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Your ref: Docket No. 52-006
Our ref: DCP/NRC1557

March 26, 2003

SUBJECT: Transmittal of Westinghouse Responses to US NRC 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. A list of the RAI responses that are transmitted with this letter is provided in Attachment 1. Attachment 2 provides the RAI responses. Attachment 3 provides the reference document WCAP-15993, Revision 1, "Evaluation of the AP1000 Conformance to Inter-System Loss-of-Coolant Accident Acceptance Criteria," for RAI 440.045 Revision 1.

Please contact me if you have questions regarding this submittal.

Very truly yours,

M. M. Corletti

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

/Attachments

1. Table 1, "List of Westinghouse's Responses to RAIs Transmitted in DCP/NRC1557"

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

3. WCAP-15993, Rev. 1, "Evaluation of the AP1000 Conformance to Inter-System Loss-of-Coolant Accident Acceptance Criteria," dated March 2003

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Attachment 1

"List of Westinghouse's Responses to RAIs Transmitted in DCP/NRC1557"

Attachment 1

Table 1

"List of Westinghouse's Responses to RAIs Transmitted in DCP/NRC1557"

241.001, Rev. 1	440.045, Rev. 1
280.011, Rev. 1	440.092, Rev. 1
410.007, Rev. 2	440.106, Rev. 1
420.046, Rev. 1	720.035, Rev. 1

Attachment 2

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

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 241.001 (Response Revision 1)

Question:

The staff's review of Table 2-1 identified the following issues:

A. Shear wave velocity of 3,500 ft/sec is defined for soil. All other references to shear wave velocity refer to rock or hard rock. The DCD does not specifically clarify the definition of assumed foundation properties for the design. Please clarify your position regarding the shear wave velocity versus restriction of the AP1000 design to rock or hard rock site.

B. The "average allowable static soil bearing capacity" of 8,400 pounds-per-square foot (psf) was specified in this table. If the DCD is applicable to hard rock sites only, Westinghouse needs to demonstrate the appropriateness of this definition. In addition, it is not clear if the definition is based on an assessment of the average strength of the hard rock or if it refers to the load associated with a given relative displacement of the foundation. Please clarify.

Westinghouse Response:

A. Westinghouse is requesting design certification based on the fixed base seismic analyses. Table 5.0-1 in Tier 1 and Table 2-1 in Tier 2 will be revised to show that the shear wave velocity should exceed 8000 feet per second. Westinghouse expect that the nuclear island design using the results of the fixed-base seismic analyses will be adequate for sites with lower shear wave velocities. However, such justification is not part of the current application and may be provided as part of a Combined License application.

B. Table 2-2 of the AP600 DCD provided typical net allowable static bearing capacities for various soils. It shows an allowable bearing capacity of 220 kips per square foot for soft rock and 450 kips per square foot for hard rock. The nuclear island analyses described in Section 3.7 show that the maximum membrane vertical compression in the walls of the nuclear island is less than 200 kips per square foot (for dead, live and seismic loads). The maximum bearing reaction on the hard rock will be smaller than the compressive stress in the walls since the reactions will be distributed through the basemat which is 22 feet thick below the shield building where the maximum wall load occurs. Thus bearing strength at a hard rock site exceeds the demand. The site interface parameter for bearing capacity will be removed.

Design Control Document (DCD) Revision:
these revisions were included in DCD Revision 3

2.5.4.2 Bearing Capacity

The maximum vertical stress in the nuclear island walls is less than 200,000 pounds per square foot under all combined loads including the safe shutdown earthquake. The maximum bearing reaction on the hard rock will be smaller than the compressive stress in the walls since the reactions will be distributed through the thickness of the basemat. Bearing capacity at a hard rock site will exceed this demand.
~~The average bearing reaction of the AP1000 is about 8,400 pounds per square foot. The minimum average allowable static soil bearing capacity is 8,400 pounds per square foot over the footprint of the nuclear island at its excavation depth (see Table 2-1).~~

~~The Combined License applicant will perform field and laboratory investigations to establish the material type and the associated strength parameters in order to determine the site-specific bearing capacity value.~~

2.5.4.6.7 ~~Bearing Capacity – The Combined License applicant will verify that the site-specific soil static bearing capacity is equal to or greater than the value documented in Table 2-1 of the DCD. The Combined License applicant will verify that the dynamic site specific bearing capacity is equal or greater than the seismic bearing demand.~~**Deleted.**

Table 2-1

~~Average allowable static soil bearing capacity~~	~~Greater than or equal to 8,400 pounds per square foot over the footprint of the nuclear island at its excavation depth~~

For other related revisions see response to RAI 240.002

PRA Revision:

None

NRC Additional Comments:

The additional comments by NRC on part (A) of RAI 220.018 are also addressed in this revised response to RAI 241.001 since both RAIs relate to bearing and lift off between the nuclear island and the underlying rock.

RAI 220.018

(A) The effects of potential lift-off of the basemat on building response and floor response spectra have not been considered in the evaluation. Westinghouse agreed to consider such potential lift-off effects and perform nonlinear time history evaluations using simplified structural models of the basemat, hard rock springs and structural stick models of the NI.

RAI 241.001

Part A. The response provided by Westinghouse satisfied the staff's concern and no information is needed.

Part B. In its response, Westinghouse indicated that the design will be acceptable for hard rock having an allowable bearing capacity of 450 ksf. The staff raised a concern that this is considered an extremely high value of "allow bearing capacity", even for hard rocks, and will be difficult for the COL applicant to substantiate. The staff also identified that the response also does not indicate whether this definition refers to strength or displacement considerations. In addition, the review of the Civil/Structural Criteria document indicated that hard crystalline bedrock is to have an allowable bearing capacity of 4 ksf. Westinghouse agreed to clarify these discrepancies.

Westinghouse Response (Revision 1)

RAI 220.018 Part A

The dynamic analyses of the nuclear island for the hard rock site use a stick model on a fixed base. DCD subsection 3.8.5.4.1 is being revised as shown in this response and describes linear and non-linear static analyses of the nuclear island on soil springs with a stiffness of 6,260,000 pounds per square foot per foot, corresponding to hard rock with a shear wave velocity of 8000 fps. These analyses apply dead load and equivalent static accelerations based on the maximum accelerations from the dynamic analyses. These equivalent static accelerations are very conservative for the overturning moment in the east-west direction. The overturning moment at grade in the auxiliary and shield building stick is 34% higher in the equivalent static analyses than in the time history analyses. The non-linear analyses permit the soil springs to resist compression only and show approximately two thirds of the mat lifting off on the tension side. Results of a simplified non-linear time history analysis will be presented in the meeting in April.

241.001 Part B

The linear analyses of the nuclear island basemat on hard rock described in DCD subsection 3.8.5.1 show a maximum bearing reaction of 13 ksf at the edge of the shield building under dead and live load and 46 ksf under dead and live load combined with the SSE. The maximum bearing reactions under the auxiliary building are less than those under the edge of the shield building. The auxiliary building reactions are largest directly below the walls and are significantly lower below the six foot thick mat midway between the walls.

The non-linear analyses permit the soil springs to resist compression only and permit lift off on the tension side. The maximum bearing reaction under dead and live load combined with the SSE is 85 ksf below the shield building using conservative seismic input. The bearing strength at a hard rock site exceeds this demand.

The allowable bearing capacity of 4 ksf for hard crystalline bedrock in the Civil/Structural Criteria document is taken from Table 18-1-A of the Uniform Building Code. This value of allowable bearing pressure is for footings having a minimum width of 12 inches and a minimum depth of 12 inches into natural grade. An increase of 20 percent is allowed for each additional foot of width and/or depth to a maximum value of three times the designated value. The allowable bearing capacity of 4 ksf to 12 ksf in the Uniform Building Code is very conservative for hard rock. In Reference 1, allowable soil bearing pressures are tabulated from several American city-building codes for different soil conditions. For rock, **allowable** bearing pressures from 80 ksf to excess of 200 ksf are given (Boston, Denver, Newark, New York, Philadelphia, and New York city). Therefore, the maximum AP1000 85 ksf SSE bearing reaction on a hard rock foundation is well within these allowable values. The allowable bearing pressure provided in the Civil / Structural Design Criteria will be modified to provide more realistic guidance for non-safety related structures. In conclusion, there are local building codes that provide allowable rock bearing pressures that are in excess of the maximum AP1000 calculated SSE bearing reaction, and it is Westinghouse's opinion that it will not be difficult for the COL applicant to substantiate the acceptability of rock bearing pressure allowables in excess of 85 ksf.

Reference 1: Terzaghi, Karl and Ralph B. Peck, Soil Mechanics in Engineering Practice, John Wiley & Sons, Inc., © 1948.

Design Control Document (DCD) Revision:

Revise subsection 2.5.4.2 as follows:

2.5.4.2 Bearing Capacity

~~The maximum vertical stress in the nuclear island walls is less than 200,000 pounds per square foot under all combined loads including the safe shutdown earthquake.~~ The maximum bearing reaction on the hard rock **determined from the analyses described in subsection 3.8.5.1 is less than 85,000 pounds per square foot under all combined loads including the safe**

shutdown earthquake ~~will be smaller than the compressive stress in the walls since the reactions will be distributed through the thickness of the basemat~~. Bearing capacity at a hard rock site will exceed this demand.

Replace subsection 3.8.5.4.1 and Figure 3.8.5-2 with the following:

3.8.5.4.1 Analyses for Loads during Operation

The analyses of the basemat use the three-dimensional ANSYS finite element models of the auxiliary building and containment internal structures which are described in subsection 3.7.2.3 and shown in Figures 3.7.2-1 and 3.7.2-2. The model considers the interaction of the basemat with the overlying structures and with the soil. Provisions are made in the model for two possible uplifts. One is the uplift of the containment internal structures from the lower basemat. The other is the uplift of the basemat from the soil.

The three-dimensional finite element model of the basemat includes the structures above the basemat and their effect on the distribution of loads on the basemat. The finite element models of the auxiliary building above elevation 106' and the containment internal structures inside containment are reduced to substructures (superelements) within ANSYS. These superelements are then included in the detailed finite model of the basemat which includes the auxiliary building below elevation 106' and the mat below the containment vessel. The finite element model of the basemat is shown on sheet 1 of Figure 3.8.5-2. The model of the basemat including the superelements is shown on sheet 2.

The subgrade is modeled with one vertical spring and two horizontal springs at each node of the basemat. The vertical springs act in compression only. The horizontal springs are active when the vertical spring is closed and inactive when the vertical spring lifts off. The vertical and horizontal stiffness of the springs represent a rock foundation with a shear wave velocity of 8000 feet per second. Horizontal bearing reactions on the side walls below grade are conservatively neglected.

The nuclear island basemat below the containment vessel and the containment internal structures basemat above the containment vessel are simulated with solid tetrahedral elements. Nodes on the two basemats are connected with spring elements normal to the theoretical surface of the containment vessel.

Normal and extreme environmental loads and containment pressure loads are considered in the analysis. The normal loads include dead loads and live loads. Extreme environmental loads include the safe shutdown earthquake.

Dead loads are applied as inertia loads. Live loads and the safe shutdown earthquake loads are applied as concentrated loads on the nodes. The safe shutdown earthquake loads are applied using the assumption that while maximum response from one direction occurs, the responses from the other two directions are 40 percent of the maximum. Combinations of the three directions of the safe shutdown earthquake are considered.

Linear analyses are performed for all specified load combinations assuming that the soil springs can take tension. Critical load cases are then selected for non-linear analyses with basemat lift-off based on the results of the linear cases. The results from the analysis include forces, shears, and moments in the basemat, bearing pressures under the basemat, and the area of the basemat that is uplifted. Reinforcing steel areas are calculated from the member forces for each load combination case.

The required reinforcing steel under the shield building is determined by considering both the reinforcement envelope for the linear analyses which do not consider liftoff and the reinforcement envelope for the full non-linear iteration of the most critical load combination cases.

The required reinforcing steel for the portion of the basemat under the auxiliary building is calculated from shears and bending moments in the slab obtained from separate calculations. Beam strip models of the slab segments are loaded with the bearing pressures under the basemat from the three-dimensional finite element analyses. Figure 3.8.5-3 shows the basemat reinforcement.

Revise Table 3.8.5-3 to show reinforcement required based on basemat design calculation to be reviewed in NRC meeting on April 2, 2003.

PRA Revision:

None

Table 3.8.5-3

DEFINITION OF CRITICAL LOCATIONS AND THICKNESSES FOR NUCLEAR ISLAND BASEMAT(1) (5)

Wall or Section Description	Applicable Column Lines	Applicable Elevation Level or Elevation Level Range	Concrete Thickness(2)	Reinforcement Required Vertical (in^2/ft) (3)	Reinforcement Required Horizontal (in^2/ft) (3)	Reinforcement Provided Vertical (in^2/ft)(4)	Reinforcement Provided Horizontal (in^2/ft)(4)
Auxiliary Building Basemat							
Auxiliary Basemat Area	Column line K to L and from Col. Line 11 wall to the intersection with the shield building	From level 0 to 1	6'-0"	**Shear Reinforcement** 0.26	**Bottom Reinforcement** ~~2.7~~**1.6** (East-West Direction) **Top Reinforcement** ~~2.7~~**1.6** (East-West Direction)	**Shear Reinforcement** 0.31	**Bottom Reinforcement** 2.7 (East-West Direction) **Top Reinforcement** 2.7 (East-West Direction)
Auxiliary Basemat Area	Column line 1 to 2 and from Column Line K-2 to N wall	From level 0 to 1	6'-0"	**Shear Reinforcement** ~~0.74~~**0.34**	**Bottom Reinforcement at column line 2** ~~4.5~~**2.2** (North-South Direction) **Top Reinforcement at mid-span** ~~3.12~~**2.5** (North-South Direction)	**Shear Reinforcement** 0.78	**Bottom Reinforcement** 4.5 (North-South Direction) **Top Reinforcement** 3.12 (North-South Direction)

Notes:

1. The applicable column lines and elevation levels are identified and included in Figures 1.2-9, 3.7.2-12 (sheets 1 through 12), 3.7.2-19 (sheets 1 through 3) and on Table 1.2-1.
2. These thicknesses have a construction tolerance of +1 inch, -3/4 inch.
3. These concrete reinforcement values represent the minimum reinforcement required for structural requirements except for designed openings, penetrations, sumps or elevator pits.
4. These concrete reinforcement values represent the provided reinforcement for structural requirements except for designed openings, penetrations, sumps or elevator pits.

~~5. The results shown are representative and are based on the AP600 design. AP1000 results will be provided by the Combined License applicant.~~

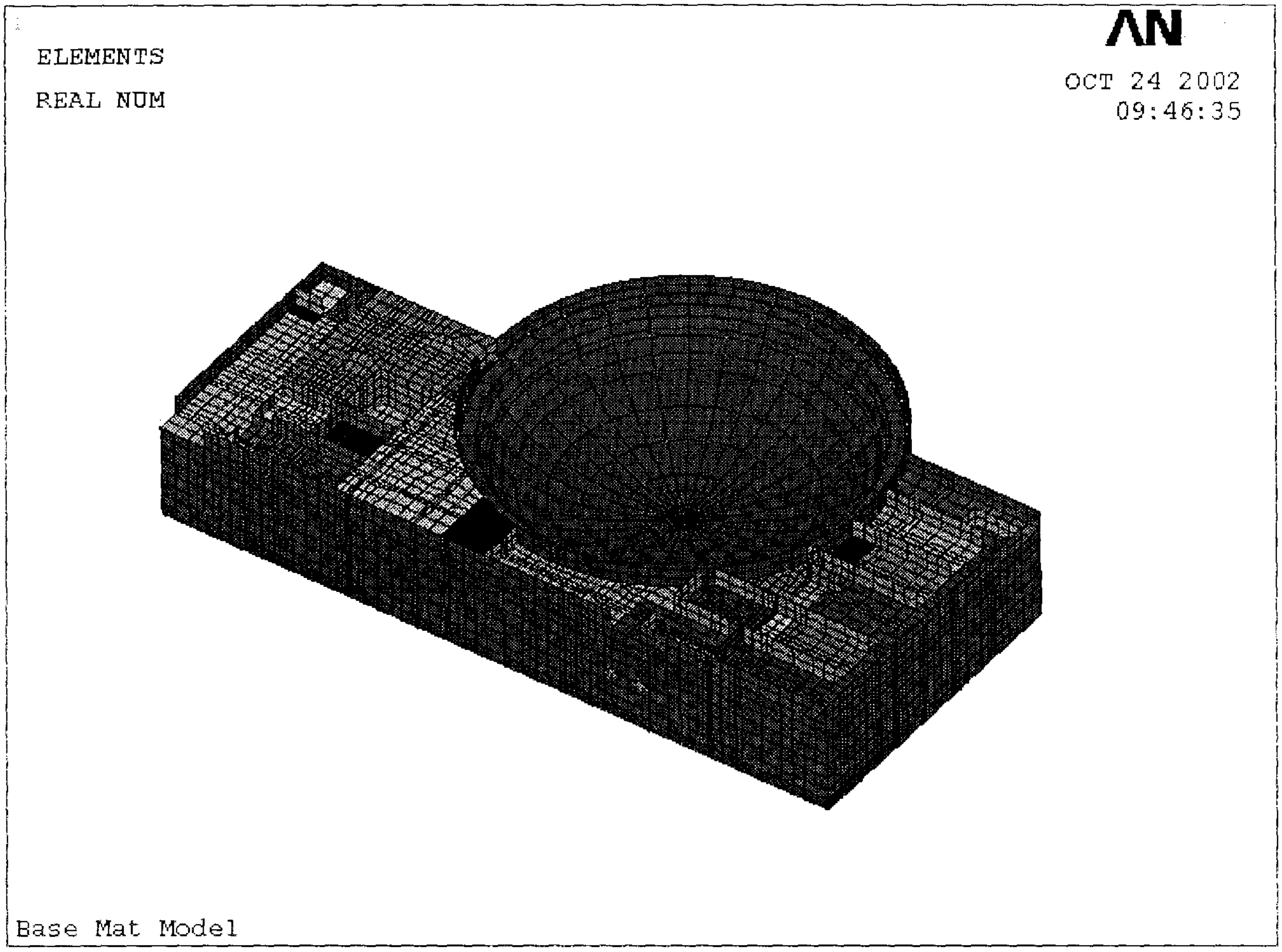


Figure 3.8.5-2 (Sheet 1 of 2)
Isometric view of finite element model

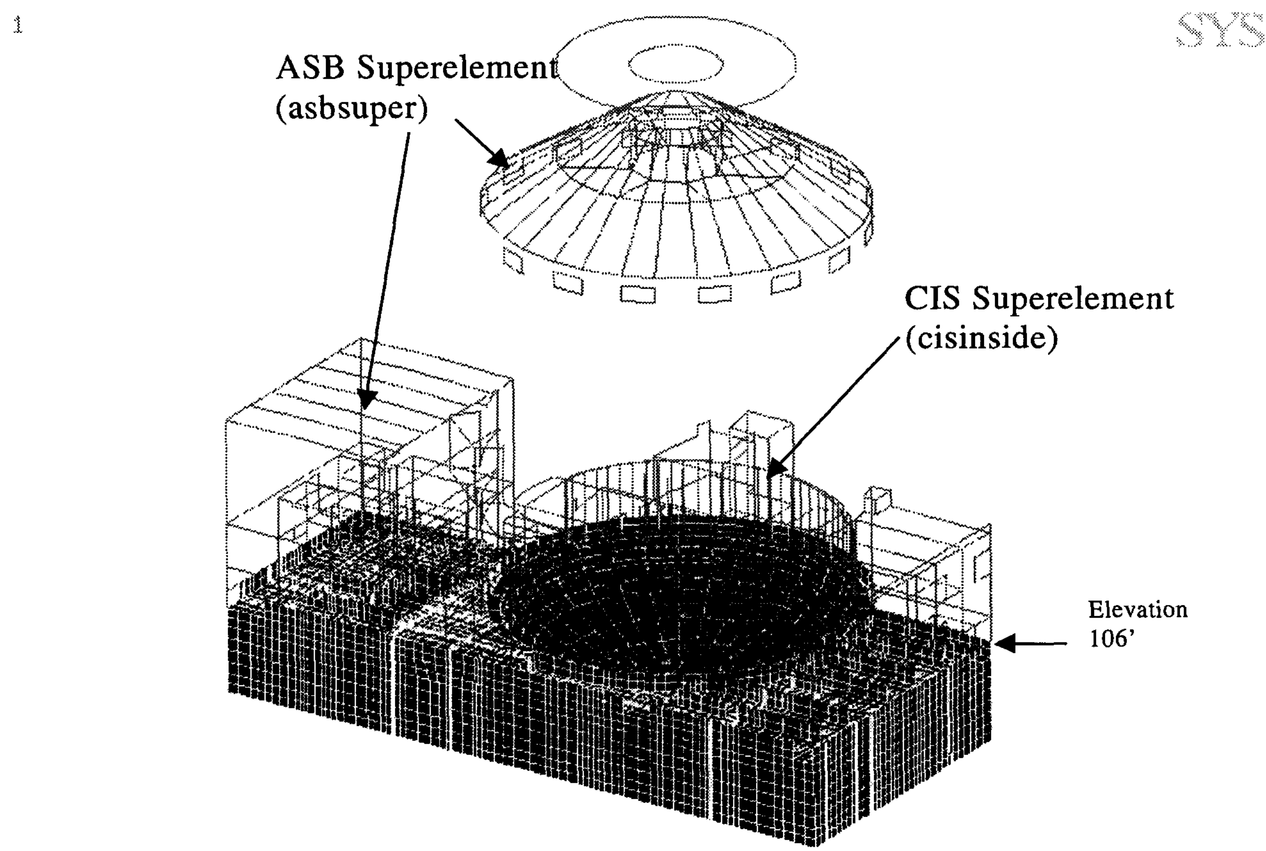


Figure 3.8.5-2 (Sheet 2 of 2)
Isometric view of finite element model including superelements

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 280.011 (Response Revision 1)

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

NRC Additional Comment:

To discuss technical concerns related to this RAI, the staff had a teleconference with Westinghouse on 12/17/02, which raised the following technical concerns.

With respect to the general design of fire areas in Containment, Westinghouse confirmed during the call that the screening criteria applied for the fire PRA assumes that if the total combustible loading in each fire area is less than 20,000 Btu/ft2, that the fire area was screened out using the FIVE methodology. On Page 5-8 of the EPRI report which explains the FIVE methodology, bullet 4 states that a fire area can be screened out if it has less then 20,000 BTU per sq. ft. Westinghouse also stated that this is considered a very low quantify of combustible loading per the NFPA Fire Protection Handbook (FPH). However, Westinghouse did not address that Page 7-78 of the FPH, 18th Edition, also states that "the original concepts of fire severity and fire load (combustible load) are very important even though they are technically obsolete." The information contained in the FPH regarding combustible loading was first published in 1997, and the FIVE methodology, which makes use of the "combustible loading" concept was published as a final in 1992.

Combustible load is a measure of the maximum heat that would be released if all the combustibles in a given fire area burned and does not consider other factors such as heat release rate (HRR), room configuration, ventilation rate, or other parameters which describe the fire dynamics over a period of time. The National Institute of Standards and Technology (NIST) Technical Report NISTIR 5842NISTIR 5842, "Methodology for Developing and Implementing Alternative Temperature-Time Curves for Testing the Fire Resistance of Barriers for Nuclear Power Plant Applications," by Cooper, L., and Steckler, K. May 1996, page 3., page ix, also identifies that the technical shortcomings of this method are the following:

I. No technical basis for the equal-area hypothesis. The equal-area hypothesis is that the area beneath a temperature-time curve is a measure of the intensity or severity of a fire, and all fires with equal-area exposures are equally severe.

II. Real room fire intensities are not a sole function of fire (combustible) load

III. Temperatures of real fires can rise much faster then the standard time-temperature curve ASTM E 119-98, "Standard Test Methods for Fire Tests of Building Construction and Materials," ASTM Fie Test Standard, Fifth Edition, American Society of Testing and Materials, West Conshohocken, PA, 1999, pp 793-813.

