Westinghouse

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

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

Direct tel. 412-374-5355
Direct fax. 412-374-5456
e-mail corletmm@westinghouse.com

Your ref. Docket No. 52-006
Our ref: DCP/NRC1566

April 7, 2003

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

This letter transmits the revised 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

Docket No. 52-006
DCP/NRC1566

April 7, 2003

Correspondence with respect to the application for withholding should reference AW-03-1621, 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 if you have questions regarding this submittal.

Very truly yours,

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-03-1621

/Attachments

1. Table 1, "List of Westinghouse's Responses to RAIs Transmitted in DCP/NRC1566"

2. Westinghouse Revised Proprietary Response to US Nuclear Regulatory Commission Requests for Additional Information dated March 2003

3. Westinghouse Revised Non-Proprietary Responses to US Nuclear Regulatory Commission Requests for Additional Information dated March 2003

			RAIs
bcc:	C. B. Brinkman	- Westinghouse, Rockville, MD	1
	M. M. Corletti	- Westinghouse, Pittsburgh, PA, EC E3-08	2
	W. E. Cummins*	- Westinghouse, Pittsburgh, PA, EC E3	
	Document Control	- US NRC	1 (original)
	J. Segala	- US NRC, Rockville, MD	10
	H. A. Sepp	- Westinghouse, Pittsburgh, PA, EC E4-07A	1
	R. P. Vijuk*	- Westinghouse, Pittsburgh, PA, EC E3-06	
	J. W. Winters*	- Westinghouse, Pittsburgh, PA, EC E3-08	

* Letter only

DCP/NRC1566

April 7, 2003

Enclosure 1

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

Westinghouse

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

April 7, 2003

AW-03-1621

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

ATTENTION: Mr. John Segala

APPLICATION FOR WITHHOLDING PROPRIETARY
INFORMATION FROM PUBLIC DISCLOSURE

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

Dear Mr. Segala:

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-03-1621 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-03-1621 and should be addressed to the undersigned.

Very truly yours,

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

/Enclosures

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COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF ALLEGHENY:

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

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

Sworn to and subscribed
before me this 7th day
of April, 2003

Notary Public

Notarial Seal
Lorraine M. Piplica, Notary Public
Monroeville Boro, Allegheny County
My Commission Expires Dec. 14, 2003
Member, Pennsylvania Association of Notaries

(1) I am Manager, Passive Plant Projects & Development, in the Nuclear Power Plants 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 2 as Proprietary Class 2 in the Westinghouse document DCP/NRC1566 for submittal to the Commission: (1) "AP1000 Design Certification Review – Revised 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-03-1621) and to the Document Control Desk, Attention: John Segala, 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.

DCP/NRC1566

Docket 52-006
April 7, 2003

Copyright Notice

The documents transmitted herewith each bear a Westinghouse copyright notice. The NRC is permitted to make the number of copies of the information contained in these reports which are necessary for its internal use in connection with generic and plant-specific reviews and approvals as well as the issuance, denial, amendment, transfer, renewal, modification, suspension, revocation, or violation of a license, permit, order, or regulation subject to the requirements of 10 CFR 2.790 regarding restrictions on public disclosure to the extent such information has been identified as proprietary by Westinghouse, copyright protection notwithstanding. With respect to the non-proprietary versions of these reports, the NRC is permitted to make the number of copies beyond these necessary for its internal use which are necessary in order to have one copy available for public viewing in the appropriate docket files in the public document room in Washington, DC and in local public document rooms as may be required by NRC regulations if the number of copies submitted is insufficient for this purpose. Copies made by the NRC must include the copyright notice in all instances and the proprietary notice if the original was identified as proprietary.

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).

Attachment 1

Table 1, "List of Westinghouse's Responses to RAIs Transmitted in DCP/NRC1566"

Docket No. 52-006

DCP/NRC1566

April 7, 2003

Attachment 1

Table 1 "List of Westinghouse's Revised Responses to RAIs as Transmitted in DCP/NRC1566"
252.001, Rev. 1 440.162P, Rev. 1 440.162, Rev. 1 440.184, Rev. 1 630.021, Rev. 1 720.080, Rev. 1

DCP/NRC1566

April 7, 2003

Attachment 3

Westinghouse Revised Non-Proprietary Responses
to US Nuclear Regulatory Commission
Requests for Additional Information
dated March 2003

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 252.001 (Revision 1 Response)

Original Question:

Recent NRC generic communications, including NRC Bulletins 2001-01, 2002-01 and 2002-02, have addressed issues related to cracking of vessel head penetration (VHP) nozzles and degradation of the reactor pressure vessel (RPV) head in operating PWRs. Describe how this operational experience has been incorporated into the AP1000 design. Specifically, address the differences in the AP1000 design compared to the current fleet of PWRs, including the following specific items:

a. geometry of the VHP nozzle weld joint,

b. processes used for fabrication of the nozzle base material,

c. accessibility for inspection of the VHP nozzles and the RPV head - describe any impediments or limitations in the AP1000 design,

d. materials used for both the nozzle base material and the welds, and

e. operating conditions, including the operating temperature of the RPV head, provisions for bypass flow to cool the head, etc. (Section 4.5.1)

Revision 0 Westinghouse Response:

In relation to the Alloy 600 issues of the current fleet of PWRs, operational experiences in materials, fabrication processes, and inspection methods were considered in the design of AP1000 RPV head. A comparison of some key design and fabrication features in the AP1000 compared to the current fleet of Westinghouse PWRs are as follows:

a. The geometry of the AP1000 vessel head penetration nozzle weld joint is the same as in current Westinghouse PWRs.

b. The main process used for fabrication of the AP1000 nozzle base material is the automatic welding process. It will be combined with manual welding processes as necessary. In current Westinghouse PWRs the main welding processes were manual.

c. The AP1000 design contains the following penetrations through the reactor vessel head: CRDM penetrations, top-mounted in-core instrumentation and a vent line. There are no top-mounted in-core instrumentation penetrations in the current fleet of Westinghouse-designed PWRs. Accessibility to the AP1000 penetrations for inspection is the same as that for current PWRs, e.g. under the head. Inspection accessibility has been increased to the ID surface of the CRDM penetrations in that the thermal sleeves have been eliminated.

Thus the small gap access is eliminated and substituted for an open access tube. The top-mounted in-core instrumentation penetrations and vent line are also open access tubes. Open access tubes allow for easier insertion of inspection probes/end effectors into the penetration and greater flexibility in the implementation of a multitude of inspection approaches.

The AP1000 design has an integrated head package permanently attached to the reactor vessel head. This acts to reduce access to the top of the vessel head for inspection as compared to the current fleet of PWRs. However, the integrated head package has doors just above the vessel head that allow inspection access. Vessel head insulation configuration and access ports through this insulation allow for the implementation of visual inspection approaches across the vessel head.

d. Alloy 690 is used for both the base material and weld filler metal in the AP1000. Alloy 600 was used in current Westinghouse PWRs.

e. The operating temperature for the AP1000 reactor vessel head is approximately 560 F, which is between Tcold (537 F) and Thot (610 F). This temperature is in the colder range of current Westinghouse PWR plants that run at various temperatures between Tcold and Thot. The bypass flow to cool the vessel head is provided through spray nozzles similar to current plants. The AP1000 nozzle area has been sized to provide approximately 1.5% of the total flow to the head.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

NRC Additional Questions:

The response stated, in part, that "(t)he AP1000 design has an integrated head package permanently attached to the reactor vessel head. This acts to reduce access to the top of the vessel head for inspection as compared to the current fleet of PWRs. However, the integrated head package has doors just above the vessel head that allow inspection access. Vessel head insulation configuration and access ports through this insulation allow for the implementation of visual inspection approaches across the vessel head."

a. Please clarify what is meant by "visual inspection approaches across the vessel head". Does the access allow for bare metal visual examination of the vessel head penetration to RPV junction at the top of the RPV as discussed in NRC Bulletin 2002-02, "Reactor Pressure Vessel Head and Vessel Head Penetration Nozzle Inspection Programs".

b. Explain how the design permits for a visual examination of 360° around each reactor vessel head penetration.

c. Provide drawing/diagrams of the integrated head package showing access, shroud, insulation and penetrations. Discuss how is the insulation fixed and the extent to which it is removable. What is the minimum offset of the insulation from the surface of the RPV head?

d. In response to a prior RAI, you stated that the geometry of the AP1000 VHP nozzle weld joint is the same as in current Westinghouse PWRs. Would the changes Westinghouse described to the fabrication and installation process of the penetration nozzles reduce residual stresses or the effects of work hardening? Have there been any changes to the volume of weld metal, surface conditioning, etc. that could serve to reduce the residual stresses in the welds? Have any calculations been performed to compare the residual stresses in these welds to the current Westinghouse PWRs, and if so, how do the stresses compare for the AP1000 and current PWRs on the nozzle ID and OD (magnitudes and directions)?

e. Relative to the inspections of present heads, interpretation of inspection findings have at times been complicated by a need to determine if an indication is service-induced cracking or an artifact from fabrication. What preservice examinations will be applied to the VHP nozzles, i.e., will the VHP nozzles be subjected to a volumetric examination? Will the welds be examined using either surface or volumetric techniques, or both?

f. How was the head operating temperature of 560 F determined, and has this been reviewed separately by the NRC?

