Westinghouse
Electric Corporation

Energy Systems

Box 355
Pittsburgh Pennsylvania 15230-0355

NSD-NRC-97-5031
DCP/NRC0792
Docket No.: STN-52-003

April 1, 1997

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

ATTENTION: T. R. QUAY

SUBJECT: INFORMAL CORRESPONDENCE

Dear Mr. Quay:

Please find enclosed a formal transmittal of correspondence we have previously sent to you informally. This informal correspondence was sent over the period March 11, 1997 through March 27, 1997.

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

Brian A. McIntyre, Manager
Advanced Plant Safety and Licensing

jml

Attachment
Enclosure

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

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Attachment 1 to Westinghouse Letter DCP/NRC0792

DATE	ADDRESSEE	DESCRIPTION
3/12/97	Sebrosky/ Huffman	SSAR markup pages for subsections 6.3.2.1.2, 6.3.2.2.7.7 and 6.3.2.2.7.9. Will be in Revision 12 unless we hear otherwise
3/19/97	Scaletti	Information on open item #204. Request to acknowledge receipt and either provide definitive action for Westinghouse or change status to action N.
3/19/97	Scaletti	Information on open item #205. Request to acknowledge receipt and either provide definitive action for Westinghouse or change status to action N.
3/19/97	Scaletti	Information on open item #305.
3/19/97	Jackson	Markup of SSAR section 3.9.3.1.1. Will be in SSAR revision 12 unless we hear otherwise.
3/20/97	Jackson	Markup of SSAR section 3.7.4.2.1. Will be in SSAR revision 12 unless we hear otherwise
3/20/97	Scaletti	Open items for Chapter 12. Request for acknowledgement of receipt of information for open item # 1210.
3/20/97	Scaletti	Open items for Chapter 13. Request to acknowledge receipt and either provide definitive action for Westinghouse or change status for items 1222, 1225, 1226 and 2033.
3/17/97	Scaletti	Open item # 184. Third request to acknowledge receipt and either provide definitive action for Westinghouse or change status to action N
3/20/97	Kenyon	Additional information related to open item 21. Request for NRC review.
3/17/97	Scaletti	Open item # 302. Request to acknowledge receipt and either provide definitive action for Westinghouse or change status to action N or closed
3/17/97	Scaletti	Open item # 706. Resubmittal of information to obtain NRC acknowledgment of receipt and either provide definitive action for Westinghouse or change status to action N
3/17/97	Scaletti	Open item # 681. Resubmittal of information to obtain NRC acknowledgment of receipt and either provide definitive action for Westinghouse or change status to action N
3/18/97	Huffman	Draft responses to NRC questions of Technical Specifications
3/19/97	Fineman	Use of quench model in NOTRUMP final validation report.
3/20/97	Kenyon	Information to close open item # 5011

3/20/97	Jackson	Information on actions from 3/7/97 fire protection meeting. Will also be attached to a letter stating we have COMPLETED all actions on fire protection
3/20/97	Scaletti/Quay	Open item status charts
3/21/97	Kenyon	Markup of PMS ITAAC to close open item #1044. Request to change status.
3/18/97	Huffman	Response to two questions on Chapter 18 minimum inventory
3/21/97	Jackson	Markup of SSAR to close open items # 472 and 1172. Will be in SSAR revision 12 unless we hear otherwise. Request to change NRC status
3/20/97	Throm	Material to support 3/25 WGOTHIC meeting
3/21/97	Jackson	Comments on draft notes from 3/3 senior management meeting
3/20/97	Huffman	Advanced draft copy of DCP/NRC0776 on tech spec comments from NRC letter of 12/24/96
3/21/97	Throm	Material to support 3/25 WGOTHIC meeting
3/21/97	Throm	Material to support 3/25 WGOTHIC meeting
3/21/97	Throm	List of open items to be discussed at 3/25 WGOTHIC meeting
3/25/97	Jackson	SSAR markup to close open items 1171 and 1179 from 3/19/97 telecon. Will be in SSAR revision 12 unless we hear otherwise.
3/27/97	Kenyon	SSAR markup to close open item 19, the final open item in Chapter 12. Will be in SSAR revision 12 unless we hear otherwise
3/11/97	Jackson	Actions from 3/6 PCS WGOTHIC meeting
3/27/97	Sebrosky/ Scaletti	Request to confirm that the responses to 146 Level 2 PRA and severe accident open items (1652-1678, 1682-1689, 1691, 1692, 1695-1704, 1706, 1707, 2141-2146, 2148-2150, 2152-2154, 2156-2207, 2209-2219, 4123-4144) have been received. Request to change status to Action N.

Date: Thu, 27 Mar 1997 15:04:39 -0500
To: jms3@nrc.gov
From: Cindy Haag <haagcl@wcsmail.com>
Subject: List from OITS need NRC Status
Cc: mcintyha, HAAGCL, dcs1@nrc.gov

Email addressed to:
Joe Sebrosky (NRC)
cc: Dino Scalletti (NRC)
From: Cindy Haag (Westinghouse)

Joe,

As we've discussed in the past, there are a number of Level 2 PRA and severe accident topic open items in the OITS that Westinghouse has provided a response to and consider the item resolved/closed, but the NRC status is still labeled as "Action W". There is a total of 146 items.

Please talk with the staff and let me know that you (1) have indeed received the responses, and (2) that we can thus change the NRC Status column at least to "Action N".

>From the OITS, here's the item numbers in question:
1652 - 1678, 1682 - 1689, 1691, 1692, 1695 - 1704, 1706, 1707, 2141 - 2146, 2148 - 2150, 2152 - 2154, 2156 - 2207, 2209 - 2219, 4123 - 4144.

Many of the responses were provided as far back as 9/28/95. I believe most (>90%) of them are within the Containment Systems Branch area.

I've got a list of which ones were sent when if that's of use.

Please get back to me on the NRC Status of these open items in the near future. If there is indeed something we owe you from these RAI responses that we don't know about, then Westinghouse needs to fully understand what we still owe the NRC.

Thanks much.

Cindy

Westinghouse

FAX COVER SHEET

RECIPIENT INFORMATION		SENDER INFORMATION	
DATE:	3-11-97	NAME:	BRUCE RARIG
TO:	D. JACKSON	LOCATION:	ENERGY CENTER - EAST
PHONE:	FACSIMILE: (301) 415-2002	PHONE:	Office: 4358
COMPANY:		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:

Diane,

These are the actions we are taking as a result of our meeting on 3/6.

Bruce

** TX CONFIRMATION REPORT **　　AS OF　MAR 11 '97 14:16　PAGE.01

AP600 DESIGN CERT

	DATE	TIME	TO/FROM	MODE	MIN/SEC	PGS	STATUS
01	3/11	14:15	813014152002	G3--S	01"17	03	OK

Rarig, Bruce

From: Rarig, Bruce
To: Ofstun, Richard P; Loftus, Mike; Rarig, Bruce; Winters, James; Andreychek, Tim S.; Gresham, Jim A.; Hicks, William W. 001; Mcintyre, Brian A.; Spencer, Daniel R.; Vijuk, Robert M.; Woodcock, Joel 001
Subject: ACTION ITEMS FROM 3/6 NRC MEETING ON PCS/WGOTHIC
Date: Monday, March 10, 1997 4:43PM

SUBJECT: Action Items from NRC Meeting on 3/6/97

ATTENDEES: T. Quay, B. McIntyre, D. Jackson, J. Gresham, E. Throm, J. Woodcock, B. Gitnick, D. Spencer, D. Prelewicz, M. Loftus, J. Kudrick, B. Rarig, C. Berlinger

PURPOSE:

1. Present PCS PIRT (WCAP-14812) and discuss closure path
2. Present PCS Scaling Report (WCAP-14845)
3. Discuss process for closure of PCS/WGOTHIC RAIs

ACTIONS:

1. Westinghouse will include a description of the expert review process in Section 2 of the PIRT.

2. Westinghouse will include a synopsis of the experts' review comments and Westinghouse's resolution of the comments in an Appendix to the PIRT.

3. NRC/WEC telecon is scheduled for 3/13 (1 PM) to receive additional NRC comments on the PIRT.

4. Westinghouse will provide the following additional information on the bases and justification for the PIRT rankings: references to specific scaling Pi groups; description of engineering judgement bases; and reference to specific tests and types of data used to draw conclusions for rankings.

5. Westinghouse will move information from Chapter 2 of WCAP-14407 (Tables 2-3 and 2-4) into an Appendix in the PIRT. These tables provide a summary of test and analyses bases for each phenomenon and how the phenomenon is addressed in the evaluation model.

6. NRC/WEC working level meeting to discuss closure paths for the WGOTHIC Application Report Chapter 7 and 9 RAIs, and informal questions is tentatively scheduled for 3/25. The firm date will be determined during the 3/13 telecon.

7. NRC/WEC working level meeting to reach agreement on revisions to the PCS PIRT is tentatively planned for early April. Westinghouse will send proposed revisions to address items 1 through 5, above, at least one week

prior to the meeting.

8. Westinghouse will prepare a letter that describes the basis for eliminating Chapter 13 from the WGOTHIC Application Report. This letter will also close all RAIs related to only Chapter 13.

Westinghouse

FAX COVER SHEET

RECIPIENT INFORMATION		SENDER INFORMATION	
DATE:	MARCH 27, 1997	NAME:	Jim Winters
TO:	Tom Kenyon	LOCATION:	ENERGY CENTER - EAST
PHONE:	FACSIMILE:	PHONE:	Office:
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:

Tom.

THIS MARKUP SHOULD RESOLVE OUR FINAL CHAPTER 12 QUESTION. WE AGREED TO ADD A POST ACCIDENT EVALUATION SECTION TO CLOSE OUT OPEN ITEM 19. I HOPE THIS IS WHAT MR. HINSON ASKED FOR. IT WILL GO INTO REVISION 12 OF THE SSAR UNLESS WE HEAR FROM YOU

Jim Winters

cc: LINDGREN
CUMMINS
MCINTYRE
NOVAK VIJUK
WINTERS
SEJUAR
M. WILLS
JEANNE EVANS.

of advanced technology into the refueling process also reduces doses. Table 12.4-11 lists some of the AP600 features that reduce doses during refueling operations.

Table 12.4-12 provides dose estimates for the various refueling activities.

12.4.1.7 Overall Plant Doses

The estimated annual personnel doses associated with the six activity categories discussed above are summarized below:

Category	Percent of Total	Estimated Annual Dose (man-rem)
Reactor operations and surveillance	20.6	13.8
Routine inspection and maintenance	18.0	12.1
Inservice inspection	24.6	16.5
Special maintenance	22.4	15.0
Waste processing	7.8	5.2
Refueling	6.6	4.4
Total	100.0	67.0

These dose estimates are based on operation with an 18-month fuel cycle and are bounding for operation with a 24-month fuel cycle.

INSERT 12.4

12.4.2 Radiation Exposure at the Site Boundary

12.4.2.1 Direct Radiation

The direct radiation from the containment and other plant buildings is negligible. The AP600 design also provides storage of refueling water inside the containment instead of in an outside storage tank that eliminates it as a radiation source.

12.4.2.2 Doses due to Airborne Radioactivity

Subsection 11.3.3 discusses doses at the site boundary due to activity released as a result of normal operations.

