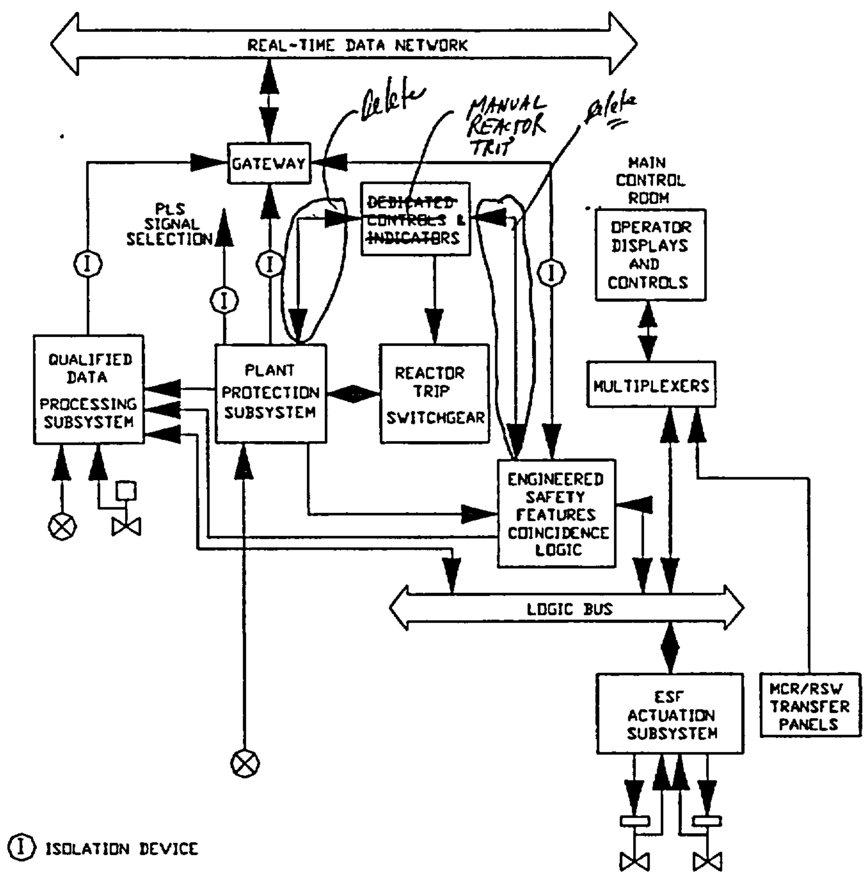


Figure 7.1-2

Protection and Safety Monitoring System

PRA Revision:

None

RAI Number: 420.022

Question:

420.22 (DCD 7.1.2.11)

The AP600 protection and safety monitoring system performs surveillance testing via a portable tester. Describe the provision provided for the AP1000 surveillance testing. Identify the tasks tested by a portable tester and the tasks tested by the built-in circuit in the protection cabinets.

Westinghouse Response:

Surveillance testing can be divided into three categories:

- On line Diagnostics Monitoring using AC 160 Platform features
- On line surveillance testing using built in test features
- Refueling Interval testing using an I/O simulator or similar device.

1. Automatic On-Line Diagnostics Monitoring

As defined in Section 6.4.1 of Reference 7.1.7.13, the Common Qualified Platform Topical Report, the AC 160 platform possesses an extensive set of on-line diagnostics. Automatic testing is an integral part of the Common Q system. It is used to monitor the integrity of the application as it performs its function. Diagnostics run continuously as background tasks during normal AC 160 operation. Any resulting errors are recorded in the processor log, and will provide appropriate alarms.

The status of failed modules is flagged to downstream components, and appropriate, conservative actions taken. The AC 160 platform is also equipped with both hardware and software watchdog timers. Fatal errors will result in a processor halt condition, and force a fail-safe watchdog timer timeout. In addition to processor tests, the AC 160 also verifies operation of the I/O modules, high-speed data-links and intra-channel AF100 bus communications

The AC 160 platform diagnostic tests are a background feature of the platform, and no separate internal or external-testing device is required.

2. Surveillance Testing

As described in Section 7.1.2.11, the Protection System function is tested by a series of overlapping tests, such that the entire protection system is verified, up to the final actuation device. This is under the control of the built in test features within the cabinet. Such testing includes four-channel comparison of input data and system setpoints. Other tests are periodically performed in a manually initiated automatic test sequence, in which test details are not under direct operator control. These tests include verification of the operability of the various trip and interlocking functions, including two out of four trip logic, up to, but not including, the final actuated device. Some test features, such as opening of the reactor trip breakers would be under manual control only. In all cases, the test system provides suitable prompting and appropriate diagnostic messages. The combination of on-line diagnostics and automatic surveillance testing and monitoring greatly reduce the need for manual initiated surveillance testing during normal operation.

3. Refueling Interval testing using an I/O simulator or similar device.

The built in test features of the PMS are adequate to monitor the system during normal operation. However, periodically, it may be necessary to inject external signals into the PMS to verify system performance. An example of this is refueling interval response time verification and system testing following major maintenance. An input/output simulator or similar device is used in such situations to provide simulated inputs and monitor appropriate outputs.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 420.031

Question:

420.31 (DCD 7.1.7, item 8)

DCD 7.1.7 Reference 8, CENPD-396-P, Rev. 01, "Common Qualified Platform," May 26, 2000, has 4 Appendices. Appendix 4 states that the purpose of this appendix (Appendix 4) is to describe the implementation of the Common Qualified Platform for an integrated configuration when digital upgrades are incorporated for multiple safety systems. However, the original intend of the Common Q design is for digital upgrade on an active operating plant, and not for a passive plant. In order for Reference 8 become more useful for a passive plant application, such as AP1000, please provide another appendix document specifically describe the implementation of the Common Qualified Platform for a passive plant with multiple safety systems.

Westinghouse Response:

Appendix 4 of the Common Qualified Platform Topical Report provides a conceptual design for a fully integrated Advant-based safety systems design. A high-level description is also included for a non-safety control system to provide an example for discussion of the interfaces between the non-safety control system and the safety system in order to address the concern regarding a postulated common-mode failure in the safety systems. The high-level description of components and interfaces provided in the main body of the topical report and Appendix 4 are applicable for the Instrumentation and Control Systems in the new AP1000 plant design. Therefore, no new appendix is necessary. Additional implementation details of the Common Q Platform for a passive plant with multiple safety systems are provided in Chapter 7 of the DCD.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 420.043

Question:

420.43 (DCD Tier 1, Section 2.5.2)

As mentioned in ITAAC No. 3 of Table 2.5.2-8, describe the report that exists or will exist and concludes or will conclude that class 1E equipment will be able to withstand SWC, EMI, RFI and ESD conditions.

Westinghouse Response:

The electromagnetic compatibility of Class 1E equipment is demonstrated by one of three techniques:

- Testing to determine the levels of electromagnetic disturbances through which reliable operation can be demonstrated,
- Analysis of the equipment to compare it to other equipment that has been previously tested, or
- Showing that the equipment is inherently immune to the required electromagnetic disturbances.

Electric Power Research Institute technical report TR-102323, Revision 1, "Guidelines for Electromagnetic Interference Testing in Power Plants" has been used for guidance regarding electrical disturbances, required levels, test methods and acceptance criteria.

DCD Tier 1 (ITAAC) item 3 of Table 2.5.2-8 refers to the protection and safety monitoring system (PMS) equipment listed in Table 2.5.2-1. The following is the current status of qualification of the listed equipment to SWC, EMI, RFI and ESD conditions.