Westinghouse Response to Additional Questions:

a. " Visual inspection approaches across the vessel head" was meant to include the bare metal visual examination of 100% of the RPV head surface (including 360-degrees around the RPV head penetration nozzle) as mandated by NRC Order EA-03-009 dated February 11, 2003. The more general term was used because it allows the inclusion of other visual examinations inside the insulation, i.e. insulation integrity. The seven shroud ports (shown as 'A' in the attached Figures 1 and 2) provide direct access to seven removable insulation doors (shown as 'C' in Figures 1 and 2). These insulation doors provide direct access to the bare vessel head and penetrations (CRDM and instrumentation) for use of a remote, mobile visual inspection manipulator. In addition, twelve removable insulation panels (shown as 'K' in the attached Figures 1 and 2) provide other, larger access windows for a remote, mobile visual inspection manipulator. These twelve removable panels allow better access to areas near the integrated head package support lugs and direct access to the regions at the interface of the insulation and the head.

b. The plan view drawing (Figure 2) shows the seven shroud ports, the corresponding seven removable insulation doors, and the twelve removable insulation panels. Removal of the shroud ports/insulation doors provides direct access to visual inspection lanes (as shown on Figure 2). Such lanes can be used by a remote, mobile visual inspection manipulator for inspecting 360-degrees around each penetration through the head (CRDM and instrumentation) and for looking on the vessel head in general. The primary lanes run parallel to the 90-270 degree axis and can be accessed by the six openings centered about the 90 and 270 degree positions. The seventh opening provides access to lanes that run parallel to the 0-90 degree axis. Selected lanes along the 0-90 axis may be needed to provide coverage in the two outer areas at 90-degrees and at 180-degrees. Removal of the insulation panels that interface with the head allows for better visual examination access to the areas near the twelve integrated head package support lugs as well as the region coinciding with the insulation/head interface. In addition these panels provide another access window for the remote, mobile visual inspection manipulator.

c. The attached two figures are included to show the AP1000 reactor vessel head and integrated head package, and associated access features related to head inspection. Figure 1 shows a cross-sectional view of the vessel head and integrated head package. Figure 2 shows a plan view from below the head vessel insulation. These drawings are intended to show access to under the head insulation as well as to appropriate levels within the shroud above the insulation. Access doors/ports/windows/panels are identified on the drawings. The limiting access gaps are also indicated.

The seven shroud ports/insulation doors and twelve removable insulation panels ('A' and 'C', and 'K', respectively) provide direct access to the bare vessel head metal including the interface between the CRDM and instrumentation penetrations and the top of the head. The four shroud manways ('B') provide access to the lower region of the integrated head package above the insulation. At this level, the required surface or volumetric examinations of the ASME Section XI Examination Category B-O welds (10% of the outer periphery) can be accomplished as well as a general visual examination of the lower integrated head package compartment. The access point ('H') provides direct access to the upper integrated head package compartment for general visual examinations and for any required ASME Section XI examinations of the upper instrumentation tube connections.

The insulation is of a panel-type construction (reflector type insulation using thin metal sheeting) that forms a cap over the head, with holes to allow for the penetration of CRDM and instrumentation into the upper part of the integrated head package. Removable doors are to be included in the insulation design such that they align with the shroud ports. The minimum offset of the insulation from the surface of the RPV head is 3". Removable, curved insulation panels below the integrated head package shroud are also included.

d. The use of automatic welding processes provides much better control of the J-groove weld for the head adapters than the manual process utilized on the original reactor vessel closure heads for operating plants and results in the improvement of residual stresses. A narrow gap for the J-groove weld edge preparation is utilized to reduce the residual stresses in the weld. This design improves the residual stresses by reducing the volume of weld metal deposited.

The use of spray cooling on the inside surface of the head adapter during J-groove welding improves the stress distribution through the adapter wall thickness. The stresses are balanced by thermal elongation due to the temperature difference between the inner and outer surfaces, and by the large shrinkage of the outer portion of the weld metal, which improves residual stresses on the inner surface.

Incorporation of these fabrication methods on replacement heads for current plants has resulted in an improvement in residual stresses of approximately 30%.

e. Preservice examinations (PSI) are to be consistent with at least the inservice inspection requirements mandated by NRC Order EA-03-009 dated February 11, 2003. As a guideline the recommendations offered by the MRP Alloy 600 Inspection Working Group on March 18, 2003 are to be applied to define these PSI requirements.

The preservice examinations to be applied to the AP1000 program are as follows:

1. A baseline top-of-the head visual examination will be performed with the results stored on electronic media. This will be conducted at the time the head is installed. This examination will include 360-degrees around each vessel penetration.

2. Qualified ultrasonic examinations will be conducted from the ID surface of each vessel head penetration from the point coincident with the top of head to the bottom

of the penetration. The purpose of these examinations will be to examine the volume of the penetration tube (particularly the ID and OD surfaces), to profile the J-groove weld to penetration tube interface, to identify any welding defects lying at the interface between the J-groove weld and the penetration tube, and to fingerprint the tube-to-vessel head interference fit region. All indications, regardless of length, orientation or configuration, will be properly located and recorded using a data file format that can be transferred to future systems.

3. Qualified eddy current examinations will be conducted on the surface of the J-groove welds, on the OD surface of the vessel penetrations, and on the ID surface of the penetrations. The ID surface of the penetration region will coincide with the UT coverage. All indications, regardless of length, orientation or configuration, will be properly located and recorded using a data file format that can be transferred to future systems.

4. Post-hydro liquid penetrant examinations will be conducted on accessible surfaces that had undergone PSI ET examinations. All indications, regardless of length, orientation or configuration, would be properly located and recorded. Any indications exceeding the ASME Code Section III requirements would be removed.

f. The reactor vessel head temperature is determined by the amount of cooling provided by core bypass flow through spray nozzles. These nozzles are flow paths between the reactor vessel and core barrel annulus and the fluid volume in the vessel closure head region above the upper support plate. A fraction of the flow that enters the vessel inlet nozzles and into the vessel/barrel downcomer passes through these head cooling nozzles and into the vessel closure head region.

Calculation of the head operating temperature for the AP1000 has been performed according to the standard design practice for reactor vessel evaluations that has been utilized in the design of Westinghouse operating plants. The core bypass flow fraction to the spray nozzles and the resulting head operating temperature are calculated utilizing an integrated computer code that models Westinghouse reactor vessels and internals. The code calculates reactor vessel pressure losses, baffle-barrel flows and velocities, core bypass flows, and hydraulic lift forces. This code has been utilized extensively in reactor vessel hydraulic calculations for Westinghouse operating plants.

The computer code utilized in the calculations is a Westinghouse design code, and has not been reviewed separately by the NRC. The reactor vessel head temperature calculated by the computer code has been verified by comparison to operating plant data. Thermocouples have been installed in the upper head region of reactor vessels in several operating Westinghouse plants. The normal operating temperature data obtained from this instrumentation have been used to validate the average head region temperatures calculated by the Westinghouse design computer code.

Design Control Document (DCD) Revision:

From DCD Revision 3 page 5.3-17:

5.3.4.7 Inservice Surveillance

The internal surfaces of the reactor vessel are accessible for periodic inspection. Visual and/or nondestructive techniques are used. During refueling, the vessel cladding is capable of being inspected in certain areas of the upper shell above the primary coolant inlet nozzles, and if deemed necessary, the core barrel is capable of being removed, making the entire inside vessel surface accessible.

The closure head is examined visually during each refueling. Optical devices permit a selective inspection of the cladding, control rod drive mechanism nozzles, and the gasket seating surface. **Access to the top head surface is provided by seven ports around the circumference of the integrated head package shroud and by twelve removable insulation panels that interface with the head under the integrated head package shroud. Both the ports and the insulation panels provide access to the bare vessel head and CRDM and instrumentation penetrations for use of a remote, mobile visual inspection manipulator to perform a 360 ° inspection around each penetration. The head insulation is a stand-off design with a minimum offset from the head surface of three inches.**

The knuckle transition piece, which is the area of highest stress of the closure head, is accessible on the outer surface for visual inspection, dye penetrant or magnetic particle testing, and ultrasonic testing. The closure studs and nuts can be inspected periodically using visual, magnetic particle, and ultrasonic techniques.

From DCD Revision 3, page 5.3-19, Section 5.3.4.7:

- After the shop hydrostatic testing, full-penetration ferritic pressure boundary welds (with the exception of the closure head welds), as well as the nozzles to safe end welds, are ultrasonically examined from both the inside and outside diameters in addition to ASME Code, Section III requirements. ~~The closure head ferritic pressure boundary welds are examined from the outside diameter only.~~

- **Preservice examinations for the closure head will include a baseline top-of-the head visual examination; ultrasonic examinations of the inside diameter surface of each vessel head penetration; eddy current examinations of the surface of head penetration welds, the outside diameter surface of the vessel penetrations, and the inside diameter surface of the penetrations; and post-hydro liquid penetrant examinations of accessible surfaces that have undergone preservice inspection eddy current examinations.**

PRA Revision:

None

Figure 1: AP1000 Integrated Head Package and Reactor Vessel Head Elevation View

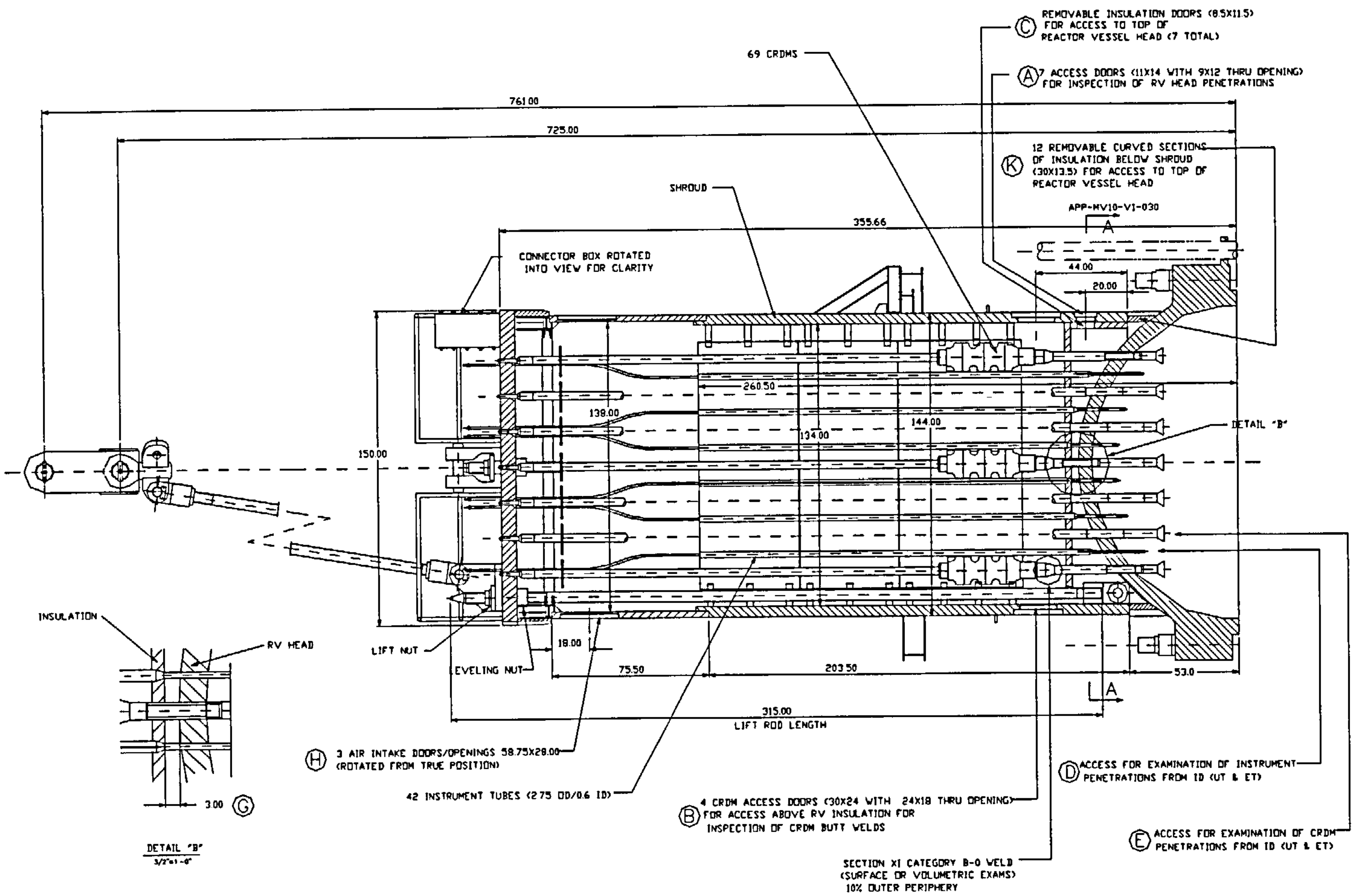
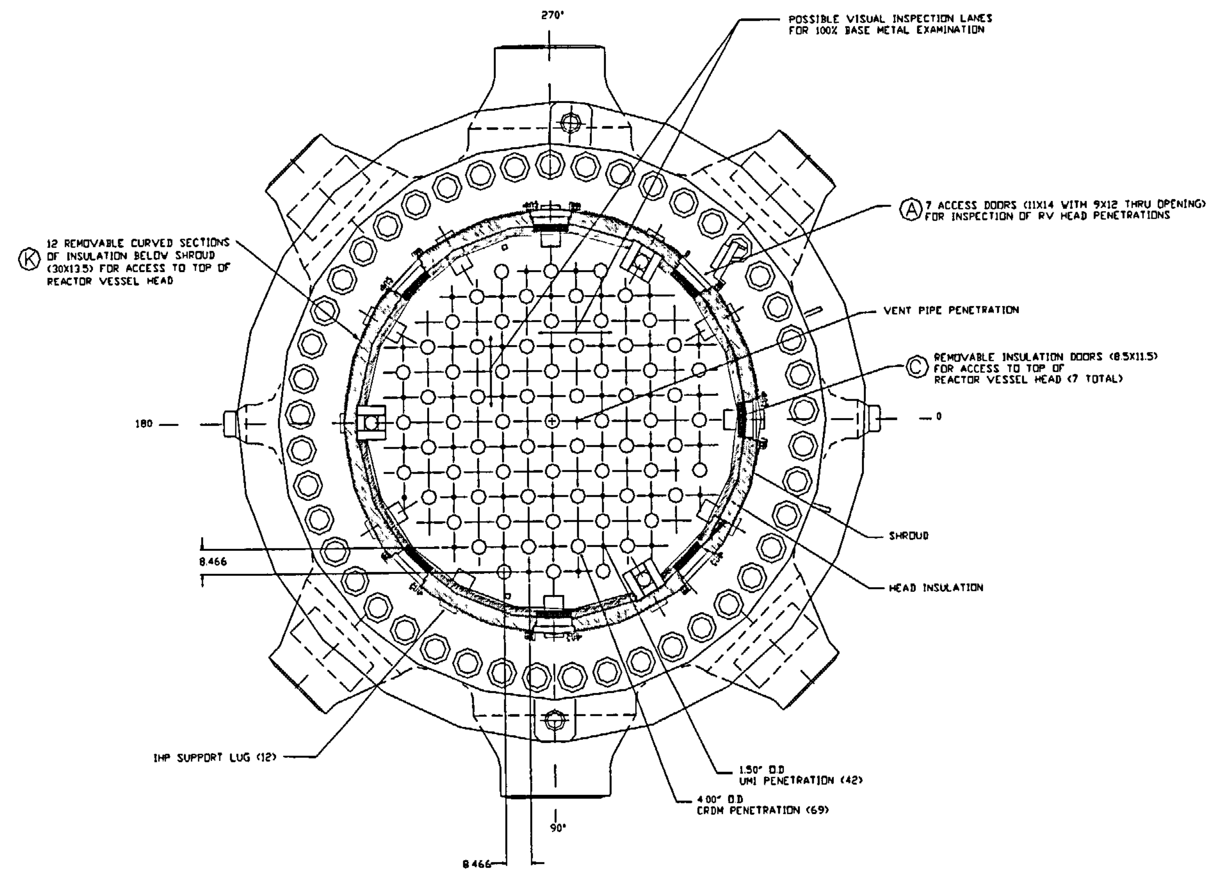


Figure 2: AP100 Reactor Vessel Head Plan View



SECTION A-A

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 440.162 (Response Revision 1)

Question:

Section 2.3.2 provides an assessment of WCOBRA/TRAC-AP using APEX Test SB18. This is a small cold leg break with a simulated failure of one of the ADS-4 lines. Please provide information for the following:

(a) For the comparison of predicted and measured pressurizer levels shown in Figure 2-29, justify the claim that the WC/T level agrees "extremely well" with the data through 1150 seconds, although WC/T clearly underpredicts the level for most of this period and does not capture the oscillations in level that are seen in the data.

(b) The predicted collapsed liquid levels in the downcomer, core, and upper plenum for Test SB18 are shown in Figures 2-31, 2-32, and 2-33, respectively. On page 2-52 the claim is made that the relatively constant code predicted levels are "consistent with the test data." However, no test data are presented in these three Figures. Please provide a meaningful comparison of predicted and measured results to validate this claim.

(c) Page 2-52 describes a "detailed comparison of vessel mass inventory with the test inventory" to show that the WCOBRA/TRAC prediction is in "excellent" agreement with the measured mass reduction during the ADS-4/IRWST initiation phase. There are no Figures comparing the predicted and measured inventories for Test SB18. Please provide this comparison.

(d) Section 2.3.2 concludes that the WCOBRA/TRAC prediction is in reasonable agreement with Test SB18 data and the code can be used in AP1000 calculations. This conclusion is reached with only three comparisons between the predicted and measured results; pressurizer level in Figure 2-29, integrated liquid flow in Figure 2-30, and downcomer pressure in Figure 2-34. Since the system pressure is primarily set by input to the BREAK Components in the model, Figure 2-34 may not be a true indication of code performance. Section 2.2.2 in WCAP-15833 showed that condensation heat transfer is underpredicted and steam flow rates in the hot leg are overpredicted. An overprediction of steam velocities in the hot legs for Test SB18 would result in an overprediction of ADS-4 flows. Thus, the apparently reasonable agreement in Figure 2-30 ADS-4 flow may be right for the wrong reasons. It remains to be shown therefore, that the simulation of Test SB18 is reasonable in comparison to experimental data and free of compensating errors. Please provide sufficient comparisons between predicted and measured results to demonstrate adequate simulation of Test SB18. Included in the comparisons and evaluation of code performance should be ADS-4 steam and liquid flows (not just the total integral), ADS-4 quality, hot leg levels, upper plenum two-phase level, and fluid temperatures throughout the system. Provide information sufficient to characterize how WCOBRA/TRAC predicted entrainment in the upper plenum and hot legs during the simulation of Test SB18.

Westinghouse Response:

a) The WCOBRA/TRAC result is within the fluctuations in the test data through the 1150 second time. Therefore, the agreement is characterized as "reasonable" according to the WCAP-15833 Revision 1 Section 2.3.2 assessment criteria.

b) Figures 440.162-1, 2 and 3 provide the requested comparisons with test data. The code agreement with the upper plenum collapsed liquid level shows that the margin to core uncovery is predicted in a "reasonable" manner by the code. The code predicts less liquid in the downcomer than the data. In Figure 440.162-3, the WCOBRA/TRAC collapsed level shown encompasses a greater span than the length present between the APEX core region pressure taps over which the data was measured. When this is taken into account, the collapsed levels agree even better than the figure indicates. Overall, the level comparisons indicate that the vessel mass prediction is in adequate agreement with the Test SB18 data; the section (c) response provides further information.

c) A plot of the change in reactor vessel mass inventory during the Test SB18 ADS-4 IRWST initiation phase (Figure 440.162-4) is provided for comparison with the WCOBRA/TRAC prediction of this mass inventory change (Figure 440.162-5). Both the test data and the code indicate that a small decrease in the vessel mass inventory occurs by the time of IRWST initiation during Test SB18.

d) Many of the requested comparisons cannot be provided because the necessary test data does not exist for the APEX Facility. Specifically, the ADS-4 steam flow rate was not accurately measured because the flow meters were out of range during the tests. Only the integral liquid flow rates through the ADS-4 flow paths are available. In the absence of the instantaneous steam and liquid flow rate data, the ADS-4 flow quality cannot be calculated. Refer to RAI 440.165 for an estimate of the ADS-4 flow quality for Test SB18.