NISTIR 5842, page ix, also states that the NFPA acknowledges that the fire load method is technically obsolete. Westinghouse stated during the call that NFPA 805, which is the latest industry performance-based standard for fire protection endorsed by the NRC, permits the use of the FIVE methodology. FIVE was approved by the NRC in the early 1990's primarily as a tool to provide a qualitative assessment of fire risk for the IPEEE to perform fire PRAs.

Appendix Section C.2.2., "Fire Model Features and Limitation" of NFPA 805 specifically states that the limitations of each fire model should be taken into consideration in order to produce reliable results that will be useful in decision making. This section specifically states that "Some models may not be appropriate for certain conditions and can produce erroneous results if applied incorrectly." The intent of the Appendix C, Table C.2.2.(b) which lists all of the fire models, is to compare the features available in each mathematical model. This enables the user to select the appropriate model for a particular fire area, in order to obtain useful estimates to best approximate the conditions within an enclosure as a result of an internal fire. Table C.2.2.(b) of NFPA 805, compares the following features available for ten different mathematical models:

I. What type of program is it (Zone, CFD, Network Flow)
II. Number of rooms that can be modeled
III. Wall heat transfer
IV. Lower Level Gas Temperature
V. Heat Targets
VI. Fire
VII. Gas Concentrations
VIII. Oxygen depletion
IX. Vertical connections
X. HVAC Fans and Ducts

The staff notes the following technical concerns with the use of the FIVE methodology for the Containment area:

I. Section 2.0, "Definitions," of the FIVE report provide definitions for "fire area boundary." Typically a fire area boundary is completely sealed with floor-to-ceiling and/or wall-to-wall fire barriers. The FIVE methodology is limited in that large open areas, such as those in containment, are not capable of being realistically modeled. The AP1000 DCD identifies that there are open areas in containment, specifically for 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. The safe shutdown evaluation provided in the DCD for these zones discuss the migration of hot gases beyond the area of fire origin and make deterministic assumptions that a fire will not propagate beyond the zone without technical justification. Hot gases and flames could also damage seals in the area of fire origin which would open a path for propagation to adjacent fire zones. However, without the proper selection of a fire model which allows the user to input more realistic data to estimate fire growth, Westinghouse may not have realistically demonstrated that propagation will not occur within certain zones in Containment, on the basis of "combustible load" assumptions. The state-of-the-art for fire protection has increased since the development of FIVE, and where practical mathematical fire modeling should be used to reduce unnecessary conservatism.

Furthermore, selection of a fire model solely on the basis that it is allowed by NFPA 805, without analyzing the limitations of each fire model for certain conditions could produce erroneous and unreliable results. Please note that NFPA 805 does not recommend any specific fire model over another. In fact, it only states that the limitations of each fire model should be taken into consideration in order to produce reliable results that will be useful in decision making. Using the FIVE methodology, Westinghouse has screened out areas in Containment on the basis of the combustible loading concept when other computer fire modeling techniques are available, which allow the user to input more useful data to make realistic determinations regarding fire growth and smoke propagation. <u>In light of the limitations noted with the combustible loading method and the inability of FIVE to model large, open areas, the staff requests that Westinghouse address the appropriateness of the FIVE methodology to screen out large open areas such as Containment for the AP1000 Fire Protection review.</u>

Westinghouse Additional Response:

Based upon a subsequent teleconference with NRC staff to clarify the nature and extent of the additional comment, its scope was reduced to the portion <u>underlined</u> above. In addition, it was agreed that the original additional comment contained elements of both deterministic and probabilistic fire analyses and that Westinghouse need only address the use of the FIVE methodology relative to the probabilistic fire analysis.

Westinghouse considers the FIVE Methodology appropriate for inside containment and did not screen out Containment from its analysis. The FIVE Methodology states in Chapter 6.3.1 p.6-6 that:

> "Containment fires are not considered in the FIVE fire frequency data table because:
>
> - the small number of events, and
> - previous fire PRAs did not show that Containment fires are risk-significant. Also, most of the other fires occurred during shutdown not during plant operation.

However, plant-unique features of various designs may not provide the same level of protection against fire as those plants previously examined. Thus, at least a qualitative assessment should be performed in order to determine if containment needs to be analyzed in the more detailed manner described by FIVE for other plant compartments. For example, consideration should be given to conducting an analysis if: (1) the plant experience indicates that fires in containment during power operation have occurred on a recurring basis; and (2) redundant trains of critical equipment within containment might be exposed to the same fire plume or be in a confined space and susceptible to damage by a hot gas layer."

Westinghouse did perform a fire PRA evaluation at power for inside the AP1000 containment, using the same assumptions as for outside the containment in the same way it was done for the AP600. Thus, Westinghouse performed more than a qualitative analysis of the fire risk inside the containment. It is a quantitative evaluation based on reasonable assumptions:

- Westinghouse assumed the probability of propagation from one zone to another (0.01) is similar to area propagation outside the containment for the zones with a combustible loading above 20000 Btu/sqft. It is based on the design that includes distances and barriers between the different combustible loadings to avoid the propagation from zone to zone.
- Westinghouse also assumed that all the components in the zone with a fire fail.
- No propagation was assumed from one zone that has a combustible loading under 20000 Btu/sqft.

But because the second assumption may be too conservative, Westinghouse also studied in more detail what components would fail or not for the following zones: 1100 AF 11300B and 1100 AF 11500.

The PRA chapter (57.4.2.2- last sentence) will be corrected accordingly.

Use of 20000 Btu/sqft as assumption limit for propagation outside the containment:

Westinghouse did for AP1000 exactly what was done for the AP600.

The FIVE Methodology also states in Chapter 5.3.6 p.5-6 that:

"A common Boundary can be evaluated and screened from further consideration based on the following (extract) criteria:

- Boundaries where the exposing compartment has a very low combustible loading < 20000 Btu/sqft, and automatic fire detection on the basis that manual suppression will prevent fire spread to the adjacent compartment."

This statement is applicable to all kinds of physical boundaries, as well for immaterial boundaries (distance). Outside containment, there are only 2 or 3-hour rated barriers as boundaries of the fire areas. Then it is reasonable to assume that there is no possible propagation from one area with a combustible loading under 20000 Btu/sqft.

The PRA chapter (57.4.2.1) will be corrected accordingly:
"According to the FIVE methodology assumptions, we do not credit any propagation in case of a combustible loading under 20000 Btu/sqft."

Clarification about the propagation in case of high combustible loading:

The FIVE Methodology states in Chapter 5.3.6 p.5-6 that:

"A common Boundary can be evaluated and screened from further consideration based on the following (extract) criteria:

- Boundaries that consist of a 2-hour or 3-hour rated barrier on the basis of barrier effectiveness
- Boundaries that consist of a 1-hour rated fire barrier with a combustible loading in the exposing compartment <80000 Btu/sqft"

Because there is no other superior limit, Westinghouse conservatively assumes that if the combustible loading in an area is above 80000 Btu/sqft and if the automatic fire suppression fails, then the propagation is credited with a probability of 1.

This assumption impacts the areas 2003 AF 01, 2040 AF 01, 2050 AF 01, 4035 AF 01 (area screened out because propagation may occur only to areas without PRA credited systems), 6030 AF 03 and 6030 AF 04.

The PRA chapter (57.4.2.1- last sentence) will be corrected accordingly:
"A fire barrier is conservatively not credited in case of a highly combustible loading (>80000 Btu/sqft) and if the automatic fire suppression fails."

The Westinghouse use of the FIVE Methodology to evaluate potential fires inside containment is an appropriate method.

Design Control Document (DCD) Revision:

None

PRA Revision:

57.4.2.1 Outside the Containment

According to the FIVE **methodology assumptions, we do not credit any propagation in case of a combustible loading under 20000 Btu/sqft.**

Fire doors, piping or cable penetrations, and ventilation ducts are major fire propagation pathways. To assign a failure probability for any given fire barrier, generic failure data pertaining to the fire barrier elements and relevant plant-specific data (for example, number of doors) have to be known. The generic failure data can be obtained from many sources including NUREG/CR-4840 (Reference 4), which is presented in Table 57-5.

The barrier-specific failure probability can be obtained by determining the total number of each element in the barrier, multiplying by the corresponding failure probability, and summing the contributions from different elements. However, according to the assumptions made in AP600 and also applicable to AP1000, the total failure probability of a barrier (independent of the type of element and number of elements in each barrier) is assumed to be 0.01. This is considered to be a realistic value since, as presented in Table 57-5, failure probability of a barrier is dominated by the fire door failure probability. It is also a conservative value for fire barriers without doors.

Unlike early fire door designs, which contribute to the NUREG/CR-4840 data, the AP1000 fire doors are designed to have alarms that annunciate in the control room if they were to be left open. Thus, for the AP1000 design, the probability of a fire door being left open, facilitating fire propagation is expected to be less than 7.40E-3. However, in some cases, after a specific examination (showing, for example, there is no penetration, no door in the fire barrier), it may be assumed that there is no possibility of propagation through a fire barrier.

A fire barrier is not credited in case of a highly combustible loading (>80000 Btu/sqft) and/~~or~~ if the automatic fire suppression system fails.

57.4.2.2 In the Containment

The combustible loading in the containment zones are generally very low. The design philosophy is to avoid propagation by having a certain distance between combustible materials. In consequence, it is assumed:

- No propagation from one zone to another zone in case of combustible loading under 20000 Btu
- A propagation with a probability of 0.01 from the two zones that have a combustible loading above 20000 Btu to all the adjacent areas~~.~~ is **similar to area propagation outside the containment for the zones with a combustible loading above 20000 Btu/sqft. It is**

based on the design that includes distances between the different combustible loadings to avoid the propagation.

- All the components in the zone fail.

Because the last assumption may be too conservative, we also studied in more details what components would fail or not for the following zones: 1100 AF 11300B and 1100 AF 11500.

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 410.007 (Response Revision 2)

Question:

(DCD, Tier 2, Section 6.4, 9.4. through 9.4.3 and 9.4.6 through 9.4.11) The required aspects of a control room for nuclear power reactors are documented in 10 CFR Part 50, Appendix A,"General Design Criteria for Nuclear Power Plants." GDC 19, "Control Room," requires that a control room be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions.

Section 6.4.5.4 states that "[t]esting for main control room in-leakage during VES [main control room emergency habitability system] operation will be conducted once every 10 years. This testing will be conducted in accordance with ASTM [American Society for Testing and Materials] E741, 'Standard Test Method for Determining Leakage Rate by Tracer Dilution'." The staff is currently working with the industry to address control room habitability issues including air in-leakage testing. It is anticipated that the testing frequency will be on the order of 5 to 6 years. The staff expects that testing requirements for the AP1000 design will be consistent with the resolution of the control room habitability issues currently pursued by the industry and the staff. Therefore, the AP1000 design should include a commitment to resolving the in-leakage testing in accordance with the anticipated outcome of the joint effort between the NRC staff and industry. Please provide such a commitment and revise Section 6.4.8 to add the ASTM E741 standard.

In addition, consistent with the SRP, Westinghouse should commit to complying with the guidance contained in the latest versions of RG 1.52, "Design, Testing, and Maintenance for Post-Accident Engineered-Safety-Feature Atmospheric Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants," and RG 1.140, "Design, Testing, and Maintenance Criteria for Normal Ventilation Exhaust System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants."

Westinghouse Response:

Westinghouse recognizes that the NRC staff and the industry are working on in-leakage testing, however it is not reasonable to commit to a standard that does not currently exist. Westinghouse therefore is not providing a commitment to have the Main Control Room Emergency Habitability System (VES) meet the anticipated requirements currently being pursued. The VES design addresses in-leakage and meets the codes and standards that were in effect six months prior to the date of the AP1000 design certification application (March 28, 2002).

Westinghouse is revising the DCD, subsection 6.4 to include ASTM E741.

Westinghouse is including Regulatory Guide (RG) 1.140 (Rev. 2, 06/2001) in the DCD in subsections 1A, 3.2 and 9.4. Please see the corresponding DCD revisions below.

RG 1.52, is not applicable to the AP1000 as the AP1000 has no safety-related air filtration systems.

Design Control Document (DCD) Revision:

- ***Changes to DCD 6.4:***

6.4.5.1 Preoperational Inspection and Testing

Preoperational testing of the main control room emergency habitability system is performed to verify that the air flow rate of 65 ±5 scfm is sufficient to maintain pressurization of the main control room envelope of at least 1/8-inch water gauge with respect to the adjacent areas. The positive pressure within the main control room is confirmed via the differential pressure transmitters within the control room. The installed flow meters are utilized to verify the system flow rates. The pressurization of the control room limits the ingress of radioactivity to maintain operator dose limits below regulatory limits. Air quality within the MCR environment is confirmed to be within the guidelines of Table 1 and Appendix C, Table C-1, of Reference 1 by analyzing air samples taken during the pressurization test.

The storage capacity of the compressed air storage tanks is verified to be in excess of 314,132 scf of compressed air at a minimum pressure of 3400 psig. This amount of compressed air will assure 72 hours of air supply to the main control room.

An inspection will verify that the heat loads within the rooms identified in Table 6.4-3 are less than the specified values.

Preoperational testing of the main control room isolation valves in the nuclear island nonradioactive ventilation system is performed to verify the leaktightness of the valves.

Preoperational testing for main control room inleakage during VES operation will be conducted in accordance with ASTM E741, ~~"Standard Test Method for Determining Air Leakage Rate by Tracer Dilution."~~**(Reference 4).**

Testing and inspection of the radiation monitors is discussed in Section 11.5. The other tests noted above are discussed in Chapter 14.

6.4.5.4 Air Inleakage Testing

Testing for main control room inleakage during VES operation will be conducted once every ten years. This testing will be conducted in accordance with ASTM E741, ~~"Standard Test Method for Determining Leakage Rate by Tracer Dilution."~~**(Reference 4).**

6.4.8 References

1. "Ventilation for Acceptable Indoor Air Quality," ASHRAE Standard 62 - 1989.
2. "Human Engineering Design Guidelines," MIL-HDBK-759C, 31 July 1995.
3. "Human Engineering," MIL-STD-1472E, 31 October 1996.
4. **"Standard Test Methods for Determining Air Change in a Single Zone by Means of a Tracer Gas Dilution," ASTM E741, 2000**

- ***Changes to DCD compliance table for RG 1.140 in DCD Appendix 1A. The following <u>replaces</u> the existing compliance:***

APPENDIX 1A

CONFORMANCE WITH REGULATORY GUIDES

Criteria Section	Referenced Criteria	AP1000 Position	Clarification/Summary Description of Exceptions
DIVISION 1 – Power Reactors			
Reg. Guide 1.140, Rev. 2, 06/01 - Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Normal Atmosphere Cleanup System in Light-Water-Cooled Nuclear Power Plants			
C.1		Conforms	Regulatory Guide 1.140 endorses ASME Standard N509-1989 (Reference 39), ASME Standard N510-1989 (Reference 40) and ASME AG-1-1997 (Reference 38). The AP1000 uses the latest version of the industry standards (as of 3/2002).
C.2.1-2.4		Conforms	

Criteria Section Exceptions	Referenced Criteria	AP1000 Position	Clarification/Summary Description of
C.3.1-3.2		Conforms	
C.3.3	ERDA 76-21, Section 5.6; ASME N509-1989 Section 4.9	Conforms	
C.3.4	Regulatory Guide 8.8	Conforms	
C.3.5		Conforms	
C.3.6	ASME AG-1-1997 Article SA-4500	Exception	Exhaust ductwork upstream of the containment air filtration system exhaust filters that has a negative operating pressure are designed to meet at least SMACNA design standards.
	ASME AG-1-1997, Section TA	Conforms	
C.4.1	ASME AG-1-1997, Section FB	Conforms	
C.4.2	ASME AG-1-1997, Section CA	Conforms	
C.4.3	ASME AG-1-1997, Section FC, and Section TA	Conforms	
C.4.4	ASME AG-1-1997, Section FG	Conforms	
C.4.5	ERDA 76-21, Section 4.4; ASME AG-1a-2000, Section HA	Conforms	

Criteria Section Exceptions	Referenced Criteria	AP1000 Position	Clarification/Summary Description of
C.4.6	ASME N509-1989, Section 5.6; ASME AG-1a-2000, Section HA	Conforms	
C.4.7	ASME AG-1-1997, Section CA	Conforms	
C.4.8	ASME AG-1-1997, Section FD or FE	Conforms	
C.4.9	ASME AG-1-1997, Section FD and FE or, Section FF	Conforms	
C.4.10	ASME AG-1-1997 Section SA	Exception	Exhaust ductwork upstream of the containment air filtration system exhaust filters that has a negative operating pressure are designed to meet at least SMACNA design standards.
C.4.11		Conforms	
C.4.12	ASME AG-1-1997 Section DA	Conforms	
C.4.13	ASME AG-1-1997, Section BA and SA	Conforms	
C.5.1	ERDA 76-21, Section 2.3.8; ASME AG-1a-2000, Section HA	Conforms	
C.5.2		Conforms	
C.6	ASME N510-1989	Conforms	
C.7	ANSI N509-1989	Conforms	

- ***Add new reference to DCD Appendix 1A:***

1A.1 References

38. ASME AG-1-1997, "Code on Nuclear Air and Gas Treatment" 1997

- ***Changes to DCD 3.2:***

1. Changes to Subsection 3.2.6 References

18. ASME/ANSI ~~N509-89~~**AG-1-1997**, "**Code on Nuclear Air and Gas Treatment**~~Nuclear Power Plant Air Cleaning Units and Components~~."

2. Changes to Table 3.2-3 sheet 54

Table 3.2-3 (Sheet 54 of 67)

AP1000 CLASSIFICATION OF MECHANICAL AND FLUID SYSTEMS, COMPONENTS, AND EQUIPMENT

Tag Number	**Description**	**AP1000 Class**	**Seismic Category**	**Principal Construction Code**
Nuclear Island Nonradioactive Ventilation System (VBS) (Continued)				
n/a	MCR/TSC Supplemental Air Filtration Units	Note 2	NS	ASME ~~N509~~**AG-1, Note 4**

3. Changes to DCD Table 3.2-3 sheet 60

Table 3.2-3 (Sheet 60 of 67)

AP1000 CLASSIFICATION OF MECHANICAL AND FLUID SYSTEMS, COMPONENTS, AND EQUIPMENT

Tag Number	Description	AP1000 Class	Seismic Category	Principal Construction Code
Containment Air Filtration System (Continued)				
n/a	Air Exhaust Filtration Units	R	NS	ASME ~~N509~~**AG-1, Note 4**
n/a	Fans, Ductwork	L or R	NS	SMACNA or ASME ~~N509~~**AG-1, Note 4**

4. Changes to Notes as the end of DCD Table 3.2-3

Notes:

1. Component performs a safety-related function equivalent to AP1000 equipment Class C. The component is constructed using the standards for Class R and a quality assurance program in conformance with 10 CFR Part 50 Appendix B.
2. Component performs an AP1000 equipment Class D function and is constructed using the standards for Class L or Class R.
3. Fire dampers are constructed to the requirements of UL-555 or UL-555S if they are fire and smoke dampers and are located in Class D, Class L, and Class R ducts.
4. **Construction is non-seismic and meets applicable portions of ASME AG-1 consistent with RG 1.140.**

- ***Changes to Section 9.4***

9.4.1.1.1 Safety Design Basis

The nuclear island nonradioactive ventilation system provides the following nuclear safety-related design basis functions:

- Monitors the main control room supply air for radioactive particulate and iodine concentrations

- Isolates the HVAC penetrations in the main control room boundary on high-high particulate or iodine concentrations in the main control room supply air or on extended loss of ac power to support operation of the main control room emergency habitability system as described in Section 6.4

Those portions of the nuclear island nonradioactive ventilation system which penetrate the main control room envelope are safety-related and designed as seismic Category I to provide isolation of the main control room envelope from the surrounding areas and outside environment in the event of a design basis accident. Other functions of the system are nonsafety-related. HVAC equipment and ductwork whose failure could affect the operability of safety-related systems or components are designed to seismic Category II requirements. The remaining portion of the system is nonsafety-related and nonseismic. The equipment is procured to meet the environmental qualifications used in standard building practice.

The nuclear island nonradioactive ventilation system is designed to control the radiological habitability in the main control room within the guidelines presented in Standard Review Plan (SRP) 6.4 and NUREG 0696 (Reference 1), if the system is operable and ac power is available.

Portions of the system that provide the defense-in-depth function of filtration of main control room/technical support center air during conditions of abnormal airborne radioactivity are designed, constructed, and tested to conform with Generic Issue B-36, as described in Section 1.9 and Regulatory Guide 1.140 (Reference 30), as described in Appendix 1A, **and the applicable portions of ASME AG-1 (Reference 36),** ASME N509 (Reference 2) and ASME N510 (Reference 3).

9.4.1.2.2 Component Description

The nuclear island nonradioactive ventilation system is comprised of the following major components. These components are located in buildings on the Seismic Category I Nuclear Island and the Seismic Category II portion of the annex building. The seismic design classification, safety classification and principal construction code for Class A, B, C, or D components are listed in Section 3.2. Tables 9.4.1-1, 9.4.1-2 and 9.4.1-3 provide design parameters for major components in each subsystem.

Supply Air Handling Units

Each air handling unit consists of a mixing box section, a low efficiency filter bank, high efficiency filter bank, an electric heating coil, a chilled water cooling coil bank, and supply and return/exhaust air fans.

Supply and Return/Exhaust Air Fans

The supply and return/exhaust air fans are centrifugal type, single width single inlet (SWSI) or double width double inlet (DWDI), with high efficiency wheels and backward inclined blades to produce non-overloading horsepower characteristics. The fans are designed and rated in accordance with ANSI/AMCA 210 (Reference 4), ANSI/AMCA 211 (Reference 5) and ANSI/AMCA 300 (Reference 6).

Ancillary Fans

The ancillary fans are centrifugal type with non-overloading horsepower characteristics. Each can provide a minimum of 1,530 cfm. The fans are designed and rated in accordance with ANSI/AMCA 210 (Reference 4), ANSI/AMCA 211 (Reference 5), and ANSI/AMCA 300 (Reference 6).

Supplemental Air Filtration Units

Each supplemental air filtration unit includes a high efficiency filter bank, an electric heating coil, a charcoal adsorber with upstream HEPA filter bank, a downstream postfilter bank and a fan. The filtration unit configurations, including housing, internal components, ductwork, dampers, fans and controls, and the location of the fans on the filtered side of units are designed, constructed, and tested to meet the **applicable** performance requirements of **ASME AG-1,** ASME N509 and ASME N510 (References **36,** 2 and 3) to satisfy the guidelines of Regulatory Guide 1.140 (Reference 30).

Low Efficiency Filters, High Efficiency Filters, and Postfilters

The low efficiency filters and high efficiency filters have a rated dust spot efficiency based on ASHRAE 52 and 126 (References 7 and 35). Filter minimum average dust spot efficiency is shown in Table 9.4.1-1 and 9.4.1-2. High efficiency filter performance upstream of HEPA filter banks meet the design requirements of ASME ~~N509~~**AG-1** (Reference ~~2~~**36**), Section ~~5.3~~**FB**. Postfilters downstream of the charcoal filters have a minimum DOP efficiency of 95 percent. The filters meet UL 900 (Reference 8) Class I construction criteria.

HEPA Filters

HEPA filters are constructed, qualified, and tested in accordance with UL-586 (Reference 9) and ASME ~~N509~~**AG-1** (Reference ~~2~~**36**), Section ~~5.1~~**FC**. Each HEPA filter cell is individually shop tested to verify an efficiency of at least 99.97 percent using a monodisperse 0.3-μm aerosol **in accordance with ASME AG-1 (Reference 36), Section TA.**

Charcoal Adsorbers

Each charcoal adsorber is designed, constructed, qualified, and tested in accordance with ASME ~~N509~~**AG-1 (Reference 36),** Section ~~5.2~~**FE**~~; ASME 510, Sections 11, 12, and 16~~; and Regulatory Guide 1.40. Each charcoal adsorber is a single assembly with welded construction and 4-inch deep Type III rechargeable adsorber cell, conforming with IE Bulletin 80-03 (Reference 29).