- The PMS cabinets and MCR safety-related displays used in the AP600 (Westinghouse Eagle product) have been tested to generic levels for the listed conditions and found to successfully withstand them. Reports exist that document this testing, test results, and any installation or operating limitations required to support the qualification of the equipment. The alternative PMS equipment and safety-related displays, the Westinghouse Common Q product, have also been tested to generic levels for the listed conditions and found to successfully withstand them. The report describing the testing, test results and any installation or operating limitations required to support the qualification of the Common Q product was submitted to the NRC by Westinghouse letter LTR-NRC-02-41 dated August 14, 2002, Additional Information Regarding the Westinghouse Common Qualified Platform Application - August 2002, Sepp to Shukla. The electromagnetic environment of the AP1000 PMS equipment is expected to be within the generic levels tested.

- The reactor trip switchgear is inherently immune to the listed conditions and does not require special testing.
- The MCR/RSW transfer panels use equipment that is inherently immune to the listed conditions and do not require special testing.
- Some of the MCR safety-related controls may be inherently immune to the listed conditions and, therefore, not require special testing. Prior to installation or usage in the AP1000, the qualification of the MCR safety controls and indicators not inherently immune will be verified to conservatively envelope the limiting plant conditions. If required, the safety controls and indicators will be re-tested or plant design or operating conditions will be modified to ensure that their qualification does envelope the limiting plant conditions.

The type tests, analyses, or combination of type tests and analyses referenced in ITAAC 2.5.2, item 3, will verify that the qualification of the chosen I&C platform envelopes the limiting plant conditions. The testing, analyses or justification of inherent immunity to the listed electromagnetic disturbances will be documented in the report cited in the acceptance criteria for ITAAC 2.5.2, item 3.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

RAI Number. 440.035

Question:

In Tier 2 Information, Section 5.2.2.1 states that the sizing of the pressurizer safety valves (for overpressure protection of the RCS during power operation) is based on the analysis of a complete loss of steam flow to the turbine, with the reactor operating at 102 percent of rated power.

Provide the safety analysis to demonstrate that the relieving capacity and set pressure of the pressurizer safety valves specified in Table 5.4-17 are appropriate for overpressure protection with sufficient margin to account for uncertainties in the design and operation of the plant. The description should include the analysis methodology, assumptions, uncertainties in the design and operation of the plant, and the analysis results.

Westinghouse Response:

Analyses for complete loss of steam flow to the turbine is supplied in DCD Section 15.2.3. Specifically the following cases are performed to evaluate overpressure protection.

Case C1 - Turbine trip event with offsite power available
Case C2 - Turbine trip event without offsite power available

Sequence of events for these cases are provided in Table 15.2-1 (Sheet 3). Transient results are provided in Figures 15.2.3-15 through 15.2.3-20. These tables and figures are provided in Section 15.2.3 of the DCD.

In the case where offsite power is lost, the loss of offsite power is assumed to be a consequence of the turbine trip occurring while the plant is at power. The loss of offsite power is assumed to occur 3 seconds after the turbine trips.

The analyses of these events are performed using the LOFTRAN code. Mitigation of the event is provided by reactor trip, opening of the pressurizer safety valves and opening of the steam generator safety valves. A reactor trip occurs on high pressurizer pressure in the case with offsite power available. The loss of offsite power in Case C2 causes a loss of power to the reactor coolant pumps. A reactor trip occurs on low reactor coolant pump speed in the case where offsite power is lost.

These analyses are performed using the following assumptions

- The plant is assumed operating at full power. Initial plant parameters with the inclusion of measurement uncertainties are assumed. See Table 440.035-1 for specific values used.

- Main feedwater is lost at the initiation of the event

- No credit for the rapid power reduction control system is assumed

- No credit for the turbine bypass (steam dump) is assumed

- No credit for automatic rod control is assumed.

- No credit for pressurizer pressure control (pressurizer spray) to reduce RCS pressure is assumed.

- Minimum core reactivity feedback coefficients are assumed.

Case C1 with offsite power available results in a peak reactor coolant pressure of 2680 psia. Case C2, which assumes a loss of offsite power results in a peak pressure of 2694 psia. Reactor coolant pressure is maintained within 110% of design pressure (2748.5 psia). The analyses demonstrate that the sizing of the pressurizer safety valves is adequate for overpressure protection of the RCS during power operation.

Table 440.035-1 Initial Conditions Used for Peak RCS Pressure Analyses

Initial Condition	Value Used
Initial Power	102% of 3415 MWt (note that AP1000 calorimetric uncertainty is ≤ 1%; 2% uncertainty was assumed in this analysis)
Initial RCS Average Temperature	nominal value plus uncertainty 573.6 °F +6.5 °F = 580.1 °F
Initial Pressurizer Pressure	nominal value ± uncertainty (see Note 1 below)
Initial Vessel Total Flow	296000. gpm, Thermal Design Flow corresponding to 10% steam generator tube plugging

(1) Nominal pressurizer pressure is 2250 psia. An uncertainty of ± 50 psi is assumed. Cases that trip on high pressurizer pressure assume the pressure is initially at 2200 psia. Assuming the initial pressurizer pressure is low delays reaching the high pressurizer pressure trip. Cases that do not trip on high pressurizer pressure assume the initial pressurizer pressure is 2300 psia.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number. 440.053

Question:

Section 6.3.2.5.2 states that the passive core cooling system can sustain a single passive failure during the long-term phase and still retain an intact flow path to the core to supply sufficient flow to keep the core covered and to remove decay heat.

Describe your AP1000-specific analysis to confirm this conclusion.

Westinghouse Response:

In the long-term after a LOCA, the PXS will be operating in the recirculation mode. In this mode, water recirculates from the containment to the RCS and a steam water mixture is vented from the RCS back to the containment through the ADS. The containment water enters the PXS piping through recirculation screens and then passes through the PXS piping / valves to the direct vessel injection (DVI) nozzle on the reactor vessel (RV). The elevation of water in the containment is higher than the water head in the reactor vessel and provides the driving force for this flow.

The PXS recirculation piping has two different design pressures. The low pressure portion consists of all the recirculation pipe and the IRWST injection lines up to the normally open MOVs (V121A/B). This piping is constructed of schedule 40 stainless steel pipe. It is capable of design conditions of 800 psig at 300 F. The high pressure portion of this piping, from the IRWST MOV through the DVI line to the RV, is constructed of schedule 160 stainless steel pipe. It is capable of design conditions of 2485 psig at 680 F.

Both the low and high pressure portion of this piping system can withstand much higher pressures than they will see during post-accident conditions. Once ADS has been actuated, the pressure in these pipes will drop to less than 15 psig (relative to containment) during IRWST injection and to less than 5 psig during recirculation. As a result, the chance of having a leak will be extremely unlikely. In addition, if there is a leak in these piping systems, the leak rate will be very small.

The effects of passive failures in the PXS vary based on their location. The PXS piping can be divided into three different areas because of the different effects of a passive failure. These areas are:

- Locations above the post accident flood level
- Locations below the flood level in normally flooded areas
- Locations below the flood level in normally unflooded areas

Piping locations above the post-ADS flood level include the PRHR HX inlet lines and the CMT balance lines. If a passive failure were to occur in one of these lines the only effect would be to vent additional steam / water from the RCS which would help the ADS reduce the RCS pressure and improve core cooling.

Piping locations below the post-ADS flood level in normally flooded areas include portions of the DVI lines and the PRHR HX discharge line. If a passive failure were to occur in one of these lines it would be under water such that the differential pressure would be essentially zero. In this situation, there would be no impact on the operation of the PXS.