As regards the requested level comparisons, hot leg levels in the horizontal pipe section are not available for comparison with WCOBRA/TRAC to characterize the entrainment prediction through the ADS-4 offtake. Also, two-phase level in the upper plenum cannot be determined from the available data. The upper plenum collapsed liquid level prediction agrees well with the data as shown in Figure 440.162-1. In the WCOBRA/TRAC test simulation, temperatures are initialized to the NOTRUMP-predicted values to correspond to the method used for the AP1000 calculations. The lower plenum temperature from the test data is presented in Figure 440.162-6 for the ADS-4 IRWST initiation phase of Test SB18. The WCOBRA/TRAC lower plenum temperature is shown in Figure 440.162-7; it exceeds the test value, due to the higher initial value specified at the time that ADS-4 actuates.

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

NRC Additional Comment:

The response supplied in the December 2, 2002, transmittal memo (W Ref.: DCP/NRC1525) was evaluated with the following conclusions:

(a) The WCAP must be revised characterizing the agreement between WCOBRA/TRAC and data as being within oscillations exhibited by the data rather than "extremely well."

(b) The new three new figures (440.162-1 to 440.162-3) must be added to WCAP-15833. Section

Westinghouse Response:

(a) The WCAP will be revised as shown characterizing the agreement between WCOBRA/TRAC and data as being within oscillations exhibited by the data rather than "extremely well."

(b) WCAP-15833 will be updated to include the Westinghouse responses to NRC RAI related to WCAP-15833 and thus the figures will be included.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

WCAP Revision:

Section 2.3.2 will be revised as follows:

The WC/T level **is within the oscillations exhibited by** ~~agrees extremely well with~~ the data through [ ]^{a,c}, at which time it falls below; overall, the agreement is judged as reasonable.

OSU Test Sb18 2 Inch Cold Leg Break

Upper Plenum Collapsed Levels (Relative to Bottom of Lower Plenum)

Figure 440.162-1

(a,b,c)

Rev. 0

OSU Test Sb18 2 Inch Cold Leg Break

Collapsed Downcomer Levels (Relative to Bottom of Lower Plenum)

Figure 440.162-2

Rev 0

(a,b,c)

OSU Test Sb18 2 Inch Cold Leg Break

Core Collapsed Liquid Levels (Relative to Bottom of Lower Plenum)

Figure 440.162-3

(a,b,c)

Rev 0

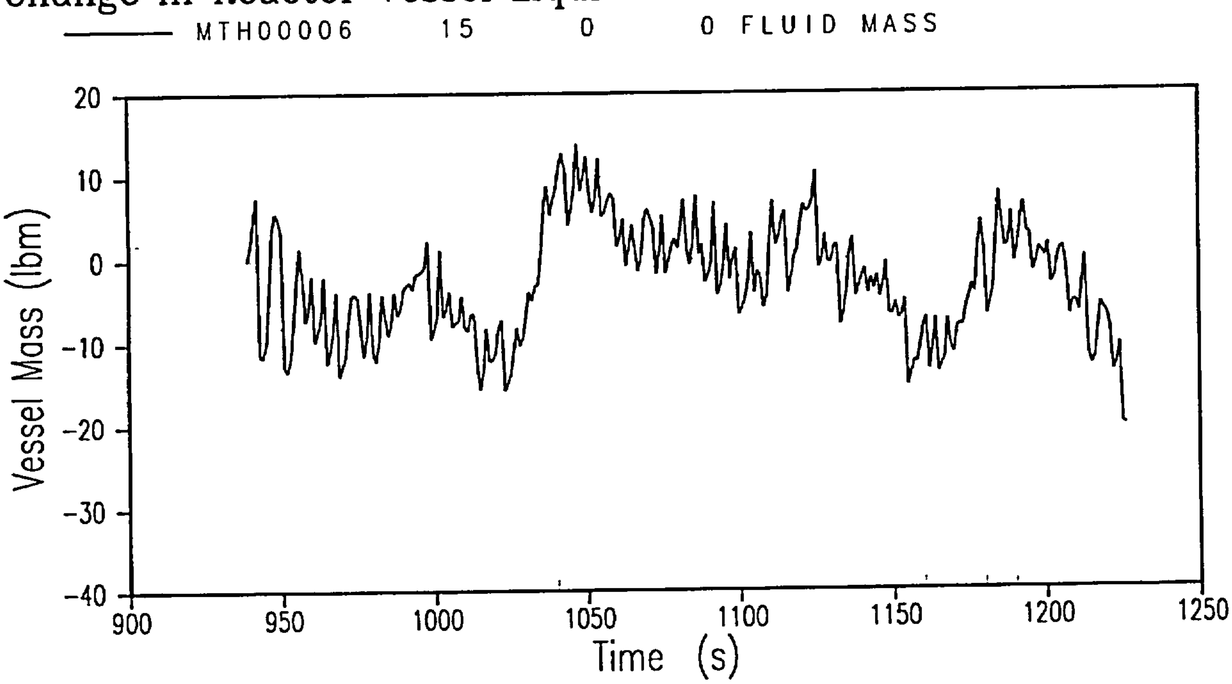
OSU Test Sb18 2 Inch Cold Leg Break

Change In Reactor Vessel Liquid Mass From ADS-4 Actuation

(a,b,c)

Figure 440.162-4

Figure 440.162-5 WCOBRA/TRAC Simulation of OSU Test SB18
Change in Reactor Vessel Liquid Mass from ADS-4 Actuation



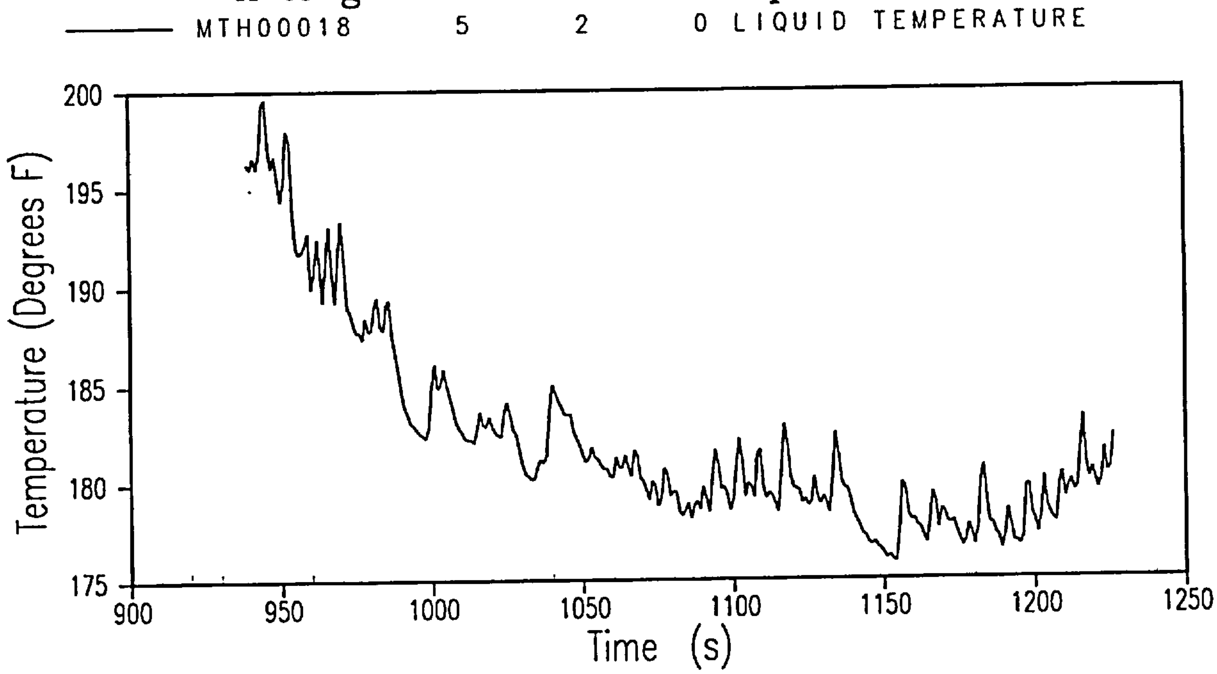
OSU Test Sb18 2 Inch Cold Leg Break
Core Inlet Temperature

Figure 440.162-6

(a,b,c)

Rev. 0

Figure 440.162-7 WCOBRA/TRAC Simulation of OSU Test SB18
Average Lower Plenum Temperature



AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 440.184 (Response Revision 1)

Question:

New Generic Issue 163, "Multiple Steam Generator [SG] Tube Leakage," in NUREG-0933, "A Prioritization of Generic Safety Issues," identifies a safety concern associated with potential multiple steam generator tube leaks triggered by a main steam line break outside containment that cannot be isolated. This sequence of events could lead to core damage due to the loss of all primary system coolant and safety injection fluid in the refueling water storage tank. The Nuclear Regulatory Commission (NRC) has given this issue HIGH priority ranking, and is working toward a resolution of the issue. The AP1000 design control document (DCD) Section 1.9, "Compliance with Regulatory Criteria," does not address this issue, except that Table 1.9-2 indicates that Generic Issue 163 is unresolved pending generic resolution.

(A) Please provide an evaluation of the AP1000 design with respect to coping with the safety concern of Issue 163 regarding multiple SG tube leakage resulting from a steam line break outside containment.

(B) Please discuss how any subsequent requirements that may be imposed by the NRC as a resolution of this issue will be identified to a prospective combined license applicant that references the AP1000 design, i.e., how are we given assurance that a combined license applicant will commit to complying with any requirements that may be imposed by the NRC as a resolution of Issue 163.

Westinghouse Original Response:

A. The AP1000 plant response to a main steam line break (MSLB) scrams the reactor automatically and removes decay heat via the intact generator or the passive RHR heat exchanger. If the MSLB is not isolated the RCS will continue to lose coolant after shutdown through leaking steam generator tubes, the plant responds to the scenario as a small loss of coolant accident. The core makeup tanks drain and produce a low level signal. The plant protection and monitoring system depressurizes the RCS via the automatic depressurization system (ADS). The core remains covered throughout the scenario. Once the RCS is depressurized, the much lower containment pressure stops the containment water loss through the leaking steam generator tubes. Therefore, no long-term core uncovery is expected.

B. Based on the above discussion, this issue should be considered closed for the AP1000, and no additional action item for the Combined Licensing Applicant should be required in the DCD.