12.4.3 Combined License Information

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

Insert 12.4

12.4.1.8 Post Accident Actions

Requirements of 10 CFR 52.79(b) relative to plant area access and post-accident sampling (10 CFR 50.34 Item (2)(viii)) are included in Section 1.9.3. If procedures are followed, the design limits radiation exposures to any individual to not exceed 5 rem to the whole-body or 75 rem to the extremities. Plant areas for post-accident personnel access are addressed in Section 12.3, including the radiation zone maps included as Figure 12.3-2. This figure shows projected radiation zones in areas requiring access and access routes for ingress, egress and performance of actions at these locations. The radiation zone maps reflect maximum radiation fields over the course of an accident. The analyses that confirm that the dose limits are not exceeded reflect the time-dependency of the area dose rates and the required post-accident access times. The areas that require post-accident accessibility are:

1) Main control room
2) Primary sampling room
3) Class 1E regulating transformer areas
4) Ventilation control area for I&C rooms with PAMS equipment
5) Valve area to align spent fuel pool makeup
6) Ancillary diesel room
7) Passive containment cooling water inventory make-up area

The area which results in the highest individual personnel exposures is the primary sampling room. The design provides for access to the primary sampling room as early as eight hours after the accident when radiation fields are high compared to 64 hours or later for the other areas requiring access outside the main control room. In addition to the design provisions, individual exposure for this early sampling operation may be minimized by proper administrative operational controls. Special operational controls would only be considered in the event that radiation fields associated with access to the primary sampling room reach the conservatively high levels considered in the evaluations. These conservatively high levels include activity releases as defined in NUREG-1465, maximum design basis leak rate from containment into the access areas and no operable building ventilation systems.

Westinghouse

FAX COVER SHEET

RECIPIENT INFORMATION		SENDER INFORMATION	
DATE:	MARCH 25, 1997	NAME:	JIM WINTERS
TO:	DIANE JACKSON	LOCATION:	ENERGY CENTER - EAST
PHONE:	FACSIMILE:	PHONE:	Office:
COMPANY:	USNRC	Facsimile:	win: 284-4887 outside: (412)374-4887
LOCATION:			

Cover + Pages 1 + 5

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

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

COMMENTS:

DIANE

THIS MARKUP SHOULD RESOLVE ITEMS 1171 AND 1179 FROM OUR 3/19/97 TELECON. "W STATUS" WILL BE CHANGED TO "CLOSED." THIS MARKUP WILL BE IN REVISION 12 UNLESS WE HEAR FROM YOU.

Jim Winters

CC: LINDGREN
MCINTYRE
CUMMINS
RON VIJUK
WINTERS (2)
ISRAELSON
JEANNE EVANS

Table 11.1-2 lists the resulting reactor coolant radionuclide concentrations. The values presented are the maximum values calculated to occur during the fuel cycle from startup through the equilibrium cycle. Thus, the source term does not represent any particular time in the fuel cycle but is a conservative composite.

INSERT 11.1-3

11.1.1.2 Corrosion Products

The reactor coolant corrosion product activities are based on operating plant data and are independent of fuel defect level. The concentrations of corrosion products are included in Table 11.1-2.

11.1.1.3 Tritium

A number of tritium production processes add tritium to the reactor coolant:

- Fission product formation in the fuel (ternary fission) forms tritium which can diffuse through the fuel clad or leak through fuel clad defects
- Neutron reactions with soluble boron in the reactor coolant
- Burnable neutron absorber
- Neutron reactions with soluble lithium in the reactor coolant
- Neutron reactions with deuterium in the reactor coolant

The first two processes are the principal contributors to tritium in the reactor coolant. Table 11.1-3 lists the tritium introduced to the reactor coolant from each of the processes.

Tritium exists in the reactor coolant primarily as tritium oxide (that is, a tritium atom replaces a hydrogen atom in a water molecule) and thus cannot be readily separated from the coolant by normal processing methods. The maximum concentration of tritium in the reactor coolant is less than 3.5 microcuries per gram as a result of losses due to leakage and the controlled release of tritiated water to the environment.

11.1.1.4 Nitrogen-16

Activation of oxygen in the coolant results in the formation of N-16 which is a strong gamma emitter. Because of its short half-life of 7.11 seconds, N-16 is not of concern outside the containment. Table 12.2-3 provides N-16 concentrations at various points in the reactor coolant system. After shutdown, N-16 is not a source of radiation inside of containment.

11.1.2 Design Basis Secondary Coolant Activity

Steam generator tube defects cause the introduction of reactor coolant into the secondary cooling system. The resulting radionuclide concentrations in the secondary coolant depend

Table 11.1-2

DESIGN BASIS REACTOR COOLANT ACTIVITY

Nuclide	Activity (μCi/g)	Nuclide	Activity (μCi/g)
Kr-83m	8.9×10^{-2}	Rb-89	3.5×10^{-2}
Kr-85m	4.7×10^{-1}	Rb-88	7.6×10^{-1}
Kr-85	2.1	Sr-89	4.1×10^{-4}
Kr-87	2.5×10^{-1}	Sr-90	2.5×10^{-5}
Kr-88	7.7×10^{-1}	Sr-91	9.8×10^{-4}
Kr-89	1.7×10^{-2}	Sr-92	2.0×10^{-4}
Xe-131m	7.5×10^{-1}	Y-90	6.5×10^{-6}
Xe-133m	9.8×10^{-1}	Y-91m	5.2×10^{-4}
Xe-133	$6.9 \times 10^{+1}$	Y-91	5.4×10^{-5}
Xe-135m	7.5×10^{-2}	Y-92	1.8×10^{-4}
Xe-135	2.1	Y-93	4.9×10^{-5}
Xe-137	3.4×10^{-2}	Zr-95	6.4×10^{-5}
Xe-138	1.2×10^{-1}	Nb-95	6.4×10^{-5}
Br-83	1.5×10^{-2}	Mo-99	7.8×10^{-2}
Br-84	7.9×10^{-3}	Tc-99m	7.2×10^{-2}
Br-85	9.6×10^{-4}	Ru-103	5.5×10^{-5}
I-129	6.7×10^{-9}	Rh-103m	5.7×10^{-5}
I-130	4.0×10^{-3}	Rh-106	2.0×10^{-5}
I-131	2.8×10^{-1}	Ag-110m	1.8×10^{-4}
I-132	4.8×10^{-1}	Te-129m	1.0×10^{-3}
I-133	5.8×10^{-1}	Te-127m	3.1×10^{-4}
I-134	1.0×10^{-1}	Te-129	1.6×10^{-3}
I-135	3.2×10^{-1}	Te-131m	2.7×10^{-3}
Cs-134	2.3×10^{-1}	Te-131	2.0×10^{-3}
Cs-136	5.3×10^{-1}	Te-132	3.1×10^{-2}
Cs-137	2.1×10^{-1}	Te-134	5.0×10^{-3}
Cs-138	1.8×10^{-1}	Ba-137m	2.0×10^{-1}
Cr-51	1.3×10^{-3}	Ba-140	4.1×10^{-4}
Mn-54	6.7×10^{-4}	La-140	1.1×10^{-4}
Mn-56	1.7×10^{-1}	Ce-141	6.4×10^{-5}
Fe-55	5.0×10^{-4}	Ce-143	5.3×10^{-5}
Fe-59	1.3×10^{-4}	Pr-143	5.9×10^{-5}
Co-58	1.9×10^{-3}	Ce-144	4.8×10^{-5}
Co-60	2.2×10^{-4}	Pr-144	4.8×10^{-5}

Note:
These activities are used for shielding and radwaste system design. For maximum release calculations, multiply activities by 4 except for iodine, noble gases, and corrosion products.

REPLACE WITH INSERT 11.1-8

INSERT 11.1-3

This design basis source term based on 0.25 percent fuel defects is used to ensure a consistent set of design values for interfaces among the radioactive waste processing systems. The Technical Specifications in Chapter 16 which are related to fuel failure are also based upon 0.25 percent fuel defects. In addition, the liquid and gaseous radioactive waste processing systems have the capability to process wastes based upon 1.0 percent fuel defects.

INSERT 11.1-8

These activities are used for shielding and radwaste system interface design. For 1 percent fuel defect calculations (maximum release and liquid and gaseous radwaste system capability) multiply the activities above by 4 except for iodine, noble gases and corrosion products.

The radioactivity of the dry active waste is expected to normally range from 0.1 curies per year to 8 curies per year with a maximum of about 16 curies per year. This waste includes spent HVAC filters, compressible trash, non-compressible components, mixed wastes and solidified chemical wastes. These activities are produced by relatively long lived radionuclides (such as Cr-51, Fe-55, Co-58, Co-60, Nb-95, Cs-134 and Cs-137), and therefore, radioactivity decay during processing and storage is minimal. These activities thus apply to the waste as generated and to the waste as shipped.

The estimated expected and maximum annual quantities of waste influents by source and form are listed in Table 11.4-1 with disposal volumes. The influent volumes are conservatively based on an 18-month refueling cycle. Annual quantities based on a 24-month refueling cycle are less than those for an 18-month cycle. The estimated expected isotopic curie content of the primary spent resin and filter cartridge wastes to be shipped offsite are presented in Table 11.4-4 based on 90 days of decay before shipment. The same information is presented in Table 11.4-5 for the estimated maximum activities based on 30 days of decay before shipment.

Section 11.1 provides the bases for determination of liquid source terms used to calculate several of the solid waste management system influent source terms. The influent data presented in Tables 11.4-2 and 11.4-3 are conservatively based on Section 11.1 (design basis (Technical Specification) values.

Shipped volumes of radwaste for disposal are estimated in Table 11.4-1 from the estimated expected or maximum influent volumes by making adjustments for volume reduction processing by mobile systems and the expected container filling efficiencies. For drum compaction, the overall volume reduction factor, including packaging efficiency, is 3.6. For box compaction, the overall volume reduction factor is 5.4. These adjustments result in a packaged internal waste volume for each waste source, and the number of containers required to hold this volume is based on the container's internal volume. The disposal volume is based on the number of containers and the external (disposal) volume of the containers.

The disposal volumes of wet and dry wastes are approximately 377 and 1254 cubic feet per year, respectively. The wet wastes include 315 cubic feet per year of spent ion exchange resins and deep bed filter carbon, which fills two 158 cubic feet high-integrity containers. The spent resin waste container fill station at the west end of the rail car bay of the auxiliary building provides about 5 months of storage. Solidified chemical wastes fill about three 55-gallon drums per year (about 20 cubic feet per year) and are stored in the packaged waste storage room of the radwaste building for up to 3 years as evaluated below. The mixed liquid wastes fill less than three drums per year (about 17 cubic feet per year) and are stored on containment pallets in the waste accumulation room of the radwaste building until shipped offsite for processing. One four-drum containment pallet provides nearly 2 years of storage capacity for the liquid mixed wastes as well as for the 7.5 cubic feet per year (one drum per year) of solid mixed wastes.