Electric Heating Coils

The electric heating coils are multi-stage fin tubular type. The electric heating coils meet the requirements of UL-1995 (Reference 10). The coils for the supplemental air filtration subsystem are constructed, qualified, and tested in accordance with ASME ~~N509~~ **AG-1** (Reference ~~2~~**36**), Section ~~5.5~~**CA**.

Electric Unit Heaters

The electric unit heaters are single-stage or two-stage fin tubular type. The electric unit heaters are UL-listed and meet the requirements of UL-1996 (Reference 26) and the National Electrical Code NFPA 70 (Reference 28).

Cooling Coils

The chilled water cooling coils are counterflow, finned tubular type. The cooling coils are designed and rated in accordance with ASHRAE 33 (Reference 11) and ANSI/ARI 410 (Reference 12).

Humidifiers

The humidifiers are packaged electric steam generator type which converts water to steam and distributes it through the air handling system. The humidifiers are designed and rated in accordance with ARI 620 (Reference 13).

Isolation Dampers and Valves

Nonsafety-related isolation dampers are bubble tight, single- or parallel-blade type. The isolation dampers have spring return actuators which fail closed on loss of electrical power. The isolation dampers are constructed, qualified, and tested in accordance with ANSI/AMCA 500 (Reference 14) or ASME ~~N509~~ **AG-1** (Reference ~~2~~**36**), Section ~~5.9~~**DA**.

The main control room pressure boundary penetrations include isolation valves, interconnecting piping, and vent and test connection with manual test valves. The isolation valves are classified as Safety Class C (see subsection 3.2.2.5 and Table 3.2-3) and seismic Category I. Their boundary isolation function will be tested in accordance with ASME N510 (Reference 3).

The main control room pressure boundary isolation valves have electro-hydraulic operators. The valves are designed to fail closed in the event of loss of electrical power. The valves are qualified to shut tight against control room pressure.

Tornado Protection Dampers

The tornado protection dampers are split-wing type and designed to close automatically. The tornado protection dampers are designed against the effect of 300 mph wind.

Shutoff, Balancing and Backdraft Dampers

Multiblade, two-position remotely operated shutoff dampers are parallel-blade type. Multiblade, balancing dampers are opposed-blade type. Backdraft dampers are of the counterbalanced type and are provided to delay smoke migration through ductwork in case of fire. The backdraft dampers meet the Leakage Class II requirements of ASME N509 (Reference 2). Air handling unit and fan shutoff dampers are designed for maximum fan static pressure at shutoff flow and meet the performance requirements in accordance with ANSI/AMCA 500 (Reference 14). The supplemental air filtration subsystem dampers are constructed, qualified, and tested in accordance with ANSI/AMCA 500 or ASME ~~N509~~**AG-1** (Reference ~~23~~**36**), Section ~~5.9~~**DA**.

Combination Fire/Smoke Dampers

Combination fire/smoke dampers are provided at duct penetrations through fire barriers to maintain the fire resistance ratings of the barriers. The combination fire/smoke dampers meet the design, leakage testing, and installation requirements of UL-555S (Reference 25).

Ductwork and Accessories

Ductwork, duct supports, and accessories are constructed of galvanized steel. Ductwork subject to fan shutoff pressures is structurally designed to accommodate fan shutoff pressures. Ductwork, supports, and accessories meet the design and construction requirements of SMACNA Industrial Rectangular and Round Duct Construction Standards (References 16 and 34) and SMACNA HVAC Duct Construction Standards – Metal and Flexible (Reference 17). The supplemental air filtration and main control room/technical support center HVAC subsystem's ductwork, including the air filtration units and the portion of the ductwork located outside of the main control room envelope, that maintains integrity of the main control room/technical support center pressure boundary during conditions of abnormal airborne radioactivity are designed in accordance with ASME ~~N509~~**AG-1** (Reference ~~23~~**36**), ~~Section 5.10~~**Article SA-4500** to provide low leakage components necessary to maintain main control room/technical support center habitability.

9.4.7.2.2 Component Description

The containment air filtration system is comprised of the following components. These components are located in buildings on the Seismic Category I Nuclear Island and the Seismic Category II portion of the annex building. The seismic design classification, safety classification and principal construction code for Class A, B, C, or D components are listed in Section 3.2. Table 9.4.7-1 provides design parameters for the major components of the system.

Supply Air Handling Units

Each supply air handling unit consists of a low efficiency filter bank, a high efficiency filter bank, a hot water heating coil bank, a chilled water cooling coil bank and a supply fan.

Exhaust Air Filtration Units

Each exhaust air filtration unit consists of an electric heater, an upstream high efficiency filter bank, a HEPA filter bank, a charcoal adsorber with a downstream postfilter bank, and an exhaust fan. The filtration unit configurations, including housing, internal components, ductwork, dampers, fans, and controls, are designed, constructed, and tested to meet the **applicable** performance requirements of ASME **AG-1,** N509 and ASME N510 (References **36,** 2 and 3) to satisfy the guidelines of Regulatory Guide 1.140 (Reference 30) except as noted in Appendix 1A. The filtration unit housings maximum leakage rates do not exceed one percent of the design flow in accordance with ASME ~~N509~~**AG-1**. Refer to Table 9.4-1 for a summary of the containment air filtration system filtration efficiencies and Appendix 1A for a comparison of the containment air filtration system exhaust air filtration units with Regulatory Guide 1.140 (Reference 30).

Isolation Dampers

Isolation dampers are bubble tight, single-blade or parallel-blade type. The isolation dampers have spring return actuators which fail closed on loss of electrical power or instrument air. The design and construction of the isolation dampers is in accordance with ANSI/AMCA 500 or ASME ~~N509~~ **AG-1** (References 14 and ~~2~~**36**).

Pressure Differential Control Dampers

Pressure differential control dampers utilize opposed-blade type construction and meet the performance requirements of ANSI/AMCA 500 (Reference 14) or ASME ~~N509~~ **AG-1** (Reference ~~2~~**36**), Section ~~5.9~~**DA**. The dampers maintain a slight negative pressure within the fuel handling building area, with respect to the environment and adjacent non-radiologically controlled plant areas.

Supply and Exhaust Fans

The supply and exhaust air fans are centrifugal type, single width single inlet (SWSI), with high efficiency wheels and backward inclined blades to produce non-overloading horsepower characteristics. Fan performance is rated in accordance with ANSI/AMCA 210 (Reference 4), ANSI/AMCA 211 (Reference 5) and ANSI/AMCA 300 (Reference 6).

Containment Penetrations

The containment penetrations include containment isolation valves, interconnecting piping, and vent and test connections with manual test valves. The containment isolation components that maintain the integrity of the containment pressure boundary after a LOCA are classified as Safety Class B and seismic Category I. Seismic Category I debris screens are mounted on Safety Class C, seismic Category I pipe to prevent entrainment of debris through the supply and exhaust openings that may prevent tight valve shutoff. The screens are designed to withstand post-LOCA pressures.

The containment isolation valves inside and outside the containment have air operators. The valves are designed to fail closed in the event of loss of electrical power or air pressure. The valves are controlled by the protection and plant safety monitoring system as discussed in subsection 7.1.1. The valves shut tight against the containment pressure following a design basis accident.

Ductwork and Accessories

Ductwork, duct supports and accessories are constructed of galvanized steel. Ductwork subject to fan shutoff pressures is structurally designed to accommodate fan shutoff pressures. The system air ductwork inside containment meets seismic Category II criteria so that it will not fall and damage any safety-related equipment following a safe shutdown earthquake. Ductwork, supports and accessories meet the design and construction requirements of SMACNA Rectangular and Round Industrial Duct Construction Standards (References 16 and 34) and SMACNA HVAC Duct Construction Standard - Metal and Flexible (Reference 17). The exhaust air ductwork and supports meet the design and construction requirements of ASME ~~N509~~**AG-1** (Reference ~~2~~**36**), ~~Section 5.10~~**Article SA-4500**.

Shutoff and Balancing Dampers

Multiblade, two-position remotely operated shutoff dampers are parallel-blade type. Multiblade, balancing dampers are opposed-blade type. Air handling unit and fan shutoff dampers are designed for maximum fan static pressure at shutoff flow and meet the performance requirements of ANSI/AMCA 500 (Reference 14). The containment exhaust air dampers meet the design and construction criteria of ASME ~~N509~~**AG-1** (Reference ~~2~~**36**), Section ~~5.9~~**DA**.

Fire Dampers

Fire dampers are provided where the ductwork penetrates a fire barrier to maintain the fire resistance rating of the fire barriers. The fire dampers meet the design and installation requirements of UL-555 (Reference 15).

Low Efficiency Filters, High Efficiency Filters, and Postfilters

Low and high efficiency filters are rated in accordance with ASHRAE Standard 52 and 126 (References 7 and 35). The minimum average dust spot efficiencies of the filters are shown in Table 9.4.7-1. High efficiency filter performance upstream of HEPA filter banks meet the design requirements of ASME ~~N509~~**AG-1** (Reference ~~2~~**36**), Section ~~5.3~~**FB**. Postfilters located downstream of the charcoal adsorbers have a minimum DOP efficiency of 95 percent. The filters meet UL 900 Class I construction criteria (Reference 8).

HEPA Filters

HEPA filters are constructed, qualified, and tested in accordance with ASME ~~N509~~ **AG-1** (Reference ~~2~~**36**), Section ~~5.1~~**FC**. Each HEPA filter cell is individually shop tested to verify an efficiency of at least 99.97 percent using a monodisperse 0.3-μm aerosol **in accordance with ASME AG-1, Section TA**.

Charcoal Adsorbers

Each charcoal adsorber is designed constructed, qualified, and tested in accordance with ASME ~~N509~~**AG-1 (Reference 36)**, Section ~~5.2~~**FE** ~~(Reference 2); ASME 510, Sections 11, 12, and 16 (Reference 3)~~; and Regulatory Guide 1.40. Each charcoal adsorber is a single assembly with welded construction and 4-inch deep Type III rechargeable adsorber cell, conforming with 1E Bulletin 80-03 (Reference 29).

Electric Heating Coils

The electric heating coils are fin tubular type. The electric heating coils meet the requirements of UL-1995 (Reference 10). The coils are constructed, qualified and tested in accordance with ASME ~~N509~~ **AG-1** (Reference ~~2~~**36**), Section ~~5.5~~**CA**.

Heating Coils

The heating coils are hot water, finned tubular type. The heating coils are provided with integral face and bypass dampers to prevent freeze damage when modulating the heat output. Coils are performance rated in accordance with ANSI/ARI 410 (Reference 12).

Cooling Coils

The chilled water cooling coils are counterflow, finned tubular type. The cooling coils are designed and rated in accordance with ASHRAE 33 (Reference 11) and ANSI/ARI 410 (Reference 12).

9.4.12 Combined License Information

The Combined License applicants referencing the AP1000 certified design will implement a program to maintain compliance with **ASME AG-1 (Reference 36),** ASME N509 (Reference 2), ASME N510 (Reference 3) and Regulatory Guide 1.140 (Reference 30) for portions of the nuclear island nonradioactive ventilation system and the containment air filtration system identified in subsection 9.4.1 and 9.4.7. The Combined License applicant will also provide a description of the MCR/TSC HVAC subsystem's recirculation mode during toxic emergencies, and how the subsystem equipment isolates and operates, as applicable, consistent with the toxic issues to be addressed by the Combined License applicant as discussed in DCD subsection 6.4.7.

- ***Add new reference 36 to DCD 9.4.13:***

9.4.13 References

36. "Code on Nuclear Air and Gas Treatment," ASME/ANSI AG-1-1997

PRA Revision:

None

NRC Additional Comments: (Revision 1)

The staff is currently working with the industry to address control room habitability issues including air in-leakage testing. It is anticipated that the testing frequency will be on the order of 5 to 6 years. The staff expects that testing requirements for the AP1000 design will be consistent with the resolution of the control room habitability issues currently pursued by the industry and the staff. Therefore, the AP1000 design should include a commitment to resolving the in-leakage testing in accordance with the anticipated outcome of the joint effort between the NRC staff and industry.

AP600 Design Certification was based upon the ASTM E741 tracer gas dilution testing every 10 years interval after its initial testing for the control room envelope (MCRE) to determine its unfiltered inleakages. During the AP600 design Certification period, ASTM E741 tracer gas dilution testing was a first of a kind testing for the MCRE. During the period following the AP600 design Certification, the NRC staff and industry learned more about tracer gas testing and the staff is currently working with the industry to address control room habitability issues including air in-leakage testing. It is anticipated that the testing frequency will be on the order of 5 to 6 years. Therefore, the AP1000 design should include a commitment to resolving the inleakage testing in accordance with the anticipated outcome of the joint effort between the NRC staff and industry.

Westinghouse Additional Response: (Response Revision 1)

Westinghouse did not interpret that the original staff comment was limited to only the testing frequency of the control room leakage test. Since this has been clarified, and it is understood that testing will remain in accordance with ASTM E741, Westinghouse will revise the DCD as follows:

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

6.4.5.4 Air Inleakage Testing

Testing for main control room inleakage during VES operation will be conducted~~ once every ten years. This testing will be conducted~~ in accordance with ASTM E741, (Reference 4).

6.4.7 Combined License Information

At the end of DCD section 6.4.7, add the following new paragraph...

The Combined License applicant will provide the testing frequency for the main control room inleakage test discussed in section 6.4.5.4.

Add the following to DCD Table 1.8-2:

Table 1.8-2 (Sheet 3 of 6)

SUMMARY OF AP1000 STANDARD PLANT COMBINED LICENSE INFORMATION ITEMS

Item No.	Subject	Subsection
6.4-3	Main Control Room Inleakage Test Frequency	6.4.7

PRA Revision: (Response Revision 1)

None

NRC Additional Comments: (Revision 2)

Westinghouse needs to:

(a) verify that chemicals listed in SSAR Table 6.4-1, "Onsite Chemicals," were evaluated using the methodology in NUREG-0570, "Toxic Vapor Concentrations in the Control Room Following A Postulated Accidental Release," to conclude that these chemicals do not represent a toxic hazard to control room operators;

(b) verify that combined license applicants are responsible for the amount and location of possible sources of toxic chemicals (as shown in SSAR Table 6.4-1, and their locations, as shown in SSAR Figure 1.2-2) in or near the plant and for seismic Category I Class 1E toxic gas monitoring, as required and assess control room protection for toxic chemicals, and for evaluating offsite toxic releases (including the potential for toxic releases beyond 72 hours) in accordance with RG 1.78-December 2001, Revision 1, "Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release" to meet the requirements of TMI Action Plan Item IIID.3.4 and GDC 19 ;

(c) add RG 1.78-December 2001, Revision 1 reference to SSAR Section 6.4.8, "References" Since "Regulatory Guide 1.78-December 2001, Revision 1" replaces the both "Regulatory Guide 1.78-June 1974, Revision 0" and "Regulatory Guide 1.95-January 1977, Revision 1";

(d) delete reference of "Regulatory Guide 1.95" from SSAR Section 6.4.7;

(e) revise Appendix 1A to assess the conformance with RG 1.78-December 2001, Revision 1, and revise DCD Tier 2 Sections 2.2, 6.4, 9.4.1, 9.5.1, and Table 1.9-1 (Sheet 7 of 15) to correctly state the reference as "RG 1.78-December 2001, Revision 1"; and

(f) revise references list in Technical Specifications Bases B.3.7.6 to add a reference of ASHRAE Standard 62-1989.

Westinghouse Additional Response: (Response Revision 2)

The following responses correspond to the items in the "NRC Additional Comments: (Revision 2)".

(a) Westinghouse confirms that the chemicals listed in SSAR Table 6.4-1, "Onsite Chemicals," were evaluated using the methodology in NUREG-0570, "Toxic Vapor Concentrations in the Control Room Following A Postulated Accidental Release."

(b) Westinghouse confirms that the combined license applicants are responsible for the amount and location of possible sources of toxic chemicals (as shown in SSAR Table 6.4-1, and their locations, as shown in SSAR Figure 1.2-2) in or near the plant and for seismic Category I Class 1E toxic gas monitoring, as required and assess control room protection for toxic chemicals, and for evaluating offsite toxic releases (including the potential for toxic releases beyond 72 hours) in accordance with RG 1.78-December 2001, Revision 1, "Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release" to meet the requirements of TMI Action Plan Item IIID.3.4 and GDC 19. See DCD change below. (In particular the changes to subsection 6.4.7.)

(c) Westinghouse will add RG 1.78-December 2001, Revision 1 reference to SSAR Section 6.4.8, "References." See DCD change below.

(d) Westinghouse will delete the reference of "Regulatory Guide 1.95" from SSAR Section 6.4.7. See DCD change below.

(e) Westinghouse will revise Appendix 1A to assess the conformance with RG 1.78-December 2001, Revision 1, and revise other DCD Tier 2 Sections to correctly state the reference as "RG 1.78-December 2001, Revision 1." See DCD changes below.

(f) Westinghouse will revise the references list in Technical Specifications Bases B.3.7.6 to add a reference of ASHRAE Standard 62-1989. See DCD changes below.

Design Control Document (DCD) Revision: (Response Revision 2)

Change DCD Appendix 1A as follows:

Reg. Guide 1.78, Rev. 1, 12/01~~Rev. 0, 6/74~~ - ~~Assumptions for~~ Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release

C.1	**N/A**	**This criterion is site-specific. Therefore, this is not applicable to AP1000 design certification. It is the Combined License applicant's responsibility.**
C.2	**N/A**	**This criterion is site-specific. Therefore, this is not applicable to AP1000 design certification. It is the Combined License applicant's responsibility.**
C.3.1	**N/A**	**This criterion is site-specific. Therefore, this is not applicable to AP1000 design certification. It is the Combined License applicant's responsibility.**
C.3.2	**Conforms**	
C.3.3	**Exception**	**For AP1000 design certification the atmospheric dispersion factors are not calculated (since there are no specific site data) but are selected so as to bound the majority of existing sites. Section 2.3 provides additional information.**
C.3.4	**Conforms**	
C.4.1	**N/A**	**This criterion is site-specific. Therefore, this is not applicable to AP1000 design certification. It is the Combined License applicant's responsibility.**
C.4.2	**Conforms**	

C.4.3	Conforms	
C.5	N/A	**Not applicable to AP1000 design certification. This is the Combined License applicant's responsibility.**
~~C.1~~	~~N/A~~	~~This criterion is site specific. Therefore, this is not applicable to AP1000 design certification.~~
~~C.2~~	~~N/A~~	~~This criterion is site specific. Therefore, this is not applicable to AP1000 design certification.~~
~~C.3~~	~~Exception~~	~~In the event of a hazardous chemical spill occurring onsite during normal operation, the main control room emergency habitability system may be manually actuated from the main control room. In addition, the main control room is supplied with self-contained portable breathing equipment for operator protection.~~
		~~The Combined License applicant is responsible for the amount and location of possible sources of toxic chemicals near the plant, and toxic gas monitoring, as required. The Combined License applicant is also responsible for plant-specific procedures and training in support of control room habitability.~~
~~C.4~~	~~N/A~~	~~Refer to discussion on item C.3~~
~~C.5.a~~	~~Conforms~~	
~~C.5.b~~	~~Exception~~	~~Refer to discussion on item C.6~~
~~C.6~~	~~Exception~~	~~For AP1000 design certification the atmospheric dispersion factors are not calculated (since there are no specific site data) but are selected so as to bound the majority of existing sites. Section 2.3 provides additional information.~~
~~C.7~~	~~Conforms~~	
~~C.8~~	~~Conforms~~	
~~C.9~~	~~Exception~~	~~Although the anticipated operating mode for the AP1000 in the event of a toxic gas release is for~~

		~~100 percent recirculation, there is the potential for operation with a pressurized main control room using bottled air. The design pressurization is 1/8 in. water gauge.~~
~~C.10~~	~~N/A~~	
~~C.11~~	~~Conforms~~	
~~C.12~~	~~Conforms~~	
~~C.13~~	~~Conforms~~	~~Onsite toxic substances conform to these guidelines. Offsite toxic chemicals are site specific and are the Combined License applicant's responsibility.~~
~~C.14~~	~~Conforms~~	
~~C.15~~	~~N/A~~	~~Not applicable to AP1000 design certification. This is the Combined License applicant's responsibility.~~

Reg. Guide 1.95~~, Rev. 1, 1/77 - Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release~~ -Withdrawn

~~General~~	~~N/A~~	~~The AP1000 does not have onsite chlorine sources. Therefore, these guidelines are not applicable to the AP1000. Offsite chlorine sources are site specific and are the Combined License applicant's responsibility.~~

Change DCD Table 1.9-1 (Sheets 7 and 8 of 15) as follows:

Division 1 Regulatory Guide		DCD Chapter, Section or Subsection
1.78	~~Assumptions for~~ Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Releases (~~Rev. 0, June 1974~~**Rev. 1 December 2001**)	2.2 6.4 9.4.1 9.5.1
1.95	~~Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release (Rev. 1, January 1977)~~**Withdrawn**	~~This regulatory guide is not applicable to AP1000 design certification.~~

Change DCD 6.4 as follows:

Next to last paragraph in 6.4.4

The protection of the operators in the main control room from offsite toxic gas releases is discussed in Section 2.2. The sources of onsite chemicals are described in Table 6.4-1 and their locations are shown on Figure 1.2-2. Analysis of these sources are in accordance with Regulatory Guide 1.78 **(Reference 5)** and shows that these sources do not represent a toxic hazard to control room personnel.

Revise paragraph as shown below. The revision incorporates the changes from Response Revision 1.

6.4.7 Combined License Information

Combined License applicants referencing the AP1000 certified design are responsible for the amount and location of possible sources of toxic chemicals in or near the plant and for seismic Category I Class 1E toxic gas monitoring, as required. Regulatory Guide~~s~~ 1.78 **(Reference 5)** ~~and 1.95~~ addresses control room protection for toxic chemicals, and ~~for evaluating~~ **evaluation of** offsite toxic releases (including the potential for toxic releases beyond 72 hours) ~~in accordance with the guidelines of Regulatory Guides 1.78 and 1.95~~ in order to meet the requirements of TMI Action Plan Item III.D.3.4 and GDC 19.

Combined License applicants referencing the AP1000 certified design are responsible for verifying that procedures and training for control room habitability are consistent with the intent of Generic Issue 83 (see Section 1.9).

The Combined License applicant will provide the testing frequency for the main control room inleakage test discussed in section 6.4.5.4.

Add new Reference 5 to 6.4.8:

5. **"Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release", Regulatory Guide 1.78, Revision 1, December 2001.**

Revise Technical Specifications Bases B3.7.6 as follows:

BACKGROUND

The Main Control Room Habitability System (VES) provides a protected environment from which operators can control the plant following an uncontrolled release of radioactivity. The system is designed to operate following a Design Basis Accident (DBA) which requires protection from the release of radioactivity. In these events, the Nuclear Island

Non-Radioactive Ventilation System (VBS) would continue to function if AC power is available. If AC power is lost or a High-2 main control room (MCR) radiation signal is received, the VES is actuated. The major functions of the VES are: 1) to provide forced ventilation to deliver an adequate supply of breathable air **(Ref. 4)** for the MCR occupants; 2) to provide forced ventilation to maintain the MCR at a 1/8 inch water gauge positive pressure with respect to the surrounding areas; and 3) to limit the temperature increase of the MCR equipment and facilities that must remain functional during an accident, via the heat absorption of passive heat sinks.

REFERENCES

4. ASHRAE Standard 62-1989, "Ventilation for Acceptable Indoor Air Quality"

PRA Revision: (Response Revision 2)

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 440.045 (Response Revision 1)

Question:

Section 5.4.7.2.2 describes the AP1000 normal residual heat removal system (RNS) design features addressing intersystem LOCA issue described in SECY-90-016, "Evolutionary Light Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements," dated January 12, 1990. Also, Section 1.9.5.1.7 addresses AP1000's compliance with the NRC position regarding the inter-system LOCA issue. It states that AP1000 has similar fluid system design to the AP600; therefore, the conclusions of topical report WCAP-14425, "Evaluation of the AP600 Conformance to Inter-System Loss-of-Coolant Accident Acceptance Criteria," dated July 1995, are applicable to the AP1000.