NRC Additional Comment:

Comments on RAI 440.184 response:

1. Please clarify the statement in the response that "Once the RCS is depressurized, the much lower containment pressure stops the containment water loss through the leaking steam generator tubes. Therefore, no long-term core uncovery is expected."

2. Provide analyses, including the assumed number of ruptured tubes and basis, to support your conclusion that the core remains covered throughout the scenario of the main steamline break with consequential multiple steam generator tube rupture.

Westinghouse Additional Response:

1. The event postulates that an unisolated main steam line beak outside containment results in primary side tube leakage to the secondary side and out the broken steam line. With the AP1000 automatic depressurization system, the primary side pressure is reduced, and thus the tube leakage from the primary to secondary system will not occur. The statement is describing the situation that once ADS is actuated, the primary side pressure is the same as containment pressure, and therefore the much lower containment pressure will result in no leakage from the primary side to the secondary side through the leaking steam generators.

2. The elevation of the steam line, in comparison to the ADS-4 discharge pipe has been evaluated. The elevation of the high point of the steam line is approximately 80 feet higher than the elevation of the ADS-4 discharge. Based on this elevation difference, once ADS-4 is actuated, leakage from the primary side through the steam generator tubes will stop after the RCS becomes depressurized. Based on this analysis, Westinghouse concludes that the ADS-4 operation sufficiently addresses the Generic Safety Issue discussed in this RAI.

   Westinghouse has performed other analyses that can also be used to address aspects of this question. A multiple steam generator tube rupture analysis that was included in our response to RAI 440.043 was performed that illustrated the plant response to a hypothetical simultaneous rupture of 5 steam generator tubes. In these cases, the passive safety systems are shown to maintain adequate core cooling, and prevent steam generator overfill, for the case of either a failed SG PORV, or a stuck open SG safety valve. For these events, the automatic depressurization system is not actuated. Actuation of the ADS would reduce any postulated primary to secondary leakage for a hypothetical main steam line break followed by steam generator tube leakage. The Generic Safety Issue is still under consideration by the NRC staff for the current fleet of reactors. The passive safety systems provide unique advantages in dealing with this beyond design basis event. The automatic depressurization system can be relied upon to terminate the primary to secondary side leak, and prevent long-term core uncovery for this even. Therefore we conclude this issue can be resolved for AP1000.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 630.021 (Response Revision 1)

Question:

(Section 16.1, Bases References for TSs 3.3.1 and 3.3.2) Reference 7 for TS 3.3.1 and Reference 6 for TS 3.3.2 seem to cite the same document, WCAP-10271-P-A, "Evaluation of Surveillance Frequencies and Out of Service Times for the Reactor Protection Instrumentation System," Supplement 2, Revision 1. However the document dates cited in TS 3.3.1 and TS 3.3.2 are not the same (June 1996 and June 1990, respectively). Please correct this discrepancy. The reactor trip system (RTS) and ESFAS instrumentation designs for the AP1000 are different from those addressed in the STS and the AP600. Explain how WCAP-10271-P-A applies to the AP1000 instrumentation test intervals and allowed outage times.

Westinghouse Response:

June 1990 is the correct date for WCAP-10271-P-A, Supplement 2, Revision 1. Reference 7 of Technical Specification (TS) Bases 3.3.1 will be corrected as shown below.

WCAP-10271-P-A, Supplement 2, Revision 1, is applied to the AP1000 as the basis for some of the TS Completion Times. The application of WCAP-10271-P-A follows the template of the Standard Technical Specifications. Although some aspects of the digital systems are different for AP600 and AP1000, the functional design of the AP1000 protection system is the same as that for AP600. WCAP-10271-P-A, Supplement 2, Revision 1, is applicable to the reactor trip and engineered safety features actuated by the protection and safety monitoring system for both the AP1000 and the AP600. The NRC, via a safety evaluation report issued on the Sequoyah Nuclear Plant, Unit 1, Docket, has approved the applicability of the methodologies provided in this WCAP to digital equipment. The AP1000 PRA also provides justification for the TS Completion Times.

Design Control Document (DCD) Revision:

3.3.1 Reactor Trip System (RTS) Instrumentation

REFERENCES

7. WCAP-10271-P-A, "Evaluation of Surveillance Frequencies and Out of Service Times for the Reactor Protection Instrumentation System," Supplement 2, Revision 1, June ~~1996~~**1990**. |

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

NRC Additional Comments:

AP1000 technical specification Sections TS 3.3.1 and TS 3.3.2 are referring the topical report WCAP-10271-P-A, Supplement 2, Revision 1, "Evaluation of Surveillance Frequencies and Out of Service Times for the Reactor Protection Instrumentation System" as the basis for some of the TS Completion Times. However, the topical report WCAP-10271 is based on the protection system uses analog system hardware. Many aspects of the functional design are different for the AP1000 digital systems. In response to RAI 630.021, Westinghouse did not adequately demonstrate the applicability of the generic analyses of WCAP-10271 for AP1000 protection system. The COL applicant should provide detailed plant protection system FMEA and component reliability data to justify the TS Completion Time. Please provide justification for the applicability of WCAP-10271 for the AP1000.

Westinghouse Additional Response:

The response to RAI 420.028 added a COL item to perform a plant specific FMEA for the protection system. DCD Revision 3 incorporates the changes identified in the original response to this RAI and RAI 420.028.

Because the plant specific FMEA will be performed by the COL, it is appropriate to use brackets in the technical specifications for those values that rely on the FMEA for their basis.

Design Control Document (DCD) Revision: (Response Revision 1)

DCD Chapter 16 will be revised to include the following changes:

3.3.1 Reactor Trip System (RTS) Instrumentation

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. One or two Power Range Neutron Flux – High channels inoperable.	D.1.1 Reduce THERMAL POWER to ≤ 75% RTP. AND	12 hours
	D.1.2 Place one inoperable channel in bypass or trip. AND	[6] hours

CONDITION	REQUIRED ACTION		COMPLETION TIME
	D.1.3	With two inoperable channels, place one channel in bypass and one channel in trip.	[6] hours
	<u>OR</u>		
	D.2.1	Place inoperable channel(s) in bypass.	[6] hours
E. One or two channels inoperable.	E.1.1	Place one inoperable channel in bypass or trip.	[6] hours
	<u>AND</u>		
	E.1.2	With two channels inoperable, place one channel in bypass and one channel in trip.	[6] hours
F. THERMAL POWER between P-6 and P-10, one or two Intermediate Range Neutron Flux channels inoperable.	F.1.1	Place one inoperable channel in bypass or trip.	[2] hours
	<u>AND</u>		
	F.1.2	With two channels inoperable, place one channel in bypass and one channel in trip.	[2] hours
K. One or two channels inoperable.	K.1.1	Place one inoperable channel in bypass or trip.	[6] hours
	<u>AND</u>		
	K.1.2	With two channels inoperable, place one channel in bypass and one channel in trip.	[6] hours
L. One or two channels inoperable.	L.1.1	Place one inoperable channel in bypass or trip.	[6] hours

CONDITION	REQUIRED ACTION	COMPLETION TIME
	AND	
	L.1.2 With two channels inoperable, place one channel in bypass and one channel in trip.	[6] hours
N. One or two interlock channels inoperable.	N.1 Verify the interlocks are in required state for existing plant conditions.	1 hour
	OR	
	N.2.1 Place the Functions associated with one inoperable interlock channel in bypass or trip.	[7] hours
	AND	
	N.2.2 With two interlock channels inoperable, place the Functions associated with one inoperable interlock channel in bypass and with one inoperable interlock channel in trip.	[7] hours
O. One or two interlock channels inoperable.	O.1 Verify the interlocks are in required state for existing plant conditions.	1 hour
	OR	
	O.2.1 Place the Functions associated with one inoperable interlock channel in bypass.	[7] hours
	AND	

CONDITION	REQUIRED ACTION		COMPLETION TIME
	O.2.2	With two interlock channels inoperable, place the Functions associated with one inoperable interlock channel in bypass and with one inoperable interlock channel in trip.	[7] hours
S. One or two Source Range Neutron Flux channel inoperable.	S.1	Restore three of four channels to OPERABLE status.	[48] hours
	OR		
	S.2	Open RTBs.	[49] hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.1.6	- NOTE - Not required to be performed for source range instrumentation prior to entering MODE 3 from MODE 2 until 4 hours after entry into MODE 3.	
	Perform RTCOT.	[92] days

3.3.2 Engineered Safety Feature Actuation System (ESFAS) Instrumentation

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. One or two channels or divisions inoperable.	B.1	Place one inoperable channel or division in bypass or trip.	[6] hours
	AND		

		B.2	With two inoperable channels or divisions, place one inoperable channel or division in bypass and one inoperable channel or division in trip.	[6] hours
C.	One channel inoperable.	C.1	Place inoperable channel in bypass.	[6] hours
I.	One or two channels inoperable.	I.1	Place one inoperable channel in bypass or trip.	[6] hours
		<u>AND</u>		
		I.2	With two inoperable channels, place one channel in bypass and one channel in trip.	[6] hours
J.	One or two interlock channels inoperable.	J.1	Verify the interlocks are in the required state for the existing plant conditions.	1 hour
		<u>OR</u>		
		J.2.1	Place the Functions associated with one inoperable interlock channel in bypass or trip.	[7] hours
		<u>AND</u>		
		J.2.2	With two interlock channels inoperable, place the Functions associated with one inoperable interlock channel in bypass and with one inoperable interlock channel in trip.	[7] hours

B 3.3.1 Reactor Trip System (RTS) Instrumentation

D.1.1, D.1.2, D.1.3, D.2.1, D.2.2, and D.3

With one or two channels inoperable, one affected channel must be placed in a bypass or trip condition within [6] hours. If one channel is bypassed, the logic becomes two-out-of-three, while still meeting the single failure criterion. (A failure in one of the three remaining channels will not prevent the protective function.) If one channel is tripped, the logic becomes one-out-of-three, while still meeting the single failure criterion. (A failure in one of the three remaining channels will not prevent the protective function.) If one channel is bypassed and one channel is tripped, the logic becomes one-out-of-two, while still meeting the single failure criterion. The [6] hours allowed to place the inoperable channel(s) in the bypassed or tripped condition is justified in Reference [7].