Piping locations below the post-ADS flood level in normally unflooded areas include portions of the DVI lines back to the accumulators, CMTs, and the IRWST. If a passive failure were to occur in one of these lines there would be a small differential pressure that would result in some leakage from the PXS line. The maximum differential pressure is 5 psi due to the water level in the containment. If the passive failure is assumed to be the complete rupture of a instrument line (1" sch 40) the leakage would be less than 7 lb/sec leak. This flow is small relative to the PXS recirculation flow rates of 170 lb/sec (DCD section 15.6.4C.2). Its effect on core cooling would be negligible considering that if such a passive failure were to occur then we would not have to assume an active failure and all four ADS stage 4 valves would be open (instead of three assumed in the DCD analysis in 15.6.4C.2).

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 440.054

Question:

Regarding Tier 2 Information, Chapter 15, "Accident Analysis," and Appendix 19E, "Shutdown Evaluation,"

A. Provide a list of the methodologies and computer codes used in the LOCA and non-LOCA transient analyses and Appendix 19E shutdown evaluation for the AP1000 design certification application, and reference the associated NRC acceptance letters to confirm the acceptance of the methodologies and codes used in the safety analyses.

B. Address the compliance with each of applicable limitations regarding the methodologies and codes and verify that the fuel performance, nuclear physics and thermal-hydraulics conditions of the analyses are within the applicable ranges of the approved computer codes.

Westinghouse Response:

A. The methodologies and analysis codes used in the AP1000 LOCA and non-LOCA transient analysis are presented in AP1000 DCD Section 15.0. Table 15.0-2 provides a listing of the analysis codes used for each transient and accident presented in Chapter 15. The references for each of these codes are also provided in DCD Section 15.0.16. The analysis presented in Appendix 19E is a LOFTRAN analysis of a loss of offsite power event that is performed with the same LOFTRAN code used for the transient analyses presented in Chapter 15 of the DCD. The purpose of the analysis presented in Appendix 19E is to demonstrate the ability of the AP1000 passive safety systems to bring the plant to safe shutdown temperature within 36 hours. This analysis is similar to the analysis performed for AP600 with LOFTRAN.

B. The compliance with the applicable limitations regarding the methodologies and codes that were developed and approved specifically for the AP600 Design Certification were addressed in the AP1000 pre-certification review. The analysis codes LOFTRAN, LOFTTR2, NOTRUMP and WCOBRA-TRAC were approved by the NRC in NUREG-1512, with limitations as noted. Westinghouse documented the applicability of these codes to the AP1000 and addressed the limitations of their use in WCAP-15644 "AP1000 Code Applicability Report." The NRC documented their review in the NRC letter "Applicability of AP600 Standard Plant Design Analysis Codes, Test Program and Exemptions to the AP1000 Standard Plant Design" dated March 25, 2002 (Reference 1), and found these codes acceptable for use for the AP1000, with limitations identified. The following summarizes the NRC conclusions regarding the applicability of the analysis codes and provides our response to the limitations that are identified in Reference 1.

LOFTRAN & LOFTTR2

Reference 1 identified the following limitations regarding the applicability of LOFTRAN & LOFTTR2 to AP1000

"Given the forgoing considerations, the NRC staff concludes that use of the LOFTRAN code as described in References 4, 5, and 6 is acceptable for licensing calculations of the AP1000 standard plant design, subject to the following condition and limitation:

- The transients and accidents that Westinghouse proposes to analyze with the LOFTRAN code are listed in Table 2 of this report, and the NRC staff's review of LOFTRAN usage by Westinghouse was limited to this set. Use of the code for other analytical purposes will require additional justification.

- The NRC staff requested that Westinghouse perform MSLB analyses for the AP1000 standard plant design. In particular, the staff wanted to assess the ability of the code to model the resulting steam formation in the reactor coolant loops. Westinghouse responded that an MSLB analysis to evaluate possible reactor system voiding will not be performed until the Phase 3 review of the AP1000 design. The NRC staff will therefore defer its review and approval of the LOFTRAN code for an MSLB analysis to the Phase 3 review."

Westinghouse Response:

Westinghouse has used the codes for only those events identified in Reference 1. Westinghouse has submitted the MSLB analysis in Section 15.1.5 of the DCD. The analysis results demonstrate that voiding in the reactor coolant loops is not a concern for this event.

NOTRUMP

Reference 1 identified the following limitations regarding the applicability of NOTRUMP to AP1000:

"The NRC staff is continuing its review of the NOTRUMP code for analysis of SBLOCA events for the AP1000 standard plant design. The discussions in the preceding sections state that the NRC staff has determined that many code features are acceptable for use in the AP1000 analysis. Nonetheless, final code approval will be contingent on the resolution of the following open issues:

- The ability of the NOTRUMP code to adequately predict liquid entrainment from the upper plenum and from stratified water in the hot legs into the ADS-4 is a concern to the staff since the amount of entrainment will affect the ability of the ADS-4 to depressurize the reactor and will affect the reactor vessel liquid inventory. Westinghouse proposes to address these issues as part of a WCOBRA-TRAC benchmark, which would be conducted as part of the Phase 3 review. Prediction of flow through the ADS-4 remains an open issue for the NOTRUMP code.

- Westinghouse has stated that the NOTRUMP PRHRHX model contains a deficiency that produces non-conservative results for high heat flows. High heat flows are identified by a velocity through the primary side of the PRHR tubes of greater than 1.5 ft/sec for any "significant period of time." Westinghouse proposes to reduce the PRHR heat transfer area in NOTRUMP by 50 percent during these periods. The flow velocity for the AP1000 design may exceed 1.5 ft/sec for much of the time during an SBLOCA event. The NRC staff therefore requires that Westinghouse define and justify what is considered to be a "significant period of time" to trigger a reduction in PRHR surface area and to justify that a 50-percent reduction of heat transfer area is conservative given comparisons with data appropriate for the AP1000 design.

- Westinghouse has not provided a complete small-break spectrum for the AP1000 standard plant design, but proposes to submit a complete break spectrum during Phase 3 of the review. Core uncovery may be predicted for certain small break sizes and locations. Westinghouse has not provided the NRC staff with justification for using either the NOTRUMP code or the SBLOCTA code, which is used to evaluate peak cladding temperatures for the AP1000 conditions when the core is uncovered. If Westinghouse calculates core uncovery during Phase 3 of the AP1000 review, the NRC staff will require that both NOTRUMP and SBLOCTA be qualified for the predicted conditions.

Westinghouse Response:

The following provides our responses to the three open items from the NRC staff's review of NOTRUMP documented in Reference 1:

- Entrainment - WCAP-15833 Revision 1 (Appendix A) provides our evaluations of the entrainment issue, as it pertains to AP1000, and its importance during the small break LOCA transients. Based on our evaluations and analyses presented in this report, Westinghouse concludes that the effects of upper plenum and hot leg entrainment are not significant. The conservatism in the NOTRUMP code and methodology is sufficient to acceptably perform analysis of small break LOCA events for AP1000, in accordance with the requirements of 10 CFR 50.46.

- PRHR HX Model – As stated in section 15.6 of the AP1000 DCD, the small break LOCA analysis performed for AP1000 that are presented in Chapter 15 of the DCD use the heat transfer penalty on PRHR heat transfer that was identified for the AP600, for cases when the velocity in the PRHR tubes is greater than 1.5 ft/sec. For AP1000, this penalty was applied for the entire transient, regardless of the velocity in the PRHR tubes. The following provides our justification for this penalty.

  The Thom correlation in NOTRUMP slightly overpredicts the heat transfer relative to the modified Rosenhow correlation that was developed from the AP600 PRHR test data by 6% to 8% depending on primary side inlet conditions. Reducing the heat transfer area by 50% and using the Thom correlation results in a reduction in the heat transfer relative to the modified Rosenhow correlation of 11% to 13% for the same conditions. The details of this evaluation are shown in Attachment A.

  Therefore, the penalty on heat transfer for the PRHR as applied to the AP1000 small break LOCA analysis is conservative.