Identify design differences between the AP1000 and AP600, in terms of the design and design pressure of the primary or secondary systems and subsystems that directly or indirectly interfacing the RCS, that could affect the inter-system LOCA conclusions. For each of these differences identified, justify why the conclusions of WCAP-14425 are applicable to the AP1000.

Westinghouse Response:

There are no significant differences between AP1000 and AP600 in terms of intersystem LOCA related features. However, we have determined that a new WCAP should be issued, with AP1000-specific descriptions and illustrations. Therefore, WCAP-15993, Revision 0, "Evaluation of AP1000 Conformance to Inter-System Loss-of-Coolant Acceptance Criteria" is being provided.

Design Control Document (DCD) Revision:

Update DCD Section 1.9.5.1.7 to reflect the new WCAP, as shown on the attached pages.

PRA Revision:

None

Follow On Question:

1. P. 3-4, Section 3.1.2, RNS Relief Valve, Line 7:
   The motor-operated CIV is listed as V04.
   Should it be V011 according to Fig. 3-1?

2. P. 3-9, Section 3.3.2, Line #6:
CVS Makeup pump discharge line check valves are listed as V156A and B. (V156A and B are not check valves, but are isolation valves, according to Fig. 3-2).

   Should they should be V160A and B?

3. P. 3-11, Section 3.4.1 Primary Sampling System Description, 2nd para. Lines 2 thru 6: It states that each connection of the PSS to RCS contains a flow-restricting orifice. However, the orifices are not shown in Fig. 3-4. Why?

4. Fig. 3-4 shows the only low-pressure components in PSS are Eductor water storage tank (EWST) and demineralized water supply line.

   Are the Eductor supply pump seal, EWST drainage line, and EWST level indication line also low-pressure components?

   Explain the PSS design differences between AP600 and AP1000.

5. In Section 3.4.2, it states that "Even in the unlikely event that overpressurization would occur, leakage flow from the RCS would be well within the makeup capability of the normally operating makeup system."

   Since the ISLOCA concern is LOCA outside containment, rather than makeup capability, the statement appears to be irrelevant.

6. Section 3.6.1 describes the demineralized water transfer and storage system interface with the primary sampling system. It references Figure 3-4, and discusses the DWS isolation valve V007, check valve V013, and isolation valve V037 (to the liquid waste system degasification). They are not shown in Figure 3.4. (Figure 3-4 and DCD Fig 9.3.3-1 do not show demineralized water transfer system).

   Discuss and modify the figure, if necessary.

7. Section 3.6.2 states that a relief valve has been added to the DWS header inside containment to preclude the possibility of overpressurizing the DWS.

   Is the relief valve described in DCD Section 9.2.4?

Westinghouse Response:

All page numbers below refer to WCAP-15993 Revision 0. Changes are reflected in Revision 1. The item numbers below correspond to the questions and comments made above.

1. On page 3-4, the reference to the RNS relief valve has been corrected, from V004 to V011.

2. On page 3-9, the reference to the CVS makeup pump discharge check valves has been corrected, from V156A/B to V160A/B.

3. The flow restricting orifices cited are in the Reactor Coolant System, and are shown there by means of a note on the piping and instrumentation diagram.

4. The PSS eductor supply pump seal and lines interconnected to the EWST are also low pressure components.

   There are no differences between the AP600 and the AP1000 PSS design.

5. The comments about makeup being within the capability of the normal makeup system are intended to indicate that this event is a "leak" rather than a "LOCA."

6. The DWS interface with the PSS is inside containment. After discussions between Westinghouse and the NRC by phone it as determined that no modifications are required.

7. This relief valve is shown on the DWS piping and instrumentation diagram.

WCAP-15993 has been updated to Revision 1 to incorporate these plus some other corrections and clarifications.

Additional Design Control Document (DCD) Revision:

None.

Additional PRA Revision:

None.

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 420.046 (Response Revision 1)

Question:

420.46 (DCD Tier 1, Sections 2.5.1, 2.5.3, 2.5.4, 2.5.5, 2.5.6, and 2.5.7)

Describe the architecture of the real-time data network and how the information is used to control and monitor the plant. This includes the network in the PLS and the one that interfaces with the DDS, DCS PLS, IIS, SMS and the DAS.

Westinghouse Response:

Chapter 7 of the DCD currently depicts the real-time data networks in two segments. The first segment is located entirely within the Plant Control System (PLS). The second segment resides primarily in the Data Display and Processing System (DDS), but also interfaces to the Protection and Safety Monitoring System (PMS), the Incore Instrumentation System (IIS), and the Special Monitoring System (SMS). That segmentation is not driven by a functional requirement and will be removed from DCD Figure 7.1-1.

There is one real-time data network that provides the communication backbone for all of the non-Class 1E instrumentation and control systems. The network will also interface to the Class-1E systems through electrical isolation devices and communication buffering devices that are part of the safety system and prevent degradation of the safety function. While the exact implementation and topology of the real-time data network will depend on the communication technology available at the time of deployment, the following characteristics are required:

1. Availability commensurate with the requirements of functions using the network
2. Maximum Data Latency commensurate with the requirements of functions using the network

The real-time data network is primarily used to provide the following control and monitoring functions:

1. Distributed Non-Safety Plant Control System (PLS) - The PLS is used to control the reactor, the turbine, and balance of plant functions. The network is used for manual soft control, supervisory control, alarming, data logging, maintenance, and diagnostics.

2. Data Display and Processing System (DDS) – The DDS provides the traditional plant computer functions including historical data storage and retrieval, data display, data logging, nuclear application programs (plant models and associated calculations). These functions are implemented using a number of computational and display resources that are distributed on the real-time data network. These resources include workstations, storage devices, and computational processors.

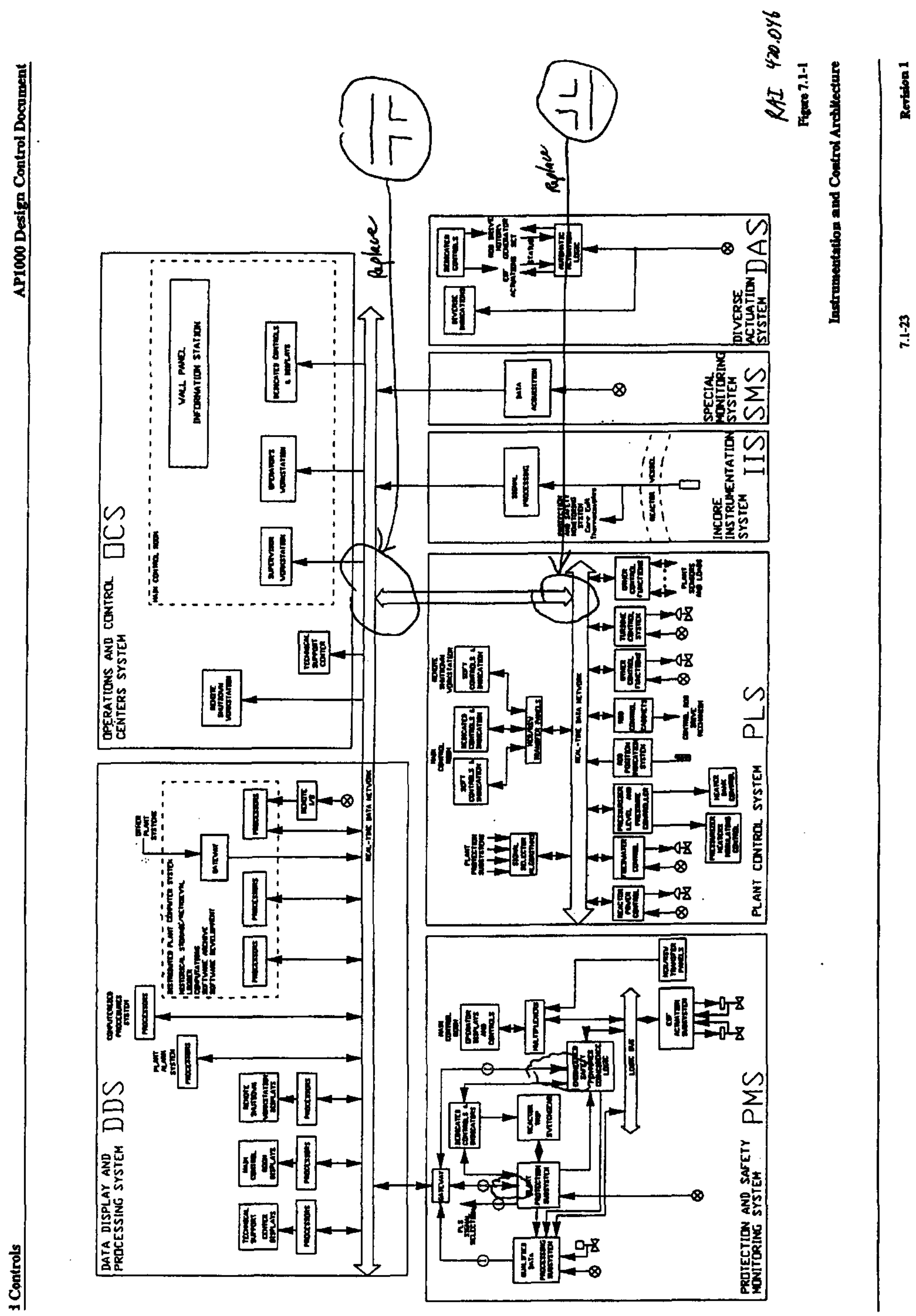
3. Supervision and Limited Control of the Safety Systems - The real-time data network is connected to a Gateway in each of the safety divisions. The Gateway provides electrical and functional isolation between the safety system and the non-safety system. The primary data flow between the systems is the transfer of plant status, safety system status, and diagnostic information from the safety system to the non-safety systems. The primary use of this data is for plant computer functions and historical data logging. Data flow from the non-safety system into the safety system is used primarily for manual control of the ESF system. These control signals are validated by the safety system prior to action being taken. The validation either takes the form of logic that prevents inhibiting of an automatic safety function or confirmation of action with the operator where there is no automatic safety function. Further details are available in Appendix 4 of the Common Q Topical Report. DCD Figure 7-1.1 will be revised within the PMS to make the arrows between the gateway and both the plant protection subsystem and the engineered safety features coincidence logic bi-directional. Figure 7.1-2 will be revised to make the arrow between the gateway and the plant protection subsystem bi-directional.

4. Interfaces to Other Systems – Gateways and firewalls provide for the controlled transfer of data between the real-time data network and external system.

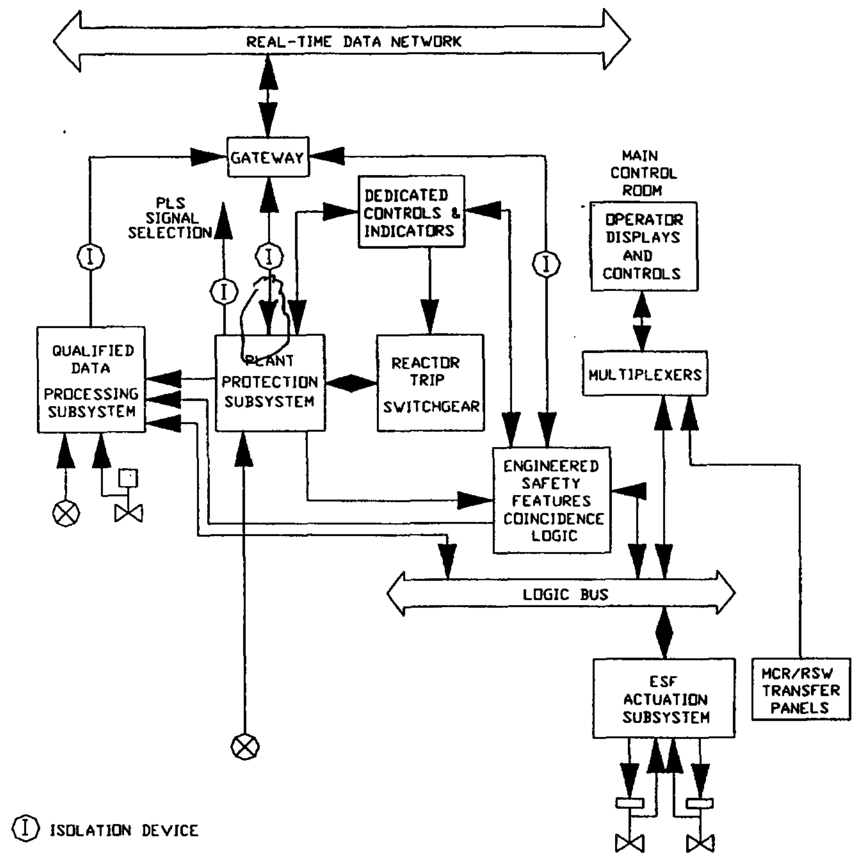
It should be noted that the Diverse Actuation System is a completely standalone system and does not interface to the real-time data network.

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

These changes were incorporated in Revision 3 of the AP1000 DCD.



AP1000 Design Control Document
I Controls
OPERATIONS AND CONTROL CENTERS SYSTEM
OCS
DATA DISPLAY AND PROCESSING SYSTEM
DDS
PROTECTION AND SAFETY MONITORING SYSTEM
PMS
PLANT CONTROL SYSTEM
PLS
INCORE INSTRUMENTATION SYSTEM
IIS
SPECIAL MONITORING SYSTEM
SMS
DIVERSE ACTUATION SYSTEM
DAS
WALL PANEL INFORMATION STATION
REAL-TIME DATA NETWORK
Replace
RAI 420.046
Figure 7.1-1
Instrumentation and Control Architecture
7.1-23
Revision 1



RAI 420.046

Figure 7.1-2

Protection and Safety Monitoring System

PRA Revision: (Original Response)

None

NRC Additional Comments:

Revise RAI response and Tier 1 to add new ITAAC to address limits to the communication between the nonsafety PLS and safety PMS so that nonsafety systems will not adversely impact the safety system.

Revise RAI response and Tier 2 with new discussion on functional requirements of the gateway. Also must capture the functional independence between the nonsafety systems and the PMS.

Westinghouse Additional Response:

The DCD will be revised as shown.

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

DCD Tier 1 Revision:

2.5.2 Protection and Safety Monitoring System

Design Description

7. The PMS provides the following nonsafety-related functions:

    a) The PMS provides process signals to the plant control system (PLS) through isolation devices.

    b) The PMS provides process signals to the data display and processing system (DDS) through isolation devices.

    c) Data communication between safety and nonsafety systems does not inhibit the performance of the safety function.

    d) The PMS ensures that the automatic safety function and the Class 1E manual controls both have priority over the non-Class 1E soft controls.

Table 2.5.2-8 (cont.)
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
7.a) The PMS provides process signals to the PLS through isolation devices.	Type tests, analyses, or a combination of type tests and analyses of the isolation devices will be performed.	A report exists and concludes that the isolation devices prevent credible faults from propagating into the PMS.
7.b) The PMS provides process signals to the DDS through isolation devices.	Type tests, analyses, or a combination of type tests and analyses of the isolation devices will be performed.	A report exists and concludes that the isolation devices prevent credible faults from propagating into the PMS.
7.c) Data communication between safety and nonsafety systems does not inhibit the performance of the safety function.	**i) Inspection of the as-built PMS gateways will be performed.**	**i) Each network interface between safety and nonsafety systems includes a buffering circuit.**
	ii) An operational test of the as-built PMS gateways will be performed. Power will be removed from the nonsafety components that communicate with the gateways. Real or simulated signals will be used. The automatic and manual actions listed in Tables 2.5.2-2, 2.5.2-3, and 2.5.2-4 will be tested.	**ii) With power removed from the nonsafety components, appropriate PMS output signals are generated after the test or manual actuation signal reaches the specified limit.**
	iii) An operational test of the as-built PMS gateways will be performed. Attempts will be made to send messages to the PMS from the DDS. Some of the messages will be from a predefined list of valid commands and some will not be on the list.	**iii) The gateways filter the incoming message streams and only accept commands from a predefined list of valid commands. All other messages are discarded.**

Table 2.5.2-8 (cont.) Inspections, Tests, Analyses, and Acceptance Criteria		
Design Commitment	**Inspections, Tests, Analyses**	**Acceptance Criteria**
7.d) The PMS ensures that the automatic safety function and the Class 1E manual controls both have priority over the non-Class 1E soft controls.	**i) An operational test of the as-built PMS will be performed. Real or simulated signals will be used. An attempt will be made to block an automatic action as listed in Tables 2.5.2-2 and 2.5.2-3 using the non-Class 1E controls.** **ii) An operational test of the as-built PMS will be performed using the PMS manual actuation controls. An attempt will be made to block a manual action as listed in Table 2.5.2-4 using the non-Class 1E controls.**	**i) Appropriate PMS output signals are generated after the test signal reaches the specified limit. These output signals remain following an attempt to block these signals using the non-Class 1E controls.** **ii) PMS output signals are generated for manually actuated functions as identified in Table 2.5.2-4 after the manual actuation controls are operated. These output signals remain following an attempt to block these signals using the non-Class 1E controls.**

DCD Tier 2 Revision:

[Note to reviewers: RAI 420.008 (Response Revision 1) provided some revisions to DCD subsection 7.1.2.8 regarding Gateway design. This RAI revised response provides additional changes to DCD subsection 7.1.2.8. The changes shown below include the changes from RAI 420.008, as well as those required by this RAI, and represent the complete set of pending revisions to DCD, Revision 3, subsection 7.1.2.8.]

7.1.2.8 Communication Functions

The communication functions provide information from the plant protection subsystem, the ESF coincidence logic, the ESF actuation subsystems, and the QDPS subsystems to external systems. This includes outputs to the plant control system and the data display and processing system. Isolation devices provide electrical isolation between the protection and safety monitoring system and the external systems. **The communication functions also provide soft control information from the non-safety system to the safety system for operator-initiated actuation and component control.**

The communication functions are accomplished via channelized gateways as shown in Figure 7.1-1.

The PMS Gateway interfaces 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 low-level 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.

The gateway provides both electrical and communication isolation between the non-safety systems and the PMS. Other than the isolation function, the gateway is not required for any PMS safety function. There is no potential signal from the non-safety system than will prevent the PMS from performing its safety functions.

Specifically, the Gateway will provide the following isolation features:

1) Electrical isolation between the Class 1E and non-Class 1E ports of the Gateway, as required by IEEE 603-1991 (Reference 1).

2) Communication isolation between the Class 1E and non-Class 1E ports of the Gateway, as envisioned by IEEE 7-4.3.2-1993, Annex G (Reference 15). This includes:

   a) Class 1E communications buffering circuits to process the low-level interface signals.

   b) Use of only simple connectionless protocols between the Class 1E and non-Class 1E ports of the Gateway. (Connectionless protocols do not use connection establishment/ management/ termination nor do they use acknowledgements/ negative-acknowledgements/ retransmission.)

   c) Software within the Class 1E portion of the gateway will filter the incoming message stream and only accept valid soft control commands from a predefined list of valid commands. All other messages will be discarded.

Application software running in the safety system will ensure the functional independence of the Class 1E functions from the soft control demands received from the non-safety systems.

Specifically, the application software will provide the following features:

1) **In cases where a component is controlled by an automatic safety function, the PMS application software will ensure that the automatic safety function and the Class 1E soft controls both have priority over the non-Class 1E soft controls.**

2) **In cases where a Class 1E component is not controlled by an automatic safety function, the PMS application software will ensure that the Class-1E controls have priority over the non-Class 1E soft controls.**

Analog inputs required for both control and protection functions are processed independently with separate input circuitry. The input signal is classified as safety-related and is, therefore, isolated in the protection and safety monitoring system cabinet before being sent to the control system.

The plant protection and safety monitoring system also provides data to the plant control system pertaining to signals calculated in the subsystems, and to the data display and processing system.

Non-process signals are also provided to external systems. The non-process outputs inform the external systems of cabinet entry status, cabinet temperature, dc power supply voltages, and subsystem diagnostic status. Cabinet temperature sensing does not affect the safety-related function. The information is gathered for the sole purpose of analysis by external systems.

7.1.7 References

15. **IEEE 7-4.3.2-1993, "IEEE Standard Criteria for Digital Computers in Safety Systems of Nuclear Power Generating Stations."**

PRA Revision: (Response Revision 1)

None

RAI Number: 440.092 (Response Revision 1)

Question:

In the case of the DEDVI break and wall-to-wall floodup (Section 15.6.5.4C.3), it was estimated that 28.5 days will be required to attain this condition.

How was this time estimated? How was the inleak rate derived? Would the long-term cooling be sustainable if the floodup was assumed to occur at the end of the IRWST injection?

Westinghouse Response:

The following assumptions have been included in determining the time to reach wall-to-wall floodup following a DEDVI break in the AP1000.

- The break occurs in the PXS B room. This is more limiting than a break in the RCS loop compartment or in the PXS A room because it results in lowest initial post-LOCA containment flood level.
- All volumes inside containment, below the recirculation flood level, are assumed to flood in the long term.
- Both CMTs, both accumulators, and the IRWST either inject or spill.
- The RCS is water filled water solid up to 80% of the RCS hot leg.
- The containment is pressurized to the resultant pressure following a DEDVI break and a water film exists on surfaces in containment.
- The CMTs are not assumed to refill after injection since they are located above the floodup elevation.
- The accumulators are not assumed to refill after injection. Although they are located below the floodup elevation, enough N2 will remain in the tanks to balance the flood pressure. In addition, series check valves are located in their discharge lines.

All volumes below the recirculation flood level are assumed to flood based on one or a combination of the following reasons:

- Back leakage occurs through the check valves in each room drain line. Note, this is conservative since each drain line has two check valves in series such that failure of both check valves is required to open the drain line.

- Leakage occurs through the concrete walls separating the normally flooded areas from the normally unflooded areas. Again, this is conservative.

Based on the above assumptions, an initial in-leakage rate of 9.0 gpm was assumed at the time of the DVI break. A resistance was assumed between the flooded and normally unflooded areas allowing determination of the time to reach wall to wall flooding considering the initial in-leakage rate of 9.0 gpm. The time to reach wall to wall flooding with an initial leakage rate of 9.0 gpm is about 29 days. A time of 28.5 days was used in the Chapter 15 safety analysis calculations for conservatism.

Note that at 28.5 days, the DCD long-term core cooling analysis shows significant margin. This margin would allow adequate core cooling assuming that wall-to-wall flooding occurred earlier than 28.5 days. However adequate core cooling would most likely not be demonstrated for the hypothetical case where wall-to-wall flooding was assumed immediately after the IRWST injection phase using the conservative methodology used in the DCD analyses.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

NRC Additional Comments:

The calculation for the wall-to-wall flooding is based on the assumption of a 9.0 gallons-per-minute (gpm) in-leakage rate. You conclude that "...core cooling would most likely not be demonstrated for the hypothetical case where wall-to-wall flooding was assumed immediately after the IRWST [in-containment refueling water storage tank] injection phase..."

Please address the basis for the assumption of a 9.0 gpm in-leakage under flooding conditions and that the walls of the dry spaces will not deform under the hydrostatic pressure.

Westinghouse Revised Response:

The AP1000 containment is designed to preferentially flood compartments to maximize the volume of available containment recirculation fluid and to maximize the gravity recirculation elevation head. As discussed below, containment compartments are designed to be leak-tight, which minimizes the long-term leakage of recirculation fluid from flooded compartments into unaffected compartments, reducing the floodup volume available for recirculation.

The containment compartments are designed so that the expected long-term seepage from a flooded compartment into unaffected compartments is well below the passive in-leakage rate of 9.0 gpm assumed in the DCD Chapter 15 safety analyses. This assumed in-leakage rate assures that the time to reach wall-to-wall flooding does not occur before 28.5 days conservatively assumed in safety analyses. Passive leakage is not expected to occur for 24 hours following the event, and in the event of a passive failure causing in-leakage, an active failure is not assumed. RAI 440.053 also discusses passive failures during long-term recirculation.

During discussions with the NRC reviewers on this issue, NRC requested that the response to this RAI include a discussion on the containment design features that enhance containment floodup and recirculation performance, and minimize seepage.

DCD 3.4.1.1.2 discusses the general philosophy for protection against internal flooding and DCD 3.4.1.2.2 provides background information on the assumptions for the internal flooding evaluation. DCD 3.4.1.2.2.1 provides a discussion of containment internal flooding events in each of the seven compartments that extend below the maximum floodup elevation, including a summary of the potential flooding sources in each compartment and a discussion of the consequences of room flooding from these sources.