High-activity filter cartridges fill three drums per year (22.5 cubic feet per year) and are stored in portable processing or storage casks in the rail car bay of the auxiliary building. One three-drum cask provides storage for 1 year. The other spent filter cartridges may be compacted

Table 11.3-4 (Sheet 1 of 2)

COMPARISON OF CALCULATED OFFSITE AIRBORNE CONCENTRATIONS WITH 10 CFR 20 LIMITS

Radionuclide	Effluent Concentration Limit μCi/ml[(a)]	Expected Site Boundary[(b)] Concentration Limit μCi/ml	Fraction of Concentration Limit[(b)] (expected)	Maximum Site Boundary Concentration Limit μCi/ml[(c)]	Fraction of Concentration Limit[(c)] (maximum)
Kr-85m	1.0E-7	1.8E-11	1.8E-4	6.0E-11	6.0E-4
Kr-85	7.0E-7	1.8E-9	2.6E-3	4.0E-9	5.7E-3
Kr-87	2.0E-8	7.9E-12	4.0E-4	1.5E-11	7.5E-4
Kr-88	9.0E-9	2.4E-11	2.7E-3	7.3E-11	8.1E-3
Xe-131m	2.0E-6	8.7E-10	4.4E-4	8.7E-10	4.4E-4
Xe-133m	6.0E-7	4.2E-11	7.0E-5	6.3E-10	1.1E-3
Xe-133	5.0E-7	2.2E-9	4.4E-3	6.1E-8	1.2E-1
Xe-135m	4.0E-8	3.2E-12	8.0E-5	3.2E-12	8.0E-5
Xe-135	7.0E-8	1.7E-10	2.4E-3	4.8E-10	6.9 E-3
Xe-138	2.0E-8	2.4E-12	1.2E-4	2.8E-12	1.4E-4
I-131	2.0E-10	5.2E-14	2.6E-4	7.5E-13	3.8E-3
I-133	1.0E-9	1.9E-13	1.9E-4	1.5E-12	1.5E-3
H-3	1.0E-7	6.1E-11	6.1E-4	6.1E-11	6.1E-4
C-14	3.0E-9	5.8E-12	1.9E-3	5.8E-12	1.9E-3
Av-41 Ar	1.0E-8	2.7E-11	2.7E-3	2.7E-11	2.7E-3
Cr-51	3.0E-8	4.8E-16	1.6E-8	4.8E-16	1.6E-8
Mn-54	1.0E-9	3.4E-16	3.4E-7	3.4E-16	3.4E-7
Co-57	9.0E-10	6.5E-18	7.2E-9	6.5E-18	7.2E-9
Co-58	1.0E-9	1.8E-14	1.8E-5	1.8E-14	1.8E-5
Co-60	5.0E-11	6.9E-15	1.4E-4	6.9E-15	1.4E-4
Fe-59	5.0E-10	6.3E-17	1.3E-7	6.3E-17	1.3E-7
Sr-89	2.0E-10	2.4E-15	1.2E-5	6.6E-14	3.3E-4
Sr-90	6.0E-12	9.5E-16	1.6E-4	2.6E-14	4.3E-3
Zr-95	4.0E-10	7.9E-16	2.0E-6	1.3E-15	3.3E-6
Nb-95	2.0E-9	2.0E-15	1.0E-6	4.3E-15	2.2E-6
Ru-103	9.0E-10	6.3E-17	7.0E-8	6.3E-17	7.0E-8

March 19, 1997

Subject: Informal Transmittal of Information on
- WCAP-14407 Chs 7 & 9 list of clarification questions

To: Ed Throm Fax: 301-415-3577 (1 page)

cc: Jim Gresham Dick Haessler
Brian McIntyre Tim Andreychek

Below is a list of open items related to WCAP-14407 Chs 7 & 9, which Westinghouse would like to discuss during the 3/25/97 PCS DBA meeting. Clarification would allow preparation of an effective response.

Ch 7 Clarification is desired for 480.905.

Ch 9 Informal questions:

a. An Executive Summary was added to the NTD-NRC-95-4459 in preparing Section 9, to clarify that the approach for PCS DBA was a bounding approach. It would seem that questions 1, 2, and 3 are no longer relevant.

b. The basis for usage of Froude numbers to examine stratification within open volumes has been issued in WCAP-14845, section 6.5. Since the approach taken for DBA is to qualitatively compare the LST to AP600, and to evaluate an extreme stratification gradient, no functional relation between stratification and Froud number is required. In light of this, questions 15, 16, 17, 20, 25, 27, 28, 29, 30, and 33 may not be entirely relevant.

c. Clarification is desired for questions 5, 6, 26, 48, 93 and RAI 480.918.

I expect to transmit a draft of proposed approaches to respond to the remainder of the RAIs for Section 7, and the remainder of the informal questions on Section 9 tomorrow. It is planned to also discuss at the meeting, in general, followed up with a telecon, the questions related to Section 9 from the March 4, 1997 NRC letter and Section 9 RAIs 480.910 - 480.944 which came in recently with RAIs on Section 4 and Section 7 (I don't have the cover letter in hand, but am tracking it down.)

Joel Woodcock

prelim.wp

March 21, 1997

Subject: Informal Transmittal of Information on
- WCAP-14407 Ch 9 questions 1 - 66, 72, and 91

To: Ed Throm Fax: 301-415-3577 (9 pages)

cc: Jim Gresham Dick Haessler
Brian McIntyre Tim Andreychek

Attached are proposed approaches to respond to the subject NRC review questions. The ones marked with a "T" are those wnich Westinghouse recommends to have technical discussions during the 3/25/97 PCS DBA meeting.

It should be noted that WCAP-14407 Section 4, WCAP-14812, and WCAP-14845 have been issued subsequent to the issuance of NRC review comments on the preliminary draft Section 9. The attached responses assume the reviewers have access to all three WCAPs at this time.

Joel Woodcock

Joel Woodcock

prelim.wp

C = Clarification at meeting w/NRC
S = Straight forward response
T = Technical discussion @ meeting w/NRC

Responses to NRC Questions on Section 9 of WCAP-14407

Several preliminary reports on AP600 mixing and stratification effects have been issued. The knowledge base was consolidated into NTD-NRC-96-4763 (Reference 1) which was reviewed by the NRC. Subsequent to the NRC review (Reference 2) of NTD-NRC-96-4763, the report was incorporated into WCAP-14407 (Reference 3) as Section 9. Throughout the responses, section number references (i.e., Section 9) refer to the section in WCAP-14407. In addition, Appendices A, B, and C refer to new appendices which will be added to Section 9 in WCAP-14407 in response to NRC requests for additional detail. Also note that the Figure numbers given in the responses are those from WCAP-14407.

In general, Section 9 will be revised in the following way to respond to NRC review comments:

- Appendices A, B, and C will be added to Section 9 of WCAP-14407. Appendix A will provide a description of the CMT calculations performed.
- Appendix B will provide a description of the effects of stratification on heat sink utilization within a volume.
- Appendix C will provide a summary of test data available from which to judge acceptability of the circulation predicted by lumped parameter modeling.
- Additional information will be added to figures for clarification.
- Section 9 sensitivity cases will be rerun using the evaluation model in Section 4 Revision 1.
- More references will be added to the text to support the observations and conclusions, as discussed below for specific questions.
- Time phases will be made consistent with the final PIRT and Scaling time phases.

C #1, #2, #3

Discuss at meeting with NRC.

T #4 The mixing and stratification effects that must be bounded are mentioned in Section 9.1, paragraph 3 and will be summarized in Table 9-1. In summary, the pressure transient is potentially affected by parameters which influence the dominant heat removal mechanism, mass transfer. Mass transfer has as its primary parameters steam concentration, and, in the case of forced convection conditions, velocity. Large-scale circulation and entrainment into jets or plumes can drive mixing and can affect local values of steam concentration and velocity near heat transfer surfaces. Jet and plume entrainment and wall boundary layer entrainment within compartments or the above-deck region can also result in stratification, or the existence of a vertical steam concentration gradient. Therefore, an assessment of the effects of mixing and stratification should focus on how the steam concentration and velocity fields are affected by circulation and stratification. Since the evaluation model assumes only free convection inside the containment, the potential benefit of forced convection, when it exists, is neglected. Therefore, the assessment can be further focused on the potential effects of circulation and stratification on steam concentration distributions.

Responses to NRC Questions on Section 9 of WCAP-14407

C #5, #6

Discuss at meeting with NRC.

S #7 (W) will update Table 9-1 to clarify the time phases.

S #8 Table 9-1 was intended to state that for open-ended compartments (e.g. compartments with multiple flow paths which have a circulation pattern) heat sink utilization is not sensitive to stratification. This is discussed in Section 9.3.1.3. Clarification will be provided to Table 9-1.

S #9 Sensitivity cases presented in Section 9.3.1.3 (last paragraph on page 16) and shown in Figures 9-6, 9-7, and 9-8 provide the justification for the utilization of heat sinks. The 3-region CMT stratification calculation considered the bottom one third to be exposed to the air rich concentration. Appendix B will contain information about this calculation.

S #10 "Well-mixed" is a conservative assumption when the PCS is the dominant heat removal mechanism. This is discussed in Section 9.3.2.3.

S #11 The applicability of WGOTHIC for long term containment modeling is discussed in Section 9.3.2 and in References 5 and 6 from Section 9.6.

S #12 Refer to the response to #4 for the definition of important physical quantities. The gradient referred to in Table 9-1 is the steam concentration gradient above- and below-deck for the WGOTHIC analysis. The effect of a bounding gradient or integrated heat sink energy removal is discussed in Section 9.3.2.3 and concluded to be insignificant. Also refer to the response to #106 for Section 9 clarification.

S #13 Qualitative discussions for the DECLG LOCA indicate that the high kinetic energy jet case is non-limiting. A qualitative discussion of the three LOCA break scenarios is provided in Section 9.3.1. A relatively low energy break would result in higher peak containment pressures since the induced circulation patterns would result in reduced utilization in the internal heat sinks.

S #14 Table 9-1 will include break positions and locations considered in the subsequent sections.

S Pg. 2: Section 1.1 Definitions

Additional text will be added to the mixing definition stating that diffusion also contributes to mixing under stratified conditions. Also, the text in the third paragraph in Section 9.1 will be updated to clarify the intent of the statement.

S Pg. 2: Last Paragraph

The comment concerning stagnation in dead-ended compartments is noted. This is discussed in Section 9.3.2.2. The objective here is to define 'segregation' and give a simplified example of how segregation occurs.

C #15, #16, #17

Discuss at meeting with NRC.

S Pg. 3: Section 2.1 LOCA Configuration

(W) will provide reference to the Scaling Analysis report for the Froude numbers and will provide discussion noting that the LST does not cover the blowdown phase of the LOCA.

S #18 (W) will provide a brief discussion of the Froude number and provide a reference to the appropriate section of the Scaling Analysis report.

S #19 Section 10.0 of the Scaling Analysis Report provides Fr number data for the blowdown, transition, and post-blowdown phases for the LOCA transient. Reference to this section will be included.

C #20 Discuss at meeting with NRC.

S #21 The Fr-number range quoted covers the post-blowdown LOCA time phases. See the response to #18 above.

S #22 (W) will provide discussions of the LST methods with respect to LOCA modeling. Use of a diffuser is not intended to provide a "LOCA" model, but rather a known uniform velocity profile. This known quantity can then be used to draw test conclusions through the data reduction.

S #23 The post-blowdown portion of the transient is being addressed with Figure 9-1. See the response to #18 above.

S #24 Since there is no simulated flow path between the open and steam generator compartments in the LST, data for mixing down to the heel of the LST vessel cannot be used to draw conclusions directly for AP600. LST stratification data within the above-deck region has been used to estimate axial density gradients that may occur in the AP600 above-deck region. Based on scaling, the vertical steam density gradient in AP600 would be shallower than that in the LST for the same Froude number. The actual flow path into the AP600 steam generator compartment would allow large scale circulation which would reduce stratification gradients. However, because of the atypicality of the LST for assessing the influence of large scale mixing on stratification gradients, the evaluation model assessment conservatively considered a degree of stratification greater than that observed in the LST. The text in Section 9.2.1 will be revised to include the above discussion.

