RECEIVED
MAR - 7 1997
Brian A. Mc Intyre

COVER SHEET FAX

To: Bill Huffman (NRC)

cc: B. McIntyre (Informal NRC File), Larry Hochreiter (Fax), Bob Osterrieder, Mike Young, Andy Gagnon, Dan Garner, Bob Kemper, File 7.6

Subject: NOTRUMP & WC/T

Date: February 7, 1997

Pages: ~~Two~~ Three, including this cover sheet.

COMMENTS:

Bill,

1. Attached is proposed agenda for our WC/T meeting on 3/13. Please give a copy to Lambrose and let me know if he has any comments.

2. Attached is the revised NOTRUMP agenda for the 3/12 meeting, which we discussed today.

3. Non-LOCA SAR revision is in final stages of publication and is scheduled to be shipped tomorrow.

4. The questions that we are working on from the original set are numbered:

   1c,d,e,i(1)
   2e,f,g; 3; 7e,f
   8a,b,f,g,i
   9a,c; 10a,b,c,d; 11; 12c,d,e,g,h,i,j,k,l,m
   13; 14; 15; 16; 17

Please pass on to Cliff for prioritization. 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) 374-4011

March 3 1997 I: NRC NT_CT 397

AGENDA

March 13, 1997
Thursday, 8:00 am
Westinghouse Rockville Office

NOTRUMP MEETING

1. Introduction

2. SPES Results

3. OSU Results

4. ACRS Meeting
   a. Executive Summary
   b. Proposed Agenda
   c. NRC Feedback on Approach

5. Documentation Closure
   a. Report
   b. RAIs/Open Items/DSER

6. Wrap-up

AGENDA

March 12, 1997
Wednesday, 12:00 pm
Westinghouse Rockville Office

LONG TERM COOLING MEETING

1. Introduction

2. PIRT

3. WC/T Plant Model

4. Summary of Westinghouse Topical Report

5. Recent Extended Time Calculation Results

6. Summary

7. ACRS Agenda

FAX to DINO SCALETTI

February 14, 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 #172 (M5.2.5-29)

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 #172 (M5.2.5-29) 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 (#172) 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 #172. 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: 2/14/97

Selection: [item no] between 172 And 172 Sorted by Item #

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

M5.2.5-29 (REACTOR COOLANT PRESSURE BOUNDARY LEAKAGE) STS 3.4.15 states that, should the containment air cooler condensate flow rate monitor become inoperable, a channel check should be performed on the containment atmosphere radioactivity monitor once per 8 hours. The AP600 TS 3.4.9 states that a grab sample should be performed once per 24 hours. Westinghouse should provide justification regarding the acceptability of the alternate action.

Action: submit T.S. 3.4.9 with June 96 rev. rkn 3/28

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&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.

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

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)

3047A

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.

6.17

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

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

Westinghouse

FAX COVER SHEET

RECIPIENT INFORMATION		SENDER INFORMATION	
DATE:	FEBRUARY 17, 1997	NAME:	JIM WINTERS
TO:	BILL HUFFMAN	LOCATION:	ENERGY CENTER - EAST
PHONE:	FACSIMILE:	PHONE:	Office: 412 374-5290
COMPANY:	USNRC	Facsimile:	win: 284-4887 outside: (412)374-4887
LOCATION:			

Cover + Pages 1 + 2

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 OUR RECOMMENDED WORDING FOR CABLE MIXING IN TRAYS. IT IS A COUNTER RECOMMENDATION TO THE ONE YOU SENT US LAST WEEK. IT WILL GO INTO SSAR REVISION 11 ONLY IF WE GET YOUR AGREEMENT BY CLOSE OF BUSINESS ON WEDNESDAY 2/19. OTHERWISE, WHATEVER WORDS WE AGREE ON WILL GO INTO REVISION 12.

JIM

CC: LINDGREN
MCINTYRE
CUMMINS
RONVISUK
WINTERS
HAYES
JEANNE EVANS

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

INSERT 8.3-Y

A tray designed for a single class of cables shall contain only cables of the same class except that low voltage power cables may be mixed with high level signal and control cables if their respective sizes do not differ greatly and if they have compatible operating temperatures. When this is done in trays, the power cable ampacity should be calculated as if all cables in the tray were power cable, unless position and grouping are controlled.

FAX to DINO SCALETTI

February 18, 1997

CC: Sharon or Dino, please make copies for: Bill Huffman
Ted Quay
Don Lindgren
Chip Suggs
Ed Cummins
Bob Vijuk
Brian McIntyre

OPEN ITEM #177 (M5.2.5-34)

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 #177 (M5.2.5-34) is attached. We provided the original responses to RAIs 410.16 through 410.20 with ET-NRC-93-3840 on March 18, 1993. We then provided a revision to the SSAR describing our conformance with Position C.9 of the Reg Guide on December 20, 1996. this information is consistent with the technical specifications. We believe that this information resolves the concerns of item #177. 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

1 of 6

AP600 Open Item Tracking System Database: Executive Summary

Date: 2/18/97

Selection: [item no] between 177 And 177 Sorted by Item #

Item No.	Branch	DSER Section/ Question	Type	Title/Description Detail Status	Resp Engineer	(W) Status	NRC Status	Letter No. /	Date
177	NRR/SPLB	5.2.5	MTG-OI	M5.2.5-34 (REACTOR COOLANT PRESSURE BOUNDARY LEAKAGE) Add the responses to the following RAIs to the SSAR. 410.16, 410.17, 410.18, 410.19, 410.20 Closed - SSAR Rev. 3 included the information from the RAI responses Action W - Describe the conformance with Position C.9 of Reg Guide. See response to RAI 410.17 for conformance information.	Lindgren,D.	Closed	Action W		

2 of 6

Total Records: 1

Westinghouse Electric Corporation **Energy Systems**

Box 355
Pittsburgh Pennsylvania 15230-0355

ET-NRC-93-3840
NSRA-APSL-93-0078
Docket No.: STN-52-003

March 18, 1993

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

ATTENTION: R. W. BORCHARDT

SUBJECT: WESTINGHOUSE RESPONSES TO NRC REQUESTS FOR ADDITIONAL INFORMATION ON THE AP600

Dear Mr. Borchardt:

Enclosed are three copies of the Westinghouse responses to NRC requests for additional information on the AP600 from your letters of November 16, 1992 and January 26, 1993. This transmittal completes the responses to the November 16, 1992 letter. A listing of the NRC requests for additional information responded to in this letter is contained in Attachment A. Attachment B is a complete listing of the questions associated with the November 16, 1992 letter and the corresponding Westinghouse letters that provided our response.

If you have any questions on this material, please contact Mr. Brian A. McIntyre at 412-374-4334.

Brian A. McIntyre /for

Nicholas J. Liparulo, Manager
Nuclear Safety & Regulatory Activities

/nja

Enclosure

cc: B. A. McIntyre - Westinghouse
F. Hasselberg - NRR

61

3 of 6

6854A

ET-NRC-93-3840
ATTACHMENT A
AP600 RAI RESPONSES
SUBMITTED MARCH 18, 1993

RAI No.	Issue
410.016	Reactor Coolant Leakage
410.017	Reg. Guide 1.45, Position C.9
410.018	Reg. Guide 1.45, Position C.8
410.019	Reg. Guide 1.45, Position C.7
410.020	Reg. Guide 1.45, Position C.6
410.023	First stage ADS hydrostatic loads
410.025	Reg. Guide 1.52
410.027	Equipment requiring protection from flooding
410.028	Potential sources of flooding
410.030	Maximun flood level
410.033	Flood protection
410.034	Flood protection
410.037	PXS equipment location
410.040	Multi-door passageways leakage prevention
410.043	CCW layout
410.044	Flood hazards
410.046	Break protection from open cycle systems
410.047	Water tight doors
410.048	SFP cooling pumps & heat exchangers flood prot.
410.049	Flood consequences
410.050	Flooding protection for remote shutdown panel
410.051	Equipment requiring missile protection
410.052	Turbine missiles
410.053	Secondary missiles
410.054	Equipment protection
410.059	Stored energy - nuts, bolts and studs

4f6

Question 410.17

Position C.9 of RG 1.45 states that the technical specifications should address the availability of various types of instruments for RCPB leakage to ensure adequate coverage at all times. Describe how the AP600 design will meet this regulatory position (Section 5.2.5).

Response:

SSAR Chapter 16, Technical Specification 3.4.9, defines the operability requirements for RCS leakage detection instrumentation. In addition, instrumentation used to identify reactor coolant pressure boundary leakage is designed so that its operability may be determined at all times. Should a detector fail (signal outside its calibrated range or self-monitored trouble detected), the plant instrumentation system will alarm in the main control room that the specific leak detection monitor readout is questionable. The alarm prompts the operators to observe other sensors providing leak detection information. Technical Specification 3.4.9 allows leakage to be averaged over 24 hours; therefore, operators have sufficient time to determine if small leaks are from the reactor coolant system and to take corrective action in an orderly manner.

SSAR Revision: NONE

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 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.

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.

FAX to DINO SCALETTI

February 18, 1997

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

Don Lindgren
Richard Orr
Ed Cummins
Bob Vijuk
Brian McIntyre

OPEN ITEMS FOR SSAR SECTION 3.8.3

This is a background package for the remaining open items for SSAR section 3.8.3. SSAR section 3.8.3 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 18 Open Items with NRC Status of Action W. Two (2) of these items (711 and 725) still require some Westinghouse action. Westinghouse believes the other sixteen (16) items were addressed in or prior to the January 16, 1997 meeting with NRC. Currently, our records show no additional outstanding Westinghouse action required for section 3.8.3, except items 711 and 725, 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: 2/18/97

Selection: [nrc st code]='Action W' And [DSER Section] like '3.8.3*' Sorted by Item #

Item No.	Branch	DSER Section/ Question	Type	Title/Description Detail Status	Resp Engineer	(W) Status	NRC Status	Letter No. /	Date
710	NRR/ECGB	3.8.3.1-1	DSER-OI	Westinghouse should provide in the SSAR the connection details between "M" modules, and between "M" modules and other types of modules. Module behavior study is in progress. Design calculations for modules will be updated following completion of the behavior study to include any changes in methodology defined by the study. Additional connection details will be developed during this update and will be included in design data to be audited during a meeting scheduled for September/October of 1996. Typical connection details will be described in SSAR. Closed in meeting with NRC 1/16/97 - minor SSAR change shown in draft revision	Orr / Bechtel / NRCSM	Closed	Action W		
711	NRR/ECGB	3.8.3.1-2	DSER-OI	Westinghouse should demonstrate that the structure will not lift up during an SSE. Liftoff of the CIS basemat from the containment vessel and NI basemat was included in the nuclear island basemat analyses. Additional analyses of the CIS and NI basemat response to seismic loads is in progress. These analyses will demonstrate that liftoff of one side of the CIS basemat is not significant. Result will be available at structural audit.	Orr / INI / NRCBM	Action W	Action W		
716	NRR/ECGB	3.8.3.2-5	DSER-OI	Westinghouse should justify the use of the ANSI/AISC N690 Standard and the ACI 349 Code for concrete-filled steel M modules. Closed - This issue is addressed in the module behavior study and included in SSAR Rev. 7. Based on review by the NRC in a meeting on May 22, this issue is closed. Meeting notes dated July 1, 1996 show this item as still open. Westinghouse to finalize all design criteria for structural modules. Closed in meeting with NRC 1/16/97	Orr / NRCSM	Closed	Action W		
717	NRR/ECGB	3.8.3.3-1	DSER-OI	Westinghouse should address in the SSAR the entire construction process, from off-site fabrication to final on-site placement. Closed - NRC will review revision 7 of SSAR, subsections 3.8.3 and 3.8.4. See NRC letter dated 7/15/96 - Address use of sections 1.23, 1.25, and 1.28 of AISC N690. Action W - See NRC letter of 12/9/96. Closed in meeting with NRC 1/16/97 - minor SSAR change shown in draft revision	Orr / Ritz / NRCSM	Closed	Action W		
718	NRR/ECGB	3.8.3.3-2	DSER-OI	Westinghouse should address the construction-induced stress following the curing of the concrete. Closed - SSAR subsection 3.8 was revised to address stress in module due to concrete placement. NRC meeting notes 7/1/96 show this as Action W - expand SSAR description of the methods for considering the hydrostatic pressure due to construction in the design. Closed in meeting with NRC 1/16/97 - minor SSAR change shown in draft revision	Orr / NRCSM	Closed	Action W		
719	NRR/ECGB	3.8.3.3-3	DSER-OI	Westinghouse should consider, in the design of the IRWST, the combination of the load from ADS actuation and the SSE load. In addition, the thermal loading should be considered in the internal structural steel frame design. SSAR Revision 7 subsection 3.8.3.3.1 combines ADS and SSE loads. Thermal loading on steel structures is considered as shown in Table 3.8.4-1. Calculations will be reviewed during the structural module audit.	Orr / NRCSM	Resolved	Action W		
722	NRR/ECGB	3.8.3.4-3	DSER-OI	Westinghouse should demonstrate the adequacy of the design based on the assumption of a composite section. Resolved based on information in the module behavior study.	Orr / NRCSM	Resolved	Action W		

Hp2

Selection: [nrc st code]='Action W' And [DSER Section] like '3.8.3*' Sorted by Item #

Item No.	Branch	DSER Section/ Question	Type	Title/Description Detail Status	Resp Engineer	(W) Status	NRC Status	Letter No. /	Date
724	NRR/ECGB	3.8.3.4-5	DSER-OI	Westinghouse should use a local 3D solid model of the module geometry and materials as the basis for developing equivalent isotropic shell properties, or for justifying the equations currently used. Closed - This issue was addressed in the module behavior study NRC meeting notes 7/1/96 show this as Action W - to provide the analysis and design results to demonstrate and confirm the adequacy of the method used for design. Design calculations are available for audit. Closed in meeting with NRC 1/16/97	Orr	Closed	Action W		
725	NRR/ECGB	3.8.3.4-6	DSER-OI	Westinghouse should acceptably address issues relating to the seismic modeling of the containment internal structures. Closed - This issue was addressed in the module behavior study NRC meeting notes 7/1/96 show this as Action W - design calculations to be audited by NRC	Orr	Action W	Action W		
729	NRR/ECGB	3.8.3.4-10	DSER-OI	Westinghouse should revise the combined stress equations in Section 3A.3.1.3 of the SSAR to reflect realistic action of the walls if biaxial bending is required. Closed - This issue was addressed in the module behavior study NRC meeting notes 7/1/96 show this as Action W - to reexamine interaction equations described in SSAR. Closed in meeting with NRC 1/16/97	Orr	Closed	Action W		
730	NRR/ECGB	3.8.3.4-11	DSER-OI	Westinghouse should complete the design of the connection details and provide the design for staff review. Resolved - Selected connection details will be available for review during the structural module audit.	Orr / NRCSM	Resolved	Action W		
731	NRR/ECGB	3.8.3.4-12	DSER-OI	Westinghouse should compile design summary reports using the format and attributes described in Appendix C to Section 3.8.4 of the SRP, and should submit the reports for staff review. Resolved - The design report will be available for review during the structural module audit.	Orr / NRCSM	Resolved	Action W		
732	NRR/ECGB	3.8.3.4-13	DSER-OI	The staff will perform a structural design audit of the containment internal structures. Resolved - The structural module audit is planned for late 1996	Orr / NRCSM	Resolved	Action W		
2347	NRR/ECGB	3.8.3	MTG-OI	Westinghouse should describe the design process used for the structural module design in the SSAR. This is part of the module behavior study in progress as well as the update to the hydrodynamic analyses. See open item 3.8.3.4-10 (item # 729) and item # 2348. Closed in meeting with NRC 1/16/97	Orr / NRCSM	Closed	Action W		
2348	NRR/ECGB	3.8.3	MTG-OI	Westinghouse should revise Appendix 3F to address questions related to analysis methods and ADS loads for the structural module design. Appendix 3F has been replaced by material in subsection 3.8.3. Detailed questions have been addressed in design calculations which are available for audit in late 1996. Review of documentation not completed during audit.	Orr / NRCSM	Audit N	Action W		

Selection: [nrc st code]='Action W' And [DSER Section] like '3.8.3*' Sorted by Item #

Item No.	Branch	DSER Section/ Question	Type	Title/Description Detail Status	Resp Engineer	(W) Status	NRC Status	Letter No. /	Date
2349	NRR/ECGB	3.8.3	MTG-OI		Orr / NRCSM	Closed	Action W		
				Westinghouse should complete analysis of a 30 inch wall in the M-1 structural module and make the analysis available for audit.					
				Analyses of 30" wall are being finalized and will be available for audit in late 1996. Closed in meeting with NRC 1/16/97 - minor SSAR change shown in draft revision					
3057	NRR/ECGB	3.8.3	MTG-OI		Orr	Closed	Action W	NSD-NRC-96-4732	5/31/96
				Describe how concrete cracking is considered in the thermal analysis and provide justification for the adaquacy of the methods used.					
				Closed - Response provided in item 1 of letter NSD-NRC-96-4732, dated May 30, 1996					
3247	NRR/ECGB	3.8.3.4	RAI-OI		Orr	Closed	Action W		
				RAI 230.98 April 5, 1996 letter: Westinghouse should complete the new design of structural modules (using shear studs) and submit the design for staff review.					
				Closed - The structural module design with shear studs and other changes is described in SSAR subsection 3.8.3.1 Rev. 7					

4 of 4

Westinghouse

FAX COVER SHEET

RECIPIENT INFORMATION		SENDER INFORMATION	
DATE:	2-18-97	NAME:	Cindy Haag
TO:	Joe Sebrosky	LOCATION:	ENERGY CENTER - EAST
PHONE:	FACSIMILE:	PHONE:	Office: 412-374-4277
COMPANY:	US NRC	Facsimile:	win: 284-4887 outside: (412)374-4887
LOCATION:			

Cover + Pages 1 + 2

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:

Joe -

Here are some minor editorial changes to the markup of SSAR section 6.2.4.2.3 that was faxed to you on 2/13/97. These changes will be in SSAR Revision 11 unless we hear from you.

Cindy

cc Winters
Cummins
Jeanne Evans

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 (± 2.5 feet) with the final locations governed by the installation details. The igniter locations identified are considered approximations (± 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

Table 6.2.4-6

IGNITER LOCATION CRITERIA

- A sufficient number of igniters ~~should be~~ are 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~~ are located in all 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~~ are 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~~ are placed near the ceiling of such compartments. Igniter coverage ~~should be~~ is 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~~ are considered. Percent

- To the extent possible, igniters ~~should be~~ are 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~~ are 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~~ is provided to contol combustion in all areas where oxygen rich air may enter into an inerted region with combustible hydrogen levels during any accident scenario.

- Igniters ~~should be~~ are 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~~ is provided as close to the source as feasible.

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

Westinghouse

FAX COVER SHEET

RECIPIENT INFORMATION		SENDER INFORMATION	
DATE:	2-18-97	NAME:	Cindy Haag
TO:	Joe Sebrosky / Bill Huffman	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 + ~~17~~ 16

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:

Joe/Bill -

Attached is the Sandia squib valve failure rate information Westinghouse agreed to supply to the Staff, based on the 2/12/97 telecon with Nick Saltos and John Flack. Westinghouse considers the telecon action item as closed.

Cindy

cc: Tim Bieter
Cindy Haag

A telecon was held on Wednesday, February 12, 1997 between Westinghouse and NRC Probabilistic Safety Assessment Branch to discuss NRC questions on an AP600 PRA sensitivity study. One of the questions the NRC asked during the telecon related to the failure rate Westinghouse used for squib valve failure to operate. Westinghouse accepted an telecon action item to provide to NRC the Sandia data which was used to develop the AP600 PRA squib valve failure rate.

Attached is a copy of the Sandia data for squib valve failure to operate. This information is being provided to NRC in response to the 2/12/97 telecon action item.

UNCLASSIFIED

Sandia National Laboratories

Livermore California 94551-0969

date: February 19, 1996

to: Tim Bueter, Westinghouse

Charles J. DeCarli

from: C. J. DeCarli - 8116

subject: Explosive Valve Reliability Information

Jere Harlan asked me to send you information on the reliability of our explosively actuated valves.

Explosively actuated valves that cut tubes or punch membranes have assessed failure rates that range from .0002 to .0006. No failures have been observed in these valves. There are differences in assessed failure rates because of different quantities of test data, not because failures have occurred.

I am enclosing the data assessment sheets for our standardized mini-valves. These are cut valves that have standardized internal features. The assessed failure rate is .0002 because there have been no failures in over 3600 post-development destructive tests combined over the several valves in the family.

Feel free to call me at 510-294-2561 if you have questions.

cjd:8116

Copy to:

MS 1452	J. G. Harlan, 1552
MS 9202	R. L. Bierbaum, 8116

STANDARDIZED MINI-VALVES

WEAPON SYSTEM: Several Weapon Systems

Components: 13 valves: MC3006, MC3205, MC3206, MC3294/MC3784, MC3295, MC3297/MC3785 Side B, MC3298, MC3425, MC3427/MC3427A, MC3428/MC3428A, MC3570/MC4232, MC3604, MC4241

MAJOR ASSEMBLY: Gas Transfer System

FAILURE EVENT
General: Failure of the standardized mini-valve to properly cut one or two tubes and transfer gas, given the proper input to the MC3004/MC2949A/MC3479,MC3753 actuators.

ASSESSED FAILURE RATE: 0.0002

ASSESSMENT DATE: January 1996

RELIABILITY ENGINEER: ______________________ Date ________
C. J. DeCarli, 8116

REVIEWER: ______________________ Date ________
R. S. Tilley, 8116

Reliability Assessment Data:
Cumulative data for thirteen mini-valves is summarized below. The sampling rate for production acceptance D-testing of all mini-valves was 5%. This was adopted in October 1981 because of the success history of the MC3006, MC3205 and MC3206 valves and because so many mini-valves were to be produced. Prior to this time, mini-valves were tested at a much higher rate to accumulate a data base.

UNCLASSIFIED

Distribution 3

CUMULATIVE VALVE TEST DATA FOR THIRTEEN MINI-VALVES

Data Source	No. Tested	No. Failed	Comments
A. Development	1048	0	See Table 1
B. Production D-tests	2703	0	See Table 1
C. Surveillance			
C1 NMLT/SLT	1362	0	See Table 1
C2 NMFT/SFT	607	0	
Total	5720	0	See Table 1
Total w/o development	4672		

The 0.0002 assessment is a 50% upper binomial confidence limit based on zero failures in 4672 post-development tests.

