Westinghouse Electric Corporation
Energy Systems

Box 355
Pittsburgh Pennsylvania 15230-0355

NSD-NRC-97-4998
DCP/NRC0752
Docket No.: STN-52-003

February 25, 1997

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

ATTENTION: T. R. QUAY

SUBJECT: INFORMAL CORRESPONDENCE

Dear Mr. Quay:

Please find attached a formal transmittal of correspondence we have previously sent to you informally. This informal correspondence was sent over the period February 1, 1997 through February 17, 1997.

One informal correspondence from the NRC to Westinghosue is included in this package.

Attachment 1 provides the index of the attached material as you requested.

Brian A. McIntyre

Brian A. McIntyre, Manager
Advanced Plant Safety and Licensing

jml

Attachment

cc: N. J. Liparulo, Westinghouse (w/o Attachment)
T. T. Martin, NRC (w/o Attachment)

E004 1/1

Attachment 1 to Westinghouse Letter DCP/NRC0752

DATE	ADDRESSEE	DESCRIPTION
2/3/97	Scaletti	Request to provide definitive action for Westinghouse, or to close open item 142
2/4/97	Scaletti	Request to provide definitive action for Westinghouse, or to close open item 143
2/4/97	Bongarra/ Huffman	Markup of Table 18.12.2-1 for 2/5 phone call discussion
2/5/97	Scaletti	Open items for Chapter 17. Request to provide definitive action for Westinghouse, or to close open items
2/5/97	Scaletti	Request to acknowledge recipt of information previously transmitted
2/5/97	Scaletti	Reminder list of open items where there is a difference in Westinghouse and NRC status
2/5/97	Scaletti	Request to provide definitive action for Westinghouse, or to close open item 144
2/5/97	Jackson	Clarification of information for open item 243
2/5/97	Huffman	Information missing from SSAR Revision 10, Chapter 17
2/6/97	Scaletti	Open items for Chapter 12. Request to provide definitive action for Westinghouse, or to change status.
2/6/97	Jackson	Protection of fire pumps in response to questions raised in 1/7/97 meeting
2/6/97	Quay	Open item closure status
2/4/97	Huffman	NOTRUMP and WC/T
2/7/97	Jackson	SSAR LBB changes
2/11/97	Scaletti	Open items for SSAR Section 3.7. Request to provide definitive action for Westinghouse, or to change status.
2/11/97	Scaletti	Open items for SSAR Appendix 3F
2/11/97	Scaletti	Open items for Chapter 13
2/11/97	Scaletti	Request to provide definitive action for Westinghouse, or to change status of open item 158.
2/11/97	Bongarra/ Higgans	Proposed makeups of SSAR Chapter 18 and 7.
2/12/97	Scaletti	Request to provide definitive action for Westinghouse, or to change status of open item 157.

2/12/97	Scaletti	Reminder list of open items where there is a difference in Westinghouse and NRC status
2/12/97	Scaletti	Request to provide definitive action for Westinghouse, or to change status of open item 164.
2/12/97	Scaletti	Request to acknowledge recipt of information previously transmitted
2/12/97	Huffman	SSAR Section 3.8.3.4.2
2/12/97	Jackson	Information to close open item 346
2/13/97	Scaletti	Open items for SSAR Section 3.10. Request to provide definitive action for Westinghouse, or to change status.
2/13/97	Quay	Open item closure status
2/13/97	Scaletti	Open items for PRA Chapter 42. Request to provide definitive action for Westinghouse, or to change status.
2/7/97	Huffman	Information to support 2/10/97 phone call
2/14/97	Winters	Request for restrictions on power and control cables
2/14/97	Scaletti	Open items for SSAR Chapter 18. Request to provide definitive action for Westinghouse, or to change status.
2/14/97	Scaletti	Open items 21 and 4617. Request to provide definitive action for Westinghouse, or to change status.
2/17/97	Scaletti	Open items for SSAR Section 3.8.2. Request to provide definitive action for Westinghouse, or to change status.
2/13/97	Huffman	Markup of SSAR section 6.2.4.2.3. WIll be in Revision 11 of the SSAR unless we hear otherwise
2/17/97	Scaletti	Open items 173. Request to provide definitive action for Westinghouse, or to change status.

FAX to DINO SCALETTI

February 17, 1997

CC: Sharon or Dino, please make copies for: Bill Huffman
Ted Quay

Robin Nydes
Chip Suggs
Ed Cummins
Bob Vijuk
Brian McIntyre

OPEN ITEM #173 (M5.2.5-30)

In my quest to make sure we have provided NRC with everything needed to prepare an FSER, I am researching open items from the smallest item number on. The relevant documentation related to Open Item #173 (M5.2.5-30) is attached. We provided the original comparison to STS with NSD-NRC-96-4833 on October 11, 1996. We then provided probability risk assessment information related to the differences from STS with NSD-NRC-97-4939 on January 14, 1997. This was reiterated in the RAI responses provided by NSD-NRC-97-4972 of February 6, 1997. This item (#173) was asked by a technical branch other than the Tech Spec branch. The letters identified above were in response to questions asked by the Tech Spec branch. Please help us provide the branch to branch coordination required to obtain proper review of this information. We believe that the letter identified above resolve the concerns of item #173. It seems a reasonable request that NRC acknowledge receipt of the information. We request that NRC provide a definitive action for Westinghouse or provide direction to change the status of this item. We recommend "Action N". Thank you.

Jim

Jim Winters
412-374-5290

AP600 Open Item Tracking System Database: Executive Summary

Date: 2/17/97

Selection: [item no] between 173 And 173 Sorted by Item #

Item No.	Branch	DSER Section/ Question	Type	Title/Description Detail Status	Resp Engineer	(W) Status	NRC Status	Letter No. /	Date
173	NRR/SPLB	5.2.5	MTG-OI		TECHSPEC/Suggs, C.	Closed	Action W		

M5.2.5-30 (REACTOR COOLANT PRESSURE BOUNDARY LEAKAGE) STS 3.4.15 includes SR 3.4.15.2, which states that a channel operational test (COT) should be performed on the containment atmosphere radioactivity monitor every 31 days. AP600 TS 3.4.5 includes SR 3.4.9.2 which states that the COT should be performed every 92 days. Westinghouse should provide justification for the deviation from STSs.

action W: justification of differences between AP600 TS and STS will be provided with TS. rkn 3/29

Closed - With issuance of the Tech Specs in SSAR Rev. 9.

Action W - Need an explanation of Action Times as they relate to STS.

2 of 7

Total Records: 1

Westinghouse
Electric Corporation

Energy Systems

Box 355
Pittsburgh Pennsylvania 15230-0355

NSD-NRC-96-4833
DCP/NRC0616
Docket No.: STN-52-003

October 11, 1996

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

ATTENTION: T. R. QUAY

SUBJECT: CLOSING THE LAST DSER OPEN ITEM FOR AP600 SSAR SECTION 16.1, TECHNICAL SPECIFICATIONS (TS)

Dear Mr. Quay:

This letter is written to close the last DSER open item for AP600 SSAR Section 16.1, Technical Specifications (TS). Westinghouse committed to provide written explanation of technical differences between the AP600 TS and those presented in NUREG-1431, the Standard TS (STS). Attached are:

1. A roadmap which identifies the sections comprising the STS versus those included in the AP600 TS. For any TS that are included in the STS but not in the AP600 TS, an explanation is provided. For any TS that are included in the AP600 TS but not in the STS, those sections are shaded in the roadmap and explained. Explanations are also provided for other content differences between the STS and AP600 TS.

2. A description of general or overall changes whose explanations apply to multiple TS.

3. A list of technical differences between the STS and AP600 TS. The TS and BASES are grouped by section and an explanation of each difference is provided.

4. A table of and explanation for those LCOs whose endpoint is defined as MODE 4 for the AP600, rather than MODE 5 or "Go to LCO 3.0.3" per the STS.

Discussions regarding ties between the AP600 PRA and the Technical Specifications will be provided in the response to RAI 630.10.

2963A

3 of 7

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This submittal closes Open Item Tracking System (OITS) item 2353, which is the final open item for the AP600 Technical Specifications. If you have any questions regarding this transmittal, please contact Robin K. Nydes at (412) 374-4125.

Brian A. McIntyre, Manager
Advanced Plant Safety and Licensing

/nja

Attachment

cc: W. Huffman, NRC
A. Chu, NRC
C. Grimes, NRC
N. Liparulo, Westinghouse (w/o Attachments)

2943A

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Westinghouse Electric Corporation
Energy Systems
Box 355
Pittsburgh Pennsylvania 15230-0355

NSD-NRC-97-4939
DCP/NRC0705
Docket No.: STN-52-003

January 14, 1997

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

ATTENTION: T. R. QUAY

SUBJECT: WESTINGHOUSE RESPONSE TO RAI 630.10

Dear Mr. Quay:

Enclosed are three copies of the Westinghouse response to RAI 630.10 regarding AP600 Technical Specification deviations from NUREG-1431 based on probability risk assessment. The NRC technical staff should review this response as part of their review of the AP600 Technical Specifications. This closes DSER open item tracking system item #3054. If there are any questions regarding this transmittal, please contact Robin K. Nydes at (412) 374-4125.

Brian A. McIntyre, Manager
Advanced Plant Safety and Licensing

/jml

enclosure

cc: Angela Chu, NRC - (w/enclosure)
W. C. Huffman, NRC - (w/enclosure)
Nicholas Liparulo, Westinghouse - (w/o enclosure)

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9701230265

Question 630.10. Provide a list of proposed AP600 Technical Specification requirements that deviate from NUREG-1431 based either totally or partially on probabilistic risk assessment (PRA) or PRA insights.

Response: The deviations from NUREG-1431 are explained in Reference 1. There are no AP600 Technical Specifications which deviate from NUREG-1431 with the PRA as the basis.

However, selection of a standardized Completion Time or Surveillance Frequency considers available PRA results as described in Reference 2. Per NRC request, attached is a list comparing the NUREG-1431 Standardized Technical Specification (STS) completion times and surveillance frequencies to the AP600 TSs. Deviations from STS times which are less restrictive than STS times are highlighted and any PRA relationship is given in the comment column.

SEE ATTACHED LIST

SSAR Revision: NONE

References:

1. NSD-NRC-96-4833, Closing the Last DSER Open Item for AP600 SSAR Section 16.1, Technical Specifications (TS), 10/11/96.

2. NSD-NRC-96-4699, Westinghouse AP600 Technical Specifications Approach, 5/3/96.

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Westinghouse Electric Corporation
Energy Systems

Box 355
Pittsburgh Pennsylvania 15230-0355

NSD-NRC-97-4972
DCP/NRC0732
Docket No.: STN-52-003

February 6, 1997

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

TO: T. R. QUAY

SUBJECT: RESPONSE TO RAIs 630.11 THROUGH 630.14

REFERENCE: LETTER FROM NRC TO WESTINGHOUSE (HUFFMAN TO LIPARULO), "REQUEST FOR ADDITIONAL INFORMATION ON WESTINGHOUSE AP600 TECHNICAL SPECIFICATIONS OPTIMIZATION METHODOLOGY", DATED DECEMBER 12, 1996.

Enclosed for NRC review are the Westinghouse responses to the following Technical Specification RAIs, provided by the above Reference.

630.11	Completion Time Anchor Point
630.12	Surveillance Frequency Baseline
630.13	Request for Response to RAI 630.10
630.14	Differences Between the Proposed Tech Specs Approach and Tech Specs Rev. 2

This completes Westinghouse activity for Open Item Tracking System items 4224 through 4227, a report for which is attached. Please advise as to the NRC status for these items. If you have any questions regarding this transmittal, please contact Robin K. Nydes (412) 374-4125.

Brian A. McIntyre, Manager
Advanced Plant Safety and Licensing

/jml
enclosure
attachment

cc: W. Huffman, NRC (w/enclosure/attachment)
A. Chu, NRC (w/enclosure/attachment)

3075A

9702190016

Westinghouse

FAX COVER SHEET

RECIPIENT INFORMATION		SENDER INFORMATION	
DATE:	2-13-97	NAME:	Cindy Haag
TO:	Bill Huffman/Joe Sebrosky	LOCATION:	ENERGY CENTER - EAST
PHONE:	FACSIMILE:	PHONE:	Office: 412-374-4277
COMPANY:	USNRC	Facsimile:	win: 284-4887 outside: (412)374-4887
LOCATION:			

Cover + Pages 1 + 12

The following pages are being sent from the Westinghouse Energy Center, East Tower, Monroeville, PA. If any problems occur during this transmission, please call:

WIN: 284-5125 (Janice) or Outside: (412)374-5125.

COMMENTS:

Bill / Joe -

Here is the markup of SSAR section 6.2.4.2.3, Hydrogen Ignition Subsystem. It will be in SSAR Revision 11 unless we hear from you.

Cindy

cc McIntyre
Winters
Cummins
RanVink
Lindgren
Jeanne Evans
McDermott
Sicbel

50.46 and 50.44 for a loss of coolant accident. The existence of significant levels of poisons mandates consideration of events and hydrogen generation rates for which other design attributes of the hydrogen control system are specifically provided. Events which generate high levels of iodine and tellurium, for example, are the result of gross fuel clad damage and cladding/water reactions. The environments in which safety-related components are designed and qualified to function are discussed in Section 3.11. The pressure, temperature, and chemical environment conditions for which components are designed to function have been based on analysis of the design basis event and the systems response. The radiation environments have, in contrast, been the result of a deterministic application of the accident source term. As specified in NUREG 1465, to determine the accident source term for regulatory purposes, the staff examined a range of severe accidents that have been analyzed for light water reactors. The environmental qualification guidance and practice is conservatively based on the effects of *radiation* due to a severe accident source term.

To illustrate the margin available and the tolerance to catalytic poisons, Figure 6.2.4-2 demonstrates the effects of the presence of elevated concentrations of poisons in containment. An estimate of the effect of iodine is documented in Reference 18 and is projected to be less than a 30 percent decrease in depletion. This value is also projected to envelope the potential impact of other potential poisons such as tellurium.

The passive autocatalytic recombiner testing and reporting of test data, conducted under the NIS quality assurance program, is appropriate for design certification. An evaluation and summary of the quality assurance program for the Battelle tests is provided in Reference 21.

The depletion rate assumed in the analysis is based on a generic passive autocatalytic recombiner application as described in Reference 19, and is expected to be representative of a number of vendor's recombiners. The calculated containment hydrogen concentration presented in Figures 6.2.4-1 and 6.2.4-2 is based on the assumptions and analysis discussed in subsection 6.2.4.3. The results demonstrate abundant margin for system performance. Further, the hydrogen concentration following an accident with only one of the two available passive autocatalytic recombiners operating within containment demonstrates significant margin to maintaining hydrogen concentrations below the recommendations of Regulatory Guide 1.7, Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident.

The recombiners are safety-related equipment. They are seismic Category I and are qualified for the post-loss of coolant accident environment. The recombiners require no power supply and are self-actuated by the presence of the reactants (hydrogen and oxygen).

A summary of component data for the hydrogen recombiners is provided in Table 6.2.4-2.

6.2.4.2.3 Hydrogen Ignition Subsystem

The hydrogen ignition subsystem is provided to address the possibility of an event that results in a rapid production of large amounts of hydrogen such that the rate of production exceeds

the capacity of the recombiners. Consequently, the containment hydrogen concentration will exceed the flammability limits. This massive hydrogen production is postulated to occur as the result of a degraded core or core melt accident (severe accident scenario) in which up to 100 percent of the zirconium fuel cladding reacts with steam to produce hydrogen.

The hydrogen ignition subsystem consists of ~~58~~60 hydrogen igniters strategically distributed throughout the containment. Since the igniters are incorporated in the design to address a low-probability severe accident, the hydrogen ignition system is not Class 1E. Although not class 1E, the igniter coverage, distribution and power supply has been designed to minimize the potential loss of igniter protection globally for containment and locally for individual compartments. The igniters have been divided into two power groups. Power to each group will be normally provided by offsite power, however should offsite power be unavailable, then each of the power groups is powered by one of the onsite non-essential diesels and finally should the diesels fail to provide power then approximately 4 hours of igniter operation is supported by the non-Class 1E batteries for each group. Assignment of igniters to each group is based on providing coverage for each compartment or area by at least one igniter from each group.

The locations of the igniters are based on evaluation of hydrogen transport in the containment and the hydrogen combustion characteristics. Locations include compartmented areas in the containment and various locations throughout the free volume, including the upper dome.

For enclosed areas of the containment at least two igniters are installed. The separation between igniter locations is selected to prevent the velocity of a flame front initiated by one igniter from becoming significant before being extinguished by a similar flame front propagating from another igniter. The number of hydrogen igniters and their locations are selected considering the behavior of hydrogen in the containment during severe accidents. The likely hydrogen transport paths in the containment and hydrogen burn physics are the two important aspects influencing the choice of igniter location.

The primary objective of installing an igniter system is to promote hydrogen burning at a low concentration and, to the extent possible, to burn hydrogen more or less continuously so that the hydrogen concentration does not build up in the containment. To achieve this goal, igniters are placed in the major regions of the containment where hydrogen may be released, through which it may flow, or where it may accumulate. The criteria utilized in the evaluation is provided in Table 6.2.4-6. The location of igniters throughout containment is provided in Figures 6.2.4-5 through 6.2.4-12. The location of igniters is also summarized in Table 6.2.4-7 identifying subcompartment/regions and which igniters by power group provide protection. The locations identified are considered approximations ($\pm$ 2.5 feet) with the final locations governed by the installation details. The igniter locations identified are considered approximations ($\pm$ 2.5 feet) with the final locations governed by the installation details.

The igniter assembly is designed to maintain the surface temperature within a range of 1600 to 1700°F in the anticipated containment environment following a loss of coolant accident. A spray shield is provided to protect the igniter from falling water drops (resulting from

condensation of steam on the containment shell and on nearby equipment and structures). Design parameters for the igniters are provided in Table 6.2.4-3.

6.2.4.2.4 Containment Purge

Containment purge is not part of the containment hydrogen control system. The purge capability of the containment air filtration system (see subsection 9.4.7) can be used to provide containment venting prior to post-loss of coolant accident cleanup operations.

6.2.4.3 Design Evaluation (Design-Basis Accident)

6.2.4.3.1 Hydrogen Production and Accumulation

Following a loss of coolant accident, hydrogen may be added to the reactor containment atmosphere by reaction of the zirconium fuel cladding with water, by radiolysis of water, by corrosion of materials of construction, and by release of the hydrogen contained in the reactor coolant system. The assumptions used in calculating the hydrogen release to containment are listed in Table 6.2.4-4.

6.2.4.3.1.1 Zirconium-Water Reaction

Zirconium fuel cladding reacts with steam according to the following equation:

$$Zr + 2\ H_2O \rightarrow ZrO_2 + 2\ H_2 + \text{heat}$$

There is 8.5 standard cubic feet (SCF) of hydrogen produced for each pound of zirconium that is reacted.

The extent of the zirconium-water reaction is dependent on the effectiveness of the core cooling. An evaluation of the AP600 design shows that there is no zirconium-water reaction during a design basis accident. The NRC model presented in Regulatory Guide 1.7 conservatively assumes that the cladding oxidizes to a depth of 0.00023 inch. For the 0.0225 inch cladding thickness used for AP600 fuel, this constitutes 1.09 percent of the zirconium. The hydrogen produced by the reaction of zirconium is 3000 standard cubic feet. This hydrogen is assumed to be released to the containment atmosphere at the beginning of the accident.

6.2.4.3.1.2 Radiolysis of Water

Water radiolysis is a complex process involving reactions of numerous intermediates. However, the overall radiolytic process may be described by the equation:

$$H_2O \rightleftarrows H_2 + \frac{1}{2}\ O_2$$

Table 6.2.4-6

IGNITER LOCATION CRITERIA

- A sufficient number of igniters should be placed in the major transport paths (including dominant natural circulation pathways) of hydrogen so that hydrogen can be burned continuously close to the release point. This prevents hydrogen from preferentially accumulating in a certain region of the containment.

- Igniters (minimum of 2) should be located in major regions or compartments where hydrogen may be released, through which it may flow, or where it may accumulate.

- It is preferable to ignite a hydrogen-air mixture at the bottom so that upward flame propagation can be promoted at lean hydrogen concentrations. Igniters within each subcompartment should be located in the vicinity of, and above, the highest potential release location within the subcompartment.

- In compartments with relatively small openings in the ceiling, the potential may exist for the hydrogen-air mixture to rise and to collect near the ceiling. Therefore, one or more igniters should be placed near the ceiling of such compartments. Igniter coverage should be provided within the upper 10% of the vertical height subcompartments or 10 ft from the ceiling whichever is less. In cases where the highest potential release point is low in the compartment, both this and the previous criteria should be considered.

- To the extent possible, igniters should be placed away from walls and other large surfaces so that a flame front created by ignition at the bottom of a compartment can travel unimpeded up to the top.

- A sufficient number of igniters should be installed in long, narrow compartments (corridors) so that the flame fronts created by the igniters need to travel only a limited distance before they merge. This limits the potential for significant flame acceleration.

- Igniter coverge should be provided to contol combustion in all areas where oxygen rich air may enter into an inerted region with combustible hydrogen levels during and accident scenario.

- Igniters should be located above the flood level, if possible. Those which may be flooded should have redundant fuses to protect the power supply.

- In locations where the potential hydrogen release location can be defined, i.e. above the IRWST spargers, at IRWST vents, etc igniter coverage should be provided as close to the source as feasible.

- Provisions for installation, maintenance, and testing must also be considered.

Table 6.2.4-7

SUBCOMPARTMENT/AREA IGNITER COVERAGE

Subcompartment	Igniter Coverage (Elevation)[1] Power Group 1	Power Group 2
Reactor Cavity	1(El 91′) 3 (El 95′) 13, 5 (El 120′) 8, 12 (El 139′)	4 (El 95′) 2 (El 99′) 11, 7 (El 120′) 6, 14 (El 139′)
Loop Compartment 01	13 (El 120′) 12 (El 139′)	11 (El 120′) 14 (El 139′)
Loop Compartment 02	5 (El 120′) 8 (El 139′)	7 (El 120′) 6 (El 139′)
Pressurizer Compartment	49 (El 154′) 60 (El 135′)	50 (El 154′) 59 (El 135′)
Tunnel connecting Loop Compartments	1 (El 91′) 3 (El 95′) 31 (El 120′)	4 (El 95′) 2 (El 99′) 30 (El 120′)
Southeast Valve Room	21 (El 105′)	20 (El 105′)
Southeast Accumulator Room	21 (El 105′)	20 (E. 105′)
East Valve Room	18 (El 105′)	19 (El 105′)
Northeast Accumulator Room	18 (El 105′)	17, 19 (El 105′)
Northeast Valve Room	18 (El 105′)	17 (El 105′)
North CVS Equipment Room	34 (El 105′)	33 (El 105′)
Lower Compartment Area (CMT and Valve area)	22 (El 133′) 27, 28, 29, 31, 32 (El 120′)	23, 24, 25 (El 133′) 26, 30 (El 120′)
IRWST Compartment	35, 37 (El 135′)	36, 38 (El 135′)
IRWST Interior	9 (El 133′)	10 (El 133′)
IRWST Inlet	16 (El 133′)	15 (El 133′)
Refueling Cavity	55 (El 120′) 58 (El 132′)	56 (El 120′) 57 (El 132′)
Upper Compartment		
Lower Region	39, 42, 44, 43, 47 (El 162′)	40, 41, 45, 46, 48 (El 162′)
Mid Region	51, 54 (El 210′)	52, 53 (El ′210)

Note:
1. Elevations are approximate.

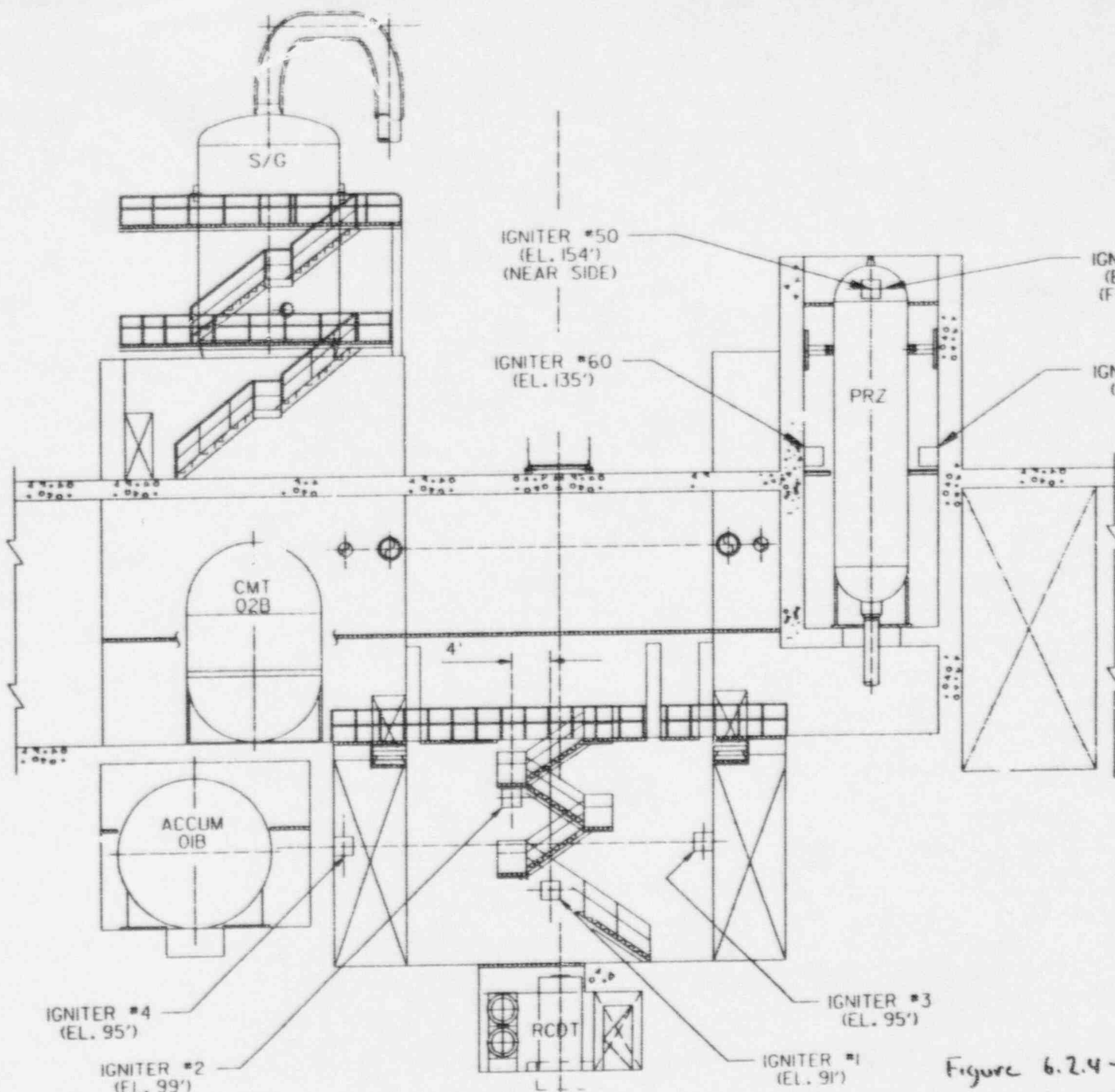


Figure 6.2.4-5
Hydrogen Igniter Locations Sheet of 7
Section View

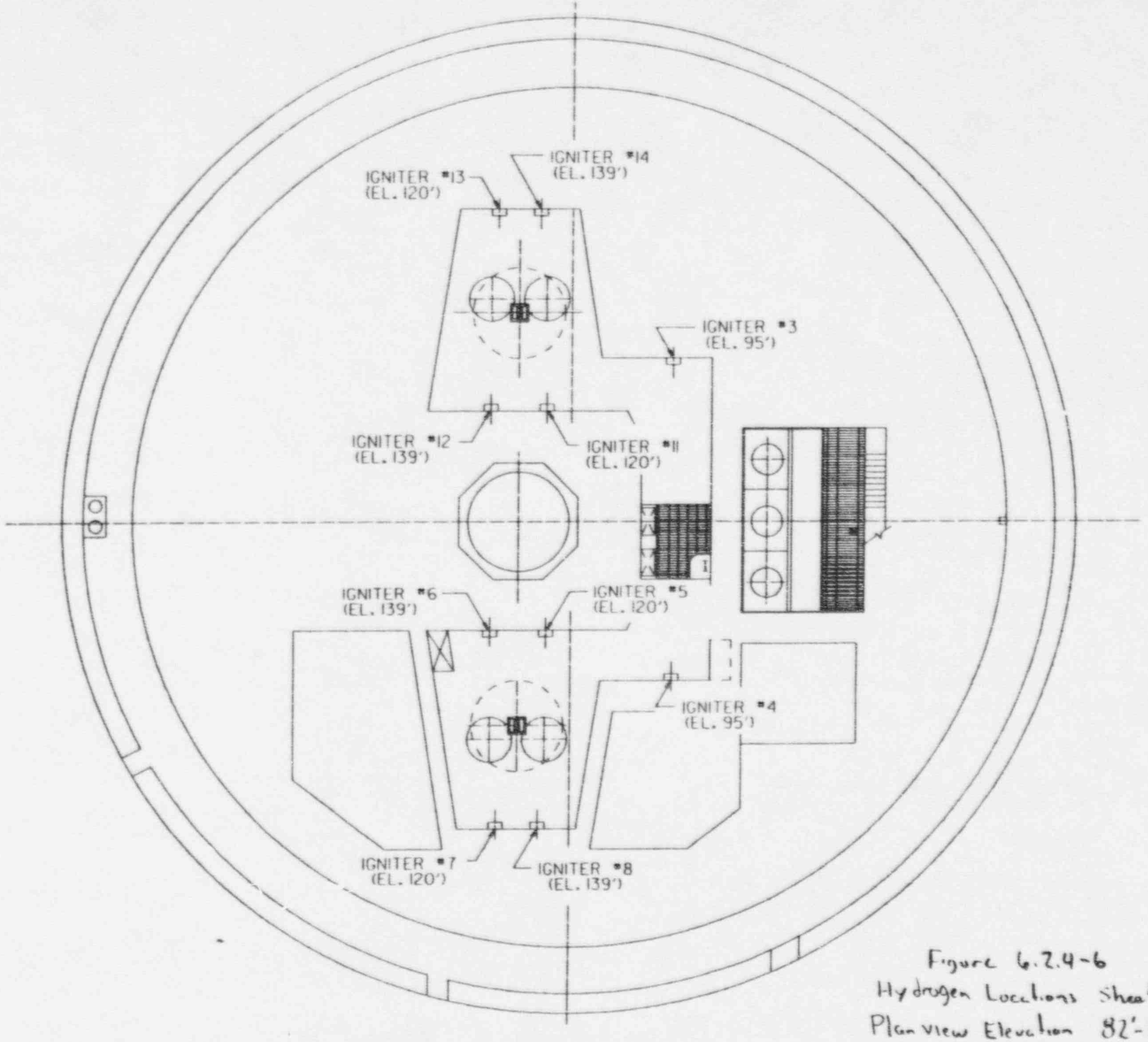


Figure 6.2.4-6
Hydrogen Locations Sheet 2 of 7
Planview Elevation 82'-6"

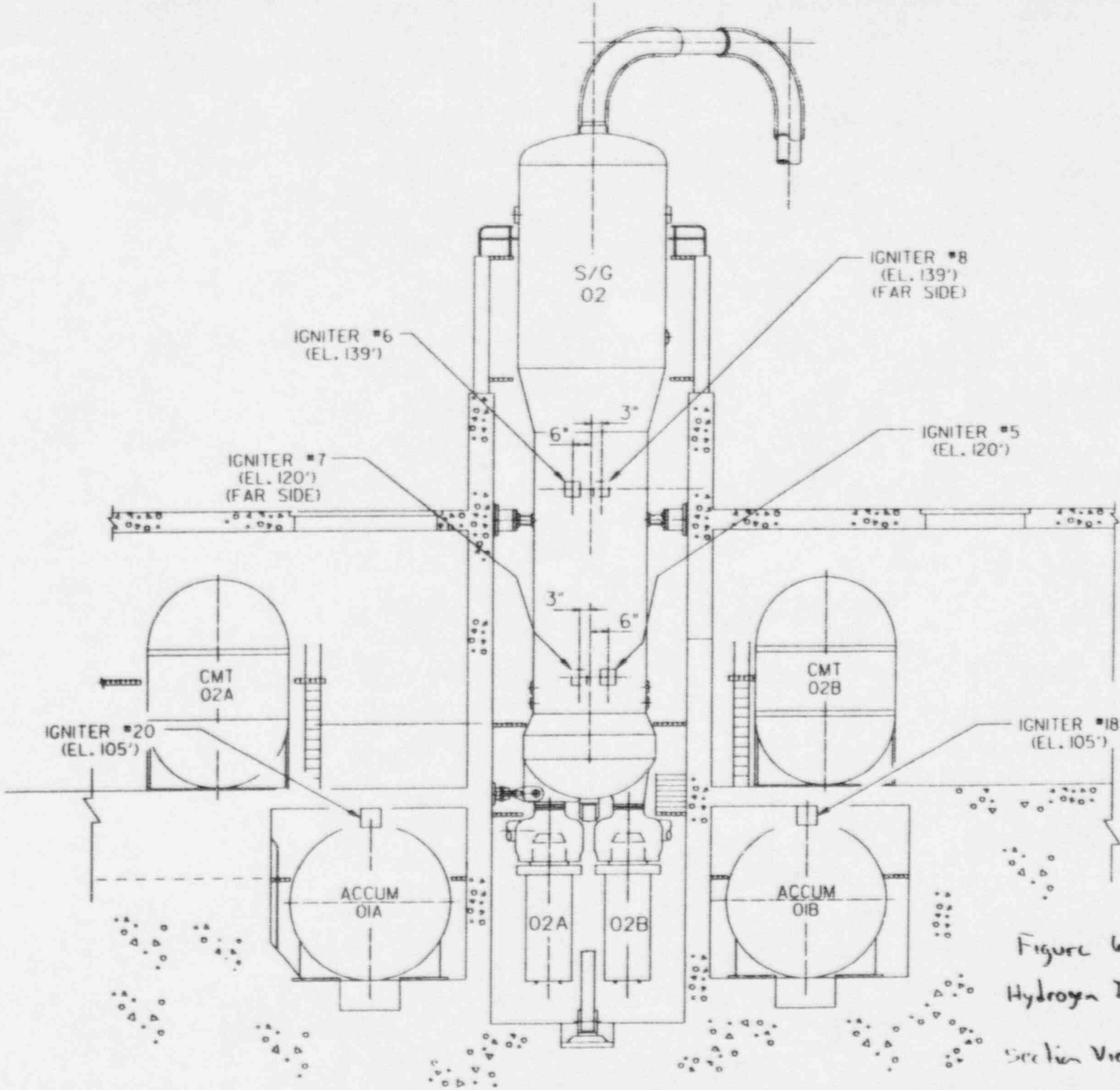


Figure 6.2.4-7
Hydrogen Igniter Locations
Sheet 3 of 7
Section View

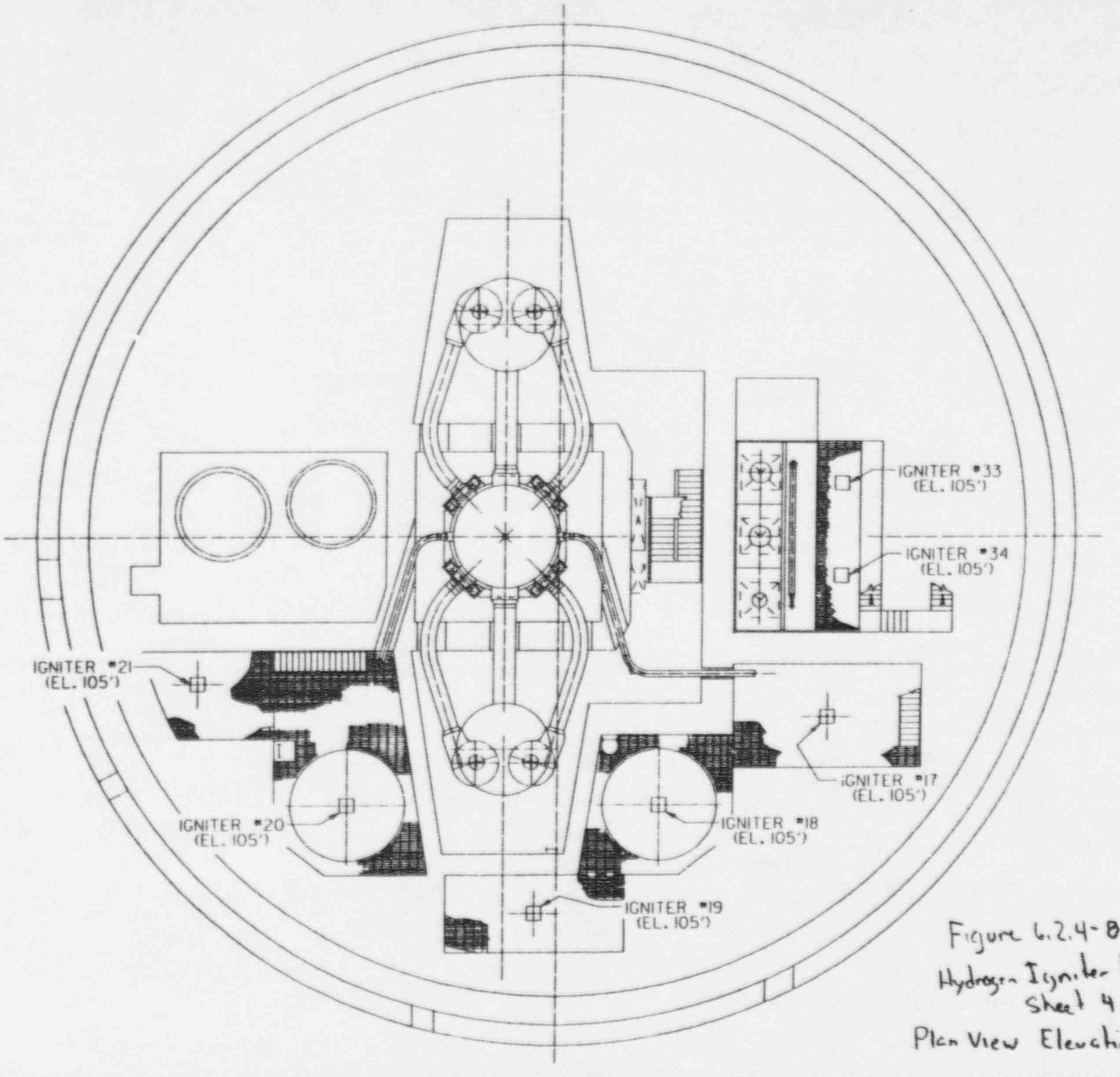
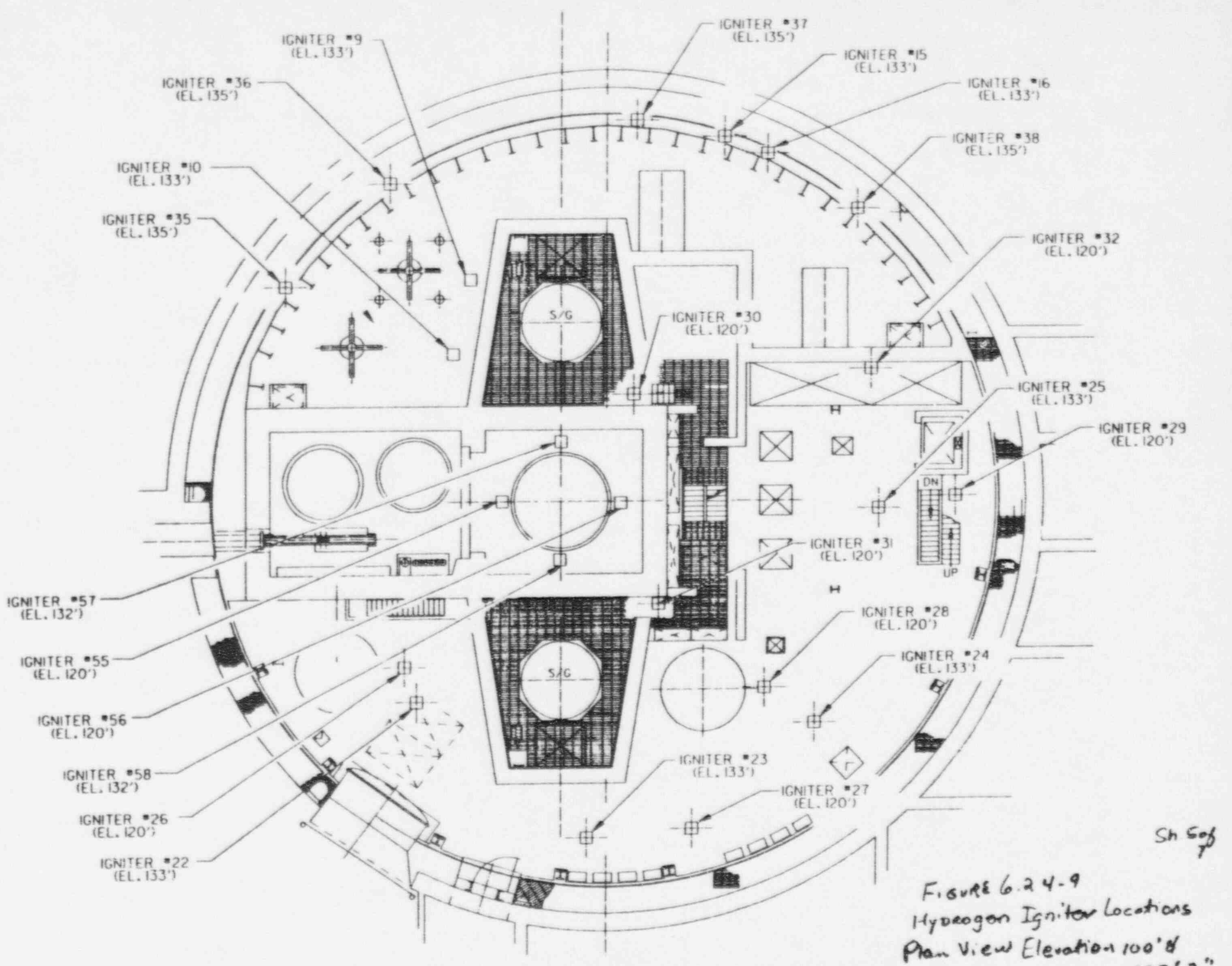


Figure 6.2.4-8
Hydrogen Igniter Locations
Sheet 4 of 7
Plan View Elevation 46'-6"



Sh. 6 of 7

FIGURE 6.2.4-9
Hydrogen Ignitor Locations
Plan View Elevation 100' & 107'-2"

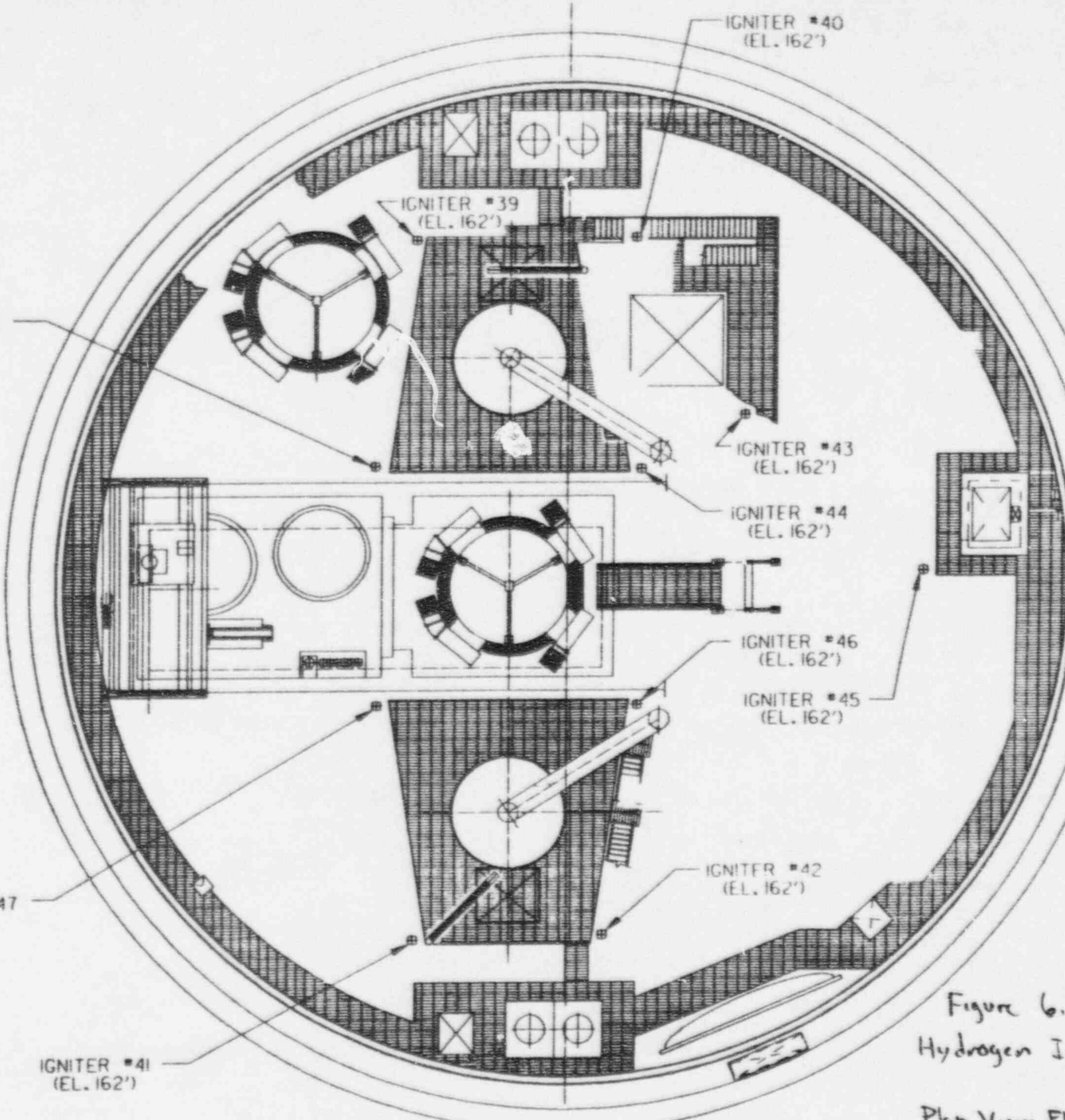


Figure 6.2.4-10
Hydrogen Igniter Location Sheet 6 of 7
Plan View Elevation 160'-6" & 153'-0"

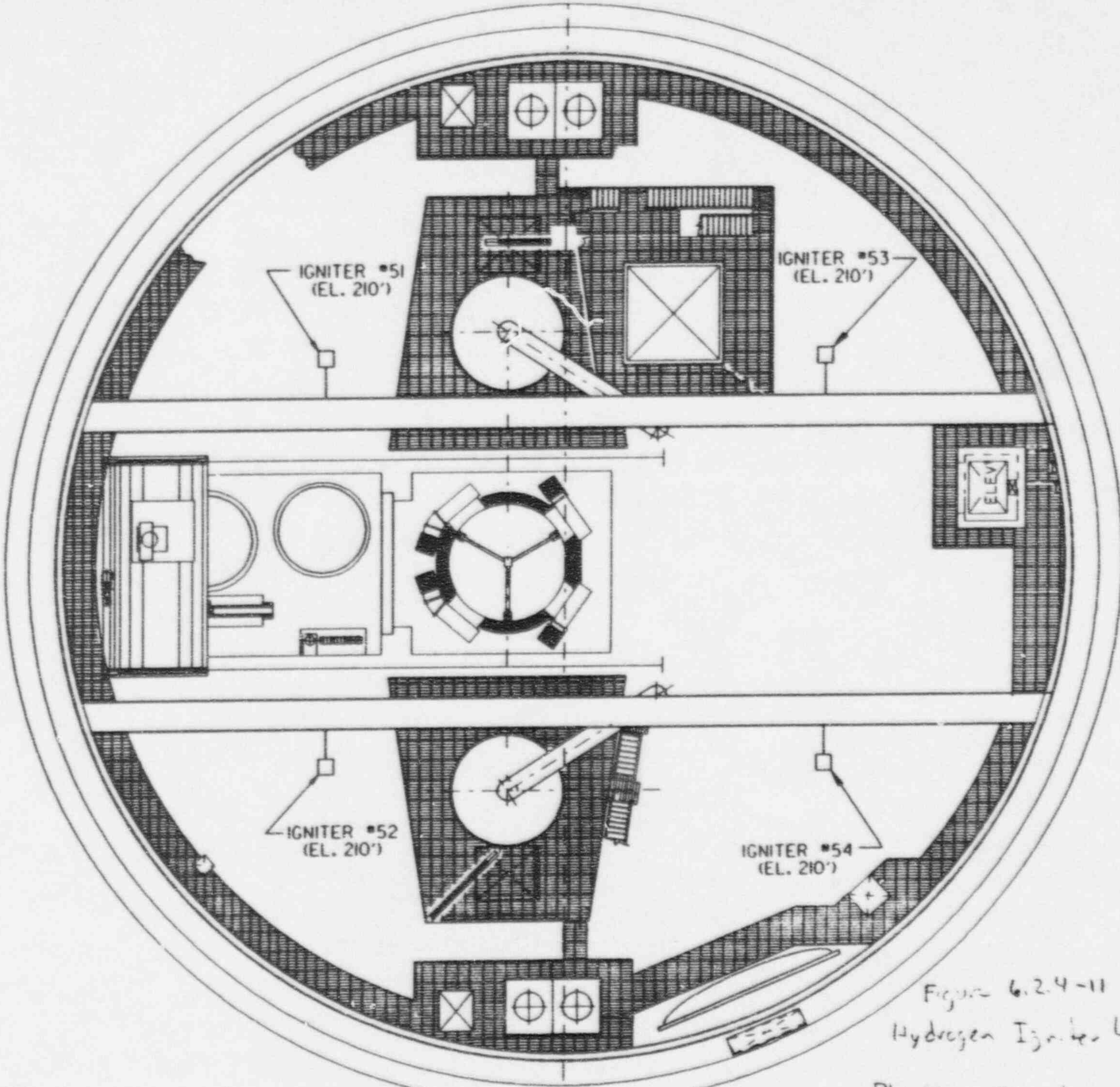


Figure 6.2.4-11
Hydrogen Igniter Locations
Sheet 7 of 7
Plan View Elevation 210'

FAX to DINO SCALETTI

February 17, 1997

CC: Sharon or Dino, please make copies for: Diane Jackson
Ted Quay

Don Lindgren
Richard Orr
Cindy Haag
Bob Lutz
Ed Cummins
Bob Vijuk
Brian McIntyre

OPEN ITEMS FOR SSAR SECTION 3.8.2

This is a background package for the remaining open items for SSAR section 3.8.2. SSAR section 3.8.2 is of interest because by our joint NRC/W schedule, the FSER for this section should be turned into Projects by the end of March. There are 7 Open Items with NRC Status of Action W. One (1) of these items (708) still requires some Westinghouse action. Westinghouse believes the other six (6) items (681, 706, 1888, 3269, 3270 and 3271) were resolved by the issue of letter NSD-NRC-97-4981 of 2/11/97. We request that NRC acknowledge receipt of this letter by changing the "NRC Action" of these items to something other than "Action W". Note that 3 of these same items were addressed in the fax i sent you covering PRA Chapter 42. Currently, our records show no additional outstanding Westinghouse action required for section 3.8.2, except item 708, and we request that NRC provide a definitive action for Westinghouse or provide direction to change the status of these items. We recommend "Action N". Thank you.