E.1.1, E.1.2, and E.2

With one or two channels inoperable, one affected channel must be placed in a bypass or trip condition within [6] hours. If one channel is bypassed, the logic becomes two-out-of-three, while still meeting the single failure criterion. (A failure in one of the three remaining channels will not prevent the protective function.) If one channel is tripped, the logic becomes one-out-of-three, while still meeting the single failure criterion. (A failure in one of the three remaining channels will not prevent the protective function.) If one channel is bypassed and one channel is tripped, the logic becomes one-out-of-two, while still meeting the single failure criterion. The [6] hours allowed to place the inoperable channel(s) in the bypassed or tripped condition is justified in Reference [7].

F.1.1, F.1.2, F.2, and F.3

With one or two channels inoperable, one affected channel must be placed in a bypass or trip condition within [2] hours. If one channel is bypassed, the logic becomes two-out-of-three, while still meeting the single failure criterion. (A failure in one of the three remaining channels will not prevent the protective function.) If one channel is tripped, the logic becomes one-out-of-three, while still meeting the single failure criterion. (A failure in one of the three remaining channels will not prevent the protective function.) If one channel is bypassed and one channel is tripped, the logic becomes one-out-of-two, while still meeting the single failure criterion. The [2] hours allowed to place the inoperable channel(s) in the bypassed or tripped condition is justified in Reference [7].

K.1.1, K.1.2, and K.2

With one or two channels inoperable, one affected channel must be placed in a bypass or trip condition within [6] hours. If one channel is bypassed, the logic becomes two-out-of-three, while still meeting the single failure criterion. (A failure in one of the three remaining channels will not prevent the protective function.) If one channel is tripped, the logic becomes one-out-of-three, while still meeting the single failure criterion. (A failure in one of the three remaining channels will not prevent the protective function.) If one channel is bypassed and one channel is tripped, the logic becomes one-out-of-two, while still meeting the single failure criterion. The [6] hours allowed to place the inoperable channel(s) in the bypassed or tripped condition is justified in Reference [7].

L.1.1, L.1.2, and L.2

With one or two channels inoperable, one affected channel must be placed in a bypass or trip condition within [6] hours. If one channel is bypassed, the logic becomes two-out-of-three, while still meeting the single failure criterion. (A failure in one of the three remaining channels will not prevent the protective function.) If one channel is tripped, the logic becomes one-out-of-three, while still meeting the single failure criterion. (A failure in one of the three remaining channels will not prevent the protective function.) If one channel is bypassed and one channel is tripped, the logic becomes one-out-of-two, while still meeting the single failure criterion. The [6] hours allowed to place the inoperable channel(s) in the bypassed or tripped condition is justified in Reference [7].

N.1, N.2.1, N.2.2, and N.3

Condition N applies to the P-6, P-10, and P-11 interlocks. With one or two channels inoperable, the associated interlock must be verified to be in its required state for the existing plant condition within 1 hour, or the Functions associated with inoperable interlocks placed in a bypassed or tripped condition within [7] hours, or the unit must be placed in MODE 3 within 13 hours. Verifying the interlock manually accomplishes the interlock condition.

If one interlock channel is inoperable, the associated Function(s) must be placed in a bypass or trip condition within [7] hours. If one channel is bypassed, the logic becomes two-out-of-three, while still meeting the single failure criterion. (A failure in one of the three remaining channels will not prevent the protective function.) If one channel is tripped, the logic becomes one-out-of-three, while still meeting the single failure criterion. (A failure in one of the three remaining channels will not prevent the protective function.)

If two interlock channels are inoperable, one channel of the associated Function(s) must be bypassed and one channel of the associated Function(s) must be tripped. In this state, the logic becomes one-out-of-two, while still meeting the single failure criterion. The [7] hours allowed to place the inoperable channel(s) in the bypassed or tripped condition is justified in Reference [7].

O.1, O.2.1, O.2.2, and O.3

Condition O applies to the P-8 interlock. With one or two channels inoperable, the associated interlock must be verified to be in its required state for the existing plant condition within 1 hour, or the Functions associated with inoperable interlocks placed in a bypassed or tripped condition within [7] hours, or the unit must be placed in MODE 2 within 13 hours. Verifying the interlock manually accomplishes the interlock condition.

If one interlock channel is inoperable, the associated Function(s) must be placed in a bypass or trip condition within [7] hours. If one channel is bypassed, the logic becomes two-out-of-three, while still meeting the single failure criterion. (A failure in one of the three remaining channels will not prevent the protective function.) If one channel is tripped, the logic becomes one-out-of-three, while still meeting the single failure criterion. (A failure in one of the three remaining channels will not prevent the protective function.)

If two interlock channels are inoperable, one channel of the associated Function(s) must be bypassed and one channel of the associated Function(s) must be tripped. In this state, the logic becomes one-out-of-two, while still meeting the single failure criterion. The [7] hours allowed to place the inoperable channel(s) in the bypassed or tripped condition is justified in Reference [7].

S 1 and S.2

Condition S applies to one or two inoperable Source Range Neutron Flux channels in MODE 3, 4, or 5 with the RTBs closed and the PLS capable of rod withdrawal. With the unit in this Condition, below P-6, the NIS source range performs the monitoring and protection functions. With one or two of the source range channels inoperable, [48] hours is allowed to restore three of the four channels to an OPERABLE status. If the channels cannot be returned to an OPERABLE status, [1] additional hour is allowed to open the RTBs. Once the RTBs are open, the core is in a more stable condition and the unit enters Condition L. The allowance of [48] hours to restore the channel to OPERABLE status, and [the additional hour] to open the RTBs, are justified in Reference [7].

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3 1.6

SR 3.3.1.6 is the performance of a REACTOR TRIP CHANNEL OPERATIONAL TEST (RTCOT) every [92] days.

This test frequency of [92] days is justified based on Reference [7] and the use of continuous diagnostic test features, such as deadman timers, cross-check of redundant channels, memory checks, numeric coprocessor checks, and tests of timers, counters and crystal time bases, which will report a failure within the protection and safety monitoring system cabinets to the operator within 10 minutes of a detectable failure.

REFERENCES

[7. WCAP-10271-P-A, "Evaluation of Surveillance Frequencies and Out of Service Times for the Reactor Protection Instrumentation System," Supplement 2, Revision 1, June 1990.]

B 3.3.2 Engineered Safety Feature Actuation System (ESFAS) Instrumentation

B.1 and B.2

With one or two channels or divisions inoperable, one affected channel or division must be placed in a bypass or trip condition within [6] hours. If one channel or division is bypassed, the logic becomes two-out-of-three, while still meeting the single failure criterion. (A failure in one of the three remaining channels or divisions will not prevent the protective function.) If one channel or division is tripped, the logic becomes one-out-of-three, while still meeting the single failure criterion. (A failure in one of the three remaining channels or divisions will not prevent the protective function.) If one channel or division is bypassed and one channel or division is tripped, the logic becomes one-out-of-two, while still meeting the single failure criterion. The [6] hours allowed to place the inoperable channel(s) or division(s) in the bypassed or tripped condition is justified in Reference [6].

C.1

With one channel inoperable, the affected channel must be placed in a bypass condition within [6] hours. The [6] hours allowed to place the inoperable channel in the bypass condition is justified in Reference [6]. If one CVS isolation channel is bypassed, the logic becomes one-out-of-one. A single failure in the remaining channel could cause a spurious CVS isolation. Spurious CVS isolation, while undesirable, would not cause an upset plant condition.

I.1 and I.2

Condition I applies to IRWST containment recirculation valve actuation on safeguards actuation coincident with IRWST Level Low 3 (Function 23.b). With one or two channels inoperable, one affected channel must be placed in a bypass or trip condition within [6] hours. If one channel is bypassed, the logic becomes two-out-of-three, while still meeting the single failure criterion. (A failure in one of the three remaining channels will not prevent the protective function.) If one channel is tripped, the logic becomes one-out-of-three, while still meeting the single failure criterion. (A failure in one of the three remaining channels will not prevent the protective function.) If one channel is bypassed and one channel is tripped, the logic becomes one-out-of-two, while still meeting the single failure criterion. The [6] hours allowed to place the inoperable channel(s) in the bypassed or tripped condition is justified in Reference [6].

J.1 and J.2

Condition J applies to the P-6, P-11, P-12, and P-19 interlocks. With one or two required channel(s) inoperable, the associated interlock must be verified to be in its required state for the existing plant condition within

1 hour, or any Function channels associated with inoperable interlocks placed in a bypassed condition within [7] hours. Verifying the interlock state manually accomplishes the interlock role.

If one interlock channel is inoperable, the associated Function(s) must be placed in a bypass or trip condition within [7] hours. If one channel is bypassed, the logic becomes two-out-of-three, while still meeting the single failure criterion. (A failure in one of the three remaining channels will not prevent the protective function.) If one channel is tripped, the logic becomes one-out-of-three, while still meeting the single failure criterion. (A failure in one of the three remaining channels will not prevent the protective function.)

If two interlock channels are inoperable, one channel of the associated Function(s) must be bypassed and one channel of the associated Function(s) must be tripped. In this state, the logic becomes one-out-of-two, while still meeting the single failure criterion. The [7] hours allowed to place the inoperable channel(s) in the bypassed or tripped condition is justified in Reference [6].

REFERENCES

[6. WCAP-10271-P-A, Supplement 2, Rev. 1, "Evaluation of Surveillance Frequencies and Out of Service Times for the Reactor Protection Instrumentation System," dated June 1990.]

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 720.080 (Response Revision 1)

Question:

Gamma and Beta Doses in Figures D-1 and D-2 of the AP1000 PRA are less than the corresponding figures for the AP600. Considering the power rating has gone up, one would expect these doses to increase not decrease. Why is this less?

Westinghouse Response:

Relative to the AP600, the increase in the power rating would tend to result in higher doses for the AP1000 with all other parameters of equal values. A compensating design feature of the AP1000 is the larger containment volume.