- Core Uncovery– In the AP1000 small break LOCA analysis, core uncovery is not predicted to occur for the spectrum of small breaks performed. However, the 10-inch cold leg break exhibited the potential for core dryout for a brief period of time without the prediction of a traditional core uncovery (for example, core two-phase mixture level dropping into the active fuel region). To conservatively account for this potential core dryout period, a composite core mixture level was created as described in the 10-Inch DCD section. For this analysis, the LOCTA code was not used to calculate the maximum Peak Clad Temperature. Instead, to conservatively estimate the effects of this dryout period, an adiabatic heat-up calculation was performed to predict the maximum fuel cladding temperature. The results are presented in the DCD.

WCOBRA-TRAC

Reference 1 identified the following limitations regarding the applicability of WCOBRA-TRAC to AP1000:

"The NRC conducted a full-scope review of the WCOBRA/TRAC Code, which was previously approved as the LBLOCA code for the AP600 standard plant design and as a "best-estimate" code for an LOCA analysis for conventional Westinghouse plants (Reference 32). During the AP600 review, the NRC staff gained considerable understanding of WCOBRA/TRAC modeling and approximations and, thus, the staff agrees with Westinghouse's statement in WCAP-15613 (Reference 5) that scaling from the AP600 to the AP1000 is not required. The reason is that the passive cooling system does not participate in the LBLOCA portion of the transient that is covered by WCOBRA/TRAC. Following the LBLOCA blowdown, the core is reflooded and the fuel temperature increase is terminated by the accumulator injection. The AP1000 LBLOCA recovery is similar to that of the AP600 and, thus, there are no new phenomena and no need for additional data or AP1000 scaling. However, the limitations described in Section 21.6.3.17 of NUREG-1512 for application of WCOBRA/TRAC for the AP600 LBLOCA analysis should also apply for the AP1000."

Westinghouse Response

Westinghouse performed the AP1000 large break LOCA analysis in accordance with the limitations outlined in Reference 1 as described in DCD Chapter 15.

Ancillary Codes

The fuel codes, such as TWINKLE, FACTRAN and VIPRE are identified in section 15.0. The approved reference for these codes are referenced in section 15.0. These codes have been reviewed and approved by the NRC for Westinghouse fuel. The AP1000 fuel design is similar to the fuel designs that are in operation in Westinghouse plants, and is within the approved licensing basis of these codes.

Reference:

1. "Applicability of AP600 Standard Plant Design Analysis Codes, Test Program and Exemptions to the AP1000 Standard Plant Design" dated March 25, 2002

Design Control Document (DCD) Revision: None

PRA Revision: None

ATTACHMENT A

Evaluation of the PRHR Heat Exchanger Tube External Heat Transfer

The PRHR heat exchanger is a C-tube design with 689 tubes. The heat exchanger is located in the in-containment refueling water storage tank (IRWST), and serves as the safety-grade decay heat removal mechanism for design basis accidents. The heat exchanger is normally isolated from the reactor coolant system. In the event of an "S" signal, the isolation valves are opened and RCS water enters the heat exchanger from the hot leg. Cold water is returned to the cold leg at the reactor coolant pump suction. Natural circulation flow is generated in the heat exchanger by the density difference between the hot inlet flow and the cold outlet flow and the separation between the thermal center of the heat exchanger and the core.

PRHR Heat Exchanger Tube Heat Transfer Model

At any point along the length of the PRHR tube, the resistance to heat transfer from the fluid inside the tubes to the IRWST water outside the tubes is comprised of three components; the film drop inside the tubes, the thermal conductivity of the tube wall and the film drop outside the tubes.

$$q = (Tin - T\infty) / (R1 + R2 + R3) \qquad (1)$$

where Tin is the temperature of the fluid inside the tubes
$T\infty$ is the local bulk temperature in the pool outside the tubes
and R1, R2, and R3 are the three resistances described above

$$R1 = 1 / (h_{in} * \pi * Di * \Delta L) \qquad (2)$$

$$R2 = \ln (Do / Di) / (2\pi * ktube * \Delta L) \qquad (3)$$

$$R3 = 1 / (h_{out} * \pi * Do * \Delta L) \qquad (4)$$

where h_{in} is the heat transfer coefficient inside the tubes
h_{out} is the heat transfer coefficient outside the tubes
Do is the outside tube diameter
Di is the inside tube diameter
k_{tube} is the thermal conductivity of the tube wall
and ΔL is the differential length of the tube segment

Heat Transfer Inside the PRHR Tubes

The inlet flow to the PRHR heat exchanger can be either single-phase liquid or two-phase mixture. For single-phase liquid inside the tubes, the heat transfer coeffieicnt is described by the Dittus-Boelter correlation

$$h_{db} = 0.023 * Re^{0.8} * Pr^{0.4} \tag{5}$$

where Pr is the Prandtl number of the fluid inside the tube
and Re is the Reynolds number given by

$$Re = 4 * m / (\pi * Di * \mu) \tag{6}$$

where m is the flow rate of the fluid in the tube
and μ is the dynamic viscosity of the fluid in the tube

For two-phase mixture, the Shah condensation model is used (Ref. 3).

$$h_{shah} = h_{db} * (1 - x)^{0.8} + 3.8 * x^{0.04} / (p / 3208)^{0.38} \tag{7}$$

where p is the saturation pressure inside the tube
and x is the flow quality

Thus,

$$h_{in} = \begin{cases} h_{db} & \text{for } x = 0 \\ h_{shah} & \text{for } x > 0 \end{cases} \tag{8}$$

Heat Transfer Outside the PRHR Tubes

An extensive test program was conducted to provide heat transfer characteristics for the PRHR heat exchanger (Ref. 1). The results of these tests showed that the heat transfer from the outside of the tubes is characterized by either free convection or nucleate boiling depending on the outer wall temperature of the tubes and the local pool conditions. Free convection is described by McAdams' correlation

$$h_c = 0.13 * k / L * [Gr * Pr]^{1/3} \tag{9}$$

where k is the water thermal conductivity
L is the characteristic dimension
Pr is the Prandtl number of the fluid outside the tube
and Gr is the Grashof number which is given by

$$Gr = g * \beta * (Tw - T\infty) * L^3 / \nu^2 \tag{10}$$

where g is the gravitational constant
β is the liquid volumetric expansion coefficient
T_w is the outer tube wall temperature
T_∞ is the bulk temperature in the pool
and ν is the liquid kinematic viscosity

Combining equations 9 and 10,

$$h_c = 0.13 * k * [g * \beta * (Tw - T\infty) * Pr / \nu^2]^{1/3} \quad (11)$$

For the case where the tube outer wall temperature is greater than the local saturation temperature in the pool, the water will boil. Reference 1 showed that the boiling heat transfer was degraded somewhat at the top of the tube bundle as steam generated further down blanketed the upper portions of the tubes. A correlation was generated from the test data based on the Rosenhow correlation and used in LOFTRAN (Ref. 2)

$$q/A = \mu_f * h_{fg} *[g*(\rho_f - \rho_g)/(g_c*\sigma)]^{0.5} * [c_p * \Delta T / (C_{sf}*Pr*h_{fg}]^{1/0.4523} \quad (12)$$

where q/A is the heat flux
μ_f is the liquid dynamic viscosity
h_{fg} is the heat of vaporization
g is the acceleration due to gravity
g_c is the gravitational constant
ρ_f is the liquid density
ρ_g is the vapor density
c_p is the liquid specific heat
σ is the liquid surface tension
Pr is the liquid Prandtl number
C_{sf} is a constant derived from the test data = 0.0413
and ΔT is the temperature difference Tw - Tsat