UNCLASSIFIED

Distribution 4

TABLE 1 - SUMMARY OF MINI-VALVE D-TEST DATA (JANUARY 1996)

MC No	Develop.	Product Acceptance No.	Product Acceptance Lot No.	Stockpile Evaluation Lab.	Stockpile Evaluation Flight
3006	143	547	107	161	100
3205	354	318	45	355	86
3206	194	210	33	223	86
3294/ 3784	97	161 78	71 37	31	59
3295	41	116	44	31	72
3297/ 3785	66	227 415	14 35	31	72
3298	65	174	36	31	36
4 Valves*	29	279		421	48
3570 4232	59 46	76 13	34	78	48
4241		56	30		
TOTALS	**1048**	**2703**		**1362**	**607**

GRAND TOTAL WITHOUT DEVELOPMENT TESTS = 4672

* Includes 4 valves: MC3425, MC3427/A, MC3428/A, MC3604

PACIFIC SCIENTIFIC

Fax Transmittal

ENERGY DYNAMICS DIVISION

7403 West Boston Street
Chandler, Arizona 85226
(520) 796-1100
Fax (520) 796-0754

Fax to #: 505-844-5924

Date: 2-15-96

Company: Sandia

Attn: Jere Harlan

From: Mike Gerace

Subject: Tech Memorandum

Number of Pages: 10

Attached is the Reliability of Pyrotechnically Actuated Valves per your request.

Should you have any questions, please contact me.

PACIFIC®
SCIENTIFIC

Energy Dynamics Division

TECHNICAL MEMORANDUM

THE RELIABILITY OF PYROTECHNICALLY ACTUATED VALVES

14 FEBRUARY 1996

Last page shows
P: 7.?E-4

Prepared by: John T Greenslade
John Greenslade, Senior Staff Engineer

THE RELIABILITY OF PYROTECHNICALLY ACTUATED VALVES

1.0 INTRODUCTION

Pacific Scientific/Energy Dynamics Division (PS/EDD) has prepared this technical memorandum in response to an informal request for information relative to the general reliability of pyrotechnically actuated valves, one of this company's product lines. Such valves, in both Normally-Open (N-O) and Normally-Closed (N-C) configurations, have been widely used for many years in military and aerospace applications where rapid and positive valving of a liquid or gaseous working medium is required in a one-time control event, such as fuel shut-off, fire suppressant deployment or gas sampling. The characteristics which have made pyro actuated valves so suitable for such applications, in fact the only reasonable choice in many cases, are their small size and weight (compared with all other competitive approaches), their ability to valve very high pressure working media (in some cases as high as 10,000 psig), their extremely rapid actuation time (typically <5 msecs) and their relative lack of complexity. The latter attribute no doubt contributes to their high operational reliability which typically is well in excess of 0.999, even after long periods of dormancy. PS/EDD produced a number of high reliability pyro valves for the DOE, for use in nuclear weapons.

All pyro actuated devices (PADs), including valves, by their very nature are "one-shot" devices. Consequently, the reliability of a PAD cannot be established empirically by conducting a large number of operational tests repeatedly on the same individual unit, as is done with electrical and electronic components for instance. Instead, the predicted reliability of PADs must be derived using data from tests of similar assemblies, the results of stress analyses and recorded failure rate data relative to similar individual components of the device.

In order to illustrate the methods used to arrive at the predicted reliability levels of a pyro actuated valve, a reliability analysis is presented in the following section relative to a typical PS/EDD 2-way pyro actuated valve, somewhat more complex than most, as it might be used in a ground-based application such as a nuclear reactor cooling system. An estimate is derived of its operational reliability, as well as its mission reliability. The latter takes into account an assumed period of dormancy.

2.0 THE RELIABILITY ANALYSIS OF A TYPICAL PS/EDD PYRO ACTUATED VALVE

2.1 The Assumed Scenario

The valve selected as an example for analysis is PS/EDD's 51-5875-2 Fuel Valve, shown in Figure 1, which was qualified for use in the TSSAM program. In this 2-way valve, which performs both N-O and N-C functions, the inlet and by-pass ports are in contact with the fuel and are interconnected, prior to actuation. During actuation of the valve, a nipple covering the outlet port is sheared away and the inlet and outlet ports are then interconnected by way of a transverse bore through the piston. The piston blocks off the by-pass port as it completes its stroke.

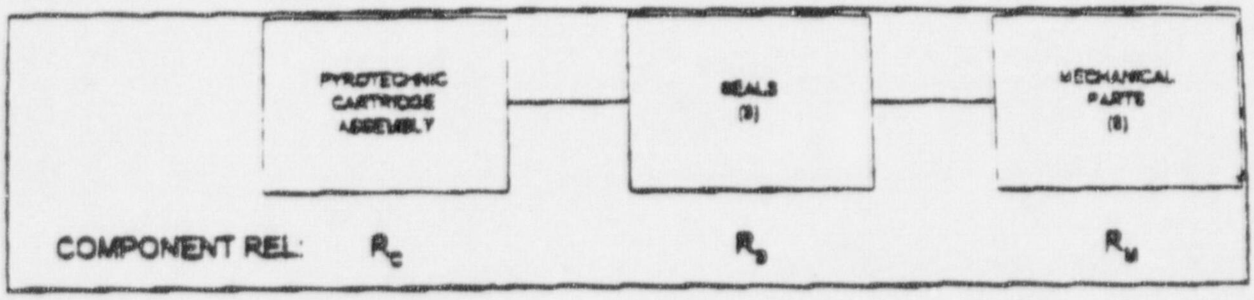
In the following sub-sections, a reliability analysis of this device is presented which is based on the assumptions below, which are believed to be consistent with a ground-based application at a nuclear energy facility.

a)	Dormant (Non-Operational) Period	10 years
b)	Valve Actuation Time	5 msecs
c)	Valve Function* Time	20 minutes

(*During this period the actuated valve would handle pressurized flow-down without loss of structural or sealing integrity. A function time of 20 minutes may be longer than required but will provide a conservative estimate of reliability.)

2.2 The Reliability Model

The valve can be considered as an assembly of "mechanical" components, both functional and structural, a set of interface seals and a mating self-contained assembly which provides the required actuation gas pressure, namely, the pyro cartridge. Since all of the components of the overall assembly must function correctly and/or maintain their structural integrity during the overall life of the device, the components are considered series dependent. This permits the following very simple Reliability Model.



Valve Assembly (Incl. Cartridge) Operational Rel: $R_V = (R_c)(R_s)(R_M)$ ---(1)

2.3 Reliability Analysis

2.3.1 Operational Reliability (R_V)

2.3.1.1 The Pyro Cartridge Reliability (R_C)

An approach often used for predicting the operational reliability of pyro cartridges is based on a relationship between the number of tests conducted without failure (N) the Confidence Level (CL) and the Reliability (R). This relationship, which is derived from the binomial theorem (Ref. 1) is expressed by the equation

$$N = \frac{Log\,(1-CL)}{Log\,(R)} \quad ----(2)$$

which gives:

$$R = e^{(\ln(1-CL))/N} - - - - (3)$$

Through the years, <u>at least</u> 25,000 <u>similar</u> pyro cartridges have been successfully fired by PS/EDD alone, therefore 25,000 would not be an unreasonable value for N in Equation 3. Then, at a CL of 90%, which is also a reasonable level, the operational reliability of the pressure cartridge (R_C), as given by Equation 3, would be

$$R_C = e^{(\ln(1-.9))/25{,}000} = .999908$$

2.3.1.2 <u>The Seal Reliability (R_s)</u>

The 52-5875-2 valve incorporates nine (9) O-ring seals. Failure rate data for O-rings is given in NPRD-91 (Ref. 2). That document gives a generalized failure rate (λ) of <u>6.5890 failures/10^6 hours</u> for "MIL" type O-rings subjected to a "Ground Mobile" (G.M.) environment. In the currently assumed scenario the valve would be subjected to a more benign "Ground Fixed" (GF) environment. In the absence of specific λ data for the GF environment it is assumed to be half that of the GM environment i.e., $\lambda_{GF} = 0.5\ \lambda_{GM}$, or, <u>$\lambda_{GF} = 0.5\ (6.5890) = 3.2945$ failures/10^6</u>. As stated in 2.1, the assumed Function Time (t) could be 20 minutes, i.e., .3333 hours. As stated in MIL-STD-756 (Ref. 3), if the failure rate and operational time are known the reliability is given by the equation

$$R = e^{-\lambda t} - - - - (4)$$

This equation, which is for a single component can be modified for (n) like components as follows

$$R = e^{-n\lambda t} - - - - (5)$$

Substituting for n, t and λ is (5) gives the seal operational reliability

$$R_s = e^{-[illegible]/10^6} = .999990$$

2 3 1 3 <u>The Mechanical Parts Reliability (R_M)</u>

In the 51-5875-2 Valve there are nine (9) "mechanical" parts, all of which must retain their structural integrity during the operation of the valve and three of them must perform certain functions. Thus:

<u>Components (9)</u>	<u>Functions (5)</u>
Body	
Adaptor	
Plug	
Lee Plug	
Inlet Fitting	
By-Pass Fitting	
Outlet Fitting →	Shear the Closure Nipple
Piston →	Shear the Closure Nipple; Release the Initial Lock; Provide Metal/Metal Seal; Provide Final Lock
Seal Plug →	Must Shear Open

We can consider the overall operational reliability of the mechanical components (R_M) as:

$R_M = (R_{MS})(R_{MF}) \cdot \cdot \cdot \cdot$ (6)

Where R_{MS} is the structural rel.

R_{MF} is the functional rel.

Then, if we assume that each component has been designed with a structural Safety Factor of at least 1.5 (verified by stress analysis), experience has shown that its structural reliability will be at least .999999. Therefore:

$R_{MS} = (.999999)^9 = .999991$

For each of the five listed functions we will conservatively assume a functional reliability of .999990. Thus

$R_{MF} = (.999990)^5 = .999950$

The mechanical parts reliability follows:

$R_M = (.999991)(.999950) = .999941$

2.3.1.4 Calculation of R_V

Substituting from 2.3.1.2, 2.3.1.2 and 2.3.1.3 in Equation (1) we obtain the valve's operational reliability:

$R_V = (.999908)(.999990)(.999941) = .999839$

2.3.2 Mission Reliability (R_{MV})

According to MIL-STD-756 (Ref. 3) the "Mission Reliability" is given by the equation:

$R_{MV} = (R_V)(R_{DV})$ - - - - (7)

Where R_{DV} is the probability of the unit functioning as required after being dormant for a specified period of time. NPRD-91 (Ref. 2) gives the following definition for "dormant".

"Dormant - Component or equipment is connected to a system in the normal operational configuration and experiences non-operational and/or periodic operational stresses and environmental stresses. The system may be in a dormant state for prolonged periods before being used in a mission."

A value for R_{DV} for the entire assembly can be derived from Equation 4, given failure rate (λ) data for similar equipment under dormant conditions. The closest available data in NPRD-91 (Ref. 2) is for a "Valve, Hydraulic" at a "MIL" quality level, which has a $(\lambda)D$ of .0018 failures/10^6 hours. For the assumed dormancy period of 10 years, i.e., 87,648 hours (which includes 2 extra days for leap years) we obtain:

$$R_{DV} = e^{-(.0018)(87,648)10^{-6}} = .999842$$

Then from Equation 7, the estimated Mission Reliability for the valve is

$$R_{MV} = (R_V)(R_{DV}) = (.999839)(.999842) = \underline{.999681}$$

REFERENCES:

1. Technical Memorandum - "Pyromechanical Reliability" - N. Butterfield, June 1981, Un-numbered, Martin Marietta T.M.
2. NPRD - 91 - "Non-Electronic Parts Reliability Data 1991" - Reliability Analysis Center, Rome, NY. Document generated under contract to Rome Laboratory, Griffis AFB, NY.
3. MIL-STD-756 - Reliability Modeling and Prediction.

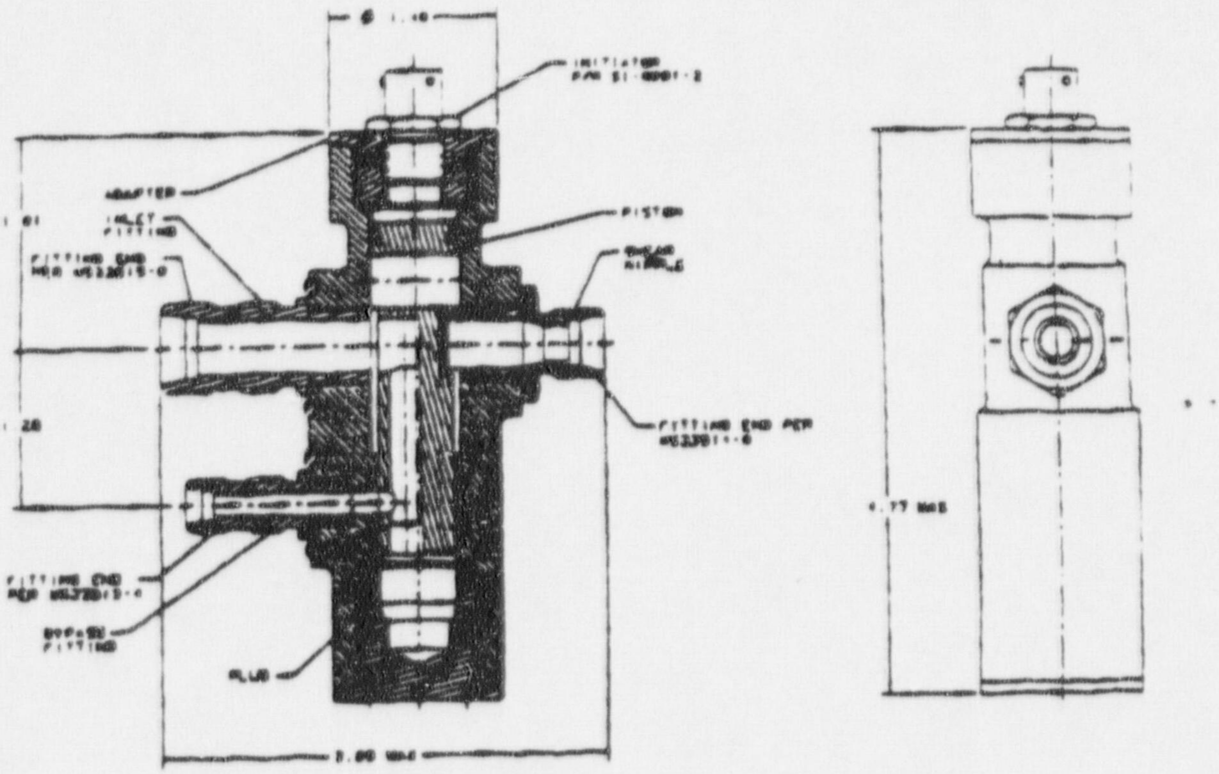


FIG. 1 <u>**THE PS/EDD Part No. 51-5875-2 FUEL VALVE**</u>

[redacted]

RECEIVED
FEB 19 1997
Brian A. Mc Intyre

February 18, 1997

Subject: Informal Transmittal of Information on
- Revision to WCAP-14407 Chapter 2 Tables

To: Ed Throm
Fax: 301-415-3577 (8 pages)

cc: Jim Gresham
Brian McIntyre
Mike Loftus

This transmittal provides to NRC a draft of a sample revision of WCAP-14407, Section 2, Tables 2-3 and 2-4 which provide the link between the PIRT phenomena and the containment pressure Evaluation Model. A revision to Tables 2-3 and 2-4 is an action from a telecon between Westinghouse and NRC on January 17, 1997.

The sample phenomena chosen cover a range of methods used to bound phenomena, and are Density of Break Source (1D), Mixing and Stratification in Containment Volume (2A), Gas Compliance in Containment Volume (2C), and Evaporation on Steel Shell (7N). The following are being addressed :

- Consistency in terminology and numbering with December 19, 1996 PIRT;
- Minor updates for consistency;
- "Page format" rather than the more cumbersome "table format."

This table is being provided early to aid in the review process. It is anticipated that a draft of revised tables 2-3 and 2-3 could be available in time for the March 6, 1997 meeting.

Joel Woodcock

Table 2-3 PIRT Application to Evaluation Model: Inside Containment-LOCA-All Phases and MSLB

PIRT ITEM	Module	PIRT Phenomena	Ranking for Containment	AP600 BCs or Phenomena Models	Test Bases	Report Submitted to NRC	Report Conclusions	Applicability of LST with Respect to Phenomena	Validation of Modeling Method and/or WGOTHIC	Use of Validation Results in this Evaluation Model	How Uncertainty is Handled
2C	I. Volume	A. Multi-Component Compressible Gases	H	Gas constituents in the governing equations	All tests analyzed with WGOTHIC	NTD-NRC-95-4563 Enclosure 1: GOTHIC Qualification Report provides large database of tests with air, hydrogen, and helium Enclosure 2: GOTHIC Technical Manual describes governing equations Enclosure 3: GOTHIC User's Manual describes how to invoke various gases NTD-NRC-95-4462 EPRI Report RA-93-10, GOTHIC Design Review, Final Report WCAP-14382 validates WGOTHIC with separate effects, integral tests with steam and air	Effects of multi-component compressible gases are correctly included in governing equations	LST includes air and steam	WGOTHIC has been validated with the LST	Governing equations in WGOTHIC are a valid representation of compressible, multi-component gas behavior. Maximum Technical Specification pressure used in conjunction with 0% relative humidity	Bounded
1D		B. Buoyancy	H	Buoyancy forces are included in the lumped parameter junction governing equations	LST internal buoyant flows	WCAP-14326, Separate effects test WCAP-14382, for integral tests	Lumped modeling overmixes noncondensables above operating deck, thereby reducing heat removal from vessel when PCS is dominant. Distributed parameter modeling shows good agreement with 550 node LST model. Modeling of buoyancy and entrainment is acceptable.	Steam injection point elevation and direction effects tests were performed. LST has prototypical buoyancy driving forces and covered the range of Proude numbers for LOCA	WGOTHIC has been validated with the LST	See boxes for line IV.A.	Bounded
2A		C. Flow Field Stability or Stratification	L	Mixing within the containment upper regions and mixing between the upper and lower portions of the containment	LST	See Section 9.	Blowdown is the same as standard plants. Long-term LOCA is driven by buoyant plume and LST covers range for AP600. MSLB is well mixed due to high velocity jet. Distributed parameter modeling shows good agreement with 550 node LST model. Modeling of buoyancy and entrainment is acceptable.	Upper and lower regions of containment represented in the LST	WGOTHIC model has been validated with the LST	See boxes for line IV.A	Bounded

Current Format for Reference

Revised Format

Table 2-?? Summary Bases and Report Cross Reference for PIRT Phenomena

Phenomena - Density of Break Source (**item 1D in PIRT**)

Ranking - High for all phases

AP600 BC's or Phenomena Models - Buoyancy forces are included in the lumped parameter junction governing equations.

Test Bases - LST internal buoyant flows.

Report Submitted to NRC:

- o WCAP-14326, Separate Effects Tests
- o WCAP-14382, LST validation results
- o WCAP-14407 Section 9, Mixing within containment

Report Conclusions:

- o Lumped modeling overmixes noncondensibles above operating deck, thereby reducing heat removal from vessel when PCS is dominant
- o Distributed parameter modeling shows good agreement with 550 node LST model. Modeling of buoyancy and entrainment is acceptable.

Applicability of LST with respect to Phenomena:

- o Steam injection point elevation, direction, and momentum affects tests performed.
- o LST had prototypical buoyancy driving forces and covered the range of Froude numbers for LOCA and MSLB.

Validation of Modeling Method and/or WGOTHIC:

- o WGOTHIC has been validated with the LST.
- o Effect of density on break source is evaluated in WCAP-14407, Section 9.

Use of Validation Results in this Evaluation Model:

- o Buoyant plume rising from the SG compartment is shown to be a limiting scenario for pressure (WCAP-14404, Section 9).

How Uncertainty is Handled:

- o Bounded by selection of limiting scenario with respect to circulation and stratification effects relative to break density.

Table 2-??

Phenomena - Mixing/Stratification in Containment Volume **(item 2A in PIRT)**

Ranking - High for all phases

AP600 BC's or Phenomena Models - Mixing within the containment upper regions and mixing between the upper and lower portions of the containment, as it is influenced by circulation and stratification, is examined outside of the evaluation model (WCAP-14407, Section 9) and methods are developed to bound the potential effects.

Test Bases - Large Scale Tests

Report Submitted to NRC:

- o WCAP-13566, "AP600 1/8th Large Scale Passive Containment Cooling System Heat Transfer Baseline Data Report"
- o WCAP-14135, "Final Data Report for PCS Large-Scale Tests, Phase 2 and Phase 3"
- o WCAP-14382 shows code influence on mixing/stratification using measured and nominal inputs
- o WCAP-14407 Section 9, Mixing evaluation

Report Conclusions:

- o Blowdown is the same as standard plants.
- o Post-blowdown LOCA is driven by buoyant plume and LST covers range for AP600.
- o MSLB is well mixed above the operating deck due to high velocity jet.
- o Distributed parameter modeling shows good agreement with 550 node LST model.
- o Buoyancy and entrainment effects on condensation to internal sinks are bounded.
- o Effects of circulation on steam distribution were ranged to select a bounding scenario.

Applicability of LST with respect to Phenomena:

- o Upper and lower regions of containment represented in the LST,
  - stratification data from above deck region is applicable
  - lack of SG flow path in LST prevents its use for studying circulation effects

Validation of Modeling Method and/or WGOTHIC:

- o WGOTHIC model has been validated with the LST (WCAP-14382)

Use of Validation Results in this Evaluation Model:

- o Evaluation model bounds effects of mixing/stratification as discussed in Section 9.

How Uncertainty is Handled:

- Bounded

Table 2-??

Phenomena - Gas Compliance in Containment Volume **(item 2C in PIRT)**

Ranking - High for all phases

AP600 BC's or Phenomena Models - Gas constituents in the governing equations.

Test Bases - All tests analyzed with WGOTHIC

Report Submitted to NRC:

- o NTD-NRC-95-4563
  Enclosure 1 - GOTHIC Qualification Report provides large database of tests with air, hydrogen, and helium
  Enclosure 2 - GOTHIC Technical Manual describes governing equations
  Enclosure 3 - GOTHIC User's Manual describes how to invoke various gases
- o NTD-NRC-95-4462, EPRI Report RA-93-10, GOTHIC Design Review Final Report
- o WCAP-14382 validates WGOTHIC with separate effects, integral tests with steam and air

Report Conclusions:

- o Effects of multi-component compressible gases are correctly included in governing equations.