As discussed in DCD 3.4.1.1.2, the protection mechanisms related to minimizing the consequences of internal flooding include the following:

- Structural enclosures
- Structural barriers
- Curbs and elevated thresholds
- Leak detection systems
- Drain systems

The AP1000 containment includes the following eight containment volumes that extend below the maximum floodup elevation and can, therefore, potentially be subject to flooding:

- Reactor vessel cavity / adjoining equipment room (part of the RCS compartment)
- Two steam generator compartments (part of the RCS compartment)
- Vertical access tunnel (part of the RCS compartment)
- Two passive core cooling system (PXS) compartments A and B (not expected to flood)
- Chemical and volume control system (CVS) compartment (not expected to flood)
- Refueling canal / cavity (not expected to flood)

The refueling canal / cavity is not evaluated with the other seven compartments that extend below the maximum floodup elevation as part of the internal flooding analysis. The refueling canal / cavity provides a limited overflow volume for the maintenance floor, as discussed below.

For the majority of RCS pipe breaks that result in a LOCA, the AP1000 is expected to directly flood the RCS compartment from the break. LOCAs cannot originate in the CVS compartment since CVS compartment piping is automatically isolated on low pressurizer level, which terminates the break. There are several specific pipe breaks in either of the PXS rooms identified in DCD 3.4.1.2.2.1 that can cause LOCA flooding in these rooms, such as a break in a DVI or RNS line. A LOCA in a PXS room is expected to flow through the floor drains to the equipment room and begin filling the RCS compartment. Since break flow is expected to be significantly greater than the floor drain capacity, a LOCA is expected to fill an affected PXS compartment, which will overflow its access curb onto the maintenance floor and flow into the RCS compartment.

Therefore, the AP1000 design features ensure that pipe break or LOCA floodup water from either PXS compartment and from the CVS compartment can flow to the RCS compartment, while preventing the LOCA flow from flooding any of the other non-flooding areas that do not contain the break location.

Water flooding onto the maintenance floor at elevation 107'-2" preferentially drains into the RCS compartment before it can overflow to any other compartment, since there are no curbs or elevated thresholds around the RCS compartment openings on the maintenance floor. The maintenance floor has a large overflow pipe (centered at the 110'-0" elevation) into the refueling canal / cavity, so this begins to pass flow at an elevation slightly above 109 feet. The maintenance floor entrances into the other three compartments have elevated curbs above the 110-foot elevation, and with staggered curb heights to sequence overflow into the CVS room, PXS-B, and PXS-A compartments, in that order.

Therefore, following a LOCA, only the RCS compartment and potentially one PXS compartment would flood, and the PXS compartment would only be affected if it were the source of the break. The affected compartment(s) would experience rapid flooding until equilibrium floodup water levels are established. As discussed below, the compartment boundaries are expected to prevent the floodup water from exceeding the assumed leakage rate during a floodup condition.

Compartment Construction

The containment compartments are formed by the reinforced concrete mat of the containment base and by structural steel wall modules. In general, the various compartments in containment have wall surfaces that are the structural steel plates forming the vertical surface of the wall module. The floors in each compartment are concrete. The ceilings for a compartment are the structural steel plates that form the horizontal bottom of the structural floor module above the room. The steel plate in the ceiling is part of the reinforced structure used to support the poured concrete floor above. Therefore, the compartment arrangement typically consists of vertical steel wall plates structurally welded as necessary to both the horizontal steel ceiling plates and to the other vertical steel wall plates. The compartments have a concrete floor. There may also be parts of a compartment where a portion of the wall is formed of reinforced concrete. For the PXS and CVS compartments, the only openings for compartments are the access openings and the floor drain lines. All wall and ceiling electrical, piping, and HVAC penetrations are sealed at least up to the top of the curbs or elevated thresholds.

As discussed in DCD 3.8.3.1, the containment internal structures that form the walls of the seven containment compartments are concrete-filled, steel plate structural modules. The wall modules are supported on the concrete containment floor with the steel surface plate on each side of the wall module extending down to the concrete floor. The steel surface plates and interconnecting support steel in the structural modules provide the reinforcement for the concrete that is placed in the walls after the structural module is set in place. The structural modules are anchored to the base concrete by mechanical connections welded to the steel plate, before the wall concrete is poured.

As discussed in DCD 6.2.1.1.2, the structural walls are a minimum of 2.5 feet thick, and all containment wall modules, with the exception of the IRWST west steel wall modules, are concrete-filled. The walls, floors, and penetrations are designed to withstand the maximum anticipated hydrodynamic loads associated with a pipe failure as described in DCD section 3.6. In addition, the walls of the compartments below elevation 107' 2" are designed for the hydrostatic loads associated with flooding of any one compartment up to elevation 110-2". The loads and acceptance criteria are described in DCD subsections 3.8.3 and 3.8.4 and deformations due to hydrostatic pressure are negligible.

There are a number of design features and construction / maintenance processes that minimize the potential for any seepage from a flooded room to a non-flooded room, that also helps to support continued operation of the passive core cooling system.

The installation of the structural walls involves first setting the wall modules in place. The specific construction techniques that enhance adhesion and bonding between the wall and floor concrete are the standard methods to prepare construction joints required by the ACI Code.

As discussed in DCD 6.1.2.1.2, part of the final containment concrete surface preparations following construction includes the application of coatings primarily intended to prevent concrete from dusting, to protect it from chemical attack, and to enhance decontaminability. Exposed concrete surfaces inside containment are coated with an epoxy sealer to help bind the concrete surface together and reduce dust that can become contaminated and airborne. Concrete floors inside containment are coated with a self-leveling epoxy. Exposed concrete walls inside containment are coated to a minimum height of 7 feet with an epoxy applied over an epoxy surfacer that has been struck flush.

As discussed in DCD 6.1.2.1.4, carbon steel is coated with inorganic zinc. An epoxy top coat is used in areas subject to decontamination such as a 7 foot wainscot in high traffic areas or on surfaces subject to radiologically contaminated liquid spray, splash, or spills. Floors subject to heavy traffic or contaminated liquid spills are coated with self-leveling epoxy. An epoxy top coat is applied a minimum of 1 foot up the wall where liquid spills might splash. Floors subject to light traffic and not subject to contaminated liquid spills are coated with an epoxy top coat. The epoxys applied to the concrete surfaces are the same epoxy used as a top coat for the inorganic zinc-coated steel. A 7-foot wainscot on exposed concrete walls in high-traffic areas and any surfaces of walls subject to spray, splash or spills of contaminated liquids are coated with epoxy top coat applied over an epoxy surfacer that has been struck flush. Remaining concrete walls are coated with an epoxy sealer to reduce or eliminate dusting. Exposed concrete ceilings are coated with an epoxy sealer to reduce dusting.

The weight of the structural walls, along with the bonding of the floor surface at the wall module internal construction joints are expected to provide well-sealed interfaces. The application of the extensive epoxy treatment of the concrete and steel surfaces is expected to essentially eliminate the potential for seepage from one compartment to another through or around the wall modules. In addition, dust and fine dirt particles from construction, as well as from continuing operations to maintain the epoxy coatings and to clean containment following initial construction and periodic outage maintenance, are expected to clog any potential cracks or other seepage paths through or around the wall modules.

Compartment Penetrations

The containment arrangement for the floor drains from the PXS and CVS compartments provides a drain path for each compartment to the lowest level of containment (elevation 71'-6") where the containment sump is located. Any leakage that occurs within the containment drains by gravity to the elevation 71'-6" equipment room, which is part of the RCS compartment. Therefore, flooding in the RCS compartment is not limited to the plant systems contained only within this compartment. Reverse flow into the PXS and CVS compartments is prevented by redundant backflow preventers in each of the three compartment drain lines, so that a single failure does not result in reverse flooding of an unaffected, non-flooded compartment following a LOCA.

As discussed earlier, the PXS and CVS compartments have access penetrations that are protected by a curb or elevated threshold, to prevent flooding from the maintenance floor above the compartment. The curb arrangement also helps to properly direct overflow from any of these rooms to the RCS compartment in the event of a LOCA in these rooms that exceeds the floor drain capacity.

The AP1000 minimizes the number of penetrations through enclosure or barrier walls below the flood level. There are HVAC ducts, cable trays, and process pipes that penetrate the maintenance floor into the PXS and CVS compartments. In addition, there are several process pipes that penetrate the walls for these compartments, as discussed in the floodup discussion of DCD 3.4.2.2.1. The penetrations through compartment ceilings and walls that are below the maximum flood level are watertight. All of the process penetrations below the maximum flood level either are embedded in the wall or floor, or are welded to a steel sleeve embedded in the wall or floor. For these compartments, the process pipes are expected to be circumferentially welded to one of the steel wall or ceiling plates to form the isolation boundary. The HVAC penetrations in the compartment ceilings also have perimeter welding of the entrance pipe to the steel ceiling plate.

Cable trays enter a compartment from the ceiling through an elevated pipe or other entrance threshold device that is at least as high as the maintenance floor entrance curb for that compartment, and that is perimeter-welded to the ceiling plate. There are no watertight doors in the AP1000 used for internal flood protection because, as described in DCD subsection 3.4.1.2.2, they are not needed to protect safe shutdown components from the effects of internal flooding.

Conclusion

Therefore, the overall design and construction of the containment compartment walls, ceilings, access openings, and compartment penetrations is expected to result in minimal seepage, well within the assumed in-leakage rate, and there will not be any adverse structural impact from floodup hydrostatic pressure.

RAI Number: 440.106 (Response Revision 1)

Question:

TS LCO 3.4.9 specifies that at least one reactor coolant pump (RCP) shall be in operation with a total flow through the core of at least 10,000 gpm while in MODES 3, 4 and 5, whenever the reactor trip breakers are open. SR 3.4.9.1 requires verification that at least one RCP is in operation at ≥ 25 percent rated speed or equivalent. TS BASES 3.4.9 provide a table of pump percentage rated speeds as a function of number of pumps operating that will deliver the required minimum flow.

The minimum RCS flow limit is an initial condition in the design-basis analysis of a possible boron dilution event to provide a mixing of the inadvertent diluted water with the primary flow. In the safety analysis of boron dilution events during MODES 3, 4, or 5, operation, Section 15.4.6.2 states that the RCS dilution volume is considered well-mixed. The TSs require that when in MODES 3, 4, 5, at least one RCP shall be operable, which provides sufficient flow through the system to maintain the system well-mixed. As shown in Table 5.4-1, the AP1000 RCP design flow is 78,750 gpm per pump.

A. Provide analysis or test data to demonstrate that the 10000 gpm minimum mixing flow specified in LCO 3.4.9 is sufficient to provide well-mixed flow condition in the boron dilution events, to validate the safety analysis assumptions.

B. Provide the characteristics or specification of the variable speed pump design that ensure the minimum mixing flow will be delivered with the pump percentage rated speeds shown in the TS BASES.

Westinghouse Response:

A. NUREG-1431, Rev. 2, Technical Specifications include specific requirements for RCS flow during shutdown MODES to provide adequate heat removal and boron dilution event mixing assumptions. The minimum RCS flow requirements in these approved Technical Specification are satisfied by operation of a single RHR pump.

   The operation of one AP1000 RCP in the specified reduced speed operation, with an RCS flow rate of at least [10,000 gpm], provides significantly greater RCS flow than the single RHR pump flow in current plants. Since AP1000 and current plants have the same boron dilution event design basis and minimum RCS flow requirements for boron mixing, the AP1000 RCS flow is significantly more than required to achieve the required boron mixing.

The flow mixing assumptions for boron dilution analyses for both current plants and for AP1000 are based on NUREG/CR-2733, "Experimental Data Report for LOFT Boron Dilution Experiment L6-6," June 1982, conducted by Idaho National Engineering Laboratory for the U.S. NRC. This testing modeled the Trojan Nuclear Power plant design, assuming a base case RHR flow of 3000 gpm, and a second case with twice the flow of the base case. As stated in Section 3 of EGG-LOFT-5867, "Quick-Look Report on LOFT Boron Dilution Experiment L6-6," May 1982, stated that for both flow cases "the close agreement between the measurement and the core criticality value implies that the reactor vessel volume was well mixed." The Quick-Look Report abstract states that "the results of the boron dilution simulations [for both flow rates] showed that the direct flow path volume was well mixed and the boron concentration as a function of time was characterized by the perfect mixing model."

B. The AP1000 RCPs are described in DCD Sections 5.1.3.3 and 5.4.1. The RCPs are single-stage, canned motor centrifugal pumps. A variable frequency drive provides speed control to reduce RCP speed and motor power requirements during pump startup from cold conditions below 450°F. The variable speed controller is only operated in Mode 5 with the reactor trip breakers open. During other plant conditions including power operation, the variable frequency drive is isolated from the RCP so that the RCP operates at a constant (full) speed.

As discussed in the Bases for LCO 3.4.9 and for SR 3.4.9.1, the minimum flow requirement of [10,000 gpm] assures adequate mixing of the RCS in the event of a boron dilution event. SR 3.4.9.1 requires confirming RCS flow for the RCP combination and speed specified (one RCP operating at 25% speed), although the minimum flow is satisfied for the various pump combinations and speeds discussed in the Bases for SR 3.4.9.1.

As indicated in Table 5.4-1, the best estimate RCP design flow (during constant, full-speed operation) is 78,750 gpm per pump, or a total reactor vessel flow of 315,000 gpm with all four pumps operating. This flow can be used to calculate the pump flow at other lower operating speeds.

For a variable-speed centrifugal pump, the flow rate change is directly proportional to the pump rotational speed, and the head produced by the pump is proportional to the square of the pump speed change. Therefore, if the pump speed is reduced to the speeds indicated in the table below from Bases for Surveillance Requirement 3.4.9.1, the flow can be calculated based on the proportional change in pump speed. The table below calculates flow changes considering only changes in RCP speed, and assuming 4 RCPs continue operating. Calculating pump flow this way is conservatively low since it ignores the significant reduction in RCS system flow resistance when RCPs are stopped, and it also simplifies the approximation of RCS flow for this RAI response.

Number of RCPs	% Rated Speed	Calculated Flow (gpm, based on 4 RCPs running)
1	25%	19,688 x 1 = **19,688***
2	20%	15,750 x 2 = 31.500
3	15%	11,813 x 3 = 35,438
4	10%	7,875 x 4 = 31,500

* The first RCP combination is used as the flow value for SR 3.4.9.1 since it is the normal minimum pump flow combination expected during plant cooldown prior to securing RCPs at about 160°F.

The SR test condition provides significantly more flow than the required minimum flow for boron mixing (as discussed in Item A above) and meets the LCO requirements, even with no benefit from the reduction in RCS system flow resistance. The RCP flow provides mixing in the reactor vessel and core, the operating loop, and the idle loop.

As shown in the table above, the large RCS flow rates for the other three operating conditions discussed in the Bases for SR 3.4.9.1 significantly exceed both the boron dilution minimum flow mixing requirements and the specified LCO 3.4.9 flow requirements. These flows are also conservatively calculated without consideration of the RCS system flow resistance reduction.

NRC Follow-on Question:

As stated in the paragraph 2 of Item B of the response to RAI 440.106, the minimum flow of 10,000 gallons-per-minute (gpm) (less than a full reactor coolant pump [RCP] flow) required in technical specification (TS) limiting condition for operation (LCO) 3.4.9 assures adequate mixing of the reactor coolant system (RCS) in the event of a boron dilution event. Item A of the RAI response indicates that the required flow rates are based on NUREG/CR-2733, "Experimental Data Report for LOFT Boron Dilution Experiment L6-6." The LOFT test data show a residual heat removal system (RHR) flow of 3000 gpm provides a sufficient flow for the adequate mixing of the fluid in the reactor vessel in the LOFT facility. The staff notes that the LOFT was conducted in a small- scaled test facility. It is not clear how the LOFT test data are applied to AP1000 in deriving the minimum required flow for supporting the perfect mixing model used in the analysis for AP1000. The staff also notes that the minimum flow requirement for AP1000 deviates from the basis for the required flow for AP600, which requires (by LCO 3.4.9) at least one full RCP flow (about 52,000 gpm)

(A) Explain how the required minimum flow of 10,000 gpm is derived. If LOFT data are used, explain how the effect of the geometry differences in the LOFT facility and AP1000, and the scaling factors for both systems are determined.

(B) Justify why the basis of the required flow for AP1000 (which deviates from that of the AP600 design) is acceptable.

Westinghouse Additional Response:

Note: At the time when the above question was written, the associated technical specification was "3.4.9 Minimum RCS Flow." Since that time Revision 3 of the DCD has been issued, and the number of that technical specification has been changed to 3.4.8.

(A) The process of selecting 10,000 gpm as the minimum required core flow to support the boron mixing assumptions used in the safety analysis for the boron dilution event included general consideration of the results reported in NUREG/CR-2733, "Experimental Data Report for LOFT Boron Dilution Experiment L6-6," June 1982 (conducted by the Idaho National Engineering Laboratory for the U.S. NRC). However, there was not an explicit scaling factor based assessment to quantify a precise comparison between the physical characteristics of the AP1000 and those of the LOFT test facility. Such an assessment was not considered necessary. Rather the basis for the choice of the 10,000 gpm minimum flow requirement was a qualitative assessment of information from a variety of sources, including the LOFT test. Details supporting the acceptability of the 10,000 gpm value are provided below, under Item (B).

(B) As discussed in EGG-LOFT-5867, "Quick-Look Report on LOFT Boron Dilution Experiment L6-6," May 1982, the key parameters of this specific series of tests were scaled based on the characteristics of the Westinghouse 4-loop Trojan PWR. While the specific scaling of that test is not based on the AP1000 design, the overall conclusions of the LOFT testing with regard to boron mixing are no less applicable to the AP1000 than to operating PWRs that differ from the exact configuration of the Trojan reactor vessel. It should also be noted that with its four cold legs, the general configuration of the AP1000 inlet plenum region is similar to that of a 4-loop plant.

In order to assess the effect of Reynolds number/RHR flow rate on the mixing that occurs, the LOFT test considered two low–pressure injection system flow rates that were scaled to provide equivalence to 3000 and 6000 gpm RHR flow rates in the Trojan plant. Typical RHR related technical specifications that are intended to ensure adequate boron mixing in current Westinghouse designed plants, allow operation in the applicable modes with a single operating RHR pump. The 3000 gpm RHR flow rate used as the reference value for the LOFT test was not only appropriate for the Trojan plant, but is a representative single RHR pump value for operating Westinghouse plants.

In a subsequent paper documenting results from the LOFT boron dilution tests, (EGG-M-03783, DE83 013666, "PWR Response to an Inadvertent Boron-Dilution Event, Presented at the Third Multiphase Flow and Heat-Transfer Symposium Workshop, April 18-20, 1983) the following conclusions were reached with respect to the LOFT test mixing results. For the base 3000 gpm RHR flow equivalent case it was concluded

"...that the fluid volume in the reactor vessel was well mixed and that the assumption of perfect mixing, though not strictly correct, is adequate for calculational purposes." For the 6000 gpm RHR flow equivalent case, the reported test results showed an even closer approach to perfect mixing.

From the above discussion, it can be concluded that actual results from the LOFT boron dilution tests have confirmed that a test RHR flow scaled to be equivalent to 3000 gpm in a representative Westinghouse plant produced mixing results very close to those associated with perfect mixing. Doubling the test flow to simulate 6000 gpm in an operating plant produced increased mixing that more closely approached perfect mixing. These results support typical plant technical specifications that generally accept an RHR flow in the vicinity of 3000 gpm as being sufficient to justify the perfect mixing assumption modeled in the boron dilution safety analyses.

The selection of 10,000 gpm core flow as the minimum acceptable core flow to preserve the required mixing in the RCS is somewhat of an arbitrary choice. However, this value is well in excess of the flow rates considered in the LOFT test and currently accepted as providing adequate mixing in operating plants.

With a single RCP running, a significant portion of the flow from the loop with an operating RCP passes into the reactor vessel inlet plenum and then flows "backwards" through the inactive loop, thereby bypassing the core. However, even flow that initially bypasses the core contributes to the overall mixing with the RCS. Though Technical Specification 3.4.8 states that the minimum flow requirement is 10,000 gpm through the core, the associated surveillance requirement (SR 3.4.8.1) places an operating speed requirement on a single operating RCP. Specifically, the surveillance requirement dictates that in order to be considered as an operating RCP, the single pump involved must be operating at a minimum of 25% rated speed.

As documented in the original Westinghouse response to RAI 440.106, a single operating RCP at 25% speed is actually predicted to produce 19,688 gpm. This means the total minimum RCP flow is almost twice the stated 10,000 gpm core flow requirement and far in excess of the 3000 gpm value that is typically applied to operating plants. Again, this information supports the conclusion that the flow requirements of AP1000 Technical Specification 3.4.8 are sufficient to validate the boron mixing assumptions associated with the analysis for the boron dilution event found in Section 15.4.6 of the AP1000 Design Control Document (DCD).

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 720.035 (Response Revision 1)

Question:

In Chapter 26 on the Protection and Safety Monitoring System (PMS), in Chapter 27 on the Diverse Actuation System (DAS), and in Chapter 28 on the Plant Control System (PLS), the following statement is made: "Because of the rapid changes that are taking place in the digital computer and graphic display technologies employed in the modern human systems interface, design certification of the AP1000 focuses upon the process used to design and implement instrumentation and control systems for the AP1000, rather than on the specific implementation." To be able to take advantage of such changes in technology, design options in additions to the ones used in the AP600 design certification are proposed for the safety-related PMS and the non-safety related DAS and PLS. For the safety-related PMS, the option to use the Common Qualified Platform (Common Q) is proposed. For the non-safety-related DAS and PLS, the option to use commercial off-the-shelf hardware and software which will be current at the time of construction is proposed. Please provide more detailed information regarding the implementation of the proposed options by responding to the following questions:

A. Regarding the PMS, it is stated that the AP600 Instrumentation and Control (I&C) functional requirements, which have received design certification, will be retained to the maximum extent compatible with the Common Q hardware and software. Also, it is stated that although the details of the AP1000 PRA model follow the AP600 design, the Common Q hardware and software provide a degree of redundancy that is equivalent to the redundancy modeled in the AP1000 PRA. Please explain the process that will be used to verify that a PMS designed with the "Common Q" option will have equivalent or better reliability with the system modeled in the PRA. Also, please explain how the introduction of the "Common Q" option will affect important PRA-based insights about the PMS, such as the ones identified during the AP600 PRA review (i.e., the design certification information "PRA-based insights" documented in Table 19.59-29 of the AP600 DCD).

B. Regarding the non-safety-related DAS and PLS, it is stated that the AP1000 PRA is based on "one possible ....configuration designed to meet the requirements of DCD Chapter 7" and that "the functional requirements and the degree of redundancy modeled in the PRA are representative of the expected final ....design." Please explain the process that will be used to verify that the DAS and PLS ,designed with the "commercial off-the-shelf hardware and software current at the time of construction" option, will have equivalent or better reliability with the systems modeled in the AP1000 PRA. Also, please explain how the introduction of such an option will affect important PRA-based insights about the DAS and PLS, such as the ones identified during the AP600 PRA review (i.e., the design certification information "PRA-based insights" documented in Table 19.59-29 of the AP600 DCD).

Westinghouse Response:

The responses to questions A and B are given below. Note that in general, all risk-important components, including those in PMS, DAS and PLS systems, of an operating AP1000 plant will be subject to the maintenance rule and their performance will be monitored and kept within the established goals. Moreover, the maintenance rule also requires assessment of plant configuration as components become out-of-service (planned or forced) on a continuous basis, further strengthening the adherence to the component performance goals.