Jim

Jim Winters
412-374-5290

AP600 Open Item Tracking System Database: Executive Summary

Date: 2/17/97

Selection: [nrc st code]='Action W' And [DSER Section] like '3.8.2*' Sorted by Item #

Item No.	Branch	DSER Section/ Question	Type	Title/Description Detail Status	Resp Engineer	(W) Status	NRC Status	Letter No. / Date
681	NRR/ECGB	3.8.2.4-3	DSER-OI	Westinghouse should demonstrate that calculated stresses in the vicinity of the concentrated masses based on an equivalent static analysis bound the local stresses computed by the dynamic analysis. Local dynamic analyses are performed for the responses of the local masses using as input the floor response spectra at the appropriate elevation of the containment vessel. The local analyses have been added in SSAR Rev. 3. Action W - Discussed at meeting at CBI 8/30 - 31/95: Westinghouse stated that detailed analyses and design of the containment vessel in the vicinity of concentrated masses are beyond the scope of the AP600 standard design. However, Westinghouse agreed to expand SSAR Section 3.8.2.4.1.2 to include (1) a detailed description of methods to be used for the dynamic analysis of local masses, (2) the approach for analyzing the local buckling potential of the containment shell adjacent to major penetrations, (3) the stress redistribution criteria to be applied for the shell adjacent to local masses, and (4) methods for evaluating the compressive strength of the containment shell in the vicinity of major penetrations. Closed - Response provided by NSD-NRC-97-4981 of 2/11/97.	Orr /WSL / NRCCV	Closed	Action W	NTD-NRC-95-4464
706	NRR/ECGB	3.8.2.4-28	DSER-OI	Westinghouse should provide in the SSAR an assessment of the pressure capability of the main steamline and main feedwater line bellows, a corresponding failure probability distribution curve, and the impact on the overall cumulative failure probability curve. The effect of containment pressure on the bellows was addressed in the response to RAI 720.206. This response has been incorporated in SSAR Subsection 3.8.2.4.2.6, Rev 3. The bellows remain intact when the containment shell remains elastic and imposed deflections remain close to the design conditions. Failure of the bellows is assumed to occur once the containment cylinder yields. This mechanism is already included in the failure probability curve for the cylinder. Discussed in meeting at CBI 8/30 - 31/95. Additional discussion was provided in draft SSAR or PRA report that bellows failure mode and fragility is included in the containment vessel cylinder yield failure mode. Closed: further review is under new RAI 220.99 transmitted by NRC letter of April 4, 1996 NRC Status Update provided in September 5, 1996 letter: This staff does not agree that this item is closed or resolved due to RAI# 220.99. Both this open item and OI# 3268 (RAI #220.99) should be tracked individually to resolution. Action Westinghouse Closed - Response provided by NSD-NRC-97-4981 of 2/11/97.	Orr	Closed	Action W	NTD-NRC-95-4464
708	NRR/ECGB	3.8.2.4-30	DSER-OI	Westinghouse should increase the thickness or use stiffeners (as in the ABB-CE System 80+ design) to meet the ASME Service Level C limits at the ambient temperature of 908 kPa (117 psig) for a 6.7 m (22-ft) diameter hatch, and 763.2 kPa (96 psig) for a 4.9-m (16- ft) diameter hatch. ASME have confirmed that the method used for the AP600 complies with ASME Code Case N 284. Westinghouse position is that use of code case N284 satisfies the deterministic Service Level C criteria approved by the commissioners. NRC staff will review N284, Revision 1 and the ASME confirmation of the AP600 interpretation. Closed: further review is under new RAI 220.100 transmitted by NRC letter of April 4, 1996 NRC Status Update provided in September 5, 1996 letter: This staff does not agree that this item is closed or resolved due to AI# 220.100. Both this open item and OI 3269 (RAI #220.100) should be tracked individually to resolution. Action Westinghouse	Orr/CBI	Action W	Action W	
1888	NRR/ECGB	3.8.2.4-1	DSER-COL	3.8.2.4-1 The COL applicant should demonstrate that EPAs to be used shall be at least as strong as the AP600 SCV. Discussed in meeting at CBI 8/30 - 31/95. Expand COL information to include demonstration that EPA satisfies Service Level C pressure and temperature requirement. Revise SSAR 3.8.6.1 to change "ultimate capacities" to "ultimate pressure capacities" Closed: additional clarification is requested under RAI 220.102 transmitted by NRC letter dated April 4, 1996. Closed - Response provided by NSD-NRC-97-4981 of 2/11/97.	Orr	Closed	Action W	NSD-NRC-97-4981

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Total Records: 7

Item No.	Branch	DSER Section/ Question	Type	Title/Description Detail Status	Resp Engineer	(W) Status	NRC Status	Letter No. /	Date
3269	NRR/ECGB	3.8.2	RAI-OI		CH42/Orr/Lutz	Closed	Action W	NSD-NRC-97-4981	

220.100 In SSAR Section 3.8.2.4.2.3, the factor of safety (FS) of 1.67 is used for equipment hatch covers ASME Service Level C limits.

Westinghouse estimated the critical buckling pressures for equipment hatches as 1.45 MPa (196 psig) for a 6.7 m (22 ft) diameter hatch and 1.21 MPa (161 psig) for a 4.9 m (16 ft) diameter hatch based on the classical buckling capacity of spherical shells subjected to external pressure and the capacity reduction factors specified in Baker et al., "Structural Analysis of Shells," pp. 253-254, McGraw-Hill, 1972, and in ASME Code Case N-284. The corresponding ASME Service Level C limits are 908 kPa (117 psig) and 763.2 kPa (96 psig) using the factor of safety (FS) of 1.67 as specified in Code Case N-284, respectively.

For the FS to be applied to the Service Level C pressure capacity, Westinghouse considered the equipment hatch cover buckling due to external pressure as the local buckling (FS = 1.67 from Code Case N-284). The hatch cover is a complete shell by itself with its own independent boundary and is subjected to pressure on its convex side due to the containment internal pressure. Therefore, the staff position is that the global buckling (FS = 2.5 from NE-3222) is the appropriate value. The ASME Service Level C pressure capacity is 763.2 kPa (96 psig) with FS of 1.67 and 545.4 kPa (64.4 psig) with an FS of 2.5.

Based on Code Case N-284, the local buckling is defined as the buckling of the shell plate between stiffeners. The flange of the cover can act as a stiffening element around the periphery of the spherical cap. However, the stiffening effect is limited to $(Rt)^{1/2}$ or 35.3 cm (13.9 in) from the edge. The entire arc length from the center of the hatch cover to the flange is 255.3 cm (100.5 in). The remaining 218.4 cm (86 in) arc should be considered as unstiffened, therefore, the global buckling criteria should be applied to this unstiffened region. In the draft safety evaluation report (DSER), the staff noted that Westinghouse's assumption of local buckling for the equipment hatch cover under external pressure was not acceptable. The staff requested that Westinghouse increase the thickness or use stiffeners (e.g., ABB-CE System 80+ design) to meet the ASME Service Level C limits at the ambient temperature of 908 kPa (117 psig) for a 6.7 m (22 ft) diameter hatch and 763.2 kPa (96 psig) for a 4.9 m (16 ft) diameter hatch. This was Open Item 3.8.2.4-30.

The staff performed independent analysis for the equipment hatch covers using the ALGOR computer code with fixed boundary conditions and no imperfection. Using ALGOR, the staff predicted the buckling pressure, Pbuckling, as 1.38 MPa (185.12 psig) and 1.57 MPa (212.96 psig) for 4.9 m (16 ft) and 6.7 m (22 ft) equipment hatch covers, respectively. In both cases, the buckling was predicted to occur near the top portion.

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Item No	Branch	DSER Section/ Question	Type	Title/Description Detail Status	Resp Engineer	(W) Status	NRC Status	Letter No /	Date
				For the reasons discussed above, the staff considers the equipment hatch covers buckling as a global failure mode. There is a potential for radioactive gas leakage through the equipment hatch sleeve/gasket once buckling occurs. Thus, the leaktight integrity of the containment is jeopardized. On this basis, the staff finds that a higher FS of 2.5 based on NE-3222 should be applied.					
				Closed - Response provided by NSD-NRC-97-4981 of 2/11/97.					
3270	NRR/ECGB	3.8.2	RAI-OI		CH42/Orr/Lutz	Closed	Action W	NSD-NRC-97-4981	
				220.101 Westinghouse evaluated an additional BOSOR-5 analysis with stress-strain curves accounting for the effects of residual stresses on the buckling of cylindrical shells due to axial compression and/or external pressure. The failure mode was found to be an axisymmetric plastic collapse resulting from excessive vertical displacements at the pole. The maximum displacement was 1.09 m (43 in) at 1.45 MPa (195 psig). This information was requested by the staff to be provided in SSAR as discussed in RAI 6 (NRC letter dated September 14, 1995). In its response (NTD-NRC-96-4617 dated January 4, 1996), Westinghouse stated that the plastic collapse is bounded by the case for knuckle buckling without specific information. Provide this information in the SSAR.					
				Closed - Response provided by NSD-NRC-97-4981 of 2/11/97.					
3271	NRR/ECGB	3.8.2	RAI-OI		CH42/Orr/Lutz	Closed	Action W	NSD-NRC-97-4981	
				220.102 In SSAR Section 3.8.2.4.2.5 mechanical and electrical penetrations are designed for a pressure of 90 psig at design temperature (280 F) for ASME Service Level C limits. In SSAR Section 3.8.2.4.2.8, however, the ASME Service Level C limit is 92 psig at 280 F from the containment ellipsoidal head plastic buckling. Clarify which pressure represents the ASME Service Level C limit at design temperature for the containment.					
				Closed - Response provided by NSD-NRC-97-4981 of 2/11/97.					

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FAX to DINO SCALETTI

February 14, 1997

CC: Sharon or Dino, please make copies for: Bill Huffman
Tom Kenyon
Ted Quay

Don Lindgren
Jim Grover
Gordon Israelson
Tom Hayes
Brian McIntyre
Ed Cummins
Bob Vijuk

OPEN ITEMS 21 and 4617

This is a background package for open items 21 and 4617. They are of interest because by our joint NRC/W schedule, the FSER for Chapters 8 and 12 should have been turned into Projects by the end of February. Attached is a copy of the letter requesting exemptions from regulations required to resolve these and other open items. Item 21 is covered by exemption request 8 and item 4617 is covered by exemption request 2. It seems to be a reasonable request that NRC acknowledge receipt of this exemption request. Our records show no other outstanding Westinghouse actions for Chapters 8 or 12 and we request that NRC provide a definitive action for Westinghouse or provide direction to change the status of Items 21 and 4617. We recommend "Action N". Thank you.

Jim

Jim Winters
412-374-5290

Westinghouse Electric Corporation
Energy Systems

Box 355
Pittsburgh Pennsylvania 15230-0355

NRC-NSD-97-4986
DCP/NRC0741
Docket No.: STN-52-003

February 14, 1997

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

TO: T. R. QUAY

SUBJECT: AP600 EXEMPTIONS FROM REGULATIONS

Dear Mr. Quay:

In response to a request from the NRC staff and DSER Open Item 1.8-1, Westinghouse is submitting the enclosed summary of AP600 design and operational features that require exemptions from current regulations in 10 CFR Parts 50, 70, and 100. The exemptions from current regulations are required because of the incorporation of passive safety systems into the AP600 and other changes in technology. These exemptions have been provided for review by the NRC staff and included in the AP600 Standard Safety Analysis Report (SSAR).

The enclosure summarizes the basis for each and the exemptions consistent with the requirements of NRC regulations. It is Westinghouse's understanding that the acceptability of the exemptions will be documented in the Final Safety Evaluation Report and codified in the AP600 Design Certification Rule.

If you have any questions please contact Donald A. Lindgren at (412) 374-4856.

Brian A. McIntyre

Brian A. McIntyre, Manager
Advanced Plant Safety and Licensing

jml

Enclosure

cc: T. Kenyon, NRC (1 Enclosure)

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AP600 DOCUMENT COVER SHEET

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0058.FRM

AP600 CENTRAL FILE USE ONLY: TDC: ______ IDS: I ______ S ______
RFS#: RFS ITEM #:

AP600 DOCUMENT NO.	REVISION NO.		ASSIGNED TO
GW-GL-010	0	Page 1 of 7	B. A. McIntyre

ALTERNATE DOCUMENT NUMBER: WORK BREAKDOWN #: 2.1

DESIGN AGENT ORGANIZATION: Westinghouse Electric

TITLE: Exemption to NRC Regulations

ATTACHMENTS:	DCP #/REV. INCORPORATED IN THIS DOCUMENT REVISION:
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ORIGINATOR	SIGNATURE/DATE	
D. A. Lindgren	D. A. Lindgren	2/14/97
AP600 RESPONSIBLE MANAGER	SIGNATURE*	APPROVAL DATE
B. A. McIntyre		2/14/97

*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.

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Form 58202G(5/94)

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CATEGORY "F" – Consists of administrative plans and administrative reports.

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EXEMPTIONS TO NRC REGULATIONS

In some cases the regulations in 10 CFR Parts 50, 70, and 100 are not appropriate for the design certification of the AP600. These rules are not appropriate because of the incorporation of passive safety systems into the AP600 and other changes in technology.

1. Exemption from dedicated containment penetration requirement. SSAR Section 1.9.3

This requirement is as specified as 10 CFR 50.34 (3)(iv) Dedicated Containment Penetrations (NUREG-0660 Item II.B.8) *Provide one or more dedicated containment penetrations, equivalent in size to a single 3-foot diameter opening, in order not to preclude future installation of systems to prevent containment failure, such as a filtered vented containment system.*

The containment analysis for the AP600, including PRA and severe accident assessments, demonstrate that the containment, with its passive heat rejection capability, does not need a future installation of such a system to prevent overpressurization.

This evaluation of AP600 containment integrity meets the exemption criteria of 10 CFR 50.12 (a)(1) and (a)(2)(ii).

2. Exemption from General Design Criteria 17 requirement for physically independent circuit (second off-site electrical power source) SSAR Section 3.1

This requirement is found in 10 CFR 50 Appendix A, General Design Criterion 17 - Electrical Power Systems

An onsite electric power system and an offsite electric power system shall be provided to permit functioning of structures, systems, and components important to safety. The safety function for each system (assuming that the other system is not functioning) shall be to provide sufficient capacity and capability to assure that (1) specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded as a result of anticipated operational occurrences and (2) the core is cooled and containment integrity and other vital functions are maintained in the event of postulated accidents.

The onsite electric power supplies, including the batteries, and the onsite electric distribution system shall have sufficient independence, redundancy, and testability to perform their safety functions, assuming a single failure.

Electric power from the transmission network to the onsite electric distribution system shall be supplied by two physically independent circuits (not necessarily on separate rights-of-way) designed and located so as to minimize, to the extent practical, the likelihood of their simultaneous failure under operating and postulated accident and environmental conditions. A switchyard common to both circuits is acceptable. Each of these circuits shall be designed to be available in sufficient time, following a loss of all onsite alternating current power supplies and other offsite electric power circuit, to assure that specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded. One of these circuits shall be designed to be available

within a few seconds following a loss of coolant accident to assure that core cooling, containment integrity, and other vital safety functions are maintained.

Provisions shall be included to minimize the probability of losing electric power from any of the remaining supplies as a result of, or coincident with, the loss of power generated by the nuclear power unit, the loss of power from the transmission network, or the loss of power from the onsite electric power supplies.

The AP600 plant design supports an exemption to the requirement of GDC 17 for two physically independent offsite circuits by providing safety-related passive systems for core cooling and containment integrity, and multiple nonsafety-related onsite and offsite electric power sources for other functions. See SSAR Section 6.3 for additional information on the systems for core cooling.

A reliable dc power source supplied by batteries provides power for the safety-related valves and instrumentation during transient and accident conditions.

The Class 1E dc and UPS system is the only safety-related power source required to monitor and actuate the safety-related passive systems. Otherwise, the plant is designed to maintain core cooling and containment integrity, independent of nonsafety-related ac power sources indefinitely. The only electric power source necessary to accomplish these safety-related functions is the Class 1E dc and UPS power system which includes the associated safety-related 120V ac distribution switchgear.

Although the AP600 is designed with reliable nonsafety-related offsite and onsite ac power that are normally expected to be available for important plant functions, nonsafety-related ac power is not relied upon to maintain the core cooling or containment integrity.

The nonsafety-related ac power system is designed such that plant auxiliaries can be powered from the grid under all modes of operation. During loss of offsite power, the ac power is supplied by the onsite standby diesel-generators. Preassigned loads and equipment are automatically loaded on the diesel-generators in a predetermined sequence. Additional loads can be manually added as required. The onsite standby power system is not required for safe shutdown of the plant.

This evaluation of power sources required for AP600 meets the exemption criteria of 10 CFR 50.12 (a)(1) and (a)(2)(ii).

3. **Exemption from use of ASME Code, Section III for Class C gas storage vessels** SSAR Subsection 3.2.2

10 CFR 50.55a requires that safety-related systems and components that are designated Quality C should meet the requirements of ASME Code Section III. For Class C air and gas storage tanks fabricated without welding, ASME Code, Section VIII, Appendix 22 provides a more appropriate set of criteria than does ASME Code, Section III, Class 3. The air storage tanks are constructed of forged, seamless pipe, with no welds.

The rules of Section VIII for gas storage vessels of the type included in AP600 for safety-related functions are found in Section VIII mandatory Appendix 22 and have been developed specifically for the storage of gas under high pressure. These tanks are used in a variety of industries to store gas.

These tanks are fabricated without welding. Because there is no weld joint to consider, weld strength reduction factors do not apply which increase the wall thickness.

The material for the integrally forged tanks is ordered to a material specification (SA-372) developed specifically for that type of tank. The material specification is included in Section II of the ASME Code. The tanks are formed from a forging that is essentially a seamless pipe. The ends are swaged down to reduce the size of the opening. After completion of the forming operation the vessel is heat treated. No welding is permitted in the fabrication of the vessel by the Section VIII, Appendix 22 rules required for this type of vessel. 10 CFR 50, Appendix B requirements and 10 CFR 21 apply to the manufacture of safety-related air and gas storage tanks.

This evaluation of AP600 air tank integrity meets the exemption criteria of 10 CFR 50.12 (a)(1), (a)(2)(ii), and (a)(2)(iii).

4. Exemption from operating-basis earthquake design requirement in 10 CFR Part 100, Appendix A. SSAR Section 3.7

Item I.M of SECY-93-087 established the following policy: The staff recommends that the Commission approve the approach to eliminate the OBE from the design of systems, structures, and components. When the OBE is eliminated from the design, no replacement earthquake loading should be used to establish the postulated pipe rupture and leakage crack locations. The SECY also established the requirements for consideration of seismic events for equipment qualification and fatigue evaluations.

The operating basis earthquake (OBE) has been eliminated as a design requirement for the AP600. Low-level seismic effects are included in the design of certain equipment potentially sensitive to a number of such events based on a percentage of the responses calculated for the safe shutdown earthquake. Criteria for evaluating the need to shut down the plant following an earthquake are established using the cumulative absolute velocity approach according to EPRI Report NP-5930 (Reference 1) and EPRI Report TR-100082 (Reference 17). For the purposes of the shutdown criteria in Reference 1 the operating basis earthquake for shutdown is considered to be one-third of the safe shutdown earthquake. The AP600 satisfies the policy outlined in SECY-93-087.

Removal of the OBE decreases the number of analyses required and results in improved piping and equipment support systems.

This evaluation of elimination of the OBE for the AP600 meets the exemption criteria of 10 CFR 50.12 (a)(1), (a)(2)(ii), and (a)(2)(iii).

5. Exemption from use of ASME OM Code 1987 Edition. SSAR Subsection 3.9.6

10CFR 50.55a specifies that the OMa-1988 Addenda to the 1987 Edition of the OM requirements be used for the requirements for inservice testing of active safety-related valves. The 1990 OM Code represented a significant restructuring of the format of the requirements. The information in the AP600 SSAR on inservice testing has been developed and reviewed using the 1990 Edition. The use of the 1990 requirements does not represent a diminution of the inservice testing requirements.

This evaluation of AP600 inservice testing requirements meets the exemption criteria of 10 CFR 50.12 (a)(1) and (a)(2)(ii).

6. **Exemption from ASME OM Code valve inservice testing frequency**. SSAR Subsection 3.9.6.3

10CFR 50.55a specifies the use of ASME OM requirements for inservice testing of active safety-related valves. The ASME OM Code frequency requirement for most valves, unless a deferral is justified, is three months. The inservice testing frequency for the AP600 automatic depressurization system valve is six months; .

Considerable experience has been used in designing and locating systems and valves to permit preservice and inservice testing required by Section XI of the ASME Code. Relief from the testing requirements of the ASME OM Code is requested when full compliance with requirements of the ASME OM Code of the Code is not practical or constructive. In such cases, specific information is provided which identifies the applicable code requirements, justification for the relief request, and the testing method to be used as an alternative. The following relief request has been identified to be included in design certification.

Automatic depressurization system stage 1 through 3 valve exercise testing represents a risk of loss of reactor coolant and depressurization of the reactor coolant system if the proper test sequence is not followed. For this reason, the frequency of this valve exercise testing should be minimized. Conversely, the probabilistic risk assessment assumes that valve reliability for these valves is a function of test frequency. The recommended test frequency considers these two factors. The recommended test frequency for the stage 1 through 3 automatic depressurization systems valve is every six months. The PRA results show that the AP600 meets its safety goals. The assumptions in the PRA are consistent with a six month test frequency for the ADS valves.

This evaluation of AP600 inservice testing relief request meets the exemption criteria of 10 CFR 50.12 (a)(1), (a)(2)(ii), and (a)(2)(iv).

7. **Exemption from leak rate testing requirements of Appendix J to 10 CFR Part 50.** SSAR Subsection 6.2.5

10 CFR 50, Appendix J provides testing requirements for containment leak rate testing. The AP600 containment leak rate testing program outlined in SSAR subsection 6.2.5 satisfies 10 CFR 50, Appendix J requirements except as described in SSAR Table 6.2.5-1 which is repeated as Table 1 of this document and lists specific exemptions and justifications. The exemptions include changes that reflect industry practice, updated industry standards, and provisions to extend testing intervals.

This evaluation of AP600 containment leak rate testing meets the exemption criteria of 10 CFR 50.12 (a)(1), (a)(2)(ii), and (a)(2)(iii).

8. Exemption from 10 CFR 70.24 requirement for criticality meters in the fuel storage area.
SSAR Subsection 11.5.6

10 CFR 70.24 requires monitoring for criticality in areas where more than specified amounts of special nuclear material are stored. The AP600 does not include criticality meters in the fuel storage area. The fuel rack limits the criticality of stored fuel to a K_{eff} less than 1.0 and obviates the need for criticality meters. Subsections 9.1.1.3 and 9.1.2.3 provide information on criticality. The design of the new fuel storage rack is such that K_{eff} remains less than or equal to 0.95 with new fuel of the maximum design basis enrichment. For a postulated accident condition of flooding of the new fuel storage area with unborated water, K_{eff} does not exceed 0.98.

The design of the spent fuel racks is such that K_{eff} remains less than or equal to 0.95 under design basis conditions, including fuel handling accidents. Because of the close spacing of the cells, it is impossible to insert a fuel assembly in other than design locations. Inadvertent insertion of a fuel assembly between the rack periphery and the pool wall or placement of a fuel assembly across the top of a fuel rack is considered a postulated accident, and as such, realistic initial conditions such as boron in the pool water are assumed. These accident conditions have an acceptable K_{eff} of less than 0.95.

This evaluation of AP600 fuel rack criticality meets the exemption criteria of 10 CFR 70.14 (a).

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Table 1
EXCEPTIONS TO 10 CFR 50 APPENDIX J LEAK TESTING REQUIREMENTS

Appendix J Requirement	AP600 Exception and Justification
Paragraph III.A.1.(a) - Type A tests are required to be terminated if excessive leak paths, which would interfere with satisfactory completion of the test, are identified.	Type A tests are to be conducted in accordance with ANSI-56.8, which permits testing to proceed provided that the leak(s) can be isolated and that subsequent local leak rate testing is performed to demonstrate that the Type A test criteria are met. This approach can potentially reduce plant outage time and is in accordance with ANSI-56.8 and industry practice.
Paragraph III.A.3.(a) - Type A tests shall be conducted in accordance with the provisions of ANSI N45.4-1972.	Type A tests are to be conducted in accordance with ANSI-56.8, which superseded ANSI N45.4. Leak rate testing will be conducted pursuant to 10 CFR 50, Appendix J, Option A or Option B. Option A will be conducted in accordance with ANSI-56.8 while Option B will be conducted with the methodology defined in NEI 94-01, Revision 0 as modified by Regulatory Guide 1.163 and in accordance testing procedures of ANSI-56.8.
Paragraph III.A.3.(a) - A Type A test duration of 24 hours is required.	Type A tests are to be conducted for a minimum of 8 hours, in accordance with ANSI-56.8. Industry experience has shown that accurate test results can be achieved in less than 24 hours
Paragraph III.D.1.(a) - Three Type A tests are to be performed at approximately equal intervals during each 10-year service period.	Type A tests under 10 CFR 50, Appendix J, Option A are to be conducted at intervals not exceeding four years, except that, if the test interval ends while containment integrity is not required or is required solely for cold shutdown or refueling activities, the test interval may be extended indefinitely provided all deferred testing is successfully completed prior to the time containment integrity is required. The 4-year interval accommodates extended fuel cycles without requiring Combined License applicants to perform excessive Type A tests. The exception complies with Appendix J and meets the intent of Section XI of the ASME Boiler and Pressure Vessel Code.
Paragraph III.D.2 - Type B tests are required to be performed at intervals not greater than 2 years.	Type B tests under 10 CFR 50, Appendix J, Option A are to be conducted at intervals not exceeding 30 months, except that if the test interval ends while containment integrity is not required or is required solely for cold shutdown or refueling activities, the test interval may be extended indefinitely provided all deferred testing is successfully completed prior to the time containment integrity is required.
Paragraph III.D.2.(b)(i) - Air locks shall be tested prior to initial fuel loading and at 6-month intervals thereafter at a pressure not less than P_a.	Type B testing under 10 CFR 50, Appendix J, Option A is to be conducted in accordance with ANSI-56.8.
Paragraph III.D.3 - Type C tests are required to be performed at intervals not greater than 2 years.	Type C tests under 10 CFR 50, Appendix J, Option A are to be conducted at intervals not exceeding 30 months, except that if the test interval ends while containment integrity is not required or is required solely for cold shutdown or refueling activities, the test interval may be extended indefinitely provided all deferred testing is successfully completed prior to the time containment integrity is required.

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FAX to DINO SCALETTI

February 14, 1997

CC: Sharon or Dino, please make copies for: Bill Huffman
Ted Quay

Robin Nydes
Steve Kerch
Bob Vijuk
Ed Cummins
Brian McIntyre

OPEN ITEMS FOR CHAPTER 18

This is a background package for the remaining open items for Chapter 18. Dino, this is a chapter that can really use your help. We think we have NRC staff buy in on most stuff, but the administrivia is lagging. I hope you can prove this belief correct and then help with the administrivia part. Chapter 18 is of interest because by our joint NRC/W schedule, the FSER for this chapter should be turned into Projects by the middle of March. Attached are copies of the OITS printout for the 17 items that show "Action W" in NRC Status. We still believe we have to provide NRC with additional information for two (2) of them (1395 and 1397). The others (15 of them) have had their additional information submitted. It seems a reasonable request that NRC acknowledge receipt of the change. Our records show no outstanding Westinghouse actions, except the two identified above, on this Chapter 18. We request that NRC provide a definitive action for Westinghouse or provide direction to change the status of the other 15 items. We recommend "Action N". If you need additional information or one of the background packages on any particular item, please send me an E-Mail or call. Thank you.

Jim

Jim Winters
412-374-5290

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AP600 Open Item Tracking System Database: Executive Summary

Date: 2/14/97

Selection: [nrc st code]='Action W' And [DSER Section] like '18*' Sorted by Item #

Item No	Branch	DSER Section/ Question	Type	Title/Description Detail Status	Resp Engineer	(W) Status	NRC Status	Letter No / Date
1354	NRR/HHFB	18.8.1.3-1	DSER-OI		MMIS	Closed	Action W	NTD-NRC-97-4961
1356	NRR/HHFB	18.8.1.3-3	DSER-OI		MMIS	Closed	Action W	NTD-NRC-97-4961

Item 1354

Westinghouse should provide information regarding the HSI design process guidance. Westinghouse should describe how evaluation results will be communicated to designers, incorporated into design guidance, and reflected in final design documentation. The process by which implementation guidance will be developed must also be described.

Meeting of 3/10/95:

Action N: NRC will review our design process as described in the SSAR, RAIs and in WCAP 12601, and WCAP 9817 and provide us feedback. The NRC will review the SSD procedures of WCAP 12601 to determine if they cover issues like "providing guidance to HSI designers for addressing / utilizing design inputs such as the FBTA results". Note that the procedures will be at a higher level, not specifically referring to the FBTA or other specific HFE analyses or reports. The staff will also review the Alarm System Design Guidelines and the Interim Report on Technical Principles for Computer Based Displays of Data (both on file in Rockville office). Informal feedback from NRC was that the display document does provide the intended guidance.

Meeting of 3/10/95: NRC reviewed our design process as described in the SSAR, RAIs and in WCAP 12601, and WCAP 9817 and provided feedback via a letter dated 7/25/95.

Action W: Review and address NRC feedback issues.

Revised SSAR Chapter 18 submitted in Revision 9, 7/31/96. This includes SSAR 18.8, Human System Interface Design and SSAR 18.2, Human Factors Engineering Program Management.
NRC letter of Dec 19 says W action for element 7. rkn 12/19
Closed with submittal of WCAP-14396 Rev 2. rkn 1/30/97

cof R

Item 1356

Westinghouse should provide information regarding HSI characteristics. Westinghouse should describe how potential problems associated with high workload will be identified early in the design process, and how the concerns noted in the evaluation above will be addressed. Westinghouse should also describe how the design of workstations (inside and outside the MCR) ensure support of optimal operator performance under a range of conditions.

Meeting of 3/10/95:

Same as the "Action N" note in the open item description field for dbase item number 1354.

Meeting of 3/10/95: NRCreviewed our design process as described in the SSAR, RAIs and in WCAP 12601 and WCAP 9817 and provided feedback. The NRC reviewed the SSD procedures of WCAP 12601 with respect to covering issues like "providing guidance to HSI designers for addressing / utilizing design inputs such as the FBTA results". Note that the procedures will be at a higher level, not specifically referring to the FBTA or other specific HFE analyses or reports. The staff also reviewed the Alarm System Design Guidelines and the Interim Report on Technical Principles for Computer Based Displays of Data (both on file in Rockville office). Feedback from NRC was that the display document does provide the intended guidance.

6/7/95: Fax of a response to this open item was sent to J. Bongarra. NRC reviewed the response and provided feedback indicating that the response is acceptable. See NRC letter dated 7/25/95. SSAR revision required for closure.

Revised SSAR Chapter 18 submitted in Revision 9, 7/31/96. This includes SSAR 18.8, Human System Interface Design and SSAR 18.2, Human Factors Engineering Program Management.

NRC letter rec'd 12/19 statusing Action w for El 7. rkn 12/19
Westinghouse activities completed with submittal of WCAP-14396 Rev 2. rkn 1/28/97

Item No	Branch	DSER Section/ Question	Type	Title/Description Detail Status	Resp Engineer	(W) Status	NRC Status	Letter No. /	Date
1360	NRR/HHFB	18.8.1.3-7	DSER-OI		ITAAC/MMIS	Closed	Action W	NSD-NRC-4875	
				Westinghouse should provide information regarding the HSI evaluation. Westinghouse should describe the rationale for the HSIs, design elements, and procedures selected for evaluation, and for the points in the design process at which the evaluations are to occur. Westinghouse should also describe the process for identifying and resolving conflicts in guidance, as well as the rationale for design decisions that conflict with guidance. Meeting of 3/10/95: Same as the "Action N" note in the open item description field for dbase item number 1354.					
				NRC reviewed the quideline documents and found them acceptable. See NRC letter dated 7/25/95. ITAAC required for closure. Revised SSAR Chapter 18 submitted in Revision 9, 7/31/96. This includes SSAR 18.8, Human System Interface Design and SSAR 18.2, Human Factors Engineering Program Management. Element 7. SSAR Rev 9 Ch 18 and WCAP-14396 Rev 1 submittals. Closed - ITAAC submitted by NSD-NRC-96-4875 of 11/7/96. Action W per NRC letter of Dec 19 for El 7. rkn 12/19					
1361	NRR/HHFB	18.8.1.3-8	DSER-OI		ITAAC/M..'IS	Closed	Action W	NSD-NRC-4875	
				Westinghouse should describe how the HSI design will be documented. Westinghouse should describe how the final HSI design will be documented, incorporating the bases given in the criterion. Meeting of 3/10/95: Same as the "Action N" note in the open item description field for dbase item number 1354.					
				NRC reviewed the quideline documents and found them acceptable. See NRC letter dated 7/25/95. ITAAC required for closure. Revised SSAR Chapter 18 submitted in Revision 9, 7/31/96. This includes SSAR 18.8, Human System Interface Design and SSAR 18.2, Human Factors Engineering Program Management. Element 7. SSAR Rev 9 Ch 18 and WCAP-14396 Rev 1 submittals. Closed - ITAAC submitted by NSD-NRC-96-4875 of 11/7/96. Action W per NRC letter of 12/19 for El 7. rkn 12/19					

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AP600 Open Item Tracking System Database: Executive Summary

Date: 2/14/97

Selection: [nrc st code]='Action W' And [DSER Section] like '18*' Sorted by Item #

Item No.	Branch	DSER Section/ Question	Type	Title/Description Detail Status	Resp Engineer	(W) Status	NRC Status	Letter No. /	Date
1365	NRR/HHFB	18.9.3-3	DSER-OI	Westinghouse should provide information regarding the writer's guide. Westinghouse should describe how the writer's guide will address the unique features of a paper- and computer-based presentation of procedures. Per 2/16/95 conference call between Jim Bongarra, John O'Hara & Kerch, Easter, Roth, Mumaw: Action W -- NRC agreed to resolution path "in principle" and we need to send revision to SSAR section. Action W Per 2/16/95 conference call between Jim Bongarra, John O'Hara & Kerch, Easter, Roth, Mumaw: Action W -- NRC agreed to resolution path "in principle" and we need to send revision to SSAR section. Westinghouse sent (via fax or fedex) to the NRC a draft revision to SSAR 18.9.8 on 8/18/95 which will provide the response to this open item. Action N: NRC to review draft of 18.9.8 ; see NRC fax of 2/20/96. ACTION W: to respond to concerns of the fax; was discussed in conference call of 3/21/96. Element 8. With submittal of SSAR Rev 9 ch 18 and WCAP-14690, the MMIS portion of this item is closed. Need to determine what is required to close ERG portion of item. Resolved - Per DCP/NRC0589, this item will be closed with submittal of the at-power ERG's. No ERG changes required. rkn 12/3	MMIS/ERG	Closed	Action W	NSD-NRC-96-4805	
1366	NRR/HHFB	18.9.3-4	DSER-OI	Westinghouse should provide information regarding the contents of procedures. Westinghouse should describe and provide a rationale for the differences, if any, between the paper- and computer-based presentations of the items in this criterion (or in NUREG-0899). Per 2/16/95 conference call between Jim Bongarra, John O'Hara & Kerch, Easter, Roth, Mumaw: Action W -- NRC agreed to resolution path "in principle" and we need to send revision to SSAR section. Action W Per 2/16/95 conference call between Jim Bongarra, John O'Hara & Kerch, Easter, Roth, Mumaw: Action W -- NRC agreed to resolution path "in principle" and we need to send revision to SSAR section. Westinghouse sent (via fax or fedex) to the NRC a draft revision to SSAR 18.9.8 on 8/18/95 which provided the response to this open item. Action N: NRC to review draft of 18.9.8 ; see NRC fax of 2/20/96. ACTION W: to respond to concerns of the fax, was discussed in conference call of 3/21/96. Element 8. With submittal of SSAR Rev 9 ch 18 and WCAP-14690, the MMIS portion of this item is closed. Need to determine what is required to close ERG portion of this item. Resolved - Per DCP/NRC0589, this item will be closed with submittal of the at-power ERG's. No ERG changes required. rkn 12/3	MMIS/ERG	Closed	Action W	NSD-NRC-96-4805	

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AP600 Open Item Tracking System Database: Executive Summary

Date: 2/14/97

Selection: [nrc st code]='Action W' And [DSER Section] like '18*' Sorted by Item #

Item No	Branch	DSER Section/ Question	Type	Title/Description Detail Status	Resp Engineer	(W) Status	NRC Status	Letter No. /	Date
1367	NRR/HHFB	18.9.3-5	DSER-OI		MMIS/ERG	Closed	Action W	NSD-NRC-96-4794	

Westinghouse should provide information regarding the symptom- based emergency operating procedures (EOPs). Westinghouse should submit the AP600-specific ERGs so that the staff can verify that the EOPs will be symptom-based.

Per 2/16/95 conference call between Jim Bongarra, John O'Hara & Kerch, Easter, Roth, Mumaw:

Action W -- NRC agreed to resolution path "in principle" and we need to send the ERGs and background documents to NRC.

Action W
Per 2/16/95 conference call between Jim Bongarra, John O'Hara & Kerch, Easter, Roth, Mumaw:
Action W -- NRC agreed to resolution path "in principle" and we need to send the ERGs and background documents to NRC. These will be sent via a phased approach with phase 1 ERGs to be sent 5/31/95.
Status update provided by phone (D. Jackson 8/21):
Action W is to complete analytical basis for AE-1, AES-1.2, AE-2, and supporting documentation for shutdown ERG. Closed with submittal of ERGs Rev 2 on Jan 10, 1997. rkn 1/15/97.

Closed - In Response to letter NSD-NRC-96-4794.
Action W - See NRC letter dated 12/9/96.

This item is closed with submittal of the ERGs Rev 2 Jan 10. rkn 1/15/97.

Item No	Branch	DSER Section/ Question	Type	Title/Description Detail Status	Resp Engineer	(W) Status	NRC Status	Letter No. /	Date
1368	NRR/HHFB	18.9.3-6	DSER-OI		MMIS/ERG	Closed	Action W	NSD-NRC-96-4805	

Westinghouse should provide information regarding the V&V procedure. Westinghouse should clarify the relationship of the EOP V&V to the M-MIS evaluation issues. The V&V process for hardcopy procedures should also be described.

Per 2/16/95 conference call between Jim Bongarra, John O'Hara & Kerch, Easter, Roth, Mumaw:

Action W -- NRC agreed to resolution path "in principle" and we need to send revision to SSAR section.

Action W
Per 2/16/95 conference call between Jim Bongarra, John O'Hara & Kerch, Easter, Roth, Mumaw: Action W -- NRC agreed to resolution path "in principle" and we need to send revision to SSAR section.
Westinghouse sent (via fax or fedex) to the NRC a draft revision to SSAR 18.9.8 on 8/18/95 which provided the response to this open item.
NRC to review draft of 18.9.8 ; see NRC fax of 2/20/96.
ACTION W: to respond to concerns of the fax; was discussed in conference call of 3/21/96.
Element 8: With submittal of SSAR Rev 9 ch 18 and WCAP-14690, the MMIS portion of this item is closed. Need to determine what is required to close ERG portion.
Resolved - Per DCP/NRC0589, this item will be closed with submittal of the at-power ERG's.

No ERG changes required. WCAP-14401 provides the programmatic procedure V&V. rkn 12/3

AP600 Open Item Tracking System Database: Executive Summary

Date: 2/14/97

Selection: [nrc st code]='Action W' And [DSER Section] like '18*' Sorted by Item #

Item No.	Branch	DSER Section/ Question	Type	Title/Description Detail Status	Resp Engineer	(W) Status	NRC Status	Letter No. /	Date
1369	NRR/HHFB	18.9.3-7	DSER-OI		MMIS/ERG	Closed	Action W	NSD-NRC-96-4805	

Westinghouse should provide information regarding the computer- based procedures. Westinghouse should describe the process by which human engineering issues associated with computer-based procedures will be resolved (e.g., concept testing, and other analyses).

Per 2/16/95 conference call between Jim Bongarra, John O'Hara & Kerch, Easter, Roth, Mumaw:

Action W -- NRC agreed to resolution path "in principle" and we need to send revision to SSAR section.

Action W
Per 2/16/95 conference call between Jim Bongarra, John O'Hara & Kerch, Easter, Roth, Mumaw: Action W -- NRC agreed to resolution path "in principle" and we need to send revision to SSAR section.
Westinghouse sent (via fax or fedex) to the NRC a draft revision to SSAR 18.9.8 on 8/18/95 which provided the response to this open item.
NRC to review draft of 18.9.8 ; see NRC fax of 2/20/96.
ACTION W: to respond to concerns of the fax; was discussed in conference call of 3/21/96.
Element 8. With submittal of SSAR Rev 9 ch 18 and WCAP-14690, the MMIS portion of this item is closed. Need to determine what is required to close ERG portion.
Resolved - Per DCP/NRC0589, this item will be closed with submittal of the at-power ERG's.

No ERG changes required. Refer to SSAR Sections 18.2 and 18.9 and Figure 18.2-3 for inputs to HSI design. rkn 12/3

Item No.	Branch	DSER Section/ Question	Type	Title/Description Detail Status	Resp Engineer	(W) Status	NRC Status	Letter No. /	Date
1370	NRR/HHFB	18.9.3-8	DSER-OI		MMIS/ERG	Closed	Action W	NSD-NRC-96-4805	

Westinghouse should provide information regarding procedure maintenance. Westinghouse should describe the administrative procedures that will ensure that hardcopy procedures remain current and consistent with the computer-based procedures.

Per 2/16/95 conference call between Jim Bongarra, John O'Hara & Kerch, Easter, Roth, Mumaw:

Action W -- NRC agreed to resolution path "in principle" and we need to send revision to SSAR section.

Action W
Per 2/16/95 conference call between Jim Bongarra, John O'Hara & Kerch, Easter, Roth, Mumaw: Action W -- NRC agreed to resolution path "in principle" and we need to send revision to SSAR section.
Westinghouse sent (via fax or fedex) to the NRC a draft revision to SSAR 18.9.8 on 8/18/95 which provided the response to this open item.
NRC to review draft of 18.9.8; see NRC fax of 2/20/96.
ACTION W: to respond to concerns of the fax; was discussed in conference call of 3/21/96.
Element 8. With submittal of SSAR Rev 9 ch 18 and WCAP-14690, the MMIS portion of this item is closed. Need to determine what is required to close ERG portion.
Resolved - Per DCP/NRC0589, this item will be closed with submittal of th at-power ERG's.

No ERG changes required. SSAR 13.5 notes procedure development , including administrative procedures, are COL applicant responsibility. See also SSAR 18.9 which references WCAP 14690. rkn 12/3

Selection: [nrc st code]='Action W' And [DSER Section] like '18*' Sorted by Item #

Item No.	Branch	DSER Section/ Question	Type	Title/Description Detail Status	Resp Engineer	(W) Status	NRC Status	Letter No. /	Date
1371	NRR/HHFB	18.9.3-9	DSER-OI		MMIS/ERG	Closed	Action W	NSD-NRC-96-4805	

Westinghouse should provide information regarding procedure use. Westinghouse should describe provisions for access to, and use of, hardcopy procedures, as backups either in the control room or at locations outside the control room. Westinghouse should also describe how disruption of ongoing activity by automatically accessed procedures will be minimized.

Per 2/16/95 conference call between Jim Bongarra, John O'Hara & Kerch, Easter, Roth, Mumaw:

Action W -- NRC agreed to resolution path "in principle" and we need to send revision to SSAR section.

Action W
Per 2/16/95 conference call between Jim Bongarra, John O'Hara & Kerch, Easter, Roth, Mumaw: Action W -- NRC agreed to resolution path "in principle" and we need to send revision to SSAR section.
Westinghouse sent (via fax or fedex) to the NRC a draft revision to SSAR 18.9.8 on 8/18/95 which provided the response to this open item.
NRC to review draft of 18.9.8 ; see NRC fax of 2/20/96.
ACTION W: to respond to concerns of the fax; was discussed in conference call of 3/21/96.
Element 8. With submittal of SSAR Rev 9 ch 18 and WCAP-14690, the MMIS portion of this item is closed. Need to determine what is required to close ERG portion.
Resolved - Per DCP/NRC0589, this item will be closed with submittal of th at-power ERG's.

No ERG changes required. SSAR 13.5 notes procedure development , including administrative procedures, are COL applicant responsibility. See also SSAR 18.9 which references WCAP 14690. rkn 12/3

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Item No.	Branch	DSER Section/ Question	Type	Title/Description Detail Status	Resp Engineer	(W) Status	NRC Status	Letter No. /	Date
1392	NRR/HHFB	18.11.3.4-1	DSER-OI		MMIS	Closed	Action W	NTD-NRC-97-4961	

Westinghouse should commit to developing a methodology for integrated system validation and related criteria. Westinghouse should describe the tools to be used in evaluating dynamic task performance in the V&V methodology. Westinghouse should describe how the V&V methodology will address the objectives listed as part of this criterion. Westinghouse should describe how the testing of critical human actions will be addressed in the V&V methodology. Westinghouse should describe how the V&V methodology will address the categories identified in Appendix A to RG 1.33 regarding procedure-related activities. Westinghouse should describe how the V&V methodology will evaluate performance under a range of operational conditions and upsets, and provide additional information about the Evaluation 17 test scenarios. Westinghouse should describe how the validation scenarios will be made realistic as part of the V&V methodology. Westinghouse should describe how the V&V methodology will address performance measures to test the achievement of all objectives, design goals, and performance requirements.

Per 2/16/95 conference call between Jim Bongarra, John O'Hara & Kerch, Easter, Roth, Mumaw:

Action W -- NRC agreed to resolution path "in principle" and we need to issue document & SSAR revision.

Action N -- NRC to provide clarification on which procedures per RG 1.33 should be covered by V&V.

Meeting of 3/10/95:

Clarification provided (in writing) by NRC to Westinghouse (Emilie). Brief discussion followed. Westinghouse to issue SSAR revision and document.

Resolved

4/13/95 - Fax of the "Programmatic Level Description of the AP600 Human Factors Verification and Validation Plan" was sent to J.Bongarra and J.O'Hara. A mapping of each element 10 open item to its response/answer was provided. Action N: Review the document and determine whether the element 10 open items 18.11's are adequately addressed.

Resolved: 5/17 phoncon with Jim Bongarra; Jim considers all the element 10 V&V open items resolved. Need to submit the revised 18.8.2.3 of chapter 18 of the SSAR as part of the formal SSAR revision.

Closed - The Human Factors Verification and Validation is addressed in revised SSAR Section 18.11, submitted in Rev. 9, 7/31/96.
Action W - See NRC letter dated 12/9/96.
Completed with submittal of WCAP-14401 Revision 2. rkn 1/28/97

8/12

AP600 Open Item Tracking System Database: Executive Summary

Date: 2/14/97

Selection: [nrc st code]='Action W' And [DSER Section] like '18*' Sorted by Item #

Item No	Branch	DSER Section/ Question	Type	Title/Description Detail Status	Resp Engineer	(W) Status	NRC Status	Letter No. /	Date
1395	NRR/HHFB	18.12.3-1	DSER-OI		MMIS	Action W	Action W	NSD-NRC-4875	

Westinghouse should submit an acceptable minimum inventory of fixed-position controls, displays, and alarms for transient mitigation. Westinghouse should describe how the task analysis will define a minimum inventoryof alarms, displays and controls necessary to perform crew tasks. Westinghouse should describe the technical basis for the minimum inventory. Westinghouse should describe how an inventory will be identified of fixed-position controls, displays, and alarms necessary to permit execution of the operator tasks to place and maintain the plant in a safe-shutdown condition. Westinghouse should describe how additional detailed characteristics of these controls, displays, and alarms (e.g., ranges, scales, physical dimensions, and actual information presentation) will be identified, defined, and implemented.

Per 2/16/95 conference call between Jim Bongarra, John O'Hara & Kerch, Easter, Roth, Mumaw:
Westinghouse to include this on the March meeting agenda.
Action N -- Give Westinghouse feedback on our proposed resolution (proposed during 2/2/95 meeting in Rockville).

3/8/95 meeting:
NRC requested Westinghouse consider that the detailed list remain completely in Tier 2. Tier 1 would include the process to select the final inventory.
ACTION W: If the inventory list is provided in chapter 7, then make the cross reference strong from chapter 18. Also, the list should include the process / criteria that was used to generate the list. Westinghouse position is that this list is an expansion of the RG 1.97 criteria and philosophy to address controls and displays. Should a Tier 1 list be required we will pursue use of critieria presented at the Feb 2 meeting versus the NRC criteria used on evolutionary plants. Also prepare a draft Tier 1 list. Need to take a stab at defining acceptable ITAAC and supporting SSAR information as to how the final inventory will be defined (Use PRA, EOPs, ERGs, FBTA). Caution from A. Sterdis -- There will be a strong push to be specific in defining these design ITAAC.

ACTION N: NRC staff to prepare a position paper for NRC senior management, proposing Tier 1 include the process / critieria. Goal is to produce the paper to support the next scheduled Senior management meeting of April 4.

2/2/95: Presentation of above made in Rockville, NRC staff to discuss and provide feedback.
2/9/95: Discussed during NRC/Westinghouse senior management meeting as one of the top 50 open items. Action N -- to provide feedback on Westinghouse proposal for resolution.
2/27/95: Conference call with NRC (J. Bongarra, G. Galletti, J. O'Hara, J.Easter, A.Sterdis & S.Kerch). 1. Agreed to following definition of "fixed position" -- unique location in the control room/control panel for alarms, displays, controls where present information from the minimum inventory; continuously available not continously displayed; doesn't have to be class 1E; always displayed at the same location, dedicated location where the operator can retrieve information that is part of the minimum inventory. 2. Scope of min. inv. -- failed to reach a mutual understanding on this; NRC stated that scope includes those controls and indications needed to execute the ERG high level operator actions including nonsafety system actions; disagreed on this. 3. Use of FBTA & ERG development task analysis I & C list. 4. When completed where does this go tier 1 or tier 2? Agreed to discuss at 3/8 meeting.
3/8/95 meeting: NRC requested Westinghouse consider that the detailed list remain completely in Tier 2. Tier 1 would include the process to select the final inventory. ACTION W: If the inventory list is provided in chapter 7, then make the cross reference strong from chapter 18. Also, the list should include the process / criteria that was used to generate the list. Westinghouse position is that this list is an expansion of the RG 1.97 critieria and philosophy to address controls and displays. Should a Tier 1 list be required we will pursue use of critieria presented at the Feb 2 meeting versus the NRC criteria used on evolutionary plants. Also prepare a draft Tier 1 list. Need to take a stab at defining acceptable ITAAC and supporting SSAR information as to how the final inventory will be defined (Use PRA, EOPs, ERGs, FBTA).
NRC staff prepared a position paper for NRC senior management, proposing Tier 1 include the process / critieria. Goal is to produce the paper to support the next scheduled Senior management meeting of April 4.
4/19/95 - Fax sent to J.Bongarra and G.Galletti of NRC that provided a preliminary (draft) description of how the total inventory list was developed and where in the tier 2 (SSAR) document it was found. A description of how the minimum inventory would be selected from the total inventory list (the criteria to be used) was also provided. This would be placed in the Tier 1 document. A very preliminary draft of a minimum inventory list, using this criteria, was provided for the NRC's information and use as backup to their position paper. The NRC (G.Galletti) has submitted the position paper for NRC management review.
NRC to Determine whether the position paper is acceptable and the proposed Westinghouse appraoch is acceptable.
Action W - see NRC response sent 8/21/95
Resolved - The minimum inventory is addressed in revised SSAR Section 18.12, submitted in Rev. 9, 7/31/96. An ITAAC will be prepared which will include the list of minimum inventory.
Closed - ITAAC submitted by NSD-NRC-96-4875 of 11/7/96.
NRC action to review the minimum inventory issue. See NSD-NRC-96-4874. rkn 12/19
Comments on Minimum Inventory rec'd by NRC letter 1/17/97.

Item No.	Branch	DSER Section/ Question	Type	Title/Description Detail Status	Resp Engineer	(W) Status	NRC Status	Letter No. /	Date
				A telecon is scheduled for Feb 5 to resolve these actions. rkn 1/30/97					
1396	NRR/HHFB	18.13.3-1	DSER-OI		ITAAC/MMIS	Closed	Action W	NSD-NRC-4875	
				Westinghouse should develop the ITAAC/DAC for certain elements of the HFE PRM. In each of the following areas, Westinghouse should provide ITAAC/DAC addressing a commitment to complete the implementation plan and provide the results to the staff for review: * Element 3 - Functional Requirements Analysis/Allocation * Element 4 - Task Analysis * Element 5 - Staffing * Element 6 - Human Reliability Analysis * Element 7 - Human-System Interface Design * Element 8 - Procedure Development * Element 9 - Training Program Development Westinghouse should also provide ITAAC/DAC addressing a V&V commitment to (a) develop a detailed implementation plan, and (b) complete the implementation plan and provide the results to the staff for review. ITAAC identifying the minimum inventory must also be developed.					
				Action W: Westinghouse will discuss with the NRC HHFB our approach to ITAACS/ Tier 1 document for chapter 18. Closed - ITAAC submitted by NSD-NRC-96-4875 of 11/7/96.					
1397	NRR/HHFB	18.13.3-2	DSER-OI		ITAAC/MMIS	Action W	Action W	NSD-NRC-4875	
10 of 12				Westinghouse should provide the specified level of detail for the DCD, ITAAC, and DAC. Westinghouse should: 1. Provide a complete set of ITAAC/DAC describing the (a) design commitments, (b) inspections, test, and analyses; and (c) acceptance criteria for Element 3, "Functional Requirements Analysis and Allocation"; Element 4, "Task Analysis"; Element 5, "Staffing"; Element 6, "Human Reliability Analysis" Element 7, "Human-System Interace Design"; Element 8, "Procedure Development"; and Element 9, "Training Program Development 2. Provide a complete set of ITAAC/DAC for all V&V activities, including HSI task support verification, human factors issue resolution verification, and final plant HFE/SHI design verification 3. Resolve the staff's concern regarding the use of HFE guidelines for verification 4. Provide ITAAC/DAC for the minimum inventory					
				Action W: Westinghouse will discuss with the NRC HHFB our approach to ITAACS/ Tier 1 document for chapter 18. Closed - ITAAC submitted by NSD-NRC-96-4875 of 11/7/96. Per NSD-NRC-96-4874, W action to respond to NRC comments on ITAACs. rkn 12/19					

Item No.	Branch	DSER Section/ Question	Type	Title/Description Detail Status	Resp Engineer	(W) Status	NRC Status	Letter No. /	Date
2043	NRR/HHFB	18	DSER-OI50		ERG	Closed	Action W	NSD-NRC-96-4805	

47. Content of ERGs
Westinghouse, in response to a staff request for an AP600 ERG submittal, stated that the low-pressure reference plant's ERGs in combination with a design differences report, identification of high level operator action strategies, and the AP600 system/event matrices are sufficient for design certification. The staff does not agree with this Westinghouse position because the passive safety system philosophy differs significantly from current plants. This was addressed in the August 1994 position paper to Westinghouse as well as in the draft safety evaluation report (DSER). (DSER open items 5.4.7.6-1, 15.3.4-3, 20.4-2, 20.4-21) Westinghouse has indicated that they will submit the ERGs by May 1995. Westinghouse met the staff on February 2 and presented and discussed the ERG development process.