The primary difference, though, is attributable to the total core inventory released and the timing of these releases. Both the AP600 and AP1000 calculations are based on NUREG-1465. For the AP600, additional considerations were made as defined by SECY-94-300 (December 1995). For the AP1000, the guidance provided in Regulatory Guide 1.183 (July 2000) was utilized. Tables 1 and 2, illustrate the releases as a function of time for the AP600 and the AP1000, respectively.

Table 1 – AP600
Core Inventory Fraction Released into Containment versus Time Period

	0-10 Minutes	10 Minutes	10-40 Minutes	40-118 Minutes	118-238 Minutes	238-718 Minutes	Total
Noble Gases	0.00	0.03	0.02	0.95	0.00	0.00	1.00
Halogens	0.00	0.03	0.02	0.35	0.25	0.10	0.75
Alkali Metals	0.00	0.03	0.02	0.25	0.35	0.10	0.75
Tellurium Metals	0.00	0.00	0.00	0.05	0.25	0.005	0.305
Ba, Sr	0.00	0.00	0.00	0.02	0.10	0.00	0.12
Noble Metals	0.00	0.00	0.00	0.0025	0.0025	0.00	0.005
Lanthanides	0.00	0.00	0.00	0.0002	0.005	0.00	0.0052
Cerium Group	0.00	0.00	0.00	0.0005	0.005	0.00	0.0055

Table 2 – AP1000 Core Inventory Fraction Released into Containment versus Time Period				
	0-10 Minutes	10-40 Minutes	40-118 Minutes	Total
Noble Gases	0.00	0.05	0.95	1.0
Halogens	0.00	0.05	0.35	0.4
Alkali Metals	0.00	0.05	0.25	0.3
Tellurium Metals	0.00	0.00	0.05	0.05
Ba, Sr	0.00	0.00	0.02	0.02
Noble Metals	0.00	0.00	0.0025	0.0025
Lanthanides	0.00	0.00	0.0002	0.0002
Cerium Group	0.00	0.00	0.0005	0.0005

Table 1 shows that an initial, instantaneous, release of a set of nuclides was simulated for the AP600 at 10 minutes. All other releases are made over a time-span. At 40 and 118 minutes, both the AP600 and the AP1000 have equivalent cumulative releases. The higher dose rate values for the AP1000 up to 118 minutes reflect the higher power rating and containment volume. After 118 minutes, no more releases are simulated for the AP1000, while they continue for the AP600. The cumulative release for all nuclides of the AP600 are larger than the AP1000 (except for the "Noble Gasses" which are equal). The higher doses for the AP600 (after approximately 3 hours) are attributable to the higher total releases.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

NRC Additional Comments:

The three release phase model used in the Westinghouse analysis is a design-basis accident model. Is it appropriate to use it for equipment survivability, or should a more conservative model be used?

Westinghouse Additional Response:

For the original AP1000 calculations, the guidance provided in Regulatory Guide 1.183 (July 2000) was used. Using this regulatory guide as a basis, the severe accident "Ex-Vessel" and

"Late In-Vessel" phases (as identified in NUREG-1465) were not included in the original AP1000 calculations.

The comparable AP600 analyses were carried out prior to issuance of Regulatory Guide 1.183 and addressed both the DBA and severe accident scenarios. Although the characterization of the radiation environment associated with a severe accident is not identified in those sections of Regulatory Guide 1.183 that address equipment qualification, Westinghouse has elected to update these figures (and supporting DCD documentation) to include the "Ex-Vessel" and "Late In-Vessel" phases. Thus, the information will be more consistent with the AP600 methodology and will provide information that could be used in addressing equipment qualification issues for a severe accident scenario. The revised figures are provided in the attached marked up sections of the AP1000 PRA.

The revised AP1000 results use the release fraction information defined in NUREG-1465. The latest information for elemental groupings provided in Table 2 of Regulatory Guide 1.183 has also been used. The key parameters used for all phases are summarized in Table 720.080R1-1 (including the time duration for each release phase).

Table 720.080R1-1
AP1000 Core Inventory Fraction Released into Containment versus Release Phase

	NUREG-1465 (Table 3.13) Gap Release 0.5 Hours	Early In-Vessel 1.3 Hours	Ex-Vessel 2 Hours	Late In-Vessel 10 Hours	Total Release Fraction
Noble Gasses	0.05	0.95	0	0	1.0
Halogens	0.05	0.35	0.25	0.1	0.65
Alkali Metals	0.05	0.25	0.35	0.1	0.65
Tellurium Metals	0	0.05	0.25	0.005	0.3
Ba, Sr	0	0.02	0.1	0	0.12
Noble Metals	0	0.0025	0.0025	0	0.005
Lanthanides	0	0.0002	0.005	0	0.0052
Cerium Group	0	0.0005	0.005	0	0.0055

Design Control Document (DCD) Revision:

None.

PRA Revision:

Revise section D.7 as shown in the attached mark-up.

RAI Number 720.080 Revision 1

D.6.4 Summary of Equipment and Instrumentation

The equipment and instrumentation used in achieving a controlled, stable state following a severe accident, and the time it operates are summarized in Tables D-3 through D-5.

D.7 Severe Accident Environments

D.7.1 Radiation Environment – Severe Accident

The radiation exposure inside the containment for a severe accident is conservatively estimated by considering the dose in the middle of the AP1000 containment with no credit for the shielding provided by internal structures.

Sources are based on the emergency safeguards system core thermal power rating and the following analytical assumptions:

- Power Level (including 2% power uncertainty) .... 3,468 MWt
- Fraction of total core inventory released to the containment atmosphere:

  Noble Gases (Xe, Kr) 1.0
  Halogens (I, Br) .... ~~0.40~~0 75
  Alkali Metals (Cs, Rb) .... ~~0.30~~0 75
  Tellurium Group (Te, Sb, Se) .... ~~0.05~~0 305
  Barium, Strontium (Ba, Sr) .... ~~0.02~~0.12
  Noble Metals (Ru, Rh, Pd, Mo, Tc, Co) .... ~~0.0025~~0.005
  Lanthanides (La, Zr, Nd, Eu, Nb, Pm, Pr, Sm, Y, Cm, Am) .... ~~0.0002~~0.0052
  Cerium Group (Ce, Pu, Np) .... ~~0.0005~~0.0055

The radionuclide groups and elemental release fractions listed above are consistent with the accident source term information presented in NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants ! Final Report."

The timing of the releases are based on NUREG-1465 assumptions. The release scenario assumed in the calculations is described below.

An initial release of activity from the gaps of a number of failed fuel rods at 10 minutes into the accident is considered. The release of 5 percent of the core inventory of the volatile species (defined as noble gases, halogens, and alkali metals) is assumed. The release period occurs over the next 30 minutes, that is, from 10 to 40 minutes into the accident. At this point, 5 percent of the total core inventory of volatile species has been considered to be released.

Over the next 1.3 hours, releases associated with an early in-vessel release period are assumed to occur, that is, from 40 minutes to 1.97 hours into the accident. This source term is a time-varying release in which the release rate is assumed to be constant during the duration time. Additional releases during the early in-vessel release period include 95 percent of the noble gases, 35 percent of the halogens, and 25 percent of the alkali metals, as well as the fractions of the tellurium group, barium and strontium, noble metals, lanthanides, and cerium group as listed above.

There is no additional release of activity to the containment atmosphere after the in-vessel release phase.

~~The above source terms are consistent with the guidance provided by the NRC in Regulatory Guide 1.183 for design basis accident (DBA) loss-of-coolant accident (LOCA) evaluations.~~

~~The resulting instantaneous gamma and beta dose rates are provided in Figures D-1 and D-2, respectively.~~

The ex-vessel and late in-vessel periods commence after the early in-vessel release period After 2 hours, the early in-vessel period is assumed to have completed The late in-vessel period continues for an additional 8 hours At the completion of the late in-vessel period, the total fraction of the core inventory defined above has been released to the containment atmosphere.

D.7.2 Thermal-Hydraulic Environments

Bounding severe accident environments are provided in this section. Five severe accident cases are analyzed with the MAAP4.04 code to generate the environment. The MAAP4 code input parameters are set to produce bounding cladding oxidation in each of the analyses.

The five cases are:

- IGN – DVI line break with vessel reflood, cavity flooding, and igniter
- IVR – DVI line break with cavity flooding and igniters, no vessel reflood
- NOIGN – 4-inch DVI line break with vessel reflood, cavity flooding, and no igniters
- CCI – Large LOCA with igniters, no vessel reflood and no cavity flooding
- GLOB – Global burning of hydrogen from 100-percent cladding reaction

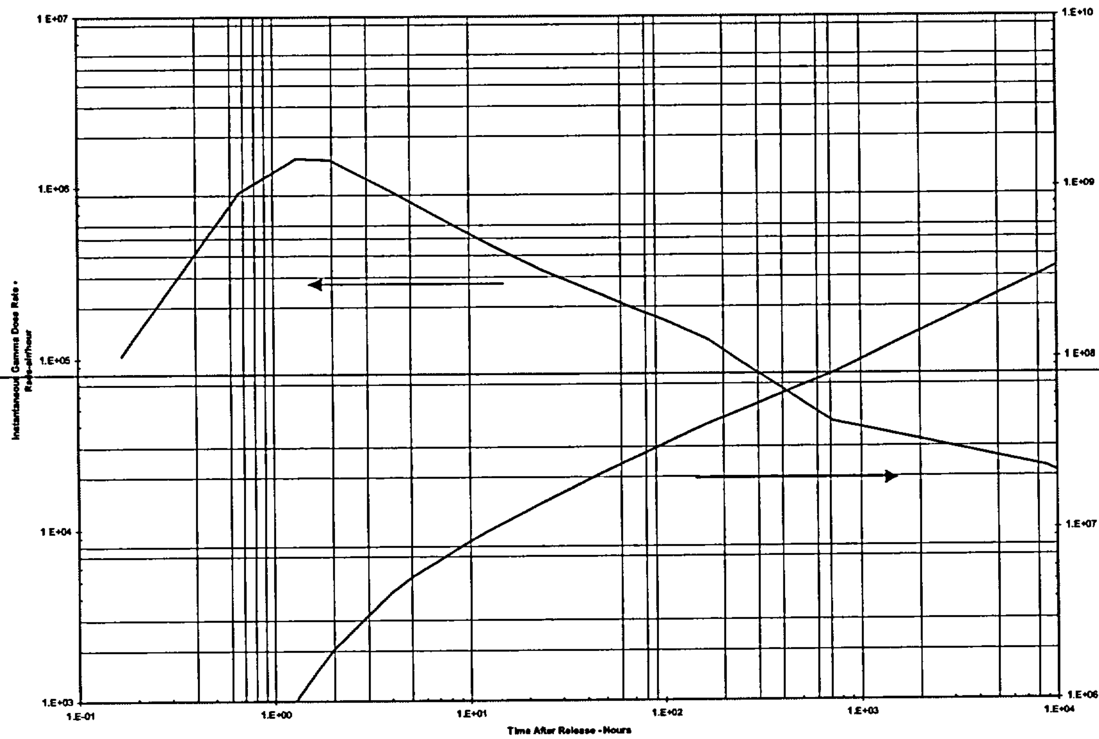
The event timing for each case is presented in Table D-6. These key events relate directly to the equipment survivability time frames.