A. Although the PMS as a system has high importance, the PRA results are not sensitive to small changes in PMS failure probabilities. For example, a factor of two increase in PMS failure probability produces less than a one-percent change in plant CDF for at-power events and the PRA results still meet the NRC safety goals. This low level of sensitivity is a result of the following:

- The PMS, using either platform, is designed to be a highly reliable system.
  - The PMS has four redundant divisions.
  - Redundancy is provided within each division.
  - The PMS uses highly reliable components.
  - The software is developed, verified and validated using processes that conform to applicable standards for safety applications.
  - Modular design enhances the rapid isolation and repair of failures.
  - The PMS has continuously running diagnostic features that will alert the operations and maintenance staffs promptly of component malfunctions that occur. This assures that malfunctions will be repaired promptly, minimizing the amount of time that the plant has to operate with malfunctioning components.
  - Circuit isolation is used to electrically isolate segments of the instrumentation and control architecture and to prevent propagation of electrical faults.
  - Physical separation is provided between the four redundant divisions of equipment for the PMS, which in turn, is separated from nonsafety systems such as the PLS.
  - PMS equipment is qualified to environmental requirements, including temperature, humidity, vibration/seismic, electromagnetic interference/radio frequency interference (EMI/RFI), and surge withstand criteria commensurate with its safety classification and intended usage.

- The AP1000 is designed using fail-safe design features.

- The AP1000 plant is designed with diverse back up for important functions.

The general architecture of the Common Q PMS is similar to that modeled in the AP1000 PRA and includes the features listed above. Based on the Common Q characteristics and AP1000 PMS design features described below, the "PRA-based insights" documented in response to RAI 720.038 are applicable to the AP1000 PMS using the Common Q product.

The AC160 and flat panel display equipment used in the Common Q product have a commercial experience base that assures that they are reliable and predictable for process control and monitoring applications.

The AP1000 Common Q based PMS application includes redundancy features to help assure system reliability, including:

- Redundant inputs for sensors
- Redundant processing of algorithms and channel bistable functions
- Inter-communication of bistable statuses so that each channel can use the information from all channels in its coincidence logic
- Redundant coincidence logic subsystems in each channel
- Actuation of any required safety function by the output from either of the redundant coincidence logic systems in each channel
- Redundant network busses in each channel to communicate information between PMS subsystems.

This summary indicates that the AP1000 with the Common Q option is expected to meet the NRC safety goals because the functional requirements and the basic design features are essentially the same. It is unlikely that modeling Common Q in the PRA would have a significant impact on the PRA results and, as stated above, even a factor of two increase in PMS failure probability has a minimal impact on the PRA.

B. The manual DAS controls are implemented in a manner that bypasses the signal processing equipment. Manual DAS controls are also subject to Technical Specification requirements. Automatic DAS is one of the systems that are subject to availability controls of DCD Section 16.3. Both DAS and PLS have medium importance in plant risk as shown by the sensitivity analyses performed for AP1000. Since PLS is a normally operating system, its reliability and availability are crucial to plant performance, and are expected to be kept at prescribed goal levels. DAS is a standby system and its reliability and availability are controlled by the availability controls. The PRA results and insights derived from the current AP1000 PRA are discussed in response to RAI 720.038.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

NRC Additional Comments:

The staff requested Westinghouse to explain the process that will be used to verify that a PMS designed with the "Common Q" option will have equivalent or better reliability than the system modeled in the PRA and how the introduction of the "Common Q" option will affect important PRA-based insights about the PMS. Westinghouse responded that "the PRA results are not sensitive to small changes in PMS failure probabilities" and "The general architecture of the Common Q PMS is similar to that modeled in the AP1000 PRA and includes the features listed above." The staff needs further clarification, including a direct comparison of the design features found to be important in the PRA between the "Common Q" option and the PMS modeled in the PRA. In addition, a direct comparison of the "design certification requirements" for the two cases can help clarify the issue. Based on the results of these comparisons, the identification of new "design certification requirements" to ensure PMS reliability may be required. The same comments apply also for DAS and PLS designed with the "commercial off-the-shelf hardware and software current at the time of construction" option.

Westinghouse Additional Response:

A comparison chart comparing Common Q to the design features found to be important in the PRA and the design certification requirements follows. As shown in the chart, most of the features are the same for both the PRA model and Common Q. Where there are differences, either the difference should not affect reliability or the Common Q feature will result in higher reliability.

The DAS and PLS will be designed to meet design reliability requirements as assumed in the PRA. This is no change from AP600.

Comparison of AP1000 Protection System Design Features in the PRA Model to Common Q			
Design Feature	**AP1000 Design Features Modeled in the PRA**	**AP1000 Common Q Design Features**	**Comments**
Number of divisions	4 – Reactor trip and ESF actuation 2 – Post-accident monitoring	4 – Reactor trip and ESF actuation 2 – Post-accident monitoring	
Voting logic	2-out-of-4, becoming two-out-of-three during testing. Subsequent channel bypass results in one-out-of-two logic and upon additional bypass, the system trips.	2-out-of-4, becoming two-out-of-three during testing. Subsequent channel bypass is not allowed. An additional failed channel must be put into partial trip, resulting in one-out-of-two logic for one channel in partial trip and one channel in bypass.	If two or more redundant channels need to be removed from service (bypassed), the PRA model (Eagle) handles the logic changes automatically. Common Q requires the operator to place the second redundant channel in partial trip. This difference should not affect reliability.
Power source	Class 1E dc and UPS System	Class 1E dc and UPS System	4 independent divisions of electrical power
Testing	Automatic testing every 3 months	Combination of automatic and manual testing every 3 months	The Common Q testing philosophy is different from that modeled in the PRA. The PRA assumes a thorough front-to-back test (including physical signal injection) every three months. Common Q uses fully redundant inputs and continuous (5-minute) comparisons of the two signal streams. The Common Q approach should result in better reliability because failures will be detected more quickly.
Fail-safe design features	Yes	Yes	

Comparison of AP1000 Protection System Design Features in the PRA Model to Common Q			
Design Feature	**AP1000 Design Features Modeled in the PRA**	**AP1000 Common Q Design Features**	**Comments**
Channel checks	Channel check every 24 hours per Tech Specs.	Channel check every 24 hours per Tech Specs.	
Separation and isolation	Separation and isolation are provided between Class 1E divisions and between Class 1E and non-Class 1E.	Separation and isolation are provided between Class 1E divisions and between Class 1E and non-Class 1E.	
Reactor trip circuit	Dynamic trip bus	Relay logic and watchdog timers	In the event of a processor lock-up, either the Dynamic Trip Bus (PRA model) or the watchdog timer (Common Q) will trip the reactor. This difference should not affect reliability.
Reactor trip breakers	4 divisions of reactor trip arranged in a two-out-of-four logic.	4 divisions of reactor trip arranged in a two-out-of-four logic.	
Self diagnostics	Yes	Yes	
Mean repair time	Assumed mean repair time of 4 hours	Modular design with mean repair time expected to be 4 hours	
Sensor input redundancy	One input for each sensor	Redundant sensor inputs	Common Q is more tolerant to input card failures and, because of the input redundancy and self-check features, will detect input card failures more quickly.

Comparison of AP1000 Protection System Design Features in the PRA Model to Common Q			
Design Feature	**AP1000 Design Features Modeled in the PRA**	**AP1000 Common Q Design Features**	**Comments**
Processing redundancy	Functionally-diverse reactor trip logic processing, fully-redundant ESF actuation logic processing	Fully-redundant logic processing for both reactor trip and ESF actuation	Common Q is more tolerant to processor failures and, because of the processor redundancy and self-check features, will detect processor failures more quickly.
Communications	Redundant data highways using fiber optics	Redundant data highways using fiber optics	The communications topology for Common Q is different from the PRA model, but this difference should not affect reliability.
Environmental qualification	Qualified to environmental requirements, including temperature, humidity, vibration/seismic, EMI/RFI, and surge withstand criteria	Qualified to environmental requirements, including temperature, humidity, vibration/seismic, EMI/RFI, and surge withstand criteria	
Temperature qualification	Qualified to operate with loss of normal HVAC, using passive heat sinks for cooling.	Qualified to operate with loss of normal HVAC, using passive heat sinks for cooling.	
Component reliability	Highly reliable	Highly reliable	
D-RAP	Included in D-RAP	Included in D-RAP	
Software verification and validation (V&V)	The PMS software is designed, tested, and maintained to be reliable under a controlled V&V program.	The PMS software is designed, tested, and maintained to be reliable under a controlled V&V program.	

Comparison of AP1000 Protection System Design Features in the PRA Model to Common Q			
Design Feature	**AP1000 Design Features Modeled in the PRA**	**AP1000 Common Q Design Features**	**Comments**
Diversity	Diverse from DAS	Diverse from DAS	
Fire separation	The PMS cabinets, in which the automatic functions are housed, are located in fire areas separate from the main control room. In each division, the PMS cabinets, in which the automatic functions are housed, are located in fire areas separate from the redundant cabinets in the other three divisions.	The PMS cabinets, in which the automatic functions are housed, are located in fire areas separate from the main control room. In each division, the PMS cabinets, in which the automatic functions are housed, are located in fire areas separate from the redundant cabinets in the other three divisions.	
Prevention of spurious squib valve actuation	Spurious actuation of squib valves is prevented by: 1) the use of a squib valve controller circuit which requires multiple hot shorts for actuation, 2) physical separation of potential hot short locations (e.g., routing of squib valve cables in low voltage cable trays), and 3) provisions for operator action to remove power from the fire zone.	Spurious actuation of squib valves is prevented by: 1) the use of a squib valve controller circuit which requires multiple hot shorts for actuation, 2) physical separation of potential hot short locations (e.g., routing of squib valve cables in low voltage cable trays), and 3) provisions for operator action to remove power from the fire zone.	

Design Control Document (DCD) Revision:

None

PRA Revision:

None

Attachment 3

WCAP-15993, Rev. 1

"Evaluation of the AP1000 Conformance to Inter-System Loss-of-Coolant Accident Acceptance Criteria"

dated March 2003

Westinghouse Non-Proprietary Class 3

WCAP-15993
Revision 1

March 2003

Evaluation of the AP1000 Conformance to Inter-System Loss-of-Coolant Accident Acceptance Criteria

Westinghouse

AP1000 DOCUMENT COVER SHEET

TDC: ______ Permanent File: ______ S ______

RFS#: ______ RFS ITEM #: ______

AP1000 DOCUMENT NO.	REVISION NO.		ASSIGNED TO
APP-GW-GLR-002	1	Page 1 of 32	W- WINTERS

ALTERNATE DOCUMENT NUMBER: WCAP-15993, Revision 1 WORK BREAKDOWN #: GW

ORIGINATING ORGANIZATION: Westinghouse Electric Company, LLC

TITLE: **Evaluation of the AP1000 Conformance to Inter-System Loss-of-Coolant Accident Acceptance Criteria**

ATTACHMENTS:	DCP #/REV. INCORPORATED IN THIS DOCUMENT REVISION:
CALCULATION/ANALYSIS REFERENCE:	

ELECTRONIC FILENAME	ELECTRONIC FILE FORMAT	ELECTRONIC FILE DESCRIPTION
Appgwglr002r1.doc		

ORIGINATOR T. K. Meneely	SIGNATURE/DATE Timothy K. Meneely 3/10/03	
REVIEWERS	SIGNATURE/DATE	
INDEPENDENT VERIFIER M. M. Corletti	SIGNATURE/DATE 3/10/03	VERIFICATION METHOD
AP1000 RESPONSIBLE MANAGER R. P. Vijuk	SIGNATURE* R P Vijuk	APPROVAL DATE 3/12/03

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

WCAP-15993
Revision 1

Evaluation of the AP1000 Conformance to Inter-System Loss-of-Coolant Accident Acceptance Criteria

T. K. Meneely

March 2003

AP1000 Document: APP-GW-GLR-002, Revision 1

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1 INTRODUCTION

1.1 BACKGROUND

In conducting studies directed at finding vulnerabilities of pressurized water reactor (PWR) plants to inter-system loss-of-coolant accidents (ISLOCAs), the Nuclear Regulatory Commission (NRC) staff concluded that the core damage frequency caused by ISLOCAs could be substantially greater than previous Probabilistic Risk Assessment (PRA) estimates.[1] In NRC Information Notice 92-36 (Reference 1), the NRC staff indicated that these PRAs have typically been limited to modeling ISLOCA sequences that include only the catastrophic failures of check valves that isolate the Reactor Coolant System (RCS) from low-pressure systems. Also, the PRAs included little consideration of human errors leading to an ISLOCA and the effects of the accident-caused harsh environment or flooding on plant equipment and recovery activities.

The results of these NRC studies have suggested that ISLOCA precursors most likely would be initiated by human errors or because of procedural deficiencies. This may be attributed to the general lack of awareness of the possibility or consequences of an ISLOCA.

The NRC has developed a position on design requirements necessary to minimize the potential for ISLOCAs. The staff position is addressed in numerous NRC documents, including References 1 through 7. Westinghouse has evaluated the AP1000 design and concludes that it complies with the stated NRC position.

1.2 OBJECTIVE

The purpose of this report is to perform a systematic evaluation of the systems that interface with the RCS and to demonstrate that the design of the systems meets the ISLOCA acceptance criteria, which are described in section 2.2 of this report.

1.3 SCOPE

The scope of this evaluation is applicable to the AP1000 systems and subsystems that interface directly or indirectly with the RCS and are susceptible to ISLOCA challenges.

1. AP1000 PRA results show that ISLOCAs provide only a minor contribution to core damage frequency. In current calculations, this contribution accounts for approximately 5.0E-11 per reactor year, which is less than one-tenth of one percent of the overall AP1000 core damage frequency at-power, calculated to be approximately 2.4E-7 per reactor year.

2 DESIGN REQUIREMENTS AND APPROACH

2.1 ISLOCA DEFINITION

An ISLOCA is defined in NRC Information Notice 92-36 (Reference 1) as a class of events in which a break occurs outside containment in a system connected to the RCS, causing a loss of primary system inventory outside containment. This is interpreted as a beyond-design-basis event for systems connected directly or indirectly to the RCS. The pressurization pathway can be established by an inadvertent opening of a valve or valves, a failure of containment isolation, or the postulation that valves are fully open (for example, check valves). This interpretation is believed to address all sources that may challenge low-pressure systems. Based on this definition of an ISLOCA, an evaluation was performed to assess the ability of the AP1000 design to withstand an overpressure event.

2.2 ISLOCA ACCEPTANCE CRITERIA

The design of systems that interface with the RCS is evaluated against acceptance criteria consistent with the following NRC guidance provided in SECY-90-016 (Reference 5).

- All systems and subsystems connected to the RCS are to be designed to withstand the full RCS pressure to the extent practicable.
- Systems that are not designed to full RCS pressure should include:
  - the capability for leak-testing of the pressure isolation valves
  - valve position indication that is available in the control room when isolation valve operators are deenergized, and
  - a high-pressure alarm to warn control room operators when rising reactor coolant pressure approaches the design pressure of attached low-pressure systems and both isolation valves are not closed.
- Systems not designed in the above methods should include other proper design features to prevent ISLOCAs to the extent practicable.

2.3 ISLOCA EVALUATION PROCESS

The systematic evaluation performed for this study of ISLOCA challenges and the subsequent determination of appropriate design responses can be summarized in the following steps:

1. The AP1000 RCS piping and instrumentation diagram (P&ID) was reviewed to identify systems or subsystems that directly interface with the RCS. The P&IDs of these primary interfacing systems were also reviewed to identify secondary interfacing systems or subsystems that directly interface with the primary interfacing systems.

2. The design pressure of each of the primary and secondary interfacing systems was identified and categorized as follows:

   A – Design pressure = RCS Design Pressure
   B – Ultimate Rupture Strength (URS) = RCS Design Pressure
   C – Low-Pressure System

3. Any system or subsystem that interfaces with a primary interfacing system categorized as A or B above was then itself evaluated as a primary interfacing system for the following reason: If a primary interfacing system is designed for full RCS pressure, then it can be considered (for this study) an extension of the reactor coolant pressure boundary (RCPB), and therefore, any system interfacing with it should be subjected to the ISLOCA evaluation criteria.

   Systems interfacing with a category C system were not evaluated as primary interfacing systems because it was assumed that the justification and design response for the category C system would also protect any system connected to that system.

   For each interfacing system or subsystem categorized as B or C above, justification for ISLOCA compliance is identified and categorized as follows:

   (1) All parts of system or subsystem are located inside the containment.

   (2) System or subsystem is designed to a URS at least equal to the full RCS pressure.

   (3) System or subsystem includes the following design features:

      – the capability for leak-testing of the pressure isolation valves

      – valve position indication that is available in the control room when isolation valve operators are deenergized, and

      – a high-pressure alarm to warn control room operators when rising reactor coolant pressure approaches the design pressure of attached low-pressure systems and both isolation valves are not closed.

   (4) System or subsystem includes other design features specific to them that prevent an ISLOCA to the extent practicable. These design features are discussed in section 3 of this report.

4. A design evaluation is performed for all category B and C primary and secondary interfacing systems with compliance justification other than (1). Each interface in the pressurization pathways was analyzed relative to the ISLOCA acceptance criteria.

Tables 2-1 through 2-11 summarize the results of the evaluation process described above. The first system in each table is the primary interfacing system. The remaining systems in each table are the secondary interfacing systems that interface with that primary system. Section 3 of this report contains design evaluations for the category B or C systems identified with justifications that do not include justification (1) (system is located entirely inside containment).

Table 2-1 Normal Residual Heat Removal System

Interfacing System	Design Pressure(1)	Justification(2)				Design Evaluation (Section)
		1	2	3	4	
Normal Residual Heat Removal System (RNS) Pump seal	A/B C		X	 X		3.1
Passive Core Cooling System (PXS) test header	A	X	X			
Chemical and Volume Control System (CVS) purification return line	A	X	X			
PXS direct vessel injection line	A	X	X			
In-containment refueling water storage tank (IRWST) sparger	C	X				
CVS purification line	A	X	X			

1. See subsection 2.3.2 for an explanation of design pressure codes.
2. See subsection 2.3.3 for an explanation of justification codes.

Table 2-2 Chemical and Volume Control System Purification Loop

Interfacing System	Design Pressure	Justification				Design Evaluation (Section)
		1	2	3	4	
CVS purification loop	A	X				
RNS discharge and return headers	A	X				
Solid Radwaste System (WSS)	C				X	3.5
Containment sump	C	X				
Demineralized Water Transfer and Storage System (DWS)	C	X			X	3.6
Primary Sampling System (PSS)	A		X			3.4
CVS makeup line Makeup pump suction line	A C		X		 X	3.3
Hydrogen addition line	A		X			
RCS pressurizer spray	A	X				

Table 2-3 Chemical and Volume Control System Letdown Line						
Interfacing System	**Design Pressure**	**Justification**				**Design Evaluation (Section)**
		1	**2**	**3**	**4**	
CVS letdown line	C			X	X	3.2
Liquid Radwaste System (WLS) degasifier	C			X	X	3.2
WLS effluent holdup tank	C			X	X	3.2

Table 2-4 Chemical and Volume Control System Makeup Pump Discharge Line						
Interfacing System	**Design Pressure**	**Justification**				**Design Evaluation (Section)**
		1	**2**	**3**	**4**	
CVS makeup pump discharge line	A		X			
PXS test header	A	X	X			
Spent fuel pool	C				X	3

Table 2-5 Chemical and Volume Control System Makeup Pump Suction Line						
Interfacing System	**Design Pressure**	**Justification**				**Design Evaluation (Section)**
		1	**2**	**3**	**4**	
CVS makeup pump suction line	C				X	3.3
Spent fuel pool	C				X	3
Waste holdup tank	C				X	3.3
Demineralized water storage tank	C				X	3.3

Table 2-6 Chemical and Volume Control System Hydrogen Injection Line						
Interfacing System	**Design Pressure**	**Justification**				**Design Evaluation (Section)**
		1	**2**	**3**	**4**	
CVS hydrogen injection line	A		X			

Table 2-7 Primary Sampling System						
Interfacing System	**Design Pressure**	**Justification**				**Design Evaluation (Section)**
		1	**2**	**3**	**4**	
PSS	A		X			3.4
Grab sample panel	C				X	
DWS	C	X			X	3.6
PXS accumulators	C	X				
PXS core makeup tanks (CMTs)	A	X	X			
CVS demineralizers	A	X	X			
WLS degasifier	C				X	3.4

Table 2-8 Passive Core Cooling System Core Makeup Tanks						
Interfacing System	**Design Pressure**	**Justification**				**Design Evaluation (Section)**
		1	**2**	**3**	**4**	
PXS CMTs	A	X	X			
Reactor coolant drain tank (RCDT)	C	X				
PSS	A		X			
CVS makeup line	A		X			

Table 2-9 Passive Core Cooling System Direct Vessel Injection Line						
Interfacing System	**Design Pressure**	**Justification**				**Design Evaluation (Section)**
		1	**2**	**3**	**4**	
PXS direct vessel injection line	A	X				
IRWST	C	X				
PXS accumulators	C	X				

Table 2-10 Passive Residual Heat Removal Heat Exchangers						
Interfacing System	**Design Pressure**	**Justification**				**Design Evaluation (Section)**
		1	**2**	**3**	**4**	
Passive residual heat removal (RHR) heat exchangers	A	X	X			

Table 2-11 Passive Core Cooling System Test Header

Interfacing System	Design Pressure	Justification				Design Evaluation (Section)
		1	2	3	4	
PXS test header	A	X	X			
CVS makeup line	A		X			
PXS accumulators	C	X				
RNS suction and discharge RCPB valves	A	X	X			
WLS RCDT	C	X				
PXS CMTs	A	X				
IRWST	C	X				

3 DESIGN EVALUATIONS

This section presents evaluations of the systems and subsystems identified in section 2.3 as requiring evaluation with regard to ISLOCA criteria. These systems or subsystems are connected directly to the RCS, or connect to high-pressure systems that connect directly to the RCS during some mode of operation, such that they must be evaluated for susceptibility to an ISLOCA. Based on the results of the evaluation process described in section 2.3, the following systems were selected for a detailed design evaluation:

- Normal Residual Heat Removal System (RNS)
- CVS letdown line to the WLS
- CVS makeup pump suction line
- Primary Sampling System (PSS)
- Solid Radwaste System (WSS)
- Demineralized Water Transfer and Storage System (DWS)

This section provides a detailed evaluation of each of these systems and subsystems. Each subsection is structured as follows:

- Description of Primary System Interface – A brief overview of the interfacing system under evaluation, the potential ISLOCA pathway, and operating conditions and failures necessary to create the ISLOCA pathway.

- Design Evaluation – An evaluation of the design against the ISLOCA criteria, and a description of any additional design features that address the ISLOCA issue.

- Justification of Design – A summary of the adequacy of the AP1000 system or subsystem design with respect to potential ISLOCA challenges.

In addition to describing systems under evaluation, these sections describe portions of systems designed to full RCS pressure, designed to a URS equal to full RCS pressure, or designed for low pressures. A system or portion of a system designed to full RCS pressure will have a design pressure of at least 2485 psig. A system or portion of a system designed to an URS equal to full RCS pressure will have a design pressure of at least 900 psig. A low-pressure system will have a design pressure less than 900 psig. These sections also describe piping lines from point A to point B. When this term is used, it can be assumed that all piping, valves, fittings, components, and instrument lines located in a "line" from point A to point B are designed to the pressure of the "line," unless otherwise specified.

SECY-90-016 (Reference 5) provides practical guidance in upgrading systems to URS design pressure. As discussed in Reference 10, it is impractical to design the large, low-design-pressure tanks and tank structures that are vented to the atmosphere to URS design pressure. Tanks included in this category are as follows:

- Spent fuel pool and fuel transfer canal
- CVS boric acid tank
- Demineralized water storage tank

- WLS effluent holdup and monitor tanks
- WLS waste holdup and monitor tanks

Table 3-1 provides the approximate sizes of these tanks to show the impracticality of increasing their design pressure. Increasing the design pressure of these tanks to the URS value would result in an unnecessary dollar cost burden. In addition, the tanks that contain radioactive waste are typically designed with features such as sloped bottoms to reduce crud deposition. Such features cannot be used in tanks designed to high pressure. Tanks such as the spent fuel pool and fuel transfer canal have no top cover and are open to the auxiliary building so that their pressure cannot be increased above the static head for which they are designed.