C #25, #26, #27, #28, #29, #30
Discuss at meeting with NRC.

S Pg. 3 and 4: MSLB Configuration

The test data is provided in Figure 9-1. All conclusions discussed in this section are derived from this figure. However, it is noted that Figure 9-1 is not specifically called out in this section. (W) will add a reference in the text to Figure 9-1 to clarify the discussion.

Responses to NRC Questions on Section 9 of WCAP-14407

S #31 Figure 3.3 of WCAP-14190 shows the steam volumetric flow rate as a function of time for the DECLG break, from which circulation times can be determined for any time of interest. This will be shown for the above-deck region in Appendix C.

S #32 The information in Appendix C will address the entrainment in the above-deck region.

C #33 Discuss at meeting with NRC.

S Pg. 5/6: Mixing and Stratification Assessment for LOCA

The text in this section was updated for the WGOTHIC Applications Report. Table 9-1 will also be edited for further clarification. These changes will provide a better picture of the three LOCA time phases and the actual times (from transient initiation) for each phase.

S #34 Time phases will be made consistent with the final PIRT and Scaling time phases.

S #35 a. Refer to the response to #79.

b. Reference to Section 4 will be provided which details the evaluation model volumes, flow paths, and heat sinks.

S Pg. 6 The CMT room plays an important part in the transient pressure mitigation for several reasons, all of which are discussed later in this report. The CMT room contains a much larger percentage of below deck heat sinks than the accumulator compartments for instance. Even though 40% of the steel and 60% of the concrete heat sinks are not in the CMT room, no other single below-deck compartment contains as many heat sinks. Also, the CMT room is the largest (volume) of the below deck compartments and contains many flow paths. These flow paths mean that the CMT room is of significant importance with respect to both above and below deck circulation patterns. Additional text will be added to this section to clarify the thought.

S #36 For the sensitivity studies in Section 9, modeling of the steel and concrete in the below deck compartments is based on Revision 8 of the General Arrangement Drawings. Text will be revised to clarify that the basis for the sensitivity studies is the evaluation model described in Section 4 Revision 1.

S #37 The CMT room heat sinks are primarily important during the transition phase of the LOCA transient, as are all of the below deck compartments. The time constant for the steel heat sinks is relatively short (< 1200 seconds), while the concrete has a much longer time constant. Figure 9-13 presents heat sink utilization as a function of time, and shows the times which the steel and concrete heat sinks are most utilized.

S Pg. 6: LOCA Break Scenarios

(W) will update Table 9-1 to include information relevant to the three break location scenarios.

S #38 (W) will update Table 9-1 to include clarification of the time phases discussed, the relative importance of each phase, and the important parameters, phenomenon, and conclusions.

Responses to NRC Questions on Section 9 of WCAP-14407

S #39 (W) will provide more information in the section which supports the three jet scenarios discussed in Sections 9.3.1.1 through 9.3.1.3. Also refer to the response to #55.

S #40 The discussion related to whether the jet points upward through the SG compartment vs. outward through the stairwell is qualitative and considers bounding extremes. However, the calculated Froude number (Fr) from the LST data are useful in assessing the influence of high kinetic energy and break elevation, such as delineating between a high energy jet which rises out of the SG compartment relatively undissipated vs. a low energy jet which dissipates in the SG compartment. Section 10 of the Scaling Analysis report discusses the use of the Fr number for characterizing jet plumes in the LST and AP600 DBA models.

S #41 LST data relating to the three jet scenarios are discussed in Section 10.0 of the Scaling Analysis report.

S Fig. 4 displays . . .

The axes on Figure 9-4 will be corrected.

S #42 The code used for the mass flow rate is listed in Section 4, Table 4-107. The final LOCA mass and energy release model will be described in SAR Section 6.2.1.3.2.

S #43 Figure 9-4 is a conservative upper bound and will be revised to be consistent with Section 4 Revision 1.

S #44 Containment back pressure was accounted for by assuming the containment pressure was 45 psig (the design value). This assumption will be discussed in SAR Section 6.2.1.3.2.3.

S The scenario developed . . .

(W) will add more detail to the discussion of the locally dissipated jet and will refer to the WGOTHIC Evaluation Model description in Section 4 for specific room geometry. The estimated SG pressurization can be confirmed for the forced-flow blowdown by a simple nodal network solution.

S #45 Section 4 Figures 4-94, 4-95, 4-96, and 4-97 will be updated to more clearly show the liquid and steam portions of the transient. Post blowdown steam released from the break for this dissipated jet case is assumed to rise up through the SG compartment while the liquid falls.

S #46 The LST does not simulate an AP600 transient as discussed in WCAP-14845, Section 11. Rather, in the context of Section 9, the data is used to assess stratification within the above and below deck regions based on the appropriate scaling of the LST Fr numbers to the AP600. The applicability of specific test to LOCA post blowdown and MSLB are discussed more fully in Section 6.5 of WCAP-14845. References will be added to Section 9.2.

S #47 The CMT room flow calculation will be presented in Appendix A. In addition, WGOTHIC break sensitivities discussed in Section 9.3.2.4 and presented in Figures 9-17, -19, -21 show the calculated circulation rates associated with the CMT room. Reference will be

made to these evaluations.

C #48 Discuss at meeting with NRC.

S #49 The answer to this question is contained in Section 10 of the Scaling Analysis report. As a measure of stability, or lack thereof, the volumetric Fr number can be used to assess vertical density gradients. Fr numbers orders of magnitude greater than unity imply Reynolds number (or kinetic energy) dominated phenomena, while Fr numbers much less than unity imply the Reynolds number is not important for mixing in containment. WCAP-14845, Figure 6-2, shows that LOCA blowdown is sufficiently strong to break up any pre-existing stratification.

S #50 (W) will provide a clearer connection between the jet scenarios discussed in this report and the data in the Scaling Analysis Report. A high-energy jet would vigorously mix the above deck region (as well as the below deck region) because the jet is assumed to be undissipated at the exit of the SG compartment, thus it will entrain a large volume of the above deck atmosphere as it rises from the SG compartment exit. Additional information provided in response to comment #39 will further help to resolve this comment.

S #51 Section 10 of the Scaling Analysis Report forms the basis for considering the effects of mixing and stratification. Reference to this report will be added.

S #52 Section 10 of the Scaling Analysis Report provides the basis for this scenario, which is based on the volumetric Fr number. Reference to this report will be added.

S #53 Refer to the response to #39.

S #54 Refer to the response to #50.

S #55 The qualitative discussion concerning containment circulation patterns for various break scenarios are confirmed via break location sensitivity studies documented in Section 9.3.2.4. (W) will provide a summary of a simplistic jet-pump calculation representing the broken-loop SG compartment. This calculation will provide quantitative supporting evidence for below deck circulation associated with the jet-up scenario.

S #56 Section 9.3.2.4 and Figures 9-17, -19, -21 discuss and present the calculated circulation patterns for a spectrum of break scenarios at the time of peak containment pressure. Reference will be provided to this section of the report.

S #57 Section 9.3.2.4 will be updated to include circulation pattern data at significant transient times (i.e. end of blowdown, beginning of the long term cooling phase, for the base case transient).

T #58 Due to the extensive structure in the SG compartment which provides support for not only the Steam Generator, but also the hot and cold leg piping, the ADS piping, and maintenance manways it is highly doubtful that a double-ended guillotine break of one of the RCS cold leg pipes would result in the pipe deflecting in such a manner that break fluid would have an unobstructed pathway into the stairwell. However, the DBA is not a mechanistic model. Rather the DBA considers extreme circumstances in order to bound

the range of accident possibilities. Therefore, the jet into the stairwell scenario has been postulated as one of the extreme cases and is evaluated in this analysis.

S #59 Descriptions of the AP600 compartments and vent paths as modeled in the WGOTHIC evaluation model are provided in Section 4. Reference will be added.

S #60 Post-blowdown circulation patterns are given for all three break scenarios in Figures 9-17, -19, and -21. These figures not only identify the circulation flows, but also the steam concentrations in the major below deck volumes.

S #61 Refer to the response to #58.

S #62 (W) will provide an additional level of detail concerning these calculations in Appendix A.

S #63 The information necessary to calculate the mixing time constant is provided in Section 9.3.1.3, and is simply the CMT room volume divided by the entrainment flow rate in minutes. This number represent the time it takes to cycle the CMT room volume. Appendix A will contain information on the CMT room calculation.

S #64 A description of the AP600 compartments and vent paths as modeled in the WGOTHIC evaluation model is provided in Section 4. Reference will be added. In addition, Appendix A will contain information on the CMT room calculation.

S #65 Refer to the response to #62.

S #66 Refer to the response to #63

S #G1 through #G11

See the responses for general questions #A1 through #A11

T #72 WCAP-14236, Section 3.9, shows that the nominal PCS correlations provide a reasonable prediction of mass transfer for free convection, or low Froude (Fr), conditions over the whole range of steam concentrations, and Section 4.3 notes that the correlation is slightly conservative with a 0.983 mean. It is desirable to assess the AP600 physics using the best available correlation. It is not believed to be useful to perform this calculation using a correlation which may have a bias which varies over the steam concentration range when evaluating AP600 physics.

S #91 a. The evaluation model described in Section 4 was used for the results in Figure 9-13.

b. Yes, upward facing surfaces in all compartments have been insulated and condensation for dead-end compartment heat sinks during the post-blowdown phases is not included in the evaluation model. When applicable for a given node, this is discussed in the "Special Modeling Assumptions" subsections in Section 4 for each compartment. In the model, the condensation heat transfer is multiplied by zero, however, convective and radiant heat transfer is still calculated by the code. Figure 10-4 of WCAP-14845 shows that this is a negligible contributor to the overall heat transfer. A similar figure will replace Figure 9-13 based on the evaluation model of Section 4 Revision 1.

Responses to NRC Questions on Section 9 of WCAP-14407

References

1. NSD-NRC-96-4763, Docket No.: STN-52-003, July 1, 1996,"Assessment of Mixing and Stratification Effects on AP600 Containment."

2. NRC informal review questions - Part 1 from D. Jackson (NRC) to J. Butler (Westinghouse), 8-22-96; Part 2 from D. Jackson (NRC) to J. Butler (Westinghouse), 11-8-96.

3. WCAP-14407, "WGOTHIC Application to AP600," September 1996.

March 21, 1997

Subject: Informal Transmittal of Information on
- WCAP-14407 Ch 7 RAIs 480.871 - 480.909

To: Ed Throm Fax: 301-415-3577 (41 pages)

cc: Jim Gresham Dick Haessler
Brian McIntyre Tim Andreychek

Attached are proposed approaches to respond to the subject NRC review questions. The ones marked with a "T" are those which Westinghouse recommends to have technical discussions during the 3/25/97 PCS DBA meeting.

It should be noted that WCAP-14407 Section 4, WCAP-14812, and WCAP-14845 have been issued subsequent to the issuance of NRC review comments on the preliminary draft Section 9. The attached responses assume the reviewers have access to all three WCAPs at this time.