Equation 12 can be written as

$$q/A = h_{nb\text{-}loft} * (Tw - Tsat) \quad (13)$$

where $h_{nb\text{-}loft}$ is the nucleate boiling heat transfer coefficient used in LOFTRAN.

$$h_{nb\text{-}loft} = a * (Tw - Tsat)^b \quad (14)$$

where the constant a is dependent on the local pool conditions

$$a = \mu_f * h_{fg} *[g*(\rho_f - \rho_g)/(g_c*\sigma)]^{0.5} * [c_p / (C_{sf}*Pr*h_{fg}]^{1/0.4523} \qquad (15)$$

and the constant b is given by

$$b = 1 / 0.4523 - 1 = 1.2109 \qquad (16)$$

The NOTRUMP code uses a global nucleate boiling model for all heat transfer surfaces and does not allow differentiation between the PRHR tubes and other surfaces such as the fuel rods. The code uses the Thom correlation (Ref. 3)

$$h_{nb\text{-}not} = (0.072)^{-2} * e^{(P/630)} * (Twall - Tsat) \qquad (17)$$

where P is the local pressure in the pool

Either equation 14 or 17 can be used to calculate the nucleate boiling heat transfer coefficient if the tube outer wall temperature is greater than the local saturation temperature. The nucleate boiling coefficient is compared to the natural circulation coefficient from equation 11 and the maximum is used.

$$h_{out} = MAX (h_c , h_{nb}) \qquad (18)$$

This is the value used in Equation 4 to calculate the resistance to heat transfer outside the tube.

Overall Heat Transfer in the Tubes

After the overall heat transfer is calculated for a tube segment using Equation 1, the outside wall temperature is calculated by

$$Tw = Tin - q * (R1 + R2) \qquad (19)$$

The process is repeated until the heat transfer, q, converges, and the solution for the tube segment has been determined depending on whether the flow is single-phase or two-phase. For single-phase flow, the temperature of the fluid inside the tube exiting this segment is lower due to this heat transfer.

$$Tin_{i+1} = Tin_i - q / (m * c_p) \qquad (20)$$

where Tin_{i+1} is the fluid temperature for the next segment
Tin_i is the fluid temperature of the previous segment
and c_p is the specific heat of the fluid inside the tube

For two-phase flow, the enthalpy change is given by

$$h_i+1 = h_i - q / m \quad (21)$$

where h_{i+1} is the fluid enthalpy for the next segment
and h_i is the fluid enthalpy for the previous segment

The quality for the next segment is given by

$$x_{i+1} = (h_{i+1} - h_f) / (h_g - h_f) \quad (22)$$

where h_f is the saturated liquid enthalpy at the pressure inside the tube
and h_g is the saturated vapor enthalpy at the pressure inside the tube

If the enthalpy for the next segment is less than or equal to the saturated liquid enthalpy, the flow is assumed to be single-phase liquid.

The process is repeated for all segments of the tube.

The overall heat transfer from the PRHR is calculated by summing the individual segments over all of the tubes

$$Q_{tot} = [\Sigma q_i] * N_{tubes} \quad (23)$$

where N_{tube} is the total number of tubes in the heat exchanger

Determining the Effect of the Nucleate Boiling Correlation – Single Phase

Several calculations were made to determine the effect of the nucleate boiling correlation. A typical PRHR flow rate of 500,000 lbm/hr is assumed along with an inlet temperature of 300F. The inlet flow is assumed to be single-phase liquid. This corresponds to 0.2 lbm/s per tube and 1.52 ft/s velocity.

The heat transfer calculation described in the previous section was performed using both the NOTRUMP and LOFTRAN nucleate boiling correlations. In both cases, the top of the tubes experience boiling and transition to natural convection as the fluid temperature inside the tubes decreases and the pool water pressure increases along the vertical portion of the tubes. The heat transfer coefficient as a function of length along the tubes is shown in Figure 1. This figure shows that the Thom correlation predicts significantly higher film coefficients for nucleate boiling than the modified Rosenhow correlation.

Figure 2 shows the local heat transfer rate as a function of length along the tubes. This plot shows that although the film coefficient in the boiling region is higher, the wall temperature is lower and the heat transfer rates are only moderately higher. In addition, the higher heat removal in the beginning of the tubes results in lower fluid temperatures inside the tubes in the lower region as is shown in Figure 3. Thus, more heat is removed in the lower region for the case where the modified Rosenhow correlation is used. The overall heat removal for the heat exchanger was 11.9 MW for the Thom case and 11.2 MW for the modified Rosenhow case. Thus, the current NOTRUMP model overpredicts the PRHR heat transfer by about 6% for these typical conditions.

Reference 3 recommends a reduction in the PRHR heat transfer area of 50% when the fluid velocity inside the tubes exceeds 1.5 ft/s. A separate calculation was performed to determine the effect of using the Thom correlation with a 50% reduction in the heat transfer area. The resulting heat removal for the heat exchanger is 9.8 MW, which is a reduction of about 13% from the modified Rosenhow case. Thus, it is conservative to reduce the PRHR heat exchanger area by 50% to account for the use of the Thom correlation.

Determining the Effect of the Nucleate Boiling Correlation – Two Phase

For the case of two-phase mixture entering the PRHR heat exchanger, the heat transfer is higher in the condensing region. For this case, the same conditions are assumed; 500,000 lb/hr inlet flow at 300F with a IRWST temperature of 212F. However, for this case, the inlet flow is assumed to be two-phase with a flow quality of 0.05. As before, two cases are analyzed; one using the Thom correlation for boiling on the outside of the tubes, and one using the modified Rosehow correlation.

Figure 4 shows that the heat transfer coefficient is higher for a larger portion of the tube length using the Thom correlation. Figure 5 shows that there is significantly higher heat transfer rates when the tubes are condensing two-phase mixture for the case using the Thom correlation. However, the vapor is condensed within a shorter tube length for this case, and in the natural convection region the higher fluid temperature inside the tubes results in higher heat transfer for the case where the modified Rosenhow correlation is used. This result is also shown in Figure 6 where the fluid temperature remains at the inlet temperature until the vapor is condensed, then falls more rapidly using the Thom correlation.

Using the Thom correlation, the overall heat removal was 17.3 MW, as compared with 16.3 MW using the modified Rosenhow correlation (~6% increase). An additional run was made using the Thom correlation and reducing the tube heat transfer area by 50%. The overall heat removal for this case is 14.5 MW which is approximately 11% lower than the modified Rosenhow correlation. Thus, for two-phase flow into the PRHR heat exchanger, the Thom correlation with a 50% decrease in the PRHR heat transfer area conservatively underpredicts the PRHR heat transfer when compared to the modified Rosenhow correlation.

Conclusions

The results of this study show that the use of NOTRUMP with the Thom nucleate boiling correlation slightly overpredicts the heat removal by the PRHR heat exchanger for both single-phase and two-phase inlet flow. By reducing the heat transfer area by 50%, the heat removal rate is conservatively underpredicted by the correlations in NOTRUMP by 11 - 13 % when compared to the modified Rosenhow correlation used in LOFTRAN.