Applicability of LST with respect to Phenomena:

- o LST includes air and steam in an enclosed volume.

Validation of Modeling Method and/or WGOTHIC:

- o WGOTHIC has been validated with the LST (WCAP-14382).

Use of Validation Results in this Evaluation Model:

- o Governing equations in WGOTHIC are a valid representation of compressible, multi-component gas behavior.
- o Maximum Technical Specification pressure used in conjunction with low estimate of containment volume.

How Uncertainty is Handled:

- o Bounded

Table 2-??

Phenomena - Evaporation on Steel Shell **(item 7N in PIRT)**

Ranking - High for all phases

AP600 BC's or Phenomena Models - Empirical correlation for the Sherwood number which is derived by dimensional analysis using the heat and mass transfer analogy and Colburn j factors. Application of a correction for mass transfer rate gives the AP600 forced convection mass transfer correlation.

Test Bases - Gilliland and Sherwood evaporation tests and Westinghouse STC flat plate evaporation tests.

Report Submitted to NRC:

- o NTD-NRC-95-4397, "Supporting Information for the Use of Forced Convection in the AP600 PCS Annulus"
- o WCAP-14326, Separates Effects Tests gives correlation (sections 2.0, 2.1), entrance effect used for separate effect test (section 2.2), and correlation validation with tests (sections 3.6, 3.7, and 4.2)

Report Conclusions:

- o AP600 shown to operate in forced convection dominant regime.
- o Correlation is biased 6.4% conservative with reasonable scatter over the range.
- o Once the outer shell heats up to at least 2F above ambient, the AP600 annulus operates in forced convection.

Applicability of LST with respect to Phenomena:

- o LST includes tests with and without fan on, covering the annulus from mixed convection through forced convection regimes.
- o WCAP-14382: Predictions of total evaporation (page 8-3)and wall heat flux (page 8-6) validate models in an integral setting.

Validation of Modeling Method and/or WGOTHIC:

- o WCAP-14382 summarizes WGOTHIC separate effects validation results (sections 3.2.1, and 4.4)

Use of Validation Results in this Evaluation Model:

- o Forced convection correlation, modified for mixed convection effects to allow transient startup is appropriate for AP600.
- o A conservative bias of 0.83 times the nominal correlation is used.

How Uncertainty is Handled:

- o Bounded

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FAX COVER SHEET

RECIPIENT INFORMATION		SENDER INFORMATION	
DATE:	2/15/07	NAME:	Robin Nydes
TO:	Tom Kenyon	LOCATION:	ENERGY CENTER - EAST
PHONE:	FACSIMILE:	PHONE:	Office:
COMPANY:		Facsimile:	win: 284-4887 outside: (412)374-4887
LOCATION:			

Cover + Pages 1 + 55 (last page is DSER item 4257.)

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:

Tom - Here is the complete package of Ch 7 changes for Rev 11. Pls pass to Hulbert. Thanks

Robin

Changes Previously Approved by NRC
to be Incorporated in SSAR Revision 11

Operability, Availability, and Testing

The diverse actuation system is designed to provide protection under all plant operating conditions in which the reactor vessel head is in place. The automatic actuation processors, in each of the two redundant automatic subsystems of the diverse actuation system, are provided with the capability for channel calibration and testing while the plant is operating. To prevent inadvertent DAS actuations during online calibration, testing activities or maintenance, the normal activation function is bypassed. Testing of the diverse actuation system is performed on a periodic basis.

Equipment Qualification and Quality Standards

The diverse actuation system is capable of functioning during and after normal and abnormal events and conditions that include:

- Excessive temperature
- Ambient vibration
- Radio frequency and electromagnetic interference

, including actuated devices,

The diverse actuation system equipment ~~is~~ designed and qualified in accordance with the industry standards listed in subsection 7.1.4.1.8. The adequacy of the hardware and software is demonstrated through the verification and validation program discussed in subsection 7.1.2.15. This program provides for commercial dedication of commercial off-the-shelf hardware and software. As the diverse actuation system performs many of the protection functions associated within the ATWS systems used in existing plants, the diverse actuation system is designed to meet the quality guidelines established by Generic Letter 85-06, "Quality Assurance Guidelines for ATWS Equipment that is not Safety-Related."

7.7.1.12 Signal Selector

The plant control system for the AP600 derives some of its control inputs from signals that are also used in the protection and safety monitoring system. The advantages of this design are:

- The nonsafety-related plant systems are controlled from the same measurements which provide protection. This permits the control system to function in a manner which maintains margin between operating conditions and safety limits, and reduces the likelihood of spurious trips.

- Reducing the number of redundant measurements for any single process variable reduces the overall plant complexity at critical pressure boundary penetrations. This leads to a reduction in separation requirements within the containment, as well as to a decrease in plant cost and maintenance requirements.

To obtain these advantages, measures are taken to provide the independence of the protection and control systems. The criteria for these measures are contained in the Standard IEEE

Operation procedures prohibit testing two divisions at the same time. There are no built-in interlocks to prevent simultaneous testing of two integrated protection cabinets. However, the use of bypasses by the tester provides that the protection and safety monitoring system cannot be placed in an unsafe condition if the procedure prohibiting simultaneous testing is violated. For example, testing two divisions results in two bypasses, which causes the voting logic to revert to a one-out-of-two coincidence for the remaining two unbypassed divisions. Attempting to test three or four divisions at the same time causes a plant trip. The operational procedure restricting simultaneous testing of two or more divisions is for operability reasons to avoid unnecessary trips.

In addition to periodic tests, the system performs error detection and data link testing as part of its normal operation. Where practical, the on-line error detecting features are designed to automatically place the channel in which the error was detected into a trip or bypass state (either by direct bypass or reconfiguration). When a channel is automatically placed into a trip state, the operator has the option to subsequently place that channel in a bypass state. If the automatic configuration of the channel is not practical, the on-line error detecting feature causes alarm annunciation to the operator.

7.1.2.13 Safety-Related Display Instrumentation

Safety-related display instrumentation provides the operator with information to determine the effect of automatic and manual actions taken following reactor trip due to a Condition II, III, or IV event as defined in Chapter 15. This instrumentation also provides for operator display of the information necessary to meet Regulatory Guide 1.97. A description of the equipment used to provide this function is provided in subsection 7.1.2.6. A description of the data provided to the operator by this instrumentation is provided in Section 7.5.

7.1.2.14 Auxiliary Supporting Systems

The safety-related system equipment is supported by the supply of uninterruptable electrical energy. This electrical power is supplied by the Class 1E dc and UPS system discussed in Chapter 8.

change on next page

7.1.2.15 Verification and Validation

Adequacy of the hardware and software is demonstrated for the protection and safety monitoring system through a verification and validation (V&V) program. Details on the verification and validation program are provided in WCAP-13383 (Reference 4). The software development process which is documented in this document is consistent with the following standards:

- ANSI/IEEE ANS-7-4.3.2 (1993); "Application Criteria for Programmable Digital Computer Systems in Safety Systems for Programmable Digital Computer Systems in Safety Systems of Nuclear Power Generating Stations"

- IEC 880-1986: "Software for Computers in the Safety Systems for Nuclear Power Generating Stations"
- IEEE 828-1983: "IEEE Standard for Software Configuration Management Plans"
- IEEE 829-1983: "IEEE Standard for Software Test Documentation"
- IEEE 830-1984: "IEEE Standard for Software Requirements Specifications"
- IEEE 1012-1986; "IEEE Standard for Software Verification and Validation Plans"
- IEEE 1042-1987; "IEEE Guide to Software Configuration Management (ANSI)"

WCAP-13383 also provides for the use of commercial off-the-shelf hardware and software through a commercial grade dedication process.

7.1.3 Plant Control System

The plant control system is a nonsafety-related system that provides control and coordination of the plant during startup, ascent to power, power operation, and shutdown conditions. The plant control system integrates the automatic and manual control of the reactor, reactor coolant, and various reactor support processes for required normal and off-normal conditions. The plant control system also provides control of the nonsafety-related decay heat removal systems during shutdown. The plant control system accomplishes these functions through use of the following:

- Rod control
- Pressurizer pressure and level control
- Steam generator water level control
- Steam dump (turbine bypass) control
- Rapid power reduction

The plant control system provides automatic regulation of reactor and other key system parameters in response to changes in operating limits (load changes). The plant control system acts to maximize margins to plant safety limits and maximize the plant transient performance. The plant control system also provides the capability for manual control of plant systems and equipment. Redundant control logic is used in some applications to increase single-failure tolerance.

The plant control system includes the equipment from the process sensor input circuitry through to the modulating and nonmodulating control outputs as well as the digital signals to other plant systems. Modulating control devices include valve positioners, pump speed controllers, and the control rod equipment. Nonmodulating devices include motor starters for motor-operated valves and pumps, breakers for heaters, and solenoids for actuation of air-operated valves. The control cabinets contain the process sensor inputs and the modulating and nonmodulating outputs. The plant control system also includes equipment to monitor and control the control rods.

7.1.4.2.22 Conformance to the Requirements for Identification of Redundant Safety System Equipment (Paragraph 4.22 of IEEE 279-1971)

Distinctive markings are applied to redundant divisions of the protection and safety monitoring system.

The color coded nameplates described below provide identification of equipment, associated with protective functions and their divisions associations.

Division	Color Coding
Division A	BROWN with WHITE lettering
Division B	GREEN with BLACK lettering
Division C	BLUE with WHITE lettering
Division D	YELLOW with BLACK lettering

Non-cabinet mounted protective equipment and components have an identification tag or nameplate. Small electrical components such as relays, have nameplates on the enclosure that houses them.

7.1.5 AP600 Protective Functions

Protective functions are those necessary to achieve the system responses assumed in the safety analyses, and those needed to shut down the plant safely. The protective functions are grouped into two classes, reactor trip and engineered safety features actuation.

Reactor trip is discussed in Section 7.2. Engineered safety features actuation is discussed in Section 7.3.

7.1.6 Combined License Information

For ~~This section, has no~~ the only requirement ~~for information to be provided in support of the Combined License application,~~ is for calculation of setpoints for protective functions ~~setpoints~~ consistent with the methodology presented in Reference 8.

Combined Licence applicants referencing the AP600 certified design will provide a

7.1.7 References

1. IEEE 603-1991, "IEEE Criteria for Safety Systems for Nuclear Power Generator Stations."

2. IEEE 796-1983, "IEEE Microcomputer System Bus."

3. WCAP-13382 (P), WCAP-13391 (NP), "AP600 Instrumentation and Control Hardware Description."

4. WCAP-13383 (P), WCAP-13392 (NP), "AP600 Instrumentation and Control Hardware and Software Design, Verification, and Validation Process Report."

5. IEEE 279-1971, "IEEE Criteria for Protection Systems for Nuclear Power Generating Stations."

6. IEEE 384-1981, "IEEE Criteria for Independence or Class 1E Equipment and Circuits."

7. WCAP-8897 (P), WCAP-8898 (NP), "Bypass Logic for the Westinghouse Integrated Protection System."

8. WCAP-14605(P), WCAP-14606(NP), "Westinghouse Setpoint Methodology for Protection Systems, AP600."

Table 1.6-1 (Sheet 11 of 15)

MATERIAL REFERENCED

SSAR Section Number	Westinghouse Topical Report Number	Title
6.2	WCAP-14382	WGOTHIC Code Description and Validation
	WCAP-8077 (P) WCAP-8078	Ice Condenser Containment Pressure Transient Analysis Methods
	WCAP-8264-P-A (P) WCAP-8312-A	Westinghouse Mass and Energy Release Data for Containment Design
	WCAP-10325 (P)	Westinghouse LOCA Mass and Energy Release Model for Containment Design - March 1979 Version
	WCAP-8822 (P) WCAP-8860	Mass and Energy Releases Following A Steam Line Rupture
	WCAP-7907-P-A (P) WCAP-7907-A	LOFTRAN Code Description
	WCAP-12945-P (P)	Code Qualification Document for Best Estimate Analysis
	WCAP-14407 (P) WCAP-14408	WGOTHIC Application to AP600
6.3	WCAP-8966	Evaluation of Mispositioned ECCS Valves
7.1	WCAP-13382 (P) WCAP-13391	AP600 Instrumentation and Control Hardware Description
	WCAP-13383 (P) WCAP-13392	AP600 Instrumentation and Control Hardware and Software Design, Verification, and Validation Process Report
	WCAP-8897 (P) WCAP-8898	Bypass Logic for the Westinghouse Integrated Protection System
7.2	WCAP-13594 (P) WCAP-13662	FMEA of Advanced Passive Plant Protection System
8.3	WCAP-13856	AP600 Implementation of the Regulatory Treatment of Nonsafety-Related Systems Process
10.2	WCAP-11525	Probabilistic Evaluation of Reduction in Turbine Valve Test Frequency
	WCAP-14605 (P) WCAP-14606 (NP)	Westinghouse Setpoint methodology for Protection Systems - AP600

(P) Denotes Document is Proprietary

Table 1.8-2 (Sheet 3 of 4)

SUMMARY OF AP600 STANDARD PLANT COMBINED LICENSE INFORMATION ITEMS

Item No.	Subject	Subsection
6.4-2	Local Toxic Gas Services and Monitoring	6.4.7
6.4-3	Procedures for Training for Control Room Habitability	6.4.7
6.6-1	Inspection Programs	6.6.9.1
6.6-2	Construction Activities	6.6.9.2
7.1-1	Setpoint Calculations for Protective Functions	7.1.6
8.2-1	Offsite Electrical Power	8.2.4
8.3-1	Onsite Electrical Power	8.3.3
9.1-1	Fuel Storage and Handling	9.1.6
9.5-1	Offsite Communications Interfaces	9.5.2.5.1
9.5-2	Emergency Response Facility Communications	9.5.2.5.2
9.5-3	Security Communications	9.5.2.5.3
9.5-4	Cathodic and Environmental Protection for Fuel Oil Tanks	9.5.4.7
10.1-1	Erosion-Corrosion Monitoring	10.1.3
10.2-1	Turbine Maintenance and Inspection	10.2.6
10.4-1	Circulating Water Supply	10.4.12.1
10.4-2	Condensate, Feedwater and Auxiliary Steam System Chemistry Control	10.4.12.2
10.4-3	Potable Water	10.4.12.3
11.2-1	Liquid Radwaste Processing by Mobile Equipment	11.2.4.1
11.2-2	Cost Benefit Analysis of Population Doses from Liquid Effluents	11.2.4.2
11.2-3	Identification of Ion Exchange and Adsorbent Media for Liquid Radwaste	11.2.4.3
11.2-4	Dilution and Control of Boric Acid Discharge	11.2.4.4
11.3-1	Cost Benefit Analysis of Population Doses from Gaseous Effluents	11.3.4.1
11.3-2	Identification of Adsorbent Media for Gaseous Radwaste	11.3.4.2
11.4-1	Solid Waste Management System Process Control Program	11.4.6
11.5-1	Plant Offsite Dose Calculation Manual (ODCM)	11.5.7
12.1-1	ALARA and Operational Policies	12.1.3
12.2-1	Additional Contained Radiation Sources	12.2.3
12.3-1	Administrative Controls, Criteria and Methods for Radiological Protection	12.3.5

core cooling monitor. The incore instrument assemblies house both fixed incore detectors and core exit thermocouples. The incore instrumentation system is described in subsection 4.4.6.1.

7.1.2 General Protection Subsystem Configuration

The protection and safety monitoring system is illustrated in Figure 7.1-2. The functions of the protection and safety monitoring system have been decomposed into physically and electrically separate microprocessor based subsystems. Each subsystem is located on an independent computer bus to prevent propagation of failures and to enhance availability. In most cases, each subsystem is implemented in a separate card chassis. Subsystem independence is maintained through the use of the following:

- Separate dc power sources with output protection to prevent interaction between subsystems upon failure of a subsystem.
- Separate input or output circuitry to maintain independence at the subsystem interfaces.
- Deadman signals: A device, circuit, or function that forces a predefined operating condition upon the cessation of a normally dynamic input parameter to improve the reliability of hard-wired data that crosses the subsystem interface.
- Optical coupling or resistor buffering between two subsystems or between a subsystem and an input/output (I/O) module.

WCAP-13382 (Reference 3) provides a description of the hardware elements which comprise the protection and safety monitoring system configuration. WCAP-14080 Reference 8 provides a description of the software architecture and operation.

7.1.2.1 Functional Components

The type and number of boards used to implement the functions of a microprocessor based subsystem are purposely limited to aid serviceability and to restrict the number of spares. In addition, the basic function of a particular board remains fixed among subsystems to facilitate the development and maintenance of the subsystem software. IEEE 796 (Reference 2) bus cards are typically used to provide functions as listed below.

Functional Processor

The functional processor performs the major computations required to achieve the specific function of the microprocessor based subsystem. Tasks performed by the functional processor include movement of data between subsystem memories or I/O registers for the purpose of input or output, on-line compensation of the analog inputs, conversion of input data to engineering units, and diagnostic testing. A functional processor is included in each subsystem.

7.1.7 References

1. IEEE 603-1991, "IEEE Criteria for Safety Systems for Nuclear Power Generator Stations."

2. IEEE 796-1983, "IEEE Microcomputer System Bus."

3. WCAP-13382 (P), WCAP-13391 (NP), "AP600 Instrumentation and Control Hardware Description."

4. WCAP-13383, Revision 1 (NP), "AP600 Instrumentation and Control Hardware and Software Design, Verification, and Validation Process Report."

5. IEEE 279-1971, "IEEE Criteria for Protection Systems for Nuclear Power Generating Stations."

6. IEEE 384-1981, "IEEE Criteria for Independence or Class 1E Equipment and Circuits."

7. WCAP-8897 (P), WCAP-8898 (NP), "Bypass Logic for the Westinghouse Integrated Protection System."

8. WCAP-14080 (P), WCAP-14081 (NP), "AP600 Instrumentation and Control Software Architecture and Operation Description", Revision 0

Table 1.6-1 (Sheet 11 of 15)

MATERIAL REFERENCED

SSAR Section Number	Westinghouse Topical Report Number	Title
6.2	WCAP-14382	WGOTHIC Code Description and Validation
	WCAP-8077 (P) WCAP-8078	Ice Condenser Containment Pressure Transient Analysis Methods
	WCAP-8264-P-A (P) WCAP-8312-A	Westinghouse Mass and Energy Release Data for Containment Design
	WCAP-10325 (P)	Westinghouse LOCA Mass and Energy Release Model for Containment Design - March 1979 Version
	WCAP-8822 (P) WCAP-8860	Mass and Energy Releases Following A Steam Line Rupture
	WCAP-7907-P-A (P) WCAP-7907-A	LOFTRAN Code Description
	WCAP-12945-P (P)	Code Qualification Document for Best Estimate Analysis
	WCAP-14407 (P) WCAP-14408	WGOTHIC Application to AP600
6.3	WCAP-8966	Evaluation of Mispositioned ECCS Valves
7.1	WCAP-13382 (P) WCAP-13391	AP600 Instrumentation and Control Hardware Description
	WCAP-13383 (P) WCAP-13392	AP600 Instrumentation and Control Hardware and Software Design, Verification, and Validation Process Report
	WCAP-8897 (P) WCAP-8898	Bypass Logic for the Westinghouse Integrated Protection System
7.2	WCAP-13594 (P) WCAP-13662	FMEA of Advanced Passive Plant Protection System
8.3	WCAP-13856	AP600 Implementation of the Regulatory Treatment of Nonsafety-Related Systems Process
10.2	WCAP-11525	Probabilistic Evaluation of Reduction in Turbine Valve Test Frequency
	WCAP-14080 (P) WCAP 14081	AP600 Instrumentation and Control Software Architecture and Operation Description

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Changes Reflecting Resolution of NRC PAM/ERG Comments

Table 7.5-5

Summary of Type B Variables

Function Monitored	Variable	Type/Category
Reactivity Control	Neutron flux	B1
	Control rod position	B3
	Boric acid concentration	B3
Reactor Coolant System Integrity	RCS pressure	B1
	RCS wide range T_{hot}	B1
	RCS wide range T_{cold}	B1
	Containment water level	B1
	Containment pressure	B1
Reactor Coolant Inventory Control	Pressurizer level	B1
	Pressurizer reference leg temperature	B1
	Pressurizer pressure	B1
	Reactor vessel - hot leg water level	B3
Reactor Core Cooling	Core exit temperature	B1
	RCS subcooling	B1
	RCS wide range T_{hot}	B2
	RCS wide range T_{cold}	B2
	RCS pressure	B2
	Reactor vessel - hot leg water level	B2
Heat Sink Maintenance	IRWST water level	B1
	PRHR flow	B1
	PRHR outlet temperature	B1
	PCS storage tank water level	B1
	Passive containment cooling water flow	B1
	IRWST to RNS suction valve status	B1
Containment Environment	Containment pressure	B1
	Remotely operated containment isolation valve ~~position~~ status	B1

Table 7.5-7 (Sheet 1 of 4)

Summary of Type D Variables

System	Variable	Type/Category
Reactivity Control System	Reactor trip breaker status	D2
	Control rod position	D3
Pressurizer Level and Pressure Control	Pressurizer safety valve status	D2
	Pressurizer level	D2
	RCS pressure	D2
	Pressurizer pressure	D2
	Reference leg temperature	D2
RCS Loops	RCS wide range T_{hot}	D2
	RCS wide range T_{cold}	D2
	RCP breaker status	D2
Secondary Pressure and Level Control	Steam generator PORV status	D2
	Steam generator PORV block valve status	D2
	Steam generator safety valve status	D2
	Main feedwater isolation valve status	D2
	Steam generator level (wide range)	D2
	Steam generator level (narrow range)	D2
	Steam generator blowdown isolation valve status	D2

RCP bearing water temperature D2

Table 7.5-7 (Sheet 2 of 4)