Joel Woodcock

Joel Woodcock

prelim.wp

RAI Binning Code

C ⇒ Question not understood; clarification by NRC is requested

S ⇒ Straightforward response offered

T ⇒ Technical discussion with NRC requested

AP600 - REQUEST FOR ADDITIONAL INFORMATION ON WCAP-14407, SECTION 7, "METHOD FOR CALCULATING THE PCS FILM COVERAGE INPUT FOR THE AP600 CONTAINMENT DBA EVALUATION MODEL"

page vii

WCAP-14407 states: "The PCS test data was examined with respect to the parameters that determined coverage. The range of the test data was compared with the estimated AP600 range during DBA. The test data was found to be acceptable for application to the development of a film stability model."

This statement implies that Westinghouse's criteria for acceptability was limited to tests that spanned the expected ranges of the key parameters. However, the "goodness" of a test should be assessed as well as its range. Criteria such as consistency and repeatability of the measurement technique, standard error and variance of data obtained, and an assessment of whether the test procedure had inherent biases of large measurement errors are also essential to judge the acceptability of test results.

S 480.871 Please expand the discussion presented to include the criteria Westinghouse used to judge the "goodness" of the test data for each of the tests whose results are summarized in Subsection 7.2.

RESPONSE:

Two actions are taken in response to this item;

1) The identified text in the Executive Summary will be amended to more clearly state the basis for including data in the evaluation of water coverage.

2) The "goodness" of the test data for each test summarized in Subsection 7.2 is considered a test acceptance question and will be addressed in a separate letter to NRC. Response to this item will not be incorporated into a revision of Section 7.

AP600 - REQUEST FOR ADDITIONAL INFORMATION ON WCAP-14407, SECTION 7, "METHOD FOR CALCULATING THE PCS FILM COVERAGE INPUT FOR THE AP600 CONTAINMENT DBA EVALUATION MODEL"

WCAP-14407 states: "A film stability model, based on a modified form of the Zuber-Staub model for determining dry spot stability, was developed to determine a maximum value for the minimum stable film flow rate. The film stability model was compared with PCS test data for both evaporating and subcooled films. This model bounds the test data."

The Zuber-Staub model (Ref. Zuber and Staub, Int. J. Heat Mass Transfer, 9 pp. 897, 1966) is theoretical. While the use of the Zuber-Staub model may have value as a means to identify the important forces and parameters that determine film flow stability, its simplifying assumptions (e.g., that the liquid film is always saturated, the flow is always laminar, and the surface is free of imperfections) do not represent the physical conditions expected for the AP600 PCS film. Westinghouse has not presented separate effects test data to support the use of the Zuber-Staub model as a quantitative predictor. It is not enough to bound the limited integral test data. Westinghouse must show that the minimum stable film thickness obtained with the modified Zuber-Staub model is conservatively predicted under the worst expected AP600 operation conditions.

T

480.872 Please provide a discussion and summary plots which clearly show the expected conservatism in the Westinghouse water coverage model, as applied to the worst combination of operating conditions for AP600. Include the following parameters:

- Location (dome, sidewall)
- Heat flux
- PCS liquid subcooling
- Flow rate/regime (laminar, wavy laminar, turbulent)
- Outside air and shell/baffle/shield building temperature
- Coating degradation and surface decontamination
- Surface roughness and plate misalignment
- Contact angle uncertainty (include aging)

RESPONSE:

Westinghouse has bounded test data which accounts for the following parameters;

- Location (dome, sidewall)
- Heat flux
- PCS liquid subcooling
- Flow rate/regime (laminar, wavy laminar, turbulent)
- Outside air and shell/baffle/shield building temperature
- Plate misalignment

Coating degradation will be controlled under the AP600 inspection and maintenance program. Preliminary examination suggest wetting characteristics of the surface improve with aging. Westinghouse requests further clarification and discussion on this item.

AP600 - REQUEST FOR ADDITIONAL INFORMATION ON WCAP-14407, SECTION 7, "METHOD FOR CALCULATING THE PCS FILM COVERAGE INPUT FOR THE AP600 CONTAINMENT DBA EVALUATION MODEL"

480.873 Please provide details of the calculations Westinghouse performed to assure that the evaporation-limited PCS flow is equivalent to using the actual PCS film flow with a time and elevation dependent coverage fraction.

RESPONSE:

S

Westinghouse will include calculations in the amended Section 7.

AP600 - REQUEST FOR ADDITIONAL INFORMATION ON WCAP-14407, SECTION 7, "METHOD FOR CALCULATING THE PCS FILM COVERAGE INPUT FOR THE AP600 CONTAINMENT DBA EVALUATION MODEL"

480.874 Please provide the mathematical definition of "evaporation-limited" PCS flow rate.

RESPONSE:

S

The mathematical definition of "evaporation-limited" PCS flow is;

$$\dot{m}_{APPLIED} = \dot{m}_{PCS\ ACTUAL} - \dot{m}_{PREDICTED\ RUNOFF}$$

This definition limits the applied PCS flow rate in the evaporation-limited model to be that which is predicted to be evaporated. Westinghouse will amend the text of Section 7 to include this definition statement

The last sentence of the last paragraph states: "As expected, when the actual PCS flow rate was used (not the evaporation limited flow rate), the calculated peak pressure increased as the coverage area was decreased."

480.875 Provide details of the WGOTHIC calculations and results which support this statement. Please clarify the apparent contradiction between this statement and the previous statement that the "evaporation-limited PCS flow is equivalent to using the actual PCS film flow with a time and elevation dependent coverage fraction."

RESPONSE:

S

The last sentence in the last paragraph is incorrect. Westinghouse will correct the sentence in the amended Section 7.

AP600 - REQUEST FOR ADDITIONAL INFORMATION ON WCAP-14407, SECTION 7, "METHOD FOR CALCULATING THE PCS FILM COVERAGE INPUT FOR THE AP600 CONTAINMENT DBA EVALUATION MODEL"

pg 7-1

480.876 Please provide the basis for the estimate that 80 per cent of the energy removal is through evaporation. Identify the events and the phases for which this condition is estimated to occur.

RESPONSE:

S

The actual per cent of the energy removal is through evaporation varies over time with changes in the PCS flow rate and the decay heat generation rate. The 80% value is based on steady state large scale test observations. Westinghouse will clarify the text of the amended Section 7 to indicate the basis for the partioning of heat removed by the various modes of heat transport.

AP600 - REQUEST FOR ADDITIONAL INFORMATION ON WCAP-14407, SECTION 7, "METHOD FOR CALCULATING THE PCS FILM COVERAGE INPUT FOR THE AP600 CONTAINMENT DBA EVALUATION MODEL"

WCAP-14407 states in the second paragraph: "The rapid heating of the cold water as it spread out onto the surface of the dome causes the film to become unstable and break down into multiple streams. In addition, surface irregularities caused by plate misalignment during welding also help to break down the film at the top of the dome." This discussion highlights the importance of the heat flux on subcooled film stability.

T

480.877 How does Westinghouse justify the values, based on data taken from the unheated Water Distribution Tests, for the water coverage fractions for the top two AP600 model "climes" that represent the dome region, when the AP600 dome will be heated?

RESPONSE:

The water coverage fractions used in WCAP-14407 are based on a PCS flow of 220 gpm. The design calls for a PCS flow of 440 gpm. The larger AP600 PCS flow provides for larger water coverage fractions than reported in WCAP-14407. Westinghouse will add explanatory text to the amended Section 7 to note the increase in PCS flow, and therefore the water coverage fractions used in the analyses are bounded by those of the AP600 plant.

AP600 - REQUEST FOR ADDITIONAL INFORMATION ON WCAP-14407, SECTION 7, "METHOD FOR CALCULATING THE PCS FILM COVERAGE INPUT FOR THE AP600 CONTAINMENT DBA EVALUATION MODEL"

S 480.878 What is meant by "help break down the film?" Is this a benefit?

RESPONSE:

1.) Westinghouse will rephrase and clarify the text in the amended Section 7.

2.) Breaking down the film is not a benefit.

AP600 - REQUEST FOR ADDITIONAL INFORMATION ON WCAP-14407, SECTION 7, "METHOD FOR CALCULATING THE PCS FILM COVERAGE INPUT FOR THE AP600 CONTAINMENT DBA EVALUATION MODEL"

The third paragraph states: "After some minimum value of film thickness is reached, further evaporation causes the film width to decrease." Westinghouse assumes that the evaporation will cause the film width (wetted perimeter) to decrease smoothly in an exponential fashion. Zuber and Staub (Ref. Zuber and Staub, Int. J. Heat Mass Transfer, 9 pp 897, 1966.) assumed that once the minimum film stability criterion is exceeded, the film would split into fingers around dry spots.

T 480.879 How does the data show that a minimum film thickness has been reached? The above statements indicate that the flow could remain as thick rivulets.

RESPONSE:

Film thickness is not a measured quantity. Film coverage is measured. Knowing the applied flow and the film coverage, a film thickness is inferred.

No change to the text of Section 7 will be implemented in response to this RAI.

AP600 - REQUEST FOR ADDITIONAL INFORMATION ON WCAP-14407, SECTION 7, "METHOD FOR CALCULATING THE PCS FILM COVERAGE INPUT FOR THE AP600 CONTAINMENT DBA EVALUATION MODEL"

480.880 Please provide justification for the assumption that evaporation will cause the film width (wetted perimeter) to decrease smoothly in an exponential fashion.

RESPONSE:

T

From testing, it was observed that the film width decrease is approximately linear. The assumption of constant film flow rate, Γ_{MIN}, results in an exponential decrease in film width as the film evaporates. The assumption of constant Γ_{MIN} does not match the test observations and, because it predicts less coverage than observed, is conservative. Westinghouse will incorporate a statement of explanation regarding the use of the exponentially decreasing film width to the amended Section 7.

AP600 - REQUEST FOR ADDITIONAL INFORMATION ON WCAP-14407, SECTION 7, "METHOD FOR CALCULATING THE PCS FILM COVERAGE INPUT FOR THE AP600 CONTAINMENT DBA EVALUATION MODEL"

480.881 In addition to plate misalignment during welding, other surface irregularities can affect the stability of the liquid film. Please discuss Westinghouse requirements for maximum allowable surface irregularities and how conformance to these requirements will be demonstrated during the plant lifetime.

RESPONSE:

T

The fabrication and erection of the AP600 containment shell will be accomplished to ASME code requirements. Inspections will be performed during the fabrication and erection process to assure code requirements are satisfied. The fabrication, erection and inspection of the containment shell to ASME code requirements will assure surface irregularities are within prescribed limits at the time of construction. Inspection and maintenance of coatings on the outside surface of the containment shell is addressed in the AP600 inspection and maintenance plan.

No change to the text of Section 7 will be implemented in response to this RAI.

480.882 At what point in the calculation is the runoff flow subtracted? Is this flow properly considered in the PCS tank level and pressure head calculation? How is the sensible heat from the subtracted runoff flow treated? Does the runoff flow subtraction procedure preserve the distance at which the remaining PCS coolant flow reaches saturation?

RESPONSE:

T

Westinghouse will amend the text of Section 7 to include a response to the point of this RAI.

AP600 - REQUEST FOR ADDITIONAL INFORMATION ON WCAP-14407, SECTION 7, "METHOD FOR CALCULATING THE PCS FILM COVERAGE INPUT FOR THE AP600 CONTAINMENT DBA EVALUATION MODEL"

pg. 7-3:

480.883 How did Westinghouse determine that the zinc coated surface was indeed prototypical? For each of the tests identified in Section 7.2, please describe the quantification performed for surface coating thickness, surface roughness, irregularity size, etc. Identify whether each test was performed with a freshly coated or an aged surface. Describe how the surfaces were aged. For each test sample with an aged surface, estimate the simulated age of the surface in terms of service years.