References

1. WCAP-12980, Rev 3, AP600 PRHR Heat Exchanger Final Test Report, April 1997

2. WCAP-14234, Rev 1, LOFTRAN and LOFTTR2 AP600 Code Applicability Document, August 1997.

3. WCAP-14807, Rev. 2, NOTRUMP Final Validation Report for AP600, August 1998.

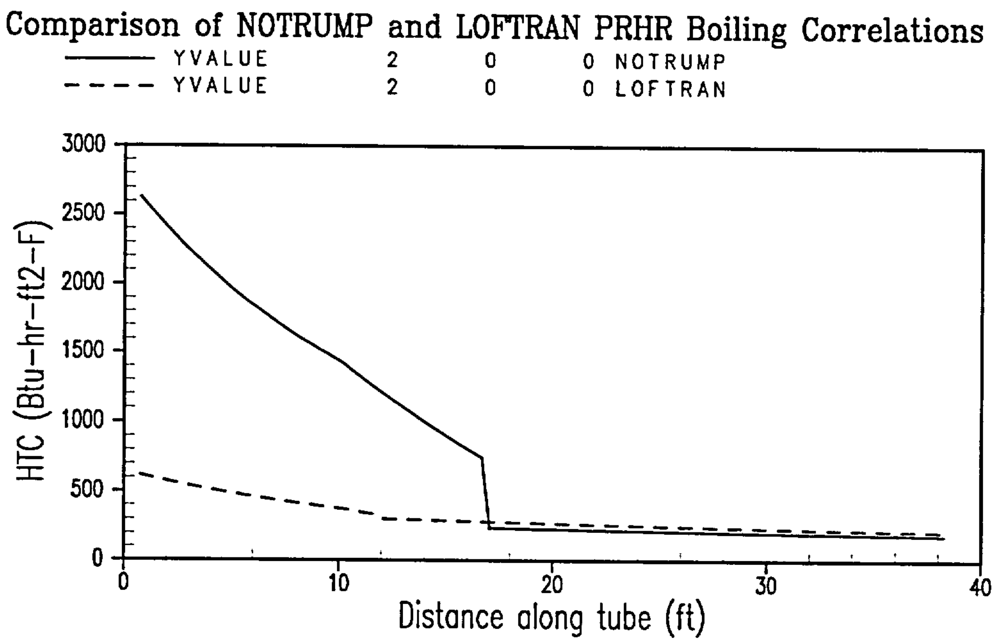


Figure 1: Heat Transfer Coefficients vs. Distance Along Tube - Single Phase

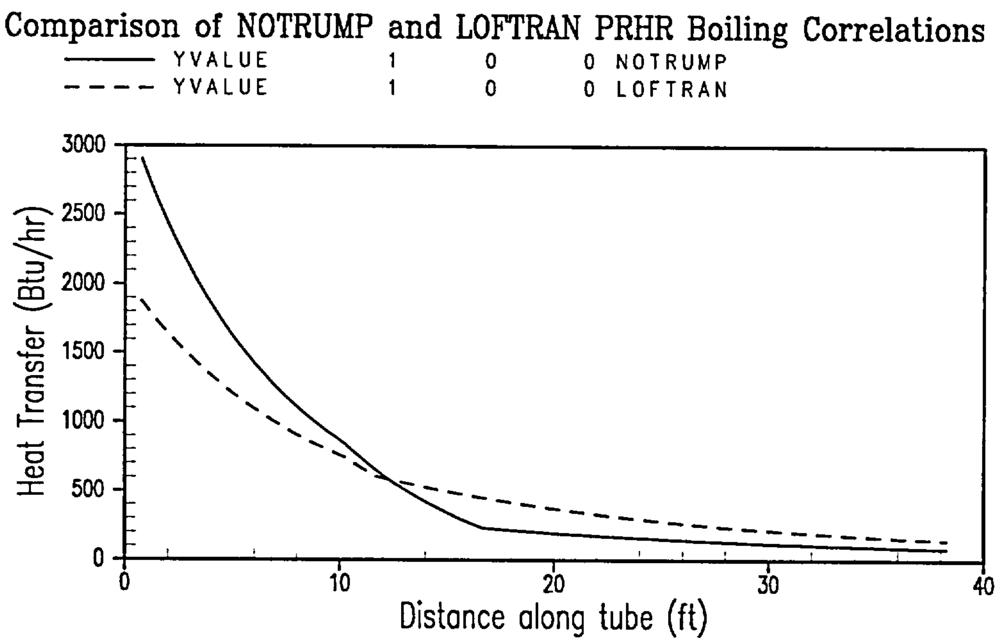


Figure 2: Heat Transfer Rate vs. Distance Along Tube - Single Phase

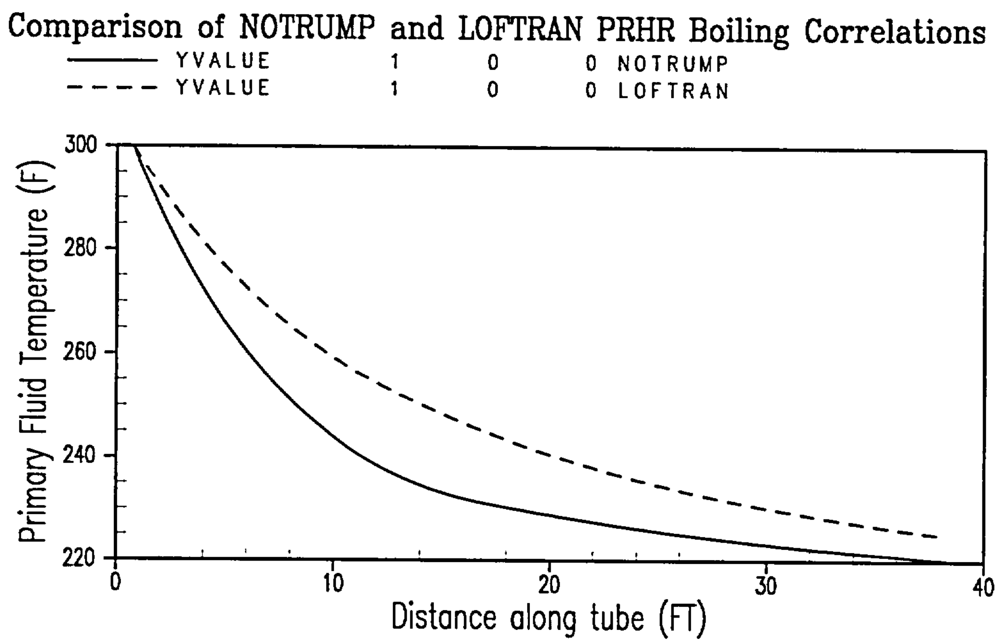


Figure 3: Primary Fluid Temperature vs. Distance Along Tube – Single Phase

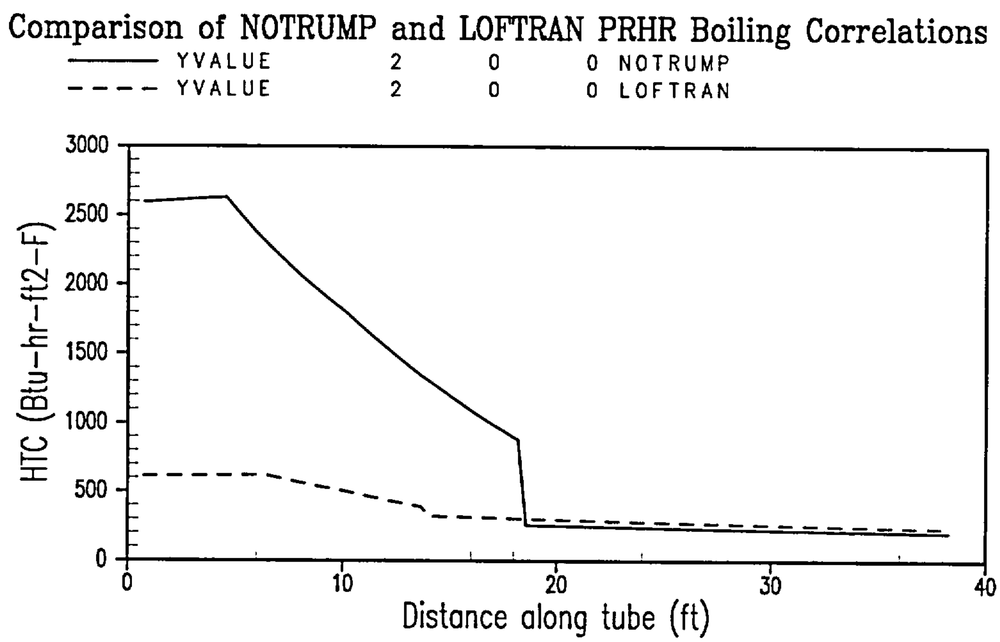