As discussed in the following evaluations, interfacing systems or subsystems that connect directly to an atmospheric tank are excluded from further ISLOCA consideration. This is limited to the piping connected directly to the atmospheric tank, up to the first isolation valve other than a locked-open, manual isolation valve. Designing these portions of the system to a higher pressure would provide no practical benefit. Designing these systems to full RCS pressure would offer no reduction in RCS inventory lost in the event that these lines were aligned to the RCS at full RCS pressure.

Other justifications for designing interfacing systems to less than full RCS pressure are provided on a case-by-case basis.

Table 3-1 AP1000 Low-Pressure Tanks Not Designed to URS Design Pressure

Tanks	Volume (gallon)
Spent fuel pit	190,000
Fuel transfer canal	61,000
CVS boric acid tank	70,000
Demineralized water storage tank	150,000
WLS effluent holdup tank	28,000
WLS waste holdup tank	15,000
WLS waste monitor tank	15,000

3.1 NORMAL RESIDUAL HEAT REMOVAL SYSTEM

3.1.1 Description of Primary System Interface

The RNS is the nonsafety-related system that provides shutdown cooling for the RCS. During normal shutdown operations, the RCS is cooled and depressurized to the RNS cut-in temperature and pressure using the steam generators as a heat sink, and using pressurizer spray to reduce RCS pressure. Once RCS pressure and temperature have been reduced to the conditions for RNS initiation, the RNS suction line isolation valves are opened, and the RNS pumps are started to provide shutdown cooling. Cooldown to refueling conditions continues with the RNS operating in this mode of shutdown cooling. Design Control Document (DCD) subsection 5.4.7 provides a complete description of the various functions and operations associated with the RNS. Figure 3-1 is the RNS P&ID modified to clearly indicate all high-pressure/low-pressure interfaces.

The RNS takes suction from an RCS hot leg and discharges to the reactor vessel direct vessel injection (DVI) lines. The lines represent the two potential paths of overpressurization for the RNS. As shown in Figure 3-1, the RNS suction line contains three normally closed isolation valves in series, with a design pressure equal to RCS design pressure. This represents the first potential pressurization pathway. The RNS inner and outer suction line isolation valves (V001A and B, and V002A and B) are RCPB valves. These valves have power removed at the valve motor control centers and are interlocked so that they cannot be opened unless RCS pressure is reduced to a pressure within the design pressure of the RNS (450 psig). The third normally closed isolation valve (V022) is designed to full RCS pressure and is a containment isolation valve. Overpressurization would occur only if either all three motor-operated gate isolation valves leaked excessively, or if the valves were inadvertently opened with the RCS pressure above the design pressure of the low-pressure portions of the RNS.

The second potential overpressurization pathway for the RNS is via the discharge branch lines, which each connect to a DVI line. Each RNS branch line contains two normally closed check valves that are RCPB valves, and as such, are designed to the RCS design pressure. The branch lines then connect to a common header that penetrates containment. The common header contains two containment isolation valves, a check valve inside containment (V013), and a motor-operated gate valve outside containment (V011). All the valves and piping up to and including the motor-operated gate valve are designed to full RCS pressure. Overpressurization would occur only if three check valves and the motor-operated gate isolation valve (in series) all leaked excessively.

3.1.2 Design Evaluation

The RNS suction line from the RCS hot leg to the outside-containment isolation valve (V022) is designed to full RCS pressure. Likewise, the RNS discharge lines from the DVI line back to the outside-containment isolation valve (V011) is designed to full RCS pressure. The portions of the RNS between these isolation valves are designed to a URS equal to the design pressure of the RCS, with the exception of the RNS pump shaft seal. The following is a summary of the specific design features incorporated in the AP1000 RNS design to address the ISLOCA issue.

Quality Assurance/Seismic Protection

The portions of the RNS located outside containment (that serve no active safety functions) are classified as AP1000 Equipment Class C so that the design, manufacture, installation, and inspection of this pressure boundary is controlled by the following industry and regulatory safety-related quality assurance requirements: 10CFR21; 10CFR50, Appendix B; Regulatory Guide 1.26 Quality Group C; and American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section III, Class 3. In addition, this pressure boundary is classified as Seismic Category I so that it is protected from failure following a safe shutdown earthquake.

Increased Design Pressure

The portions of the RNS from the RCS to the containment isolation valves outside containment are designed to the operating pressure of the RCS. The portions of the system downstream of the suction line containment isolation valve and upstream of the discharge line containment isolation valve are designed with a URS not less than RCS operating pressure. Specifically, the piping is designed as Schedule 80S, and the flanges, valves, and fittings are specified to be greater than or equal to ANS class 900. Although the design pressure of the system has been increased to 900 psig, the maximum operating pressure has remained consistent with previous designs, and therefore, the actual margin between the maximum operating pressure and the design pressure of the RNS is increased by a factor of 3 (from 150 to 450 psig).

Reactor Coolant System Isolation Valve

The AP1000 RNS contains an isolation valve in the pump suction line from the RCS. This motor-operated containment isolation valve is designed to the RCS pressure. It provides an additional barrier between the RCS and lower-pressure portions of the RNS.

Normal Residual Heat Removal System Relief Valves

The inside-containment AP1000 RNS relief valve is connected to the RHR pump suction line inside containment. This valve is designed to provide low-temperature overpressure protection of the RCS as described in DCD subsection 5.2.2. It is connected to the high-pressure portion of the pump suction line, and it will reduce the risk of overpressurizing the low-pressure portions of the system. In addition, the RNS discharge header contains a relief valve provided to prevent overpressure in the RNS pump discharge line. Overpressure could occur if the three check valves (V013, V015, and V017) and the motor-operated containment isolation gate valve (V11) leaked back to the low-pressure portions of the RNS. The discharge of this relief valve is routed to the WLS effluent holdup tanks.

Features Preventing Inadvertent Opening of Isolation Valves

An interlock is provided for the normally closed, motor-operated RNS inner and outer suction isolation valves (RNS-V001A and B, and V002A and B). The interlock prevents the suction valves for the RNS from being opened by operator action unless the RCS pressure is less than a preset pressure and the following valves are in a closed position:

- IRWST suction isolation valve (RNS-V023)
- IRWST discharge isolation valve (RNS-V024)

Alarms are also provided in the main control room and on the remote shutdown workstation to alert the operator if RCS pressure exceeds the RNS design pressure after the valves are opened.

Reactor Coolant System Pressure Indication and High Alarm

The AP1000 RNS contains an instrumentation channel that indicates pressure in each RHR pump suction line. A high-pressure alarm is provided in the main control room to alert the operator to a condition of rising RCS pressure, which could eventually exceed the design pressure of the RNS.

The only portion of the RNS not designed to full RCS pressure, or to a URS pressure equal to the RCS design pressure, is the RNS pump shaft seal. The RNS pumps contain a shaft seal that has a design pressure of 900 psig. In addition, the pump is fitted with a disaster bushing that limits seal leakage in the event of a catastrophic failure of the pump seal to within the capabilities of the normal makeup system. The seal leakoff line is routed to a floor drain that is routed to the auxiliary building sump.

3.1.3 Justification of Design

This section provides justification for the adequacy of the RNS design with regard to ISLOCA criteria. Justification for the portions of the RNS other than the RNS pump mechanical seal is provided in subsection 3.1.3.1. Subsection 3.1.3.2 contains the justification for the design of the RNS pump shaft seal.

3.1.3.1 Design Justification for Normal Residual Heat Removal System

The design of the RNS meets the acceptance criteria for ISLOCA because the system is designed to either full RCS pressure or to a URS pressure equal to the RCS design pressure. In addition, design features are provided that exceed the ISLOCA criteria. The design features of the RNS contribute to the low core damage frequency attributed to ISLOCA calculated in the AP1000 PRA.

3.1.3.2 Design Justification for Normal Residual Heat Removal Pump Mechanical Seal

The RNS pumps contain a mechanical seal that permits proper operation of the RNS pump while limiting shaft leakage. The RNS pump shaft seal has a design pressure of 900 psig, and a maximum operating pressure of ~565 psig, with an expected operating range from 450 to 0 psig.

A fundamental problem with designing an RNS pump seal that can withstand full RCS pressure is that any type of seal that can withstand full RCS pressure will likely have abnormally fast wear of the seal faces during normal plant operation at low seal pressures. This increased wear at normal plant operating conditions could prevent the seal from maintaining the pressure boundary if ever exposed to the full RCS pressure. High-pressure seals would also require more frequent maintenance during normal operation. Therefore, a seal can be designed for normal-low pressure operation or for full-RCS-pressure conditions, but it is impractical to design a seal that would maintain the RCS pressure boundary with no leakage, and also operate satisfactorily at low-pressure conditions.

The AP1000 RNS pump mechanical seal is designed to minimize the amount of leakage if exposed to full RCS pressure. NUREG/CR-5603 (Reference 9) documented an evaluation of the pumps at the Davis Besse Nuclear Power Station under potential ISLOCA conditions. This study concluded the following for the Davis Besse Decay Heat Removal (DHR) System pumps.

Based on extensive discussions with the seal manufacturer, it was found that the rotating seal would maintain its structural integrity to pressures in excess of 2500 psi. The mechanical seals are designed to withstand a pressure of 1200 to 1250 psi without leaking. At greater pressures, the rotating face begins to distort creating a rotation at the contact surface. At 2500 psi, the rotation is three times the maximum allowable value. Thus, it is recommended that the potential for leakage through the pump seals be characterized assuming a nominal leak rate of 100 to 200 mg/sec together with an uncertainty variability of about 0.20.

The Davis Besse DHR pumps use a mechanical seal of a similar design as the AP1000 RNS pumps. Furthermore, the design pressure of the AP1000 mechanical seal is 900 psig as opposed to 450 psig for the Davis Besse DHR pumps. Since the design pressure of the AP1000 RNS mechanical seal is higher than that of the DHR pump in NUREG/CR-5603 (Reference 9), the expected leakage for the AP1000 RNS pump is less than that of the Davis Besse DHR pumps. Seal manufacturers contacted would not claim as low a leakage as specified in the reference study; they claimed their seals would meet the requirement that leakage at full RCS pressure be limited to within the capabilities of the normal makeup system.

The AP1000 RNS pump also has a disaster bushing that limits the leakage from the pump to within the capabilities of the normal makeup system in case of a catastrophic mechanical seal failure. The combination of a highly reliable single-seal design, in conjunction with a sturdy disaster bushing, maximizes the reliability of the seal during normal RNS operation and minimizes maintenance and associated radiation exposure. Furthermore, this design approach minimizes RNS pump leakage in the event of catastrophic mechanical seal failure. Leakage can be controlled so that only a small portion of the water that leaks past the primary seal faces escapes to the pump cubicle and most leakage is piped to a controlled drain. This is more favorable than a seal specially designed for full RCS pressure at the expense of normal-condition reliability.

3.2 CHEMICAL AND VOLUME CONTROL SYSTEM LETDOWN LINE TO LIQUID RADWASTE SYSTEM

3.2.1 Description of Primary System Interface

The CVS is the nonsafety-related system that provides for purification and makeup flow for the RCS. Unlike current PWRs that use continuous charging and letdown flow to maintain RCS chemistry and inventory control, the AP1000 uses a high-pressure purification loop totally within containment that uses reactor coolant pump (RCP) head to provide the motive force to drive purification flow. This eliminates the need for continuous charging and letdown, and therefore, letdown operations (that is, letdown of the RCS to the WLS) are limited to off-normal situations. DCD subsection 9.3.6 provides a complete description of the various functions and operations associated with the CVS. Figure 3-2 is the CVS P&ID modified to clearly indicate all high-pressure/low-pressure interfaces.

As shown in Figure 3-2, the CVS letdown line connects to the high-pressure CVS purification loop inside containment. Immediately downstream of this connection is the high-pressure, multi-stage letdown orifice, which reduces pressure in the letdown line from RCS operating pressure to below the design pressure of the low-pressure portion of the letdown line. The letdown line also contains a locked-closed bypass line around the letdown orifice. This line contains a locked-closed manual isolation valve (V043), which is opened only at shutdown when the RCS is depressurized. The letdown orifice must be bypassed when the RCS is depressurized to ensure sufficient letdown flow when required.

Downstream of the letdown orifice are two normally closed, fail-closed containment isolation valves (V045 and V047). The portions of the letdown line, from the purification loop up to and including the second containment isolation valve, are designed to full RCS pressure. The WLS portion of the letdown line contains a three-way valve that normally routes the letdown flow to the WLS degasifier package, and can be aligned to route the letdown flow to the WLS effluent holdup tanks. The discharge of the degasifier package is also routed to the WLS effluent holdup tanks.

A potential ISLOCA overpressurization pathway could exist from the RCS through the CVS purification loop, and through the CVS letdown line to the low-pressure WLS.

3.2.2 Design Evaluation

During power operation, the WLS is protected from overpressurization by the letdown orifice. The orifice design limits WLS pressure during letdown operation. In addition, a relief valve is provided in the low-pressure portion of the CVS letdown line in case a valve in the letdown line is inadvertently mispositioned and consequently causes an overpressurization of a low-pressure line. As seen in Figure 3-2, relief valve V057 is provided to limit the pressure in the WLS if manual isolation valve V048 were inadvertently closed, and the letdown isolation valves were opened. Discharge from relief valve V057 is routed directly to the WLS waste holdup tank.

During shutdown operation, with the letdown orifice bypassed, the relief valve in the CVS letdown line is required to protect the letdown line in the event of a cold overpressure transient. If the letdown isolation valves were opened and a cold overpressure transient occurred, the pressure excursion in the RCS would be limited to the set pressure of the RNS relief valve (plus accumulation pressure). Relief valve V057 is sized to provide sufficient flow for this event such that the pressure drop in the letdown line would limit the maximum WLS pressure to within 110 percent of its design pressure.

Because of the passive features in the CVS letdown line (that is, the letdown orifice and relief valve V057), inadvertent pressurization of the low-pressure portion of the letdown line is avoided. However, other events, such as excessive letdown operation or valve mispositioning that causes relief valve V057 to open and discharge to the WLS effluent holdup tanks, could cause a depletion in reactor coolant inventory. For any event that results in the depletion in the pressurizer water level (including excessive letdown or a letdown line ISLOCA), the letdown line isolation valves receive separate automatic signals to close. The letdown line isolation valves receive a control-grade automatic signal to close on normal low pressurizer level (~40 to 55 percent based on power level). This signal will terminate any letdown line ISLOCA or other excessive letdown event. Furthermore, the letdown isolation valves and the purification loop isolation valves (V001 and V002) also receive a safety-related signal to close on an abnormally low pressurizer level (~25 percent). Finally, the letdown line isolation valves and the purification loop isolation valves also close on a safeguards actuation signal, which would occur as a result of a LOCA that continued until the low-pressure safeguards actuation setpoint was reached. These four safety-related valves isolate the letdown line and would terminate any letdown line ISLOCA before it became a challenge to core cooling.

3.2.3 Justification of Design

The CVS letdown line meets ISLOCA criteria for low-pressure systems. The letdown isolation valves are containment isolation valves, and as such, have the capability for leak-testing, and are provided with valve position indication in the control room at all times. Furthermore, the WLS degasifier column contains a high-pressure alarm (via pressure switch PS-014), which would warn the control room operators that the WLS pressure was approaching the design pressure and that rising reactor coolant pressure could result in an ISLOCA. Also, the multiple safety-related isolation valves, which close automatically on low pressurizer level and on a safeguards actuation signal, protect against a letdown line ISLOCA, which could cause a loss of core cooling.

The flow rate from any excessive letdown or letdown line ISLOCA event would be within the capabilities of the normal makeup system. If the makeup pumps operate such that RCS inventory and pressure remain within the RCS operating limits (that is, pressurizer level >25 percent, RCS pressure >1800 psig), then it is assumed that the operator would identify the break and determine the actions to terminate the leak within 30 minutes. The radioactive releases from such an event are within the design basis analysis contained in DCD subsection 15.6.2.

It is not practicable to design the low-pressure portions of the letdown line to a higher design pressure. The letdown line is routed to either the degasifier package or the effluent holdup tanks. As discussed in section 3, it is not practicable to design the WLS effluent holdup tanks to a higher design pressure. It is also not practicable to design the WLS degasifier package to a higher design pressure. This degasifier package includes a degasifier column and a degasifier separator, four low-pressure pumps, and a low-pressure heat exchanger. A significant cost and development effort would be required to redesign this equipment to withstand full RCS design pressure. In addition, the degasifier package discharges directly to the WLS effluent holdup tanks, and therefore, designing the degasifier package to high pressure, if practicable, would provide no benefit. This is because the system interfaces directly with large, low-pressure tanks for which higher design pressures are impractical.

3.3 CHEMICAL AND VOLUME CONTROL SYSTEM MAKEUP PUMP SUCTION LINE

3.3.1 Description of Primary System Interface

The AP1000 CVS makeup pumps operate intermittently to make up for RCS leakage. The pumps start automatically when the pressurizer level reaches the bottom of the normal level band, and stop when the level reaches the top of the band. The makeup pumps take suction from either the boric acid tank, the demineralized water storage tank, or both, and inject makeup into the CVS purification loop return stream. DCD subsection 9.3.6 provides a complete description of the various functions and operations associated with the CVS. Figure 3-3 is the CVS P&ID modified to clearly indicate all high-pressure/ low-pressure interfaces.

As shown in Figures 3-2 and 3-3, the CVS makeup line from the makeup pump discharge to the RCS, has a design pressure greater than or equal to the RCS design pressure. Pressurization is postulated from the RCS through the purification loop, through the makeup line connection to the purification loop, back through the makeup line and makeup pumps, and to the low-pressure makeup pump suction line. This pressurization pathway exists only if the makeup pumps are not operating. If the makeup pumps are operating, the system hydraulic phenomena prevent pressurization of the suction piping. It should be noted that two normally closed check valves in the makeup line isolate the pump suction line from the high-pressure purification loop. In addition, each makeup pump suction line contains a relief valve that protects the low-pressure piping in the event that leakage through the check valves causes the suction piping to become overpressurized.

The makeup pumps can take suction from either the boric acid tank (BAT), the demineralized water storage tank, the waste holdup tanks, or the spent fuel pool. Each suction line contains a check valve to prevent flow between water storage tanks. In addition, the spent fuel pool and waste holdup tank suction lines contain a normally closed manual valve.

3.3.2 Design Evaluation

As discussed in section 3, the tanks that the CVS can take suction from are all large, low-pressure tanks for which high-pressure designs are impractical. As such, these tanks and the piping up to the first manual isolation valve, are excluded from ISLOCA consideration. As shown in Figure 3-3, each makeup pump suction line contains a check valve and at least one manual isolation valve. To prevent overpressurization of the makeup pump suction line, relief valves V158A and B are provided in case the check valves (V064, and V160A and B) in the makeup pump discharge line leak when the pumps are not running. These relief valves prevent an ISLOCA in the makeup pump suction piping.

In the event that makeup line check valve failure causes the relief valves to open, the relief valves would discharge to the WLS effluent holdup tanks. This would eventually lead to a low normal pressurizer level signal, causing the makeup pumps to start, and effectively terminating the ISLOCA. If the nonsafety-related makeup pumps failed to start, safety-related isolation of the makeup line would be achieved by isolation of the purification loop isolation valves (V001 and V002), the makeup line containment isolation valves (V090 and V091), and the RCS boundary check valves in the makeup line (V081 and V082). These safety-related valves isolate the makeup line and would terminate any makeup suction line ISLOCA before it became a challenge to core cooling.

3.3.3 Justification of Design

The CVS suction line piping meets ISLOCA criteria for low-pressure systems. The makeup line isolation valves are containment isolation valves, and as such, have the capability for leak-testing, and are provided with valve position indication in the control room at all times. In the event of an ISLOCA, the makeup pumps would be operated (either manually or automatically on low pressurizer level), and the mechanism for overpressurizing the suction piping would not exist. If the makeup pumps did not start, and the mechanism was still available to overpressurize the suction piping, the containment isolation valves would automatically terminate the ISLOCA. These valves are closed on a safeguards actuation signal coincident with low pressurizer level. In addition, the RCS pressure boundary valves in the purification loop are also closed on a safeguards actuation signal. These multiple, safety-related isolation valves prevent an ISLOCA in the makeup pump suction line that could result in a loss of core cooling.

It is not practicable to design the low-pressure portions of the makeup pump suction line to a higher design pressure. The suction line contains relief valves that protect the low-pressure portions of the piping from overpressure in events such as leaking check valves in the discharge line or thermal expansion in case of a loss of miniflow cooling. A loss of miniflow cooling could occur if component cooling water to the miniflow heat exchanger was lost. If the design pressure of the piping were increased to the URS pressure, the relief valves would still be necessary to protect against leaking check valves or thermal expansion. An increase in the valve set pressure (to correspond to the higher design pressure) would significantly impact design pressure of the pump discharge line for cases of a loss of miniflow heat exchanger cooling. And while designing the suction piping to full RCS pressure would address the case of leaking check valves in the discharge piping, it would not solve the thermal expansion issue.

Another consideration is that the makeup pump suction lines each contain a check valve that separates the suction piping from a large atmospheric tank. The suction line check valves are designed to open on low differential pressure, and industry experience has shown that low-differential-pressure check valves have a high tendency to leak. Therefore, assuming that the two discharge line (high-differential-pressure) check valves (in series) leak, it should also be assumed that the suction piping check valves would also tend to leak. Therefore, designing the suction pipe to a higher pressure will only increase the likelihood that the RCS leak extends on to one of the atmospheric tanks.

With the AP1000 design, the relief valves provide overpressure protection and direct any leakage from the discharge line check valves to the WLS effluent holdup tanks, a satisfactory arrangement, as opposed to leaking into the clean tanks from which the makeup pumps normally take suction. The WLS effluent holdup tank is designed to handle radioactive fluids, and its level is monitored by remote instrumentation (see section 4 regarding detection of ISLOCAs). Therefore, low-pressure suction piping with appropriately sized relief valves is a preferable arrangement to higher-design-pressure suction piping.

Adding an interlock to isolate the makeup line on indication of high makeup pump suction pressure was considered but not incorporated. The added complication of potentially isolating the makeup line on spurious signals, combined with the low probability of a makeup pump suction line ISLOCA, make this interlock undesirable.

3.4 PRIMARY SAMPLING SYSTEM

3.4.1 Description of Primary System Interface

The PSS collects representative samples of fluids from the RCS and associated auxiliary system process streams, and the containment atmosphere for analysis by the plant operating staff. Since fluids are collected outside the containment, the PSS is the system that connects directly to the RCS and carries reactor coolant outside containment. DCD subsection 9.3.3 provides a complete description of the various functions and operations associated with the PSS. Figure 3-4 is the PSS P&ID modified to clearly indicate all high-pressure/low-pressure interfaces. As shown, almost the entire PSS is designed to withstand full RCS pressure.

The following portions of the PSS are designed to lower pressure than the full RCS pressure in the PSS:

- Eductor water storage tank (EWST)
- Demineralized water supply line

The PSS connects to the RCS at several locations, including the pressurizer liquid space and each hot leg. Each connection contains a flow-restricting orifice that limits the flow from the RCS in the event of a break of a sample line. These orifices also reduce the pressure in the sampling lines during sampling operations. During the sampling of the RCS, the operator opens the appropriate sample line isolation valve (for example, V003 for RCS pressurizer liquid sample) and opens the two remotely operated containment isolation valves (V010A or B, and V011). The sample passes through a sample cooler, and it is collected in the appropriate sample bottle. For a typical sample operation, the operator will purge the sample line to the WLS degasifier. When a sufficient volume of coolant has been purged, the operator closes the isolation valves downstream and upstream of the appropriate sample chamber, and then closes the remotely operated valves.

3.4.2 Design Evaluation and Justification

It is not practicable to design the low pressure portions of the PSS to a higher design pressure. These portions of the PSS are at atmospheric pressure and connect to the low-pressure demineralized water system (DWS). Designing the low-pressure EWST to high pressure, to meet ISLOCA criteria, would then require the DWS to be designed for high pressure. As discussed in section 3.6, this is not practicable.

During sampling operations, flow limiting orifices plus the small diameter of the PSS lines limit flow to approximately 0.5 gpm, and the PSS lines are never pressurized above the design pressure of the low-pressure portions of the PSS. The PSS high pressure/low pressure interface occurs within the grab sample panel, which is a standard panel with design features to prevent backflow and overpressurization of the low pressure portions of the system. Even in the unlikely event that overpressurization would occur, leakage flow from the RCS would be well within the makeup capability of the normally operating makeup system. At any time, the operator would be able to isolate the leak by closing the PSS containment isolation valves.