Summary of Type D Variables

System	Variable	Type/Category
Secondary Pressure and Level Control (continued)	Steam line pressure	D2
	Main feedwater pump status	D2
	Main feedwater control valve status	D2
	Main steam line isolation valve status	D2
	Main steam line isolation bypass valve status	D2
Startup Feedwater	Startup feedwater flow	D2
	Startup feedwater control valve status	D2
	Startup feedwater isolation valve status	D2
	Main to startup feedwater crossover valve status	D2
Safeguards	Containment pressure	D2
	Accumulator level	D2
	Core makeup tank level	D2
	IRWST ~~to reactor vessel~~ line isolation valve status (MOV)	D3
	IRWST ~~to reactor vessel~~ injection isolation valve status (~~non-MOV~~ Squib)	D2
	ADS first stage, second stage and third stage valve status	D2
	ADS fourth stage valve status (MOV)	D2
	ADS fourth stage valve status (non-MOV)	D2
	PRHR heat exchanger inlet isolation valve status	D3
	PRHR heat exchanger ~~discharge isolation~~ control valve status	D2
	Reactor vessel head vent valve status	D2
	CMT ~~to reactor vessel~~ discharge isolation valve status	D2
	CMT inlet isolation valve status	D2
	Accumulator ~~to reactor vessel~~ isolation valve status	D3

Table 7.5-7 (Sheet 4 of 4)

Summary of Type D Variables

System	Variable	Type/Category
Containment Cooling	Containment temperature	D2
	PCS water storage tank ~~to containment~~ series isolation valve status (MOV)	D2
	PCS water storage tank ~~to containment~~ isolation valve status (non-MOV)	D2
	Passive containment cooling water flow	D2
	PCS storage tank water level	D2
HVAC System Status	MCR return air isolation damper status	D2
	MCR toilet exhaust isolation damper status	D2
	MCR supply air isolation damper status	D2
	MCR air delivery isolation valve status	D2
	MCR air storage bottle pressure	D2
	MCR supply air radiation level	D2
Main Steam	Turbine stop valve status	D2
	Turbine control valve position	D2
	Condenser steam dump valve status	D2

Table 7.5-9 (Sheet 2 of 4)

Summary of Type F Variables

Variable	Type/Category
Startup feedwater control valve status	F3
Main feedwater flow	F3
Steam generator level (WR)	F3
Steam flow	F3
Main steam line isolation valve status	F3
Main feedwater pump status	F3
Startup feedwater pump status	F3
Condenser steam dump valve status	F3
Condensate storage tank level	F3
Pressurizer spray (- cold leg to pressurizer) valve status	F3
Auxiliary spray line isolation valve status	F3
Makeup flow	F3
Makeup pump status	F3
Letdown flow	F3
Circulating water pump breaker status	F3
Condenser backpressure	F3
Accumulator vent valve status	F3

Table 7.5-9 (Sheet 3 of 4)

Summary of Type F Variables

Variable	Type/Category
Boric acid tank level	F3
Boric acid flow	F3
Makeup ~~pump suction header~~ blend valve status	F3
Makeup flow control valve status	F3
RNS flow	F3
IRWST to RNS suction valve status	F3
RNS discharge to IRWST valve status	F3
CCS surge tank level	F3
CCS flow	F3
CCS pump status	F3
CCS flow to RNS valve status	F3
CCS flow to RCPs valve status	F3
CCS heat exchanger inlet temperature	F3
CCS heat exchanger outlet temperature	F3
Diesel generator status	F3
Containment fan cooler status	F3
Chilled water pump status	F3
Chilled water valve status	F3
Containment temperature	F3
Main control room supply air isolation damper status	F3
Main control room return air isolation damper status	F3
Main control room supply air radiation	F3
Service water flow	F3

Diesel generator load F3
Voltage for diesel-backed buses F3
Power supply to diesel-backed buses F3

RNS pump status F3

Table 7.5-9 (Sheet 4 of 4)

Summary of Type F Variables

Variable	Type/Category
Service water pump status	F3
Service water pump discharge valve status	F3
Service water pump discharge temperature	F3
Instrument air header pressure	F3
Spent fuel pool pump flow	F3
Spent fuel pool temperature	F3
Spent fuel pool water level	F3
Main to startup feedwater crossover valve status	F3
~~Gutter to containment sump valve status~~ IRWST gutter bypass isolation valve status	F3

Changes for Auxiliary Spray and CVS Letdown

Condition 1 results from a coincidence of two of the four divisions of reactor loop average temperature (T_{avg}) below the Low-2 setpoint coincident with the P-4 permissive (reactor trip). This blocks the opening of the steam dump valves. This signal also becomes an input to the steam dump interlock selector switch for unblocking the steam dump valves used for plant cooldown. This function may be manually blocked when the pressurizer pressure is below the P-11 setpoint. The block is automatically removed when the pressurizer pressure is above the P-11 setpoint.

Condition 2 consists of two controls. Either one of these controls can be used to manually initiate a steam dump block.

The functional logic relating to the steam dump block is illustrated in Figure 7.2-1, sheet 10.

7.3.1.2.17 Control Room Isolation and Air Supply Initiation

Signals to initiate isolation of the main control room and to initiate the air supply are generated from either of the following conditions:

1. High control room air supply radioactivity level
2. Loss of ac power sources

Condition 1 is the occurrence one of two control room air supply radioactivity monitors detecting a radioactivity level above the High-2 setpoint.

Condition 2 results from the loss of all ac power sources. A preset time delay is provided to permit the restoration of ac power from the offsite sources or from the onsite diesel generators before initiation. The loss of all ac power is detected by undervoltage sensors that are connected to the input of each of the four Class 1E battery chargers. Two sensors are connected to each of the four battery charger inputs. The loss of ac power signal is based on the detection of an undervoltage condition by each of the two sensors connected to two of the four battery chargers. The two-out-of-four logic is based on an undervoltage to the battery chargers for divisions A or C coincident with an undervoltage to the battery chargers for divisions B or D.

The functional logic relating to control room isolation and air supply initiation is illustrated in Figure 7.2-1, sheet 13.

7.3.1.2.18 Auxiliary Spray and Letdown Purification Line Isolation

A signal to isolate the auxiliary spray and letdown purification lines is generated upon the coincidence of pressurizer level below the Low-1 setpoint in any two of four divisions. This helps to maintain reactor coolant system inventory. This function can be manually blocked when the pressurizer water level is below the P-12 setpoint. This function is automatically unblocked when the pressurizer water level is above the P-12 setpoint. The functional logic relating to this is illustrated in Figure 7.2-1, sheet 12.

7.3.1.2.19 Containment Air Filtration System Isolation

A signal to isolate the containment air filtration system is generated upon the coincidence of containment radioactivity above the High-1 setpoint in any two of four divisions. This limits activity release to the environment. The functional logic relating to this is illustrated in Figure 7.2-1, sheet 13.

7.3.1.2.20 Normal Residual Heat Removal System Isolation

A signal for isolating the normal residual heat removal system lines is generated upon the coincidence of containment radioactivity above the High-2 setpoint in any two of four divisions. This signal also isolates the chemical and volume control system as discussed in subsection 7.3.1.2.15. This limits activity release to the environment. The functional logic relating to this is illustrated in Figure 7.2-1, sheet 13.

7.3.1.2.21 Spent Fuel Pool Isolation

A signal for isolating the spent fuel pool lines is generated upon the coincidence of spent fuel pool level below the Low setpoint in any one of two divisions. This helps to maintain the water inventory in the spent fuel pool due to line leakage. The functional logic relating to this is illustrated in Figure 7.2-1, sheet 13.

[ADD INSERT 7.3.1.2.22]

7.3.1.3 Blocks, Permissives, and Interlocks for Engineered Safety Features Actuation

The interlocks used for engineered safety features actuation are designated as "P-xx" permissives and are listed in Table 7.3-2.

7.3.1.4 Bypasses of Engineered Safety Features Actuations

The channels used in engineered safety features actuation that can be manually bypassed are indicated in Table 7.3-1. A description of this bypass capability is provided in subsection 7.1.2.10. The actuation logic is not bypassed for test. During tests, the actuation logic is fully tested by blocking the actuation logic output before it results in component actuations.

7.3.1.5 Design Basis for Engineered Safety Features Actuation

The following subsections provide the design bases information for engineered safety features actuation, including the information required by Section 3 of IEEE 279-1971. Engineered safety features are initiated by the protection and safety monitoring system. Those design bases relating to the equipment that initiates and accomplishes engineered safety features are given in subsection 7.1.4.1. The design bases presented here concern the variables monitored for engineered safety features actuation and the minimum performance requirements in generating the actuation signals.

[INSERT 7.3.1.1.22]

7.3.1.2.22 Chemical and Volume Control System Letdown Isolation

A signal to isolate the letdown valves of the chemical and volume control system is generated upon the coincidence of low loop 1 and loop 2 hot leg levels. This helps to maintain reactor system inventory. The functional logic relating to this is illustrated in Figure 7.2-1, sheet 16. These letdown valves are also closed by the containment isolation function as described in subsection 7.3.1.2.1.

Table 7.3-1 (Sheet 7 of 8)

ENGINEERED SAFETY FEATURES ACTUATION SIGNALS

Actuation Signal	No. of Channels/ Switches	Actuation Logic	Permissives and Interlocks
16. Main Control Room Isolation and Air Supply Initiation (Figure 7.2-1, Sheet 13)			
a. High-2 control room supply air radiation	2	1/2	None
b. Undervoltage to Class 1E battery chargers	2/charger	2/2 per charger and 2/4 chargers[5]	None
Auxiliary Spray and **17. Purification Line Isolation** (Figure 7.2-1, Sheet 12)			
a. Low-1 pressurizer level	4	2/4-BYP[4]	Manual block permitted below P-12. Automatically unblocked above P-12.
18. Containment Air Filtration System Isolation (Figure 7.2-1, Sheet 13)			
a. High-1 containment radioactivity	4	2/4-BYP[4]	None
19. Normal Residual Heat Removal System Isolation (Figure 7.2-1, Sheet 13)			
a. High-2 containment radioactivity	4	2/4-BYP[4]	None
20. Spent Fuel Pool Isolation (Figure 7.2-1, Sheet 13)			
a. Low spent fuel pool level	2	1/2	None
21. Open In-Containment Refueling Water Storage Tank (IRWST) Injection Line Valves (Figure 7.2-1, Sheet 16)			
a. Automatic reactor coolant system depressurization (fourth stage)		(See items 3d and 3e)	
b. Coincident loop 1 and loop 2 low hot leg level (after delay)	1 per loop	2/2	None
c. Manual initiation	4 switches	2/4 switches[3]	None
22. Open IRWST Containment Recirculation Valves In Series with Check Valves (Figure 7.2-1, Sheet 15)			
a. Extended undervoltage to Class 1E battery chargers	2/charger	1/2 per charger and 2/4 chargers	None

Table 7.3-1 (Sheet 8 of 8)

ENGINEERED SAFETY FEATURES ACTUATION SIGNALS

Actuation Signal	No. of Channels/ Switches	Actuation Logic	Permissives and Interlocks
23. Open All IRWST Containment Recirculation Valves (Figure 7.2-1, Sheet 16)			
b. Safeguards actuation signal (automatic or manual) coincident with		(See items 1a through 1e)	
Low IRWST level (Low-3 setpoint)	4	2/4 BYP[1]	None
c. Manual initiation	4 switches	2/4 switches	None

Notes:

1. 2/4-BYP indicates automatic bypass logic. The logic is 2 out of 4 with no bypasses; 2 out of 3 with one bypass; 1 out of 2 with two bypasses; and, automatically actuated with three or four bypasses.
2. Any two channels from either tank not in same division.
3. Two switches must be actuated simultaneously.
4. Also, closes power-operated relief block valve of respective steam generator.
5. The two-out-of-four logic is based on undervoltage to the battery chargers for divisions A or C coincident with an undervoltage to the battery chargers for divisions B or D.
6. Any two channels from either loop not in same division.
7. Any two channels from either line not in same division.

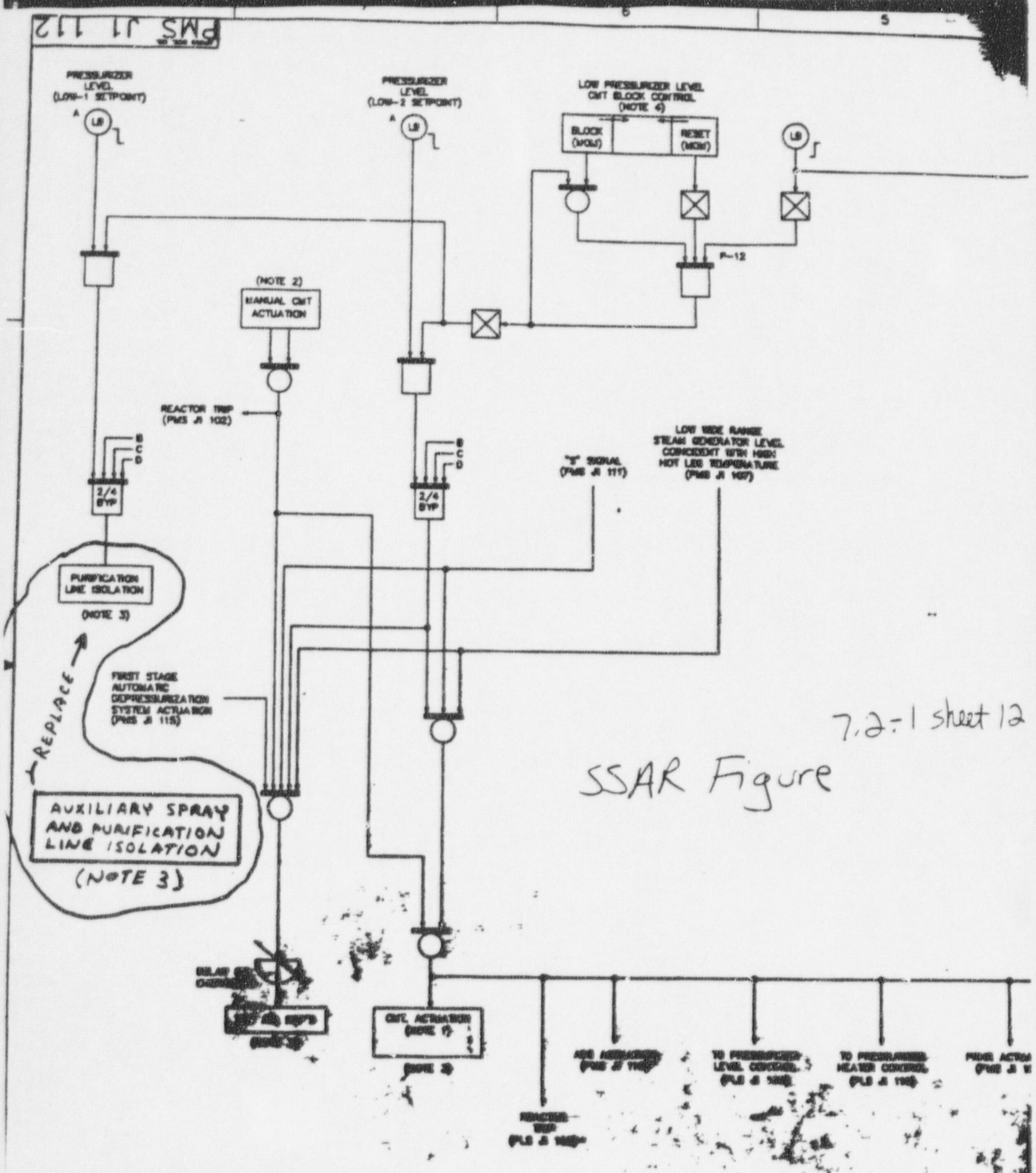
24. Chemical and Volume Control System Letdown Isolation (Figure 7.2-1, Sheet 16

a. Coincident loop 1 and loop 2 low hot leg level	1 per loop	2/2	None

Table 7.3-2 (Sheet 3 of 3)

INTERLOCKS FOR ENGINEERED SAFETY FEATURES ACTUATION SYSTEM

Designation	Derivation	Function
P-12	Pressurizer level below setpoint	(a) Permits manual block of core makeup tank actuation on low pressurizer level to allow mid-loop operation (b) Permits manual block of reactor coolant pump trip on low pressurizer level to allow mid-loop operation (c) Permits manual block of auxiliary spray and purification line isolation on low pressurizer level to allow mid-loop operation
$\overline{\text{P-12}}$	Pressurizer level above setpoint	(a) Prevents manual block of core makeup tank actuation on low pressurizer level (b) Prevents manual block of reactor coolant pump trip on low pressurizer level (c) Prevents manual block of auxiliary spray and purification line isolation on low pressurizer level (d) Provides confirmatory open signal to the core makeup tank cold leg balance lines



SSAR Figure 7.2-1 sheet 12

DWG. NO. AP6-300 309210-01 REVISION 1

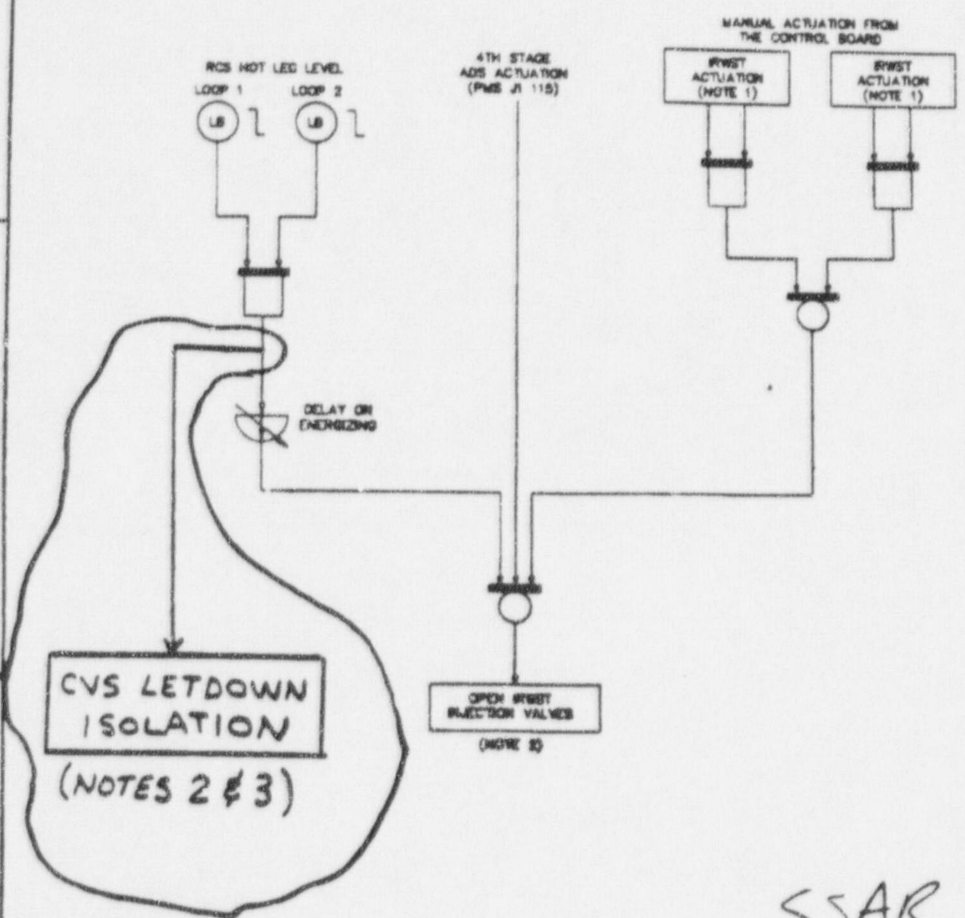
AP6-300 EDN-97823 5

ADDED PRESS LEVEL LOW-1 SETPOINT LOGIC

RELOCATED PURIF LINE ISOLATION

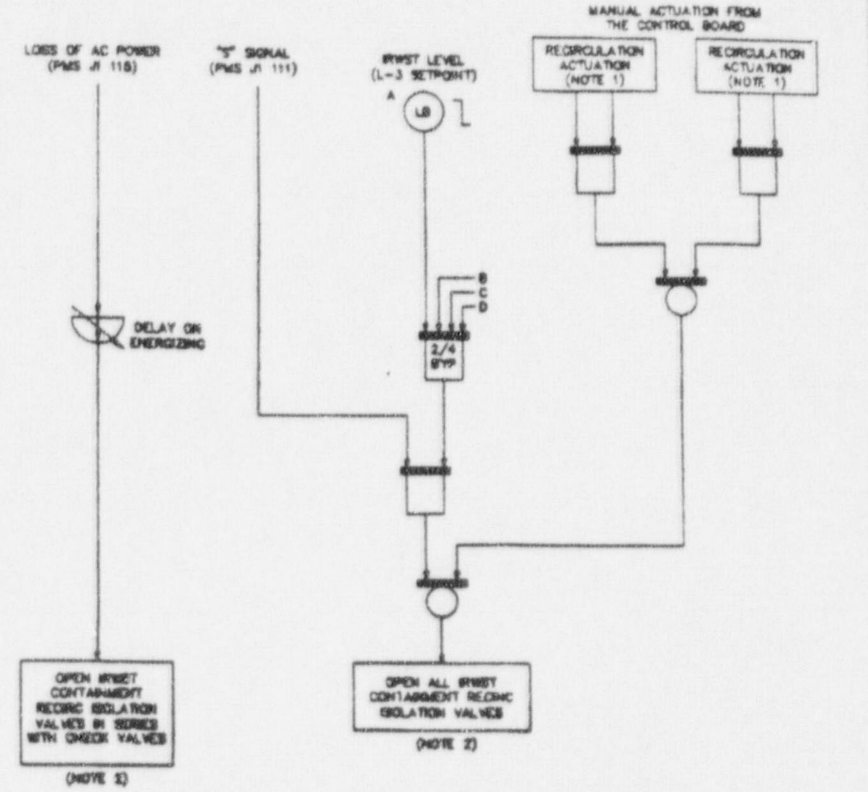
O'HARA 6-10-96

PMS J1 116



SSAR Figure 7.2-1
sheet 16

APR-300
30S210-01
1
APR-200
CON-07763
3
GW-GEE-380 REV 1
GW-GEE-384 REV 0
DELETED RESETS
IN ZONES C1,C8
ADDED NOTE 2 REF
REVISED ISO VALVE
REF IN ZONE B3
DEL ON ENERG WAS
0-30 MIN
RESTORED & REV
NOTE 2
DELETED NOTE 3
L-3 WAS L-8



SSAR
Figure 7.2-1 sheet 16

NOTES:

1. THE MANUAL ACTUATION CONSISTS OF FOUR MOMENTARY CONTROLS. IF TWO ASSOCIATED CONTROLS ARE OPERATED SIMULTANEOUSLY, ACTUATION WILL OCCUR IN ALL DIVISIONS.
2. COMPONENTS ARE ALL INDIVIDUALLY SEALED IN (LATCHED) SO THAT LOSS OF THE ACTUATION SIGNAL WILL NOT CAUSE THESE COMPONENTS TO RETURN TO THE CONDITION HELD PRIOR TO THE ABSENT OF THE ACTUATION SIGNAL.
3. CVS LETDOWN ISOLATION ALSO OCCURS DURING CONTAINMENT ISOLATION. SEE PMS J1 113.