Figure 4: Heat Transfer Coefficients vs. Distance Along Tube – Two Phase

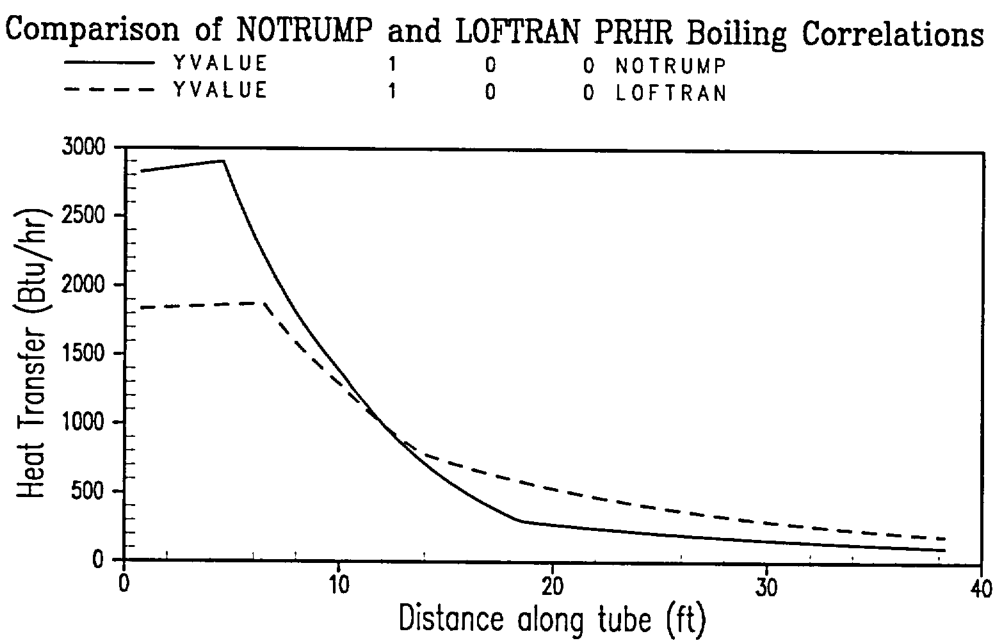


Figure 5: Heat Transfer Rate vs. Distance Along Tube - Two Phase

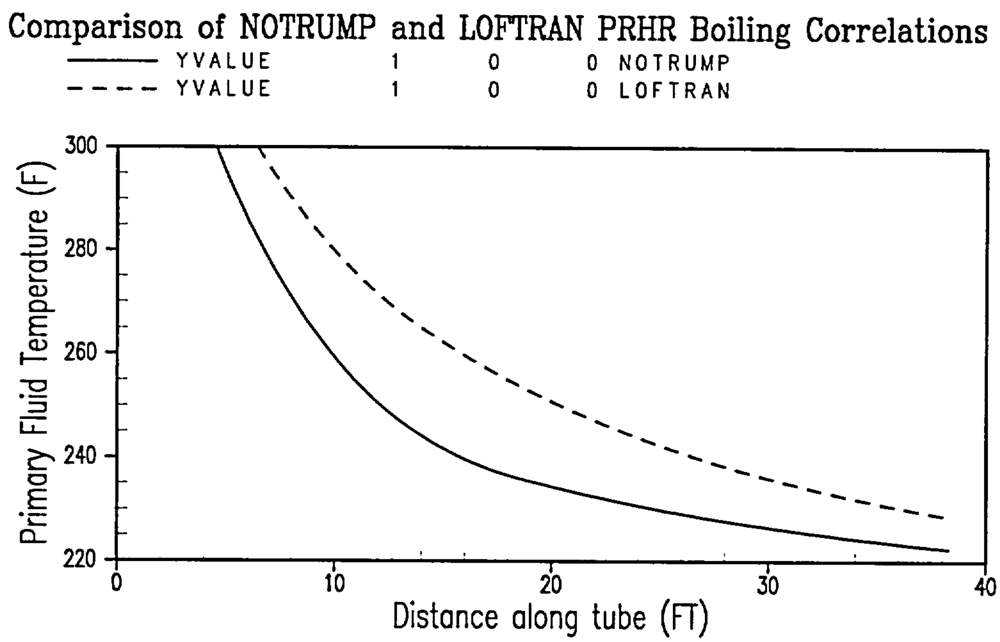


Figure 6: Primary Fluid Temperature vs. Distance Along Tube – Two Phase

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 440.063

Question:

Section 15.1.2 presents the results of an analysis for the increased feedwater flow event. No figure is presented to show that the calculated DNBRs do not exceed the specific acceptable fuel design limits during the transient.

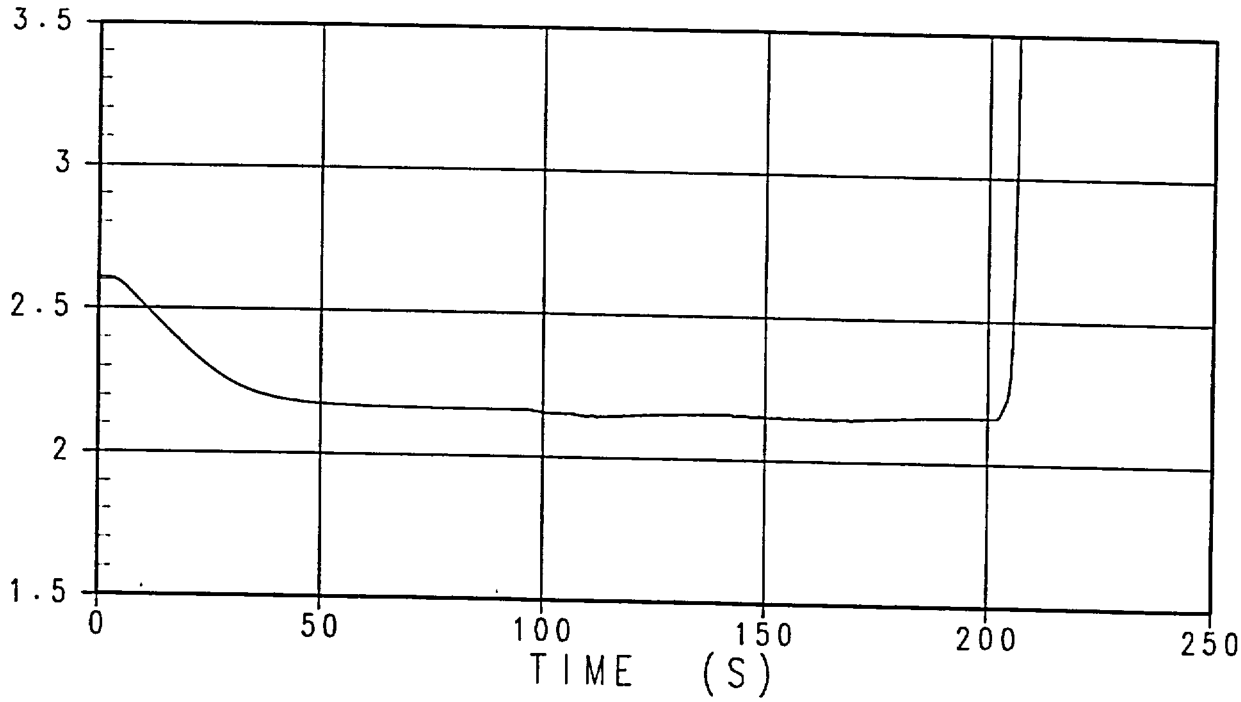
A. Provide the DNBR figure for the staff to review.