Action W
DISCUSSED AT 2/9/95 SENIOR MANAGEMENT MEETING: 47. Staff to confirm that technical agreement has been reached and implementation is the only issue. May be removed at next meeting. Westinghouse believes item 47 is resolved and waiting for staff feedback.
Action W -- NRC agreed to resolution path "in principle" and we need to send the ERGs and background documents to NRC. These will be sent via a phased approach with phase 1 ERGs to be sent 5/31/95.
Status update provided by phone (D. Jackson 8/21):
Action W is to complete analytical basis for AE-1, AES-1.2, AE-2, and supporting documentation for shutdown ERG. (need to submit background information on S/D ERGs.
Action N - Review at-power , Low power/shutdown ERGs and background for at-power and low power ERGs.
Resolved - Per DCP/NRC0589, this item will be closed with submittal of th at-power ERG's scheduled for 12/31. rkn 12/3.
Closed with ERG Rev 2 submittal Jan 10. rkn 1/15/97.

11 of 12

Item No.	Branch	DSER Section/ Question	Type	Title/Description Detail Status	Resp Engineer	(W) Status	NRC Status	Letter No. /	Date
2044	NRR/HHFB	18.	DSER-OI50		ITAAC/MMIS	Dropped	Action W	NSD-NRC-4875	

48. Minimum Inventory of Controls and Displays
Westinghouse has not submitted a minimum inventory of controls and displays for the AP600 (See DSER open item 18.12.3-1). This issue was discussed with the staff at the 2/2/95 meeting in Rockville. A resolution to this open item was proposed and the staff stated that they needed to discuss it among themselves. (For closure, this open item may first require a completed set of AP600 ERGs.)

See 18.12.3-1 for further status.

Action N: To review rev. 9 of ch. 18, specifically section 18.12.
8/96: SSAR Rev. 9, which included a revision to Ch. 18, was submitted to NRC. This included section 18.12 on "minimum inventory".

2/2/95: Presentation of above made in Rockville, NRC staff to discuss and provide feedback.
2/9/95: Discussed during NRC/Westinghouse senior management meeting as one of the top 50 open items. Action N -- to provide feedback on Westinghouse proposal for resolution.

2/27/95: Conference call with NRC (J. Bongarra, G. Galletti, J. O'Hara, J.Easter, A.Sterdis & S.Kerch). 1. Agreed to following definition of "fixed position" -- unique location in the control room/control panel for alarms, displays, controls where present information from the minimum inventory; continuously available not continously displayed; doesn't have to be class 1E; always displayed at the same location; dedicated location where the operator can retrieve information that is part of the minimum inventory. 2. Scope of min. inv. -- failed to reach a mutual understanding on this; NRC stated that scope includes those controls and indications needed to execute the ERG high level operator actions including nonsafety system actions; disagreed on this 3. Use of FBTA & ERG development task analysis I & C list. 4. When completed where does this go tier 1 or tier 2? Agreed to discuss at 3/8 meeting.

3/8/95 meeting: NRC requested Westinghouse consider that the detailed list remain completely in Tier 2. Tier 1 would include the process to select the final inventory. ACTION W: If the inventory list is provided in chapter 7, then make the cross reference strong from chapter 18. Also, the list should include the process / criteria that was used to generate the list. Westinghouse position is that this list is an expansion of the RG 1.97 critieria and philosophy to address controls and displays. Should a Tier 1 list be required we will pursue use of critieria presented at the Feb 2 meeting versus the NRC criteria used on evolutionary plants. Also prepare a draft Tier 1 list. Need to take a stab at defining acceptable ITAAC and supporting SSAR information as to how the final inventory will be defined (Use PRA, EOPs, ERGs, FBTA). Caution from A. Sterdis -- There will be a strong push to be specific in defining these design ITAAC.

ACTION N: NRC staff to prepare a position paper for NRC senior management, proposing Tier 1 include the process / criteria. Goal is to produce the paper to support the next scheduled Senior management meeting of April 4.

4/19/95 - Fax sent to J.Bongarra and G.Galletti of NRC that provided a preliminary (draft) description of how the total inventory list was developed and where in the tier 2 (SSAR) document it was found. A description of how the minimum inventory would be selected from the total inventory list (the criteria to be used) was also provided. This would be placed in the Tier 1 document. A very preliminary draft of a minimum inventory list, using this criteria, was provided for the NRC's information and use as backup to their position paper. The NRC (G.Galletti) has submitted the position paper for NRC management review.

5/2/95 Status: Discussed during March 8-9, 1995 mtg. Proposed approach under NRC management review.

Action N: Determine whether the position paper and Westinghouse approach is acceptable. May require a SSAR revision to chapter 18.
Action W - see NRC response sent 8/21/95
Closed - ITAAC submitted by NSD-NRC-96-4875 of 11/7/96.
Action N - See letter dated 12/9/96.
This item is dropped since 1395 is a repeat (action is being followed under item 1395). rkn 1/30/97

21 of 21

NRC FORM 366 (12-93)

U.S. NUCLEAR REGULATORY COMMISSION

TELECOPIER TRANSMITTAL

DATE: 2/14/97

TIME: 12:00

WARNING: Most facsimile machines produce copies on thermal paper. The image produced is highly unstable and will deteriorate significantly in a few years. Reproduce copies onto plain paper prior to filing as a record.

TO

NAME: Jim Winters

TELEPHONE: 412-374-5290

NAME AND LOCATION OF COMPANY (If other than NRC): W AP600

TELECOPY NUMBER: 412-374-4887

VERIFICATION NUMBER:

FROM

NAME: Bill Huffman

TELEPHONE: 301-415-1141

MAIL STOP: 10 D22

TELECOPY DATA

NUMBER OF PAGES

THIS PAGE + 3 PAGES = 4 TOTAL

PRIORITY

IMMEDIATE

OTHER (Specify)

SPECIAL INSTRUCTIONS

Jim,

Per our discussion

Bill Huffman

PROBLEMS

If any problems occur or if you do not receive all the pages, call:

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After telecopy has been sent, process the original as requested below. (If none are checked, the original will be discarded.)

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PROCESSED BY (INITIALS)

VERIFIED BY (INITIALS)

NRC FORM 366 (12-93)

PRINTED ON RECYCLED PAPER

February 14, 1997

TO: Jim Winters

FROM: Bill Huffman

SUBJ: ELECTRICAL FSER

Narinder Trehan has asked me to pass a request onto you. He has a problem with not specifying any restrictions on mixing 480 volt power cables and control cables in the same trays. I have attached a fax from Tom Hayes which noted that there should be some restrictions but nothing specific was provided.

Narinder has marked up the what kind of restrictions he would like to see which I have attached. Could you pass this onto Hayes (or whoever) and have them take a look. After you have thought about it, give me a call and we'll set up a telecon.

Please stick this into your informal fax documentation.

----------------------Bill

Westinghouse

FAX COVER SHEET

RECIPIENT INFORMATION		SENDER INFORMATION	
DATE:	July 24, 1996	NAME:	Tom Hayes
TO:	Narendra Trehan	LOCATION:	Energy Center E322B
PHONE:	301-415-2777	PHONE:	(412) 374-4420
COMPANY:	NRC	WIN:	284-4420
LOCATION:		FAX:	(412) 374-5535

Cover+Pages = 1 + 7

Comments: Per our telephone conversation of 7/22, attached are two documents generated during the URD change process to allow the combining of low-voltage power and high-level signal and control cables in the same raceway. Imp

The first document (5 pages) is our original request to revise the URD. It contains a review of relevant codes and standards and an overview of why we believe the change to be beneficial.

The second document (2 pages) responds to questions generated by the utilities. It tends to jump from topic to topic to follow the questions, but contains some good information about our design.

Note that there are restrictions on trays containing a combination of low-voltage power and high-level signal, so combined trays are only used where the trays are relatively empty. Imp

If you need further information, please give me a call.

Regards,
Tom Hayes

Phone Number of Receiving Equipment: 301-415-2444

7/24/96:1007

Where, (a) it is serving an ~~intermittent load or is a #12 AWG cable~~ operating at less than 50 percent of the ICEA rating, and (b) the fault current is insufficient to heat insulation to the flash point.

involve exclusively limited energy content cables (instrumentation and control), these minimum distances are reduced to 3 inches and 1 inch respectively.

- Within panels and control switchboards, the minimum horizontal separation between components or cables of different separation groups (both field-routed and vendor-supplied internal wiring) is 1 inch, and the minimum vertical separation distance is 6 inches.

The exceptions to the guidance in Regulatory Guide 1.75 are based on test results used to support exceptions to the separation guidance for operating nuclear power plants. A summary of test results from ten electrical separation test programs is documented in Reference 13. These test programs support the AP600 exceptions.

Non-Class 1E circuits are electrically isolated from Class 1E circuits, and Class 1E circuits from different separation groups are electrically isolated by isolation devices, shielding and wiring techniques, physical separation (in accordance with Regulatory Guide 1.75 for circuits in raceways), or an appropriate combination thereof.

When isolation devices are used to isolate Class 1E circuits from non-Class 1E circuits, the circuits within or from the Class 1E equipment or devices are identified as Class 1E and are treated as such. Beyond the isolation device(s) these circuits are identified as non-Class 1E and are separated from Class 1E circuits in accordance with the above separation criteria.

Power and control cables are installed in conduits or ventilated bottom trays (ladder-type). Solid tray covers are used in outdoor locations and indoors where trays run in areas where falling debris is a problem. Instrumentation cables are routed in conduit or solid bottom cable tray with solid tray covers as required. The cables are derated for specific application in the location where they are installed as stated in subsection 8.3.1.3.3. The environmental design of electrical equipment including Class 1E cables under normal and abnormal operating conditions is discussed in Section 3.11.

Separate trays are provided for each voltage service level: 4.16 kV, low voltage power (480 Vac, 120 Vac, 125 Vdc), high-level signal and control (120 Vac, 125 Vdc), and low level signal (instrumentation). 480 Vac power cables may be mixed with 120 Vac/125 Vdc signal and control cables. Vertically stacked trays are arranged from top to bottom as stated in subsection 8.3.1.3.4. In general, a minimum of 12 inches vertical spacing is maintained between trays of different service levels within the stack.

The electrical penetrations are in accordance with IEEE 317 (Reference 2). Class 1E and non-Class 1E electrical penetration assemblies are maintained in a separate nozzle. The physical separation of the Class 1E electrical penetration assemblies are in accordance with Regulatory Guide 1.75. The containment building penetrations are described in subsection 8.3.1.1.5.

Raceways installed in seismic Category I structures have seismically designed supports or are shown not to affect safety-related equipment should they fail. Trays are not attached rigidly

COVER SHEET

FAX

RECEIVED
Brian A. Mc Intyre

To: Bill Huffman (NRC)
cc: B. McIntyre (Informal NRC File), Bob Osterrieder, Larry Hochreiter, File 7.6
Subject: NOTRUMP and WC/T
Date: February 7, 1997
Pages: Four, including this cover sheet.

COMMENTS:

Bill,

Attached are the following which I would like to discuss Monday's telecon.

Attachment I - Status of NOTRUMP OIs

Attachment II - Proposed Draft of "Roadmap"

Please give a copy to Ralph.. Thanks.

Earl

From the desk of...

Earl H. Novendstern
Manager, Advanced and VVER Plant Safety Analysis
Westinghouse
PO Box 355
Pittsburgh, PA 15235

(412) 374 -4790
Fax: (412) 374-5744

Attachment 1

Item #	Type	Question	Action
3230	DSER-CN	21.6.2.4-4	RESPONSE TO BE WRITTEN: Verify heat link meth. for trans. boiling not used in AP600 NOTRMP
2615	RAI-OI	440.342	RESPONSE TO BE WRITTEN: Ident. which info from RAI responses included in documentation
3152	DSER-OI	21.6.2.4-11	RESPONSE TO BE WRITTEN: Demo acceptability of Zuber CHF correlation for AP600
3156	DSER-OI	21.6.2.5-2	DONE - V&V Report discusses inabil. to characterize CMT thermal strat.
3162	DSER-OI	21.6.2.7-2	RESPONSE TO BE WRITTEN: Address effects of nonconden. gases on PRHR heat transfer
3161	DSER-OI	21.6.2.7-1	DONE - Put PRHR comparison plots in V&V Report (done for SPES, OSU data unavailable)
3151	DSER-OI	21.6.2.4-10	DONE - Birthing not used and removed from report
3153	DSER-OI	21.6.2.4-12	DONE - Acceptability of unchoking model for ADS can be evaluated based on test results in V&V
3140	DSER-OI	21.6.2.2-1	RESPONSE TO BE WRITTEN: Ident. which info from RAI responses included in documentation
3141	DSER-OI	21.6.2.2-2	DONE - Submitted final V&V Report
3142	DSER-OI	21.6.2.4-1	RESPONSE TO BE WRITTEN: Explain how user won't override approved options for flow part.
3143	DSER-OI	21.6.2.4-2	DONE - Benchmarks/assessments included in V&V Report
3145	DSER-OI	21.6.2.4-4	RESPONSE TO BE WRITTEN: Explain how user won't override approved options
3146	DSER-OI	21.6.2.4-5	DONE - Completed all benchmarks/assessments
3154	DSER-OI	21.6.2.4-13	DONE - Stacking logic assessment benchmark in section 3 does this.
3155	DSER-OI	21.6.2.5-1	DONE - Compared measured vs predicted ADS integrated flow in V&V Report
3160	DSER-OI	21.6.2.6-3	DONE - Submitted reanalyzed integral tests
3232	DSER-CN	21.6.2.7-1	DONE - SG tube draining is discussed for OSU and SPES simulations
2608	RAI-OI	440.335	RESPONSE TO BE WRITTEN: Justify constant friction
2609	RAI-OI	440.336	RESPONSE TO BE WRITTEN: Is momentum flux off? / justify if it is
2610	RAI-OI	440.337	RESPONSE TO BE WRITTEN: Justify use of Macbeth CHF at low pressure/low flow
2611	RAI-OI	440.338	RESPONSE TO BE WRITTEN: Demo pump model predicts coastdown / justify 2-phase curves

Attachment 1 Continued

2612	RAI-OI	440.339	RESPONSE TO BE WRITTEN: Need for timestep studies / nodalization studies
2820	RAI-OI	440.466	DONE - Benchmarks in V&V Report
2821	RAI-OI	440.467	DONE - Completed in original RAI response
2823	RAI-OI	440.469	DONE - Benchmarks in V&V Report
2824	RAI-OI	440.470	DONE - Horiz. strat. flow model removed from report
2825	RAI-OI	440.471	DONE - Completed in original RAI response
2826	RAI-OI	440.472	DONE - Completed in original RAI response
2827	RAI-OI	440.473	DONE - Completed in original RAI response
2832	RAI-OI	440.478	DONE - Birthing not used and removed from report
2833	RAI-OI	440.479	DONE - Completed in original RAI response
2834	RAI-OI	440.480	RESPONSE TO BE WRITTEN: Deter. which Zuber CHF correlation used, if unmod., then done
2835	RAI-OI	440.481	DONE - completed in original RAI response
2836	RAI-OI	440.482	DONE - orig. response plus described final unchoking model in V&V
2922	RAI-OI	440.441	RESPONSE TO BE WRITTEN: Provide CMT wall temp comparisons and discussion
2923	RAI-OI	440.442	RESPONSE TO BE WRITTEN: Justify not modeling wall heat structures in CMT test sims.
2924	RAI-OI	440.443	RESPONSE TO BE WRITTEN: Justify nodalization of CMT reservior
2925	RAI-OI	440.444	RESPONSE TO BE WRITTEN: Requested CMT timestep study
2927	RAI-OI	440.446	RESPONSE TO BE WRITTEN: Describe differing uncertainties in CMT flow data
	RAI-OI	440.489	RESPONSE TO BE WRITTEN: PRHR noding study to be documented
	RAI-OI	440.480	REVISE SECTION 2.15 OF V&V REPORT: Add info from RAI response
	RAI-OI	440.481	REVISE SECTION 2.16 OF V&V REPORT: Add info from RAI response

ATTACHMENT II - Draft of NOTRUMP Roadmap

RAI #	Reference	RAI Description
xxx.yyy	WCAP-XXXXX, Section y.z	Brief Description of RAI
xxx.zzz	W. Letter #	

FAX to DINO SCALETTI

February 13, 1997

CC: Sharon or Dino, please make copies for: Joe Sebrosky
Ted Quay

Don Lindgren
Cindy Haag
Richard Orr
Bob Lutz
Ed Cummins
Bob Vijuk
Brian McIntyre

OPEN ITEMS FOR PRA CHAPTER 42

This is a background package for the remaining open items for PRA Chapter 42. PRA Chapter 42 is of interest because by our joint NRC/W schedule, the FSER for this section should be turned into Projects by the middle of March. There are 8 Open Items with NRC Status of Action W. Westinghouse believes that five (5) of these items (3264 through 3268) were resolved by responses provided by NSD-NRC-96-4904 of December 9, 1996. Westinghouse believes the other three (3) items (3269, 3270 and 3271) were resolved by responses provided by NSD-NRC-97-4981 of February 11, 1997. Currently, our records show no additional outstanding Westinghouse actions required for PRA Chapter 42. We request that NRC provide a definitive action for Westinghouse associated with the items above or provide direction to change the status of these items. We recommend "Action N". Thank you.

Jim

Jim Winters
412-374-5290

AP600 Open Item Tracking System Database: Executive Summary

Date: 2/13/97

Selection: [nrc st code]='Action W' And [resp eng] like 'ch42*' Sorted by Item #

Item No.	Branch	DSER Section/ Question	Type	Title/Description Detail Status	Resp Engineer	(W) Status	NRC Status	Letter No. /	Date
3264	NRR/ECGB	19.2	RAI-OI		CH42/Orr/Lutz	Closed	Action W	NSD-NRC-96-4904	

220.95 In PRA Chapter 42, dated December 1, 1995, a coefficient of variance (COV) of 0.06 for random material uncertainty for SA537, Class 2 is used. This does not seem to be appropriate. The staff could not find where your Reference 42-2 recommends a COV between 0.06 and 0.08 for random material uncertainty. On page 77 in Reference 42-2, the material yield strength mean and standard deviation were taken as 1.1 and 0.11 times the specified y for the uncertainty analysis of the five containments.

Based on "Development of a Probability Based Load Criterion for American National Standard A58," National Bureau of Standards Special Publication 577, US Government Printing Office, Washington, 1980, the use of 1.1 and 0.11 for the median and the COV, respectively, for material property is more appropriate. Figure 1 shows the probability of failure differences between the random material uncertainty COVs of 0.06 and 0.11.

RAI response provided in letter NSD-NRC-96-4904 of 12/9/96.

Item No.	Branch	DSER Section/ Question	Type	Title/Description Detail Status	Resp Engineer	(W) Status	NRC Status	Letter No. /	Date
3265	NRR/ECGB	19.2	RAI-OI		CH42/Orr/Lutz	Closed	Action W	NSD-NRC-96-4904	

220.96 In PRA Subsection 42.4.1, a COV of 0.1 for modeling error for containment cylindrical shell is used. This does not seem to be appropriate. The COV of 0.1 seems to come from the average of 0.12 (limit pressure calculations) and 0.08 (axisymmetric finite element analysis) in Reference 42-1. However, the COV of 0.08 is not directly applicable to AP600 since Reference 42-1 uses ANSYS with STIF 82 elements for six closed-end smooth cylinders which are differe.: from that of the AP600. The COV of 0.12 is obtained after calibrating with respect to the finite element method in Reference 42-1. In this calibration, Reference 42-1 treats the von Mises yield criterion as a constant (15 percent increase). The use of COV of 0.12 for the constant von Mises yield criterion is acceptable. However, based on test data, the von Mises yield criterion should be treated as a random variable. For the complete modeling error COV determination, provide the median and its COV values with the legitimate distribution information for the von Mises yield criterion.

RAI response provided in letter NSD-NRC-96-4904 of 12/9/96.

Item No.	Branch	DSER Section/ Question	Type	Title/Description Detail Status	Resp Engineer	(W) Status	NRC Status	Letter No. /	Date
3266	NRR/ECGB	19.2	RAI-OI		CH42/Orr/Lutz	Closed	Action W	NSD-NRC-96-4904	

220.97 In letter NTD-NRC-96-4617, dated January 4, 1996, the response to RAI 2 (NRC letter dated September 14, 1995) stated that "Distribution is primarily influenced by imperfection. Measured imperfection must be less than the American Society of Mechanical Engineers (ASME) specified limit. Use of lognormal distribution is appropriate, similar to its use for material yield which must also exceed a minimum ASME specification." Is the lognormal distribution applicable here if it exceeds the ASME minimum specifications? The distribution should come from the existing test data and not be assumed. Clarify these statements.

RAI response provided in letter NSD-NRC 96-4904 of 12/9/96.

AP600 Open Item Tracking System Database: Executive Summary

Date: 2/13/97

Selection: [nrc st code]='Action W' And [resp eng] like 'ch42*' Sorted by Item #

Item No	Branch	DSER Section/ Question	Type	Title/Description Detail Status	Resp Engineer	(W) Status	NRC Status	Letter No. /	Date
3267	NRR/ECGB	19.2	RAI-OI		CH42/Orr/Lutz	Closed	Action W	NSD-NRC-96-4904	
				220.98 In PRA Subsection 42.4.3, a COV of 0.14 is used for equipment hatch modeling error. However, this modeling error calculation in Reference 42-2 is based on internal pressure cases. Provide the justification that this COV can be used for the external pressure case.					
				RAI response provided in letter NSD-NRC-96-4904.					
3268	NRR/ECGB	19.2	RAI-OI		CH42/Orr/Lutz	Closed	Action W	NSD-NRC-96-4904	
				220.99 In PRA Section 42.1, it is stated that "Failures of the mechanical penetration bellows, and leakage of the equipment hatches due to ovalization, do not occur prior to general yielding of the cylinder." This implies that after the yield pressure is reached, the bellows will fail and the equipment hatches start to leak without restriction. Therefore, the probability of failure for bellows and leakage through equipment hatches due to ovalization beyond yield pressure should be given in PRA.					
				RAI response provided in letter NSD-NRC-96-4904.					

Item No.	Branch	DSER Section/ Question	Type	Title/Description Detail Status	Resp Engineer	(W) Status	NRC Status	Letter No. /	Date
3269	NRR/ECGB	3.8.2	RAI-OI		CH42/Orr/Lutz	Closed	Action W	NSD-NRC-97-4981	

220.100 In SSAR Section 3.8.2.4.2.3, the factor of safety (FS) of 1.67 is used for equipment hatch covers ASME Service Level C limits.

Westinghouse estimated the critical buckling pressures for equipment hatches as 1.45 MPa (196 psig) for a 6.7 m (22 ft) diameter hatch and 1.21 MPa (161 psig) for a 4.9 m (16 ft) diameter hatch based on the classical buckling capacity of spherical shells subjected to external pressure and the capacity reduction factors specified in Baker et al., "Structural Analysis of Shells," pp. 253-254, McGraw-Hill, 1972, and in ASME Code Case N-284. The corresponding ASME Service Level C limits are 908 kPa (117 psig) and 763.2 kPa (96 psig) using the factor of safety (FS) of 1.67 as specified in Code Case N-284, respectively.

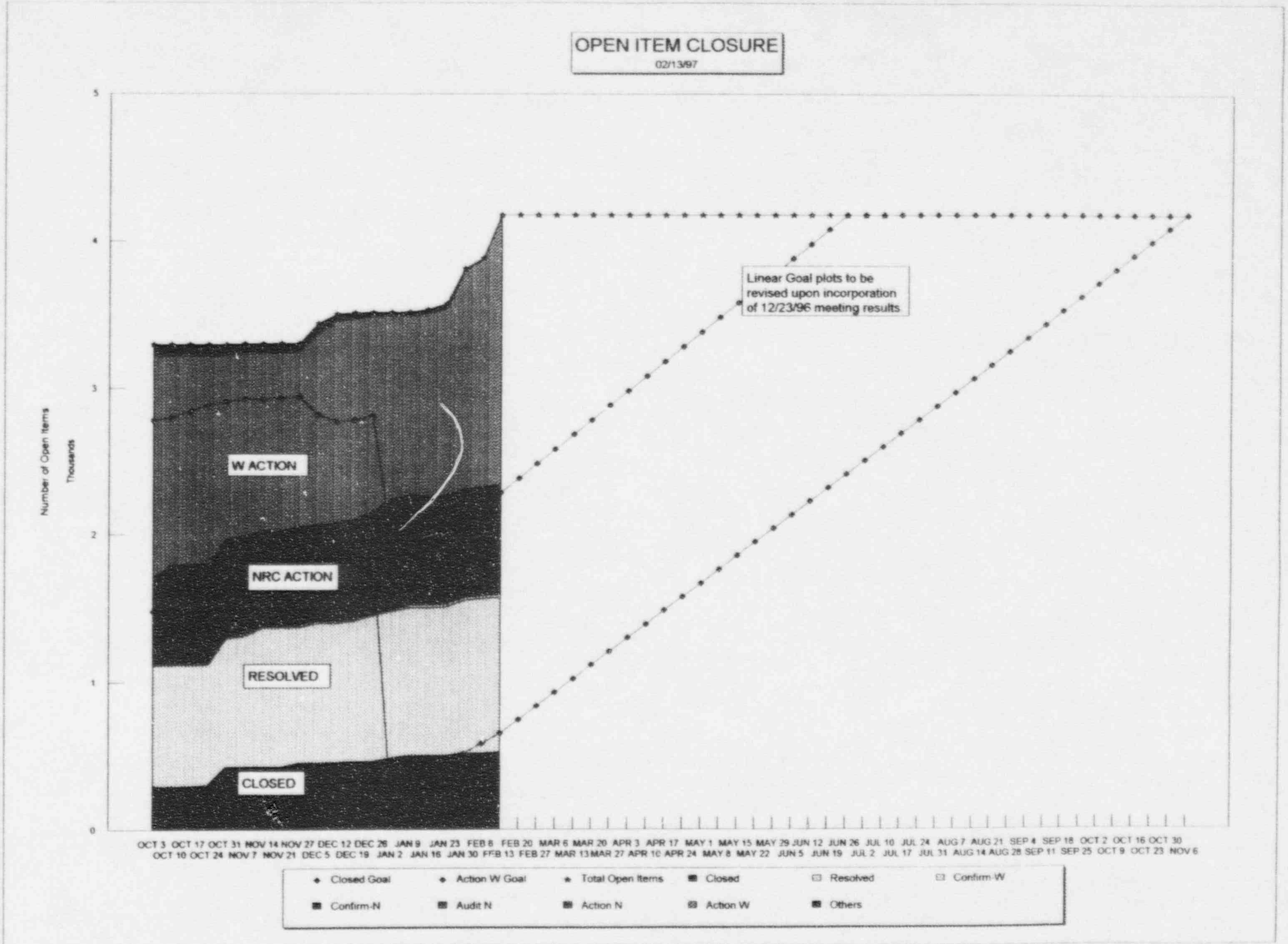
For the FS to be applied to the Service Level C pressure capacity, Westinghouse considered the equipment hatch cover buckling due to external pressure as the local buckling (FS = 1.67 from Code Case N-284). The hatch cover is a complete shell by itself with its own independent boundary and is subjected to pressure on its convex side due to the containment internal pressure. Therefore, the staff position is that the global buckling (FS = 2.5 from NE-3222) is the appropriate value. The ASME Service Level C pressure capacity is 763.2 kPa (96 psig) with FS of 1.67 and 545.4 kPa (64.4 psig) with an FS of 2.5.

Based on Code Case N-284, the local buckling is defined as the buckling of the shell plate between stiffeners. The flange of the cover can act as a stiffening element around the periphery of the spherical cap. However, the stiffening effect is limited to (Rt)½ or 35.3 cm (13.9 in) from the edge. The entire arc length from the center of the hatch cover to the flange is 255.3 cm (100.5 in). The remaining 218.4 cm (86 in) arc should be considered as unstiffened, therefore, the global buckling criteria should be applied to this unstiffened region. In the draft safety evaluation report (DSER), the staff noted that Westinghouse's assumption of local buckling for the equipment hatch cover under external pressure was not acceptable. The staff requested that Westinghouse increase the thickness or use stiffeners (e.g., ABB-CE System 80+ design) to meet the ASME Service Level C limits at the ambient temperature of 908 kPa (117 psig) for a 6.7 m (22 ft) diameter hatch and 763.2 kPa (96 psig) for a 4.9 m (16 ft) diameter hatch. This was Open Item 3.8.2.4-30.

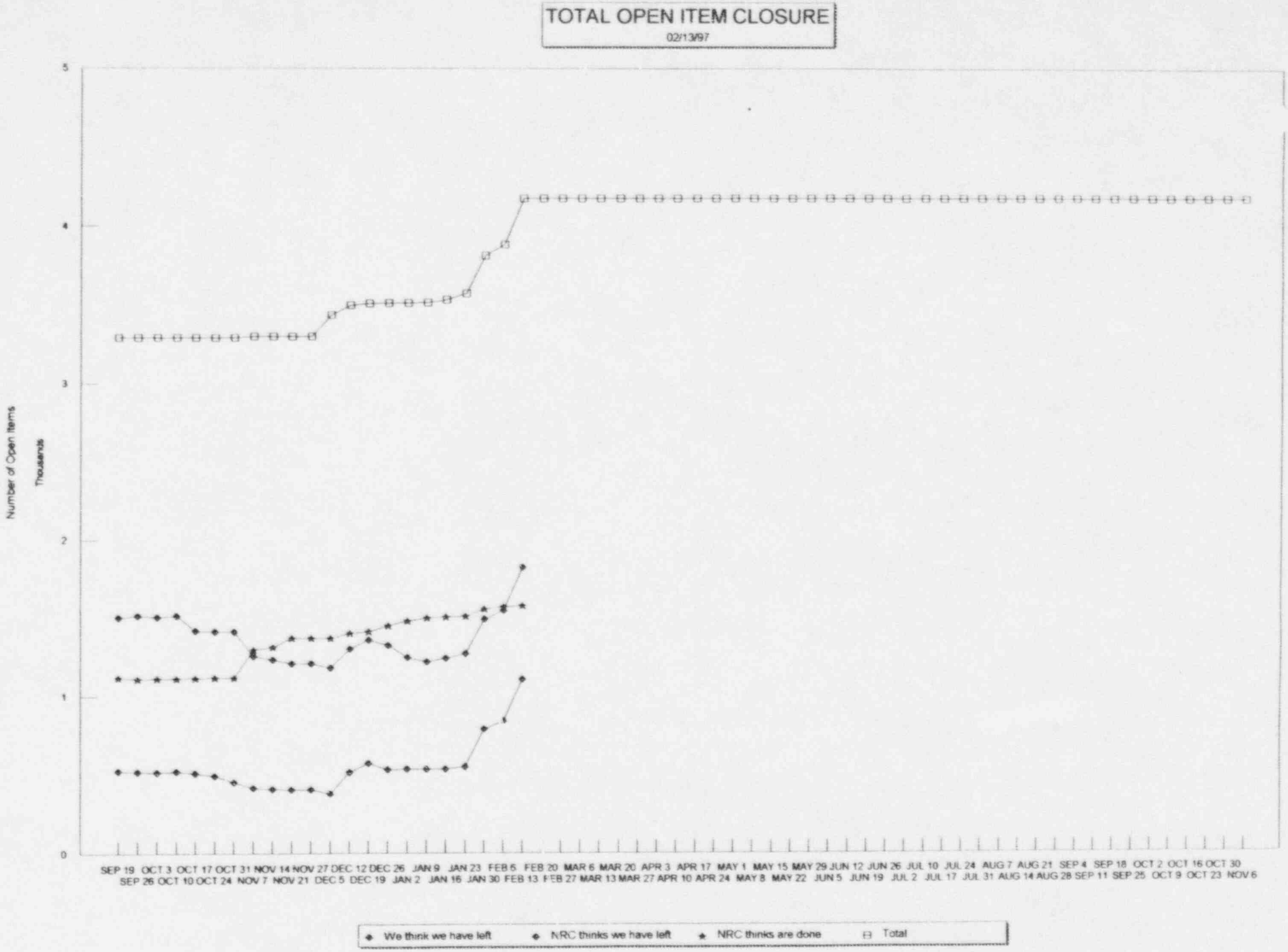
The staff performed independent analysis for the equipment hatch covers using the ALGOR computer code with fixed boundary conditions and no imperfection. Using ALGOR, the staff predicted the buckling pressure, Pbuckling, as 1.38 MPa (185.12 psig) and 1.57 MPa (212.96 psig) for 4.9 m (16 ft) and 6.7 m (22 ft) equipment hatch covers, respectively. In both cases, the buckling was predicted to occur near the top portion.

4/25

Item No.	Branch	DSER Section/ Question	Type	Title/Description Detail Status	Resp Engineer	(W) Status	NRC Status	Letter No. / Date
				For the reasons discussed above, the staff considers the equipment hatch covers buckling as a global failure mode. There is a potential for radioactive gas leakage through the equipment hatch sleeve/gasket once buckling occurs. Thus, the leaktight integrity of the containment is jeopardized. On this basis, the staff finds that a higher FS of 2.5 based on NE-3222 should be applied.				
				Closed - Response provided by NSD-NRC-97-4981 of 2/11/97.				
3270	NRR/ECGB	3.8.2	RAI-OI		CH42/Orr/Lutz	Closed	Action W	NSD-NRC-97-4981
				220.101 Westinghouse evaluated an additional BOSOR-5 analysis with stress-strain curves accounting for the effects of residual stresses on the buckling of cylindrical shells due to axial compression and/or external pressure. The failure mode was found to be an axisymmetric plastic collapse resulting from excessive vertical displacements at the pole. The maximum displacement was 1.09 m (43 in) at 1.45 MPa (195 psig). This information was requested by the staff to be provided in SSAR as discussed in RAI 6 (NRC letter dated September 14, 1995). In its response (NTD-NRC-96-4617 dated January 4, 1996), Westinghouse stated that the plastic collapse is bounded by the case for knuckle buckling without specific information. Provide this information in the SSAR.				
				Closed - Response provided by NSD-NRC-97-4981 of 2/11/97.				
3271	NRR/ECGB	3.8.2	RAI-OI		CH42/Orr/Lutz	Closed	Action W	NSD-NRC-97-4981
				220.102 In SSAR Section 3.8.2.4.2.5, mechanical and electrical penetrations are designed for a pressure of 90 psig at design temperature (280 F) for ASME Service Level C limits. In SSAR Section 3.8.2.4.2.8, however, the ASME Service Level C limit is 92 psig at 280 F from the containment ellipsoidal head plastic buckling. Clarify which pressure represents the ASME Service Level C limit at design temperature for the containment.				
				Closed - Response provided by NSD-NRC-97-4981 of 2/11/97.				



OPEN ITEM CLOSURE
02/13/97
Number of Open Items
Thousands
0
1
2
3
4
5
W ACTION
NRC ACTION
RESOLVED
CLOSED
Linear Goal plots to be revised upon incorporation of 12/23/96 meeting results
OCT 3 OCT 17 OCT 31 NOV 14 NOV 27 DEC 12 DEC 26 JAN 9 JAN 23 FEB 6 FEB 20 MAR 6 MAR 20 APR 3 APR 17 MAY 1 MAY 15 MAY 29 JUN 12 JUN 26 JUL 10 JUL 24 AUG 7 AUG 21 SEP 4 SEP 18 OCT 2 OCT 16 OCT 30
OCT 10 OCT 24 NOV 7 NOV 21 DEC 5 DEC 19 JAN 2 JAN 16 JAN 30 FEB 13 FEB 27 MAR 13 MAR 27 APR 10 APR 24 MAY 8 MAY 22 JUN 5 JUN 19 JUL 2 JUL 17 JUL 31 AUG 14 AUG 28 SEP 11 SEP 25 OCT 9 OCT 23 NOV 6
Closed Goal
Action W Goal
Total Open Items
Closed
Resolved
Confirm-W
Confirm-N
Audit N
Action N
Action W
Others



TOTAL OPEN ITEM CLOSURE
02/13/97
Number of Open Items
Thousands
5
4
3
2
1
0
SEP 19 OCT 3 OCT 17 OCT 31 NOV 14 NOV 27 DEC 12 DEC 26 JAN 9 JAN 23 FEB 6 FEB 20 MAR 6 MAR 20 APR 3 APR 17 MAY 1 MAY 15 MAY 29 JUN 12 JUN 26 JUL 10 JUL 24 AUG 7 AUG 21 SEP 4 SEP 18 OCT 2 OCT 16 OCT 30
SEP 26 OCT 10 OCT 24 NOV 7 NOV 21 DEC 5 DEC 19 JAN 2 JAN 16 JAN 30 FEB 13 FEB 27 MAR 13 MAR 27 APR 10 APR 24 MAY 8 MAY 22 JUN 5 JUN 19 JUL 2 JUL 17 JUL 31 AUG 14 AUG 28 SEP 11 SEP 25 OCT 9 OCT 23 NOV 6
We think we have left
NRC thinks we have left
NRC thinks are done
Total

FAX to DINO SCALETTI

February 13, 1997

CC: Sharon or Dino, please make copies for: Diane Jackson
Ted Quay

Don Lindgren
Preston Vock
Moshe Mahlab
Ed Cummins
Bob Vijuk
Brian McIntyre

OPEN ITEMS FOR SSAR SECTION 3.10

This is a background package for the remaining open items for SSAR section 3.10. SSAR section 3.10 is of interest because by our joint NRC/W schedule, the FSER for this section should be turned into Projects by the middle of March. There are 5 Open Items with NRC Status of Action W. Two (2) of these items (813 and 814) required some Westinghouse action. There have been a number of telephone conversations about these items recently and Westinghouse understands that NRC has prepared a letter detailing the current status of these items. We request that NRC issue this letter soon so that Westinghouse can properly address any remaining NRC concerns. Westinghouse believes the other three (3) items (1809, 1810 and 1811) were resolved by the issue of WCAP-13054 by letter NSD-NRC-96-4806 of 9/5/96 (over 5 months ago). Currently, our records show no additional outstanding Westinghouse action required for items 1809, 1810 and 1811 and we request that NRC provide a definitive action for Westinghouse or provide direction to change the status of these items. We recommend "Action N". Thank you.

Jim

Jim Winters
412-374-5290

AP600 Open Item Tracking System Database: Executive Summary

Date: 2/13/97

Selection: [nrc st code]='Action W' And [DSER Section] like '3.10*' Sorted by Item #

Item No.	Branch	DSER Section/ Question	Type	Title/Description Detail Status	Resp Engineer	(W) Status	NRC Status	Letter No. /	Date
813	NRR/EMEB	3.10.1	DSER-OI		Miller	Confrm-N	Action W	NSD-NRC-96-4841	

Westinghouse should revise the SSAR and WCAP-13054 to state that the COL applicant will submit all seismic experience data information to the staff for review and approval before including it in the equipment qualification file.

Resolved - SSAR subsections 3.10.1.1 and 3.10.6 address the requirement for the COL applicant to supply seismic experience data to the NRC for approval. SSAR Appendix 3D is consistent with this approach. This item will be closed upon issue of this same approach in WCAP - 13054.
Action W - The SSAR should be revised to state that the COL applicant will submit all of the information described in the DSER to the staff for review and approval prior to including this information in the equipment qualification file. In addition, WCAP 13054 should be revised to delete the exception to the applicable portion of SRP 3.10.
Per DCP/NRC0590, WCAP-13054 was updated to reflect COL responsibility related to SRP 3.10
Criteria 1. This item remains open pending resolution of a related SSAR comment received in an 8/20/96 letter from Jackson to Liparulo, Followon Questions and Staff Update to DSER Open Items Regarding the W AP600 Advanced Reactor Design
Action - W - 8/20/96 - Revision 5 to the SSAR revised Section 10.2 to respond to this issue. This revision is identical to the response to Q210.81. In Section 3.10 of the DSER, dated November, 1994, the staff stated that this response was not completely acceptable, and identified this issue as a DSER Open Item. The SSAR should be revised to state that the COL applicant will submit all of the information described in the DSER to the staff for review and approval prior to including this information in the equipment qualification file. In addition, WCAP 13054 should be revised to delete the exception to the applicable portion of SRP 3.10.
Action W - SSAR 3.10.6 will be revised to indicate that the COL item is addressed as part of the aplication.
Resolved - See response in Letter NSD-NRC-96-4841, dated October 14, 1996. Subsection 3.10.6 of SSAR will be revised.
Confirm-N - Subsection 3.10.6 Revision 10 addressed this issue.

c f 2

Action W - In a letter from McIntyre to Quay dated October 14, 1996, Westinghouse proposed a revision to SSAR Section 3.10.6 which states that the COL applicant, as a part of the Combined License application, will identify equipment qualified based on experience and include details of the methodology and the corresponding experience data. This agrees with the staff's request on this item, and is acceptable. However, the exception to SRP 3.10 in Revision 2 to WCAP 13054 contains statements which either need to be deleted or clarified. The first two sentences imply that IEEE 344-1987 is acceptable relative to the use of experience data. RG 1.100, Revision 2 states that this method of qualification in IEEE 344-1987 will be evaluated by the staff on a case-by-case basis. It appears to the staff that the exception in the WCAP is relative to RG 1.100, Revision 2. These two sentences should be revised to reflect the position in RG 1.100, Rev. 2. In addition, the discussion relative to Generic Issue A-46 is not applicable to new plants. The staff's position is that A-46 is only used for verification of equipment in operating plants, and is not acceptable for qualification of equipment in ALWRs. This discussion should either be deleted or revised.

Item No.	Branch	DSER Section/ Question	Type	Title/Description Detail Status	Resp Engineer	(W) Status	NRC Status	Letter No. /	Date
814	NRR/EMEB	3.10.2	DSER-OI		Vock/SSARREV	Action W	Action W	NSD-NRC-96-4841	

In addition to revising WCAP-13054, Westinghouse should revise Section 3.9.3 or 3.10 of the SSAR to describe the methodology used in the AP600 design to analyze the feedwater line valve disks when they are subjected to dynamic loads from a LOCA. (RAI210.85)

Closed - Added a statement as follows to SSAR Rev. 7 "Valve discs are evaluated for maximum design line pressure and maximum differential pressure resulting from plant operating, transient, and accident conditions. Valve operating conditions are included as part of the valve design specification and are used to evaluate the valve disc.
Action W - the SSAR should be revised to describe the methodology used in the AP600 design to analyze the feedwater line valve disks when they are subjected to dynamic loads due to a LOCA. In addition, as requested in the DSER, WCAP 13054 should be revised to delete an exception to SRP, Section 3.10.II.1.a(14)(b).
Action W - Include information in the SSAR on the method of analysis of dynamic effects.
Resolved - See response in Letter NSD-NRC-96-4841, dated October 14, 1996. Valves are analyzed using equivalent static loads. SSAR 3.10.2.2 will be revised.
Action W - Additional information is required on the method used to determine the equivalent static load.
Action W - In a letter from McIntyre to Quay dated October 14, 1996, Westinghouse responded to this item by proposing a revision to the fourth paragraph of SSAR, Subsection 3.10.2.2 to state that feedwater line valve disks are evaluated for the effect of dynamic loads of pipe breaks by considering the effect of an equivalent differential pressure. This does not appear to address the staff's concerns. The staff considers equivalent differential pressure as being a static load. The SSAR should be revised to describe the methodology used in the AP600 design to analyze the dynamic closure of feedwater line valve disks when they are subjected to dynamic loads due to a pipe break.

AP600 Open Item Tracking System Database: Executive Summary

Date: 2/13/97

Selection: [nrc st code]='Action W' And [DSER Section] like '3.10*' Sorted by Item #

Item No	Branch	DSER Section/ Question	Type	Title/Description Detail Status	Resp Engineer	(W) Status	NRC Status	Letter No. /	Date
1809	NRR/EMEB	3.10.2	DSER-CN		13054	Closed	Action W	NSD-NRC-96-4806	
				3.10-2 Westinghouse should revise WCAP-13054 to remove the exception.					
				Action W - Westinghouse will revise WCAP-13054 to remove the exception as stated in response to RAI 210.82. Closed - Revision 2 of WCAP-13054 was transmitted by letter NSD-NRC-96-4806 dated September 5, 1996.					
1810	NRR/EMEB	3.10.3	DSER-CN		13054	Closed	Action W	NSD-NRC-96-4806	
				3.10-3 Westinghouse should revise WCAP-13054 to remove the exception.					
				Action W - Westinghouse will revise WCAP-13054 to remove the exception as promised in RAI 210.82. Closed - WCAP-13054 Revision 2 transmitted by letter NSD-NRC-96-4806 dated September 5, 1996.					
1811	NRR/EMEB	3.10.4	DSER-CN		13054	Closed	Action W	NSD-NRC-96-4806	
				3.10-4 Westinghouse should revise the SSAR and WCAP-13054 as noted in Section 3.10 of this report.					
				Action W - Westinghouse will revise WCAP-13054 to remove the exception as promised in RAI 210.88. Westinghouse will incorporate the SSAR revision from the response to RAI 210.86 Closed - WCAP-13054 Rev. 2 transmitted by letter NSD-NRC-96-4806 dated September 5, 1996 revise the position on criteria 5c.					

W
of
W

Westinghouse

FAX COVER SHEET

RECIPIENT INFORMATION		SENDER INFORMATION	
DATE:	FEBRUARY 12, 1997	NAME:	TIM WINTERS
TO:	DIANE JACKSON	LOCATION:	ENERGY CENTER - EAST
PHONE:	FACSIMILE:	PHONE:	Office: 412-374-5290
COMPANY:	US NRC	Facsimile:	win: 284-4887 outside: (412)374-4887
LOCATION:			

Cover + Pages 1 + 3

The following pages are being sent from the Westinghouse Energy Center, East Tower, Monroeville, PA. If any problems occur during this transmission, please call:

WIN: 284-5125 (Janice) or Outside: (412)374-5125.

COMMENTS:

DIANE,

THIS IS A RESEND OF THE FEBRUARY 6 FAX WHICH SHOULD RESOLVE ITEM 346. IT IS BEING RESENT TO INCLUDE THE INSERT. IT WILL BE IN SSAR REVISION 11 UNLESS WE HEAR FROM YOU.

CC: LINDGREN
MCINTYRE
RON VIJUK
CUMMINS
WINTERS
HAYLS
HUTCHINGS
JEANNE EVANS

Each of the generators is directly coupled to the diesel engine. Each diesel generator unit is an independent self-contained system complete with necessary support subsystems that include:

- Diesel engine starting subsystem
- Combustion air intake and engine exhaust subsystem
- Engine cooling subsystem
- Engine lubricating oil subsystem
- Engine speed control subsystem
- Generator, static exciter, generator protection, monitoring instruments and controls subsystems

a compressor inlet air filter an in-line air filter (with dewpoint at least 10°F less than the lowest normal diesel generator room temperature

The diesel-generator starting air subsystem consists of an ac motor-driven, air-cooled compressor, an air-cooled aftercooler, refrigerant dryer, and an air receiver with sufficient storage capacity for three diesel engine starts. The interconnecting stainless steel piping from the compressor to the diesel engine dual air starter system includes air filters, moisture drainers, and pressure regulators to provide clean dry compressed air for engine starting.

at normal diesel generator room temperature

The diesel-generator combustion air intake and engine exhaust subsystem provides combustion air directly to the diesel engine and discharges exhaust gases from the engine to the outside of the diesel generator building. The combustion air circuit includes weather protected dry type inlet air filters piped directly to the inlet connections of the diesel engine-mounted turbochargers. The engine exhaust gas circuit consists of the engine exhaust gas discharge pipes from the turbocharger outlets to a single vertically mounted outdoor silencer which discharges to the atmosphere.

The diesel-generator engine cooling system is an independent closed loop cooling system, rejecting engine heat through two separate roof-mounted, fan-cooled radiators. The system consists of two separate cooling loops each maintained at a temperature required for optimum engine performance by separate engine-driven coolant water circulating pumps. One circuit cools the engine cylinder block, jacket, and head area, while the other circuit cools the oil cooler and turbocharger aftercooler. The cooling circuit, which cools the engine cylinder blocks, jacket, and head areas, includes a keep-warm circuit consisting of a temperature controlled electric heater and an ac motor-driven water circulating pump.

The diesel-generator engine lubrication system is contained on the engine skid and includes an engine oil sump, a main engine driven oil pump and a continuous engine prelube system consisting of an ac and dc motor driven prelube pump and electric heater. The prelube system maintains the engine lubrication system in service when the diesel engine is in standby mode. The lube oil is circulated through the engine and various filters and coolers to maintain the lube oil properties suitable for engine lubrication.

The diesel generator engine fuel oil system consists of an engine-mounted, engine-driven fuel oil pump that takes fuel from the fuel oil day tank, and pumps through inline oil filters to the engine fuel injectors and a separate recirculation circuit with a fuel oil cooler. The recirculation circuit discharges back to the fuel oil day tank that is maintained at the proper fuel level by the diesel fuel oil storage and transfer system.

INSERT 8.3-7-1

The design of the onsite standby diesel generators does not ensure functional operability or maintenance access or support plant recovery following design basis events. Maintenance accessibility is provided consistent with the system nonsafety-related functions and plant availability goals.

The piping and instrumentation diagrams for the onsite standby diesel generator units and the associated subsystems are shown on Figures 8.3.1-4 and 8.3.1-5.

The onsite standby power supply system is shown schematically on one line diagram, Figure 8.3.1-1.

8.3.1.1.2.2 Generator

Each generator is a direct-shaft driven, air-cooled self ventilated machine. The generator enclosure is open drip-proof type that facilitates free movement of ventilation air. The generator component design is in compliance with the NEMA MG-1 (Reference 1) requirements.

Each generator produces its rated power at 4160 V, 60 Hz. Each generator continuous rating is based on supplying the electrical ac loads listed in Tables 8.3.1-1 or 8.3.1-2. The loads shown on Tables 8.3.1-1 and 8.3.1-2 represent a set of nonsafety-related loads which provide shutdown capability using nonsafety-related systems. The generators can also provide power for additional investment protection ac loads. The plant operator would normally provide power to these loads by deenergizing one of those system components that are redundantly supplied by both the diesel generators. The diesel generator design is compatible with the step loading requirements identified in Tables 8.3.1-1 and 8.3.1-2. The generator exciter and voltage regulator systems are capable of providing full voltage control during operating conditions including postulated fault conditions.

Each generator has a set of potential and current transformers for protective relaying and metering purposes.

The following generator protection functions are provided via relays that are mounted on the local generator control panel:

> Differential (87), overcurrent (50/51), reverse power (32), underfrequency (81), under/over voltage (27/59), loss of excitation (40), ground fault (51g), negative sequence (46), synchronization check (25), voltage balance (60).

Note: The number in the parentheses identifies the ANSI device designation.

INSERT 8.3-7-1

The onsite standby diesel generators are provided with necessary controls and indicators for local or remote monitoring of the operation of the units. Essential parameters are monitored and alarmed in the main control room via the plant data display and processing system as described in Chapter 7.