RESPONSE:

T

This question is determined to be a test description/acceptance question. Westinghouse will respond to this question by separate letter. No revision to Section 7.0 is planned as a result of responding to this question.

AP600 - REQUEST FOR ADDITIONAL INFORMATION ON WCAP-14407, SECTION 7, "METHOD FOR CALCULATING THE PCS FILM COVERAGE INPUT FOR THE AP600 CONTAINMENT DBA EVALUATION MODEL"

The second paragraph of Water Film Formation Test Section (Section 7.2.1) states: "The film thickness was not uniform near the point of application; it was thinnest just below the application point and thicker on both sides. The film stripe continued to spread (more slowly as the surface became more vertical) and a very thin, wet region was created at the edges as the film traveled downwards."

480.884 This observed behavior appears to contradict the Westinghouse assumption that the film has a constant cross-sectional thickness as the water travels down the PCS shell. If the liquid film has a non-uniform cross sectional area, the equations on pages 7-27 through 7-30 may underpredict the amount of runoff. Please explain.

RESPONSE:

S

The film is uniform, wary laminar below the point of application. Westinghouse will clarify the description of the observed behavior of the water coverage near the point of application in the amended Section 7.

AP600 - REQUEST FOR ADDITIONAL INFORMATION ON WCAP-14407,
SECTION 7, "METHOD FOR CALCULATING THE PCS FILM COVERAGE INPUT FOR THE AP600 CONTAINMENT DBA EVALUATION MODEL"

480.885 Please add an equation defining the Reynolds and Marangoni numbers as used in Table 7-1.

RESPONSE:

S

Westinghouse will add the requested equations to the text for Table 7-1 in the amended Section 7.0.

480.886 In Table 7-1, the "LST Test Data Ranges" box refers to "Peak Heat Flux at Bottom." To what "Bottom" does this refer?

RESPONSE:

S

Westinghouse will amend Table 7-1 to clarify the definition of "bottom" in the amended Section 7.

T 480.887 Please explain how the water coverage fractions above the first weir, between the weirs and from the second weir down (refer to Section 7.4.2, page 7-33) were obtained from the "measured" Water D ibution Tests, phase 3 data. As shown in Table 7-2, wetted perimeter measurements were taken just above the second weir and after the springline. From this data, it is difficult to infer the average data above the first weir, between the two weirs and between the springline with any accuracy.

RESPONSE:

T The water coverage fractions reported Section 7.4.2 were developed from videotape records of the 220 gpm equivalent flow Water Distribution Tests as follows;

Location	Coverage	Method of Determining
▸ Between the dome top and 1st weir	25 %	Visual Inspection
▸ 1st weir to 2nd weir	65 %	Visual inspection and calculated using an averaged elliptical surface area
▸ 2nd weir and below	90 %	Measured

Westinghouse will include an explanation of how the water coverage fractions are determined in the amended Section 7.

AP600 - REQUEST FOR ADDITIONAL INFORMATION ON WCAP-14407, SECTION 7, "METHOD FOR CALCULATING THE PCS FILM COVERAGE INPUT FOR THE AP600 CONTAINMENT DBA EVALUATION MODEL"

T

480.888 Based on the techniques used to determine the water coverage fractions, estimate the uncertainty in the values and the impact of including these uncertainties when evaluating the PCS performance. The data provided in Table 7-2 suggests that the measurement accuracy is ± 1% in the coverage fraction.

RESPONSE:

As noted in the response to RAI 480.877, the water coverage fractions used in WCAP-14407 are based on a PCS flow of 220 gpm. The design calls for a PCS flow of 440 gpm. The larger AP600 PCS flow provides for larger water coverage fractions than reported in WCAP-14407. It is believed that the increase in flow overwhelms the measurement error. Westinghouse will add explanatory text to the amended Section 7 to note the increase in PCS flow, and therefore the water coverage fractions used for the analyses are bounded by those that would exist for the AP600 plant.

S 480.889 Table 7-4 lists the film inlet temperature for case 107A5U as 800° F. Should this be 80° F.

RESPONSE:

The film inlet temperature for case 107A5U listed in Table 7-4 is incorrectly listed as 800° F. The correct value of the inlet is 80° F. Table 7-4 will be amended to list the correct value in the amended Section 7.

S

480.890 In discussing flow oscillations, WCAP-14407 states that the flow oscillations had a small effect on the coverage. In the absence of test data without flow oscillations, how was this statement justified? What analysis had been performed to determine that the flow oscillations did not fundamentally alter the film stability and water coverage measurements?

RESPONSE:

An analysis of the maximum and minimum flow rates resulting from the oscillations is presented in Appendix A of Section 7. Observations and the analysis performed support the following conclusions;

- Water coverage was observed to have small changes at the bottom of the test article, and,
- Film stability was not affected.

These conclusions are stated in Appendix A of Section 7.

No change to the text of Section 7 will be implemented in response to this RAI.

C

480.891 The second bulleted paragraph states that a relatively uniform wall temperature and heat flux was maintained over the evaporating surface. However, a higher than proportional heat transfer rate is experienced on the dome, as discussed in the prior paragraph. Presumably, this is caused by sensible heating of the PCS film flow. How is the non-uniform heat flux distribution factored into the water coverage area calculation.

RESPONSE:

The WGOTHIC code does not calculated water coverage area; rather, it calculates a point of dryout on a fixed area that is an input (clime node) to the code. The calculation uses heat flux as a function of elevation. Westinghouse will modify the text in the amended Section 7.0 to clarify how heat fluxes vary and are used in the calculations.

AP600 - REQUEST FOR ADDITIONAL INFORMATION ON WCAP-14407,
SECTION 7, "METHOD FOR CALCULATING THE PCS FILM COVERAGE INPUT FOR THE AP600 CONTAINMENT DBA EVALUATION MODEL"

S 480.892 Why was the data for tests 216.1A and 216.1B omitted from Table 7-5, and figures 7.A-5 and 7.A-6? Please add these data points to the Table and Figures.

RESPONSE:

Data from the forced coverage tests were not used. Other forced coverage cases not included are 207.1, 207.3 and 208.1. This is stated on page 7-14.

No change to the text of Section 7 will be implemented in response to this RAI.

S 480.893 In the last paragraph of page 7-18 and in Table 7-6, the minimum applied film temperature is given as 40° F. AP600 technical specifications specify 50° F. Please clarify.

RESPONSE:

The current AP600 technical specifications identify a minimum PCS temperature of 40° F.

No change to the text of Section 7 will be implemented in response to this RAI.

AP600 - REQUEST FOR ADDITIONAL INFORMATION ON WCAP-14407, SECTION 7, "METHOD FOR CALCULATING THE PCS FILM COVERAGE INPUT FOR THE AP600 CONTAINMENT DBA EVALUATION MODEL"

S 480.894 Please add the values for "Runoff Film Temperature," "Bottom Sidewall Film Flow Rate" and "Marangoni Number" to Table 7-6 for AP600.

RESPONSE:

Westinghouse will add estimates of the requested parameters to Table 7-6 in the amended Section 7.

S

480.895 The third paragraph states that "about 10 minutes" is required to fill the water distribution system and reach steady-state for a 220 gpm flow. Previous documents used 11 minutes. Both estimates are based on video tape data from the Water Distribution Tests. Please clarify the reason for the change.

RESPONSE:

No change in time to fill and reach steady-state flow conditions was intended by the usage of "about 10 minutes" instead of 11 minutes; they were taken to be similar times inferred from a video record of the test. Westinghouse will modify the text of the amended Section 7 to be consistent with terminology previously used.

AP600 - REQUEST FOR ADDITIONAL INFORMATION ON WCAP-14407, SECTION 7, "METHOD FOR CALCULATING THE PCS FILM COVERAGE INPUT FOR THE AP600 CONTAINMENT DBA EVALUATION MODEL"

S

480.896 Please explain why 250° F was selected for the asymptotic internal containment temperature, T_∞, when 260-265° F is more appropriate based on the containment atmosphere curve shown in Figure 7-4 on page 7-23.

RESPONSE:

The purpose of the calculation was to estimate the time required to heat the exterior surface of the dry shell to the boiling point. Several approximations were made in performing the calculation, one was an average containment atmosphere temperature of 250° F over the time period of interest. Without integrating the area under the curve of Figure 7-4, the use of 250° F is judged to be a reasonable first approximation to the average containment temperature over the time period of interest.

Based on current mass and energy results, the calculation results shown in Figure 7-4 will be reviewed and recalculated, as appropriate, for inclusion in the amended Section 7.

AP600 - REQUEST FOR ADDITIONAL INFORMATION ON WCAP-14407, SECTION 7, "METHOD FOR CALCULATING THE PCS FILM COVERAGE INPUT FOR THE AP600 CONTAINMENT DBA EVALUATION MODEL"

S

480.397 Please justify the value of 75 Btu/hr-ft^2-°F for the post-LOCA internal atmosphere to PCS shell energy transfer coefficient.

RESPONSE:

The value of 75 Btu/hr-ft^2-°F for the post-LOCA internal atmosphere to PCS shell energy transfer coefficient is based on inspection of previous WGOTHIC calculations. Westinghouse will modify the text associated with identifying this value to note the origin of this approximation in the amended Section 7.

AP600 - REQUEST FOR ADDITIONAL INFORMATION ON WCAP-14407, SECTION 7, "METHOD FOR CALCULATING THE PCS FILM COVERAGE INPUT FOR THE AP600 CONTAINMENT DBA EVALUATION MODEL"

pg. 7-27 and 7.A-12:

S

480.898 In the formula for wetted perimeter, r_{WETTED}, on pg. 7-27 (or WDL on 7.A-12), the predicted circumference, r_{SPLIT}, on pg. 7-27 (or CI on 7.A-12) is averaged with the circumference at the springline. Please explain the reason for this extra step in the calculation procedure, which always increases the predicted wetter perimeter.

RESPONSE:

The formula for wetter perimeter as given on pages 7-27 and 7.A-12 is incorrect. Westinghouse will correct the error in the amended Section 7. Westinghouse will also review and correct the calculations with regard to bounding the test data that may have been affected by this error.

AP600 - REQUEST FOR ADDITIONAL INFORMATION ON WCAP-14407, SECTION 7, "METHOD FOR CALCULATING THE PCS FILM COVERAGE INPUT FOR THE AP600 CONTAINMENT DBA EVALUATION MODEL"

S 480.899 Describe the model used for the Section 7 study: number of nodes, flow paths, climes, WGOTHIC computer code version used, etc. If the computer program version differs from WGOTHIC 4.0 version to be used for the final SSAR analyses, discuss the differences and their impact on the results provided. With respect to the model described in Section 4, discuss modeling differences and their impact on the results provided.

RESPONSE:

This comment will be addressed in the amending of Section 7.0. The model and version of WGOTHIC used will be described; differences, if any, between the those used for Section 4.0 and Section 7.0 will be identified and the impact of those differences on calculated results evaluated.

AP600 - REQUEST FOR ADDITIONAL INFORMATION ON WCAP-14407,
SECTION 7, "METHOD FOR CALCULATING THE PCS FILM COVERAGE INPUT FOR THE AP600 CONTAINMENT DBA EVALUATION MODEL"

480.900 What driving forcing functions (break mass flow rate, energy addition) were used to obtain the computational results? What blowdown computer program was used to predict break flow and associated energy? Provide a plot of the mass and energy profiles used, or a reference location.