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AP600 FUNCTIONAL DIAGRAMS
IRWST ACTUATIONS

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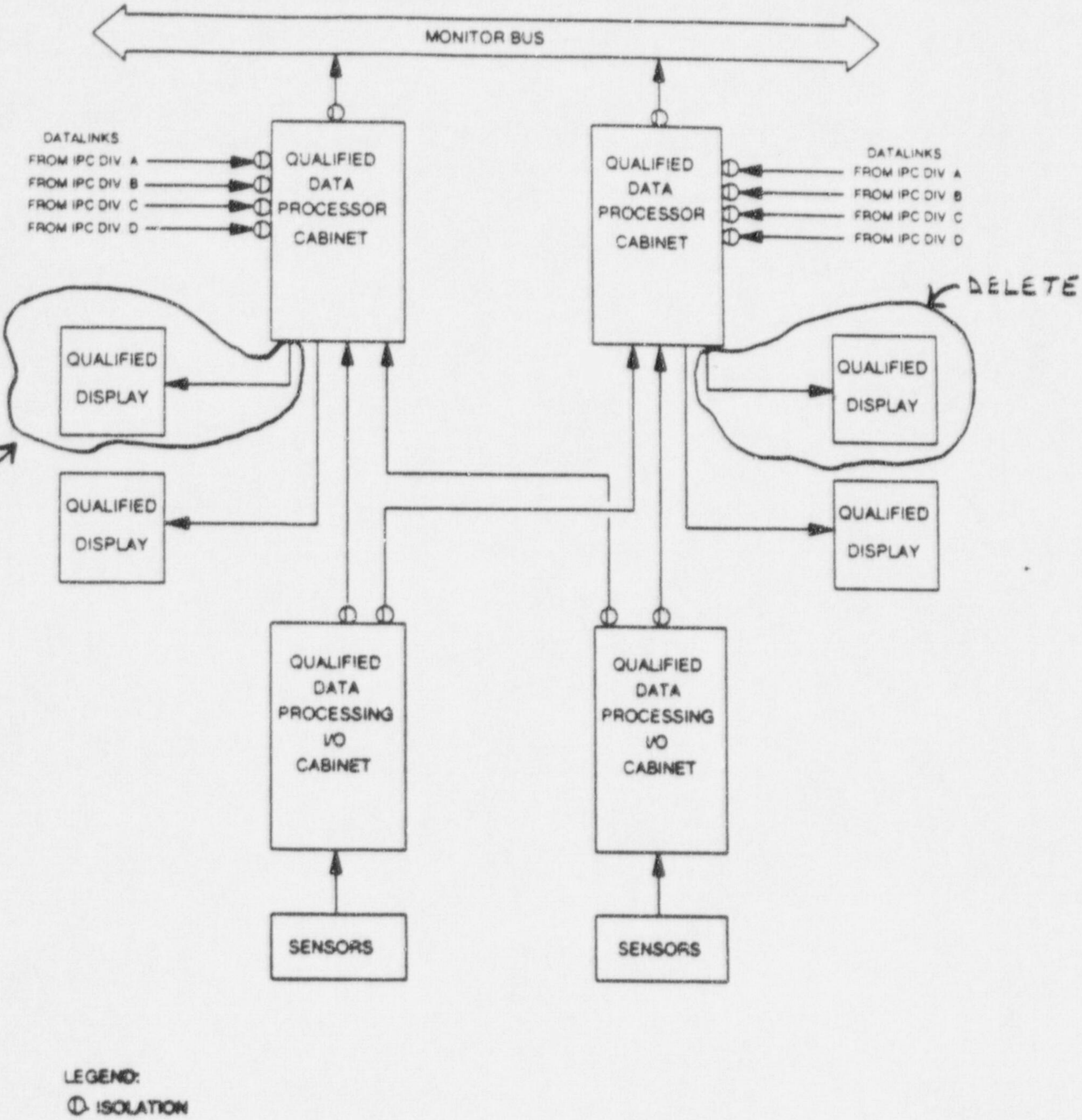
33

Changes Reflecting Resolution of NRC Minimum Inventory (Remote Shutdown Workstation) Comments



CHANGE 3 SSAR MARKUP, PAGE 3 OF 6

Figure 7.1-8

Qualified Data Processor

Condition 3 results from a coincidence of two of the four divisions of narrow range steam generator water level above the High-2 setpoint for either steam generator.

The functional logic relating to the tripping of the turbine is illustrated in Figure 7.2-1, sheet 14.

7.3.1.2.9 In-Containment Refueling Water Storage Tank Containment Recirculation

Signals to align the in-containment refueling water storage tank containment recirculation isolation valves are generated from the following conditions:

1. Automatic or manual safeguards actuation (subsection 7.3.1.1) in coincidence with low in-containment refueling water storage tank water level

2. Manual initiation

3. Extended loss of ac power sources

There are four parallel containment recirculation paths provided to permit the recirculation of the water provided by the in-containment refueling water storage tank. Two of these paths are provided with two isolation valves in series while the remaining two paths are provided with a single isolation valve in series with a check valve.

Conditions 1 and 2 result in the opening of all isolation valves in all four parallel paths. Condition 3 results in the opening of the two isolation valves that are in series with the check valves.

Condition 1 results from the coincidence of two of the four divisions of in-containment refueling water storage tank water level below the Low-3 setpoint, coincident with an automatic or manual safeguards actuation.

Condition 2 consists of two sets of two momentary controls. Manual actuation of both controls of either of the two control sets initiates recirculation in all four parallel paths. A two-control simultaneous actuation prevents inadvertent actuation.

Condition 3 results from the loss of all ac power for a period of time that approaches the 24-hour Class 1E dc battery capability to activate the in-containment refueling water storage tank containment recirculation isolation valves. The timed output holds on restoration of ac power and is manually reset after the batteries are recharged. The loss of all ac power is detected by undervoltage sensors that are connected to the input of each of the four Class 1E battery chargers. Two sensors are connected to each of the four battery charger inputs. The loss of ac power signal is based on the detection of an undervoltage condition by either of the two sensors connected to two of the four battery chargers.

[INSERT 7.3.1.2.9]

No interlocks or permissive signals apply directly to the activation of the in-containment refueling water storage tank containment recirculation isolation valves. However, automatic

[INSERT 7.3.1.2.9]

The safeguards actuation signal, which is part of condition 1, is latched-in upon its occurrence. A deliberate operator action is required to reset this latch. This feature is provided so that the actuation signal to the recirculation isolation valves is not cleared by the reset of the safeguards actuation signal as discussed in subsection 7.3.1.1.

Condition 1 results from a coincidence of two of the four divisions of reactor loop average temperature (T_{avg}) below the Low-2 setpoint coincident with the P-4 permissive (reactor trip). This blocks the opening of the steam dump valves. This signal also becomes an input to the steam dump interlock selector switch for unblocking the steam dump valves used for plant cooldown. This function may be manually blocked when the pressurizer pressure is below the P-11 setpoint. The block is automatically removed when the pressurizer pressure is above the P-11 setpoint.

Condition 2 consists of two controls. Either one of these controls can be used to manually initiate a steam dump block.

The functional logic relating to the steam dump block is illustrated in Figure 7.2-1, sheet 10.

7.3.1.2.17 Control Room Isolation and Air Supply Initiation

Signals to initiate isolation of the main control room and to initiate the air supply are generated from either of the following conditions:

1. High control room air supply radioactivity level
2. Loss of ac power sources
3. Manual initiation

Condition 1 is the occurrence one of two control room air supply radioactivity monitors detecting a radioactivity level above the High-2 setpoint.

Condition 2 results from the loss of all ac power sources. A preset time delay is provided to permit the restoration of ac power from the offsite sources or from the onsite diesel generators before initiation. The loss of all ac power is detected by undervoltage sensors that are connected to the input of each of the four Class 1E battery chargers. Two sensors are connected to each of the four battery charger inputs. The loss of ac power signal is based on the detection of an undervoltage condition by each of the two sensors connected to two of the four battery chargers. The two-out-of-four logic is based on an undervoltage to the battery chargers for divisions A or C coincident with an undervoltage to the battery chargers for divisions B or D.

[INSERT 7.3.1.2.17]

The functional logic relating to control room isolation and air supply initiation is illustrated in Figure 7.2-1, sheet 13.

7.3.1.2.18 Letdown Purification Line Isolation

A signal to isolate the letdown purification line is generated upon the coincidence of pressurizer level below the Low-1 setpoint in any two of four divisions. This helps to maintain reactor coolant system inventory. This function can be manually blocked when the pressurizer water level is below the P-12 setpoint. This function is automatically unblocked when the pressurizer water level is above the P-12 setpoint. The functional logic relating to this is illustrated in Figure 7.2-1, sheet 12.

[INSERT 7.3.1.2.17]

Condition 3 consists of two momentary controls. Manual actuation of either of the two controls will result in control room isolation and air supply initiation.

Table 7.3-1 (Sheet 7 of 8)

ENGINEERED SAFETY FEATURES ACTUATION SIGNALS

Actuation Signal	No. of Channels/ Switches	Actuation Logic	Permissives and Interlocks
16. Main Control Room Isolation and Air Supply Initiation (Figure 7.2-1, Sheet 13)			
a. High-2 control room supply air radiation	2	1/2	None
b. Undervoltage to Class 1E battery chargers	2/charger	2/2 per charger and 2/4 chargers[3]	None
c. Manual initiation	2 switches	1/2 switches	None
17. Purification Line Isolation (Figure 7.2-1, Sheet 12)			
a. Low-1 pressurizer level	4	2/4-BYP[1]	Manual block permitted below P-12. Automatically unblocked above P-12.
18. Containment Air Filtration System Isolation (Figure 7.2-1, Sheet 13)			
a. High-1 containment radioactivity	4	2/4-BYP[1]	None
19. Normal Residual Heat Removal System Isolation (Figure 7.2-1, Sheet 13)			
a. High-2 containment radioactivity	4	2/4-BYP[1]	None
20. Spent Fuel Pool Isolation (Figure 7.2-1, Sheet 13)			
a. Low spent fuel pool level	2	1/2	None
21. Open In-Containment Refueling Water Storage Tank (IRWST) Injection Line Valves (Figure 7.2-1, Sheet 16)			
a. Automatic reactor coolant system depressurization (fourth stage)		(See items 3d and 3e)	
b. Coincident loop 1 and loop 2 low hot leg level (after delay)	1 per loop	2/2	None
c. Manual initiation	4 switches	2/4 switches[3]	None
22. Open IRWST Containment Recirculation Valves In Series with Check Valves (Figure 7.2-1, Sheet 15)			
a. Extended undervoltage to Class 1E battery chargers	2/charger	1/2 per charger and 2/4 chargers	None

Table 7.3-3

SYSTEM-LEVEL MANUAL INPUT TO THE ENGINEERED SAFETY FEATURES ACTUATION SYSTEM

Manual Control	To Divisions	Figure 7.2-1 Sheet
Manual safeguards actuation #1	A B C D	2 & 11
Manual safeguards actuation #2	A B C D	2 & 11
Manual passive residual heat removal actuation #1	A B	8
Manual passive residual heat removal actuation #2	A B	8
Manual steam line isolation #1	B D	9
Manual steam line isolation #2	B D	9
Steam/feedwater isolation and safeguards block control #1	B	9
Steam/feedwater isolation and safeguards block control #2	D	9
Manual feedwater isolation #1	B D	10
Manual feedwater isolation #2	B D	10
Manual steam dump interlock selector #1	B	10
Manual steam dump interlock selector #2	D	10
Pressurizer pressure safeguards block control #1	A	11
Pressurizer pressure safeguards block control #2	B	11
Pressurizer pressure safeguards block control #3	C	11
Pressurizer pressure safeguards block control #4	D	11
Manual core makeup tank actuation #1	A B C D	12
Manual core makeup tank actuation #2	A B C D	12
Core makeup tank actuation block control #1	A	12
Core makeup tank actuation block control #2	B	12
Core makeup tank actuation block control #3	C	12
Core makeup tank actuation block control #4	D	12
Manual containment cooling actuation #1 & #2	A B	13
Manual containment cooling actuation #3 & #4	A B	13
Manual containment isolation actuation #1	A B C D	13
Manual containment isolation actuation #2	A B C D	13
Manual depressurization system stages 1, 2, and 3 actuation #1 & #2	A B C D	15
Manual depressurization system stages 1, 2, and 3 actuation #3 & #4	A B C D	15
Manual depressurization system stage 4 actuation #1 & #2	A B C D	15
Manual depressurization system stage 4 actuation #3 & #4	A B C D	15
Manual IRWST actuation #1 & #2	A B C D	16
Manual IRWST actuation #3 & #4	A B C D	16
Manual containment recirculation actuation #1 & #2	A B C D	16
Manual containment recirculation actuation #3 & #4	A B C D	16
Manual control room isolation and air supply initiation #1	ABCD	13
Manual control room isolation and air supply initiation #2	ABCD	13

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

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~~ components in the systems required for safe shutdown, using independent safety-related power divisions.

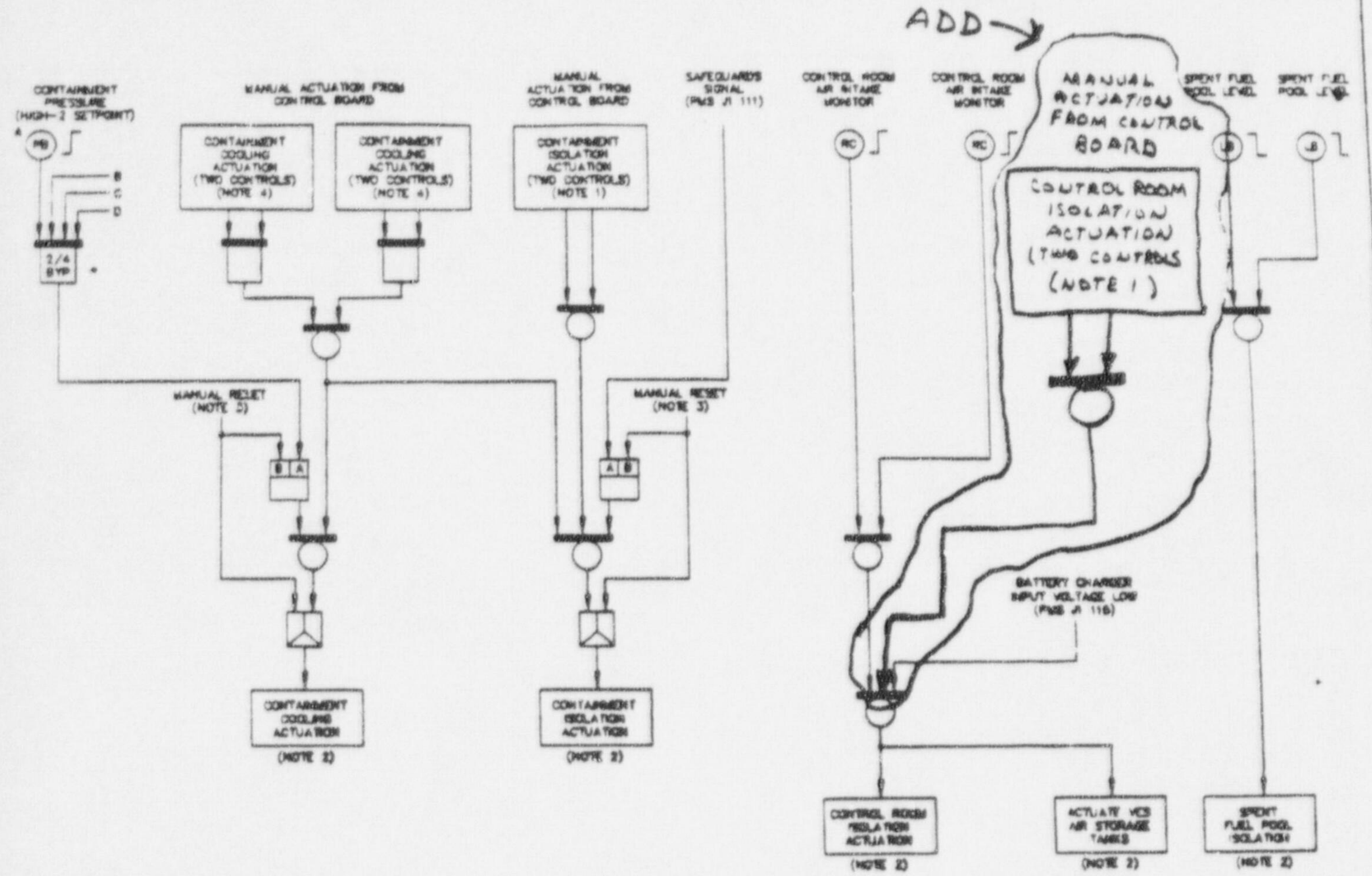
To prevent interaction between the redundant systems, the redundant control channels are wired independently and are separated from each other. Nonsafety-related circuits available for (but not required for) safe shutdown are electrically isolated from safety-related circuits.

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."



NOTES:

1. TWO MOMENTARY CONTROLS ON THE CONTROL BOARD. OPERATING EITHER CONTROL WILL ACTUATE ALL DIVISIONS.
2. COMPONENTS ARE ALL INDIVIDUALLY SEALED IN (LATCHED), SO THAT LOSS OF THE ACTUATION SIGNAL WILL NOT CAUSE THESE COMPONENTS TO RETURN TO THE CONDITION HELD PRIOR TO THE ADVENT OF THE ACTUATION SIGNAL.
3. SEPARATE MOMENTARY CONTROLS ON THE CONTROL BOARD, ONE FOR EACH DIVISION.
4. THE MANUAL CONTAINMENT COOLING ACTUATION CONSISTS OF FOUR MOMENTARY CONTROLS. IF TWO ASSOCIATED CONTROLS ARE OPERATED SIMULTANEOUSLY, ACTUATION WILL OCCUR IN ALL DIVISIONS.
5. TWO MOMENTARY CONTROLS ON THE CONTROL BOARD. OPERATING EITHER CONTROL WILL CAUSE THE REACTOR TRIP FUNCTION TO BE ACTUATED IN EACH OF THE FOUR DIVISIONS.

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AP600 FUNCTIONAL DIAGRAMS
CONTAINMENT PROTECTION AND
OTHER PROTECTION

DPMS J1 113 5

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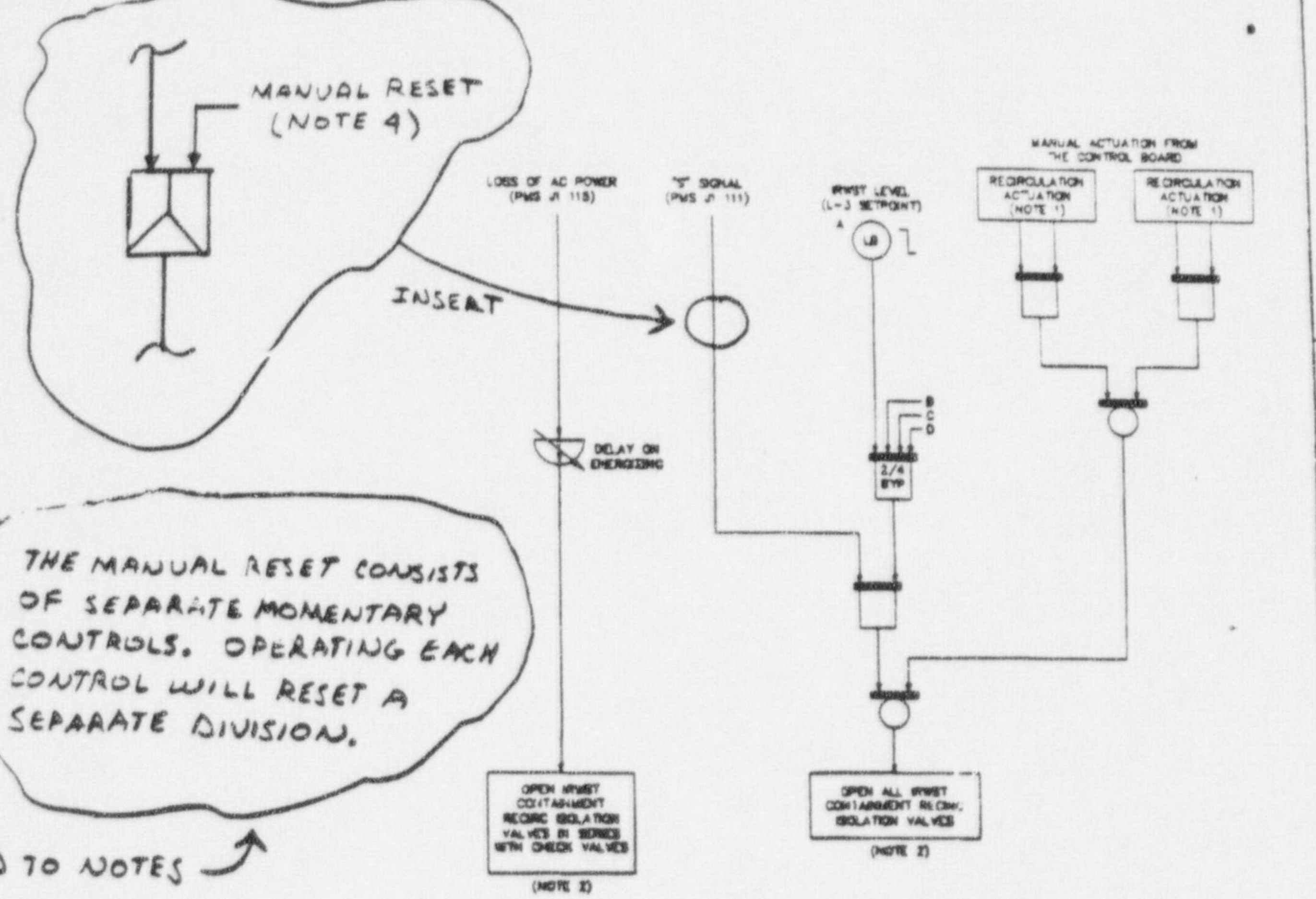
SSAR Figure 7.2-1, Sheet 13



4. THE MANUAL RESET CONSISTS OF SEPARATE MOMENTARY CONTROLS. OPERATING EACH CONTROL WILL RESET A SEPARATE DIVISION.

ADD TO NOTES

NOTES:

1. THE MANUAL ACTUATION CONSISTS OF FOUR MOMENTARY CONTROLS. IF TWO ASSOCIATED CONTROLS ARE OPERATED SIMULTANEOUSLY, ACTUATION WILL OCCUR IN ALL DIVISIONS.
2. COMPONENTS ARE ALL INDIVIDUALLY SEALED IN (LATCHED), SO THAT LOSS OF THE ACTUATION SIGNAL WILL NOT CAUSE THESE COMPONENTS TO RETURN TO THE CONDITION HELD PRIOR TO THE [illegible] OF THE ACTUATION SIGNAL.
3. ~~[illegible]~~

3. CVS LETDOWN ISOLATION ALSO OCCURS DURING CONTAINMENT ISOLATION. SEE PMS J1 113.

ADVANCED CAD TECHNOLOGY

Westinghouse Electric Corporation

AP600
ADVANCED PASSIVE LIGHT WATER REACTOR

R. O'HARA

Westinghouse Electric Corporation

5063525

AP600 FUNCTIONAL DIAGRAMS
IRWST ACTUATIONS

D PMS J1 116 3

288064

33

Changes Reflecting Resolution of NRC Post-72 Hour Position

The protection and safety monitoring system provides signal conditioning, communications, and display functions for Category 1 variables and for Category 2 variables that are energized from the Class 1E dc uninterruptible power supply system. The plant control system and the data display and processing system provides signal conditioning, communications and display functions for Category 3 variables and for Category 2 variables that are energized from the non-Class 1E dc uninterruptible power system. The data display and processing system also provides an alternate display of the variables which are displayed by the protection and safety monitoring system. Electrical separation of the data display and processing system and the protection and safety monitoring system is maintained through the use of isolation devices in the data links connecting the two systems, as discussed in subsection 7.1.2.11. The portion of the protection and safety monitoring system which is dedicated to providing the safety-related display function is referred to as the qualified data processing cabinets. These cabinets are discussed in subsection 7.1.2.6 and are illustrated in Figure 7.1-8.