B. No information is presented to address the SG overfill issue. Specifically, for the case initiated from a full opening of a feedwater isolation valves without the isolation valve reclosure because of a single failure consideration, the applicant is requested to identify the safety related equipment that are credible to isolate the feedwater in order to prevent SG overfill.

C. If non-safety related systems or components (such as the feedwater control valves or feedwater pumps) are credited to isolate or terminate the feedwater, the applicant should show that the non-safety related system or component is reliable for feedwater isolation and provide a TS LCO to meet the requirements specified in (c)2(ii)(C) of 10 CFR 50.36.

Westinghouse Response:

A. The attached figure provides the DNBR transient for the limiting feedwater malfunction event reported the DCD. Note that the minimum DNBR value and time reported in the DCD (sub-section 15.1.2.2.2 and Table 15.1.2-1) have been revised. See attached DCD changes.

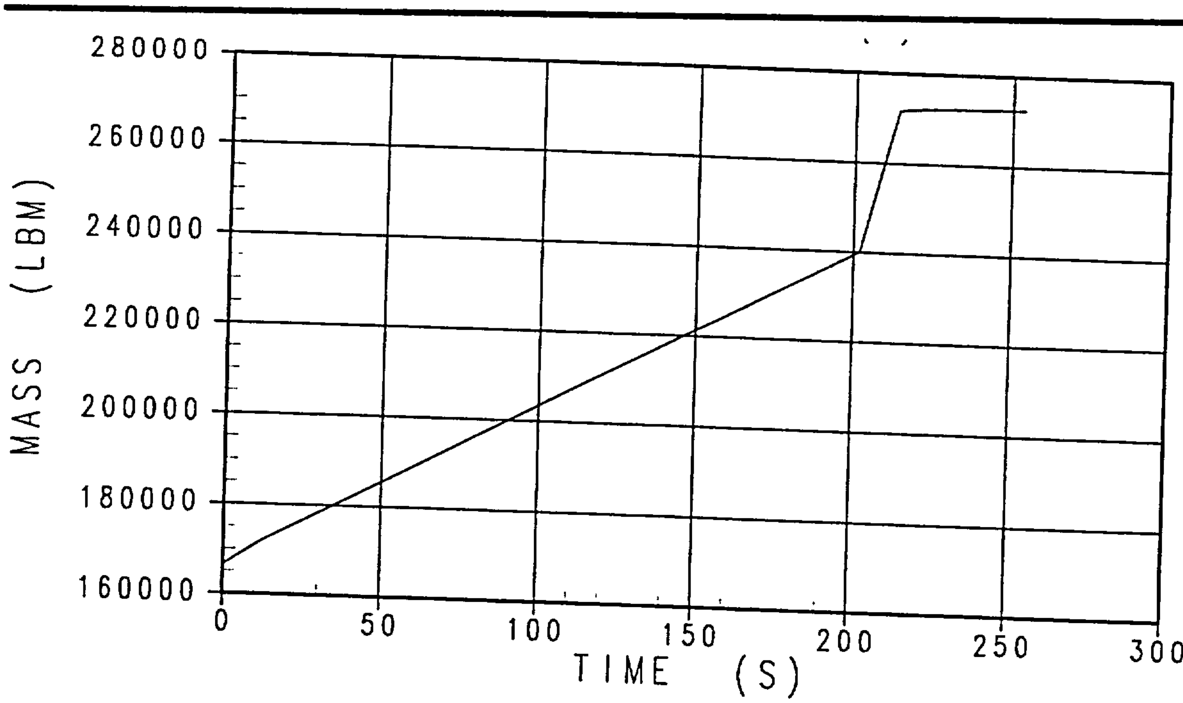


Feedwater Malfunction Event – DCD Case – DNBR Transient

B. SG Overfill Consideration

The following figure reports the behavior of the affected steam generator water mass as a function of time. It should be noted that, at the time of the turbine trip, the rate of increase in secondary inventory increases due to the mismatch between steam flow and the assumed sustained feedwater flow. Feedwater is conservatively assumed to be isolated 12 seconds after the High-2 steam generator level signal (The safety analysis setpoint has been set to 100% of the narrow range (NR) span, corresponding to an inventory of about 240,000 lbm at full power). This time bounds the time it takes to terminate MFW by tripping the MFW pumps; isolating MFW by closing the MFW isolation valves or the MFW control valves whichever is faster.

At this time there is a large margin to overfilling. Moreover, following the turbine trip and the immediate void collapse, the water level in the steam generator drum immediately drops providing an additional margin to overfill. It should be noted, in fact, that the normal level steam generator water mass, calculated at zero load condition, is evaluated, including uncertainties, in the range between 256,000 lbm and 300,000 lbm. In other terms, after reactor trip and turbine trip, the level in the steam generators is expected to drop in the range of the nominal water level.



Feedwater Malfunction Event – Affected Steam Generator secondary inventory (lbm)

C. Feedwater Line Isolation and Single Failure Considerations

The safety related portion of the feedwater system from the steam generator inlets outwards through the containment up to and including the main feedwater control (MFCV) and isolation (MFIS) valves is constructed according ASME Code, Section III for class 2 or 3 components and is designed to Seismic Category 1 requirements (DCD Section 10.4.7).

For events other than a malfunction of a MFW control valve, single failure tolerant main feedwater isolation is provided via the MFCV and the MFIV both of which are safety-related, active valves (DCD Section 3.2-70). Both valves are designed to close automatically on main feedwater isolation signals from the safety-related plant protection I&C system, within the time established within the technical specification.

For a feedwater malfunction event resulting in an increase of feedwater flow to one or more steam generators, overfilling protection is provided by the following protective safety actions performed by the protection control system following a High-2 Steam generator water level (two out of four logic) in any steam generator:

- Close all main feedwater control valves
- Close all main feedwater isolation valves

- Trip the main feedwater pumps
- Close all the feedwater bypass valves
- Close the startup feedwater control and isolation valves and trip the startup feedwater pumps

All of the above signals are safety related signals. The logic for the signals generation is a two out of four logic that is single fault tolerant.

If the initial fault is a failure in the MFW I&C control signal, it could cause one or both MFCV to open excessively. The safety-related MFW isolation signal discussed above will cause the MFCV to close even in the presence of such a malfunction. The control signal acts through the valve positioner and the safety signal acts through a solenoid that overrides / blocks the control signal.

If the initial fault is a failure of one MFCV valve, it could cause that one MFCV to open excessively. The following is a listing of failures that could affect the MFCV. For each failure an assessment is made as to whether the failure could cause the valve to open excessively and whether the safety signal could override the excessive opening.

Failure	Effect on MFCV	Can Safety Signal Override Opening
Loss air pressure	fails close	--
Rupture of air diaphragm	fails close	--
Failure of control signal	any position	yes
Failure electro-pneumatic transducer	any position	yes
Actuator spring weakening / breaking	could open	no
Positioner linkage bent / disconnected	could open	yes
Valve internal binding	fails as is	no
Clogged pressure regulator / filter	fails close	--
Valve packing leakage	no effect	--

From this evaluation, it can be seen that most valve failures would either result in the MFCV closing or if it does open, the safety signal would be able to override / block the opening and cause the valve to close. There is one failure that could cause the valve to open that the safety signal could not override / block which is the breaking of the actuator spring. Note that a failure of valve internal binding is not considered likely to cause an excessive feedwater accident.

As a result, it is considered very unlikely that a failure that causes one or both of the MFCV to open excessively could not be overridden or blocked by the safety signal.