RESPONSE:

Westinghouse will modify the text of the amended Section 7 to identify the mass and energy forcing functions used in the calculation reported in this section.

AP600 - REQUEST FOR ADDITIONAL INFORMATION ON WCAP-14407, SECTION 7, "METHOD FOR CALCULATING THE PCS FILM COVERAGE INPUT FOR THE AP600 CONTAINMENT DBA EVALUATION MODEL"

S 480.901 The sensitivity calculation for water overage location varied the coverage are as well as the location. Please reperform the sensitivity case for water coverage location (dome and sidewall coverage cases only) preserving the coverage area (e.g., 36% coverage for both cases). Please extend the calculations to show results (Figure 7-8) to 24 hours.

RESPONSE:

Westinghouse will perform the requested calculations and provide a summary of the results in the amended Section 7.

AP600 - REQUEST FOR ADDITIONAL INFORMATION ON WCAP-14407, SECTION 7, "METHOD FOR CALCULATING THE PCS FILM COVERAGE INPUT FOR THE AP600 CONTAINMENT DBA EVALUATION MODEL"

T

480.902 Section 7.5 includes four sensitivity studies. However, none of these calculations show the sensitivity of the peak pressure to Gamma min (Γ_{MIN}) - the minimum film thickness. Please add this sensitivity calculation.

RESPONSE:

The value of the minimum stable film flow rate, Γ_{MIN}, used in the calculation is a bounding value that minimizes the water coverage area. The use of larger values of Γ_{MIN} only further bound (adds additional conservatism to an already conservative or bounding assumption. The use of small values of Γ_{MIN} would increase the water coverage area and provide for increased heat removal and lower containment pressures.

Westinghouse will add text explaining the use of a bounding Γ_{MIN} value that provides for a conservatively small water coverage area to the amendment of Section 7.

AP600 - REQUEST FOR ADDITIONAL INFORMATION ON WCAP-14407, SECTION 7, "METHOD FOR CALCULATING THE PCS FILM COVERAGE INPUT FOR THE AP600 CONTAINMENT DBA EVALUATION MODEL"

pg 7-46:

S 480.903 Subsection 7.5.4 states that the base case assumed delay time is 600 seconds for a 220 gpm flow rate. This is inconsistent with the 440 gpm flow rate quoted in Sections 7.2 and 7.3 (see pages 7-14 and 7-20). Please clarify Westinghouse's position of the PCS design flow, and correct the text accordingly.

RESPONSE:

The use of a delay of 600 seconds is conservative for the 440 gpm design flow rate. Westinghouse will clarify the text to state that this is conservative in the amendment to Section 7.

AP600 - REQUEST FOR ADDITIONAL INFORMATION ON WCAP-14407, SECTION 7, "METHOD FOR CALCULATING THE PCS FILM COVERAGE INPUT FOR THE AP600 CONTAINMENT DBA EVALUATION MODEL"

Ponter, et.al., (Ponter, Davies, Beaton and Ross, Int. J. Heat Mass Transfer, 10 pp. 1633, 1967) stated, when discussing the poor agreement of experimental data to the Zuber-Staub model that: "This is not surprising when one compares the actual dynamics of the system with the assumptions made in the model. This was developed for laminar flow conditions, assuming a parabolic velocity profile. Above a Reynolds number (Re) of 20, surface waves are apparent which between the Re range of 300 to 1120, cause liquid circulation sufficient to induce mixing. In this range the simple models will not simulate actual flow conditions."

T 480.904 The expected range of Reynolds numbers for the AP600 side wall films is 0 to 2900. Please comment.

RESPONSE:

The Zuber-Staub model is used to bound the data collected for the AP600 surface.

No change to the text of Section 7 will be implemented in response to this RAI.

AP600 - REQUEST FOR ADDITIONAL INFORMATION ON WCAP-14407, SECTION 7, "METHOD FOR CALCULATING THE PCS FILM COVERAGE INPUT FOR THE AP600 CONTAINMENT DBA EVALUATION MODEL"

pg 7.A-5:

T

480.905 The advancing contact angle value was selected for the evaluation model, "because it is the upper bound of the measured, steady-state values for a heated, weathered surface." This may be the maximum value for wetted surfaces, which are not expected to exceed 180° F. However, the dry strips, which form parallel to the wet strips, are predicted to reach temperatures of 260° F. How can the Zuber-Staub model with this contact angle be used to predict rewet?

RESPONSE:

The Zuber-Staub model is used to bound the data for the AP600 surface.

No change to the text of Section 7 will be implemented in response to this RAI.

AP600 - REQUEST FOR ADDITIONAL INFORMATION ON WCAP-14407, SECTION 7, "METHOD FOR CALCULATING THE PCS FILM COVERAGE INPUT FOR THE AP600 CONTAINMENT DBA EVALUATION MODEL"

S

480.906 C., page 7.A-7, fourth paragraph, should the reference be to Section 7.A-6, not Section A.6?

RESPONSE:

The reference to Section A.6 on page 7.A-7, fourth paragraph, is incorrect. The correct reference should be to Section 7.A-6. Westinghouse will correct this typographical error in amending Section 7.

pg. 7.A-8

S 480.907 Please uniquely identify each data point on Figure 7.A-1 so the test and test conditions can be determined.

RESPONSE:

In amending Section 7.0, Westinghouse will add to the discussion of Figure 7.A-1 a table identifying test number and conditions associated with the data plotted in Figure 7.A-1.

pg. 7.A-9

T

480.908 Please uniquely identify the error ranges given in Figure 7.A-2 so the test and test conditions can be identified in Table 7-5 (pg. 7-15). Please explain how the "cold" (80° F) and "hot" (180° F) estimated film breakdown data were obtained. Please provide a single figure which combines the data shown in Figures 7.A-2 and 7.A-3 in units of lbm/hr-ft instead of Reynolds number. This figure should reflect the expected breakdown temperatures and Reynolds numbers for the AP600. Please comment on how this figure supports the Westinghouse position.

RESPONSE:

In the amended Section 7, Westinghouse will do the following;

- Add a table listing the bounding values plotted in Figures 7.A-2 and 7.A-3
- Provide text describing how the data were obtained, and,
- Provide text describing how the figure supports the Westinghouse position.

T

480.909 Describe how the circumferential average heat flux value is determined, and what is the uncertainty in this value? Is there a correlation between the local (instantaneous) exit Reynolds number and the local heat flux?

RESPONSE:

1.) The equation used to define circumferential average heat flux is;

$$\bar{h}_{CIRCUMFERENCE} = \frac{\sum h_{WETTED,\,i}}{n}$$

where

$h_{WETTED,\,i}$ is the wetted surface heat flux at an elevation

n is the number of wetted surfaces (stripes) at an elevation

Westinghouse will add this definition in the amended Section 7.

2.) Westinghouse has evaluated the uncertainty associated with this number and will provide a discussion in the amended Section 7.

3.) Westinghouse has not evaluated if there is a correlation between the local (instantaneous) exit Reynolds number and the local heat flux.

3/20/97

Bill - This is an advance copy of the letter. It may look a little different when signed but the attached responses will not change.

Brian is out this afternoon and I didn't want to wait this any longer.

Robin

p1/28

Robin - copy for informal insp file.

Westinghouse Electric Corporation
Energy Systems

Box 355
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NSD-NRC-97-5029
DCP/NRC0776
Docket No.: STN-52-003

March 19, 1997

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

ATTENTION: T. R. Quay

SUBJECT: RESPONSE TO SOME OF THE TECHNICAL SPECIFICATION COMMENTS FROM NRC LETTER OF 12/24/96

References:

1. Letter from NRC to Westinghouse (Huffman to Liparulo), Initial Comments on the AP600 Revised Technical Specifications (TS), dated 12/24/96.

2. Letter from Westinghouse to NRC, NSD-NRC-97-4984 (DCP/NRC0739), Response to NRC Question on AP600 Technical Specification LCO 3.0.3, dated 2/13/97.

3. Letter from Westinghouse to NRC, NSD-NRC-97-5001 (DCP/NRC0753), Response to Some of the Technical Specification Comments from NRC Letter of 12/24/96, dated 3/12/97.

4. Letter from Westinghouse to NRC, NSD-NRC-97-5015 (DCP/NRC0763), Schedule for AP600 Technical Specifications Comment Resolution Meetings, dated 3/11/97.

5. Letter from Westinghouse to NRC, NSD-NRC-97-5030 (DCP/NRC-0777), Revision to Responses to NRC Question on AP600 Technical Specification LCO 3.0.3, Dated 3/19/97.

Dear Mr. Quay:

Reference 1 provided 51 questions related to the AP600 Technical Specifications. The first comment was responded to in Reference 2 and Westinghouse provided a revised response to that comment in Reference 5.

The action to respond to Reference 1 comments 2 through 51 was logged as open item tracking system (OITS) item 4970. Reference 3 provided responses to comments 2, 3, 4, 5, 6, 8, 10, 19, 24, 25, 30, 39, 40, 41, 45, 49, and 50. Attached, as further completion of OITS item 4970, are responses to the following Reference 1 comments:

Comment	Topic
9	LCO 3.4.3, RCS Pressure and Temperature (P/T) Limits
11	LCO 3.4.6, Pressurizer
13	LCO 3.4.12, Automatic Depressurization System - Operating
14	LCO 3.4.13, Automatic Depressurization System - Shutdown, RCS Intact LCO 3.4.14, Automatic Depressurization System - Shutdown, RCS Open
15	LCO 3.5.1, Accumulators
16	LCO 3.5.4, Passive Residual Heat Removal Heat Exchanger - Operating
17	LCO 3.5.6, In-Containment Refueling Water Storage Tank - Operating
18	LCO 3.5.7, IRWST - Shutdown, RCS Inventory High LCO 3.5.8, IRWST - Shutdown, RCS Inventory Low
20	Section 5.6.5, Core Operating Limits Report (COLR)
21	LCO 3.4.12, Automatic Depressurization System LCO 3.5.6, In-containment Refueling Water Storage Tank
22	LCO 3.4.12, Automatic Depressurization System
26	Section 5.5.6, Secondary Water Chemistry Program
27	B 3.7.1, Main Steam Safety Valve, Mode 4
28	LCO 3.7.1, Action B.2, Mode 4
29	Draft TS 3.7.8, Secondary Coolant Leakage
31	LCO 3.7.6, Main Control Room Habitability System (VES)
32	LCO 3.7.3, Action C, Main Feedwater Isolation and Control Valves
33	B 3.7.3, Action C.1
34	LCO 3.7.7, Action B, Startup Feedwater and Control Valves
35	B 3.7.7, Startup Feedwater Isolation and Control Valves
36	B 3.7.7, Actions B.1 and B.2
37	B 3.7.7, References
48	Main steam line PORV block valves

new ¶

Westinghouse is writing responses to the other comments provided in Reference 1. In support of an upcoming comment resolution telecon with the Plant Systems Branch (see Reference 4), responses to all comments related to the agenda for that telecon have been provided.

These remaining responses (7, 12, 23, 38, 42-44, 46, 47, and 51), will be provided by March 27, 1997.

Please review the attached responses. Please contact Robin K. Nydes (412)374-4125 with any additional comments related to the AP600 Technical Specifications.