The qualified data processing cabinets are divided into two separate electrical divisions. Each of the two electrical divisions is connected to a Class 1E dc uninterruptible power system with sufficient battery capacity to provide necessary electrical power for at least 72 hours. If all ac power sources are lost for a period of time that exceeds 72 hours, the power supply system will be energized from ac power sources which are brought to the site from other locations. See Section 8.3. the ancillary diesel generators or from

Instrumentation associated with primary variables that are energized from the Class 1E dc uninterruptible power supply system are powered from one of the two electrical divisions with 72 hour battery capacity. Instrumentation associated with other variables that are energized from the Class 1E dc uninterruptible power supply system are powered from one of four electrical divisions with 24 hour battery capacity. If a variable exists only to provide a backup to a primary variable, it may be powered by an electrical division with a 24 hour battery capacity. In such cases, provisions are provided to enable this variable to be powered by an alternate source if it is needed to resolve a discrepancy between two primary variables in the event that all ac power sources are lost for a period in excess of 24 hours.

Class 1E position indication signals for valves and electrical breakers may be powered by an electrical division with 24 hour battery capacity. This is necessary to make full use of all four Class 1E electrical divisions to enhance fire separation criteria. The power associated with the actuation signal for each of these valves or electrical breakers is provided by an electrical division with 24 hour battery capacity, so there is no need to provide position indication beyond this period. The operator will verify that the valves or electrical breakers have achieved the proper position for long-term stable plant operation before position indication is lost. Once the position indication is lost, there is no need for further monitoring since the operator does not have any remote capability for changing the position of these components.

Electrically operated valves, which have the electrical power removed to meet the single failure criterion, are provided with redundant valve position sensors. Each of the two position sensors is powered from a different non-Class 1E power source.

7.3.1.2.19 Containment Air Filtration System Isolation

A signal to isolate the containment air filtration system is generated upon the coincidence of containment radioactivity above the High-1 setpoint in any two of four divisions. This limits activity release to the environment. The functional logic relating to this is illustrated in Figure 7.2-1, sheet 13.

7.3.1.2.20 Normal Residual Heat Removal System Isolation

A signal for isolating the normal residual heat removal system lines is generated upon the coincidence of containment radioactivity above the High-2 setpoint in any two of four divisions. This signal also isolates the chemical and volume control system as discussed in subsection 7.3.1.2.15. This limits activity release to the environment. The functional logic relating to this is illustrated in Figure 7.2-1, sheet 13.

7.3.1.2.21 Spent Fuel Pool Isolation

A signal for isolating the spent fuel pool lines is generated upon the coincidence of spent fuel pool level below the Low setpoint in ~~any one~~ two of ~~two~~ three divisions. This helps to maintain the water inventory in the spent fuel pool due to line leakage. The functional logic relating to this is illustrated in Figure 7.2-1, sheet 13.

7.3.1.3 Blocks, Permissives, and Interlocks for Engineered Safety Features Actuation

The interlocks used for engineered safety features actuation are designated as "P-xx" permissives and are listed in Table 7.3-2.

7.3.1.4 Bypasses of Engineered Safety Features Actuations

The channels used in engineered safety features actuation that can be manually bypassed are indicated in Table 7.3-1. A description of this bypass capability is provided in subsection 7.1.2.10. The actuation logic is not bypassed for test. During tests, the actuation logic is fully tested by blocking the actuation logic output before it results in component actuations.

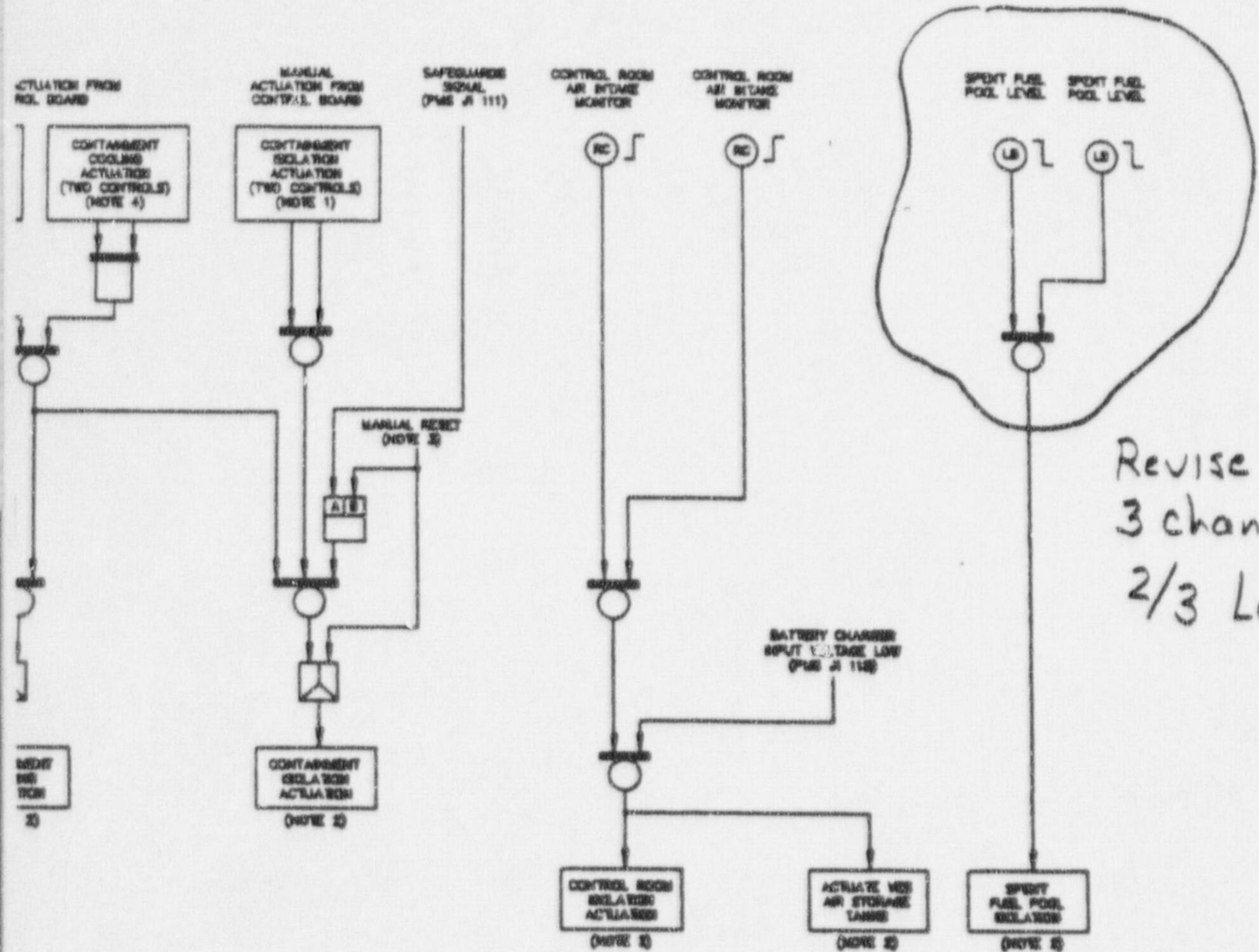
7.3.1.5 Design Basis for Engineered Safety Features Actuation

The following subsections provide the design bases information for engineered safety features actuation, including the information required by Section 3 of IEEE 279-1971. Engineered safety features are initiated by the protection and safety monitoring system. Those design bases relating to the equipment that initiates and accomplishes engineered safety features are given in subsection 7.1.4.1. The design bases presented here concern the variables monitored for engineered safety features actuation and the minimum performance requirements in generating the actuation signals.

Table 7.3-1 (Sheet 7 of 8)

ENGINEERED SAFETY FEATURES ACTUATION SIGNALS

Actuation Signal	No. of Channels/ Switches	Actuation Logic	Permissives and Interlocks
16. Main Control Room Isolation and Air Supply Initiation (Figure 7.2-1, Sheet 13)			
a. High-2 control room supply air radiation	2	1/2	None
b. Undervoltage to Class 1E battery chargers	2/charger	2/2 per charger and 2/4 chargers[5]	None
17. Purification Line Isolation (Figure 7.2-1, Sheet 12)			
a. Low-1 pressurizer level	4	2/4-BYP[1]	Manual block permitted below P-12. Automatically unblocked above P-12.
18. Containment Air Filtration System Isolation (Figure 7.2-1, Sheet 13)			
a. High-1 containment radioactivity	4	2/4-BYP[1]	None
19. Normal Residual Heat Removal System Isolation (Figure 7.2-1, Sheet 13)			
a. High-2 containment radioactivity	4	2/4-BYP[1]	None
20. Spent Fuel Pool Isolation (Figure 7.2-1, Sheet 13)			
a. Low spent fuel pool level	~~2~~ 3	~~1/2~~ 2/3	None
21. Open In-Containment Refueling Water Storage Tank (IRWST) Injection Line Valves (Figure 7.2-1, Sheet 16)			
a. Automatic reactor coolant system depressurization (fourth stage)		(See items 3d and 3e)	
b. Coincident loop 1 and loop 2 low hot leg level (after delay)	1 per loop	2/2	None
c. Manual initiation	4 switches	2/4 switches[3]	None
22. Open IRWST Containment Recirculation Valves In Series with Check Valves (Figure 7.2-1, Sheet 15)			
a. Extended undervoltage to Class 1E battery chargers	2/charger	1/2 per charger and 2/4 chargers	None



Revise to show
3 channels and
2/3 Logic

Figure 7.2-1 (Sheet 13 of 20)

Functional Diagrams
Containment & Other Protection

Table 7.5-1 (Sheet 7 of 12)

Post-Accident Monitoring System

Variable	Range/ Status	Type/ Category	Qualification		Number of Instruments Required	Power Supply	QDPS Indication (Note 2)	Remarks
			Environmental	Seismic				
Chilled water pump status	On/Off	F3	None	None	1/pump	Non-1E	No	
Chilled water valve status	Open/ Closed	F3	None	None	1/valve	Non-1E	No	
Spent fuel pool pump flow	0-1000 gpm	F3	None	None	1/pump	Non-1E	No	
Spent fuel pool temperature	50-250°F	D2, F3	Mild	None	1	Non-1E	No	
Spent fuel pool water level	0-100%	D2, F3	Mild	Yes	~~2~~ 3 (Note 4)	1E	Yes	
CMT to reactor vessel valve status	Open/ Closed	D2	Harsh	No	1/valve	Non-1E	No	
CMT inlet isolation valve status	Open/ Closed	D2	Harsh	Yes	1/valve (Note 7)	1E	Yes	
CMT level	0-100%	D2, F2	Harsh	Yes	1/tank	1E	Yes	
IRWST to reactor vessel valve status (Non-MOV)	Open/ Closed	D2	Harsh	None	1/valve	Non-1E	No	
IRWST to reactor vessel valve status (MOV)	Open/ Closed	D3	None	None	1/valve	Non-1E	No	
ADS: first, second and third stage valve status	Open/ Closed	D2	Harsh	Yes	1/valve (Note 7)	1E	Yes	

Table 7.5-1 (Sheet 12 of 12)

Post-Accident Monitoring System

Variable	Range/ Status	Type/ Category	Qualification		Number of Instruments Required	Power Supply	QDPS Indication (Note 2)	Remarks
			Environmental	Seismic				
Post accident sampling station radiation	10^{-1}- 10^{7} mR/hr	E3	None	None	1	Non-1E	No	
Main steam line radiation level	10^{-1}- 10^{3} μCi/cc	C2, E2	Mild	None	1/line	Non-1E	No	
Technical support center radiation	10^{-1}- 10^{4} mR/hr	E3	None	None	1	Non-1E	No	
Meteorological parameters	N/A	E3	None	None	N/A	Non-1E	No	Site specific

Notes:

1. Total flow measurement is obtained from the sum of ~~three~~ four branch flow devices.
2. The same information is available in the technical support center via the monitor bus. Information available on the qualified data processing system is also available at the remote shutdown workstation.
3. Noble gas: 10^{-7} to 10^{5} μCi/cc
   Particulates: 10^{-13} to 10^{-7} μCi/cc
   Iodines: 10^{-12} to 10^{-6} μCi/cc
4. The number of instruments required after stable plant conditions is two. A third channel is available through temporary connections to resolve information ambiguity if necessary (See subsection 7.5.4).
5. Noble gas: 10^{-7} to 10^{-2} μCi/cc
   Particulates: 10^{-12} to 10^{-7} μCi/cc
   Iodines: 10^{-11} to 10^{-5} μCi/cc
6. Degree of subcooling is calculated from pressurizer pressure and RCS hot leg temperature.
7. This instrument is not required after 24 hours.

DSER Open Item Tracking System Report for Chapter 7
Items not Statused "Resolved" by NRC

AP600 Open Item Tracking System Database: Executive Summary

Date: 2/18/97

Selection: [nrc st code]<>'Resolved' And [DSER Section] like '7*' Sorted by Item #

Item No	Branch	DSER Section/ Question	Type	Title/Description Detail Status	Resp Engineer	(W) Status	NRC Status	Letter No. /	Date
1038	NRR/HICB	7.1.4-1	DSER-OI		ITAAC/Deutsch, K	Action N	Action N	NSD-NRC-96-4737	

Westinghouse should describe in the SSAR, CDM, and ITAAC the digital system design process.
Westinghouse should provide a detailed descritpiton of the digital system design process in the SSAR and CDM with a corresponding ITAAC.

Action W - WCAP-13383, which describes the digital system design process is being updated. The certified design material and ITAACs will be modified. The SSAR has been modified to reference the design process and to indicate the software design standards the design process conforms to. This information is provided in Revision 3 of the SSAR, Subsection 7.1.2.15. The WCAP and ITAAC revisions must be completed before this item can be closed out. NRC has requested a presentation when all elements are completed. WCAP-13383 rev due 5/30/96 rkn 5/7/96

WCAP-13383 in repro 6/14 for 6/17 release. rkn 6/14/96
Closed - Response provided by NSD-NRC-96-4737.

Per an 11/21 W/NRC telecon, the NRC thinks the I&C ITAAC is deficient and requested that we "fix" the ITAAC or justify/explain deviations from the SRP 14.3.5 to NRC satisfaction. NRC to provide specific comments on the ITAAC. rkn 12/2

Item No	Branch	DSER Section/ Question	Type	Title/Description Detail Status	Resp Engineer	(W) Status	NRC Status	Letter No. /	Date
1039	NRR/HICB	7.1.7-1	DSER-OI		ITAAC/Deutsch	Action N	Action N	NSD-NRC-96-4737	

Westinghouse should describe a commercial grade item dedication program for digital systems.
Westinghouse has not addressed the commercial grade item dedication program that is necessary to ensure sufficient quality in the design of safety-related and nonsafety-related I&C systems using commercial of-the-shelf equipment. The design, verification, and validation process for COTS software and hardware should be clearly documented for design certification.

Action W - WCAP-13383 is being updated to include a commercial grade item dedication process. The SSAR has been modified to reference this process. This information is provided in Revision 3 of the SSAR, Subsection 7.1.2.15. The WCAP revision must be completed before this item can be closed out.

WCAP in repro 6/14 for 6/17 issuance. rkn 6/14
Closed - Response provided by NSD-NRC-96-4737.

Same as item 1038. rkn 12/2

Item No	Branch	DSER Section/ Question	Type	Title/Description Detail Status	Resp Engineer	(W) Status	NRC Status	Letter No. /	Date
1041	NRR/HICB	7.2.6-1	DSER-OI		ITAAC/Deutsch, K	Action N	Action N		

The staff has not yet completed its evaluation of the software architecture design.
because WCAP 14080 was submitted in July 1994, the staff has not completed its review of the document and is continuing its evaluation of the software architecture based on both the proposed design and the associated design process. The results from this evaluation will be presented in the final SER for AP600.

Closed - Westinghouse has completed necessary submittals to support staff review.

Per 11/21 W/NRC telecon, when the NRC agrees with the design process through their review of the ITAACs, this item will be closed. rkn 12/2

Item No	Branch	DSER Section/ Question	Type	Title/Description Detail Status	Resp Engineer	(W) Status	NRC Status	Letter No. /	Date
1043	NRR/HICB	7.2.8-1	DSER-OI		ITAAC/Deutsch	Action N	Action N	NTD-NRC-95-4464	

Westinghouse should provide a discussion concerning the qualification of digital equipment to the electromagnetic environment.
Westinghouse has not addressed the issue of electromagnetic environmental qualification and has not committed to the appropriate standards.

Closed - List of standards reviewed by NRC during meeting on May 15-16. Standards incorporated into Revision 3 of the SSAR, Subsection 7.1.4.1.6.

Per an 11/21 W/NRC telecon, the technical issues are resolved. When NRC agrees with design process thru ITAAC review, this item will be closed.

AP600 Open Item Tracking System Database: Executive Summary

Date: 2/18/97

Selection: [nrc st code]<>'Resolved' And [DSER Section] like '7*' Sorted by Item #

Item No	Branch	DSER Section/ Question	Type	Title/Description Detail Status	Resp Engineer	(W) Status	NRC Status	Letter No /	Date
1044	NRR/HICB	7 2 8-2	DSER-OI		ITAAC/Deutsch, K.	Closed	Action W	NTD-NRC-95-4464	

Westinghouse should provide information concerning environmental qualification of PMS components addressing local temperature rises above the room ambient experienced by the components during operation.
It is desirable to have additional margin built into the design. The components should, therefore, be qualified by testing to higher temperatures than specified in the SSAR for a given room environment. Westinghouse should address this concern in the SSAR. Westinghouse should also provide mild environment equipment qualification in the CDM with the corresponding ITAAC.

Closed - Technical information agreed to by NRC during meeting on May 15-16. Additional technical information regarding the equipment design margin to loss of HVAC has been incorporated into Revision 3 of the SSAR, Subsection 7.1.4.1.8. rkn 12/2
Westinghouse needs to decide approach to close this item. rkn 12/6

Action N - NRC still has the action to evaluate the Westinghouse proposal on procedural fix of instrument overheating after 24 hour period. (6/21 meeting with W/SPLB/HICB). Based on 11/21 W/NRC telecon, this approach is reasonable, see qualification program in SSAR Section 3.11.

Action W - NRC requested W provide proposed COL item for qualification margin and instrument setpoint data or document in the CDM and corresponding ITAAC (W is considering options, did not commit to either approach). rkn 12/2

Westinghouse does not consider there to be an applicable COL action to identify. Technical information related to design margin against a loss of HVAC was provided in SSAR 7.1.4.1.6 and is considered technically resolved, as waws previously agreed to by NRC. This item is considered closed since there is no Westinghouse action required at this time to address this item (since the NRC relates this comment to the PMS ITAAC, the responsible engineer is changed to ITAAC). rkn 1/14/97
Action W - (from NRC on 1/28/97) Submit revised CMD & ITAAC to include COL action to include additional design margin to accomodate a loss of the normal HVAC. Provide an alarm if internal cabinet temperatures reach an excessive value. jww1/28

This was previously closed and is still considered closed, meaning there is no Westinghouse action identified or required o close this item, all necessary submittals have been made. For background, SSAR Section 7.1.4.1.6 was revised in Feb 1996 to address this. Specifically, there is a sentence which reads, "The cabinets containing the digital equipment are provided with temperature sensors which provide an alarm if internal cabinet temperatures reach an excessive value." This is closed. rkn 1/30/97

Per telecon with Hulbert Li today, the action is for Westinghouse to include this alarm in the ITAAC. rkn 2/18/97

Item No	Branch	DSER Section/ Question	Type	Title/Description Detail Status	Resp Engineer	(W) Status	NRC Status	Letter No /	Date
1049	NRR/HICB	7 5 8-1	DSER-OI		ITAAC/Lindgren/Deuts	Action N	Action N		

Westinghouse should describe the design features of the incore instrumentation system.
In its response to Q492.5 dated July 25, 1994, Westinghouse states that information on the employment of fixed incore detectors in conjunction with an online power distribution monitoring system will be provided to the NRC to support the final SER.