Post-It™ brand fax transmittal memo 7671	# of pages ▸ 5
To Bill Huffman	From Tim Wilson
Co. USNRC	Co. Westinghouse
Dept.	Phone # 412-374-5470
Fax #	Fax #

3.8.3.4.2 Hydrodynamic Analyses

This subsection describes the hydrodynamic analyses for automatic depressurization system discharge into the in-containment refueling water storage tank. This discharge is designated as ADS_1 in the load description of subsection 3.8.3.3.1. The first three stages of the automatic depressurization system valves discharge into the tank through spargers under water, producing hydrodynamic loads on the tank walls and equipment. Hydrodynamic loads, measured in hydraulic tests of the automatic depressurization system sparger in a test tank, are evaluated using the source load approach (Reference 34). Analyses of the tests define source pressure loads that are then used in analyses of the in-containment refueling water storage tank to give the dynamic responses of the contaiment internal structures. The basic analysis approach consists of the following steps:

1. A pressure source, an impulsive forcing function at the sparger discharge, is selected from the tests using a coupled fluid structure finite element model of the test tank, taking into account fluid compressibility effects. This source development procedure is based on a comparison between analysis and test results, both near the sparger exit and at the boundaries of the test tank.

2. The pressure source is applied at each sparger location in a coupled fluid structure finite element model of the in-containment refueling water storage tank structure and of the contained water. The mesh characteristics of the model at the sparger locations and the applied forcing functions correspond to those of the test tank analysis.

3.8.3.4.2.1 Sparger Source Term Evaluation

A series of tests was conducted with discharge conditions representative of one sparger for the AP600 (References 35 and 36). Pressure traces measured during the test discharges were investigated, at both sparger exit and tank boundaries to (1) bound the expected discharge from the automatic depressurization system; (2) characterize the pressure wave transmission through the pool water; (3) determine the maximum pressure amplitudes and the frequency content; and (4) produce reference data for qualification of the analytical procedure. Pressure time histories and power spectrum densities were examined at reference sensors, both for the total duration of the discharge transient (about 50 seconds) and for critical time intervals.

Fluid-structure interaction analyses were performed with the ANSYS computer code (Reference 37). The mathematical model consists of a 3D sector finite element model, 15 degrees wide, as shown in Figure 3.8.3-9. It uses STIF30 fluid and STIF63 structural ANSYS finite elements, which take into account fluid compressibility and fluid-structure interaction. Rayleigh damping of 4 percent is used for the concrete structure, and fluid damping is neglected. Direct step-by-step time integration is used. The measured discharge pressures for single time intervals are imposed as uniform forcing functions on the idealized spherical surface of the steam/water interface, and pressures transmitted through the water to the tank boundary are calculated and compared with test measurements. The analyses of the test tank showed satisfactory agreement for the pressures at the tank boundary.

The examination of test results related to the structural design of the in-containment refueling water storage tank under automatic depressurization system hydrodynamic excitation and the comparison with the analytical procedure previously described, lead to the following conclusions regarding the sparger source term definition:

- The automatic depressurization system discharge into cold water produces the highest hydrodynamic pressures. The tests at higher water temperatures produce significantly lower pressures.

- Two pressure time histories, characterized by different shapes and frequency content, can be selected as representative of the sparger discharge pressures; they are assumed as acting on a spherical bubble centered on the sparger centerline and enveloping the ends of the sparger arms.

- The application of such time histories as forcing functions to an analytical model, simulating the fluid structure interaction effects in the test tank, has been found to predict the measured tank wall pressures, for the two selected reference time intervals.

- The two defined sparger source term pressure time histories can be used as forcing functions for global hydrodynamic analyses of the in-containment refueling water storage tank by developing a comprehensive fluid-structure finite element model and reproducing the test tank mesh pattern in the sparger region.

3.8.3.4.2.2 In-Containment Refueling Water Storage Tank Analyses

The in-containment refueling water storage tank is constructed as an integral part of the containment internal structures as described in subsection 3.8.3.1.3. It contains two depressurization spargers that are submerged approximately 9 feet below the normal water level. Transmission of the hydrodynamic pressures from the sparger discharge to the wetted in-containment refueling water storage tank is evaluated using the coupled fluid-structure interaction method similar to that described for the test tank analysis in the previous subsection.

The 3D ANSYS finite element model includes the in-containment refueling water storage tank boundary, the water within the in-containment refueling water storage tank, the adjacent structural walls of the containment internal structures, and the operating floor. The model of the in-containment refueling water storage tank, shown in Figures 3.8.3-10, 3.8.3-11, and 3.8.3-12, represents the outer steel structures, the inner concrete walls, and the water. The flexible steel outer wall is represented using beam elements; isotropic plate elements are used to represent the inner structural module walls. The water is modelled as a compressible fluid to provide an acoustic medium to transmit the source pressure. The model has two bubble boundaries representing the spargers. Pressure loads are applied to the solid element faces adjacent to the air bubbles. The forcing functions at the sparger locations are conservatively assumed to be in phase. Rayleigh damping of 5 percent is used for the concrete-filled structural modules and fluid damping is neglected. All degrees of freedom were retained in the step-by-step direct integration solution procedure for the in-containment refueling water

4.5.10 Source term evaluation

Through the comparisons between measured and predicted wall pressures, we conclude that two pressure source terms can be used for the subsequent IRWST hydrodynamic response analysis.

The first Source Term Subinterval, shown in fig. 4.5.11 lasts 0,6 sec., has a maximum and minimum values of about 0.22/- 0.17 bars and has a significant frequency content especially in the below 40 Hz frequency range.

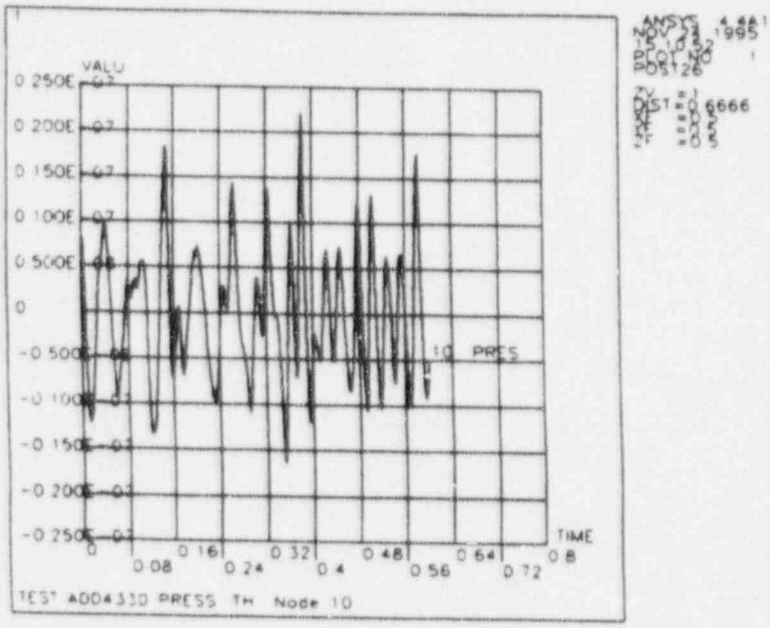
The second one, shown in fig. 4.5.12 last 1 sec, has maximum and minimum values of about 0,22/-0,2 bars and has a significant frequency content expecially in the above 40 Hz frequency range.

Although the use of such excitations leads to predict rather conservative test tank wall pressures against those measured, it is estimated that they represent an appropriate input for the IRWST hydrodynamic analysis. The implied input conservatism can infact compensate other uncertainties possibly effecting the test tank measurement, the results interpretation and the application of the source term to the IRWST environment.

FIGURE 4.5.11

Input Source Term - Low Frequency

(PE13 - 10.2 - 10.8 sec. Test 330)



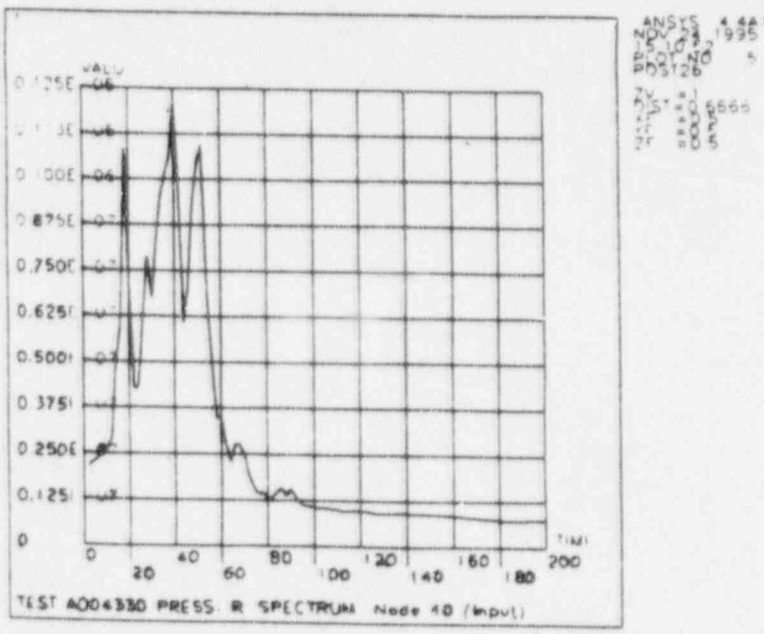
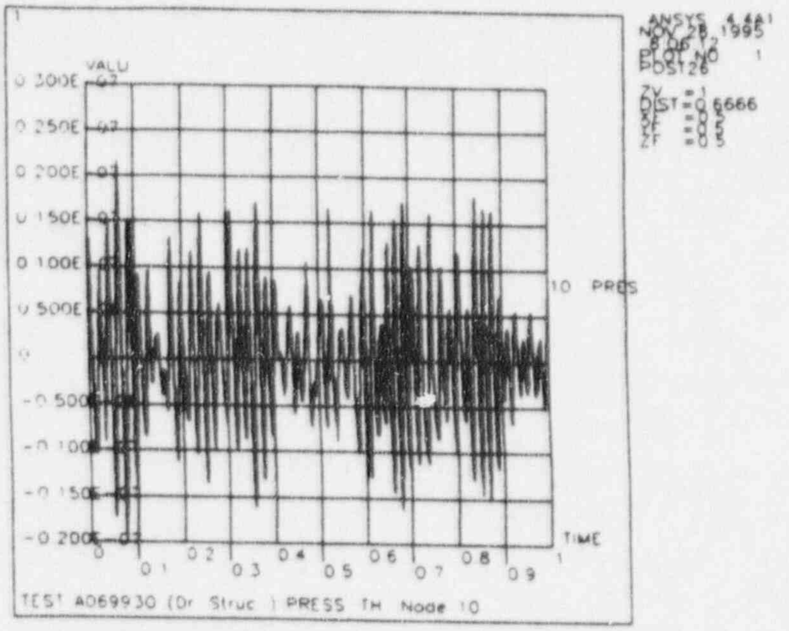
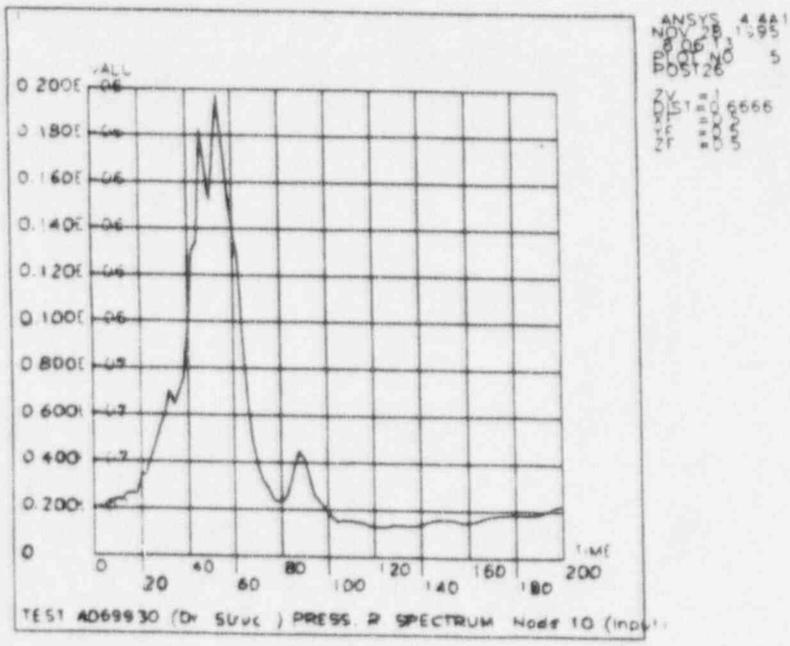


FIGURE 4.5.12

Input Source Term - High Frequency

(PE13 - 26.2 - 27.2 sec. Test 930)





FAX to DINO SCALETTI

February 12, 1997

CC: Sharon or Dino, please make copies for:

Ted Quay
Bill Huffman
Diane Jackson
Tom Kenyon
Joe Sebrosky
Robin Nydes
Cindy Haag
Don Lindgren
Bob Tupper
Bruce Rarig
Brian McIntyre
Ed Cummins
Bob Vijuk

NRC is requested to please acknowledge receipt of information related to each of the following Open Items. These are a subset of the items with "Action W" in "NRC Status" for which I have personally checked, since the first of the year, that we have submitted what we believe is the resolving information. Unlike those on the other list I sent you today, I have not prepared a background package for each of these. However, the reviewer in each case should have a submittal from Westinghouse as identified in OITS for the item. Recognizing that reviewing for completeness of the response in each case constitutes an NRC action, we recommend that receipt acknowledgement be accompanied by direction to change their "NRC Status" to "Action N". If these are truly "Action W", please provide a description of the action Westinghouse is expected to take. We know of no action required. This is the fifth weekly request of this type.

142, 157, 164, 182, 184, 262, 300, 305, 308, 333, 405, 457, 458, 628, 801, 802, 805, 807, 809, 972, 973, 1009, 1037, 1038, 1039, 1040, 1041, 1043, 1045, 1052, 1053, 1055, 1101, 1102, 1195, 1197, 1210, 1225, 1226, 1227, 1228, 1231, 1232, 1317, 1458, 1461, 1697, 1698, 1699, 1700, 1701, 1702, 1703, 1704, 1707, 1716, 1727, 1730, 1731, 1742, 1745, 1747, 1749, 1753, 1760, 1885, 1996, 1999, 2018, 2019, 2023, 2024, 2025, 2034, 2040, 2045, 2051, 2199, 2200, 2201, 2202, 2272, 2273, 2442, 2457, 2676, 2683, 2684, 2686, 2691, 2939, 2942, 2945, 2958, 2959, 2960, 2961, 2962, 2963, 2964, 2965, 2966, 2967, 2968, 2969, 2970, 2971, 2972, 2973, 2974, 2975, 2976, 2977, 2978, 2979, 2981, 2982, 2983, 2984, 2985, 2986, 3098, 3122, 3126. 3127, 3128, 3197, 3372, 3398, 3399, 3427, 3469, 3470, 3505, 3517, 3895, 3944, 3945, 3946, 3947, 3948, 3949, 3950, 3951, 3952, 3953, 3954, 3955, 3956, 3957, 3958, 4123, 4124, 4125, 4126, 4127, 4128, 4129, 4130, 4131, 4132, 4133, 4134, 4135, 4136, 4137, 4138, 4139, 4140, 4141, 4142, 4143, and 4144.

Jim

Thanks
Jim Winters
412-374-5290

FAX to DINO SCALETTI

February 12, 1997

CC: Sharon or Dino, please make copies for: Bill Huffman
Ted Quay

Don Lindgren
Mike Corletti
Ed Cummins
Bob Vijuk
Brian McIntyre

OPEN ITEM #164 (M5.2.5-20)

In my quest to make sure we have provided NRC with everything needed to prepare an FSER, I am researching open items from the smallest item number on. The relevant documentation related to Open Item #164 (M5.2.5-20) is attached. We provided this FAX response on January 10, 1997. We believed that this list of references resolved the concerns of item #164 and subsequent telephone conversations. It seems a reasonable request that NRC acknowledge receipt of the change. We request that NRC provide a definitive action for Westinghouse or provide direction to change the status of this item. We recommend "Action N". Thank you.

Jim

Jim Winters
412-374-5290

AP600 Open Item Tracking System Database: Executive Summary

Date: 2/12/97

Selection: [item no] between 164 And 164 Sorted by Item #

Item No	Branch	DSER Section/ Question	Type	Title/Description Detail Status	Resp Engineer	(W) Status	NRC Status	Letter No. /	Date
164	NRR/SPLB	5.2.5	MTG-OI	M5.2.5-20 (REACTOR COOLANT PRESSURE BOUNDARY LEAKAGE) Identify each system connected to the reactor coolant system (RCS) that could experience intersystem leakage and provide a discussion of the leak detection method, including protective features to ensure that the system does not overpressurize. Closed - Westinghouse has completed necessary submittals to support staff review. See the response for RAI 440.132 for a discussion of this issue. Action W - per 12/2 telecon, Westinghouse to provide explicit references to where we covered the systems connected to the RCS in the SSAR or other document. Action N - FAX to Huffman on 1/10/97 provided explicit references.	Corletti,M.	Closed	Action W		

2
2
3

 Total Records: 1

FAX to BILL HUFFMAN

January 10, 1997

CC: Don Lindgren
Mike Corletti
Brian McIntyre

ADDITIONAL INFORMATION FOR OPEN ITEM 164

This is in response to the 12/2/96 request to provide, explicitly, where we covered leakage from each system connected to RCS in the SSAR or other document. We explicitly cover intersystem leakage from the RCS in WCAP-14425, the ISLOCA report. This WCAP is referenced in the SSAR in a number of places. The most relevant are in section B-63 of SSAR section 1.9, and in subsection 1.9.5.1.7. We believe this completes Westinghouse actions required for Open Item 164 and request NRC direction to change its "NRC Status". We recommend "Action N".

Jim

Jim Winters
412-374-5290

3-13

FAX to DINO SCALETTI

February 12, 1997

CC: Sharon or Dino, please make copies for: Ted Quay
Bill Huffman
Diane Jackson
Tom Kenyon
Joe Sebrosky

Cindy Haag
Don Lindgren
Robin Nydes
Brian McIntyre
Ed Cummins
Bob Vijuk

This is a reminder list of the Open Items where we have recently documented the difference between "W Status" and "NRC Status". In all cases, we believe the next action has been at NRC for over 3 months and await your definitization of a Westinghouse action or your direction to change the "NRC Status" to something other than "Action W".

Open Item Number	Westinghouse Submittal	Request for Status Change
142 (M3.11-9)	2/29/96	2/3/97
157 (M5.2.5-13)	1/9/97	2/12/97
405	7/8/96	2/11/97
1210 (DSER 12.4.2-2)	4/30/96	2/6/97
1227	7/8/96	2/11/97
1228	7/8/96	2/11/97
1231	7/8/96	2/11/97
1232	7/8/96	2/11/97
2034	7/8/96	2/11/97

Note that the status was changed for Items 4, 21, 30, 37, 123, 134, 135, 137, 138, 139, 140, 141, 144, 158, 586, 969, 970, 971, 1300, and 1301 so they have been removed from the table.

Thanks for your help.

Jim

FAX to DINO SCALETTI

February 12, 1997

CC: Sharon or Dino, please make copies for: Bill Huffman
Ted Quay

Don Lindgren
Mike Corletti
Ed Cummins
Bob Vijuk
Brian McIntyre

OPEN ITEM #157 (M5.2.5-13)

In my quest to make sure we have provided NRC with everything needed to prepare an FSER, I am researching open items from the smallest item number on. The relevant documentation related to Open Item #157 (M5.2.5-13) is attached. We provided this FAX response on January 9, 1997. We believed that this list of references resolved the concerns of item #157 and subsequent telephone conversations. It seems a reasonable request that NRC acknowledge receipt of the change. We request that NRC provide a definitive action for Westinghouse or provide direction to change the status of this item. We recommend "Action N". Thank you.

Jim

Jim Winters
412-374-5290

AP600 Open Item Tracking System Database: Executive Summary

Date: 2/12/97

Selection: [item no] between 157 And 157 Sorted by Item #

Item No	Branch	DSER Section/ Question	Type	Title/Description Detail Status	Resp Engineer	(W) Status	NRC Status	Letter No. /	Date
157	NRR/SPLB	5.2.5	MTG-OI		Corletti,M.	Closed	Action W		

M5.2.5-13 (REACTOR COOLANT PRESSURE BOUNDARY LEAKAGE) Identify each system that's susceptible to intersystem leakage, discuss the method of leak detection, and protective features.

Closed - Westinghouse has completed necessary submittals to support staff review.
Action N - NRC to review RAI 440.132, 210.61, Section 5.4.7, and Section 1.9.5.
Action W - per 12/2 telecon, Westinghouse to provide explicit references to where we covered the CVS portion of ISLOCA in the SSAR or other document.
Action N - FAX to Huffman on 1/9/97 provided explicit references.

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FAX to BILL HUFFMAN

January 9, 1997

CC: Don Lindgren
Mike Corletti
Brian McIntyre

ADDITIONAL INFORMATION FOR OPEN ITEM 157

This is in response to the 12/2/96 request to provide, explicitly, where we covered the CVS portion of ISLOCA in the SSAR or other document. We explicitly cover the CVS portion of ISLOCA in WCAP-14425, the ISLOCA report. This WCAP is referenced in the SSAR in a number of places. The most relevant to the CVS potion of ISLOCA are in section B-63 of SSAR section 1.9, and in subsection 1.9.5.1.7. We believe this completes Westinghouse actions required for Open Item 157 and request NRC direction to change its "NRC Status". We recommend "Action N".

Jim

Jim Winters
412-374-5290

583

Westinghouse

FAX COVER SHEET

RECIPIENT INFORMATION		SENDER INFORMATION	
DATE:	February 11, 1997	NAME:	Steve Kerch
TO:	Jim Bongarra / Jim Higgins	LOCATION:	Monroeville, PA.
PHONE:	301-415-1046 / 516-344-3638	PHONE:	412-374-5704
COMPANY:	NRC / BNL		
LOCATION:		FAX:	(412) 374-5099

Cover + Pages = 1 + 10 11 total

Jim Bongarra and Jim Higgins,

Attached are the proposed markups of SSAR chapter 18 and chapter 7 that address the remaining issues with the DSER open item on "Minimum Inventory". These markups are consistent with the approaches that we agreed upon during our conference calls last Wednesday and Thursday and are pending review and approval by our management review team. Not included with the chapter 7 markup is a proposed change to figure 7.2-1 that will add the functional logic for the manual initiation of the main control room emergency habitability system. This is currently under review by the chapter 7 author and will be forwarded as soon as available.

Thank You,
Steve Kerch

Phone Number of Receiving Equipment: Jim B. 301-415-2222
Jim H. 516-344-4900

By accomplishing the emergency response guideline critical safety functions following a design basis event, the plant is able to mitigate the consequences of the event and to establish and maintain safe shutdown conditions. The minimum inventory list identifies sufficient controls, displays, and alarms to monitor and control operation of the safety-related systems to achieve the six critical safety functions identified in the Emergency Response Guidelines and to establish and maintain safe shutdown conditions following an accident.

Tables 7.5-4, 7.5-5, and 7.5-6 identify the instrumentation and the associated Emergency Response Guidelines critical safety functions that each instrument supports for each of the Type A, B, and C post-accident instrumentation, respectively.

Minimum Inventory Selection Criteria Implementation Process

Section 7.5 provides a discussion of the development of the requirements of Regulatory Guide 1.97 and the implementation process for the AP600 (Criteria 1, 2, and 4).

Section 18.7 and Reference 1 provide a discussion of the implementation process for identification of critical PRA operator actions (Criteria 3). Chapter 30 of the AP600 PRA describes the process for the human reliability analysis.

18.12.3 Remote Shutdown Workstation Displays, Alarms, and Controls

Subsection 7.4.3 discusses safe shutdown using the remote shutdown workstation following an evacuation of the main control room.

The main control room provides the capability to perform accident mitigation and safe shutdown tasks for design basis events. The only types of events that would require evacuation of the main control room and control from the remote shutdown workstation are localized emergencies where the main control room environment is unsuitable for the operators or where the main control room workstations and equipment become damaged.

Evacuation of the main control room is not expected to occur coincident with any other design basis events. Subsection 9.5.1 of the Standard Review Plan (NUREG-0800) specifically excludes consideration of other design basis events coincident with a fire.

The design capability for the remote shutdown workstation is to provide the capability to establish and maintain safe shutdown conditions following a main control room evacuation, as described in subsection 7.4.3.1.1. ~~The design basis for the remote shutdown workstation does not require the installation of dedicated, fixed-position displays, alarms, and controls and the AP600 minimum inventory requirements are not applicable for the remote shutdown workstation.~~ The controls, displays and alarms listed in Table 18.12.2-1 are retrievable from the remote shutdown workstation.

18.12.4 Combined License Information

This section has no requirement for additional information to be provided in support of the Combined License application.

18.12.5 References

1. WCAP-14651, "Integration of Human Reliability Analysis With Human Factors Engineering Design Implementation Plan."

2. WCAP-13793, "The AP600 System/Event Matrix," June 1994.

4 of 11

Table 18.12.2-1 (Sheet 1 of 2)

MINIMUM INVENTORY

Instrument/Display ~~Fixed Position Instrumentation~~	Control	Display	Alarm(2)	SSAR Source
Insert ① and ② → Neutron flux		~~B1~~ ← X	X	Table 7.5-5
RCS pressure		~~B1, B2, C1~~ X	X	Table 7.5-5, 6
WR T_{hot}		~~B1, B2~~ ← X		Table 7.5-5
WR T_{cold}		~~B1, B2~~ ← X	X	Table 7.5-5
Insert ③ and ④ → Containment water level		~~B1, C1~~ ← X	X	Table 7.5-5, 6
Containment pressure		~~B1, C2~~ ← X		Table 7.5-5, 6
Pressurizer water level		~~B1~~ ← X	X	Table 7.5-5
Insert ⑤ → Pressurizer reference leg temperature		~~B1~~ ← X		Table 7.5-5
Insert ⑥ → Pressurizer pressure		~~B1~~ ← X		Table 7.5-5
Core exit temperature		~~B1, C1~~ ← X	X	Table 7.5-5, 6
RCS subcooling		~~B1~~ ← X	X	Table 7.5-5
Insert ⑦ → IRWST water level		~~B1~~ ← X	X	Table 7.5-5
PRHR flow		~~B1~~ ← X	X	Table 7.5-5
PRHR outlet temperature		~~B1~~ ← X	X	Table 7.5-5
PCS storage tank water level		~~B1~~ ← X		Table 7.5-5
PCS cooling flow		~~B1~~ ← X		Table 7.5-5
IRWST to RNS suction valve status		~~B1~~ ← X	X	Table 7.5-5
Remotely Operated → Containment isolation (3) valve status		~~B1~~ ← X		Table 7.5-5
Containment area high range radiation level		~~C1~~ ← X	X	Table 7.5-6
Containment pressure (extended range)		~~C1~~ ← X	X	Table 7.5-6
Containment hydrogen concentration		~~C1~~ ← X		Table 7.5-6
CMT level (1)		X		
Manual reactor trip (also initiates turbine trip Figure 7.2-1, sheet 19.)	x			Table 7.2-4, PMS Also initiates turbine trip Figure 7.2-1 (Sheet 19), DAS MG set trip

		Control	Display	Alarm[1]
Insert 1:	Neutron Flux Doubling			X
Insert 2:	Startup Rate		X	X
Insert 3:	RCS Cooldown Rate Compared to the Limit Based on RCS Pressure		X	X
Insert 4:	Change of RCS temperature by more than 5°F in the last 10 minutes			X
Insert 5:	Pressurizer Water Level Trend		X	
Insert 6:	Reactor Vessel - Hot Leg Water Level		X	X
Insert 7:	RCS Cold Overpressure Limit		X	X
~~Insert 8:~~	~~[illegible]~~		~~[illegible]~~	

Table 18.12.2-1 (Sheet 2 of 2)

MINIMUM INVENTORY

Instrument/Display ~~Fixed Position Instrumentation~~	Control	Display	Alarm (2)	~~SSAR Source~~
Manual safeguards actuation	x			~~Table 7.2-4, PMS Also initiates reactor trip Table 7.3-3, PMS~~
Manual CMT actuation	x			~~Table 7.2-4, PMS Also initiates reactor trip Table 7.3-3, PMS Figure 7.2-1 (Sheet 19), DAS~~
Manual Main Control Room (4) emergency habitability system actuation	X			
Manual ADS actuation	x			~~Table 7.2-4, PMS Also initiates reactor trip Table 7.3-3, PMS Figure 7.2-1 (Sheet 20), DAS~~
(1-3 and 4)				
1 / 2 / 3 / 4				
Manual PRHR actuation	x			~~Table 7.3-3, PMS Figure 7.2-1 (Sheet 19), DAS~~
Manual containment cooling actuation	x			~~Table 7.3-3, PMS Figure 7.2-1 (Sheet 20), DAS~~
Manual IRWST injection actuation	x			~~Table 7.3-3, PMS Figure 7.2-1 (Sheet 20), DAS~~
Manual containment recirculation actuation	x			~~Table 7.3-3, PMS Figure 7.2-1 (Sheet 20), DAS~~
CV line / MOV line				
Manual containment isolation	x			~~Table 7.3-3, PMS Figure 7.2-1 (Sheet 20), DAS~~
Manual main steamline isolation	x			~~Table 7.3-3, PMS~~
Manual feedwater isolation	x			~~Table 7.3-3, PMS~~
Manual containment hydrogen igniter	x			~~Figure 7.2-1 (Sheet 20), DAS~~

Notes: (1) Although this parameter does not satisfy any of the selection criteria of subsection 18.12.2, its importance to manual actuation of ADS justifies its placement on this list.

(2) These parameters are used to generate visual alerts (safety related displ for the main control room; non-safety related displays for the remote shutdown workstation) that identify challenges to any of the critical safety functions.

(3) These instruments are not required after 24 hours. (Subsection 7.5.4 includes more information on the class 1E valve positio indication signals, specified as part of the post accident monitoring instrume

Notes (continued):

(4) This manual actuatiin capability is not needed at the remote shutdown workstatiin.

No changes

coolant pumps are tripped during engineered safety system actuation, the reactor coolant pumps are not available. However, the reactor coolant system is designed to provide sufficient natural circulation to achieve safe shutdown conditions with the steam generators and passive residual heat removal heat exchanger removing decay heat. Natural circulation flow is verified by monitoring the reactor coolant system temperatures.

7.4.2.5 Other Systems Required for Safe Shutdown

The other safety-related equipment and systems used to maintain the plant in safe shutdown are identified in Table 7.4-1. They are also listed below, with a reference to the respective section or subsection which discusses their operation in more detail:

- Protection and safety monitoring system — Sections 7.2, 7.3, and 7.5
- Class 1E dc and UPS system — Subsection 8.3.2

These systems are either normally operating or they start automatically when required. The instrumentation for these systems is described in the particular section containing the system description.

The monitoring instrumentation available in the main control room for safe shutdown are safety-related and are part of the qualified data processing system. The instrumentation available for safe shutdown monitoring is listed in Section 7.5.

7.4.3 Safe Shutdown from Outside the Main Control Room

7.4.3.1 Description

If temporary evacuation of the main control room is required because of some abnormal main control room condition, the operators can establish and maintain safe shutdown conditions for the plant from outside the main control room through the use of controls and monitoring located at the remote shutdown workstation. Safe shutdown is a stable plant condition that can be maintained for an extended period of time. In the event that access to the main control room is restricted, the plant is maintained in safe shutdown until the main control room can be re-entered.

7.4.3.1.1 Remote Shutdown Workstation

Safe shutdown can be established and maintained from the remote shutdown workstation. The workstation is designed to allow control of a shutdown following an evacuation of the control room, coincident with the loss of offsite power and a single active failure. No other design basis event is postulated. Subsection 9.5.1 provides a discussion of shutdown in the event of a fire. The remote shutdown workstation equipment is similar to the operator workstations in the main control room and is designed to the same standards.

The design basis for the remote shutdown workstation does not require the installation of safety related, dedicated, fixed-position displays, alarms, and controls.

9 of 11

One remote shutdown workstation is provided. The remote shutdown workstation contains controls for the safety-related equipment required to establish and maintain safe shutdown. Additionally, control of nonsafety-related components is available, allowing operation and control when ac power is available. ~~The remote shutdown workstation also receives inputs from the qualified data processing system for indication, similar to the main control room.~~

The controls, displays and alarms listed in Table 18.12.2-1 are retrievable from the remote shutdown workstation. Subsection 18.12.3 provides more discussion on the remote shutdown workstation displays, alarms, and controls.

The remote shutdown workstation is provided for use only following an evacuation of the main control room. No actions are anticipated from the remote shutdown workstation during normal, routine shutdown, refueling, or maintenance operations.

The remote shutdown workstation has sufficient communication circuits to allow the operator to effectively establish safe shutdown conditions. As detailed in subsection 9.5.2, communication is available between the following stations:

- Main control room
- Remote shutdown workstation
- Onsite technical support center
- Diesel generator local control station

Operator control capability at the remote shutdown workstation is normally disabled, and operator control functions are normally performed from workstations located inside the main control room; however, operator control capability can be transferred from the main control room workstations to the remote workstation if the control room requires evacuation. This operator control transfer capability can not be disabled by any single active failure coincident with the loss of offsite power.

The control transfer function is implemented by multiple transfer switches. Each individual transfer switch is associated with only a single safety-related or single nonsafety-related division. These switches are located behind an unlocked access panel. Entry into this access panel will result in alarms at the main control room and remote shutdown workstation. The access panel is located within a fire zone which is separate from the main control room. Actuation of these transfer switches results in additional alarms at the main control room and remote shutdown workstation, the activation of operator control capability from the remote workstation, and the deactivation of operator control capability from the main control room workstations. The ~~safety-related and nonsafety-related~~ operator displays located in the main control room and on the remote shutdown workstation are not affected by this control transfer function.

7.4.3.1.2 Controls at Other Locations

In addition to the controls and indicators provided at the remote shutdown workstation, the following controls are provided outside the main control room:

- Reactor trip capability at the reactor trip switchgear
- Turbine trip capability at the turbine

- Start/stop controls for the diesel generators, located at each diesel generator local control panel

7.4.3.1.3 Design Bases Information

According to GDC 19, the capability of establishing a shutdown condition and maintaining the station in a safe status in that mode is an essential function. The controls and indications necessary for this function are identified in subsection 7.4.2. To provide the availability of the remote shutdown workstation after control room evacuation, the following design features are provided:

- The remote shutdown workstation conforms with the guidelines provided by ANSI 58.6 1983 (Reference 1).

- ~~The remote shutdown workstation, including the safety-related controls and indications, is designed to withstand the safe shutdown earthquake with no loss of essential functions.~~

- The remote shutdown workstation achieves and maintains safe shutdown conditions from full power conditions and maintains safe shutdown conditions thereafter.

- The remote shutdown workstation achieves safe shutdown when offsite power is available and when offsite power is not available.

- The remote shutdown workstation operates safety-related systems, independent from the main control room.

- The remote shutdown workstation is designed for a single failure. When a random event, such as a fire, or an allowable technical specification maintenance results in one safety-related division being unavailable, a single failure in a redundant division is not postulated. When a random event other than fire causes a main control room evacuation, a coincident single failure in the systems controlled from the remote shutdown panel is considered.

- Access to the remote shutdown workstation is under administrative control.

7.4.3.2 Analysis

The analysis of the systems required for safe shutdown is provided in subsection 7.4.1. The following discussion is limited to the remote shutdown workstation.

Conformance to NRC General Design Criteria

General Design Criterion 19 - The remote shutdown workstation provides adequate controls and indications located outside the main control room to establish and maintain the reactor

and the reactor coolant system in a safe shutdown condition in the event that the main control room must be evacuated.

Conformance to NRC Regulatory Guides

Regulatory Guide 1.22 - The remote shutdown workstation is tested periodically during station operation.

Regulatory Guide 1.29 - ~~The~~ remote shutdown workstation is designed ~~to withstand the effects of a safe shutdown earthquake without loss of function or physical damage. The remote shutdown workstation is classified as seismic Category I. Selected instrumentation and control devices are not safety-related but are~~ as seismic Category II to prevent compromising the function of safety-related devices during or after a safe shutdown earthquake.

Conformance to IEEE 279-1971

which perform the control transfer function

The remote shutdown workstation and the design features which provide for the transfer of control capability from the main control room to the remote shutdown workstation conforms to applicable portions of IEEE 279-1971. The ~~control~~ circuits ~~at the remote shutdown workstation~~ are designed so that a single failure does not prevent maintaining safe shutdown. This is accomplished by redundant ~~controls for~~ the systems required for safe shutdown, using independent safety-related power divisions.

components in

To prevent interaction between the redundant systems, the redundant control channels are wired independently and are separated from each other. Nonsafety-related circuits available for (but not required for) safe shutdown are electrically isolated from safety-related circuits.

7.4.4 Combined License Information

This section has no requirement for information to be provided in support of the Combined License application.

7.4.5 References

1. ANSI 58.6 1983, "Criteria for Remote Shutdown for Light Water Reactors."

** TX CONFIRMATION REPORT **   AS OF FEB 11 '97 17:25   PAGE.01

WETSO/RM 468 EC EAST

	DATE	TIME	TO/FROM	MODE	MIN/SEC	PGS	CMD#	STATUS
28	02/11	17:18	516 344 4900	G3--S	06'55"	011		OK

** TX CONFIRMATION REPORT **   AS OF FEB 11 '97 17:14 PAGE.01

WETSO/RM 468 EC EAST

	DATE	TIME	TO/FROM	MODE	MIN/SEC	PGS	CMD#	STATUS
27	02/11	17:07	301 504 2222	G3--S	06'37"	011		OK

FAX to DINO SCALETTI

February 11, 1997

CC: Sharon or Dino, please make copies for: Bill Huffman
Ted Quay

Don Lindgren
Gordon Israelson
Ed Cummins
Bob Vijuk
Brian McIntyre

OPEN ITEM #158 (M5.2.5-14)

In my quest to make sure we have provided NRC with everything needed to prepare an FSER, I am researching open items from the smallest item number on. The relevant documentation related to Open Item #158 (M5.2.5-14) is attached. We provided this markup revision to subsection 5.2.5 on January 16, 1997. We believed that these changes resolved the concerns of item #158 and subsequent telephone conversations. It seems a reasonable request that NRC acknowledge receipt of the change. We request that NRC provide a definitive action for Westinghouse or provide direction to change the status of this item. We recommend "Action N". Thank you.

Jim

Jim Winters
412-374-5290

1 of 8

AP600 Open Item Tracking System Database: Executive Summary

Date: 2/11/97

Selection: [item no] between 158 And 158 Sorted by Item #

Item No.	Branch	DSER Section/ Question	Type	Title/Description Detail Status	Resp Engineer	(W) Status	NRC Status	Letter No. /	Date
158	NRR/SPLB	5.2.5	MTG-OI	M5.2.5-14 (REACTOR COOLANT PRESSURE BOUNDARY LEAKAGE) Discuss how the design satisfies RG 1.45, Positions C.1 through C.9. Closed - SSAR Revision 3 of 5.2.5 includes commitment for seismic Category I for the sump level monitor and the containment atmospheric monitor. Action W - per 12/2 telecon we need to add to SSAR "pointers" showing compliance with RegGuide 1.145. Action N - SSAR markup with "pointers" to RegGuide 1.45 faxed to NRC on 1/16/97.	Hutchings/BPC	Closed	Action W		

 Total Records: 1

5.2.4.4 Inspection Intervals

Inspection intervals are established as defined in Subarticles IWA-2400 and IWB-2400 of The ASME Code, Section XI. The interval may be extended by as much as one year so that inspections are concurrent with plant outages. It is intended that in-service examinations be performed during normal plant outages such as refueling shutdowns or maintenance shutdowns occurring during the inspection interval.

5.2.4.5 Examination Categories and Requirements

The examination categories and requirements are established according to Subarticle IWB-2500 and Table IWB-2500-1 of the ASME Code, Section XI. The preservice examinations comply with IWB-2200.

5.2.4.6 Evaluation of Examination Results

Examination results are evaluated according to IWA-3000 and IWB-3000, with flaw indications according to IWB-3400 and Table IWB-3410-1. Repair procedures, if required, are according to IWB-4000 of the ASME Code, Section XI.

5.2.4.7 System Leakage and Hydrostatic Pressure Tests

System pressure tests comply with IWA-5000 and IWB-5000 of the ASME Code, Section XI. These system pressure tests are included in the design transients defined in Subsection 3.9.1. This subsection discusses the transients included in the evaluation of fatigue of Class 1 components due to cyclic loads.

5.2.5 Detection of Leakage Through Reactor Coolant Pressure Boundary

The reactor coolant pressure boundary (RCPB) leakage detection monitoring provides a means of detecting and to the extent practical, identifying the source and quantifying the reactor coolant leakage. The detection monitors perform the detection and monitoring function in conformance with the requirements of General Design Criteria 2 and 30 and the recommendations of Regulatory Guide 1.45. Leakage detection monitoring is also maintained in support of the use of leak-before-break criteria for high-energy pipe in containment. See subsection 3.6.3 for the application of leak-before-break criteria.

Leakage detection monitoring is accomplished using instrumentation and other components of several systems. Diverse measurement methods including level, flow, and radioactivity measurements are used for leak detection. The equipment classification for each of the systems and components used for leak detection is generally determined by the requirements and functions of the system in which it is located. There is no requirement that leak detection and monitoring components be safety-related. See Figure 5.2-1 for the leak detection approach. The descriptions of the instrumentation and components used for leak detection and monitoring include information on the system.

To satisfy position 1. of Regulatory Guide 1.45

Reactor coolant pressure boundary leakage is classified as either identified or unidentified leakage. Identified leakage includes:

- Leakage from closed systems such as pump gasket or reactor vessel seal leaks that are captured and conducted to a sump or collecting tank
- Leakage into auxiliary systems and secondary systems (intersystem leakage) (This leakage is not considered to be part of the 10 gpm limit identified leakage in the bases of the technical specification 3.4.8. This additional leakage must be considered in the evaluation of the reactor coolant inventory balance.)

Other leakage is unidentified leakage.

5.2.5.1 Collection and Monitoring of Identified Leakage

Identified leakage other than intersystem leakage is collected in the reactor coolant drain tank. The reactor coolant drain tank is a closed tank located in the reactor cavity in the containment. The tank vent is piped to the gaseous radwaste system to prevent release of radioactive gas to the containment atmosphere. The liquid level in the reactor coolant drain tank and total flow pumped out of the reactor coolant drain tank are used to calculate the identified leakage rate. These parameters are available in the main control room. The reactor coolant drain tank, pumps, and sensors are part of the liquid radwaste system. The following sections outline the various sources of identified leakage other than intersystem leakage.

For positions 1 and 7. of Regulatory Guide 1.45

5.2.5.1.1 Valve Stem Leakoff Collection

Valve stem leakoff connections are not provided in the AP600.

5.2.5.1.2 Reactor Head Seal

The reactor vessel flange and head flange are sealed by two concentric seals. Seal leakage is detected by two leak-off connections: one between the inner and outer seal, and one outside the outer seal. These lines are combined in a header before being routed to the reactor coolant drain tank. An isolation valve is installed in the common line. During normal plant operation, the leak-off valves are aligned so that leakage across the inner seal drains to the reactor coolant drain tank.

A surface-mounted resistance temperature detector installed on the bottom of the common reactor vessel seal leak pipe provides an indication and high temperature alarm signal in the main control room indicating the possibility of a reactor pressure vessel head seal leak. The temperature detector and drain line downstream of the isolation valve are part of the liquid radwaste system.

The reactor coolant pump closure flange is sealed with a welded canopy seal and does not require leak-off collection provisions.

Leakage from other flanges is discussed in subsection 5.2.5.3, Collection and Monitoring of Unidentified Leakage.

5.2.5.1.3 Pressurizer Safety Relief Valves

Temperature is sensed downstream of each pressurizer safety relief valve by a resistance temperature detector on the discharge piping upstream of the rupture disc. High temperature indications (alarms in the main control room) identify a reduction of coolant inventory as a result of seat leakage through a pressurizer safety valve. These detectors are part of the reactor coolant system. This leakage is drained to the reactor coolant drain tank during normal plant operation and vented to containment atmosphere during accident conditions that rupture the disc. This identified leakage is measured by the change in level of the reactor coolant drain tank.

5.2.5.1.4 Reactor Coolant Pump Drain

Leakage from the reactor coolant pump drain is directed to the reactor coolant drain tank. This identified leakage is measured by the change in level in the reactor coolant drain tank.

5.2.5.1.5 Other Leakage Sources

In the course of plant operation, various minor leaks of the reactor coolant pressure boundary may be detected by operating personnel. If these leaks can be subsequently observed, quantified, and routed to the containment sump, this leakage will be considered identified leakage.

5.2.5.2 Intersystem Leakage Detection

In accordance with position 4 of Regulatory Guide 1.45,

Substantial intersystem leakage from the reactor coolant pressure boundary to other systems is not expected. However, possible leakage points across passive barriers or valves and their detection methods are considered. Auxiliary systems connected to the reactor coolant pressure boundary incorporate design and administrative provisions that limit leakage. Leakage is detected by increasing auxiliary system level, temperature, flow, or pressure, by lifting the relief valves or increasing the values of monitored radiation in the auxiliary system.

The normal residual heat removal system and the chemical and volume control system, which are connected to the reactor coolant system, have potential for leakage past closed valves. For additional information on the control of reactor coolant leakage into these systems, see subsections 5.4.7 and 9.3.6 and the intersystem LOCA discussion in subsection 1.9.5.1.

5.2.5.2.1 Steam Generator Tubes

An important potential identified leakage path for reactor coolant is through the steam generator tubes into the secondary side of the steam generator. Identified leakage from the steam generator primary side is detected by one, or a combination, of the following:

- High condenser air removal discharge radioactivity, as monitored and alarmed by the turbine island vent discharge radiation monitor
- Steam generator secondary side radioactivity, as monitored and alarmed by the steam generator blowdown radiation monitor
- Secondary side radioactivity, as monitored and alarmed by the main steam line radiation monitors
- Radioactivity, boric acid, or conductivity in condensate as indicated by laboratory analysis

Details on the radiation monitors are provided in Section 11.5, Radiation Monitoring.

5.2.5.2.2 Component Cooling Water System

Leakage from the reactor coolant system to the component cooling water system is detected by the component cooling water system radiation monitor, by increasing surge tank level, by high flow downstream of selected components, or by some combination of the preceding. Refer to Section 11.5, Radiation Monitoring, and subsection 9.2.2, Component Cooling Water System.

5.2.5.3 Collection and Monitoring of Unidentified Leakage

in accordance with position 3 of Regulatory Guide 1.45

To detect unidentified leakage inside containment, the following diverse methods may be utilized to quantify and assist in locating the leakage:

- Containment Sump Level
- Reactor Coolant System Inventory Balance
- Containment Atmosphere Radiation

Other methods that can be employed to supplement the above methods include:

- Containment Atmosphere Pressure, Temperature, and Humidity
- Visual Inspection

The reactor coolant system is an all-welded system, except for the connections on the pressurizer safety valves, reactor vessel head, pressurizer and steam generator manways, and reactor vessel head vent, which are flanged. During normal operation, variations in airborne radioactivity, containment pressure, temperature, or specific humidity above the

normal level signify a possible increase in unidentified leakage rates and alert the plant operators that corrective action may be required. Similarly, increases in containment sump level signify an increase in unidentified leakage. The following sections outline the methods used to collect and monitor unidentified leakage.

5.2.5.3.1 Containment Sump Level Monitor

Leakage from the reactor coolant pressure boundary and other components not otherwise identified inside the containment will condense and flow by gravity via the floor drains and other drains to the containment sump.

in accordance with position 6 of Regulatory Guide 1.45

A leak in the primary system would result in reactor coolant flowing into the containment sump. Leakage is indicated by an increase in the sump level. The containment sump level is monitored by two seismic Category I level sensors. The level sensors are powered from a safety-related Class 1E electrical source. These sensors remain functional when subjected to a safe shutdown earthquake in conformance with the guidance in Regulatory Guide 1.45. The containment sump level and sump total flow sensors located on the discharge of the sump pump are part of the liquid radwaste system.

Failure of one of the level sensors will still allow the calculation of a 0.5 gpm in-leakage rate within 1 hour. The data display and processing system (DDS) computes the leakage rate and the plant control system (PLS) provides an alarm in the main control room if the average change in leak rate for any given measurement period exceeds 0.5 gpm for unidentified leakage. Unidentified leakage is the total leakage minus the identified leakage. The leakage rate algorithm subtracts the identified leakage directed to the sump.

To satisfy positions 2 and 5 of Regulatory Guide 1.45,

The measurement interval must be long enough to permit the measurement loop to adequately detect the increase in level that would correspond to 0.5 gpm leak rate, and yet short enough to ensure that such a leak rate is detected within an hour. The measurement interval is less than or equal to 1 hour.

When the sump level increases to the high level setpoint, one of the sump pumps automatically starts to pump the accumulated liquid to the waste holdup tanks in the liquid radwaste system. The sump discharge flow is integrated and available for display in the control room, in accordance with position 7 of Regulatory Guide 1.45.

Procedures to identify the leakage source upon a change in the unidentified leakage rate into the sump include the following:

- Check for changes in containment atmosphere radiation monitor indications.
- Check for changes in containment humidity, pressure, and temperature.
- Check makeup rate to the reactor coolant system for abnormal increases.

- Check for changes in water levels and other parameters in systems which could leak water into the containment, and

- Review records for maintenance operations which may have discharged water into the containment.

5.2.5.3.2 Reactor Coolant System Inventory Balance

Reactor coolant system inventory monitoring provides an indication of system leakage. Net level change in the pressurizer is indicative of system leakage. Monitoring net makeup from the chemical and volume control system and net collected leakage provides an important method of obtaining information to establish a water inventory balance. An abnormal increase in makeup water requirements or a significant change in the water inventory balance can indicate increased system leakage.

The reactor coolant system inventory balance is a quantitative inventory or mass balance calculation. This approach allows determination of both the type and magnitude of leakage. Steady-state operation is required to perform a proper inventory balance calculation. Steady-state is defined as stable reactor coolant system pressure, temperature, power level, pressurizer level, and reactor coolant drain tank and in-containment refueling water storage tank levels. The reactor coolant inventory balance is done on a periodic basis and when other indication and detection methods indicate a change in the leak rate.

The mass balance involves isolating the reactor coolant system to the extent possible and observing the change in inventory which occurs over a known time period. This involves isolating the systems connected to the reactor coolant system. System inventory is determined by observing the level in the pressurizer. Compensation is provided for changes in plant conditions which affect water density. The change in the inventory determines the total reactor coolant system leak rate. Identified leakages are monitored (using the reactor coolant drain tank) to calculate a leakage rate and by monitoring the intersystem leakage. The unidentified leakage rate is then calculated by subtracting the identified leakage rate from the total reactor coolant system leakage rate.

Since the pressurizer inventory is controlled during normal plant operation through the level control system, the level in the pressurizer will be reasonably constant even if leakage exists. The mass contained in the pressurizer may fluctuate sufficiently, however, to have a significant effect on the calculated leak rate. The pressurizer mass calculation includes both the steam and water mass contributions.

Changes in the reactor coolant system mass inventory are a result of changes in liquid density. Liquid density is a strong function of temperature and a lesser function of pressure. A range of temperatures exists throughout the reactor coolant system all of which may vary over time. A simplified, but acceptably accurate, model for determining mass changes is to assume all of the reactor coolant system is at $T_{Average}$.

The inventory balance calculation is done by the data display and processing system with additional input from sensors in the protection and safety monitoring system, chemical and volume control system, and liquid radwaste system. The use of components and sensors in systems required for plant operation provides conformance with the regulatory guidance in Regulatory Guide 1.45 that leak detection should be provided following seismic events that do not require plant shutdown.

of position 6

5.2.5.3.3 Containment Atmosphere Radioactivity Monitor

Leakage from the reactor coolant pressure boundary will result in an increase in the radioactivity levels inside containment. The containment atmosphere is continuously monitored for airborne gaseous radioactivity. Air flow through the monitor is provided by the suction created by a vacuum pump. Gaseous and N_{13}/F_{18} concentration monitors indicate radiation concentrations in the containment atmosphere.

The gas channel can respond rapidly to reactor coolant pressure boundary leakage. N_{13} is a neutron activation product which is proportional to power levels. Additionally, N_{13} has a relatively short half life and consequently will reach equilibrium rapidly. An increase in activity inside containment would therefore indicate a leakage from the reactor coolant pressure boundary. Based on the concentration of N_{13}/F_{18} and the power level, reactor coolant pressure boundary leakage can be estimated.

position 6 of Regulatory Guide 1.45

The N_{13}/F_{18} monitoring system has a high sensitivity when the reactor is operating at a power range higher than 20 percent. The N_{13} monitor is seismic Category I. Conformance with the guidance that leak detection should be provided following seismic events that do not require plant shutdown is provided by the seismic Category I classification. Safety-related Class 1E power is not required since loss of power to the radiation monitor is not consistent with continuing operation following an earthquake. Above 20 percent power level, in one hour, a leak less than 0.5 gpm can be detected. Operating experience has indicated the average long-term leakage (from sampling losses, collected leakoffs, and unidentified leakage to the containment) from the reactor coolant system ranges between 0.1 and 0.3 gpm. The N_{13} concentration will increase by at least 25 percent above an existing 0.1 gpm leakage background and almost 10 percent for an existing 0.3 gpm leakage. Both increases are well within the sensitivity of the N_{13}/F_{18} monitor capabilities.