For this event, assuming operation of the normal makeup system, the operator would identify the break, and/or the radiation monitors and alarms in the auxiliary building, and take actions to terminate the leak within 30 minutes. The radioactive releases resulting from such a beyond-design-basis event are within the design basis analysis contained in DCD subsection 15.6.2. For this event, assuming the normal makeup system is not available, the PSS containment isolation valves would automatically close on the safeguards actuation signal resulting from the loss of coolant and terminate the event.

3.5 SOLID RADWASTE SYSTEM

3.5.1 Description of Primary System Interface

The solid radwaste system (WSS) provides the storage facilities for both wet and dry solid wastes prior to and subsequent to processing and packaging. As shown in Figure 3-2, the WSS connects to the high-pressure CVS demineralizers to facilitate transfer of the spent resin from the CVS demineralizers to the spent resin storage tanks. The spent resin header connects to each of the three high pressure CVS demineralizers with an individual, normally closed isolation valve in each line. The spent resin header then penetrates containment with two normally closed, locked-closed, containment isolation valves (V040 and V041). A manual valve, placed downstream of the second containment isolation valve, isolates the downstream piping to facilitate containment isolation leak-testing. Figure 3-2 shows the high-pressure/low-pressure interface across this valve (V039).

3.5.2 Design Evaluation and Justification

It is not practical or necessary to design the WSS to a higher design pressure. The system contains many low-pressure components, such as spent resin tanks and resin transfer and resin mixing pumps. The WSS spent resin line meets the ISLOCA criteria for low-pressure systems by providing locked-closed isolation valves and administrative procedures to protect the low-pressure portion of the system.

The WSS spent resin line is normally isolated by the locked-closed manual containment isolation valves. These containment isolation valves are administratively controlled and are leak-tested in accordance with the AP1000 In-Service Testing (IST) Plan DCD subsection 3.9.6. The CVS demineralizers are inside containment and normally circulate reactor coolant at RCS operating pressure. As such, resin transfer operations cannot be performed at normal power operations. These operations are conducted during refueling operations, when the RCS is fully depressurized. Therefore, since this spent resin line can be opened only when the RCS is depressurized, and the high-pressure valves in the spent resin line that isolates the low-pressure portion of the system are administratively locked closed and regularly leak-tested, the WSS spent resin lines are not required to be designed to a higher design pressure.

3.6 DEMINERALIZED WATER TRANSFER AND STORAGE SYSTEM

3.6.1 Description of Primary System Interface

The DWS is a low-pressure water transfer system consisting of tanks, pumps, piping, valves, and associated instrumentation and controls. It interfaces with the high-pressure CVS purification loop as shown in Figure 3-2.

The DWS supply header inside containment connects to the CVS demineralizers. During shutdown operations, demineralized water is used to sluice resin to the WSS as discussed in section 3.5. To perform these operations, the operator must open manual valves in the CVS. As discussed in section 3.5, these operations can be performed only at shutdown when the RCS is fully depressurized. A potential pressurization pathway could exist if the operators failed to reclose manual valves in the CVS (such as V022A or B) before returning to power operation. In this case, the DWS would be protected from overpressurization by a single check valve (V026). If check valve V026 subsequently leaked or failed to close, the DWS header inside containment would become overpressurized.

3.6.2 Design Evaluation

The overpressurization pathways for the DWS initiate inside containment. Therefore, an overpressurization of this system would most likely result in the rupture of the DWS header inside containment. This would not result in an ISLOCA, as discussed in section 2 of this report. Any resulting loss of coolant would be maintained inside containment. Isolation of the CVS purification loop would terminate the event.

A relief valve has been added to the DWS header inside containment to preclude the possibility of overpressurizing the DWS for these events. This relief valve, shown in Figure 3-5, discharges to the containment.

3.6.3 Justification of Design

The DWS meets the ISLOCA criteria for low-pressure systems because an overpressurization of the system from the high-pressure RCS does not result in a loss of coolant outside containment. The DWS inside-containment supply header interfaces with a potentially high-pressure system containing reactor coolant. Overpressurization can only occur if there are multiple failures and misalignments of isolation valves and check valves in the high-pressure systems. For those events, the relief valve in the DWS supply header prevents an ISLOCA.

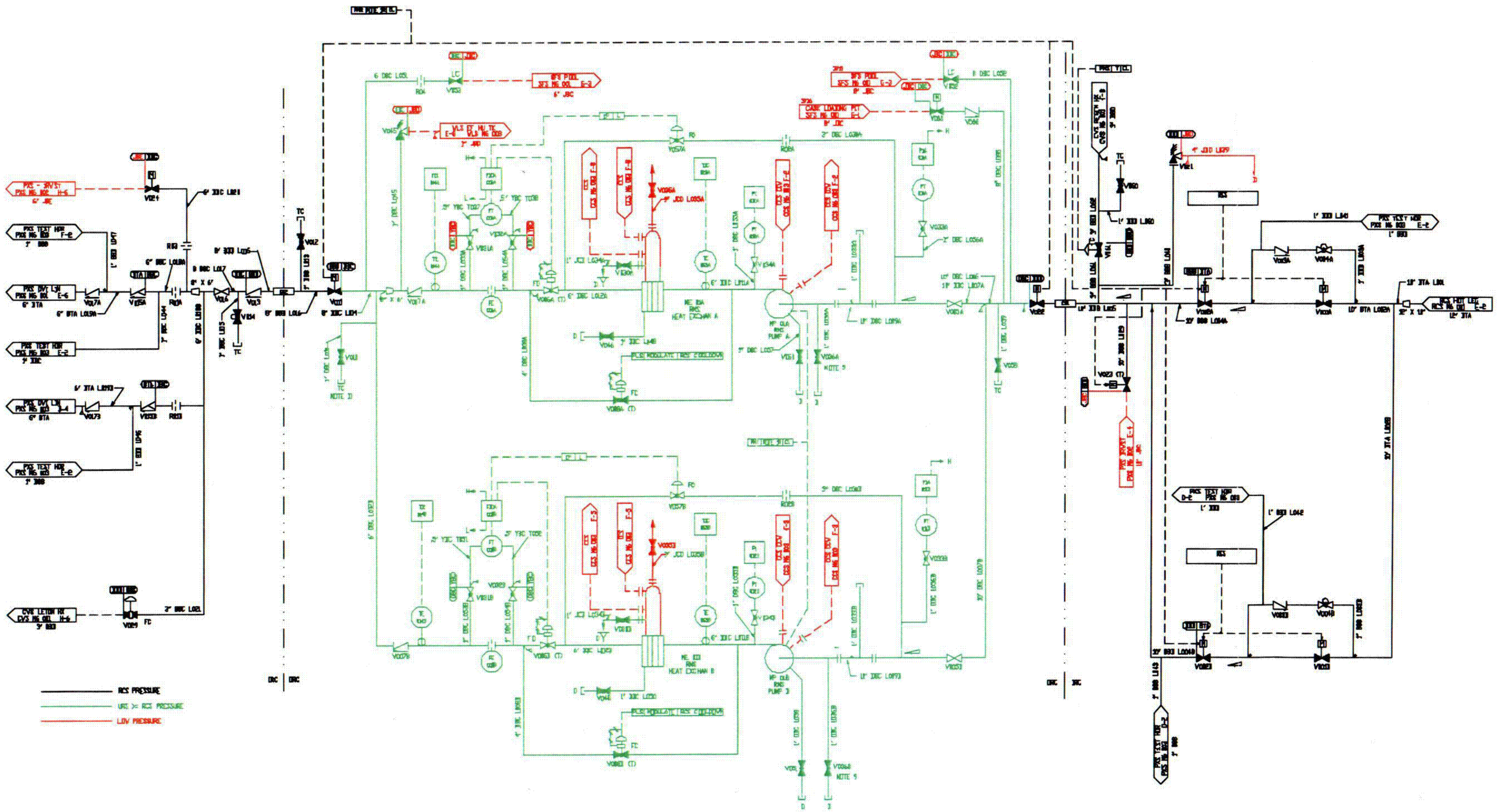

Figure 3-1
Normal Residual Heat Removal System

C01

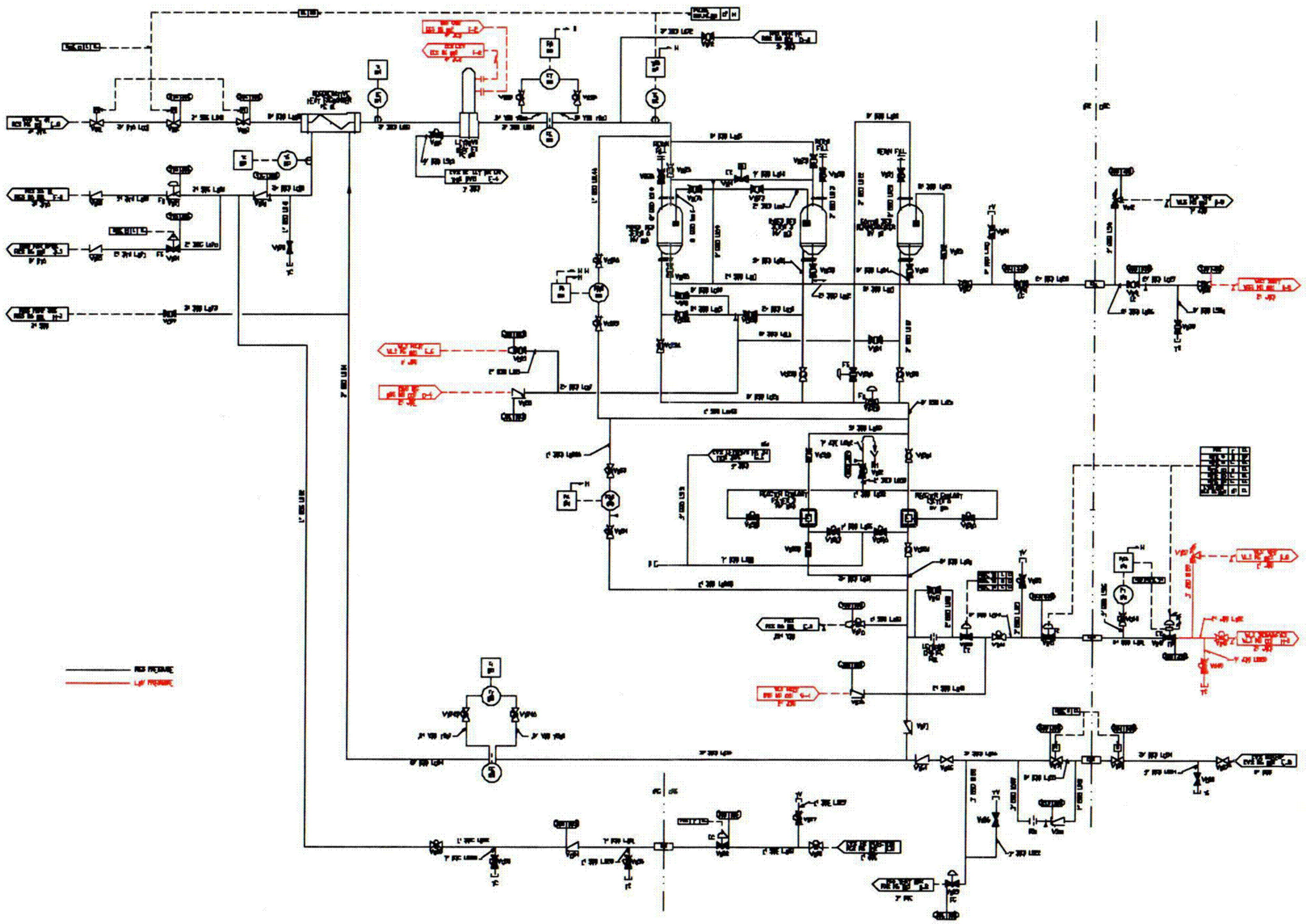

Figure 3-2
Chemical and Volume Control System
Purification Loop

C02

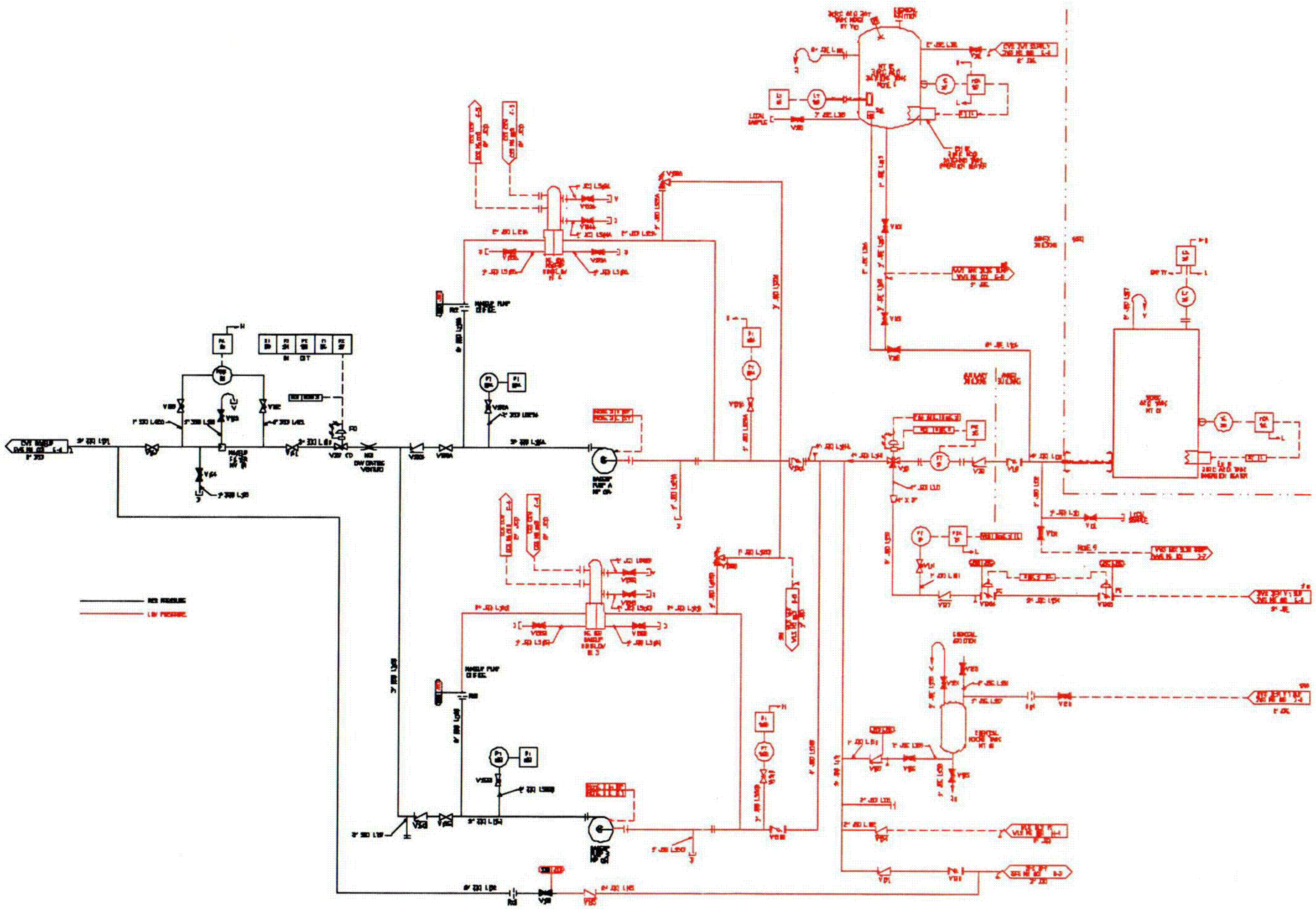

Figure 3-3
Chemical and Volume Control System
Makeup Pumps

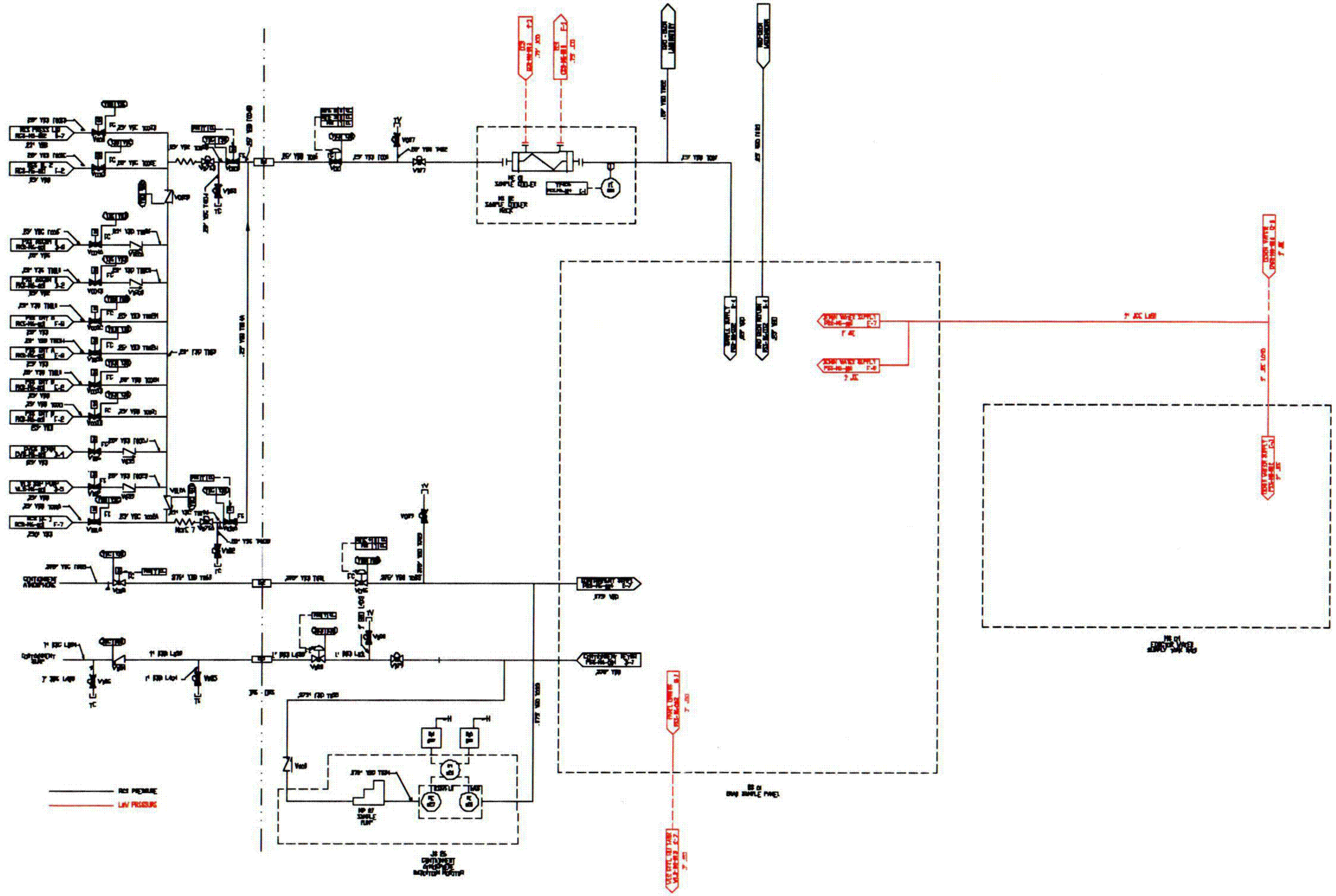

Figure 3-4
Primary Sampling System

C04

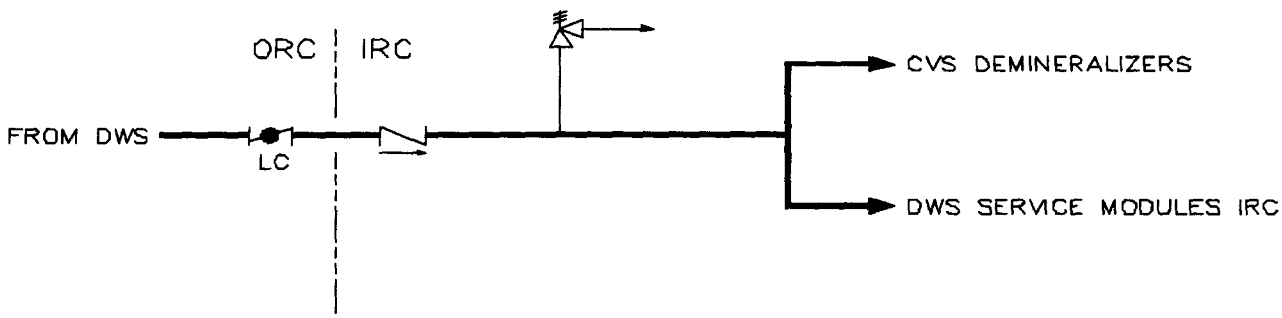

Figure 3-5 Demineralized Water System Supply Header Inside Containment

4 CONCLUSIONS

The AP1000 has incorporated various design features to address ISLOCA challenges. These design features have resulted in the low AP1000 core damage frequency for ISLOCA compared with that of current plants. These design features are primarily associated with the RNS and are discussed in detail in section 3 of this report as well as DCD subsection 5.4.7. This report was prepared to document the comprehensive systematic evaluation of the AP1000 design for conformance to the ISLOCA acceptance criteria in the various referenced NRC documents. As a result of this study, additional design features have been incorporated in the AP1000 design and are documented in the AP1000 DCD. The following table provides a summary of AP1000 design features incorporated to meet the ISLOCA acceptance criteria.

Table 4-1 Summary of AP1000 ISLOCA Design Features

System/Subsystem	Major Design Features	Figure Number
RNS	• Increased design pressure of the RNS outside containment to a URS equal to full RCS pressure	3-1
Letdown line	• High-pressure purification loop inside containment to eliminate high-energy letdown outside containment • Letdown orifice to limit leakage from a letdown line ISLOCA • Automatic isolation of letdown on safeguards actuation • Relief valve added to prevent overpressurization of letdown line	3-2
Makeup pump suction	• Relief valves added to minimize the consequences of pump suction overpressurization • High-pressure alarm in pump suction line to alert the operator to overpressurization	3-3
PSS	• Most of PSS designed to full RCS pressure • Flow-restricting orifices to limit scope of ISLOCA • Automatic isolation of PSS on a safeguards actuation signal	3-4
DWS	• Relief valve added to prevent overpressurization of DWS inside containment • Automatic isolation of DWS lines outside containment on safeguards actuation	3-5

5 REFERENCES

1. NRC Information Notice 92-36, "Intersystem LOCA Outside Containment," May 7, 1992.

2. NRC Information Notice 92-36, Supplement 1, "Intersystem LOCA Outside Containment," February 22, 1994.

3. NRC Letter, "Preliminary Evaluation of the Resolution of the Intersystem Loss-of-Coolant-Accident (ISLOCA) Issue for the Advanced Boiling Water Reactor (ABWR) - Design Pressure for Low Pressure Systems," Docket Number 52-001.

4. NUREG/CR-5102, "Interfacing System LOCA: Pressurized Water Reactors," February 1989.

5. SECY-90-016, "Evolutionary Light Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirement," January 12, 1990.

6. NUREG-0933, "A Status Report on Unresolved Safety Issues," U.S. Nuclear Regulatory Commission, April 1989.

7. SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor Designs," April 2, 1993.

8. NRC Letter, "Clarification of Issues for Evolutionary Advanced Light Water Reactors (ALWR)," to E. E. Kintner (ALWR Steering Committee), May 6, 1991.

9. NUREG/CR-5603, "Pressure-Dependent Fragilities for Piping Components," October 1990.

10. Letter, "Submittal Supporting Accelerated ABWR Schedule-ISLOCA, Issue #42 (GE ABWR SSAR Appendix 3M)," General Electric Company, San Jose, California, July 9, 1993.

11. WCAP-14425, "Evaluation of the AP600 Conformance to Inter-System Loss-of-Coolant Accident Acceptance Criteria," Westinghouse, July 1995.