Closed - The technical information was accepted by the I&C Branch of NRC during the meeting on May 15-16. This technical information has been incorporated into Revision 3 of the SSAR, Subsection 4.4.6.1.
Open for ITAAC based on fax from NRC 1/21/97. rkn

For Chapter 7 this item is resolved. (NRC/RSB to communicate any concerns with qualification of thermocouples and instrument coolant capability outside the scope of Chapter 7). rkn 12/2

Selection: [nrc st code]<>'Resolved' And [DSER Section] like '7*' Sorted by Item #

Item No	Branch	DSER Section/ Question	Type	Title/Description Detail Status	Resp Engineer	(W) Status	NRC Status	Letter No / Date
1052	NRR/HICB	7.6.2-1	DSER-OI	Westinghouse should provide additional design details of the accumulator isolation valve interlocks important to safety to confirm that the design meets the relevant requirements of the SRP, including IEEE 279. Closed - Additional technical information has been incorporated into Revision 3 of the SSAR, Subsection 7.6.2.1. Figure 7.2-1 was also modified to include additional technical detail! Action NRC - Per 11/21 telecon, NRC to review technical information already provided since this operator is nonsafety, not important to safety, has separate power, positive 3 position indications, and power removed at-power (consistent with Tech Specs) and limit switch alarms. rkn 12/2 Technical information provided. NRC to advise to resolution status. rkn 1/14/97 Per fax, NRC considers this open for interlocks concern (FSER open item 7.6.2-1). rkn 1/21/97	Schulz, T.	Closed	Action N	
1053	NRR/HICB	7.6.3-1	DSER-OI	Westinghouse should provide additional design details of the IRWST discharge valve interlocks important to safety to confirm that the design meets the relevant requirements of the SRP, including IEEE 279. Closed - Additional technical information has been incorporated into Revision 3 of the SSAR, Subsection 7.6.2.2. Figure 7.2-1 was also modified to include additional technical detail. Action NRC - See 1052. rkn 12/2 Technical information has been provided. NRC to advise regarding resolution status. rkn 1/14/97 Per fax, NRC considers this open for interlocks concern (FSER open item 7.6.2-1). rkn 1/21/97	Schulz, T.	Closed	Action N	
1055	NRR/HICB	7.7.2-1	DSER-OI	Westinghouse should provide additional information concerning the design of the DAS. Closed - Technical information accepted by NRC during meeting on May 15-16. This additional technical detail has been incorporated into Revision 3 of the SSAR, Subsection 7.7.1.11. NRC action to review ITAAC. Per 11/21 telecon, this item is now subject to DAS ITAAC comment resolution/completion. rkn 12/2	ITAAC/Delose,Frank	Action N	Action N	NTD-NRC-95-4464
2023	NRR/HICB	7	DSER-OI50	27. No Commitment to Industry Standards for Digital Systems While the SSAR references IEEE standards 279, 384, 603 and 796 for the design of AP600 I&C systems, the staff is concerned that there is no reference to digital microprocessor-related standards. Specifically they are concerned about the lack of standards related to multiplexer architecture, communications protocols, and hardware/software design. The staff wants Westinghouse to make an explicit commitment to industry hardware and software related standards. No detailed documentation of the process and no phased ITAAC for verification of the design. Action W - Item 1037 closes all but final sentence of item. Remaining action to address "No detailed documentation of the process and no phased ITAAC for verification of the design". SSAR Ch 7.1 commits to a V&V program, meeting Standards, etc., such that NRC expectations are met. When the ITAAC for PMS is complete, this item will be closed. rkn 5/7/96 Closed - ITAAC submitted by NSD-NRC-96-4875 of 11/7/96. Per 11/21 telecon for DSER Ch 7, NRC wants to discuss ITAAC approach with Westinghouse.	ITAAC/Deutsch	Action N	Action N	NSD-NRC-4875

AP600 Open Item Tracking System Database: Executive Summary

Date: 2/18/97

Selection: [nrc st code]<>'Resolved' And [DSER Section] like '7*' Sorted by Item #

Item No	Branch	DSER Section/ Question	Type	Title/Description Detail Status	Resp Engineer	(W) Status	NRC Status	Letter No. /	Date
2025	NRR/HICB	7	DSER-OI50	29. Environmental Qualification of DAS Equipment and Sensors The DSER indicates that the DAS equipment must be designed and qualified to the environment in which it needs to perform. The Westinghouse position is that the DAS equipment will be designed to function the environment in which it needs to perform. However, the DAS equipment will not be subjected to a full-blown 10 CFR 50.49 / IEEE 323 qualification program. Closed - SSAR Chapter 7 section 7.7.1.11 revised to address Per an 11/21 telecon, NRC thinks the DAS sensors and actuated devices (e.g., PRHR solenoid valve) should be qualified to a higher (PMS) standard but Westinghouse does not agree. By 12/6 fax, W proposed SSAR change to clarify qualification, NRC to review approach. rkn 12/6 Completed in SSAR Rev 10. rkn 1/14/97 Whoops! I checked and it didn't get into SSAR Rev 10. It WILL get into Rev 11. See NSD-NRC-97-4947. rkn 1/30/97	SSARREV/Miller	Confirm-W	Action N		
2272	NRR/SRXB	7.6.2	MTG-OI	APRIL 19, 1995 (HSII) DISCUSSION ITEMS 15. Availability of Safeguards - Interlocks (SSAR Section 7.6.2): Section 7.6.2 of the SSAR discusses the interlock systems to verify the availability of safeguard functions, i.e., to ensure opening of the isolation valves of the accumulators, IRWST, and PRHRHXs. These valves are motor-operated, normally open valves, and are controlled from the main control room and remote shutdown work station. a. SSAR Section 7.6.2 states that, as a result of the confirmatory safeguard open signal (which will automatically open the isolation valves, overriding bypass features to allow the isolation valves to be closed), isolation of an accumulator with the RCS at pressure (or isolation of the IRWST gravity injection line when the tank is required to be operable, or isolation of the PRHRHX inlet line when the PRHRHX is required to be operable) is acceptable. What are the design reliability of these interlocks to ensure these isolation valves will be open upon the confirmatory safeguard open signals? Is this practice acceptable for current operating reactor to allow accumulator isolated at pressure? Closed - At he Reactor System Branch Meeting on 4/25/95, Westinghouse referred to the use of identical interlocks on the accumulators and IRWST as those currently used on the accumulators at current plants (power locked-out). CMTs and PRHR interlocks are not power locked out but instead redundant controllers are provided for each valve along with three-way redundant valve positions. Westinghouse also referred to the Revision 2 SSAR 6.3 for the design details. Revision 3 of the SSAR, Section 7.6 includes the interlock information. Based on 11/21 telecon, NRC doesn't think the SSAR Section 7.6 is sufficient and has been asked to provide specific comments. rkn 12/6 Closed since there is no Westinghouse action at this time. NRC to advise regarding status. rkn 1/14/97	Deutsch	Closed	Action N		
2273	NRR/SRXB	7.6.2	MTG-OI	APRIL 19, 1995 (HSII) DISCUSSION ITEMS 15. Availability of Safeguards - Interlocks (SSAR Section 7.6.2): b. SSAR Section 7.6.2 also states that the maximum permissible time that an accumulator valve (or IRWST discharge valve, or PRHRHX inlet valve, respectively) is closed when the reactor is at pressure as specified in the TS. Where are they specified? Action W - Section 3.5.1 of the Tech Specs specifies the maximum permissble valve times. The revised Tech Specs will be submitted June 1996, at which time this item can be closed. Closed - With issuance of the Tech Specs in SSAR Rev. 9. Action NRC - Per 11/21 telecon, NRC to review Tech Specs to ensure this is resolved/closed. rkn 12/4 Westinghouse action is complete for this item. NRC to advise on status. rkn 1/14/97	TECHSPEC/Schulz	Closed	Action N		

AP600 Open Item Tracking System Database: Executive Summary

Date: 2/18/97

Selection: [nrc st code]<>'Resolved' And [DSER Section] like '7*' Sorted by Item #

Item No	Branch	DSER Section/ Question	Type	Title/Description Detail Status	Resp Engineer	(W) Status	NRC Status	Letter No /	Date
4257	NRR/HICB	7	MTG-COM	Add WCAP-14080 as reference for SSAR Section 7.1 Westinghouse has confirmed this is an appropriate reference and transmitted the SSAR markups to NRC. This item is opened to ensure the marked changes are included in the next SSAR rev. rkn 1/15/97 Refer to NSD-NRC-97-4947 for changes to be included in SSAR Rev 11. rkn 1/31/97	SSARREV/Deutsch, Ke	Confrm-W	Confrm-W		

FAX to DINO SCALETTI

February 20, 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 will send 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 sixth weekly request of this type.

5, 21, 142, 157, 164, 172, 173, 177, 182, 184, 262, 300, 305, 308, 333, 405, 457, 458, 628, 681, 698, 706, 710, 716, 717, 718, 719, 722, 724, 729, 730, 731, 732, 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, 1354, 1356, 1360, 1361, 1365, 1366, 1367, 1368, 1369, 1370, 1371, 1392, 1396, 1458, 1461, 1697, 1698, 1699, 1700, 1701, 1702, 1703, 1704, 1707, 1716, 1727, 1730, 1731, 1742, 1745, 1747, 1749, 1753, 1760, 1809, 1810, 1811, 1885, 1888, 1996, 1999, 2018, 2019, 2023, 2024, 2025, 2034, 2040, 2043, 2044, 2045, 2051, 2199, 2200, 2201, 2202, 2272, 2273, 2347, 2348, 2349, 2442, 2457, 2515, 2676, 2683, 2684, 2686, 2691, 2698, 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, 3057, 3098, 3122, 3126, 3127, 3128, 3197, 3247, 3264, 3265, 3266, 3267, 3268, 3269, 3270, 3271, 3372, 3398, 3399, 3400, 3401, 3402, 3427, 3439, 3468, 3469, 3470, 3471, 3472, 3473, 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, 4144, 4151, 4224, 4225, 4226, and 4227.

Jim

Thanks
Jim Winters
412-374-5290

FAX to DINO SCALETTI

February 20, 1997

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

Don Lindgren
Richard Orr
Ed Cummins
Bob Vijuk
Brian McIntyre

OPEN ITEMS FOR SSAR SECTION 3.8.4

This is a background package for the remaining open items for SSAR section 3.8.4. SSAR section 3.8.4 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 8 Open Items with NRC Status of Action W. Four (4) of these items (745, 750, 751 and 755) still require some Westinghouse action. Westinghouse believes the other four (4) items were addressed in or prior to the January 16, 1997 meeting with NRC. We are still waiting for the NRC letter report on the January 16, 1997 meeting. Currently, our records show no additional outstanding Westinghouse action required for section 3.8.4, except items 745, 750, 751 and 755, 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

1 of 3

AP600 Open Item Tracking System Database: Executive Summary

Date: 2/20/97

Selection: [nrc st code]='Action W' And [DSER Section] like '3.8.4*' Sorted by Item #

Item No	Branch	DSER Section/ Question	Type	Title/Description Detail Status	Resp Engineer	(W) Status	NRC Status	Letter No. /	Date
740	NRR/ECGB	3.8.4.1-3	DSER-OI	Westinghouse should provide in the SSAR a description and design details of modules located in the auxiliary building, and should indicate the difference between these modules and those located inside the containment. Additional information as provided in RAI response has been incorporated in SSAR Rev 3 for the structural modules in the fuel handling area and for the finned floors. Further discussion will be added in SSAR Revision once module behavior study is completed (see DSER OIs on section 3.8.3). Closed in meeting with NRC 1/16/97 - minor SSAR change shown in draft revision	Orr / NRCSM	Closed	Action W		
745	NRR/ECGB	3.8.4.3-1	DSER-OI	Westinghouse should acceptably address the issues regarding the consideration of live load in the seismic model. SSAR subsection 3.8.4.3.2.3 has been added and 3.7.2.3.1 has been revised. Meeting 6/15/95 - NRC will provide copy of staff position on combination of live load with SSE. Westinghouse should address this position. NRC position provided in letter of July 18, 1996. NRC Status : Action W - The staff reviewed the Westinghouse letter response with draft markup. The response is unacceptable. (1) Westinghouse does not include live load in the global seismic model. (West has calc. to show insignificance - SSAR revision is needed) (2) For local effects, the Westinghouse use of 25% of live load for SSE cases (rather than 100% in all cases) is unacceptable. (3) For global effects, Westinghouse does not include snow load as defined in the staff position (NRC letter dated July 18, 1996) - SSAR commitment is needed. (12/16/96)	Orr / Prasad	Action W	Action W	NTD-NRC-95-4464	
750	NRR/ECGB	3.8.4.4-2	DSER-OI	Westinghouse should provide for staff review the final design calculation for the shield building and the passive containment cooling water storage tank. Methodology was presented to NRC in meeting on March 2. The final design calculation for the shield building and the passive containment cooling water storage tank were reviewed during the meeting in June 1995. Westinghouse should address comments identified in meeting notes. Comments from meeting have been addressed and were discussed during meeting on March 7, 1996. Closed: remaining issues are tracked under new RAI 230.100 transmitted by letter of April 5, 1996. NRC to review design calculations during meeting in December. NRC Status: Action W - Westinghouse will provide the design of the ring beam including torsional moment and axial tension. (12/16/96)	Orr/ANSALDO	Action W	Action W		
751	NRR/ECGB	3.8.4.4-3	DSER-OI	Westinghouse should add COL Action Item 3.8.4.4-1 to the SSAR. Design calculations for shield building roof were reviewed by NRC staff during meeting in June. Tank deformations during tank filling are small and monitoring of tank deflections and comparison against predictions is not meaningful. NRC Status: Action W - Post-construction testing is necessary to confirm adequacy of the PCS tank. (12/16/96)	Orr/ANSALDO	Action W	Action W		
754	NRR/ECGB	3.8.4.4-6	DSER-OI	Westinghouse should provide analysis procedures and design details of the spent fuel pool, including fuel racks, the fuel transfer canal, and the new fuel storage area. The spent fuel pool and transfer canal are part of module M20/M21. Additional details will be provided in response to OI 3.8.4.5-1. Analysis and design details of the fuel racks are covered in SSAR Chapter 9. NRC to review design calculations during meeting in January, 1997. SSAR 3.8.4, Rev 11 includes reference to Section 9.1 for the new and spent fuel racks. SSAR subsection 9.1.6 requires the COL applicant to perform confirmatory analyses. These include reconciliation of loads on the structures.	Orr / NRCSM	Closed	Action W		

2 of 3

AP600 Open Item Tracking System Database: Executive Summary

Date: 2/20/97

Selection: [nrc st code]='Action W' And [DSER Section] like '3.8.4*' Sorted by Item #

Item No.	Branch	DSER Section/ Question	Type	Title/Description Detail Status	Resp Engineer	(W) Status	NRC Status	Letter No. /	Date
755	NRR/ECGB	3.8.4.4-7	DSER-OI		Orr / McDermott	Action W	Action W		
				Westinghouse's list of components provided in the June 30, 1994 response to Q220.83 should include both the IRWST (as part of containment internal structures), and the air baffle (as part of the shield building).					
				The description of the open item is misleading. This open item covers the design reports for the nuclear island basemat, auxiliary building, containment internal structure and the shield building. Design reports for the nuclear island basemat, auxiliary building, containment internal structure and the shield building will be available in meetings in December, 1996 and January 1997. NRC Status: Action W - The staff reviewed the summary reports shield building roof structure, basemat, and auxiliary building in the December 1996 meeting. The containment internal structures summary report will be reviewed in January 1997 audit meeting. (12/16/96) Action W - meeting 1/16/97 - determine loads on air baffle due to flow and vortex shedding and demonstrate adequacy of air baffle.					
757	NRR/ECGB	3.8.4.5-1	DSER-OI		Orr / NRCSM	Closed	Action W		
				Westinghouse should include in Appendix 3A of the SSAR, a description of criteria used for the different configurations and applications if there are differences in the details of these modules.					
				See OI 3.8.4.1-3. Closed in meeting with NRC 1/16/97 - minor SSAR change shown in draft revision					
758	NRR/ECGB	3.8.4.5-2	DSER-OI		Orr / Ritz / NRCSM	Closed	Action W		
				Westinghouse should provide requirements in the SSAR for modular construction in the auxiliary building.					
				Additional information to be added to SSAR as discussed in December, 1994 meeting. See NRC letter dated 7/15/96 - more information needed on quality control Closed in meeting with NRC 1/16/97 - minor SSAR change shown in draft revision					

FAX to DINO SCALETTI

February 20, 1997

CC: Sharon or Dino, please make copies for: Ted Quay
Bill Huffman
D:ane 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 provided background documentation showing the difference between "W Status" and "NRC Status". In all cases, we believe the next action is with NRC and await your definitization of a Westinghouse action or your direction to change the "NRC Status" to something other than "Action W". Note that we have received no information from NRC on items on this list for over a week.

Open Item Number	Westinghouse Submittal	Request for Status Change
21(RAI 471.24)	2/14/97	2/14/97
142 (M3.11-9)	2/29/96	2/3/97
157 (M5.2.5-13)	1/9/97	2/12/97
164 (M5.2.5-20)	1/10/97	2/12/97
172 (M5.2.5-29)	1/14/97	2/14/97
173 (M5.2.5-30)	1/14/97	2/17/97
177 (M5.2.5-34)	12/20/96	2/18/97
405	7/8/96	2/11/97
681 (DSER 3.8.2.4-3)	2/11/97	2/17/97
706 (DSER 3.8.2.4-28)	2/11/97	2/17/97
710 (DSER 3.8.3.1-1)	1/16/97	2/18/97
716 (DSER 3.8.3.2-5)	1/16/97	2/18/97
717 (DSER 3.8.3.3-1)	1/16/97	2/18/97
718 (DSER 3.8.3.3-2)	1/16/97	2/18/97
719 (DSER 3.8.3.3-3)	1/16/97	2/18/97

Open Item Number	Westinghouse Submittal	Request for Status Change
722 (DSER 3.8.3.4-3)	1/16/97	2/18/97
724 (DSER 3.8.3.4-5)	1/16/97	2/18/97
729 (DSER 3.8.3.4-10)	1/16/97	2/18/97
730 (DSER 3.8.3.4-11)	1/16/97	2/18/97
731 (DSER 3.8.3.4-12)	1/16/97	2/18/97
732 (DSER 3.8.3.4-13)	1/16/97	2/18/97
740 (DSER 3.8.4.1-3)	1/16/97	2/20/97
754 (DSER 3.8.4.4-6)	1/16/97	2/20/97
757 (DSER 3.8.4.5-1)	1/16/97	2/20/97
758 (DSER 3.8.4.5-2)	1/16/97	2/20/97
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
1354 (DSER 18.8.1.3-1)	1/30/97	2/14/97
1356 (DSER 18.8.1.3-3)	1/30/97	2/14/97
1360 (DSER 18.8.1.3-7)	12/19/96	2/14/97
1361 (DSER 18.8.1.3-8)	12/19/67	2/14/97
1365 (DSER 18.9.3-3)	12/3/96	2/14/97
1366 (DSER 18.9.3-4)	12/3/96	2/14/97
1367 (DSER 18.9.3-5)	1/10/97	2/14/97
1368 (DSER 18.9.3-6)	12/3/96	2/14/97
1369 (DSER 18.9.3-7)	12/3/96	2/14/97
1370 (DSER 18.9.3-8)	12/3/96	2/14/97
1371 (DSER 18.9.3-9)	12/3/96	2/14/97
1392 (DSER 18.11.3.4-1)	1/30/97	2/14/97
1396 (DSER 18.13.3-1)	11/7/96	2/14/97
1809 (DSER-CN 3.10-2)	9/5/96	2/13/97

Open Item Number	Westinghouse Submittal	Request for Status Change
1810 (DSER-CN 3.10-3)	9/5/96	2/13/97
1811 (DSER-CN 3.10-4)	9/5/96	2/13/97
1888(DSER-COL 3.8.2.4-1)	2/11/97	2/17/97
2034	7/8/96	2/11/97
2043	1/10/97	2/14/97
2044	1/30/97	2/14/97
2347	1/16/97	2/18/97
2348	1/16/97	2/18/97
2349	1/16/97	2/18/97
3057	5/30/96	2/18/97
3247 (RAI 230.98)	4/30/96	2/18/97
3264 (RAI 220.95)	12/9/96	2/13/97
3265 (RAI 220.96)	12/9/96	2/13/97
3266 (RAI 220.97)	12/9/96	2/13/97
3267 (RAI 220.98)	12/9/96	2/13/97
3268 (RAI 220.99)	12/9/96	2/13/97
3269 (RAI 220.100)	2/11/97	2/13/97 & 2/17/97
3270 (RAI 220.101)	2/11/97	2/13/97 & 2/17/97
3271 (RAI 220.102)	2/11/97	2/13/97 & 2/17/97
4617	2/14/97	2/14/97

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 20, 1997

CC: Sharon or Dino, please make copies for: Ted Quay
Bill Huffman
Diane Jackson
Tom Kenyon
Joe Sebrosky
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This is a reminder list of the Open Items where we have recently provided background documentation showing the difference between "W Status" and "NRC Status". In all cases, we believe the next action is with NRC and await your definitization of a Westinghouse action or your direction to change the "NRC Status" to something other than "Action W". Note that we have received no information from NRC on items on this list for over a week.

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Open Item Number	Westinghouse Submittal	Request for Status Change
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730 (DSER 3.8.3.4-11)	1/16/97	2/18/97
731 (DSER 3.8.3.4-12)	1/16/97	2/18/97
732 (DSER 3.8.3.4-13)	1/16/97	2/18/97
740 (DSER 3.8.4.1-3)	1/16/97	2/20/97
754 (DSER 3.8.4.4-6)	1/16/97	2/20/97
757 (DSER 3.8.4.5-1)	1/16/97	2/20/97
758 (DSER 3.8.4.5-2)	1/16/97	2/20/97
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
1354 (DSER 18.8.1.3-1)	1/30/97	2/14/97
1356 (DSER 18.8.1.3-3)	1/30/97	2/14/97
1360 (DSER 18.8.1.3-7)	12/19/96	2/14/97
1361 (DSER 18.8.1.3-8)	12/19/67	2/14/97
1365 (DSER 18.9.3-3)	12/3/96	2/14/97
1366 (DSER 18.9.3-4)	12/3/96	2/14/97
1367 (DSER 18.9.3-5)	1/10/97	2/14/97
1368 (DSER 18.9.3-6)	12/3/96	2/14/97
1369 (DSER 18.9.3-7)	12/3/96	2/14/97
1370 (DSER 18.9.3-8)	12/3/96	2/14/97
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1392 (DSER 18.11.3.4-1)	1/30/97	2/14/97
1396 (DSER 18.13.3-1)	11/7/96	2/14/97
1809 (DSER-CN 3.10-2)	9/5/96	2/13/97

Open Item Number	Westinghouse Submittal	Request for Status Change
1810 (DSER-CN 3.10-3)	9/5/96	2/13/97
1811 (DSER-CN 3.10-4)	9/5/96	2/13/97
1888(DSER-COL 3.8.2.4-1)	2/11/97	2/17/97
2034	7/8/96	2/11/97
2043	1/10/97	2/14/97
2044	1/30/97	2/14/97
2347	1/16/97	2/18/97
2[illegible]48	1/16/97	2/18/97
2349	1/16/97	2/18/97
3057	5/30/96	2/18/97
3247 (RAI 230.98)	4/30/96	2/18/97
3264 (RAI 220.95)	12/9/96	2/13/97
3265 (RAI 220.96)	12/9/96	2/13/97
3266 (RAI 220.97)	12/9/96	2/13/97
3267 (RAI 220.98)	12/9/96	2/13/97
3268 (RAI 220.99)	12/9/96	2/13/97
3269 (RAI 220.100)	2/11/97	2/13/97 & 2/17/97
3270 (RAI 220.101)	2/11/97	2/13/97 & 2/17/97
3271 (RAI 220.102)	2/11/97	2/13/97 & 2/17/97
4617	2/14/97	2/14/97

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 20, 1997

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

Don Lindgren
Richard Orr
Ed Cummins
Bob Vijuk
Brian McIntyre

OPEN ITEMS FOR SSAR SECTION 3.8.5

This is a background package for the remaining open items for SSAR section 3.8.5 for your information. SSAR section 3.8.5 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 5 Open Items with NRC Status of Action W. All 5 of these items still require some Westinghouse action. Thank you.