Radioactivity concentration indication and alarms for loss of sample flow, high radiation, and loss of indication are provided. Sample collection connections permit sample collection for laboratory analysis. The radiation monitor can be calibrated during power operation.

5.2.5.3.4 Containment Pressure, Temperature and Humidity Monitors

Reactor coolant pressure boundary leakage increases containment pressure, temperature, and humidity, values available to the operator through the plant control system.

An increase in containment pressure is an indication of increased leakage or a high energy line break. Containment pressure is monitored by redundant Class 1E pressure transmitters. For additional discussion see subsection 6.2.2, Passive Containment Cooling System.

The containment average temperature is monitored using temperature instrumentation at the inlet to the containment fan cooler as an indication of increased leakage or a high energy line break. This instrumentation as well as temperature instruments within specific areas including steam generator areas, pressurizer area, and containment compartments are part of the containment recirculation cooling system.

An increase in the containment average temperature combined with an increase in containment pressure indicate increased leakage or a high energy line break. The individual compartment area temperatures can assist in identifying the location of the leak.

Containment humidity is monitored using temperature-compensated humidity detectors which determine the water-vapor content of the containment atmosphere. An increase in the containment atmosphere humidity indicates release of water vapor within the containment. The containment humidity monitors are part of the containment leak rate test system.

The humidity monitors supplement the containment sump level monitors and are most sensitive under conditions when there is no condensation. A rapid increase of humidity over the ambient value by more than 10 percent is indication of a probable leak.

Containment pressure, temperature and humidity can assist in identifying and locating a leak. They are not relied on to quantify a leak.

5.2.5.4 Safety Evaluation

Leak detection monitoring has no safety-related function. Therefore, the single failure criterion does not apply and there is no requirement for a nuclear safety evaluation. The containment sump level monitors and the containment atmosphere monitor are seismic Category I. The components used to calculate reactor coolant system inventory balance are both safety-related and nonsafety-related components. The containment sump level monitors are powered from the Class 1E dc and UPS system (IDS). Measurement signals are processed by the data display and processing system and the plant control system (PLS).

5.2.5.5 Tests and Inspections

To satisfy position 8 of Regulatory Guide 1.45,

Periodic testing of leakage detection monitors verifies the operability and sensitivity of detector equipment. These tests include installation calibrations and alignments, periodic channel calibrations, functional tests, and channel checks in conformance with regulatory guidance.

5.2.5.6 Instrumentation Applications

satisfy position 7 of Regulatory Guide 1.45 and

The parameters tabulated below are provided in the main control room to allow operating personnel to monitor for indications of reactor coolant pressure boundary leakage. The containment sump level, containment atmosphere radioactivity, reactor coolant system inventory balance, and the flow measurements are provided as gallon per minute leakage equivalent.

Parameter	System(s)	Alarm or Indication
Containment sump level and sump total flow	WLS	Both
Reactor coolant drain tank level and drain tank total flow	WLS	Both
Containment atmosphere radioactivity	PSS	Both
Reactor coolant system inventory balance parameters	PCS, PXS, RCS, VCS, WLS	Both
Containment humidity	VUS	Indication
Containment atmospheric pressure	PCS	Both
Containment atmosphere temperature	VCS	Both
Reactor vessel head seal leak temperature	WLS	Both
Pressurizer safety relief valve leakage temperature	RCS	Both
Reactor coolant pump flange leakoff temperature	RCS	Both
Steam generator blowdown radiation	BDS	Both
Turbine island vent discharge radiation	TDS	Both
Component cooling water radiation	CCS	Both
Main steam line radiation	SGS	Both
Component cooling water surge tank level	CCS	Both

5.2.5.7 Technical Specification

which satisfy position 9 of Regulatory Guide 1.45

Limits for identified and unidentified reactor coolant leakage are identified in the technical specifications, Chapter 16. LCO 3.4.8 addresses leakage limits. LCO 3.4.10 addresses leak detection instrument requirements.

** TX CONFIRMATION REPORT ** AS OF FEB 11 '97 15:07 PAGE.01

AP600 DESIGN CERT

	DATE	TIME	TO/FROM	MODE	MIN/SEC	PGS	STATUS
01	2/11	15:00	301 504 2300	G3--S	07'00	11	OK

FAX to DINO SCALETTI

February 11, 1997

CC: Sharon or Dino, please make copies for: Tom Kenyon
Ted Quay

Robin Nydes
Steve Kerch
Ed Cummins
Bob Vijuk
Brian McIntyre

OPEN ITEMS FOR CHAPTER 13

This is a background package for the remaining open items for Chapter 13. Chapter 13 is of interest because by our joint NRC/W schedule, the FSER for this chapter should be turned into Projects by the end of February. Attached is a copy of the OITS entries for the 10 items showing "Action W" in "NRC Status". For two (1222 and 2033) we believe we owe additional information. A letter will be issued from Westinghouse by the end of February, which should resolve these two items. The other eight items (405, 1225, 1226, 1227, 1228, 1231, 1232 and 2034) have been answered in a variety of documents. These include the Security Plan, and SSAR Chapter 18. For six items of these items (405, 1227, 1228, 1231, 1232 and 2034), resolution should have been included in the Security Plan, which was issued on 7/8/97 (over six months ago). It seems a reasonable request that NRC acknowledge receipt of this information. Our records show no outstanding Westinghouse action on this Chapter (13), except for items 1222 and 2033, and we request that NRC provide a definitive action for Westinghouse or provide direction to change the status of the other eight items. We recommend "Action N". Thank you.

Jim

Jim Winters
412-374-5290

AP600 Open Item Tracking System Database: Executive Summary

Date: 2/11/97

Selection: [nrc st code]='Action W' And [DSER Section] like '13*' Sorted by Item #

Item No	Branch	DSER Section/ Question	Type	Title/Description Detail Status	Resp Engineer	(W) Status	NRC Status	Letter No. /	Date
405	NRR/TSGB	13.6	MTG-OI	(October 25, 1994 Security Meeting) Provide a vulnerabilty analysis Vulnerability analysis will be submitted with revised Security plan AP600 Security Design Vulnerability Analysis submitted 7/8/96	McIntyre, B.	Closed	Action W		7/8/96
1222	NRR/PERB	13.3-5	DSER-OI	Westinghouse should demonstrate the ability of the main control room to support the appropriate number of staff when the TSC is unavailable. Action W - The ERG development will provide the task analysis necessary to determine the number of staff required to support main control room poeration and the required TSC functions. The capability of the MRC to support this staffing level will be provided. Resolved - Per DCP/NRC0589, this item will be closed with submittal of th at-power ERG's. This is an MMIS item. Westinghouse to write a letter to the NRC explaining when the TSC would be unavailable and that the staff would not be moved to the MCR. Expect letter by 12/30. This will address items 1222 and 2033. rkn 12/3 Letter not complete by 12/30, expect by Jan 31, 1997. rkn 1/15/97.	MMIS/Kerch	Action W	Action W	NSD-NRC-96-4805	
1225	NRR/HHFB	13.5.1-1	DSER-OI	Westinghouse should add COL Action Item 13.5.1-1 to the SSAR for the COL applicant to address administrative procedures for the plant. Closed - A COL information item was added to Chapter 13.5, Revision 3, of the SSAR to address administrative proceedures for the plant 6/10/96 - NRC believes issue remains open pending resolution of the procedures section of Chapter 18.	Winters, J.	Closed	Action W		
1226	NRR/HHFB	13.5.2-1	DSER-OI	Westinghouse should add COL Action Item 13.5.2-1 to the SSAR for the COL applicant to address operating and maintenance procedures for the plant. Closed - A COL information item was added to Section 13.5, Revision 3, of the SSAR to address the development of operating and maintainance procedures for the plant. 6/10/96 - NRC believes issue remains open pending resolution of the procedures section of Chapter 18.	Winters, J.	Closed	Action W		
1227	NRR/PSGB	13.6-1	DSER-OI	Review of Westinghouse's revised security plan. Closed - see SSAR section 13.6.1 ans white paper sent to NRC on 03/21/95, "AP600 Performance Based Security Design Approach, SES-ASR-001, Rev A." Security desogn report submitted 8/20/95 Revision 2 of the AP600 Security Design Report submitted 7/8/96	McIntyre, B.	Closed	Action W		7/8/96
1228	NRR/PSGB	13.6.3.2-1	DSER-OI	Westinghouse should provide an analysis of the vulnerabilities of the design to sabotage. Action W - Westinghouse will provide a vulneravibility analysis as a part of the preliminary security plan in August 1995 The AP600 Security Design Vulnerability report was submitted 7/8/96	McIntyre, B.	Closed	Action W		7/8/96

Item No	Branch	DSER Section/ Question	Type	Title/Description Detail Status	Resp Engineer	(W) Status	NRC Status	Letter No. /	Date
1231	NRR/PSGB	13.6.7.2-1	DSER-OI		McIntyre, B.	Closed	Action W		7/8/96
				Westinghouse should re-evaluate an internal threat in the security plan.					
				Action W - SECY-93-326 will be addressed in the revised security plan (DSER-OI 13.6-1) Closed - The security report, revision 2, was submitted 7/8/96					
1232	NRR/PSGB	13.6.10-1	DSER-OI		McIntyre, B.	Closed	Action W		7/8/96
				Westinghouse should evaluate the interdiction capability of the security response force.					
				Action W - The capabilities of the security response force will be addressed as a part of the revised security plan (DSER-OI 13.6-1) Revision 2 of the AP600 Security Design Report summitted 7/8/96					
2033	NRR/PERB	13.	DSER-OI50		MMIS/Kerch	Action W	Action W		
				37. Habitability of Technical Support Center The staff is concerned with the acceptability of the habitability requirements specified by Westinghouse for the TSC under accident conditions. Westinghouse has proposed that a detailed task analysis be performed post-certification to determine disposition of TSC staff when the facility is not habitable. The staff believes that this analysis should be performed pre-certification. (See DSER Open Item 13.3-4)					
				Need to determine what is needed to close this item. rkn 10/16/96 Met on 10/24 (Wills, Schulz, Kerch, Nydes) and developed plan to research this then discuss with NRC. Appears to be a RTNSS-related item. To close this item, the letter being written for OITS item 1222 will close this (Kerch- author, Wills and Schulz to review). rkn 12/2					
2034	NRR/PSGB	13.	DSER-OI50		McIntyre	Closed	Action W		7/8/96
				38. Proposed AP600 Security Plan The AP600 has a proposed Security Plan concept that essentially eliminates the protected area. One briefing of the NRC staff has been held. A formal submittal is required.					
				Discussed at 2/9/95 SMM) W to submit a formal security plan revision. 5/2/95 Status: Discussed at April 1995 SMM. Security plan revision submitted. Staff met w/W on March 27 & April 12, 1995 to discuss review results. (Discussed at 4/4/95 SMM) 38. Safeguards The staff indicated that Westinghouse's proposal on safeguards seemed reasonable conceptually. However, the staff indicated that there may be some non-safety-related components or systems that are RTNSS-significant that should be considered. Active A meeting will be setup to discuss these issues in detail. (5/2/95 Status) Action W - Meeting held 4/12/95. RTNSS important systems are in the process of being evaluated for regulatory oversight. Closed - Security Plan submitted 8/20/95 Security Plan Revision 2 submitted 7/8/96					

FAX to DINO SCALETTI

February 11, 1997

CC: Sharon or Dino, please make copies for: Diane Jackson
Ted Quay

Don Lindgren
Ed Johnson
Ed Cummins
Bob Vijuk
Brian McIntyre

OPEN ITEMS FOR SSAR APPENDIX 3F

This is a background package for the remaining open items for SSAR Appendix 3F. SSAR Appendix 3F is of interest because by our joint NRC/W schedule, the FSER for this section should be turned into Projects by the middle of March. Currently, our records show no outstanding Westinghouse action required for Appendix 3F. We request NRC confirm this status. Thank you.

Jim

Jim Winters
412-374-5290

FAX to DINO SCALETTI

February 11, 1997

CC: Sharon or Dino, please make copies for: Diane Jackson
Ted Quay

Richard Orr
Ed Cummins
Bob Vijuk
Brian McIntyre

OPEN ITEMS FOR SSAR SECTION 3.7

This is a background package for the remaining open items for SSAR section 3.7. SSAR section 3.7 is of interest because by our joint NRC/W schedule, the FSER for this section should be turned into Projects by the middle of March. There are 7 Open Items with NRC Status of Action W. Five (5) of these items (623, 649, 662, 664 and 668) also have Action W in their W Status. Westinghouse owes information as identified in the OITS and as a result of our (NRC/W) December meeting. Westinghouse believes the other two (2) items (628 and 1885) are resolved by letter NSD-NRC-97-4956 of 1/28/97, which discusses our position on shallow soil sites. In addition, we have not received the NRC letter report on the December meeting covering section 3.7 and others. It may have additional status information to be incorporated into OITS when issued. We request expeditious issue of this letter report so that we can properly support NRC development of FSER Section 3.7. Currently, our records show no additional outstanding Westinghouse action required for items 628 and 1885 and we request that NRC provide a definitive action for Westinghouse or provide direction to change the status of these items. We recommend "Action N". Thank you.

Jim

Jim Winters
412-374-5290

AP600 Open Item Tracking System Database: Executive Summary

Date: 2/11/97

Selection: [nrc st code]='Action W' And [DSER Section] like '3.7*' Sorted by Item #

Item No.	Branch	DSER Section/ Question	Type	Title/Description Detail Status	Resp Engineer	(W) Status	NRC Status	Letter No. /	Date
623	NRR/ECGB	3.7.1-2	DSER-OI		Orr	Action W	Action W	NTD-NRC-95-4464	

Westinghouse should revise the SSAR to use acceptable damping values for cable tray and HVAC systems.

Reopened in telecon 11/26/96. W position defined in meeting in June, 1996. NRC requires further justification for electrical cable tray damping which is higher than recommended in recent BNL report. See NRC letter of 12/9/96.

Item No.	Branch	DSER Section/ Question	Type	Title/Description Detail Status	Resp Engineer	(W) Status	NRC Status	Letter No. /	Date
628	NRR/ECGB	3.7.1.1-1	MTG-OI		Orr / NRCSEIS	Closed	Action W	NSD-NRC-97-4956	

Westinghouse should commit, in the SSAR, that the potential plant site needs to meet the identified bounding parameters.

Closed - The shallow soil site (shear wave velocity < 1000 fps, depth to bedrock < 100 ft.) is excluded by the requirement that the shear wave velocity be greater than 1000 fps.
Clarification requested in NRC letter of 4/5/96 - Requirement for site specific time history and PSD criteria added in SSAR 2.5.4.5.5 which covers site specific seismic input.

Action W - Resolve the differences in NRC and Westinghouse position on site qualification. The shallow soil site (shear wave velocity < 1000 fps, depth to bedrock < 100 ft.) is excluded by the requirement that the shear wave velocity be greater than 1000 fps. COL must demonstrate that site is acceptable for the AP600 design. NRC position is that site specific analysis must use ZPA of 0.3 for SSE. Westinghouse position is that site specific earthquake should be used for sites outside the interface.

Discussed at NRC Management meeting 7-17-96 - further technical discussions required to clarify issue.
Westinghouse position identified in letter NSD-NRC-96-4804, dated 8/26/96.
NRC Status: Action W - The SSAR proposal presented at the December 1996 meeting does not satisfy the staff position. Westinghouse will respond. (12/16/96)
See NRC letter of 12/9/96.
Closed - Westinghouse provided a position on shallow soil sites in NSD-NRC-97-4956 of 1/28/97.

Item No.	Branch	DSER Section/ Question	Type	Title/Description Detail Status	Resp Engineer	(W) Status	NRC Status	Letter No. /	Date
649	NRR/ECGB	3.7.2.4-7	DSER-OI		Orr / BPC / NRCSEIS	Action W	Action W	NSD-NRC-96-4825	

Westinghouse should evaluate the localized through-soil SSI effect of non-seismic Category I structures on the design of embedded seismic Category I walls and the potential for pounding between structures. Additional issues identified in NRC Letter dated July 18, 1996.

Results of the 2D SASSI analyses to determine the loads on the exterior walls below grade are included in SSAR Revision 7, Appendix 2C. Potential for pounding between buildings is addressed in SSAR 3.7.2.8.
Action W- See meeting notes 7/18/96. Determine the effect of adjacent non-seismic Category I buildings on the lateral pressure on nuclear island walls below grade due to horizontal seismic ground motions; justify that existing analyses adequately represent the gap between the buildings.
Closed - Response provided in NSD-NRC-96-4825. Refined model has minimal effect on results.

Relative deflections from SASSI 2D analyses are available for review in December, 1996 meeting.

NRC Status: Acton W - The staff reviewed the draft SSAR 3.7.2.8 in the December 1996 meeting. The draft is incomplete. Westinghouse does not have a means, such as a COL action item, to demonstrate that non-Category I structures adjacent to NI do not interact with NI. (12/16/96)

Item No.	Branch	DSER Section/ Question	Type	Title/Description Detail Status	Resp Engineer	(W) Status	NRC Status	Letter No. / Date
662	NRR/ECGB	3.7.2.8-5	DSER-OI		Orr / SCS/ NRCSEIS	Action W	Action W	NTD-NRC-95-4433

Westinghouse should acceptably address two issues related to the design of bracing systems of structures adjacent to the NI structures.

Westinghouse to provide following additional information on turbine building:
Demonstrate that collapse of floors between the turbine building and the nuclear island in event of SSE will not impair safety functions of the nuclear island.
NRC Letter dated July 18, 1996 - B.3. Demonstrate that the turbine building frames can withstand with concentric bracing o seismic ground acceleration of 0.3g. Establish post-construction verification of structural members, connections, dimensions, etc to provide that these are consistent with the design.
Demonstrate that the turbine building foundation will not pound the nuclear island wall at the foundation level.
Classification of turbine building is discussed in letter NSD-NRC-96-4854, dated 10/22/96. Further discussion is needed on behavior of K versus X bracing.
NRC Status: Action W - The staff reviewed the Westinghouse October 21, 1996, letter response. Westinghouse provided diagrams in the December 1996 meeting. A telephone conference call is necessary. Westinghouse needs to classify the turbine building as Category II or use an eccentric bracing system in the design. (12/16/96)
Action W - See NRC letter of 12/9/96.

Item No.	Branch	DSER Section/ Question	Type	Title/Description Detail Status	Resp Engineer	(W) Status	NRC Status	Letter No. / Date
664	NRR/ECGB	3.7.2.8-7	DSER-OI		Orr/Lapay	Action W	Action W	NTD-NRC-95-4464

Westinghouse should demonstrate and document in the SSAR, for the evaluation of seismic margin, that both seismic Category II and non-seismic structures can withstand an earthquake up to 0.5g without collapse.

Seismic Category II and nonseismic structures are not directly included in the seismic margins assessment. The seismic design of these structures is described in Subsection 3.7.2.8 of the SSAR.
Seismic Category II building structures are designed for the safe shutdown earthquake using the same methods as are used for seismic Category I structures. The seismic Category II structures are the annex building and the stair tower to the shield building roof. These would have seismic capability similar to the seismic Category I structures. Therefore, it is expected that they will withstand an earthquake greater than 0.50g as shown in the seismic margin assessment for the seismic Category I structures in Appendix H of the PRA report, revison 1.
Nonseismic structures are generally analyzed and designed for seismic loads according to the Uniform Building Code requirements for Zone 2A with an Importance Factor of 1.25. The radwaste and turbine buildings are nonseismic structures. As described in Subsection 3.7.2.8, collapse of the radwaste building would not would impair the integrity of the reinforced concrete nuclear island.

As described in Subsection 3.7.2.8, the major structure of the turbine building is separated from the nuclear island by approximately eighteen feet and the seismic design of the turbine building has been upgraded to UBC Zone 3 with an Importance Factor of 1.0 in order to provide margin against collapse during the safe shutdown earthquake. The turbine building may not withstand the 0.5g earthquake without potential local collapse. However, it is separated from the nuclear island, and the equipment essential to safe shutdown is well protected by the thick concrete walls, floors and roof slab of the nuclear island. Hence the failure of the turbine building is not considered in the seismic margins assessment since its collapse is unlikely to impair the integrity of equipment essential to safe shutdown.

Staff update provided during 8/17/95 meeting: Statement made above that "collapse of the radwaste building would not impair the integrity of the reinforced concrete nuclear island" is a judgemental conclusion. Also, floors between T-building and NI may impact the safety of NI.
Discussion will be added in seismic margin report on why collapse of the nonseismic buildings is not expected to result in core damage.
Meeting 7/17-20/96 - Westinghouse evaluation of collapse of turbine building to be reviewed during seismic margins meeting.
Closed in chapter 3.7: transferred to seismic margins review
Action W - See NRC letter of 12/9/96.
Westinghouse to document that turbine building collapse does not lead to core damage.

Item No.	Branch	DSER Section/ Question	Type	Title/Description Detail Status	Resp Engineer	(W) Status	NRC Status	Letter No. /	Date
668	NRR/ECGB	3.7.2.12-1	DSER-OI		Orr/BPC	Action W	Action W		
				Westinghouse should compare the results from the response spectrum analysis method to those of the modal time-history analysis method.					
				Preliminary comparison of results from the response spectrum analysis method vs. the modal time-history analysis was presented in meeting with NRC on 6/13/95. Comparison is included in SSAR Revision 7. Editorial changes included in SSAR revision 9. Reopened in telecon 11/26/96. Westinghouse to provide additional comparisons in meeting in December, 1996. NRC Status: Action W - Westinghouse to reconciliation differences in time history and response spectrum results. (12/16/96)					
1885	NRR/ECGB	3.7.2.16-1	DSER-COL		Orr	Closed	Action W	NSD-NRC-97-4956	
				3.7.2.16-1 The COL applicant should per[illegible]rm an analysis and evaluation using the design basis earthquake ground motion and plant specific site conditions to confirm the design adequa[illegible]f the AP600 design.					
				SSAR Subsection 2.5 provides the inform[illegible]ion requirements for the COL applicant. Site-specific soil structure interaction analyses may be performed by the Combined License applicant to demonstrate acceptability by comparison of floor response spectra. These analyses would use the site specific soil conditions and safe shutdown earthquake. The COL applicant requirement is included in SSAR Section 2.5.4.5.5. See open item 3.7.1.1-1 for details and issue B.1 of NRC Letter dated July 18, 1996. NRC Status: Action W - This COL action is connected to DSER 3.7.1.1-1 (OITS# 628). The current SSAR proposal presented at the December 1996 meeting does not satisfy the staff position. (12/16/96) Closed - Westinghouse provided a position on shallow soil sites in NSD-NRC-97-4956 of 1/28/97.					

MEMORANDUM

WESTINGHOUSE FORM 2478 K

TO NRC (LOCATION) DEPT. D. JACKSON (NAME) 2/7/97 (DATE)

SUBJECT SSAR LBB CHANGES.

DIANE,

PLEASE PASS ~~T~~HIS PACKAGE ALONG TO THE LBB + PIPING REVIEWERS. THESE CHANGES ARE THE RESULT OF REMOVING MAIN FEEDWATER AND THE 4" PRESSURIZER SPRAY LINE FROM LBB. THERE IS ALSO A SMALL CHANGE IN THE DESCRIPTION OF THE BREAK EXCLUSION ZONES. THE EFFORT TO IDENTIFY THE BREAK LOCATIONS RESULTED IN ADDITIONAL CHANGES IN TABLE 3.6-3

FROM

LOCATION DEPT. NAME

3.6 Protection Against the Dynamic Effects Associated with the Postulated Rupture of Piping

The effects of a postulated pipe rupture in the AP600 are of several types. This section considers the effects that are localized to the area of the break and are a result of the dynamic effects of the pipe rupture including jet impingement, pipe whip, subcompartment pressurization, and fluid system decompression. This section describes the evaluation of the potential for and effects of these dynamic effects. It describes measures taken to protect systems and equipment from dynamic effects of pipe rupture when necessary. This section also considers the effects of spray wetting and flooding from pipe ruptures and cracks.

Chapters 6 and 15 discuss the response of the system to changes in flow and pressure and loss of coolant and the response of the containment to the pressure and temperature changes. Pressure due to a break in a high energy line in the auxiliary building is vented into an adjacent building or to the atmosphere. The design transients listed in subsection 3.9.1 are used in evaluating the components of the reactor coolant system for effects due to internal pressure and temperature changes from postulated accidents. Section 3.11 discusses the qualification of the equipment required to function in the adverse environmental conditions including temperature, humidity, pressure, and chemical consequences.

Pipe failure protection is provided according to the requirements of 10 CFR 50, Appendix A, General Design Criterion 4. In the event of a high- or moderate-energy pipe failure within the plant, adequate protection is provided so that essential structures, systems, or components are not impacted by the adverse effects of postulated piping failure. Essential systems and components are those required to shut down the reactor and mitigate the consequences of the postulated piping failure. Nonsafety-related systems, including those that are determined to be important by the regulatory treatment of nonsafety-related systems (RTNSS) process and defence in-depth systems, are not required to be protected from the dynamic and environmental effects associated with the postulated rupture of piping. See subsection 1.9.5.3 for a discussion of the regulatory treatment of nonsafety systems in the AP600. Protection against pipe rupture is not an RTNSS important mission for nonsafety-related systems in the AP600.

The criteria used to evaluate pipe failure protection are generally consistent with NRC guidelines including those in the Standard Review Plan Sections 3.6.1 and 3.6.2, NUREG-1061, Volume 3 (Reference 11) and applicable Branch Technical Positions.

Subsection 3.6.1 provides the design bases and criteria for the analysis required to demonstrate that essential systems are protected. The high- and moderate-energy systems representing the potential source of dynamic effects are listed. Additionally, the criteria for separation and the effects of adverse consequences are defined.

Subsection 3.6.2 defines the criteria for postulated break location and configuration. High-energy pipes are evaluated for the effects of circumferential and longitudinal pipe breaks and through-wall cracks. Moderate-energy pipes are evaluated for the effects of through-wall cracks. Analysis methods and criteria for evaluating pipe whip and evaluating the consequences of jet impingement, motions of the pipe, and system depressurization on

integrity and operability are provided. The evaluation of containment penetrations, pipe whip restraints, guard pipes, and other protective devices is also described. The criteria for excluding breaks in high-energy piping adjacent to containment penetrations are also provided.

Evaluation of the dynamic effects of postulated breaks in the reactor coolant loop, main steam ~~and feedwater~~ lines inside containment, and other primary piping inside containment equal to or greater than the 6 ~~4~~-inch nominal pipe size (NPS) is eliminated for AP600 based on mechanistic pipe break (leak-before-break) considerations. Those sections of high-energy piping that qualify for mechanistic pipe break are evaluated for only the effects of leakage cracks.

Subsection 3.6.3 describes the application of leak-before-break criteria to permit the elimination of pipe rupture dynamic effects considerations. Design guidelines aid in the design of piping systems that satisfy the requirements for mechanistic pipe break. Dynamic effects of postulated breaks are evaluated for those analyzable sections of high-energy piping systems that do not use the mechanistic pipe break methods.

The safety analyses in Chapter 15 and the requirements for emergency core cooling discussed in Section 6.3 and the environmental qualification of equipment discussed in Section 3.11 of this report are not changed by the use of mechanistic pipe break considerations for pipe rupture dynamic effects evaluations. Chapter 6 describes the containment subcompartment pressurization analyses including mechanistic pipe break considerations.

3.6.1 Postulated Piping Failures in Fluid Systems Inside and Outside Containment

A number of systems and components are necessary to shut the plant down in the event of a pipe rupture. These systems, termed essential systems, are protected from the postulated pipe ruptures. The essential systems for various pipe ruptures are the reactor coolant system, the steam generator system, the passive core cooling system, and the passive containment cooling system. In addition to these fluid systems, the protection and safety monitoring system and the Class 1E dc and UPS system are essential. The main control room and main control room habitability system are also protected as essential systems. In addition, containment penetrations and isolation valves (including those for nonessential systems) are essential.

Most of the equipment required for plant safety or safety-related shutdown is located inside containment. The piping inside containment also represents the most significant piping relative to plant safety and, therefore, is subject to the most stringent design and analysis requirements.

Essential equipment in the vicinity of piping that does not satisfy leak-before-break criteria is protected as required by the use of protective structures, pipe restraints, and separation. The need for protection of essential structures, systems and components is determined by evaluation of the dynamic effects. The design bases and criteria for the evaluation follow.

- For evaluation of spray wetting, flooding, and subcompartment pressurization effects, longitudinal cracks (with crack flow areas of 1 square foot) are postulated in the main steam and main feedwater piping. The dynamic effects of pipe whip and jet impingement are not evaluated for these cracks. Locations having the greatest effect on essential equipment are chosen.

- Guard pipe assemblies for high-energy piping in the containment annulus region between the containment shell and shield building that are part of the containment boundary are designed according to the rules of Class MC, subsection NE, of the ASME Code. The following requirements also apply. The design pressure and temperature are equal to or greater than the maximum operating pressure and temperature of the enclosed process pipe under normal plant conditions. Level C service limits of the ASME Code, Section III, Paragraph NE-3221(c), are not exceeded by the loadings associated with containment design pressure and temperature in combination with a safe shutdown earthquake. The guard pipe assemblies are subjected to a pressure test performed at the maximum operating pressure of the enclosed process pipe.

Areas of system piping where no breaks, except as noted in subsections 3.6.1.3 and 3.6.1.2.2, are postulated are as follows:

- The main steam piping, from the containment penetration flued head outboard weld, to the upstream weld of the auxiliary building anchor downstream of the main steam isolation valves, including the main steam safety valves and the connecting branch piping

- The main feedwater piping, from the containment penetration flued head outboard weld, to the auxiliary building anchor upstream of the isolation valve~~, including branch connections~~

- The startup feedwater piping from the containment penetration to the auxiliary building anchor upstream of the isolation valve ~~including branch connections~~

- The steam generator blowdown piping from the containment to auxiliary building anchor downstream of the isolation valve

- The chemical and volume control system makeup piping from the containment to the anchor upstream of the outboard isolation valve ~~including branch connections~~.

- The chemical and volume control system makeup piping from the containment to the anchor downstream of the inboard isolation valve ~~including branch connections~~.

All other fluid system containment penetrations are for moderate-energy systems or for pipe of 1-inch nominal diameter or smaller. See subsection 6.2.3 for a discussion of containment penetrations.

For ASME Class 1 piping terminal end locations are determined from the piping isometric drawings. Intermediate break locations depend on the ASME Code stress report fatigue analysis results. These results are not available at design certification. For the design of the AP600, breaks are postulated at locations typically associated with a high cumulative fatigue usage factor. These locations are at valves, tees, and branch connections which have significant structural discontinuities. The combined license applicant will evalute these locations as part of the as-built reconciliation, (see subsection 3.6.4.1). The following ASME Class 1 lines are evaluated to terminal end and intermediate high energy break locations if applicable.

Line	Diameter (inches)
Pressurizer Spray	4
Automatic Depressurization Stage 1	4
Chemical and Volume Control Letdown	3
Chemical and Volume Control Makeup	3
Pressurizer Auxiliary Spray	2

For ASME Class 2 and 3 piping, terminal end break locations are determined from the piping isometric drawings. The intermediate break locations depend on the stress level. The AP600 ASME Class 2 and 3 lines do not have intermediate breaks based on the low stress. The following ASME Class 2 and 3 lines have terminal end high energy break locations.

Line	Diameter (inches)
Main Feedwater	16
Startup Feedwater	6
Steam Generator Blowdown	4

For B31.1 piping, terminal end break locations are determined from the piping isometric drawings. The intermediate break locations in seismically analyzed pipe depend on the stress level. The AP600 ASME seismically analyzed B31.1 piping does not have intermediate breaks based on the low stress. For nonseismically analyzed high-energy ASME B31.1, intermediate breaks locations are postulated at each fitting.

Rooms that contain high energy pipe break locations are listed in Table 3.6-2.

Essential Systems and Components

In rooms that contain high energy pipe breaks, the systems and components that are needed to mitigate the postulated break and achieve a safe plant shutdown are identified. Rooms that contain both high energy pipe break locations and essential systems or components that must be protected are listed in Table 3.6-3.

High-energy ASME Code Section III piping that is evaluated to the leak-before-break criteria is identified in Appendix 3E. This applies to the main steam ~~and main feedwater~~ piping as follows. The main steam piping from the steam generator outlet nozzle to the anchor downstream of the isolation valve is analyzed for applicable loadings including the safe shutdown earthquake. This anchor is at the exterior wall of the auxiliary building. The portion of this piping from the containment penetration flued head inboard weld to the above anchor satisfies the break exclusion zone requirements described in subsection 3.6.2. The portion of this piping from the steam generator outlet nozzle to flued head inboard weld is evaluated to the leak-before-break criteria. ~~The main feedwater piping from the steam generator inlet nozzle to the anchor upstream of the isolation valve is analyzed for applicable loadings including the safe shutdown earthquake (SSE). This anchor is also located at the exterior wall of the auxiliary building. The portion of this piping from the containment penetration flued head inboard weld to the above anchor satisfies the break exclusion zone requirements described in subsection 3.6.2. The portion of the piping from the steam generator inlet nozzle to the flued head inboard weld is evaluated to the leak before break criteria.~~ High-energy piping that does not satisfy the leak-before-break criteria is designed to the requirements discussed in subsections 3.6.1 and 3.6.2.

The piping to which mechanistic pipe break is applied is analyzed to demonstrate that the piping has leak-before-break characteristics. The leak-before-break analysis is either a fracture-mechanics based stability analysis or a plastic-instability limit load analysis as appropriate. The analysis combines normal and abnormal (including seismic) loads to determine a critical crack size for a postulated through-wall crack. The critical crack size is compared to the size of a leakage crack for which, with appropriate margin, detection is certain. When the critical crack size is sufficiently larger than the leakage crack size the leak-before-break requirements are satisfied.

Mechanistic pipe break is not used for purposes of specifying non-structural design criteria for emergency core cooling, containment systems, or other non-structural engineered safety features, or for the evaluation of environmental effects including spray wetting, humidity, and adverse reactions with chemicals in the coolant. This includes piping for which leak-before-break is demonstrated.

A bounding analysis is performed for each piping system. The bounding analysis is used by the Combined License applicant to verify that the as-built piping satisfies the requirements for leak-before-break.

3.6.3.1 Application of Mechanistic Pipe Break Criteria

Piping systems to which mechanistic pipe break are applied are high integrity systems with well understood loading combinations and conditions. The piping systems to which it is applied satisfy the requirements of the ASME Code, Section III. ASME Code requirements also apply to the pre-service and in-service inspection which confirm continued integrity.

The mechanistic pipe break approach is applicable to high-energy piping provided plant design, operating experience, tests, or analyses have indicated low probability of failure from effects of intergranular stress corrosion cracking, water hammer, steam hammer, fatigue (thermal or mechanical), or erosion.

The plant design and operating features permit the application of the mechanistic pipe break approach. The piping to which the leak-before-break criteria is applied is evaluated for fatigue due to cyclic loads as required by the appropriate requirements of the ASME Code.

The piping in the AP600 does not operate at temperatures for which creep or creep fatigue must be considered.

The reactor coolant loop piping, branch lines, and other lines in contact with reactor coolant are fabricated of austenitic stainless steel, which is very resistant to erosion and corrosion in typical reactor coolant chemistries and flow rates. Intergranular stress corrosion cracking has not been associated with reactor coolant piping in pressurized water reactors.

The design of the reactor coolant loop is not conducive to the generation of water hammer loads. The reactor coolant loop does not have any valves that could result in a water hammer due to rapid valve closure. The steam bubble in the pressurizer is not subject to the introduction of a large volume of cold water sufficient to result in a bubble collapse water hammer.

The design and component selection of reactor coolant branch lines and other lines evaluated for mechanistic pipe break follow design guidelines intended to minimize the potential for water hammer.

Thermal stratification of water in stagnant or slowly flowing lines can result in thermal fatigue in a pipe. The piping and system design requirements for AP600 address the potential for thermal stratification. For additional information of thermal stratification, see subsections 3.9.3, 5.4.3, and 5.4.5.

~~Particular care is taken in the design of the main feedwater piping to minimize the potential failure mechanisms. Failure mechanisms in main feedwater piping attributed to erosion, corrosion, erosion/corrosion, and thermal stratification and the occurrences of water hammer transients were considered. The connection of the startup feedwater piping directly to the steam generator nozzle, rather than to the main feedwater piping, minimizes the potential for thermal stratification in the feedwater piping.~~

The water chemistry and flow velocities in the main ~~feedwater and~~ steam lines are controlled to minimize the potential for erosion and corrosion. ~~Additionally, the individual main feedwater piping is fabricated from a material that has enhanced erosion resistance compared to materials traditionally used in feedwater lines. The use of the corrosion resistant material for the main feedwater piping extends from the steam generator to the larger diameter header pipe in the turbine building to minimize the potential for a rupture outside the nuclear island that would challenge safety systems.~~

~~The routing of the main feedwater piping and the design of the steam generator internals minimize the potential for a water hammer transient in the feedwater piping. Additionally, operating procedures for main feedwater minimizes the potential for water hammer.~~ The main steam lines are not subject to water hammer or thermal stratification by the nature of the fluid transported.

The steam line is protected from being filled with water due to steam generator overfill by implementation of operating instructions or isolation requirements included in the protection system logic or both. See Section 7.3 for information on the protection system design to prevent overfill.

In addition to requirements on the design, fabrication, and inspection of the piping systems, the application of mechanistic pipe break requires a qualified leak detection capability. Leak detection systems inside containment meet the guidelines of Regulatory Guide 1.45. See subsection 5.2.5 for a discussion of the leak detection system for the reactor coolant system and connected piping.

3.6.3.2 Design Criteria for Leak-before-Break

The methods and criteria to evaluate leak-before-break in the AP600 are consistent with the guidance in NUREG-1061 (Reference 11) and Draft Standard Review Plan 3.6.3 (Reference 12). The application of the mechanistic pipe break in AP600 requires that the following design requirements are met.

- Pre-service inspection of welds is required.

- For ASME Code Class 1, Class 2, and 3 systems for which leak-before break is demonstrated, the ASME Section XI preservice and inservice inspection will provide for the integrity of each system. ~~Appendix 3B describes augmented inspection requirements for the feedwater line.~~

- Inservice inspection and testing of snubbers (if used) are performed to provide for a low snubber failure rate.

- For the maximum stress due to steady-state vibration refer to subsection 3.9.2.

- The leak-before-break bounding analysis curves are developed for each applicable piping system. The bounding analysis methods are described in Appendix 3B. These curves give the design guidance to satisfy the stress limits and leak-before-break acceptance criteria. The highest stressed point (critical location) determined from the piping stress analysis is compared to the bounding analysis curve and has to fall on or under the curve. The points on or under the bounding analysis curve satisfy the requirements for leak-before-break.

As noted in subsection 5.2.5, the rated capability of the leak detection systems for the primary coolant inside containment is 0.5 gpm in one hour. The methods used to detect leakage are described in subsection 5.2.5.3. The methods used for primary coolant are the containment sump level, inventory balance, and containment atmosphere radiation. The method used to detect leakage from the main steam ~~and main feedwater~~ line inside containment is the containment sump level. Containment air cooler condensate flow, and containment atmosphere pressure, temperature, and humidity also provide an indication of possible leakage.

Stability and Critical Flaw Sizes

The local and global failure mechanisms are evaluated, as appropriate, to provide margin on flaw size and load. The local mode of failure addresses crack tip behavior: blunting, initiation, extension, and instability. The local failure mechanism is evaluated for ferritic steel piping systems using the J-integral method. The global mode of failure addresses the behavior of the net section: initial yielding, strain hardening, and plastic hinge formation. The global failure mechanism (limit load method) is evaluated for stainless steel piping with no cast material and GTAW welding. From these evaluations a critical crack size is determined. That is, a crack larger than the critical crack size would have unstable growth characteristics.

Acceptance Standards

The results of the preceding evaluations are compared to show that the critical flaw size, which is shown to be stable when the maximum loads are combined based on individual absolute values, is at least twice the size (to satisfy margin of 2 on flaw size) of the leakage flaw size.To satisfy a margin on load of 1.0, the maximum loads are combined using absolute summation of individual values. The maximum loads are described in Appendix 3B subsection 3B.3.3.

The torsional moments are not combined with the bending moments since the torsional moment does not have a significant effect on postulated circumferential cracks.

Bounding Analyses

Evaluations are provided for each different combination of material type, pipe size, pressure, and temperature. These evaluations are used to develop a set of curves of maximum faulted stress versus the corresponding normal stress that satisfy the criteria for leak-before-break. These curves are used in the design of the piping systems and will be used by the Combined License applicant to verify that the as-built piping satisfies the requirements for leak-before-break.

Table 3.6-2 (Page 1 of 7)

SUBCOMPARTMENTS AND POSTULATED PIPE RUPTURES

Compartment		Lines Evaluated to LBB		Lines Not Evaluated to LBB	
Name	Room Number	Description	Terminal End Break Location Excluded by LBB	Description	Terminal End Break Location
Steam Generator Compartment 1	11201	22 in. Cold Leg (RCS)	RC Pump Nozzles (2)	~~None~~ 4 in. Pressurizer Spray (RCS)	Cold Leg Nozzles (2)
		12 in. Fourth Stage ADS (RCS)	Hot Leg Nozzle		
		~~4 in. Pressurizer Spray (RCS)~~	~~Cold Leg Nozzles (2)~~		
	11301	31 in. Hot Leg (RCS)	SG Nozzle	3 in. Purification (CVS)	3 in. by 10 in. PRHR Branch
		18 in. Surge Line (RCS)	Hot Leg Nozzle		
		12 in. & 10 in. Fourth Stage ADS (RCS)	Valves: V004A/C		
		10 in. PRHR Return (RCS)	SG Nozzle		
	11401	None		4 in. SG Blowdown (SGS)	4 in. SG Nozzle
	11501	None		None	
	11601	~~16 in. Feedwater (SGS)~~	~~SG Nozzle~~	16 in. Feedwater (SGS)	SG Nozzle
				6 in. Startup Feedwater (SGS)	SG Nozzle
	11701	32 in. Main Steam (SGS)	SG Nozzle	None	

Table 3.6-2 (Page 2 of 7)

SUBCOMPARTMENTS AND POSTULATED PIPE RUPTURES

Compartment		Lines Evaluated to LBB		Lines Not Evaluated to LBB	
Name	Room Number	Description	Terminal End Break Location Excluded by LBB	Description	Terminal End Break Location
Steam Generator Compartment 2	11202	22 in. Cold Leg (RCS)	RC Pump Nozzles (2)	None	
		12 in. Fourth Stage ADS (RCS)	Hot Leg Nozzle		
		20 in. Normal RHR (RCS)	Hot Leg Nozzle		
		12 in. Normal RHR (RCS)	20 in. x 12 in. Reducer (This is not a terminal end)		
	11302	31 in. Hot Leg (RCS)	SG Nozzle	None	
		12 in. & 10 in. Fourth Stage ADS (RCS)	Valves: V004B/D		
		8 in. Cold Leg to CMT (RCS)	Cold Leg Nozzles (2)		
	11402	None		4 in. SG Blowdown (SGS)	4 in. SG Nozzle
	11502	None		None	
	11602	~~16 in. Feedwater (SGS)~~	~~SG Nozzle~~	16 in. Feedwater (SGS)	SG Nozzle
				6 in. Startup Feedwater (SGS)	SG Nozzle
	11702	32 in. Main Steam (SGS)	SG Nozzle	None	
Reactor Vessel Nozzle Area	11205	31 in. Hot Leg (RCS)	Reactor Vessel Nozzles (2)	None	
		22 in. Cold Leg (RCS)	Reactor Vessel Nozzles (4)		
		8 in. Direct Vessel Injection (RCS)	Reactor Vessel Nozzles (2)		

Table 3.6-2 (Page 3 of 7)

SUBCOMPARTMENTS AND POSTULATED PIPE RUPTURES

Compartment		Lines Evaluated to LBB		Lines Not Evaluated to LBB	
Name	Room Number	Description	Terminal End Break Location Excluded by LBB	Description	Terminal End Break Location
PXS Valve and Accumulator Room A	11206	8 in. Accumulator Injection (PXS)	Accumulator Nozzle	None	
		8 in. CMT Injection (PXS)	CMT Nozzle		
		6 in. Line from Normal RHR (RNS)	Valve: V017A		
		6 in. Line from IRWST (PXS)	Valves: V125A & V123A		
PXS Valve Room B	11207 PXS	6 in. Line from Normal RHR (RNS)	Valve: V017B	None	
		6 in. Line from IRWST (PXS)	Valves: V125B & V123B		
Accumulator Room B	11207 ACCUM	8 in. Accumulator Injection (PXS)	Accumulator Nozzle	None	
		8 in. CMT Injection (PXS)	CMT Nozzle		
Vertical Access	11204	None		3 in. Line from Regen HX to SG 01 (CVS)	Anchor to Wall
				3 in. Purification from Cold Leg to Regen HX (CVS)	Anchor to Wall
RNS Valve Room	11208	10 in. Normal RHR (RNS)	Valves: V001A/B	None	

Table 3.6-2 (Page 4 of 7)

SUBCOMPARTMENTS AND POSTULATED PIPE RUPTURES

Compartment		Lines Evaluated to LBB		Lines Not Evaluated to LBB	
Name	Room Number	Description	Terminal End Break Location Excluded by LBB	Description	Terminal End Break Location
Lower Pressurizer Compartment	11303	18 in. Surge Line (RCS)	Pressurizer Nozzle	None	
Upper Pressurizer Compartment	11503	14 in. ADS (RCS)	Pressurizer Nozzle (2)	~~None~~ 4 in. Pressurizer Spray (RCS)	Pressurizer Nozzle
		~~4 in. Pressurizer Spray (RCS)~~	~~Pressurizer Nozzle~~		
Lower ADS Valve Area	11603	14 in. & 8 in. ADS (RCS)	Valves: V012B & V013B	4 in. ADS (RCS)	Valve V0011B & 14 in. x 4 in. Branch
		6 in. Pressurizer Safety (RCS)	14 in. x 6 in. Tee, Valve-V005B		
Upper ADS Valve Area	11703	14 in. & 8 in., ADS (RCS)	Valves: V012A & V013A	4 in. ADS (RCS)	Valve V0011A & 14 in. x 4 in. Branch
		6 in. Pressurizer Safety (RCS)	14 in. x 6 in. Tee, Valve-V005A		
Maintenance Floor/ Mezzanine	11400	32 in. Main Steam (SGS)	Non-terminal End Location (2) at Boundary of Break Exclusion Zone	6 in. Startup Feedwater (SGS)	Anchors (2) at Containment Penetration
		~~16 in. Feedwater (SGS)~~	~~Non-terminal End Location (2) at Boundary of Break Exclusion Zone~~		
		10 in. Passive RHR (PXS)	PRHR HX Nozzle		
		8 in. CMT Piping	CMT Nozzles (2)		

Table 3.6-2 (Page 5 of 7)

SUBCOMPARTMENTS AND POSTULATED PIPE RUPTURES

Compartment		Lines Evaluated to LBB		Lines Not Evaluated to LBB	
Name	Room Number	Description	Terminal End Break Location Excluded by LBB	Description	Terminal End Break Location
SG01 Access Room	11304	None		None	
Pressurizer Spray Valve Room	11403	None		4 in. Pressurizer Spray (RCS)	Anchor (both sides)
Maintenance Floor	11300	10 in. Passive RHR (PXS)	PRHR HX Nozzle	None	
Operating Deck	11500	None		None	
CVS Room	11209	None		3 in. Purification from Pressurizer Spray to Regen HX (CVS)	Regen HX Nozzle
				3 in. Return, Auxiliary Spray (CVS)	Regen HX Nozzle
				3 in. Return to RNS from Regen HX (CVS)	Valve: V079
				3 in. Supply from RNS to Letdown HX (CVS)	Valve: V072
				3 in. Supply from Regen HX to Letdown HX (CVS)	Nozzles: Regen HX, Letdown HX
CVS Room Pipe Tunnel	11209 PIPE	None		3 in. Purification from Anchor to Regen HX	Anchor
				3 in. Return from Regen HX to Anchor (CVS)	Anchor
				4 in. SG Blowdown (SGS)	Anchors (2) at Containment Penetration

Table 3.6-2 (Page 6 of 7)

SUBCOMPARTMENTS AND POSTULATED PIPE RUPTURES

Compartment		Lines Evaluated to LBB		Lines Not Evaluated to LBB	
Name	**Room Number**	**Description**	**Terminal End Break Location Excluded by LBB**	**Description**	**Terminal End Break Location**
Reactor Coolant Drain Tank Room	11104	None		None	
Reactor Vessel Cavity	11105	None		None	
MSIV Compartment B	12504/ 12404	None		Main Steam Main Feedwater Startup Feedwater Lines(a)	Longitudinal Cracks with Crack Flow Areas of 1 Square Foot are Postulated
MSIV Compartment A	12506/ 12504	None		Main Steam Main Feedwater Startup Feedwater Lines(a)	Longitudinal Cracks with Crack Flow Areas of 1 Square Foot are Postulated
Valve/Piping Penetration Room	12306	None		4 in. Steam Generator Blowdown(a)	Anchors (2) at Containment Penetrations Anchors (2) at Wall to Turbine Building

Note:

a. The piping in these areas is included in break exclusion zones. For additional information on the evaluation of these lines, see subsection 3.6.1.2.1 for the steam generator blowdown line; subsection 3.6.1.2.2 for information on the evaluation of lines in MSIV compartment B because of the proximity to the main control room; and subsection 3.6.2.1.1.4 for general break exclusion zone requirements.

Table 3.6-2 (Page 7 of 7)

SUBCOMPARTMENTS AND POSTULATED PIPE RUPTURES

Room #	Description	Bottom Elevation	Top Elevation
11104	RCDT Room	66'-6"	81'-0"
11105	Reactor Vessel Cavity	66'-6"	98'
11205	Reactor Vessel Nozzle Area	98'	107'-2"
11201	SG Compartment 1	83'	104'-7"
11202	SG Compartment 2	83'	104'-7"
11204	Vertical Access	83'	107'-2"
11206	PXS Valve Room A	87'6"	105'-2"
11300	Maintenance Floor	107'-2"	118'-6"
11301	SG Compartment 1	104'-7"	113'-9"
11302	SG Compartment 2	104'-7"	113'-9"
11400	Maintenance Floor/Mezzanine	118'-6"	135'-3"
11401	SG Compartment 1	113'-9"	135'-3"
11402	SG Compartment 2	113'-9"	135'-3"
11501	SG Compartment 1	135'-3"	149'-7"
11502	SG Compartment 2	135'-3"	149'-7"
11601	SG Compartment 1	149'-7"	162'-1"
11602	SG Compartment 2	149'-7"	162'-1"
11701	SG Compartment 1	162'-1"	—
11702	SG Compartment 2	162'-1"	—
11500	Operating Deck	135'-3"	256'-2 3/8"
11303	Pressurizer Lower Compartment	107'-2"	135'-3"
11304	SG01 Access Room	107'-2"	118'-6"
11403	Pressurizer Spray Valve Room	118'-6"	135'-3"
11503	Pressurizer Upper Compartment	135'-3'	163'-0"
11603	Lower ADS Valve Area	163'-0"	173'-10"
11703	Upper ADS Valve Area	173'-10"	—
11207 ACCUM	Accumulator Room B	87'-6"	105'-2"
11207 PXS	PXS Valve Room B	87'-6"	105'-2"
11208	RNS Valve Room	94'	105'-2"
11209	CVS Room	80'-6"	105'-2"
11209 PIPE	CVS Room Pipe Tunnel	100'-0"	105'-2"
12306	Valve/Piping Penetration Room	100'-0"	117'-6"
12504/12404	MSIV Compartment B (Upper/Lower)	117'-6"	153'-0"
12506/12406	MSIV Compartment A (Upper/Lower)	117'-6"	153'-0"

Table 3.6-3

ROOMS WITH HIGH ENERGY PIPE BREAKS AND POTENTIAL ESSENTIAL TARGET INTERACTION

Elevation	Room Numbers*	High Energy Break Source	Essential Equipment Protected by Whip Restraints or Jet Shields
66'-6"	None		
82'-6"	11201	RCS Press. Spray - Terminal End	RCS-ADS valves: V004A, V004C, V014A, V014C
	11204		None
	11209		None
96'-6"	11204		None
	11209		None
100'-0" and 107'-2"	11209 Pipe chase	SGS Blowdown Piping - Terminal End	CVS Makeup, CVS Letdown, CVS Hydrogen Supply and SGS steam generator blowdown piping
	11300		None
	11301		None
	11303/ 11304	RCS Makeup Piping - Intermediate Break	~~RCS and~~ SGS sg blowdown and sg drain Piping, RCS pressurizer pressure and level instrumentation, and Pressurizer support steel
117'-6"	11400	SGS Start Up Feedwater Piping - Terminal end	None ~~Raceways and cables for Divisions A/C and B/D~~
	11401		~~None~~ RCS-ADS valves: V004A, V004C, V014A, V014C are protected from a break located in room 11403
	11402		Steam Generator supports are protected from a break located in room 11400
	11403	RCS Press Spray - Terminal End	None
	11403	RCS Letdown - Intermediate Break	Raceways for Divisions A/C and B/D
135'-3"	~~None~~ 11503	RCS Press Spray - Terminal End	RCS-ADS valves: lower tier platform support steel

160'-6" and 153'-0"	11601	SGS Start Up Feedwater Piping - Terminal end SGS Main Feedwater Piping-Terminal End	RCS head vent piping SGS level instrumentation piping
	11602	SGS Main Feedwater Piping-Terminal End	~~None~~ SGS level instrumentation piping
	11603	RCS ADS Stage 1 Piping - Terminal End	RCS piping and ADS valves 002B, 003B, 012B, & 013B Raceways and cables for Divisions A/C and B/D
	11703	RCS ADS Stage 1 Piping - Terminal End	RCS piping and ADS valves 002A, 003A, 012A, & 013A Raceways and cables for Division ~~B/D~~ A/C

* See Figures 1.2-1 through 1.2-8, 1.2-10, and 1.2-11 for room numbers

APPENDIX 3B

LEAK-BEFORE-BREAK EVALUATION OF THE AP600 PIPING

General Design Criterion 4 requires that structures, systems, and components important to safety be designed to accommodate the effects of conditions associated with normal operation, anticipated transients, and postulated accident conditions. However, the dynamic effects and flooding associated with pipe rupture may be excluded when analysis demonstrates that the probability of fluid system pipe rupture is extremely low. Dynamic effects are not considered for those segments of piping that are shown mechanistically, with a large margin, not to be susceptible to a pipe rupture.