D.7.2.1 Case IGN – Large In-Vessel Hydrogen Release Burned at Igniters

Case IGN provides a containment environment with a high rate of hydrogen generation from vessel reflooding, sustained steaming from stage 4 ADS, and hydrogen burning at the igniters. The MAAP4 results are presented in Figures D-3 through D-11.

The accident sequence is initiated by a DVI line break into a PXS compartment. The compartment floods with water from the IRWST and fills above the break elevation, allowing the vessel to reflood. Reflooding the overheated core causes a large fraction of the zirconium cladding to oxidize; however, relocation of the core to the lower plenum of the reactor vessel is prevented.

The hydrogen produced in-vessel is released to the containment through the ADS and through the break. It burns at igniters placed throughout the containment.


1 E+07
1.E+06
1.E+05
1 E+04
1.E+03
1.E-01
1.E+00
1.E+01
1.E+02
1.E+03
1.E+04
Time After Release - Hours
Instantaneous Gamma Dose Rate - Rads-air/hour
1.E+10
1.E+09
1.E+08
1.E+07
1.E+06
Integrated Gamma Dose Rate Rads-air

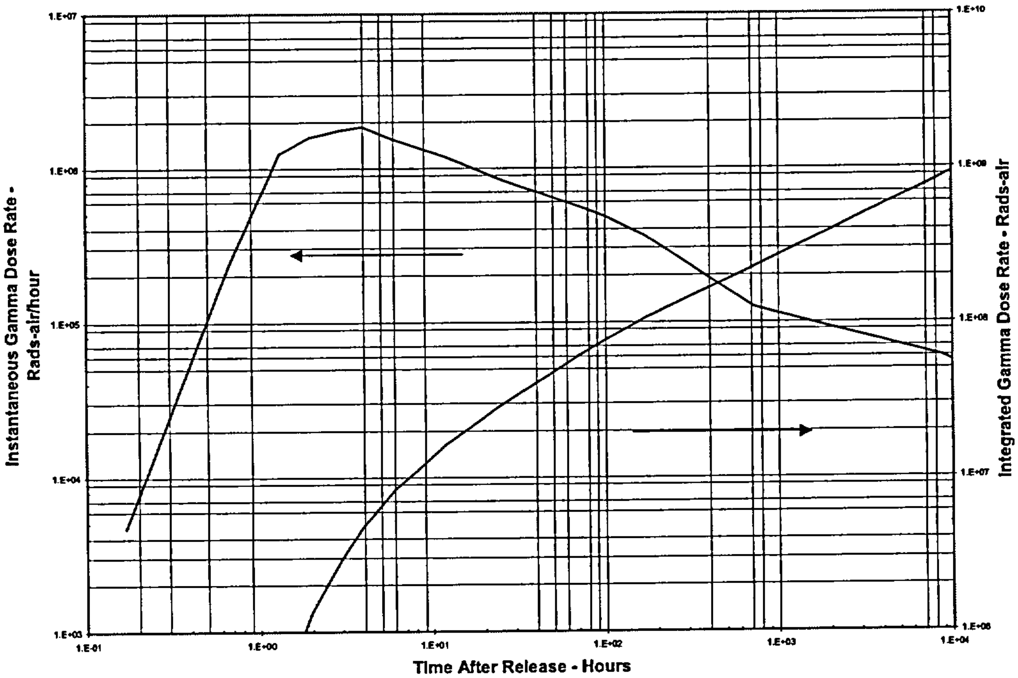
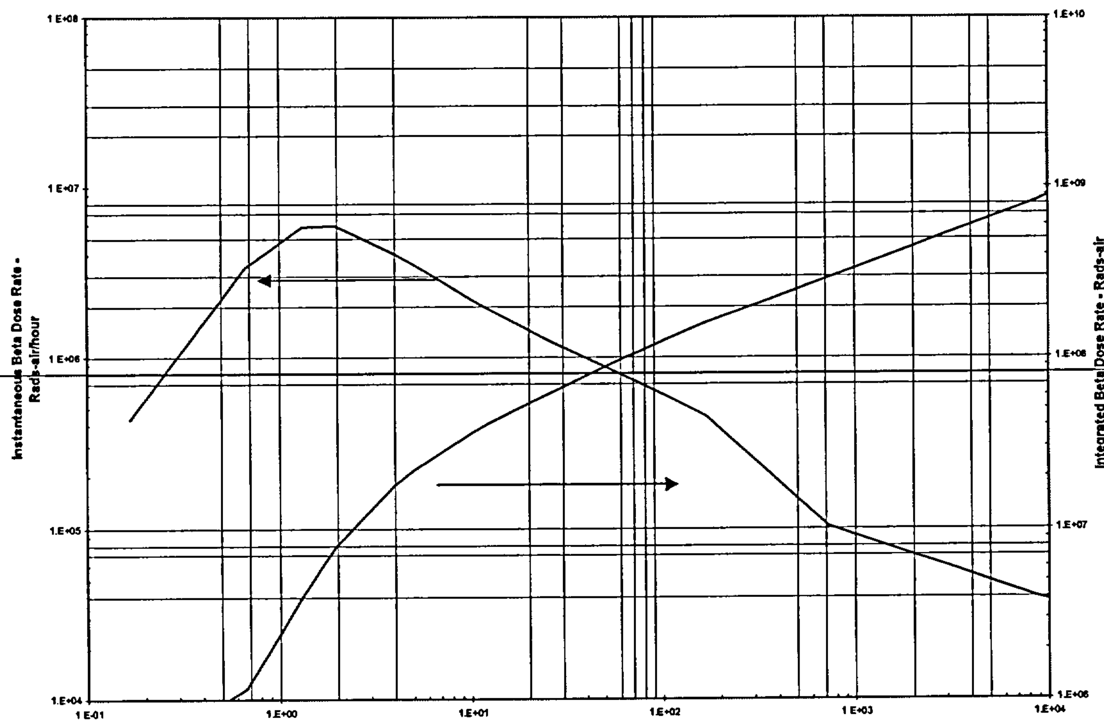


Figure D-1

Post-LOCA Gamma Dose and Dose Rate Inside Containment


1 E+08
1 E+07
1 E+06
1.E+05
1.E+04
1.E+10
1.E+09
1.E+08
1.E+07
1.E+06
1 E-01
1.E+00
1.E+01
1.E+02
1 E+03
1.E+04
Time After Release - Hours
Instantaneous Beta Dose Rate - Rads-air/hour
Integrated Beta Dose Rate - Rads-air

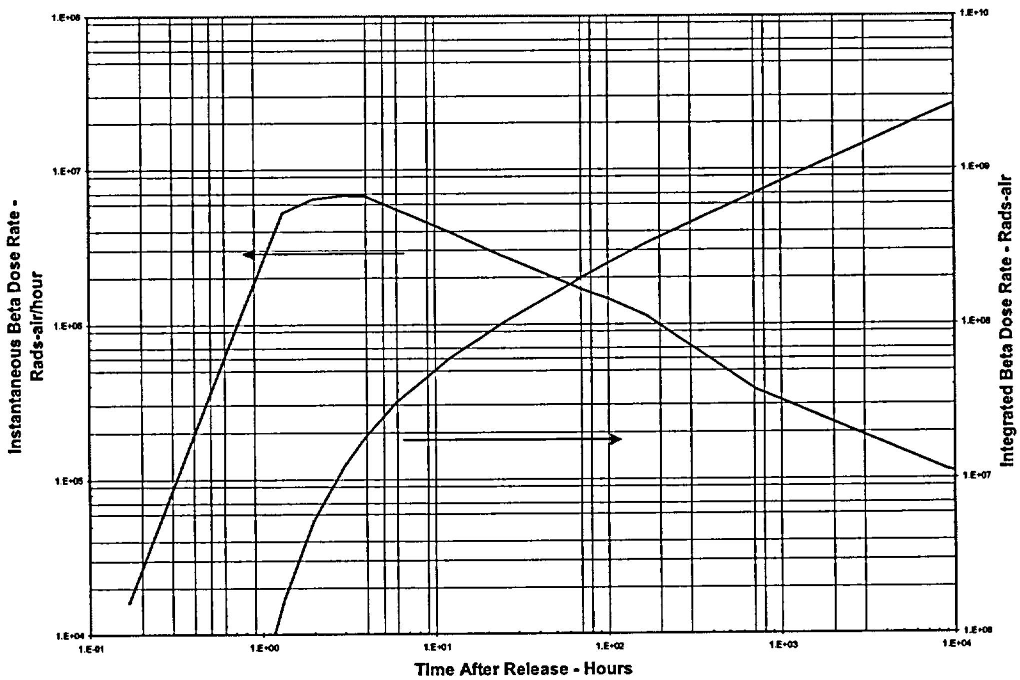


Figure D-2

Post-LOCA Beta Dose and Dose Rate Inside Containment