Jim

Jim Winters
412-374-5290

AP600 Open Item Tracking System Database: Executive Summary   Date: 2/20/97

Selection: [nrc st code]='Action W' And [DSER Section] like '3.8.5*' Sorted by Item #

Item No.	Branch	DSER Section/ Question	Type	Title/Description Detail Status	Resp Engineer	(W) Status	NRC Status	Letter No. /	Date
766	NRR/ECGB	3.8.5-8	DSER-OI	Westinghouse should provide the validation package of INITEC's in-house computer codes for review and should verify the adequacy of the post-processed results obtained from these codes. Closed - Validation package is available for review in proposed basemat meeting in December, 1996. Comparisons of results of computer code versus hand calaculation are included in revised documentation. Action W - Westinghouse will provide technical information and final design calculations for the staff to review in meeting in December. NRC Status: Action W - Westinghouse has not provided (in english) the ARMA computer code validation and verification package. (12/16/96)	Orr	Action W	Action W		
767	NRR/ECGB	3.8.5-9	DSER-OI	Westinghouse should perform additional review of the basemat analysis, and should use simplified analysis (based on ACI 336 procedures) to verify the design adequacy. Westinghouse has performed additional review. Simplified analyses have been performed to verify design adequacy. Action W - Westinghouse will provide technical information and final design calculations for the staff to review in December meeting. NRC Status: Action W - Simple analysis was reviewed and accepted. However, the December 1996 design audit identified a lack of consistency in the analysis and design of the entire structural system. (12/16/96)	Orr	Action W	Action W		
768	NRR/ECGB	3.8.5-10	DSER-OI	Westinghouse should perform additional analyses for construction loads. Analyses have been performed for construction loads as described in SSAR subsection 3.8.5.4.2. The results are used in reinforcement design. As discussed in June, 1996 meeting, Westinghouse will consider effect of long term settlement. NRC requested that the acceptable settlement during construction should be addressed in the SSAR. NRC Status: Action W - The SSAR 3.8.5 draft is incomplete; more information is needed. Westinghouse does not consider the effects of settlement during construction and dewatering during construction. (12/16/96) Resolution of this item will also resolve Item 547 (DSER OI 2.5.4.8-1)	Orr / Bechtel / NRCBM	Action W	Action W		
769	NRR/ECGB	3.8.5-11	DSER-OI	Westinghouse should perform additional analyses to evaluate the effects of (1) local soft spots of soil foundation, (2) soil springs to the foundation mat design with non-uniform stiffnesses, and (3) soil stiffness corresponding to other soil conditions used in the design. Additional analyses have been performed as described in SSAR subsection 3.8.5.4.2. Action W - Subject was discussed during meeting on July 11, 1996. Westinghouse will include in the SSAR site interface criteria related to the local variability of soil stiffness below the foundation. The allowable variability will be included in the design of the basemat. Soil varaibility has been addressed in the documentation available for review in meeting in December. A draft SSAR revision is being prepared. NRC Status: Action W - The SSAR 2.5.4 draft is incomplete; more information is needed on the geotechnical program. A cross-reference is needed in SSAR 3.8.5. (12/16/96)	Orr / NRCBM	Action W	Action W		
772	NRR/ECGB	3.8.5-14	DSER-OI	Westinghouse should commit in the SSAR to use coated reinforcing bars for the design of the NI foundation. The seismic Category I structures below grade are protected against flooding by waterstops and a waterproofing system. The waterproofing system is provided by the introduction of a cementitious crystalline waterproofing additive to the nailed soil retention wall shotcrete or to the shotcrete applied to the rock surface. This waterproofing system plus concrete cover to the reinforcement is considered to provide adequate protection. NRC Status: Action W - Westinghouse needs to ensure a consistent waterproof system, specifically the interface between the mudmat and the shotcrete sidewall. This information needs to be in SSAR 3.4.1.1.1. A cross-reference is needed in SSAR 3.8.5.6. (12/16/96)	Orr/BPC	Action W	Action W		

2 of 2

FAX to DINO SCALETTI

February 20, 1997

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

Don Lindgren
Ron Vijuk
Terry Schulz
Mike Corletti
Ed Cummins
Bob Vijuk
Brian McIntyre

OPEN ITEM #182 (M5.4.11-5)

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 #182 (M5.4.11-5) is attached. We provided the original fax of a markup on January 10, 1997 (over a month ago). We intend to include this SSAR change in Revision 11 to be issued next week. We believe that this information resolves the concerns of item #182. 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: 2/20/97

Selection: [item no] between 182 And 182 Sorted by Item #

Item No.	Branch	DSER Section/ Question	Type	Title/Description Detail Status	Resp Engineer	(W) Status	NRC Status	Letter No. /	Date
182	NRR/SPLB	5.4.11	MTG-OI		Corletti,M.	Closed	Action W		

M5.4.11-5 (PRESSURIZER RELIEF DISCHARGE) Section 5.4.11.3 states that the IRWST is sized based on the heat load and steam volume following an actuation of the ADS. Does this include steam, water, and noncondensable gases from all three ADS stages? Provide the analysis

Closed - See Section 6.3 for a discussion of the IRWST during accuation of the automatic depressurization system. DISCUSSED AT 1/25/95 MEETING BETWEEN WESTINGHOUSE AND NRC PLANT SYSTEMS BRANCH

Action N - Need to review following Westinghouse providing of specific SSAR reference. Specific reference provided by fax markup of SSAR on January 10, 1997.

5 of 2

 Total Records: 1

Westinghouse

FAX COVER SHEET

RECIPIENT INFORMATION		SENDER INFORMATION	
DATE:	JANUARY 10, 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 + 2

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

THIS SHOULD SATISFY THE NRC REQUEST ON 12/2/96 related to Open Item 182. NOTE THAT SPECIFIC SSAR SUBSECTION IS 6.3.2.2.3 but editorial conventions REQUIRE TRUNCATING THE NUMBER. THIS MARKUP WILL GO INTO SSAR REVISION 11 UNLESS WE HEAR FROM YOU.

Jim Winters

cc: BUTLER
LINDGREN
McINTYRE
CUMMINS
KUNVEUIC
SCHULZ
CORLETTI
JEANNE EVANS

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lifetime (thermal, weight, and pressure) are applied, and stresses are compared to allowable values. Subsection 3.9.3 discusses the modeling and analysis methods.

5.4.10.4 Tests and Inspections

Nondestructive examinations are performed according to the procedures of the ASME Code, Section V, except as modified by the ASME Code, Section III, Subsection NF.

5.4.11 Pressurizer Relief Discharge

The AP600 does not have a pressurizer relief discharge system. The AP600 has neither power operated pressurizer relief valves nor a pressurizer relief discharge tank. Some of the functions provided by the pressurizer relief discharge system in previous nuclear power plants are provided by portions of other systems in the AP600.

The safety valves connected to the top of the pressurizer provide for overpressure protection of the reactor coolant system. First- second-, and third-stage automatic depressurization system valves provide for depressurization of the reactor coolant system and venting of noncondensable gases in the pressurizer following an accident. These functions are discussed in subsections 5.2.2, 5.4.12, and in Section 6.3. The AP600 does not have power operated relief valves connected to the pressurizer.

The discharge of the safety valves is directed through a rupture disk to containment atmosphere.

The discharge of the first-, second-, and third-stage automatic depressurization system valves is directed to the in-containment refueling water storage tank. For the automatic depressurization system valves, the following discussion considers only the gas venting function. Only the first stage automatic depressurization valves are used to vent non-condensible gases following an accident. The sizing considerations and design basis for the in-containment refueling water storage tank for the depressurization function are discussed throughout ~~in~~ Section 6.3. The provisions to minimize the differential pressure between the containment atmosphere and the interior of the in-containment refueling water storage tank are also discussed in ~~Section 6.3~~ subsection 6.3.2.

The safety valve on the normal residual heat removal system, which provides low temperature overpressure protection, discharges into the in-containment refueling water storage tank. See subsection 5.4.7 for a discussion of the connections to and location of the safety valve in the normal residual heat removal system.

5.4.11.1 Design Bases

The containment has the capability to absorb the pressure increase and heat load resulting from the discharge of the safety valves to containment atmosphere. The in-containment refueling water storage tank has the capability to absorb the pressure increase and heat load from the discharge, including the water seal, steam and gases, from a first-stage automatic depressurization system valve when used to vent noncondensable gases from the pressurizer following an accident. The venting of noncondensable gases from the pressurizer following an accident is not a safety-related function.

5.4.11.2 System Description

Each safety valve discharge is directed to a rupture disk at the end of the discharge piping. A small pipe is connected to the discharge piping to drain away condensed steam leaking past the safety valve. The discharge is directed away from any safety related equipment, structures, or supports that could be damaged to the extent that emergency plant shutdown is prevented by such a discharge.

The discharge from each of two groups of automatic depressurization system valves is connected to a separate sparger below the water level in the in-containment refueling water storage tank. The piping and instrumentation diagram for the connection between the automatic depressurization system valves and the in-containment refueling water storage tank is shown in Figure 6.3-1. The in-containment refueling water storage tank is a stainless steel lined compartment integrated into the containment interior structure. The discharge of water, steam, and gases from the first-stage automatic depressurization system valves when used to vent noncondensable gases does not result in pressure in excess of the in-containment refueling water storage tank design pressure. Additionally, vents on the top of the tank protect the tank from overpressure as described in subsection 6.3.2.

Overflow provisions prevent overfilling of the tank. The overflow is directed into the refueling cavity. The in-containment refueling water storage tank does not have a cover gas and does not require a connection to the waste gas processing system. The normal residual heat removal system provides nonsafety-related cooling of the in-containment refueling water storage tank.

5.4.11.3 Safety Evaluation

The design of the control for the reactor coolant system and the volume of the pressurizer is such that a discharge from the safety valves is not expected. The containment design pressure, which is based on loss of coolant accident considerations, is greatly in excess of the pressure that would result from the discharge of a pressurizer safety valve. The heat load resulting from a discharge of a pressurizer safety valve is considerably less than the capacity of the passive containment cooling system or the fan coolers. See Section 6.2.

Venting of noncondensable gases, including entrained steam and water from the loop seals in the lines to the automatic depressurizations system valves, from the pressurizer into spargers below the water line in the in-containment refueling water storage tank does not result in a significant increase in the pressure or water temperature. The in-containment refueling water storage tank is not susceptible to vacuum conditions resulting from the cooling of hot water in the tank as described in subsection 6.3.2. The in-containment refueling water storage tank has capacity in excess of that required for venting of noncondensable gases from the pressurizer following an accident.

FAX to DINO SCALETTI

February 20, 1997

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

Don Lindgren
Ron Vijuk
Terry Schulz
Mike Corletti
Ed Cummins
Bob Vijuk
Brian McIntyre

OPEN ITEM #184 (M5.4.11-7)

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 #184 (M5.4.11-7) is attached. We provided the original fax of a markup on January 13, 1997 (over a month ago). We intend to include this SSAR change in Revision 11 to be issued next week. We believe that this information resolves the concerns of item #184. 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

Selection: [item no] between 184 And 184 Sorted by Item #

Item No	Branch	DSER Section/ Question	Type	Title/Description Detail Status	Resp Engineer	(W) Status	NRC Status	Letter No. /	Date
184	NRR/SPLB	5.4.11	MTG-OI	M5.4.11-7 (PRESSURIZER RELIEF DISCHARGE) Where is the instrumentation for the ADS valve discharge lines discussed? Closed - See Section 6.3 for instrumentation requirements for the automatic depressurization system. DISCUSSED AT 1/25/95 MEETING BETWEEN WESTINGHOUSE AND NRC PLANT SYSTEMS BRANCH Action N - Need to review following Westinghouse providing of specific SSAR reference Action W - Give NRC explicit, specific reference to the part of Section 6.3 that answers the question. Action N - FAX with markup of appropriate changes to Chapter 5 provided on 1/13/97.	Corletti,M	Closed	Action W		

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Westinghouse

FAX COVER SHEET

RECIPIENT INFORMATION		SENDER INFORMATION	
DATE:	JANUARY 13, 1997	NAME:	JIM WINTERS
TO:	BILL HUFFMAN	LOCATION:	ENERGY CENTER - EAST
PHONE:	FACSIMILE:	PHONE:	Office: 412-374-5290
COMPANY:	USNRC	Facsimile:	win: 284-4887 outside: (412)374-4887
LOCATION:			

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COMMENTS:

BILL

THIS MARKUP SHOULD RESOLVE OPEN ITEM 184 FROM OUR DECEMBER 2, 1996 TELECON. IT WILL BE IN SSAR REVISION 11 UNLESS WE HEAR FROM YOU. WE RECOMMEND THIS BE CHANGED TO "ACTION N" SINCE SPECIFIC REFERENCES ARE INCLUDED IN THIS FAX.

Jim Winters

cc: BUTLER
LINDGREN
MCINTYRE
CUMMINS
RON VIJUK
WINTERS
CORLETTI
SCHULZ
JEANNE EVANS

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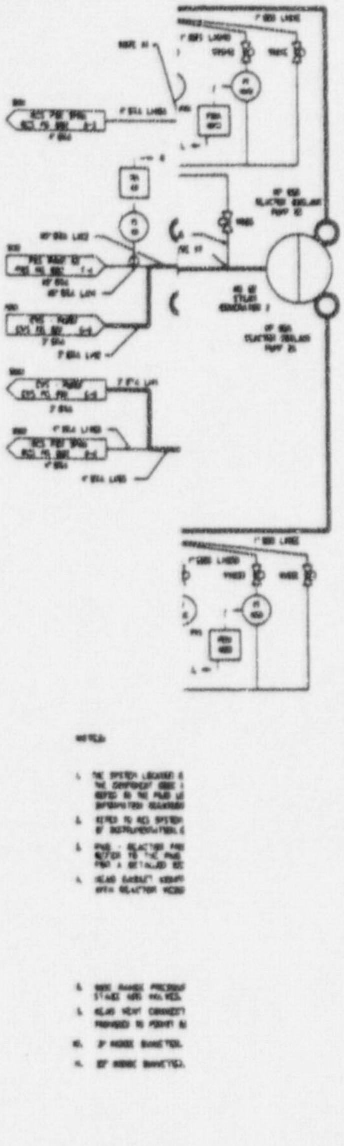

Figure 5.1-5 (Sheet 1 of 3)
Reactor Coolant System
~~Containment Internal Structures~~
Piping and Instrumentation Diagram

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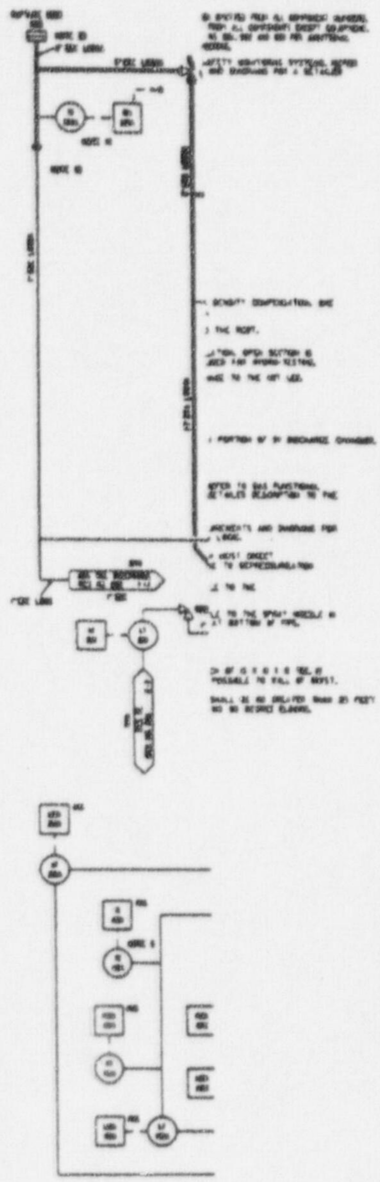

Figure 5.1-5 (Sheet 2 of 3)
Reactor Coolant System
~~Containment Internal Structures~~
Piping and Instrumentation Diagram

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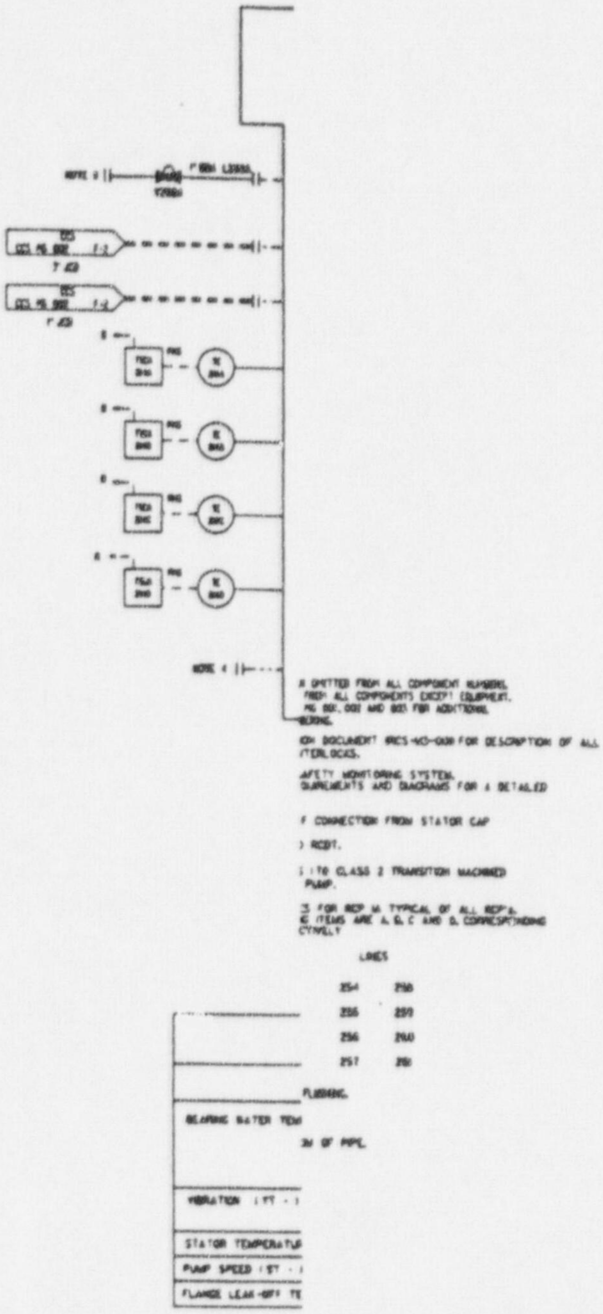

Figure 5.1-5 (Sheet 3 of 3)
Reactor Coolant System
~~Containment Internal Structures~~
Piping and Instrumentation Diagram

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

and Automatic Depressurization

5.2.5.1.3 Pressurizer Safety Relief Valves

just downstream of each valve

and each automatic depressurization valve mounted on the pressurizer

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~~ valves 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.

one of the

or the incontainment refueling water storage 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

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.

5.4.11.4 Instrumentation Requirements

5.2.5,

The instrumentation for the safety valve discharge pipe, containment, and in-containment refueling water storage tank are discussed in subsection 5.4.9 and in Sections 6.2 and 6.3, respectively. Separate instrumentation for the monitoring of the discharge of noncondensable gases in not required.

5.4.11.5 Inspection and Testing Requirements

Sections 6.2 and 6.3 discuss the requirements for inspection and testing of the containment and in-containment refueling water storage tank, including operational testing of the spargers. Separate testing is not required for the noncondensable gas venting function.

5.4.12 Reactor Coolant System High Point Vents

The requirements for high point vents are provided for the AP600 by the reactor vessel head vent valves and the automatic depressurization system valves. The primary function of the reactor vessel head vent is for use during plant startup to properly fill the reactor coolant system and vessel head. Both reactor vessel head vent valves and the automatic depressurization system valves may be activated and controlled from the main control room. The AP600 does not require use of a reactor vessel head vent to provide safety-related core cooling following a postulated accident.

The reactor vessel head vent valves (Figure 5.4-8) can remove noncondensable gases or steam from the reactor vessel head to mitigate a possible condition of inadequate core cooling or impaired natural circulation through the steam generators resulting from the accumulation of noncondensable gases in the reactor coolant system. The design of the reactor vessel head vent system is in accordance with the requirements of 10 CFR 50.34 (f)(2)(vi).

The first stage valves of the automatic depressurization system are attached to the pressurizer and provide the capability of removing noncondensable gases from the pressurizer steam space following an accident. Venting of noncondensable gases from the pressurizer steam space is not required to provide safety-related core cooling following a postulated accident. Gas accumulations are removed by remote manual operation of the first stage automatic depressurization system valves.

The discharge of the automatic depressurization system valves is directed to the in-containment refueling water storage tank. Subsection 5.4.6 and Section 6.3 discuss the automatic depressurization system valves and discharge system.

The passive residual heat removal heat exchanger piping and the core makeup tank inlet piping in the passive core cooling system include high point vents that provide the capability of removing noncondensable gases that could interfere with heat exchanger or core makeup tank operation. These gases are normally expected to accumulate when the