The dynamic effects associated with pipe rupture include effects such as pipe break reaction loads, jets and jet impingement, subcompartment pressurization loads, and transient pipe rupture depressurization loads on other components.

The use of mechanistic pipe break to eliminate evaluation of dynamic effects of pipe rupture includes material selection, inspection, leak detection, and analysis. Subsection 3.6.3 outlines considerations relative to material selection, inspections, and leak detection. Subsection 5.2.5 describes the leak detection system inside containment. This appendix describes the analysis methods used to support the application of mechanistic pipe break to high-energy piping in the AP600.

The analysis and criteria to eliminate dynamic effects of pipe breaks are encompassed in a methodology called leak-before-break (LBB). This methodology has been validated by theoretical investigations and test demonstrations sponsored by the industry and the NRC.

The primary regulatory documents for leak-before-break analyses are General Design Criterion No. 4 (GDC-4), Draft Standard Review Plan 3.6.3 (SRP 3.6.3) (Reference 1), and NUREG-1061, Volume 3 (Reference 2). Although SRP 3.6.3 has been issued only as a draft, its provisions are followed as guidelines to leak-before-break analyses.

Leak-before-break methodology has been applied to the reactor coolant loop and high-energy auxiliary line piping in operating nuclear power plants. The leak-before-break analysis used to support the piping design of the AP600 is an application of the same methodology used in leak-before-beak evaluations previously accepted by the NRC.

In the AP600, leak-before-break evaluations are performed for the reactor coolant loop, the surge line, selected other branch lines containing reactor coolant down to and including 6 ~~4~~-inch diameter nominal pipe size, and portions of the main steam line~~, and portions of the main feedwater line~~. Those lines not qualified to the leak-before-break criteria are evaluated using the pipe rupture protection criteria outlined in subsections 3.6.1 and 3.6.2.

This appendix provides a leak-before-break analysis for the applicable piping systems. Table 3B-1 provides a list of AP600 leak-before-break piping systems.

3B.1 Leak-Before-Break Criteria for AP600 Piping

The methodology used for leak-before-break analysis is consistent with that set forth in GDC-4, SRP 3.6.3 (Reference 1) and NUREG-1061, Volume 3 (Reference 2). The steps are:

- Evaluate potential failure mechanisms
- Perform bounding analysis

3B.2 Potential Failure Mechanisms for AP600 Piping

In high-energy piping, there are material degradation mechanisms that could adversely affect the integrity of the system as well as its suitability for leak-before-break analysis. The following lists potential degradation (or "failure") mechanisms:

- Erosion-corrosion induced wall thinning
- Stress corrosion cracking (SCC)
- Water hammer
- Fatigue
- Thermal aging
- Thermal stratification
- Other mechanisms

The stainless steel piping is fabricated of SA312TP316LN or SA312TP304L material. The type 304L material is used in the accumulator discharge lines. ~~The main feedwater piping is fabricated of SA335P11 (low alloy steel).~~ The main steam piping is fabricated of SA333 Grade 6. The welds are made by the gas tungsten arc welding (GTAW) method.

The various degradation mechanisms are discussed in the following subsections.

3B.2.1 Erosion-Corrosion Induced Wall Thinning

Primary Loop Piping

Wall thinning by erosion and erosion-corrosion effects does not occur in the primary loop piping because SA312TP316LN austenitic stainless steel material is highly resistant to these effects. The coolant velocity in the AP600 primary loop is about 43 feet per second, which is lower than the velocity in operating Westinghouse-designed pressurized water reactors. The bend radii in the AP600 hot and cold legs are greater than the bend radii used in the crossover legs of operating plants. There is no record of erosion-corrosion induced wall thinning in the primary loops of operating plants.

Auxiliary Stainless Steel Piping

Wall thinning by erosion-corrosion effects does not occur in the auxiliary stainless steel piping because SA312TP316LN and SA304TP304L austenitic stainless materials are highly resistant to these effects. The coolant velocity in these systems is lower than in comparable system

velocity in operating Westinghouse-designed pressurized water reactors. There is no record of erosion-corrosion induced wall thinning in the stainless steel piping of operating plants.

Main Steam Line

Main steam lines in the AP600 are fabricated from SA333 Grade 6 Carbon steel. Erosion-corrosion induced wall thinning is not expected in the main steam line. Extensive work has been done investigating erosion-corrosion in carbon steel pipes. The main steam line has low susceptibility to erosion due to the relatively high operating temperature. Susceptibility is also low due to the high quality steam in the main steam line.

~~Main Feedwater Line~~

~~The feedwater line is fabricated from SA335 P11 low alloy steel. The water chemistry and flow velocities in the feedwater line are controlled to limit the potential for erosion and corrosion. The feedwater piping material has enhanced erosion resistance compared to materials traditionally used in feedwater lines. The alloy steel was modeled utilizing Electric Power Research Institute's (EPRI's) "CHECMATE" (Reference 3) program to determine erosion-corrosion rates based on AP600 chemistry controls. "CHECMATE" is an EPRI developed computer code which quantifies expected erosion-corrosion rates based on chemistry, material, fluid conditions, and piping configurations. The calculated wear rates provide significant margin for the proposed feedwater line for the 60 year plant life. The corrosion resistant material used for the feedwater piping extends from the steam generator to the common header pipe in the turbine building.~~

Based on the above discussion, erosion-corrosion induced wall thinning does not have an adverse effect on the integrity of the AP600 leak-before-break piping systems.

3B.2.2 Stress Corrosion Cracking

Stress corrosion cracking is not expected to occur in the AP600 piping systems because the three conditions necessary for stress corrosion cracking to take place are not present. If any of these three conditions is not present, stress corrosion cracking will not take place. The three conditions are:

- There must be a corrosive environment.
- The material itself must be susceptible.
- Tensile stresses must be present in the material.

Primary Loop Piping

During plant operation, the reactor coolant water chemistry is monitored and maintained within specific limits (see subsection 5.2.3 for a discussion of reactor coolant chemistry). Contaminant concentrations are kept below the thresholds known to be conducive to stress corrosion cracking. The major water chemistry control standards are included in the plant operating procedures as a condition for plant operation.

...or plants (which have not experienced stress corrosion cracking in the auxiliary stainless steel piping).

Main Steam Line ~~and Main Feedwater Line~~

The main steam piping is constructed from ferritic steel. Stress corrosion cracking in ferritic steels commonly result from a caustic environment. A source of a caustic environment in the main steam piping would be moisture carryover from the steam generator. However, the secondary side water treatment utilizes all volatile treatment. All volatile treatment effectively precludes causticity in the steam generator bulk liquid environment. For some operating plants prior to implementing all volatile treatment, the phosphate water treatment caused a caustic chemical imbalance resulting in stress corrosion cracking of steam generator tubing. Under all volatile treatment water treatment conditions, there is no instance of caustic stress corrosion cracking on the ferritic steam lines indicating no significant caustic carryover. The operating secondary side chemistry precludes stress corrosion cracking on the ferritic main steam line.

~~Stress corrosion cracking is not expected to occur in the main feedwater line piping because of control of the oxygen to very low levels. There has been no experience with stress corrosion cracking in feedwater lines in operating plants of Westinghouse design. The operating secondary side chemistry precludes stress corrosion cracking on the main feedwater line.~~

Based on the above discussion, stress corrosion cracking does not have an adverse effect on the integrity of AP600 leak-before-break piping systems.

3B.2.3 Water Hammer

Primary Loop Piping

The reactor coolant loop is designed to operate at a pressure greater than the saturation pressure of the coolant, thus precluding the voiding conditions necessary for water hammer to occur. The reactor coolant primary system is designed for Level A, B, C, and D (normal, upset, emergency, and faulted) service condition transients. The design requirements are conservative relative to both the number of transients and their severity. Relief valve actuation and the associated hydraulic transients following valve opening have been considered in the system design. Other valve and pump actuations cause relatively slow transients with no significant effect on the system dynamic loads.

To provide dynamic system stability, reactor coolant parameters are controlled. Temperature during normal operation is maintained within a narrow range by control rod positioning. Pressure is controlled within a narrow range for steady-state conditions by pressurizer heaters and pressurizer spray. The flow characteristics of the system remain constant during a fuel cycle. The operating transients of the reactor coolant system primary loop piping are such that significant water hammer loads are not expected to occur.

Auxiliary Stainless Steel Piping

The passive core cooling system and automatic depressurization system are designed to minimize the potential for water hammer induced dynamic loads. Design features include:

- Continuously sloping core makeup tank and passive residual heat exchanger inlet lines to eliminate local high points
- Inlet diffusers in the core makeup tanks to preclude adverse steam and water interactions
- Vacuum breakers in the discharge lines of the automatic depressurization valves connected to the pressurizer

The AP600 pressurizer spray control valve is similar to what is used in the operating plants. There is no history of water hammer caused by the spray control valve.

The normal residual heat removal system isolation valves are slow closing valves, identical to operating plants, and therefore would not be a source of water hammer.

These features minimize the potential of water hammer in the auxiliary stainless steel piping system.

~~Main Feedwater Line~~

~~The feedwater piping, steam generator design details, and other component details in the feedwater system are designed to minimize the potential and severity of water hammer within the feedwater piping. The following addresses each aspect of the design incorporated to minimize water hammer.~~

~~Steam Generator Design: The AP600 steam generator design benefits from investigation of water hammer events and the resulting design changes developed to address the events (References 4 through 8).~~

- ~~Top discharge feed flow through spray tubes (similar to J-tubes) from the feedring reduces the potential of void formation when the steam generator level drops below the feedring level. Previous steam generator feedring designs had incorporated bottom discharge holes that permitted feedring draining whenever the steam generator level dropped below the feedring.~~
- ~~Separate startup feedwater and main feedwater nozzles are incorporated to provide for only heated feedwater from the deaerator entering the steam generator via the main feedwater line~~
- ~~Feedwater nozzle design incorporates a welded thermal liner attached to the feedwater nozzle forging to form a positive seal to limit the potential for feedring drainage and therefore void formation within the feedring. Previous designs had included a "close fit"~~

~~but not a complete seal at the connection to the nozzle forging. The welded thermal liner design has no leak paths in the steam generator through which the water can drain from the feedring.~~

~~Feedwater piping design: The AP600 feedwater piping layout has incorporated features to limit void formation and water hammer initiation.~~

- ~~A downward facing elbow is connected to the steam generator nozzle and thus complies with industry recommendations to minimize the horizontal feedwater piping connected to the steam generator. The short horizontal section minimizes amount of steam void which can form.~~

- ~~The main feedwater piping inside containment continuously rises to the steam generator providing for natural venting of the steam generator in the event a steam void is formed.~~

- ~~Long straight piping runs in the feedwater line are limited.~~

~~Component and system design selection:~~

- ~~A major cause of water hammer problems in pressurized water reactor feedwater systems has been control valve instability. These instabilities resulted from factors such as oversized valve, unbalanced valve trim, damage to valve components, and incompatibility of the feedwater control valve with the rest of the feedwater system. These problems are minimized on AP600 by the following:~~

  - ~~The specification of specialized valve trim to avoid instability~~

  - ~~The use of variable speed feedwater pumps to reduce the demands on the control valve requirements~~

  - ~~Reduced control requirements on the main feedwater control valve by the use of a startup feedwater line that provides feedwater flow control from either the startup feedwater pump or the main feed pump at lower feed demand (power) levels.~~

  - ~~Main feedwater control valve positioning during normal operation is the function of the plant control system (see subsection 7.7.1.8) using a refinement of a standard three element control scheme. The control scheme provides greater steam generator level stability and thus reduces potential feedwater transients.~~

- ~~Rapid closure of some types of feedwater check valves may potentially cause water hammer in main feedwater lines. The controlled closure check valve specified for the AP600 main feedwater lines limits the magnitude of the closing loads generated by valve closure caused by depressurization of the feedwater line upstream of the check valve.~~

- ~~Feedwater delivered to the main feedwater line is drawn only from the deaerator. The heated feedwater is normally at least 250°F and helps reduce the possibility of water hammer.~~

- ~~Startup feedwater is piped directly to the steam generator. This feature helps prevent the need to introduce cold water directly into the main feedwater and thus minimizes the chances of steam water counterflow or steam bubble collapse type of water hammer events.~~

- ~~Rapid resumption of feedwater flow to the steam generators is accomplished in the AP600 design. Numerous options are available to maintain or restore steam generator level with the feedwater system design. Based on the flow demand signal and level of feedwater isolation either the main feedwater pump(s) or the startup feedwater pumps can adequately provide level control. If there is no engineered safeguard features feedwater isolation signal present, the main feedwater pumps will provide adequate steam generator inventory control, via the main feedwater line or the startup feedwater line. If a main feedwater isolation signal exists then either the main feedwater pump(s) or the startup feedwater pump(s) provides startup feedwater flow via the startup feedwater line.~~

~~The above design provisions make the potential for steam generator water hammer in the feedwater line extremely low. However, with consideration for the main feedwater and steam generator design features, the susceptibility of the main feedwater line inside containment for water hammer has been evaluated. The most common historic causes were evaluated as well as the relevant modes of operation for susceptibility to the appropriate water hammer mechanisms (Reference 4). The limiting anticipated and unanticipated events were evaluated. The results of the analysis demonstrate that the system is acceptable for leak before break application.~~

Main Steam Line

The steam lines are not subject to water hammer by the nature of the fluid transported. The following system design provisions address concerns regarding steam hammer within the main steam line and identify the significant dynamic loads included in the main steam piping design.

- Design features that prevent water slug formations are included in the system design and layout. In the main steam system, these include the use of drain pots and the proper sloping of lines.

- The operating and maintenance procedures that protect against a potential occurrence of steam hammer include system operating procedures that provide for slowly heating up (to avoid condensate formation from hotter steam on colder surfaces), operating procedures that caution against fast closing of the main steam isolation valves except when necessary, and operating and maintenance procedures that emphasize proper draining.

- The stress analyses for the safety-related portion of the main steam system piping and components include the dynamic loads from rapid valve actuations, including actuation of the main steam isolation valves and the safety valves.

Based on the above discussion, water hammer does not have an adverse effect on the integrity of AP600 leak-before-break piping systems.

3B.2.4 Fatigue

Low-Cycle Fatigue

Low-cycle fatigue due to normal operation and anticipated transients is accounted for in the design of the piping system. The Class 1 piping systems comply with the fatigue usage requirements of the ASME Code, Section III. The Class 2 and 3 piping systems comply with the stress range reduction factors of the ASME Code, Section III.

~~A fatigue evaluation at the main feedwater nozzle equivalent to ASME Class 1 piping is performed. Also, a fatigue crack growth analysis at the main feedwater nozzle is performed.~~

Due to the nature of operating parameters, main steam line piping (Class 2) and the Class 3 portion of the accumulator piping, are not subjected to any significant transients to cause low-cycle fatigue.

Based on the above discussion, low-cycle fatigue is not a concern of AP600 leak-before-break piping systems.

High-Cycle Fatigue

High-cycle fatigue loads in the system result primarily from pump vibrations. The steam generator is designed so that flow-induced vibrations in the tubes are avoided (see subsection 5.4.2). The loads from reactor coolant pump vibrations are minimized by criteria for pump shaft vibrations during hot functional testing and operation. During operation, an alarm signals when the reactor coolant pump vibration is greater than the limits.

~~Main feedwater pump vibration is isolated from the leak before break feedwater line inside containment via the piping and equipment supports.~~

With these precautions taken, the likelihood of leakage due to fatigue in piping systems evaluated for leak-before-break is very small.

3B.2.5 Thermal Aging

Stainless Steel Piping

Piping used in the reactor coolant loop and other auxiliary lines are wrought stainless steel materials, rather than cast materials, so that thermal aging concerns are not expected for the

AP600 piping and fittings. The welds used in the assembly of the AP600 are gas tungsten arc welds (GTAW). These welds are essentially as resistant to the effects of thermal aging as the base metal materials. This is due to the typically low ferrite contact in welds which results in minimal impact from thermal aging. Based on this information, thermal aging of weld materials and piping used in the AP600 is not an issue.

Main Steam ~~and Main Feedwater~~ Lines

The main steam ~~and main feedwater~~ piping systems does not have cast materials. The welding process used on these lines is also gas tungsten arc weld (GTAW).

There are no thermal aging concerns for the carbon steel piping of the main steam line and the alloy steel of the main feedwater piping.

The material used for the main steam ~~and main feedwater~~ piping systems is not susceptible to dynamic strain aging effects.

3B.2.6 Thermal Stratification

Leak-before-break analyses include consideration of the loads and stresses due to thermal stratification.

Thermal stratification occurs only in a pipe that has a susceptible geometry and low flow velocities. A temperature difference between the flowing fluid and stagnant fluid is also a prerequisite.

The design of piping and component nozzles in the AP600 includes provisions to minimize the potential for and the effects of thermal stratification, cycling, and striping, pursuant to actions requested in several NRC bulletins, as discussed below.

Primary Loop Piping

Thermal stratification in the reactor coolant loops resulting from actuation of passive safety features is evaluated as a design transient. Stratification effects due to both Level B and Level D service conditions are considered. The criteria used in the evaluation of the stress in the loop piping due to stratification is the same as that applicable for other Level B and Level D service conditions.

Auxiliary Stainless Steel Piping

Pursuant to the actions requested in NRC Bulletin 88-11, the pressurizer surge line is analyzed to demonstrate that the applicable requirements of the ASME Code, Section III are met. This analysis includes consideration of plant operation, thermal stratification, and thermal striping using temperature distributions and transients developed from experience on existing plant monitoring programs.

Automatic depressurization stage 2 and 3 lines from the pressurizer to the depressurization valves
Leakage is not a concern since double isolation exists in all potential leakage flow paths.

Normal residual heat removal suction lines from the hot legs to the isolation valves
The piping from the hot legs to the isolation valves is expected to be essentially at the hot leg temperature during 100 percent power due to turbulent penetration and convective currents which heat the line. Isolation valve leakage is not a concern since hot leakage from the reactor coolant system would be entering a hot section of piping.

~~Pressurizer spray lines from the cold legs to the pressurizer~~
~~These lines have a controlled bypass flow around the pressurizer spray valves which maintains the piping at approximately cold leg temperature to prevent thermal shock transients in the spray piping and pressurizer spray nozzle. This flow is expected to stratify with pressurizer steam in the piping near the pressurizer. The stratification loads are considered in the analysis.~~

Main Steam Line

The steam lines are not subjected to thermal stratification by the nature of fluid transported.

~~Main Feedwater Line~~

~~Thermal stratification is prevented in the main feedwater line based on the flow rate limitations within the main feedwater and startup feedwater line and the flow control stability for feedwater control. Low feedwater flow duty is provided by the startup feedwater line while higher feedwater flow rates are provided and controlled via the main feedwater line. Subsection 10.4.7.2.3 provides details of the automatic switchover between main and startup feedwater lines. The switchover between the feedwater lines occurs above a minimum flow rate to prevent thermal stratification for limiting temperature deviations.~~

~~Main feedwater control valve positioning during normal operation is the function of the plant control system (see subsection 7.7.1.8). The control scheme enhances steam generator level stability and thus reduces potential feedwater thermal stratification resulting from temporary low flow transients.~~

~~For additional information about augmented inspection of the feedwater line see subsection 3B.8.~~

~~For additional information about stratification, refer to subsection 3.9.3.~~

Based on the above discussion, thermal stratification does not have an adverse effect on the integrity of AP600 leak-before-break piping systems.

3B.2.7 Other Mechanisms

The pipe evaluated for leak-before-break does not operate at temperature for which creep fatigue must be considered. Creep fatigue is a concern for ferritic steel piping operation at temperatures above 700°F and for austenitic stainless steel operation above 800°F.

Pipe degradation or failure by indirect causes such as fires, missiles, and component support failures is precluded by criteria for design, fabrication, inspection, and separation of potential hazards in the vicinity of the safety-related piping. The structures, larger pipe, and components in the vicinity of pipe evaluated for leak-before-break are safety-related and seismically designed or are seismically supported if nonsafety-related.

Cleavage type failures are not a concern for systems operating temperature and material used in the stainless steel piping systems. The material used in the main steam ~~and main feedwater~~ lines ~~are~~ is highly ductile and resistant to cleavage type failure at operating temperatures. The resistance to failure have been demonstrated by material fracture toughness tests.

3B.3 Leak-Before-Break Bounding Analysis

The methodology used for performing the bounding analysis is consistent with that set forth in GDC-4, SRP 3.6.3 (Reference 1) and NUREG-1061, Volume 3 (Reference 2).

Bounding leak-before-break analysis for the applicable AP600 piping systems is performed. The analysis criteria and development techniques of the bounding analysis curves (BAC) are described below. The bounding analysis curve allows for the evaluation of the piping system in advance of the final piping analysis, incorporating leak-before-break considerations early in the piping design process. The leak-before-break bounding analysis curve is used to evaluate critical points in the piping system. A minimum of two points are required to develop the bounding analysis curve. One point for the low normal stress case and the other point for the high normal stress case. If variations in pipe size, material, pressure or temperature occur for a specific piping system, an additional bounding analysis curve is generated. These points meet the following margins for leak-before-break analysis: (References 1 and 2).

- Margin of 10 on leak detection capability
- Margin of 2 on flaw size
- Establish margin of 1 on load by using absolute combination method of maximum loads

3B.3.1 Procedure for Stainless Steel Piping

3B.3.1.1 Pipe Geometry, Material and Operating Conditions

The following information is identified for each of the lines:

- Piping materials - 316LN/304L, Type 304L is used for the accumulator discharge line

maximum stress as determined by the pipe stress results. A comparison is made with the applicable bounding analysis curves for the analyzable piping systems. As outlined in 3B.3.1.1 and 3B.3.2.1, bounding analysis curves are calculated for different combinations of pipe size, pipe schedule, operating pressures, operating temperatures.

The bounding analysis curves are used during the layout and design of the piping systems to provide a design that satisfies leak-before-break criteria. In addition, the Combined License holder compares the results of the as-built piping analysis reconciliation to the bounding analysis curves to verify that the fabricated piping systems satisfies leak-before-break criteria. See subsection 3.6.4.2 for the Combined License information item associated with this verification.

At the critical location, load combinations for the maximum stress calculation use the absolute sum method. The load combinations include the following combinations:

(1) |Pressure| + |Deadweight| + |Thermal (100% Power)| + |Safe Shutdown Earthquake|

(2) |Pressure| + |Deadweight| + |Thermal (100% Power)| + |Valve Thrust Maximum*|

(3) |Pressure| + |Deadweight| + |Thermal Maximum*|

~~(4) |Pressure| + |Deadweight| + |Pipe Break**|; for main feedwater line only~~

~~(5) |Pressure| + |Deadweight| + |Thermal (100% Power)| + |Water Hammer Loads***|; for main feedwater line only~~

* Level A and Level B of ASME Code load conditions. Valve thrust maximum includes anticipated water hammer events resulting from rapid valve closure or opening, including pressurizer safety valve opening (Level C). Thermal maximum includes applicable stratification loads.

~~** Main feedwater pipe break in the turbine building~~

~~***Includes unanticipated water hammer events (vapor pocket collapse during feedwater refilling is the limiting unanticipated event).~~

The normal stress is calculated using the algebraic sum method at critical location and the following load combination.

(1) Pressure + Deadweight + Thermal (100% Power)

3B.3.4 Bounding Analysis Results

Table 3B-2 shows a summary of piping systems and corresponding bounding analysis figures. Figures 3B-2 to 3B-9 and 3B-11 to 3B-~~40~~ 35 show the bounding analysis curves. The curves satisfy the margins as indicated in subsection 3B.3.

3B.4 Differences in Leak-Before-Break Analysis for Stainless Steel and Ferritic Steel Pipe

The significant difference between leak-before-break analysis performed for the stainless steel (Class 1 and Class 3) systems and the ferritic steel in the Class 2 systems is in the stability analysis. In the case of stainless steel systems, stability analyses are performed by limit load approach. In the ferritic steel systems, stability analyses are performed by J-integral approach.

3B.5 Differences in Inspection Criteria for Class 1, 2, and 3 Systems

Class 1, 2 and 3 systems are subjected to in-service inspection requirements from ASME Code, Section XI. For Class 1 piping, terminal ends and dissimilar metal welds are volumetrically inspected, along with other locations, to total 25 percent of the welds. For Class 2 piping, the requirement is to volumetrically inspect the terminal ends and other locations to total 7.5 percent of the welds. For Class 3 systems (the only Class 3 piping is in the accumulator line which is always at room temperature), the system receives periodic visual examinations in conjunction with pressure testing. These requirements were developed by ASME Code, Section XI consistent with the different safety classes of these systems.

The leak-before-break evaluations are based on the ability to detect a potential leaking crack; not the ability to find cracks by inservice inspections. The criteria or methods of the leak-before-break evaluations are the same for ASME Code Class 1, 2, and 3.

3B.6 Differences in Fabrication Requirements of ASME Class 1, Class 2, and Class 3 Piping

The significant difference among Class 1, 2 and 3 seamless pipe occurs in the nondestructive examination requirements. The Class 1 seamless pipe examination requirements include an ultrasonic testing examination, whereas Class 2 and 3 do not. In addition, the Class 1 examination requirements for a circumferential butt welded joint include radioagraphic testing and magnetic particle or liquid penetrant examination where Class 2 does not. The examination requirements for Class 2 pipe require radiographic examination of the welds and normally Class 3 pipe does not. As noted in subsection 3.2.2.5, for Class 3 lines required for emergency core cooling functions, radiography will be conducted on a random sample of welds. The Class 3 leak-before-break lines are included in the lines that are radiographed.

For the fabrication of welds in the Class 1, Class 2 and Class 3 pipes there is no significant differences.

The differences in fabrication and nondestructive examination requirements do not affect the leak-before-break analyses assumptions, criteria, or methods.

~~3B.7 Monitoring of Unanticipated Dynamic Loads in the Feedwater Lines~~

~~Instrumentation for monitoring unanticipated dynamic loads in the feedwater lines inside containment will be provided in the first plant.~~

~~3B.8 Augmented In-Service Inspection at the Main Feedwater Nozzles Connected to the Steam Generators~~

~~Augmented in-service inspection (100 percent volumetric inspection every 10 years of inspection interval) at the weld connecting the piping to the steam generator feedwater nozzles will be performed.~~

3B.~~9~~ 7 References

1. Standard Review Plan 3.6.3, "Leak Before Break Evaluation Procedures," Federal Register, Volume 52, Number 167, Friday, August 28, 1987; Notice (Public Comment Solicited), pp. 32626-32633.

2. NUREG-1061, "Evaluation of Potential for Pipe Breaks, Report of the U.S. Nuclear Regulatory Commission Piping Review Committee," Volume 3, (prepared by the Pipe Break Task Group), November 1984.

3. ~~"CHECMATE (TM) Computer Program User's Manual," EPRI NSAC 145L, April 1989.~~ Deleted

4. ~~EPRI Report EPRI NP 6766 "Water Hammer Prevention, Mitigation and Accommodation," July 1992.~~ Deleted

5. ~~NUREG/CR 2069 "Compilation of Data Concerning Known and Suspected Water Hammer Events in Nuclear Power Plants," May 1992.~~ Deleted

6. ~~NUREG 1190, "Loss of Power and Water Hammer Event at San Onofre, Unit 1 on November 21, 1985," January 1986.~~ Deleted

7. ~~NUREG 0918, "Prevention and Mitigation of Steam Generator Water Hammer Events in PWR Plants."~~ Deleted

8. ~~NUREG 0800/ABS 10-2 (Branch Technical Paper), "Design Guidelines for Avoiding Water Hammers in Steam Generators."~~ Deleted

9. ASME Boiler and Pressure Vessel Code, Section III, "Rules for Construction of Nuclear Power Plant Components; Division 1 - Appendices," 1989 Edition, July 1, 1989.

Table 3B-1

AP600 LEAK-BEFORE-BREAK SCOPE OF PIPING SYSTEMS

Number	System	Pipe Diameter (Inch -Nominal)	Material
1	Primary Loop	31 ID & 22 ID	SA312 Type 316 LN
2	Main Steam A & B	32	SA333 Grade 6
3	~~Main Feedwater A & B~~ Deleted	~~16~~	~~SA335 P11~~
4	Pressurizer Surge Line	18	SA312 Type 316 LN
5	Automatic Depressurization System Stage 2, 3, and Safety	6, 8, & 14	SA312 Type 316 LN
6	Normal Residual Heat Removal	10, 12, & 20	SA312 Type 316 LN
7	Passive Residual Heat Removal Return	10	SA312 Type 316 LN
8	Passive Residual Heat Removal Supply/Automatic Depressurization System Stage 4 (West)	10 & 12	SA312 Type 316 LN
9	Automatic Depressurization System Stage 4 (East)	10 & 12	SA312 Type 316 LN
10	Direct Vessel Injection A & B (Accumulator Discharge line)	6 & 8 8	SA312 Type 316 LN or Type 304 L
11	Core Makeup Tank A & B	8	SA312 Type 316 LN
~~12~~	~~Pressurizer Spray~~	~~4~~	~~SA312 Type 316 LN~~

Table 3B-2

AP600 LEAK-BEFORE-BREAK BOUNDING ANALYSIS SYSTEMS AND CORRESPONDING FIGURES

System	Figure Number
Primary Loop	3B-2, 3B-3, 3B-4, 3B-5, 3B-6, 3B-7
Main Steam A & B	3B-8, 3B-9
~~Main Feedwater A & B~~	~~3B-10~~
Pressurizer Surge Line	3B-11, 3B-12
Automatic Depressurization System Stage 2,3/Safety	3B-13, 3B-14, 3B-15
Normal Residual Heat Removal	3B-16, 3B-17, 3B-18
Passive Residual Heat Removal Return	3B-19, 3B-20, 3B-21, 3B-22
Passive Residual Heat Removal Supply/Automatic Depressurization System Stage 4 (West)	3B-16, 3B-17, 3B-23, 3B-24
Automatic Depressurization System Stage 4 (East)	3B-16, 3B-17, 3B-23, 3B-24
Direct Vessel Injection A & B	3B-25, 3B-26, 3B-27, 3B-28, 3B-29, 3B-30, 3B-31, 3B-32, 3B-33, 3B-34
Core Makeup Tank A & B	3B-27, 3B-35
~~Pressurizer Spray~~	~~3B-36, 3B-37, 3B-38, 3B-39~~

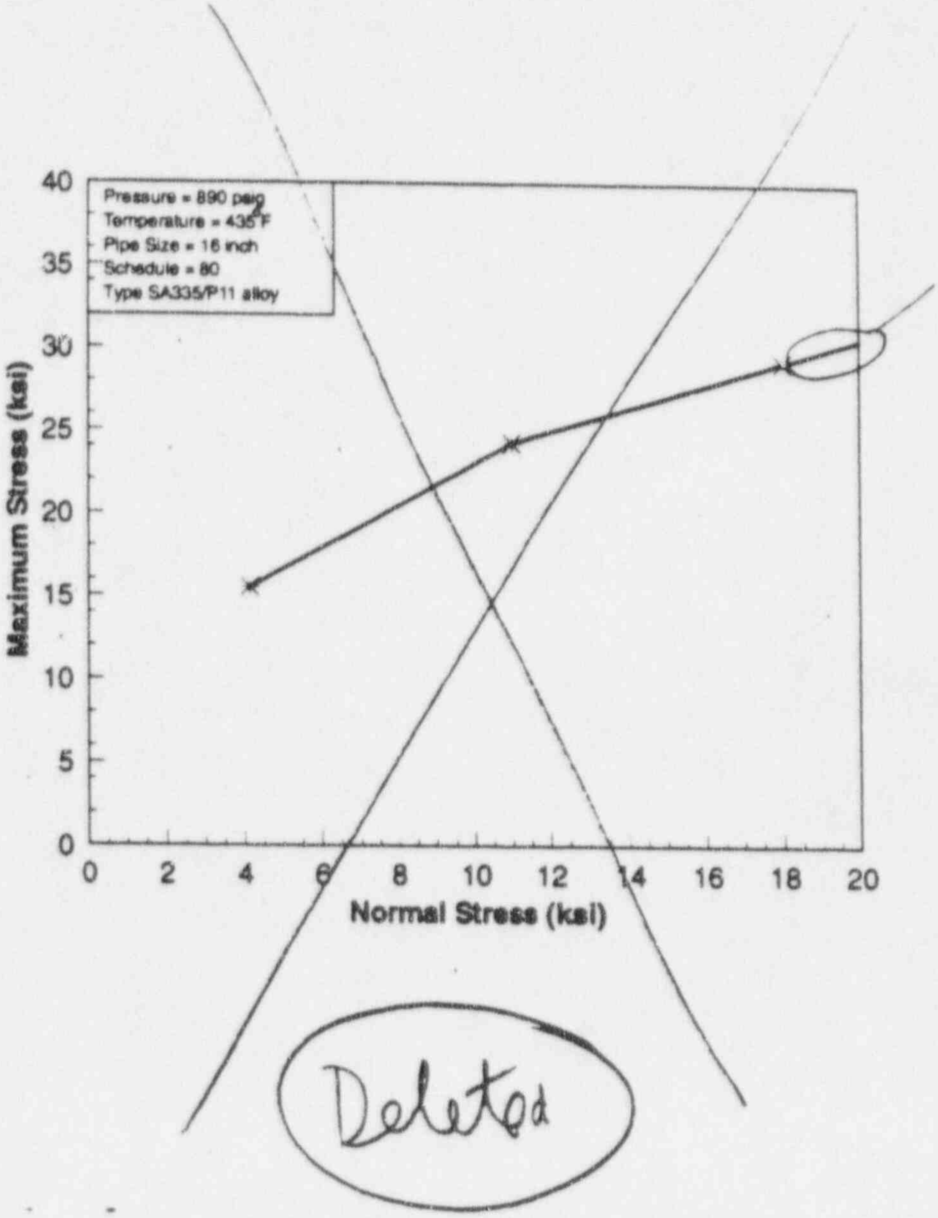

Figure 3B-10

Bounding Analysis Curve for Feedwater Line

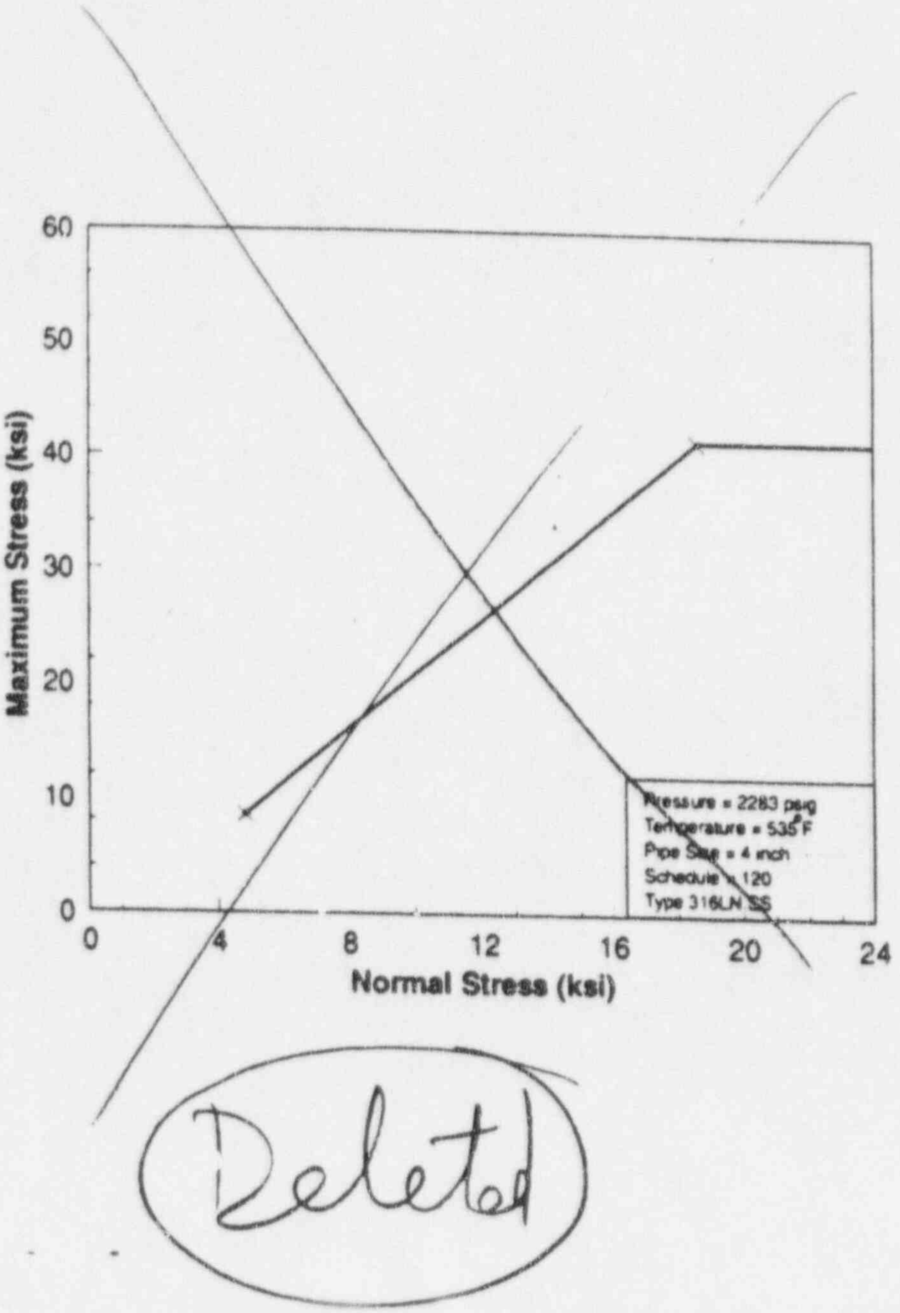

Figure 3B-36

Bounding Analysis Curve for 4" Spray

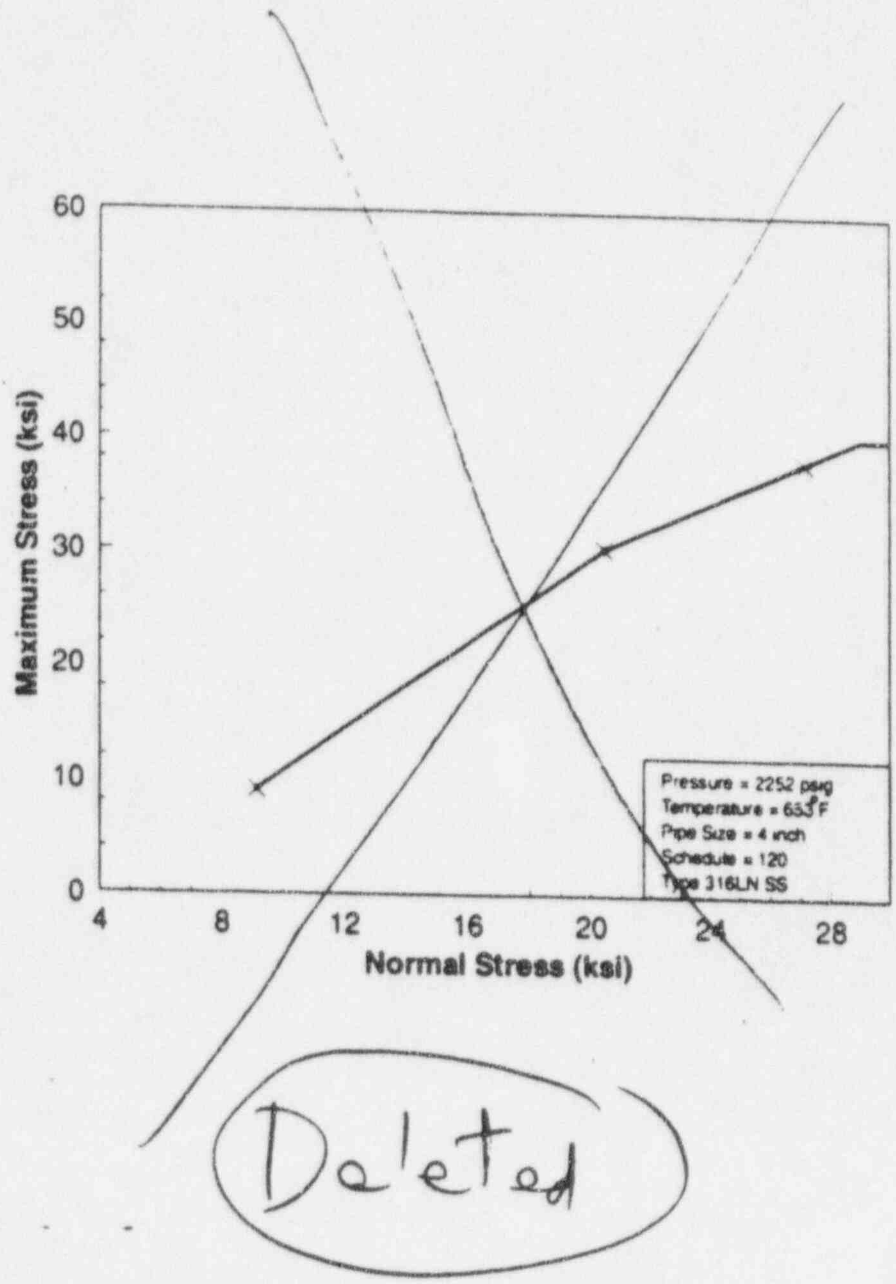

Figure 3B-37

Bounding Analysis Curve for 4" Spray

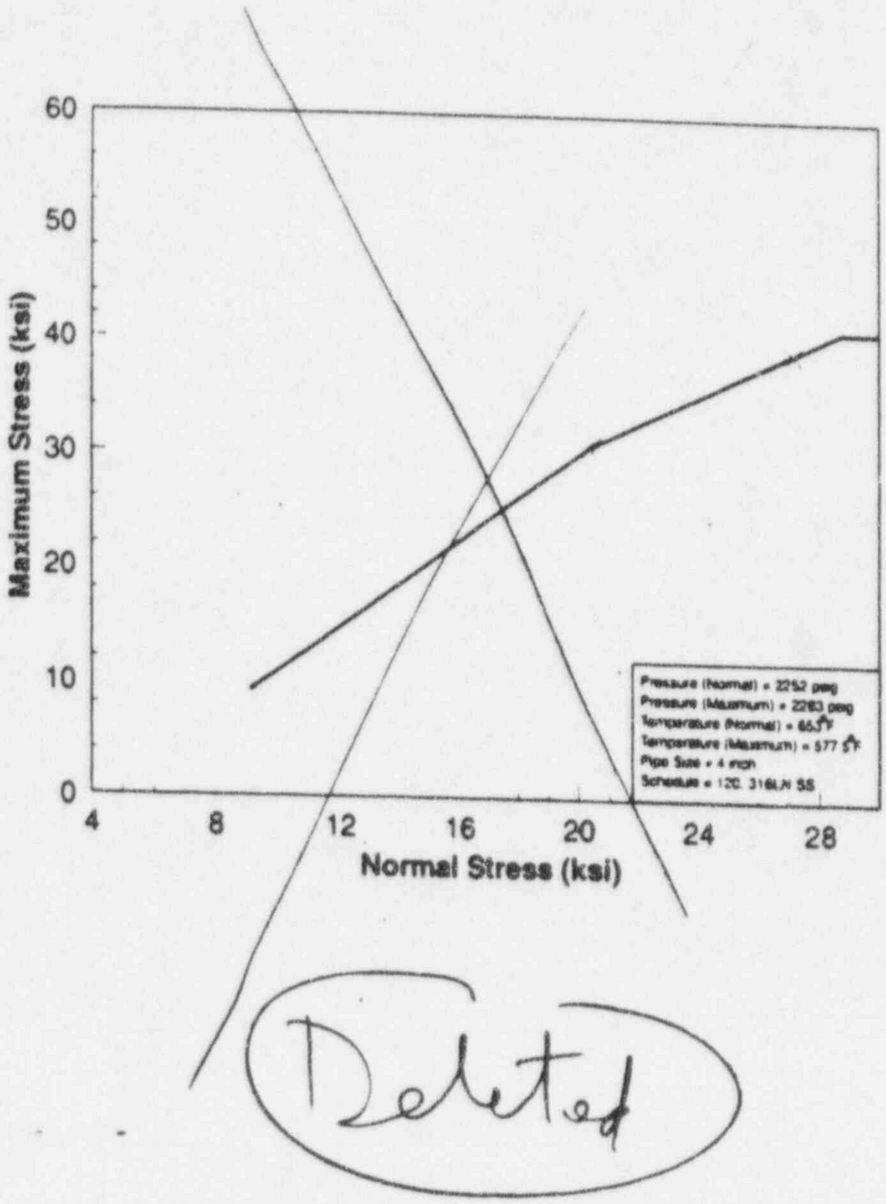

Figure 3B-38

Bounding Analysis Curve for 4" Spray

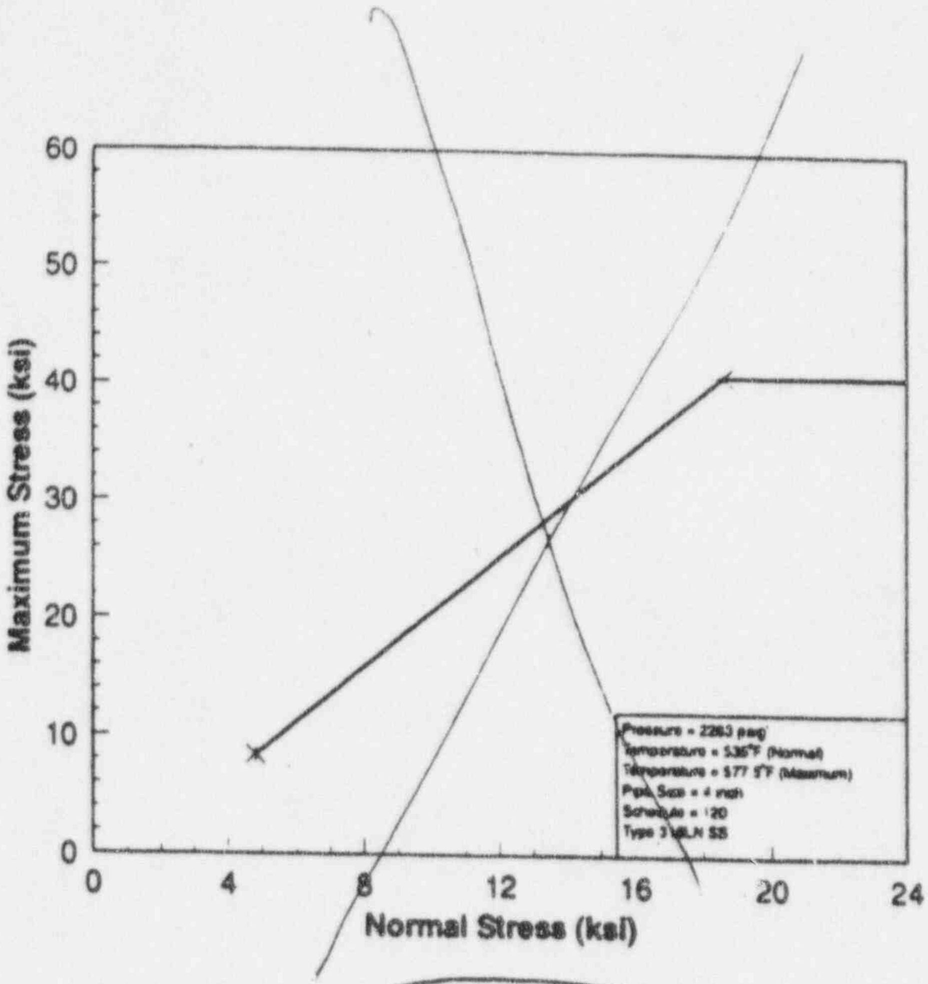

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Figure 3B-39

Bounding Analysis Curve for 4" Spray

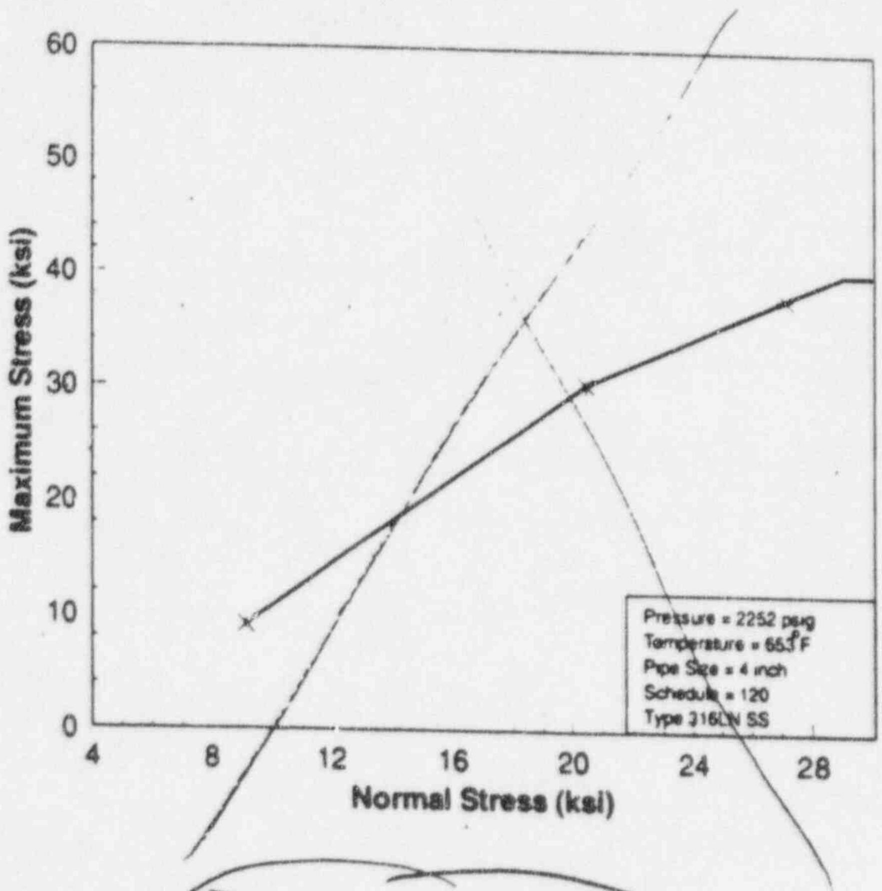

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Figure 3B-40

Bounding Analysis Curve for 4" ADS Stage 1

APPENDIX 3E

HIGH-ENERGY PIPING IN THE NUCLEAR ISLAND

This appendix identifies high-energy piping in the nuclear island with a diameter larger than 1 inch. Candidate leak-before-break piping is identified in Figures 3E-1 through 3E-5 along with other piping for which high-energy pipe failures are postulated. These figures also identify piping in the break exclusion zones outside containment. These figures do not include piping of 1 inch size and smaller. Instrumentation and instrumentation lines are not included.

In addition to the high-energy pipe identified in the figures, the hot water heating system (VYS) includes a limited amount of high-energy piping in the auxiliary building. The subject piping is the 3 inch-diameter supply and return header piping for the heating coils in HVAC equipment in the auxiliary building. ~~Table 3.6-2 identifies the compartments in which these lines are located.~~ There are no anchors or fittings on these lines in the nuclear island. Therefore, there are no postulated pipe breaks in these lines on the nuclear island.

The selection of the failure type is based on whether the system is high or moderate energy during normal operating conditions of the system. High-energy piping includes those systems or portions of systems in which the maximum normal operating temperature exceeds 200°F or the maximum normal operating pressure exceeds 275 psig. Piping systems or portions of systems pressurized above atmospheric pressure during normal plant conditions and not identified as high energy are considered moderate energy. Piping systems that exceed 200°F or 275 psig for 2 percent or less of the time during which the system is in operation or that experience high-energy pressures or temperatures for less than 1 percent of the plant operation time are considered moderate energy. In piping whose nominal diameter is greater than 1 inch but less than 4 inches, only circumferential breaks are postulated at each selected location. No breaks are postulated for piping whose nominal diameter is 1 inch or less.

The three-letter code included in the line numbering identifies the pipe specification. The letters define the pressure class, material specification, and AP600 equipment classification, respectively. The symbols used in Figures 3E-1 through 3E-5 are the same as the P&ID figures. See Figure 1.7-2 for additional information on the drawing legend and for the key for the pipe specification. Section 3.2 includes additional information on the AP600 equipment classification.

generators resulting from thermal expansion. ~~Section 3.6 addresses the applicability of leak-before-break to the main feedwater system, and branch lines.~~

The feedwater system and steam generator design minimize the potential for waterhammer and subsequent effects. Details are provided in Subsection 5.4.2.2 ~~and Appendix 3B~~. Feedwater piping analysis considers the following factors and events in the evaluation:

- Steam generators with top feed ring design (BTP ASB 10-2)
- Main feedwater check valves due to line breaks (BTP MEB 3-1)
- Spurious isolation or feedwater control valve trips
- Pump trips
- Deaerator regulating flow control valve trip
- Local feedwater piping, anchors, supports, and snubbers, as applicable

Feedwater Isolation Valves

One MFIV is installed in each of the two main feedwater lines outside the containment and downstream of the feedwater control valve. The MFIVs are installed to prevent uncontrolled blowdown from the steam generators in the event of a feedwater pipe rupture. The main feedwater check valve provides backup isolation. In the event of a secondary side pipe rupture inside the containment, the MFIVs limit the quantity of high energy fluid that enters the containment through the broken loop and limit cooldown. The MFCV provides backup isolation to limit cooldown and high energy fluid addition.

Each MFIV is a bidirectional wedge type gate valve composed of a valve body that is welded into the system pipeline. The MFIV gate valve is provided with a hydraulic/pneumatic actuator. The valve actuator is supported by the yoke, which is attached to the top of the body. The valve actuator consists of a hydraulic cylinder with a stored energy system to provide emergency closure of the isolation valve. The energy to operate the valve is stored in the form of compressed nitrogen contained in one end of the actuator cylinder. The MFIV is maintained in a normally open position by high-pressure hydraulic fluid. For emergency closure, redundant solenoids are energized resulting in the high-pressure hydraulic fluid being dumped to a fluid reservoir.

The feedwater isolation functional diagram is shown in Figure 7.2-1. To provide safety function actuation, the redundant actuation solenoid valves are powered from separate Class 1E power divisions. Redundant control and indication channels are provided for each of the isolation valves. Provisions are made for inservice inspection of the isolation valves.

Feedwater Control Valves

The MFCVs are air-operated control valves with the dual purpose of controlling feedwater flow rate as well as providing backup isolation of the feedwater system. The valve body is a globe design. Seats and trim are of an erosion resistant material. The design allows for removal and replacement of seats and other wearing parts.

- Section 3.2 delineates the quality group classification and seismic category applicable to the safety-related portion of this system and supporting systems. The controls and power supplies necessary for the safety-related functions of the condensate and feedwater system are Class 1E, and are described in Chapters 7 and 8.

- For a main feedwater line break inside the containment or a main steam line break, the MFIVs and the main feedwater control valves automatically close upon receipt of a feedwater isolation signal. The signals that produce a main feedwater isolation signal are identified and discussed in subsection 7.3.1.2.6.

- The MFIVs are provided with solenoids supplied by redundant power divisions. Failure of either of the power divisions or solenoids does not prevent closure of the MFIV. Releases of radioactivity from the condensate and feedwater system, resulting from the main feedwater line break, are minimal because of the negligible amount of radioactivity in the system under normal operating conditions. Following a steam generator tube rupture, the main steam isolation system and the passive residual heat removal heat exchanger reduce accidental releases, as discussed in Section 10.3 and Chapter 15. Detection of radioactive leakage into and out of the system is facilitated by area radiation monitoring (described in subsection 12.3.4), process radiation monitoring (described in Section 11.5), and steam generator blowdown sampling (described in subsection 10.4.8).

- For a steam generator tube rupture event, positive and redundant isolation is provided for the main feedwater (MFIV and MFCV) with isolation signals generated by the protection and safety monitoring system (PMS). Refer to subsection 7.3.1.2.6.

- Prevention and mitigation of feedline-related water hammer is accomplished through operation of the feedwater delivery system as described in subsection 5.4.2.2. The feedwater piping at the steam generators is sloped so that it does not drain into the steam generators. These features help avoid the formation of a steam pocket in the feedwater piping which, when collapsed, could create a hydraulic instability.

10.4.7.4 Tests and Inspections

10.4.7.4.1 Preservice Valve Testing

The MFIVs and feedwater control valves are checked for closing time prior to initial startup.

10.4.7.4.2 Preservice Pipe Testing

The main feedwater lines from the steam generator to the anchor at the interface between the turbine building and the auxiliary building are classified as ASME Code, Section III, Class 2 and 3 and seismic Category I piping. ~~The portion of these lines in the containment meet the leak-before-break criteria described in Section 3.6~~. The Class 2 portions of the main feedwater system piping are tested and inspected to the requirements of ASME Code, Section III, Class 2 piping ~~as described in subsection 3.6.3.2~~. The portion of the piping between the containment penetration and the anchor, which is considered as the break

shown on the diagrams titled "Feedwater Isolation" and "Steamline Isolation" in Figure 7.2-1.

- For a steam generator tube rupture event, positive and redundant isolation is provided for the startup feedwater system (startup feedwater isolation valve and startup feedwater control valve) to prevent steam generator overfill, with engineered safeguards isolation signals generated by the protection and safety monitoring system (PMS).

10.4.9.4 Tests and Inspections

10.4.9.4.1 Preservice Valve Testing

The startup feedwater isolation valves and startup feedwater control valves are checked for closing time prior to initial startup.

10.4.9.4.2 Preservice Pipe Testing

The Class 2 portion of the startup feedwater system piping is tested and inspected to the requirements of ASME Code, Section III, Class 2 piping ~~as described in subsection 3.6.3.2~~. In addition, the portion of the piping between the containment penetration and the anchor, which is traditionally considered as the break exclusion zone described in subsection 3.6.2, is subjected to 100-percent volumetric inspection at installation (that is, 100-percent volumetric examination of shop and field longitudinal and circumferential welds).

10.4.9.4.3 Preoperational System Testing

Preoperational testing of the startup feedwater system is performed as described in Chapter 14.

10.4.9.4.4 Inservice Inspections

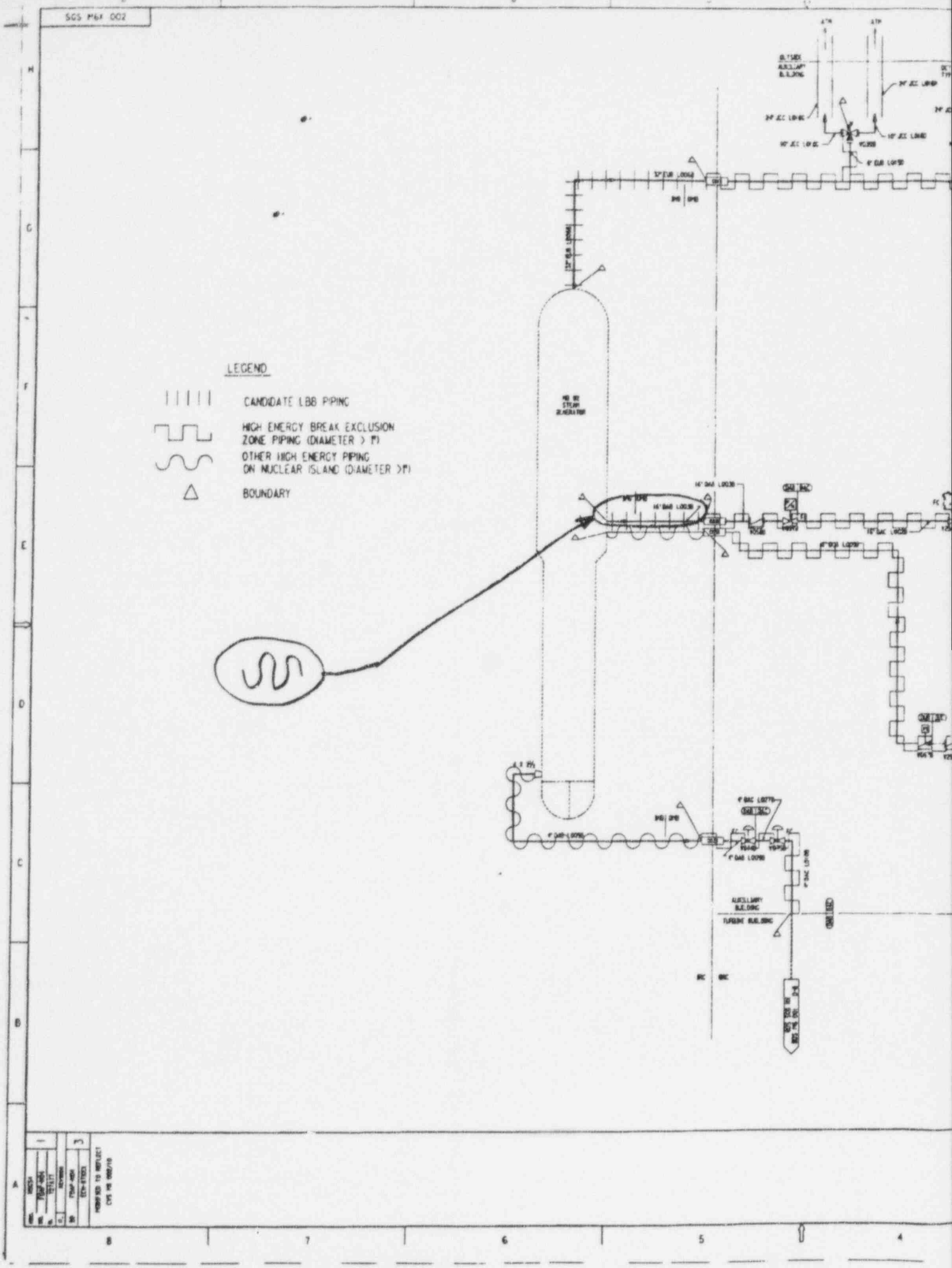
The performance and structural and leaktight integrity of the startup feedwater system components are demonstrated by normal operation.

The inservice inspection program for ASME Section III Class 2 and 3 components is described in Section 6.6. The inservice testing program, including testing for the startup feedwater isolation valve and startup feedwater control valve, is described in subsection 3.9.6.

10.4.9.5 Instrumentation Applications

The startup feedwater system instrumentation is designed to facilitate automatic operation, remote control, and continuous indication of system parameters.

The startup feedwater flow is controlled by a steam generator level demand signal modulating the startup feedwater control valve. The control valve may either be in manual or automatic control. Refer to Section 7.7. The startup feedwater flow transmitters also provide redundant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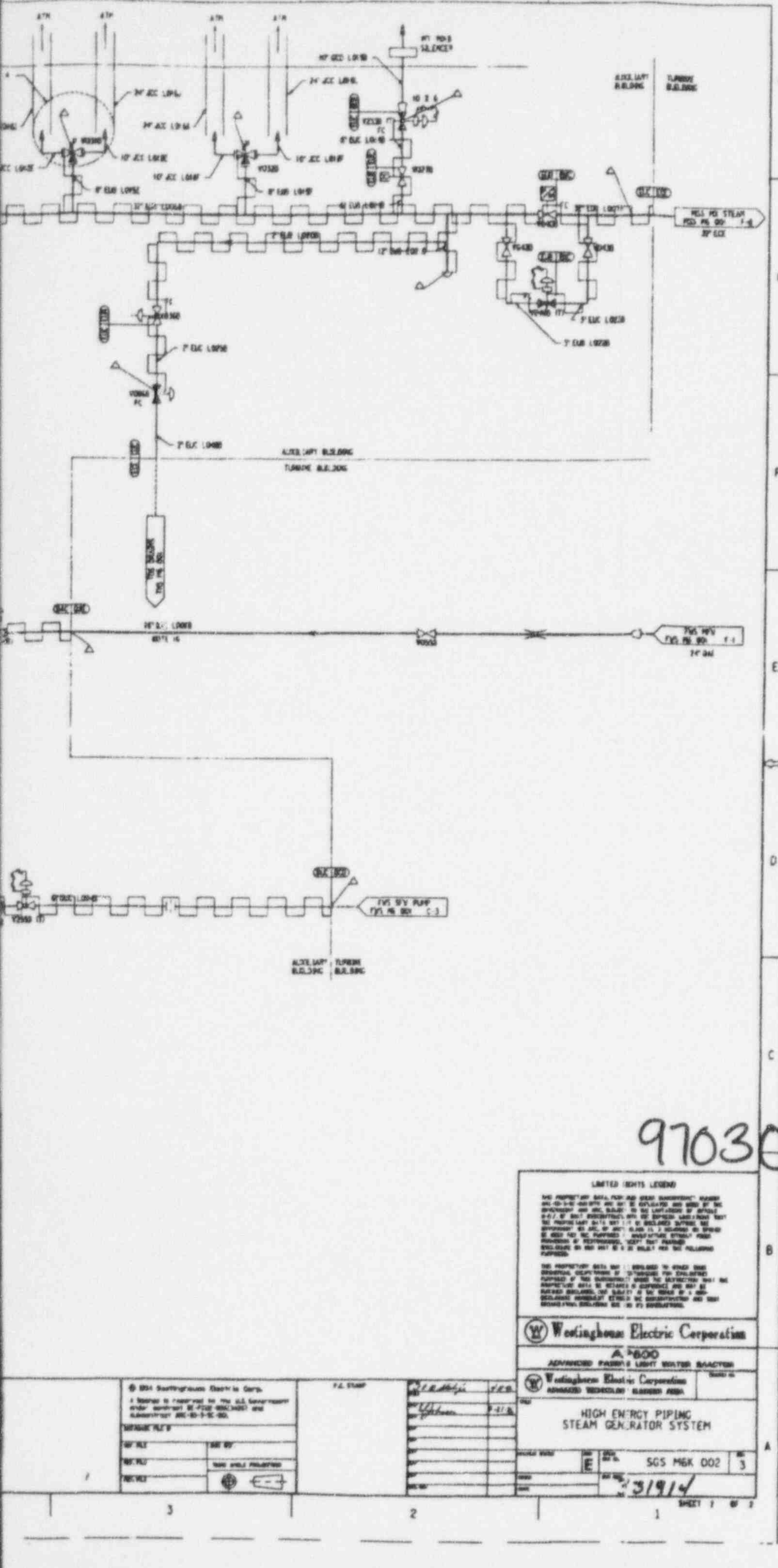
CANDIDATE LBB PIPING
HIGH ENERGY BREAK EXCLUSION ZONE PIPING (DIAMETER > 1")
OTHER HIGH ENERGY PIPING ON NUCLEAR ISLAND (DIAMETER >1")
BOUNDARY
AUXILIARY BUILDING
TURBINE BUILDING
REVISED TO REFLECT



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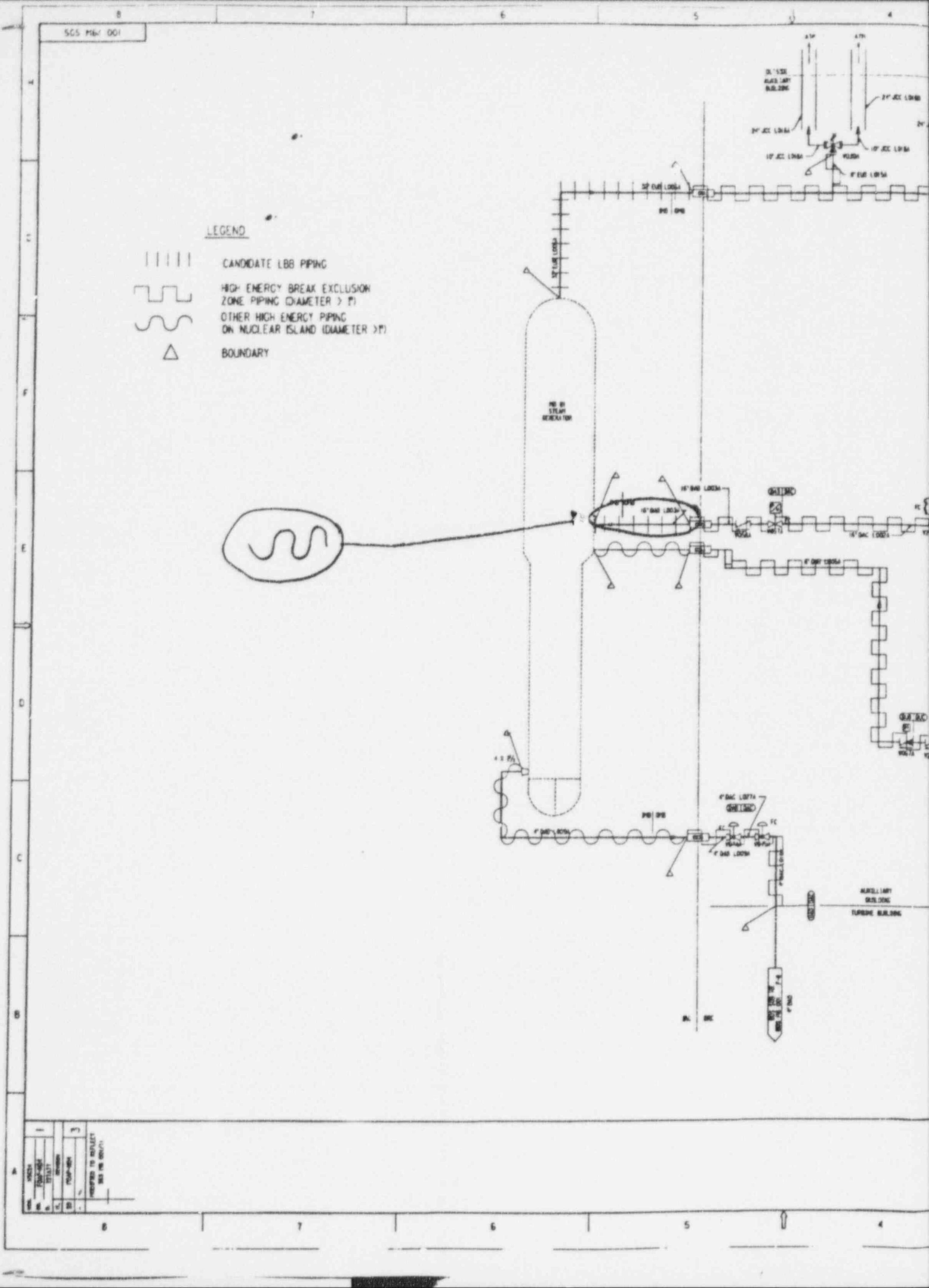
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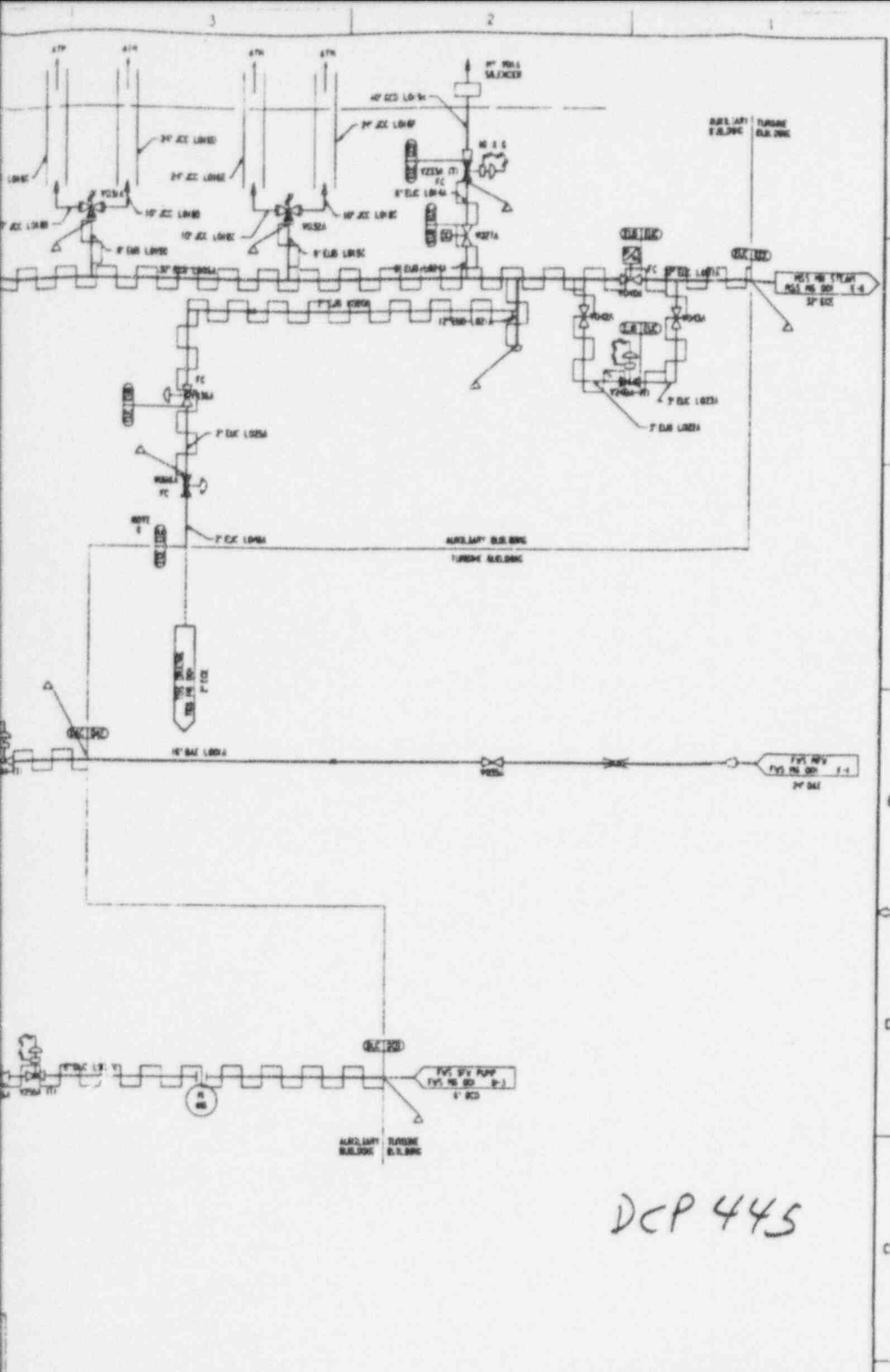
BOUNDARY
STEAM GENERATOR
AUXILIARY BUILDING
TURBINE BUILDING



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STEAM GENERATOR SYSTEM

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SGS MGK 001

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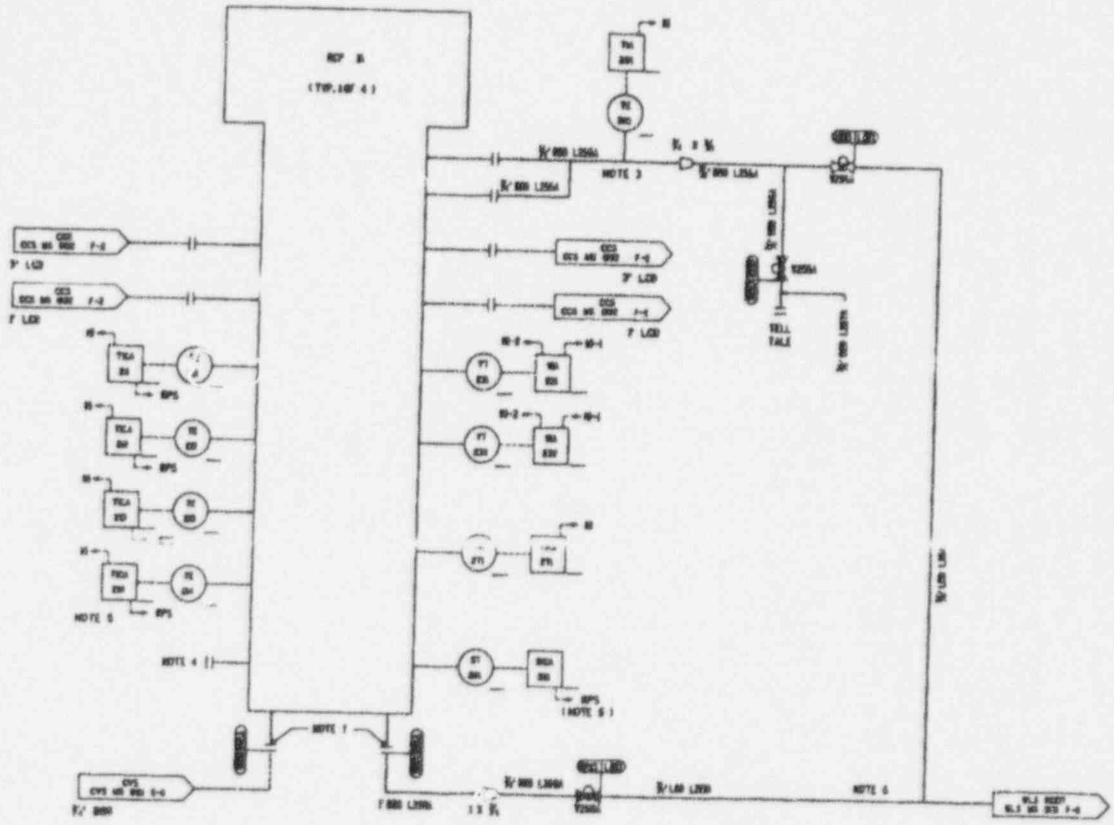
RCS M6K 003



REACTOR COOLANT PUMP INSTRUMENTATION					
	RCP A	RCS B	RCP C	RCP D	SENSOR SUPPLIED WITH PUMP
BEARING WATER TEMPERATURE (TE -)	211 212 213 214	215 216 217 218	221 222 223 224	225 226 227 228	YES YES YES YES
VIBRATION (YT -)	231 232	241 242	251 252	261 262	YES YES
STATOR TEMPERATURE (TE -)	271	272	273	274	YES
PUMP SPEED (ST -)	281	285	[illegible]	[illegible]	YES
FLANGE LEAK-OFF TEMPERATURE (TE -)	201	202	203	204	NO

3 2 1

H G F E D C B A

ANSTEC APERTURE CARD

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EQUIPMENT NUMBERS, LINES, VALVES, INSTRUMENTS, ... ETC) ARE SHOWN WITHOUT PREFIX RC.

... TO SYSTEM DESCRIPTION FOR DESCRIPTION OF ALL ...UMENTATION, CONTROLS, AND INTERLOCKS.

...AP-ON RTD LOCATED AT BOTTOM OF PIPE.

...LL BLIND FLANGE ON LEAK-OFF CONNECTION FROM STATOR CAP

... TO SYSTEM SPECIFICATION DOCUMENT FOR DESCRIPTION OF ...NSTRUMENTATION, INTERLOCKS AND CONTROLS.

... CONTINUOUSLY DOWNWARD TO RCDT.

...LOW RESTRICTOR FOR CLASS 1 TO CLASS 2 TRANSITION MACHINED ...PIPING FLANGE SUPPLIED WITH PUMP.

... LETTERS FOR THE FOLLOWING ITEMS ARE A, B, C AND D, ...SPONDING TO PUMPS A, B, C AND D RESPECTIVELY

VALVES	LINES	
255	254	258
256	265	259
260	256	260
	257	261

LEGEND

CANDIDATE LBB PIPING

OTHER HIGH ENERGY PIPING ON NUCLEAR ISLAND (DIAMETER >?)

BOUNDARY

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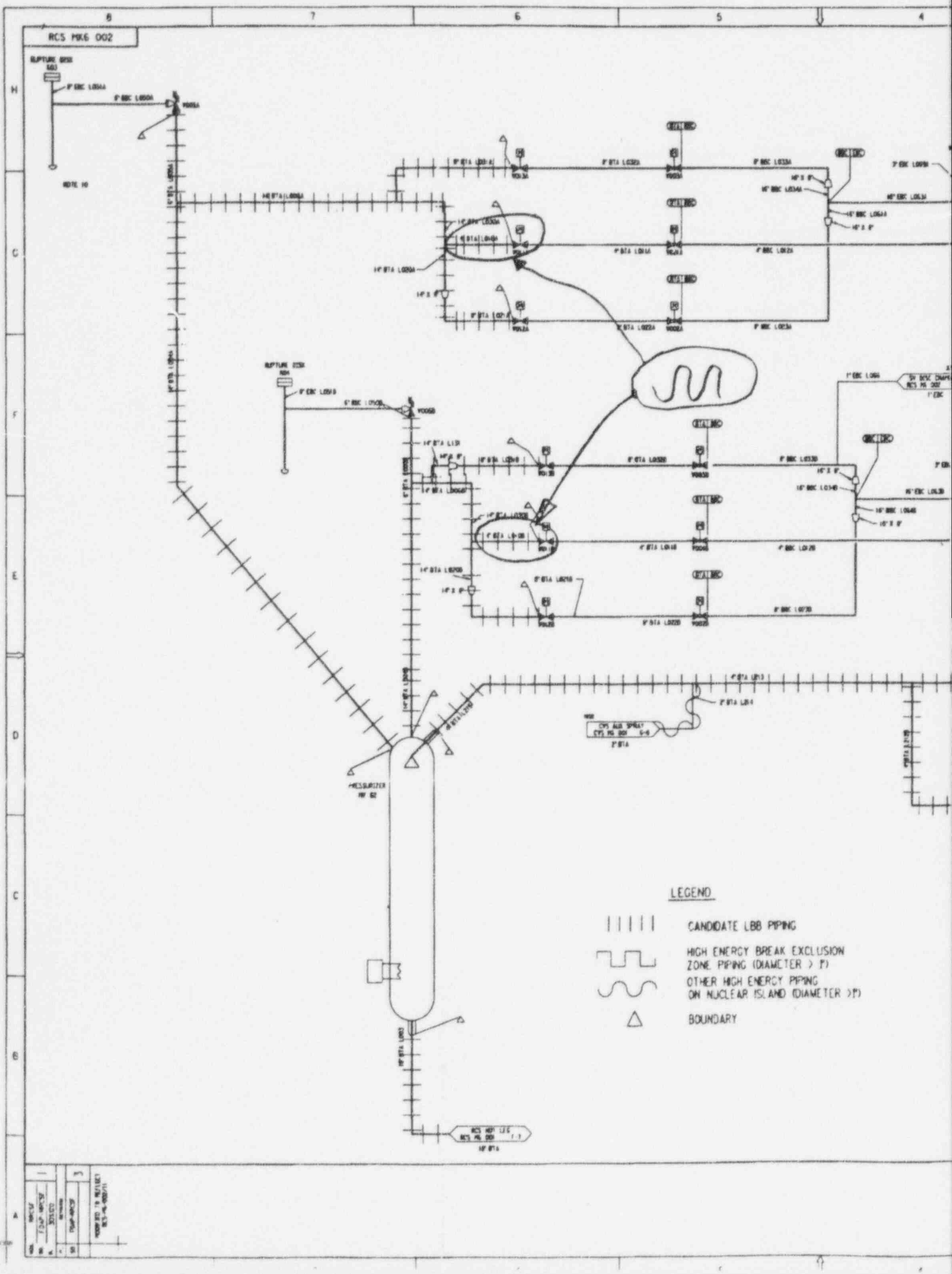
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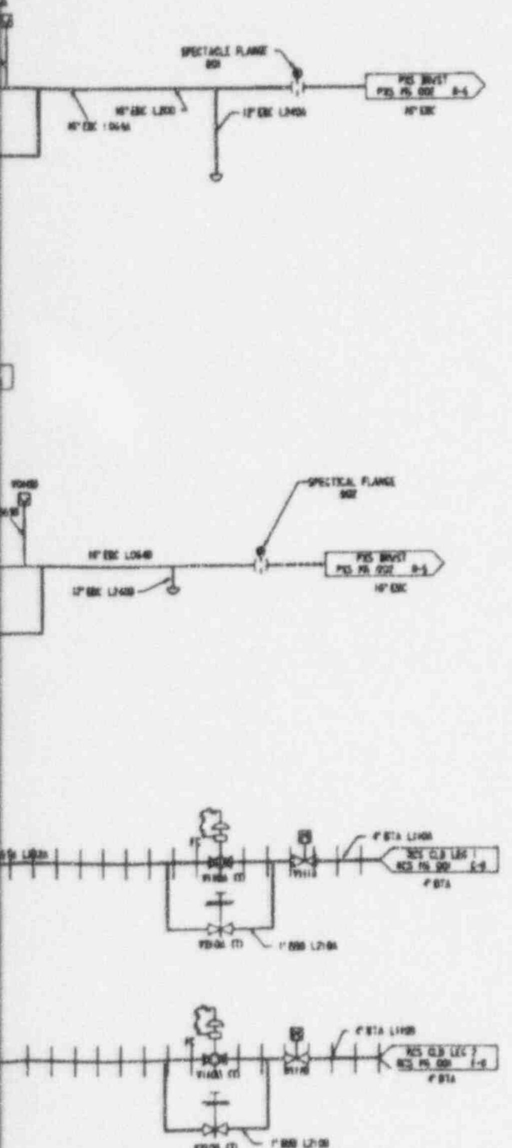
LEGEND
CANDIDATE LBB PIPING
HIGH ENERGY BREAK EXCLUSION
ZONE PIPING (DIAMETER > 1")
OTHER HIGH ENERGY PIPING
ON NUCLEAR ISLAND (DIAMETER >1")
BOUNDARY
PRESSURIZER
RCS HOT LEG



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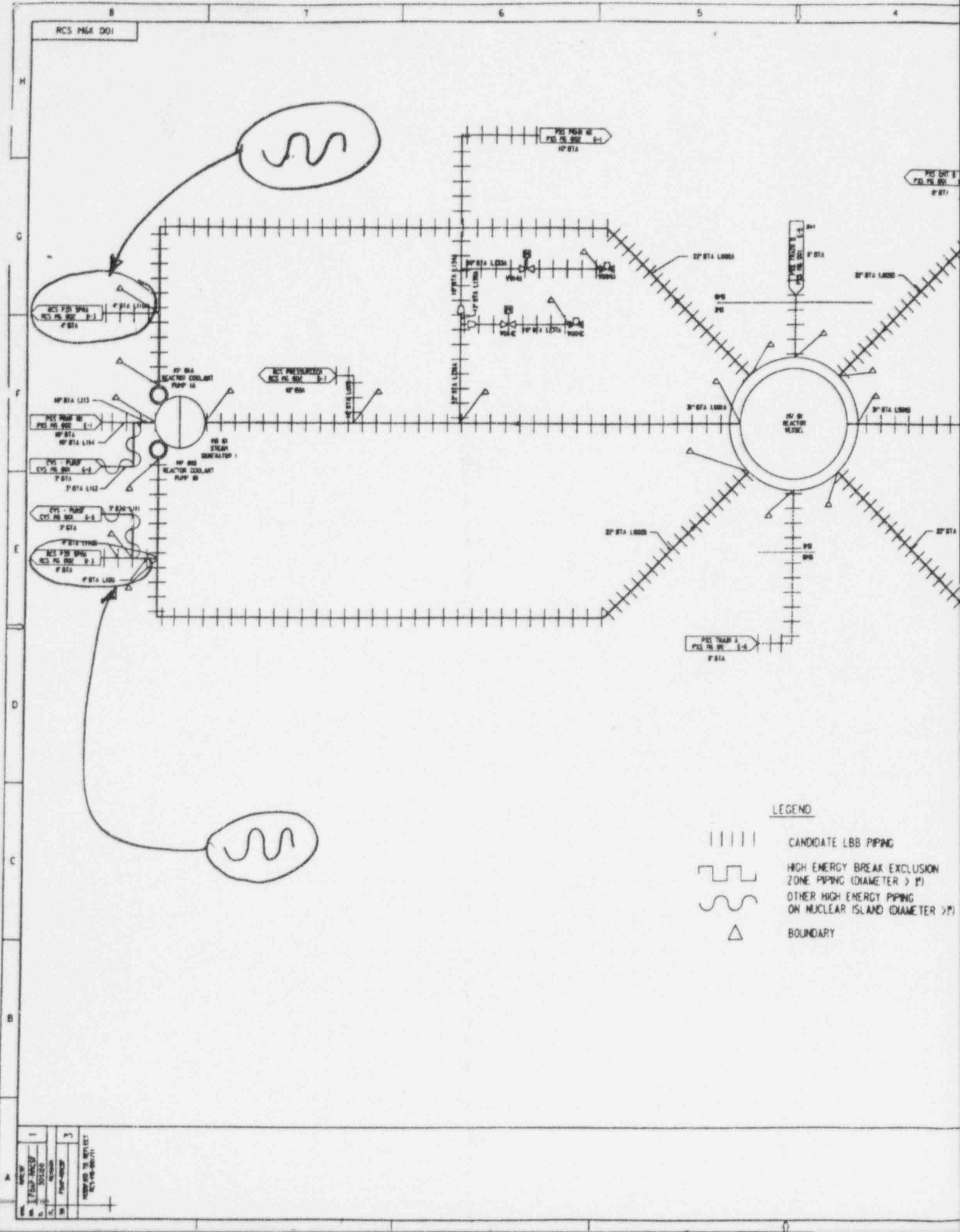
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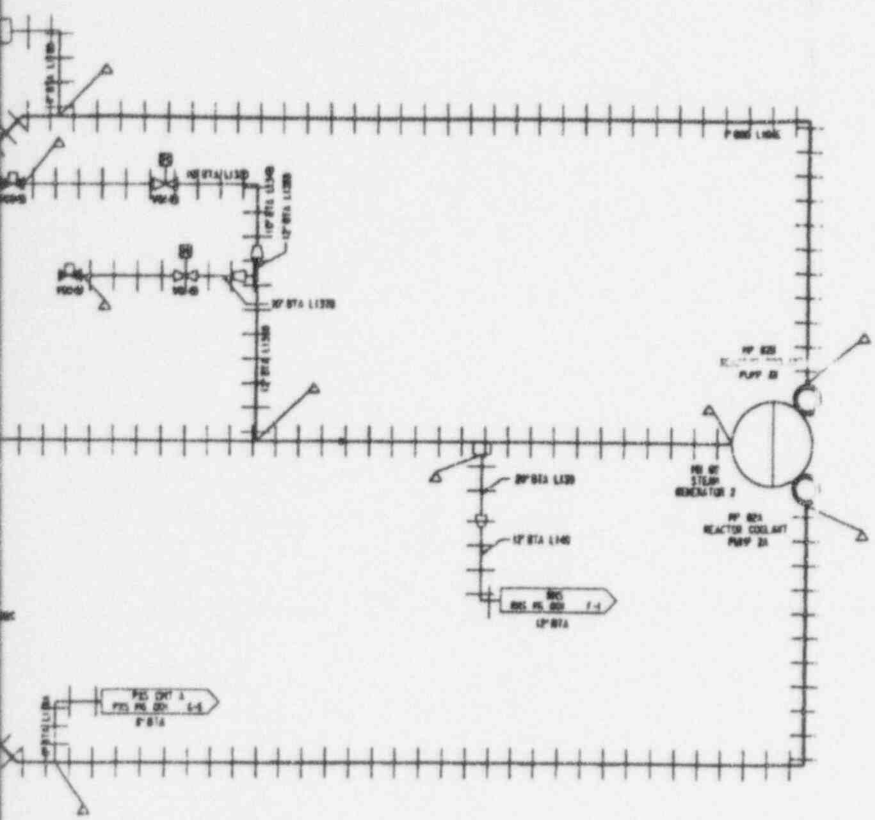

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FEB - 6 1997

Brian A. Mc Intyre

COVER SHEET FAX

To: Bill Huffman (NRC), Larry Hochreiter (PSU)
cc: B. McIntyre (Informal NRC File), R. Kemper, D. Garner, M. Young, File 7.6
Subject: NOTRUMP and WC/T
Date: February 4, 1997
Pages: Three, including this cover sheet.

COMMENTS:

Bill,

Attached are the following which were discussed in Monday's telecon.

> Attachment I - Brief evaluation of NOTRUMP ranging questions that were raised by Ralph. We need to discuss more with Ralph to better understand his concerns.
>
> Attachment II - Draft Agenda for WC/T LTC ACRS meeting. We should talk more next Monday on this.

Please give copies to Ralph and Lambrose. Thanks.

Earl

From the desk of...

Earl H. Novendstern
Manager, Advanced and VVER Plant Safety Analysis
Westinghouse
PO Box 355
Pittsburgh, PA 15235

(412) 374 -4790
Fax: (412) 374-5744

Attachment I - NOTRUMP Ranging

NOTRUMP V&V Report Section	RAI Number 440.xxx	Comments
2.6 Contact Coefficients	471	Explanation, no commitment to put into V&V report, no commitment to check adequacy of ranges.
2.7 Internally Calculated Liquid Reflux Flow Links	472	Model in original 1985 V&V report, explanation with no commitment to check adequacy of ranges.
2.15 Critical Heat Flux	480	Nothing in report as to comparison with test data. RAI indicated that refinements made by Bjornard & Griffith may be incorporated into NOTRUMP and included in final V&V report.
2.16 Two-Phase Friction Multiplier	481	Nothing in report as to how compares with data below 250 psi. However, a comparison exists in RAI response.
2.19 Transition Boiling Correlation	484	RAI asked to provid" PCT calc to show effect of changes to transition boiling correlation on PCT. RAI response stated orig. 1985 NOı RUMP still applicable.

Attachment II

LTC ACRS Agenda (Draft - 12/23/96 - Rev. 0)

I. Introduction (LEH)
   A. Why WC/T
   B. BOP vs. AP600 LTC - Equipment Pedigree

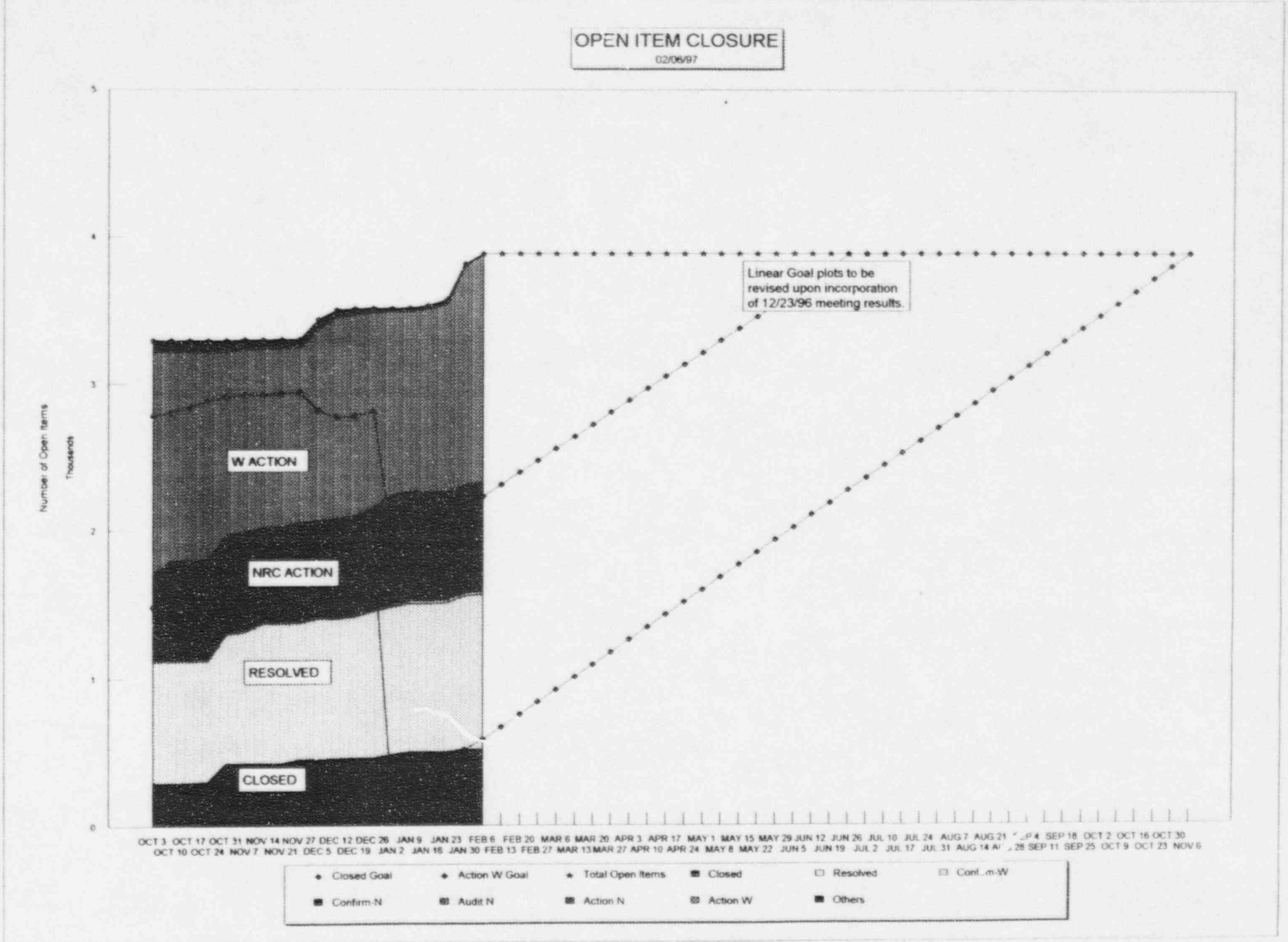
II. PIRT (LEH)

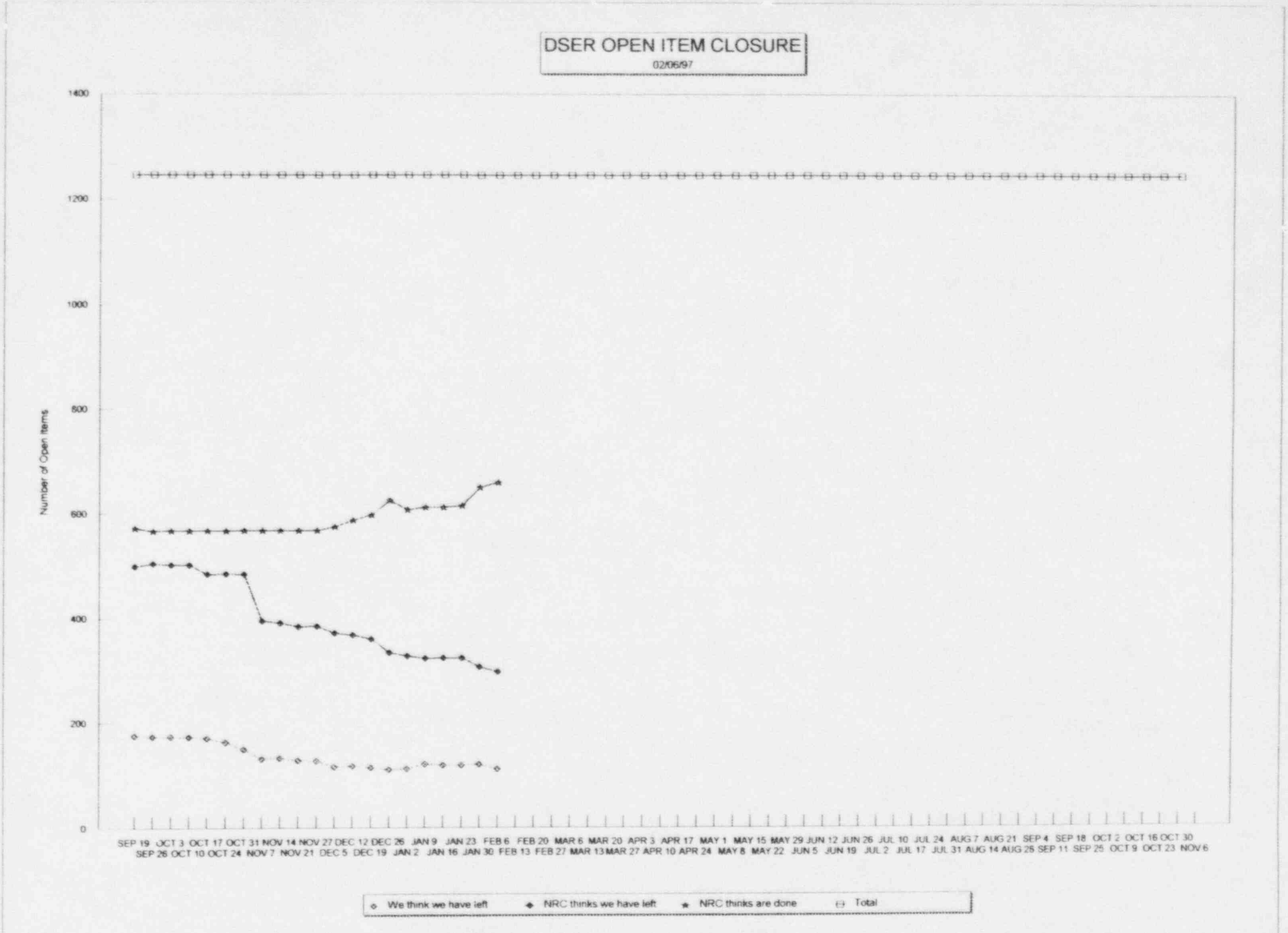
III. WC/T Model (RMK/DCG)
   A. SAR
      1. RCS/Containment Coupling (Completely integrated story)
   B. OSU
   C. Model Improvements (?) (Section 4)

IV. Window Modes (Section 3) (DCG)
   A. Description
   B. Transient Description using data
   C. WC/T Sensitivity Studies

V. W/CT Analyses vs. Data (Section 5) (DCG)
   A. Results
   B. PIRT comparison

VI. Summary (LEH)


OPEN ITEM CLOSURE
02/06/97
Number of Open Items
Thousands
0
1
2
3
4
5
W ACTION
NRC ACTION
RESOLVED
CLOSED
Linear Goal plots to be revised upon incorporation of 12/23/96 meeting results.
OCT 3 OCT 17 OCT 31 NOV 14 NOV 27 DEC 12 DEC 26 JAN 9 JAN 23 FEB 6 FEB 20 MAR 6 MAR 20 APR 3 APR 17 MAY 1 MAY 15 MAY 29 JUN 12 JUN 26 JUL 10 JUL 24 AUG 7 AUG 21 SEP 18 OCT 2 OCT 16 OCT 30
OCT 10 OCT 24 NOV 7 NOV 21 DEC 5 DEC 19 JAN 2 JAN 16 JAN 30 FEB 13 FEB 27 MAR 13 MAR 27 APR 10 APR 24 MAY 8 MAY 22 JUN 5 JUN 19 JUL 2 JUL 17 JUL 31 AUG 14 SEP 11 SEP 25 OCT 9 OCT 23 NOV 6
Closed Goal
Action W Goal
Total Open Items
Closed
Resolved
Confirm-N
Audit N
Action N
Action W
Others


DSER OPEN ITEM CLOSURE
02/06/97
Number of Open Items
1400
1200
1000
800
600
400
200
0
SEP 19 OCT 3 OCT 17 OCT 31 NOV 14 NOV 27 DEC 12 DEC 26 JAN 9 JAN 23 FEB 6 FEB 20 MAR 6 MAR 20 APR 3 APR 17 MAY 1 MAY 15 MAY 29 JUN 12 JUN 26 JUL 10 JUL 24 AUG 7 AUG 21 SEP 4 SEP 18 OCT 2 OCT 16 OCT 30
SEP 26 OCT 10 OCT 24 NOV 7 NOV 21 DEC 5 DEC 19 JAN 2 JAN 16 JAN 30 FEB 13 FEB 27 MAR 13 MAR 27 APR 10 APR 24 MAY 8 MAY 22 JUN 5 JUN 19 JUL 2 JUL 17 JUL 31 AUG 14 AUG 28 SEP 11 SEP 25 OCT 9 OCT 23 NOV 6
We think we have left
NRC thinks we have left
NRC thinks are done
Total

Westinghouse

FAX COVER SHEET

RECIPIENT INFORMATION		SENDER INFORMATION	
DATE:	February 6, 1997	Name:	Don Hutchings
TO:	Diane Jackson	LOCATION:	ENERGY CENTER - EAST
PHONE:		PHONE:	Office: 412-374-5109
COMPANY:	US NRC	Facsimile:	win: 284-5535 outside: (412)374-5535
LOCATION:			

Cover + Pages 1 + 4

The following pages are being sent from the Westinghouse Energy Center, East Tower, Monroeville, PA. If any problems occur during this transmission, please call:

WIN: 284-5489 (Wanda)/4031 (Delina) or Outside (412)374-5489/4031.

COMMENTS:

Diane,

The attached is our response related to how we ensure protection of the fire pumps and air compressors from a major fire in the turbine building. These questions were raised during our Jan. 7 1997 meeting.

WE WILL FOLLOW THIS WITH AN OFFICIAL LETTER ANSWERING YOUR LETTERS.

Jim

cc: Lindgren
McIntyre
Ron Vijuk
Winters

ISSUE

The NRC has voiced concerns relative to protecting the fire pumps which are located in the turbine building in the event of a large turbine building fire.

DISCUSSION

Postulated Loss of Fire Main in the Turbine Building

The fire protection main includes an underground yard loop and a yard main extension that distributes water to suppression systems and hose stations within the main plant buildings. If the yard main extension in the turbine building is damaged due to a fire or any other reason, the damaged portion of the main can be isolated using installed sectionalization valves. With the damaged portion isolated, the underground yard loop is still capable of supplying fire hydrants on the yard loop and supplying the other buildings connected to the yard main extension. Isolation of the yard main extension in the turbine building results in loss of the ability to fill the PCS storage tank. However, the inventory of the PCS tank remains available to supply hose stations in areas containing safety-related equipment.

Postulated Loss of Fire Pumps

The motor and diesel driven fire pumps are located in separate fire rated enclosures. These enclosures are designed to prevent the spread of fires within the turbine building to the fire pumps. The turbine building is provided with fire detection and suppression systems designed to detect fires, extinguish fires, and limit fire damage. With these systems functioning properly, a general turbine building fire that causes structural damage to the building and subsequent collapse resulting in loss of the fire pumps is considered unlikely.

Even if the fire detection and suppression systems do not function properly, it is unlikely that the turbine building structure would collapse due to fire. As the turbine building is designed for tornado and seismic loads, structural members are larger than required for other load conditions. The building is laterally braced in both the north-south and east-west directions. The structural steel members are designed for temperatures up to 1000°F. Allowable stress values for the steel members are reduced for the increased temperatures. The turbine building's large steel members would have to be exposed to very high temperatures for several hours before even minor warping or distortion might occur. This provides significant time for fire fighting activities.

The location of the fire pump rooms in the heavily braced northwest corner of the turbine building make it unlikely the fire pumps would be damaged even if building collapse did occur. The fire pumps are located on the basemat at el. 100'-0". The turbine building floors directly above the fire pumps are lightly loaded, i.e., the secondary sampling laboratory on el. 117'-6", the electrical switchgear room on el. 135'-3", and an open portion of the operating deck on el. 161'-0". There is no

heavy equipment located above the pumps that might cause the structure above the pump rooms to collapse during a fire

However, should a fire in the turbine building result in a loss of the fire pumps, this would not affect the ability of the plant to shut down. As the turbine building is separated from areas containing safety-related equipment by 3-hour rated fire walls, a fire in the turbine building is not postulated to result in fire damage in any area containing safety-related equipment. Following loss of the fire pumps, fire hose stations in safety-related areas will continue to be supplied from the PCS storage tank. The fire pumps and any damaged portions of the fire protection yard main extension in the turbine building can be isolated. It is anticipated that a fire pump truck could be used on a temporary basis to charge the fire protection main until the fire pumps were replaced.

ISSUE

The NRC has voiced concerns relative to protecting the high-pressure breathing compressor which is located in the turbine building in the event of a large turbine building fire.

REQUIREMENTS:

BTP CMEB 9.5-1, Paragraph C.3.c states in part:

> "At least 10 [self-contained breathing apparatus] masks shall be available for fire brigade personnel."
>
> "Service or rated operating life shall be a minimum of one-half hour for the self contained units."
>
> "At least two extra bottles should be located onsite for each self-contained breathing unit."
>
> "In addition, an onsite 6-hour supply of reserve air should be provided and arranged to permit quick and complete replenishment of exhausted supply air bottles as they are returned. If compressors are used as a source of breathing air, only units approved for breathing air shall be used; compressors shall be operable assuming a loss of offsite power."

10CFR50, App. R, Paragraph III.H. states in part:

> "At least 10 [self-contained breathing apparatus] masks shall be available for fire brigade personnel."
>
> "Service or rated operating life shall be a minimum of one-half hour for the self contained units."
>
> "At least a 1-hour supply of breathing air shall be located on the plant site for each self-contained breathing unit."
>
> "In addition, an onsite 6-hour supply of reserve air should be provided and arranged to permit quick and complete replenishment of exhausted supply air bottles as they are returned. If compressors are used as a source of breathing air, only units approved for breathing air shall be used and the compressors shall be operable assuming a loss of offsite power."

PRESENT AP600 SSAR STATEMENT:

AP600 SSAR Table 9.5.1-1, Fire Protection Program Compliance with BTP CMEB 9.5-1, Item 33 states in part:

> "A breathing air compressor is provided in the compressed and instrument air system (CAS) to replenish the exhausted air supply bottles used by the fire brigade."

PRESENT AP600 POSITION:

After reviewing the requirements of BTP 9.5-1 paragraph C.3.c and 10CFR50, App. R, paragraph III.H. AP600 considers that we are in compliance with these requirements.

AP600 goes beyond the minimum requirement which specifies "...an onsite 6-hour supply of reserve air should be provided..." and notes that there is no requirement that the 6-hour supply of reserve air be located in a place where a fire cannot occur.

AP600 provides a breathing-air compressor and an air receiver in the turbine building in order to replenish exhausted SCBA air bottles in the event of a fire. This design should be acceptable as it is capable of providing far more air than the required 6-hour supply.

However, in order to minimize concerns relative to a large turbine building fire damaging the "6-hour supply of reserve air" (i.e. the breathing-air compressor and an air receiver), AP600 proposes to utilize the self-contained compressed breathable air bottles stored inside the MCR pressure boundary to provide up to six additional hours of breathable air for up to eleven people. - See SSAR 6.4, section 5.4.2 Breathing Apparatus relative to control room habitability.

PROPOSED AP600 SSAR STATEMENT:

AP600 SSAR Table 9.5.1-1, Fire Protection Program Compliance with BTP CMEB 9.5-1, Item 33 states in part:

> "A breathing air compressor *and receiver* is provided in the compressed and instrument air system (CAS) to replenish the exhausted air supply bottles used by the fire brigade. *Additionally, an equivalent 6-hour supply of reserve air will be maintained in an area located outside of the turbine building. (e.g. the 6-hours of compressed breathable air bottles stored inside the MCR pressure boundary for up to 11 people.)*

FAX to DINO SCALETTI

February 6, 1997

CC: Sharon or Dino, please make copies for: Tom Kenyon
Ted Quay

Don Lindgren
Jim Grover
Gordon Israelson
Brian McIntyre
Ed Cummins
Bob Vijuk

OPEN ITEMS FOR CHAPTER 12

This is a background package for the remaining open items for Chapter 12. Chapter 12 is of interest because by our joint NRC/W schedule, the FSER for this chapter should have been turned into Projects by the end of February. Attached are copies of some of the relevant documentation related to the two remaining open items (21 and 1210). Open Item #21 requires that Westinghouse submit an exemption request for not providing criticality monitors near the spent and new fuel racks. This letter is currently in internal review and expected to be issued Friday, February 7. Open Item #21 will show "Action W" for both NRC and Westinghouse status until the letter is issued. Open Item #1210 requested information about the shielding associated with the gap around the fuel transfer tube. A revision to SSAR subsection 12.3.2.2.9 was issued on April 30, 1996 to specifically address this request. This was a straightforward change. It seems a reasonable request that NRC acknowledge receipt of the change. Our records show no other outstanding Westinghouse action on this Chapter (12) and we request that NRC provide a definitive action for Westinghouse or provide direction to change the status of Item #1210. We recommend "Action N". Thank you.

Jim

Jim Winters
412-374-5290

AP600 Open Item Tracking System Database: Executive Summary

Date: 2/6/97

Selection: [nrc st code]='Action W' And [DSER Section] like '12*' Sorted by Item #

Item No.	Branch	DSER Section/ Question	Type	Title/Description Detail Status	Resp Engineer	(W) Status	NRC Status	Letter No. / Date
21	NRR/PERB	12	RAI-OI	Question 471.24 (Radiation Protection) Section 11.5.6 of Chapter 11 of the SSAR states that criticality monitors as required in 10 CFR 70.24 and Regulatory Guide 8.12 are not provided because the design of the fuel pool racks precludes criticality under postulated normal and accident conditions. Justify why criticality monitors are not required and state the requirements that the COL applicant will need to fulfill this requirement. Closed - Response provided by NSD-NRC-96-4727. Action W - Exemption request required.	Winters	Action W	Action W	NSD-NRC-96-4727
1210	NRR/PERB	12.4.2-2	DSER-OI	The 5.08 cm (2 in) air gap between the fuel transfer tube shielding and the containment vessel is a possible source for radiation streaming. Closed - Addressed in SSAR section 12.3.2.2.9, Revision 7.	Butler/Sejvar/Johnson, F.	Closed	Action W	

2 of 3

 Total Records: 2

12.3.2.2.9 Spent Fuel Transfer Canal and Tube Shielding

The spent fuel transfer tube is shielded to within adjacent area radiation zone limits. This is primarily achieved through the use of concrete and water. The only removable shielding consists of concrete or steel hatches which reduce radiation in accessible areas to within those levels prescribed in the normal operation radiation zone maps (Figure 12.3-1).

The spent fuel transfer tube is completely enclosed in concrete and there is no unshielded portion of the spent fuel transfer tube during the refueling operation. The only potential radiation streaming path associated with the tube shielding configuration is the 2 inch (5.08 cm) seismic gap between the fuel transfer tube shielding and the steel containment wall. Shielding of this gap is provided by a water-filled bladder. This "expansion gap" radiation shield provides effective reduction of the radiation fields during fuel transfer and accommodates relative movement between the containment and the concrete transfer tube shielding with no loss in shield integrity. A removable hatch in the shield configuration provides access for inspection of the fuel transfer tube welds. The opening of this hatch is administratively controlled; this hatch is in place during the spent fuel transfer operation.

12.3.2.3 Shielding Calculational Methods

The shielding thicknesses provided for compliance with plant radiation zoning and to minimize plant personnel exposure are based on maximum equipment activities under the plant operating conditions described in Chapter 11 and Section 12.2. The thickness of each shield wall surrounding radioactive equipment is determined by approximating as closely as practicable the actual geometry and physical condition of the source or sources. The isotopic concentrations are converted to energy group sources using data from standard references (References 1 through 6).

The geometric model assumed for shielding evaluation of tanks, heat exchangers, filters, ion exchangers, and the containment is a finite cylindrical volume source. For shielding evaluation of piping, the geometric model is a finite shielded cylinder. In cases where radioactive materials are deposited on surfaces such as pipe, the latter is treated as an annular cylindrical surface source.

The computer code SHIELD-SG (Reference 11) is used to calculate dose rates. For complex geometries other computer codes such as QAD (Reference 16) are used. Buildup, calculated using Berger coefficients presented in ORNL-RSIC-10 (Reference 7) and Blizard's Method of Buildup Determination, presented in the Engineering Compendium on Radiation Shielding (Reference 8), is used for laminated shields.

The source activity (Ci) and gamma ray source strengths (MeV/sec) are calculated using one of the following computer codes: ORIGEN (Reference 17), SOURCE2/ACCUM (Reference 12), or RADGAS3 (Reference 13). ACCUM (Reference 12) is an option within SOURCE2 that computes isotope accumulation for several time periods from a given flow of isotopes in curies per second. This accumulated activity may then be decayed for any number of decay times at which gamma energy spectra and isotope Curie activity are computed. The

Westinghouse

FAX COVER SHEET

RECIPIENT INFORMATION		SENDER INFORMATION	
DATE:	FEBRUARY 5, 1997	NAME:	JIM WINTERS
TO:	BILL HUFFMAN	LOCATION:	ENERGY CENTER - EAST
PHONE:	FACSIMILE:	PHONE:	Office: 412-374-5290
COMPANY:	US NRC	Facsimile:	win: 284-4887 outside: (412)374-4887
LOCATION:			

Cover + Pages 1 + 1

The following pages are being sent from the Westinghouse Energy Center, East Tower, Monroeville, PA. If any problems occur during this transmission, please call:

WIN: 284-5125 (Janice) or Outside: (412)374-5125.

COMMENTS:

BILL

HERE IS THE CHANGE WE INADVERTANTLY LEFT OUT OF SSAR Rev 10.

IT WILL BE IN REVISON 11 UNLESS WE HEAR FROM YOU.

Jim

cc LINDGREN
MCINTYRE
CUMMINS
RON VIJUK
WINTERS
KLUES
[illegible]
JENKINS EVANS
DIPLICA

17.4 Combined License Information Items

The Combined License applicant will address its design phase Quality Assurance program, as well as its Quality Assurance program for procurement, fabrication, installation, ~~and~~ construction and testing of structures, systems and components in the facility.

The Combined License applicant will also address its Quality Assurance program for operations.

17.5 References

1. "Energy Systems Business Unit – Quality Management System," Revision 1.

2. WCAP-8370 Revision 12a, "Energy Systems Business Unit - Power Generation Business Unit Quality Assurance Plan."

3. WCAP-8370/7800, Revision 11A/7A, "Energy Systems Business Unit - Nuclear Fuel Business Unit Quality Assurance Plan."

4. Letter NSD-NRC-96-4670, dated March 26, 1996.

Westinghouse

FAX COVER SHEET

RECIPIENT INFORMATION		SENDER INFORMATION	
DATE:	FEBRUARY 05, 1997	NAME:	JIM WINTERS
TO:	DIANE JACKSON	LOCATION:	ENERGY CENTER - EAST
PHONE:	FACSIMILE:	PHONE:	Office: 412-374-5290
COMPANY:	US NRC	Facsimile:	win: 284-4887 outside: (412)374-4887
LOCATION:			

Cover + Pages 1 + 1

The following pages are being sent from the Westinghouse Energy Center, East Tower, Monroeville, PA. If any problems occur during this transmission, please call:

WIN: 284-5125 (Janice) or Outside: (412)374-5125.

COMMENTS:

DIANE,

BASED UPON INTERNAL WESTINGHOUSE REVIEW, WE WOULD LIKE TO MAKE THE CLARIFICATION SHOWN IN THE CIRCLE TO THE MARKUP WE SENT YOU ON JANUARY 28, 1997 FOR OPEN ITEM 243. I WILL GO INTO SSAR REVISION 11 UNLESS WE HEAR FROM YOU.

cc: LINDGREN
MCINTYRE
CUMMINS
RON VIJUK
WINTERS
MCDERMOTT
HUTCHINGS
JEANNE EVANS

The high-pressure air subsystem includes pressure and carbon monoxide instrumentation, automatic protection devices, and temperature indication.

9.3.2 Plant Gas System

The plant gas system (PGS) provides hydrogen, carbon dioxide, and nitrogen gases to the plant systems as required.

Other gases, such as oxygen, methane, acetylene, and argon, are supplied in smaller individual containers and are not supplied by the plant gas system.

9.3.2.1 Design Basis

9.3.2.1.1 Safety Design Basis

The plant gas system serves no safety-related function and therefore has no nuclear safety design basis.

9.3.2.1.2 Power Generation Design Basis

The nitrogen portion of the plant gas system supplies nitrogen for pressurizing, blanketing, and purging of various plant components.

The hydrogen gas portion of the plant gas system supplies hydrogen to the main plant electrical generator for cooling as well as to other plant auxiliary systems.

The carbon dioxide portion of the plant gas system supplies carbon dioxide to the generator for purging of hydrogen and air during layup or plant outages.

9.3.2.2 System Description

Classification of equipment and components is given in Section 3.2.

9.3.2.2.1 General Description

The nitrogen portion of the plant gas system is a packaged system consisting of a liquid nitrogen storage tank and vaporizers. Nitrogen gas is supplied in both a high-pressure and a low-pressure subsystem. The high-pressure subsystem uses a pump to pressurize the gas supplying the accumulators in the passive core cooling system. The high-pressure supply is then reduced to supply makeup to the reactor coolant drain tank for purging and blanketing. Low-pressure nitrogen is provided for component purging, layup/blanketing, and pressurization.

stored within the valve~~s and their~~ operators

The main steam isolation valves (MSIVs) and main feedwater isolation valves (MFIVs) use compressed nitrogen as the motive force to close the valves. The main steam isolation valves are described in subsection 10.3.2.2.4 and the main feedwater isolation valves are described

FAX to DINO SCALETTI

February 5, 1997

CC: Sharon or Dino, please make copies for: Diane Jackson
Ted Quay

Don Lindgren
Robin Nydes
Dick Miller
Brian McIntyre

OPEN ITEM #144 (M3.11-11)

In my quest to make sure we have provided NRC with everything you need to prepare an FSER, I am researching open items from the oldest on. The relevant documentation related to Open Item #144 (M3.11-11) is contained in SSAR Appendix 3D. We provided a revision to a number of subsections in 3D on February 29, 1996 (over 10 months ago). We believed that these changes resolved the concerns of item #144. It seems a reasonable request that NRC acknowledge receipt of the change. Although we have other open items on environmental qualification, our records show no outstanding Westinghouse action on this item (#144) and we request that NRC provide a definitive action for Westinghouse or provide direction to change the status of this item. We recommend "Action N". Thank you.

Jim

Jim Winters
412-374-5290

AP600 Open Item Tracking System Database: Executive Summary

Date: 2/5/97

Selection: [item no] between 144 And 144 Sorted by Item #

Item No	Branch	DSER Section/ Question	Type	Title/Description Detail Status	Resp Engineer	(W) Status	NRC Status	Letter No. /	Date
144	NRR/SPLB	3.11	MTG-OI		Miller,D.	Closed	Action W		

M3.11-11 (EQUIPMENT QUALIFICATION)
Considering the response to Q270.14, provide an explanation of what is meant in the expression at the beginning of the sixth paragraph on page 3D-1 that states "Safety-related electrical and mechanical equipment is typically qualified using analysis, testing, or a combination of these methods".
If the response to Q270.14 accurately states the intent of the AP600 design, the SSAR should be rewritten to clarify the apparent inconsistencies between the statements on pages 3D-1, sixth paragraph; 3D-19, Section 3D.6.2, second paragraph; and 3D-69, environmental qualification data package. In addition, at the beginning of Section 3D.6 it is stated that "The recognized methods available for qualifying safety-related electrical equipment are established in IEEE 323. These are type testing, operating experience, analysis, on-going qualification, or a combination of these methods". This may be true for IEEE 323-1983, however, the requirements as outlined in 10 CFR 50.49(f) do not permit qualification by analysis alone. The SSAR should be updated to be consistent with the requirements of the Code of Federal Regulations.

Closed - Cloged SSAR 3D was changed in a number of places in Revision 5 to clarify the intent of allowable methods for qualifying safety - related electrical equipment.

 Total Records: 1

FAX to DINO SCALETTI

February 5, 1997

CC: Sharon or Dino, please make copies for: Ted Quay
Bill Huffman
Diane Jackson
Tom Kenyon
Joe Sebrosky
Cindy Haag
Don Lindgren
Robin Nydes
Brian McIntyre
Ed Cummins
Bob Vijuk

This is a reminder list of the Open Items where we have recently documented the difference between "W Status" and "NRC Status". In all cases, we believe the next action has been at NRC for over 3 months and await your definitization of a Westinghouse action or your direction to change the "NRC Status" to something other than "Action W".

Open Item Number	Westinghouse Submittal	Request for Status Change
123 (M3.6.1-2)	4/30/96	1/15/97
142 (M3.11-9)	2/29/96	2/3/97

Note that the status was changed for Items 4, 21, 30, 37, 134, 135, 137, 138, 139, 140, 141, 586, 969, 970 and 971, so they have been removed from the table.

Thanks for your help.

Jim

Jim Winters

FAX to DINO SCALETTI

February 5, 1997

CC: Sharon or Dino, please make copies for:

Ted Quay
Bill Huffman
Diane Jackson
Tom Kenyon
Joe Sebrosky
Robin Nydes
Cindy Haag
Don Lindgren
Bob Tupper
Bruce Rarig
Brian McIntyre
Ed Cummins
Bob Vijuk

NRC is requested to please acknowledge receipt of information related to each of the following Open Items. These are a subset of the items with "Action W" in "NRC Status" for which I have personally checked, since the first of the year, that we have submitted what we believe is the resolving information. Recognizing that reviewing for completeness of the response in each case constitutes an NRC action, we recommend that receipt acknowledgement be accompanied by direction to change their "NRC Status" to "Action N". If these are truly "Action W", please provide a description of the action Westinghouse is expected to take. We know of no action required. This is the fourth weekly request of this type.

123, 142, 157, 158, 164, 182, 184, 262, 300, 305, 308, 319, 333, 457, 458, 801, 802, 805, 807, 809, 972, 973, 1009, 1037, 1038, 1039, 1040, 1041, 1043, 1045, 1052, 1053, 1055, 1101, 1102, 1195, 1197, 1317, 1458, 1461, 1697, 1698, 1699, 1700, 1701, 1702, 1703, 1704, 1707, 1716, 1727, 1730, 1731, 1742, 1745, 1747, 1749, 1753, 1760, 1996, 1999, 2018, 2019, 2023, 2024, 2025, 2040, 2045, 2051, 2199, 2200, 2201, 2202, 2272, 2273, 2442, 2457, 2676, 2683, 2684, 2686, 2691, 2939, 2942, 2945, 2958, 2959, 2960, 2961, 2962, 2963, 2964, 2965, 2966, 2967, 2968, 2969, 2970, 2971, 2972, 2973, 2974, 2975, 2976, 2977, 2978, 2979, 2981, 2982, 2983, 2984, 2985, 2986, 3098, 3122, 3126. 3127, 3128, 3197, 3372, 3398, 3399, 3427, 3469, 3470, 3505, 3517, 3895, 3944, 3945, 3946, 3947, 3948, 3949, 3950, 3951, 3952, 3953, 3954, 3955, 3956, 3957, 3958, 4123, 4124, 4125, 4126, 4127, 4128, 4129, 4130, 4131, 4132, 4133, 4134, 4135, 4136, 4137, 4138, 4139, 4140, 4141, 4142, 4143, and 4144.

Jim

Thanks
Jim Winters
412-374-5290

FAX to DINO SCALETTI

February 5, 1997

CC: Sharon or Dino, please make copies for: Bill Huffman
Ted Quay
Bob Tupper
Ken Kloes
Brian McIntyre

OPEN ITEMS FOR CHAPTER 17

This is a background package for the remaining open items for Chapter 17. Chapter 17 is of interest because by our joint NRC/W schedule, the FSER for this chapter should have been turned into Projects by the end of January. Attached are copies of some of the relevant documentation related to the two remaining open items (1300 and 1301). Open Item #1300 had a number of question and answer cycles, one of which included Open Item #37. We believed these cycles culminated in the SSAR Revision 8 change to Chapter 17 which added reference to NSD-NRC-96-4670. We provided this revision to SSAR subsections 17.3 and 17.5 on June 19, 1996 (over 6 months ago). Open Item #1302 requested a change to the Combined License applicant information item for Chapter 17. This subsection (17.6) was revised prior to Revision 7 (April 30, 1996). These were very straightforward changes and implemented into the SSAR as agreed. It seems a reasonable request that NRC acknowledge receipt of the change. Our records show no outstanding Westinghouse action on this Chapter (17) and we request that NRC provide a definitive action for Westinghouse or provide direction to change the status of this item. We recommend "Action N". Thank you.

Jim

Jim Winters
412-374-5290

AP600 Open Item Tracking System Database: Executive Summary

Date: 2/5/97

Selection: [item no] between 1300 And 1300 Sorted by Item #

Item No.	Branch	DSER Section/ Question	Type	Title/Description Detail Status	Resp Engineer	(W) Status	NRC Status	Letter No. /	Date
1300	nrr/hqmb	17.1.3-1	DSER-OI		RTNSS/Kloes	Closed	Action W		

The QA applied to RTNSS should be comparable to that described in Generic Letter 85-06 for ATWS, as well as Regulatory Position 3.5 and Appendix A of RG 1.155 for non-safety-related station blackout equipment.

Closed - Quality Assurance requirements are graded based on the safety classification of the item or service, as described in WCAP-8370. Appropriate industry quality assurance requirements will be applied to non-safety-related items and services.

See item no. 37

2 of 2

AP600 Open Item Tracking System Database: Executive Summary

Date: 2/5/97

Selection: [item no] between 37 And 37 Sorted by Item #

Item No.	Branch	DSER Section/ Question	Type	Title/Description Detail Status	Resp Engineer	(W) Status	NRC Status	Letter No. /	Date
37	nrr/hqmb	17	RAI-OI	Question 260.34 (Quality Assurance) Westinghouse states that non-safety-related systems identified during implementation of the process used to determine the regulatory treatment of non-safety-related systems (RTNSS) are classified as AP600 Class D. The staff considers that the QA program applied to RTNSS-identified structures, systems, and componenets (SSCs) should follow guidelines comparable to those of Generic Letter 85-06 regarding anticipated transients without scram, and Regulatory Position 3.5 and Appendix A of Regulatory Guide 1.155, "Station Blackout," for blackout non-safety-related equipment. Describe how the AP600 quality assurance program that is applied to RTNSS-identified SSCs is comparable to these guidelines. Closed - Response provided by NSD-NRC-96-4696. Action N - Per telecon between Scaletti and Winters on 2/3/97.	Fanto	Closed	Action N	NSD-NRC-96-4696	

3 of 10

Westinghouse
Electric Corporation

Energy Systems

Box 355
Pittsburgh Pennsylvania 15230-0355

NSD-NRC-96-4696
~~DCP/NRC0500~~
Docket No.: STN-52-003

April 22, 1996

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

ATTENTION: T. R. QUAY

SUBJECT: WESTINGHOUSE RESPONSES TO NRC REQUESTS FOR ADDITIONAL INFORMATION ON THE AP600

Dear Mr. Quay:

Enclosed are three copies of the Westinghouse responses to NRC requests for additional information on the AP600 topics. Responses to RAIs 260.34, 952.101 and 952.106 are included in this transmittal.

The NRC technical staff should review these responses as a part of their review of the AP600 design. These responses close the three RAIs.

Please contact Brian A. McIntyre on (412) 374-4334 if you have any questions concerning this transmittal.

Brian A McIntyre

Brian A. McIntyre, Manager
Advanced Plant Safety and Licensing

/nja

Enclosures

cc: T. Kenyon, NRC (w/o enclosures)
D. Jackson, NRC (1E1)
E. Throm, NRC (w/o enclosures)
J. Kudrick, NRC (w/o enclosures)
P. Boehnert, ACRS (4E1)
N. J. Liparulo, Westinghouse (w/o enclosures)

2747A

4 of 10

9604300350

Question 260.34

Westinghouse states that non-safety-related systems identified during implementation of the process used to determine the regulatory treatment of non-safety-related systems (RTNSS) are classified as AP600 Class D. The staff considers that the QA program applied to RTNSS-identified structures, systems, and componenets (SSCs) should follow guidelines comparable to those of Generic Letter 85-06 regarding anticipated transients without scram, and Regulatory Position 3.5 and Appendix A of Regulatory Guide 1.155, "Station Blackout;" for blackout non-safety-related equipment. Describe how the AP600 quality assurance program that is applied to RTNSS-identified SSCs is comparable to these guidelines.

Response:

This response was provided as a response to DSER open item 17.1.3-1 forwarded by letter NSD-NRC-96-4670 dated March 26, 1996. The requirements to be included in our procedures for QA applied to ~~RTNSS are included~~ as an attachment to the letter for information. These requirements are comparable to those described in Generic Letter 85-06 and Regulatory Guide 1.155.

SSAR Revision: NONE

CHAPTER 17

QUALITY ASSURANCE

17.1 Quality Assurance During the Design and Construction Phases

See Section 17.4.

17.2 Quality Assurance During the Operations Phase

See Section 17.4.

17.3 Quality Assurance During Design, Procurement, Fabrication, Inspection and/or Testing of Nuclear Power Plant Items and Services

This section outlines the quality assurance program applicable to the design, procurement, fabrication, inspection, and/or testing of items and services for the AP600 Project.

Effective March 31, 1996, activities affecting the quality of items and services for the AP600 Project during design, procurement, fabrication, inspection, and/or testing are being performed in accordance with the quality plan described in *Westinghouse Electric Corporation – Energy Systems Business Unit, Quality Management Systems*, Revision 1 (Reference 1).

Activities performed prior to March 31, 1996 were performed in accordance with the quality plan described in topical report WCAP-8370, *Energy Systems Business Unit – Power Generation Business Unit, Quality Assurance Plan*, Revision 12a (Reference 2). Activities performed prior to November 30, 1992 were performed in accordance with the quality plan described in topical report WCAP-8370/7800, *Energy Systems Business Unit – Nuclear Fuel Business Unit, Quality Assurance Plan*, Revision 11A/7A (Reference 3).

A project-specific quality plan was issued to supplement the quality management system document and the topical reports for design activities affecting the quality of structures, systems, and components for the AP600 project. This plan addresses the NQA-1-1989 edition through NQA-1b-1991 addenda.

Quality Assurance requirements may be graded based on the safety classification of the item or service. For regulatory treatment of nonsafety systems, structures, and components (RTNSS SSCs), Westinghouse will impose quality requirements as identified in Reference 4.

While Westinghouse retains the overall responsibility for the AP600 design, portions of the design are developed by external organizations. Each organization maintains a quality assurance program that meets the NQA-1 criteria that apply to its work scope. In accordance with QMS Revision 1 (Reference 1), Westinghouse performs an initial evaluation of these programs and monitors their continued effective implementation through audits, surveillance, and evaluation of the performance of external organizations.

17.4 Combined License Information Items

The Combined License applicant will address its design phase Quality Assurance program, as well as its Quality Assurance program for procurement, fabrication, installation and construction of structures, systems and components in the facility.

The Combined License applicant will also address its Quality Assurance program for operations.

17.5 References

1. "Energy Systems Business Unit – Quality Management System," Revision 1.

2. WCAP-8370 Revision 12a, "Energy Systems Business Unit - Power Generation Business Unit Quality Assurance Plan."

3. WCAP-8370/7800, Revision 11A/7A, "Energy Systems Business Unit - Nuclear Fuel Business Unit Quality Assurance Plan."

4. Letter NSD-NRC-96-4670, dated March 26, 1996.

Westinghouse
Electric Corporation

Energy Systems

Box 355
Pittsburgh Pennsylvania 15230-0355

NSD-NRC-96-4670
~~DCP/NRC0481~~
Docket No.: STN-52-003

March 26, 1996

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

ATTENTION: T. R. QUAY

SUBJECT: QUALITY ASSURANCE REQUIREMENTS FOR AP600 RTNSS SYSTEMS, STRUCTURES, AND COMPONENTS

Dear Mr. Quay:

In the Draft Safety Evaluation Report (DSER) for the AP600, the staff requested that the QA applied to RTNSS should be comparable to that described in Generic Letter 85-06 for ATWS, as well as Regulatory Position 3.5 and Appendix A of RG 1.155 for non-safety-related station blackout equipment (DSER open item 17.1.3-1). Westinghouse is preparing a QA program requirements document, based on these standards, for use in RTNSS SSC procurements. The requirements that are to be included in this document are included for information as Attachment 1 to this letter.

Please contact John C. Butler on (412) 374-5268 if you have any questions concerning this transmittal.

Brian A. McIntyre

Brian A. McIntyre, Manager
Advanced Plant Safety and Licensing

/nja

Enclosure

cc: T. Kenyon, NRC
N. J. Liparulo, Westinghouse (w/o Enclosure)

~~9603290301~~

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2716A

AP600 Open Item Tracking System Database: Executive Summary

Date: 2/5/97

Selection: [item no] between 1301 And 1301 Sorted by Item #

Item No.	Branch	DSER Section/ Question	Type	Title/Description Detail Status	Resp Engineer	(W) Status	NRC Status	Letter No. /	Date
1301	nrr/hqmb	17.1.3-2	DSER-OI	Westinghouse should add COL Action Item 17.1.3-1 to the SSAR. (When completing the detailed design during the COL design phase, a COL applicant will be responsible to submit its design phase QA program for staff review. This will be in addition to the staff review of the COL applicant's QA program for construction of the facility.) Closed - Included in SSAR Revision 3.	Kloes	Closed	Action W	NTD-NRC-95-4464	

9 of 10

17.4 Combined License Information Items

The Combined License applicant will address its design phase Quality Assurance program, as well as its Quality Assurance program for procurement, fabrication, installation and construction of structures, systems and components in the facility.

The Combined License applicant will also address its Quality Assurance program for operations.

17.5 References

1. "Energy Systems Business Unit – Quality Management System," Revision 1.

2. WCAP-8370 Revision 12a, "Energy Systems Business Unit - Power Generation Business Unit Quality Assurance Plan."

3. WCAP-8370/7800, Revision 11A/7A, "Energy Systems Business Unit - Nuclear Fuel Business Unit Quality Assurance Plan."

4. Letter NSD-NRC-96-4670, dated March 26, 1996.

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** TX CONFIRMATION REPORT **   AS OF FEB 04 '97 16:58 PAGE.01

WETSO/RM 468 EC EAST

	DATE	TIME	TO/FROM	MODE	MIN/SEC	PGS	CMD#	STATUS
09	02/04	16:54	813014152002	EC--S	01'09"	003	122	OK
10	02/04	16:56	516 344 4900	G3--S	01'50"	003	122	OK

Westinghouse

FAX COVER SHEET

RECIPIENT INFORMATION	SENDER INFORMATION
DATE: February 4, 1997	NAME: Steve Kerch
TO: Jim Bongarra / Bill Huffman / Jim Higgins	LOCATION: Monroeville, PA.
PHONE: NRC / NRC / BNL	PHONE: 412-374-5104
COMPANY: 301-415-1046 / 301-415-1141 / BNL	
LOCATION:	FAX: (412) 374-5099

Cover + Pages = 1 + 2 3 total

- REMOVE ALL STAPLES
- PENCIL WILL NOT TRANSMIT - USE BLACK PEN
- PLEASE MAKE COPIES OF TWO-SIDED PAGES

Comments: Jim B., Bill H., and Jim Higgins,

Attached is a markup of the "Minimum Inventory" table (table 18.12.2-1) of the SSAR. This markup addresses some but not all of your comments in the NRC letter of January 17, 1997 on Minimum Inventory. The markup should be helpful during our conference call scheduled for tomorrow.

Thanks,

Steve Kerch

Phone Number of Receiving Equipment:

Jim B. → 301-415-2222

Bill H. → 301-415-2002

Jim H. → 516-344-4900

** TX CONFIRMATION REPORT **　　AS OF FEB 04 '97 16:52 PAGE.01

WETSO/RM 468 EC EAST

	DATE	TIME	TO/FROM	MODE	MIN/SEC	PGS	CMD#	STATUS
08	02/04	16:50	301 504 2222	G3--S	01'44"	003		OK

Table 18.12.2-1 (Sheet 1 of 2)

MINIMUM INVENTORY

Instrument/Display ~~Fixed Position Instrumentation~~	Control	Display	Alarm	SSAR Source
Neutron flux and neutron flux doubling indication		~~B1~~ X		Table 7.5-5 ; 7.5-1
RCS pressure		~~B1, B2, C1~~ X		Table 7.5-5, 6
WR T_{hot}		~~B1, B2~~ X		Table 7.5-5
WR T_{cold}		~~B1, B2~~ X		Table 7.5-5
Containment water level		~~B1, C1~~ X		Table 7.5-5, 6
Containment pressure		~~B1, C2~~ X		Table 7.5-5, 6
Pressurizer water level and trend		~~B1~~ X		Table 7.5-5
Pressurizer reference leg temperature		~~B1~~ X		Table 7.5-5
Pressurizer pressure		~~B1~~ X		Table 7.5-5
Core exit temperature		~~B1, C1~~ X		Table 7.5-5, 6
RCS subcooling		~~B1~~ X		Table 7.5-5
IRWST water level		~~B1~~ X		Table 7.5-5
PRHR flow		~~B1~~ X		Table 7.5-5
PRHR outlet temperature		~~B1~~ X		Table 7.5-5
PCS storage tank water level		~~B1~~ X		Table 7.5-5
PCS cooling flow		~~B1~~ X		Table 7.5-5
IRWST to RNS suction valve status		~~B1~~ X		Table 7.5-5
Containment isolation valves ~~position~~		~~B1~~ X		Table 7.5-5
Containment area high range radiation level		~~C1~~ X		Table 7.5-6
Containment pressure (extended range)		~~C1~~ X		Table 7.5-6
Containment hydrogen concentration		~~C1~~ X		Table 7.5-6
CMT level *		X		
Manual reactor trip	x			Table 7.2-4, PMS Also initiates turbine trip Figure 7.2-1 (Sheet 19), DAS MG set trip

Startup Rate

RCS cooldown rate compared to a limit, based on RCS pressure.

RCS hot leg level

RCS cold overpressure limit display

Position of Remotely Operated

Indication that RCS temperature has changed by more than 5° in the last 10 minutes

* Although this parameter does not meet any of the selection criteria of subsection 18.12.2, its importance to manual ADS actuation justifies its placement on the list.

Table 18.12.2-1 (Sheet 2 of 2)

MINIMUM INVENTORY

~~Fixed Position Instrumentation~~	Control	Display	Alarm	SSAR Source
Manual safeguards actuation	x			Table 7.2-4, PMS Also initiates reactor trip Table 7.3-3, PMS
Manual CMT actuation	x			Table 7.2-4, PMS Also initiates reactor trip Table 7.3-3, PMS Figure 7.2-1 (Sheet 19), DAS
Manual ADS actuation (1-3 and 4) 1 / 2 / 3 / 4	x			Table 7.2-4, PMS Also initiates reactor trip Table 7.3-3, PMS Figure 7.2-1 (Sheet 20), DAS
Manual PRHR actuation	x			Table 7.3-3, PMS Figure 7.2-1 (Sheet 19), DAS
Manual containment cooling actuation	x			Table 7.3-3, PMS Figure 7.2-1 (Sheet 20), DAS
Manual IRWST injection actuation	x			Table 7.3-3, PMS Figure 7.2-1 (Sheet 20), DAS
Manual containment recirculation actuation CV line / MOV line	x			Table 7.3-3, PMS Figure 7.2-1 (Sheet 20), DAS
Manual containment isolation	x			Table 7.3-3, PMS Figure 7.2-1 (Sheet 20), DAS
Manual main steamline isolation	x			Table 7.3-3, PMS
Manual feedwater isolation	x			Table 7.3-3, PMS
Manual containment hydrogen igniter	x			Figure 7.2-1 (Sheet 20), DAS
Main Control Room Emergency Habitability System Actuation	X			

FAX to DINO SCALETTI

February 4, 1997

CC: Sharon or Dino, please make copies for: Diane Jackson
Ted Quay

Don Lindgren
Robin Nydes
Dick Miller
Brian McIntyre

OPEN ITEM #143 (M3.11-10)

In my quest to make sure we have provided NRC with everything you need to prepare an FSER, I am researching open items from the oldest on. Attached are copies of some of the relevant documentation related to Open Item #143 (M3.11-10). We provided a revision to SSAR subsection 3D.5 on February 29, 1996 (over 10 months ago). This was a very straightforward change and implemented into the SSAR as agreed. It seems a reasonable request that NRC acknowledge receipt of the change. Although we have other open items on environmental qualification, our records show no outstanding Westinghouse action on this item (#143) and we request that NRC provide a definitive action for Westinghouse or provide direction to change the status of this item. We recommend "Closed". Thank you.

Jim

Jim Winters
412-374-5290

AP600 Open Item Tracking System Database: Executive Summary

Date: 2/4/97

Selection: [item no] between 143 And 143 Sorted by Item #

Item No.	Branch	DSER Section/ Question	Type	Title/Description Detail Status	Resp Engineer	(W) Status	NRC Status	Letter No. /	Date
143	NRR/SPLB	3.11	MTG-OI	M3.11-10 (EQUIPMENT QUALIFICATION) Question 270.11 is directed at the description of normal, abnormal, and design basis event conditions as outlined in the first paragraph of Section 3D.5. For example, "Abnormal refers to the operating range in which the equipment is designed to operate for a period of time without any special calibration or maintenance effort"; this description also applies to normal and design basis event conditions, therefore, no meaningful information is provided with this statement. The staff reviewed, and generally approves of the information provided in Sections 3D.5.1, 3D.5.2 etc. However, the staff is suggesting the 3D.5 can be rewritten with more clarity. Closed - The first paragraph of Section 3D.5 was changed by Revision 5 to better define normal, abnormal and design basis event conditions.	Miller,D.	Closed	Action W		

2 of 4

 Total Records: 1

Question 270.11

Section 3D.5 of the SSAR states that "normal conditions are those sets and ranges of plant conditions that are expected to occur regularly and for which plant equipment is expected to perform its safety-related function, as required, on a continuous, steady-state basis. Abnormal refers to the operating range in which the equipment is designed to operate for a period of time without any special calibration or maintenance effort. Design basis event conditions refers to environmental parameters to which the equipment may be subjected without impairment of its defined operating characteristics for those conditions." The descriptions of the three conditions are in terms of expected equipment performance rather than reactor operating conditions; consequently, the descriptions provide no information on the expected environmental conditions anticipated under each of the three conditions. The SSAR should discuss the anticipated environmental conditions associated with normal, abnormal, and design basis event conditions.

Response:

SSAR Subsections 3D.5.1 (Normal Operating Conditions), 3D.5.2 (Abnormal Operationg Conditions), 3D.5.3 (Seismic Events), 3D.5.4 (Containment Test Environment), and 3D.5.5 (Design Basis Event Conditions) provide information on the expected environmental conditions.

SSAR Revision: NONE

3D.4.10.1.3 Material Link

This documentation certifies that the materials used in the equipment are represented in a materials aging analysis, such as that described in Attachment B, (Subprogram B). This link applies only to equipment whose equipment qualification data package references the materials aging analysis and reflects a comparison of the as-built drawings, baseline design document, or other documentation of the plant specific equipment to the materials aging analysis listing.

3D.4.10.2 Similarity

Where differences exist between items of equipment, analysis may be employed to demonstrate that the test results obtained for one piece of equipment are applicable to a similar piece of equipment. Documentation of this analysis conforms with guidelines in IEEE 323 and 627, and Subsection 3D.6.2.1 and Section 3D.7 of this appendix.

3D.5 Design Specifications

The conditions and parameters considered in the environmental and seismic qualification of AP600 safety-related equipment are separated into three categories: normal, abnormal, and design basis event. Normal conditions are those sets and ranges of plant conditions that are expected to occur regularly and for which plant equipment is expected to perform its safety-related function, as required, on a continuous, steady-state basis. Abnormal conditions refer to the extreme ranges of normal plant conditions for which the equipment is designed to operate for a period of time without any special calibration or maintenance effort. Design basis event conditions refers to environmental parameters to which the equipment may be subjected without impairment of its defined operating characteristics for those conditions. Equipment required to operate while subjected to the design basis event and its extreme conditions and if not replaced, may require that tests, inspections, and maintenance be performed on the equipment, before returning to normal operating conditions.

The following subsections define the basis for the normal, abnormal, design basis event, and post-design basis event environmental conditions specified for the qualification of safety-related equipment in the AP600 equipment qualification program. (these are cited in Section 1.7 of each equipment qualification data package; See Attachment A.)

The service conditions simulated by the test plan are identified in equipment qualification data package Section 3.7. (See Subsection 3D.7.4.6 and Attachment A.) In general, the parameters employed are selected to be equal to (normal and abnormal) or have margin (design basis event and post-design basis event) with respect to the specified service conditions of equipment qualification data package, Section 1.7, as recommended by IEEE 323. These conditions are conservatively derived to allow for possible alternative locations of equipment within the plant.

FAX to DINO SCALETTI

February 3, 1997

CC: Sharon or Dino, please make copies for: Diane Jackson
Ted Quay

Don Lindgren
Robin Nydes
Dick Miller
Brian McIntyre

OPEN ITEM #142 (M3.11-9)

In my quest to make sure we have provided NRC with everything you need to prepare an FSER, I am researching open items from the oldest on. Attached are copies of some of the relevant documentation related to Open Item #142 (M3.11-9). We provided a revision to SSAR subsection 3D.4.10.2 prior to February 29, 1996 (over 10 months ago) and then modified SSAR subsection 3D.6.2.1 for clarification in the February 29, 1996 revision. It seems a reasonable request that NRC acknowledge receipt of the change. Note that in the Open Item the staff has not requested explicit changes, they simply pointed out a potential future concern for Combined License applicants. Although we have other open items on environmental qualification, our records show no outstanding Westinghouse action on this item (#142) and we request that NRC provide a definitive action for Westinghouse or provide direction to change the status of this item. We recommend "Closed". Thank you.

Jim

Jim Winters
412-374-5290

AP600 Open Item Tracking System Database: Executive Summary

Date: 2/3/97

Selection: [item no] between 142 And 142 Sorted by Item #

Item No	Branch	DSER Section/ Question	Type	Title/Description Detail Status	Resp Engineer	(W) Status	NRC Status	Letter No. /	Date
142	NRR/SPLB	3.11	MTG-OI	M3.11-9 (EQUIPMENT QUALIFICATION) There is no evidence anywhere in Industry or in NRC acceptance practice to support the position stated in the SSAR or in the response to the Q270.10 in regards to similarity between equipment from different manufacturers. Similarity between manufacturers is not arbitrarily excluded, however, the staff is simply pointing out that it has not been satisfactorily demonstrated previously in order to prevent the raising of false hopes and unnecessary expense for potential COL Applicants. Closed - Revised SSAR 3.D.4.10.2 to clarify.	Miller	Closed	Action W		

9 of 2

Question 270.10

Similarity is discussed in Section 3.D.10.2 of the SSAR. It is not clear that this discussion is consistent with staff practice on similarity. One of the most important aspects of the staff's position on this issue is that it is unlikely that similarity can be adequately demonstrated between equipment from different manufacturers. This section of the SSAR should address the staff's practice on this issue.

Response:

SSAR Subsection 3D.6.2.1 discusses similarities among manufacturers models implying that similarity is most easily addressed with equipment from the same manufacturer. Similarity between manufacturers should not be arbitrarily excluded, but should be judged on a case by case basis.

SSAR Revison: NONE

3D.4.10.1.3 Material Link

This documentation certifies that the materials used in the equipment are represented in a materials aging analysis, such as that described in Attachment B, (Subprogram B). This link applies only to equipment whose equipment qualification data package references the materials aging analysis and reflects a comparison of the as-built drawings, baseline design document, or other documentation of the plant specific equipment to the materials aging analysis listing.

3D.4.10.2 Similarity

Where differences exist between items of equipment, analysis may be employed to demonstrate that the test results obtained for one piece of equipment are applicable to a similar piece of equipment. Documentation of this analysis conforms with guidelines in IEEE 323 and 627, and Subsection 3D.6.2.1 and Section 3D.7 of this appendix.

3D.5 Design Specifications

The conditions and parameters considered in the environmental and seismic qualification of AP600 safety-related equipment are separated into three categories: normal, abnormal, and design basis event. Normal conditions are those sets and ranges of plant conditions that are expected to occur regularly and for which plant equipment is expected to perform its safety-related function, as required, on a continuous, steady-state basis. Abnormal conditions refer to the extreme ranges of normal plant conditions for which the equipment is designed to operate for a period of time without any special calibration or maintenance effort. Design basis event conditions refers to environmental parameters to which the equipment may be subjected without impairment of its defined operating characteristics for those conditions. Equipment required to operate while subjected to the design basis event and its extreme conditions and if not replaced, may require that tests, inspections, and maintenance be performed on the equipment before returning to normal operating conditions.

The following subsections define the basis for the normal, abnormal, design basis event, and post-design basis event environmental conditions specified for the qualification of safety-related equipment in the AP600 equipment qualification program. (these are cited in Section 1.7 of each equipment qualification data package; See Attachment A.)

The service conditions simulated by the test plan are identified in equipment qualification data package Section 3.7. (See Subsection 3D.7.4.6 and Attachment A.) In general, the parameters employed are selected to be equal to (normal and abnormal) or have margin (design basis event and post-design basis event) with respect to the specified service conditions of equipment qualification data package, Section 1.7, as recommended by IEEE 323. These conditions are conservatively derived to allow for possible alternative locations of equipment within the plant.

3D.6.2.1 Similarity

Similarities among manufacturer's models provides several options for extending qualification to equipment without the need for a complete qualification test program.

A model series, such as that for a solenoid valve design, consists of numerous models that are identical in materials of construction and manufacturing process, but have minor variance in size, functional mode, operating voltage, electrical termination type, and mechanical interface sizing. Such variances in most cases have no impact on or relevance to the capability of the various models to perform acceptably under environmental or seismic (or both) qualification test conditions. Furthermore, the design basis document may apply equally to each member of the model series. In such cases, all members of the model series can be qualified by reference to the same testing or analysis.

There may be sufficient similarities between different model series to justify the case for similarity. A documented comparison addressing differences in the design for each, or apparent physical differences between members of each model series, may be sufficient to preclude the testing of one model series based on the testing of the other.

Similarly, different models of a manufacturer's transmitters may be identical in some respects but different in others. The justification of similarity addresses the degree of similarity for critical characteristics. Differences that are not significant to qualification are also addressed for completeness. The mechanical and electrical functional modes and configurations must be the same. The materials of construction may be different, but must demonstrate equivalent performance. Other means of assuring accuracy may be necessary. When the devices are sufficiently similar in all attributes affecting qualification, qualification testing of one item can adequately cover another.

3D.6.2.2 Substitution

The objectives are to establish a degree of similarity and equivalence of performance for parts and materials that are different and, ultimately, to preclude the need for testing. For example, a gasket material is changed or a new type of capacitor is used because the original is no longer available, economical, or inadequate. Substitution of parts and materials is acceptable if comparison or analysis supports the conclusion that equipment performance is the same or better as a result. Consideration is given to characteristics of materials and the relative degree to which each is affected (or degraded) by the environmental parameters of qualification.

3D.6.2.3 Analysis of Safety-Related Mechanical Equipment

Environmental qualification of safety-related mechanical equipment is required to preclude common mode failures due to environmental effects of a design basis accident. Requirements are based on GDC 4 and 10 CFR 50, Appendix B. These criteria mandate that safety-related structures, systems, and components be designed to accommodate both normal and accident environmental effects.

3D.6.5.1.1 Aging

Past qualification tests may provide sufficient basis to preclude new aging simulation testing as part of the AP600 program. Also, simulation of both electrical and mechanical operational cycling may be waived where existing data demonstrates equipment durability greatly in the excess of the estimated number of operating cycles for Class 1E service. Application of past qualification and other tests is considered in the development of test plans and analysis procedures. The bases and justification is provided in qualification documentation for cases where applicable aging parameters are omitted from the test sequence.

3D.6.5.1.2 Seismic

Seismic qualification generally relies on analyses and justification to verify the adequacy or applicability of generic testing to a particular installed configuration of similar equipment. Analytical methods and documentation guidelines of IEEE 344-1987, as supplemented by Regulatory Guide 1.100, Revision 2, address these needs. Attachment E of this appendix provides the AP600 equipment qualification program requirements regarding seismic qualification.

3D.6.5.1.3 High-Energy Line Break Conditions

Typically, existing qualification tests address conditions of high-energy line break environments occurring inside containment. These are used where it is demonstrated that the qualification envelops the applicable requirements.

3D.7 Documentation

The AP600 equipment qualification program documentation consists principally of three types of documents:

- "Methodology for Qualifying AP600 Safety-Related Electrical and Mechanical Equipment" is the generic program "parent" document. It describes the methods and practices employed in the AP600 equipment qualification program.
- Equipment qualification data packages are "daughter" documents to the methodology. Each is a summary of the qualification program for a specific equipment type (for example, a particular model or design series of a manufacturer, an as-provided system, or a family of equipment tested as a set). The equipment qualification data package defines the qualification program objectives, methods, applicable equipment performance specifications, and the qualification plan. It provides a summary of the results.
- Equipment Qualification Test Reports (EQTRs) are the reports that present specific methods used during the qualification process and the results of that process.

The equipment qualification data packages are developed separate from the parent document. Similarly, the equipment qualification test reports are developed separate from the equipment

CONTINUES IN SUBSEQUENT PAGES