Brian A. McIntyre, Manager
Advanced Plant Safety and Licensing

jml

Attachment

cc: Bill Huffman, NRC (1L, 1A)
Angela Chu, NRC (5L, 5A)
N. J. Liparulo, Westinghouse (w/o Attachment)

9) LCO 3.4.3, RCS Pressure and Temperature (P/T) Limits

Action B.2 requires the plant to be placed in MODE 4 within 24 hours for Condition B, i.e. the Required Action and Completion Time for Condition A (LCO not met in MODES 1, 2, 3, or 4) are not met. Therefore, if the plant is operating in MODE 4 when the LCO is not met, there would be no required action. This is inconsistent with STS, where B.2 requires the plant to be in MODE 5 with RCS pressure less than 500 psig within 36 hours. Westinghouse should provide bases or justification for this deviation.

Response

MODE 4 has been established as the AP600 safe shutdown end state (SECY-96-128, June 12, 1996). The AP600 passive, safety-related plant cool down systems are capable of shutting the plant down from MODE 1 to MODE 4 (420°F). The AP600 Technical Specification shutdown requirements have been specified so as to permit shutdown compliance using only the passive, safety-related systems. This is accomplished by specifying MODE 4 as the shutdown end state, even when the LCO Applicability includes MODE 4. This deviation from Standard Technical Specifications (NUREG-1431) practice is based on the passive cooldown capability of the AP600 plant design.

11) LCO 3.4.6, Pressurizer

a. The operability requirement for the pressurizer heaters to provide a minimum required available capacity is deleted form the AP600 TS. The BASES states that the safety analyses do not take credit for pressurizer heater operation. The October 11, 1996, letter states that for AP600, pressurizer heaters are not needed to maintain subcooling in the long term during a loss of offsite power. Therefore, the deletion of the pressurizer heater operability requirement is acceptable. However, the analysis of a steam generator tube rupture event takes credit for the trip of the pressurizer heaters. There should be a requirement on the ability to trip the pressurizer heaters in accordance with Criterion 3 of the Policy Statement.

b. There is a typo in Required Action A.1 where the word "restore" is mistyped as "reactor." Also, the Completion Times for placing the plant in MODES 3 and 4 are drastically increased from 6 and 12 hours, respectively, to 80 and 96 hours. Acceptability of these Completion Times should be further evaluated.

11) LCO 3.4.6, Pressurizer (continued)

Response

a. SSAR Section 15.6.3.1.2, Sequence of Events for a Steam Generator Tube Rupture, states that power to the pressurizer heaters is shutoff during the course of the accident. Although not specifically stated in the accident analysis discussion, automatic trip of the pressurizer heaters is available.

The SSAR Functional Diagram (sheet 12) shows that the same logic that actuates the Core Makeup Tanks also trips the pressurizer heaters. In order to ensure the availability of the pressurizer heater trip as assumed in the safety analysis, the pressurizer heater trip will be added to AP600 LCO 3.3.2, ESFAS Instrumentation.

Since there are no applicable heater performance requirements, there is no basis for change to LCO 3.4.6, Pressurizer.

b. The editorial correction to Required Action A.1 will be included in the next revision to the AP600 Technical Specifications.

Completion Time

The Required Action A.1, 72-hour Completion Time has been selected based on the expectation that Pressurizer level deviations will be minor and that the capability remains to mitigate all DBAs assuming a single failure. Control Room pressurizer level instrumentation is available to the operators providing assurance that pressurizer level deviations will be minor. Further, since there is a Reactor Trip on Pressurizer Water Level - High 3 (LCO 3.3.1, Table 3.3.1-1, Function 9) also specified at 92% (same as LCO 3.4.6 limit), it may not be possible to exceed the LCO 3.4.6 limit without automatically tripping the reactor. The automatic Reactor Trip ensures that LCO 3.4.6 deviations will be minor.

Required Actions A.2.1 and A.2.2 specify shutdown to MODE 3 and 4 within an additional 8 and 24 hours, respectively. These times are standard times applied throughout the AP600 Technical Specifications.

13) LCO 3.4.12, Automatic Depressurization System - Operating

When the LCO and other required actions are not met during MODES 1, 2, 3, and 4 operation, Action C.2 requires the plant to be placed in MODE 4 in 24 hours. The BASES indicates the reason for the end state of MODE 4 is that the probability and consequences of a design basis event are minimized. This is inconsistent with the LCO applicability MODES which include MODE 4. If the LCO is not required to be complied with in MODE 4, then MODE 4 should be removed from the applicability MODES, and another LCO with specific configurations and supporting analysis shall be developed for MODE 4. Otherwise, failure to meet this LCO and corrective actions should enter into LCO 3.0.3 immediately.

Response

MODE 4 has been established as the AP600 safe shutdown end state (SECY-96-128, June 12, 1996). The AP600 passive, safety-related plant cool down systems are capable of shutting the plant down from MODE 1 to MODE 4 (420°F). The AP600 Technical Specification shutdown requirements have been specified so as to permit shutdown compliance using only the passive, safety-related systems. This is accomplished by specifying MODE 4 as the shutdown end state, even when the LCO Applicability includes MODE 4. This deviation from Standard Technical Specifications (NUREG-1431) practice is based on the passive cooldown capability of the AP600 plant design.

LCO 3.4.12 specifies OPERABILITY requirements for the Automatic Depressurization System in MODES 1 through 4. LCOs 3.4.13 and 3.4.14 specify Automatic Depressurization System requirements applicable in MODES 5 and 6. Considering all three LCOs, there are ADS Technical Specification requirements in all MODES except the part of MODE 6 with the upper internals removed and the refueling cavity full.

MODE 4 is specified as the LCO 3.4.12, Required Action C.2 end state. Since the ADS LCO Applicabilities cover all MODES except the part of MODE 6 with the upper internals removed and the refueling cavity full, there are no practical end states available outside the collective Applicability. (In the event of an ADS inoperability, it is not considered practical to send the plant to refueling from MODE 1 through 4.)

Considering that for AP600, MODE 4 is the preferred end state and that for ADS there are no special circumstances which indicate that further cooldown to MODE 5 is necessary, the optimum end state for LCO 3.4.12, Required Action C.2, is MODE 4, as currently specified.

14) LCO 3.4.13, Automatic Depressurization System - Shutdown, RCS Intact
LCO 3.4.14, Automatic Depressurization System - Shutdown, RCS Open

LCO 3.4.13 and 3.4.14 specify the configurations and operability of various ADS stages for shutdown operation. Westinghouse should ensure the configurations and operability are consistent with the shutdown PRA assumptions.

Response

The Technical Specifications specify the following requirements during MODE 5 and 6 shutdown operation:

LCO 3.4.13 9 Flow paths OPERABLE

Applicability MODE 5 with the RCS pressure boundary intact

LCO 3.4.14 Stage 1, 2, 3 open,
Two stage 4 flow paths OPERABLE

Applicability MODE 5 with the RCS pressure boundary open
MODE 6 with the upper internals in place and the refueling cavity less than full

In MODE 6, with the upper internals removed and the cavity full, ADS is not needed to depressurize the RCS.

While it is not the purpose of Technical Specifications to ensure that PRA assumptions are met during plant operations, the ADS LCOs 3.4.13 and 3.4.14 are consistent with the shutdown PRA.

15) LCO 3.5.1, Accumulators

LCO 3.5.1 requires both accumulators to be operable while in MODES 1, 2, 3, and 4 with pressurizer pressure greater than 1000 psig.

a. In Condition A when one accumulator is inoperable due to boron concentration, nitrogen pressure, or volume outside limits, Action A.1 requires restoration within 7 days BASES A.1 indicates that this completion time is considered acceptable because deviations in these parameters are expected to be slight considering the pressure and volume are verified once per 24 hours. However, the boron concentration is verified once per 31 days per SR 3.5.1.4. Also, there is no DBA analysis with the assumption of an accumulator inoperable to demonstrate that the AP600 design can sustain an additional single failure and remain able to mitigate all DBA. Therefore, it does not meet "Single Failure" criteria of Westinghouse's proposed Completion Time Logic. Therefore the completion time should be changed to 72 hours, consistent with STS.

15) LCO 3.5.1, Accumulators (continued)

a. Response

The Surveillance Frequency of 31 days and after volume changes of 3% for boron concentration was selected based on the STS precedent (31 days and [ ]% level increase) and on the operator's ability to monitor parameters such as level and pressure which provide a continuous reliable indication of conditions associated with boron concentration changes. Therefore, only minor deviations in boron concentration are expected even though the official surveillance is only performed each 31 days.

The STS LCO 3.5.1 Action A.1 Bases state that one accumulator below the minimum boron concentration will have an insignificant effect on core subcriticality during reflood. This statement is also valid for AP600.

Further, the AP600 Action A.1 section of the LCO 3.5.1 Bases state that:

> "Deviations in these parameters are expected to be slight, considering that the pressure and volume are verified once per 24 hours. For one accumulator, one of these parameters not within limits will have an insignificant effect on the ability of the accumulators to perform their safety function. Therefore, a Completion Time of 7 days is considered to be acceptable."

With only minor deviations in boron concentration, nitrogen pressure or volume the accumulator can perform as assumed in the safety analyses. While in Condition A, since all DBAs can be mitigated, assuming a conservative analysis and a single failure, a Completion Time of 7 days is justified.

15) b. In Condition B when one accumulator is inoperable for reasons other than Condition A, the TS allows 24 hours (completion time) for accumulator restoration. The BASES considers this to be acceptable, because, with one accumulator inoperable, the remaining accumulator is capable of providing the required safety function, based on PRA success criteria thermal/hydraulic analyses. However, the design basis analyses for small break LOCA assume both accumulators to be operable. If one accumulator is inoperable, a direct vessel injection line break would render the operable accumulator useless as its contents are discharged through the break. No DBA analysis was performed for this scenario. Therefore, the 24 hour completion time is unacceptable. The completion should be changed to one hour, consistent with the STS.

15) LCO 3.5.1, Accumulators (continued)

b. Response

The direct vessel injection (DVI) line break has been evaluated and it was found that, with the very small break size, no accumulator (one spills and one inoperable) injection is needed when nominal (realistic) conditions are applied. The DVI line break is a very small break of 13 square inches (versus 144 square inches for the largest small break LOCA).

The LCO 3.5.1 Bases Background supports this conclusion:

> "The accumulator size, water volume, and nitrogen cover pressure are selected so that both of the accumulators are sufficient to recover the core cooling before significant clad melting or zirconium water reaction can occur following a large break LOCA. One accumulator is adequate during a small break LOCA where the entire contents of one accumulator can possibly be lost via the pipe break. This accumulator performance is based on design basis accident (DBA) assumptions and models (Ref. 3). The Probabilistic Risk Assessment (PRA) (Ref. 4) shows that one of the two accumulators is sufficient for a large break LOCA and that none of the accumulators are required for small break LOCAs, assuming that at least one core makeup tank (CMT) is available."

With one accumulator inoperable (Condition B), all DBAs can be mitigated based on a realistic analysis with a single failure, therefore a 24-hour Completion Time is justified.

15) c. LCO 3.5.1 requires operability of both accumulators during operation in MODES 1 and 2, and MODES 3 and 4 with pressurizer pressure greater than 1000 psig. Action D.1 requires the plant to be in MODE 3 with pressurizer pressure less than 1000 psig in 36 hours. If the plant was in MODE 4 operation, would it need to increase the RCS temperature to above 420°F when the LCO and other corrective actions are not met? Why?

c. Response

The Required Action D.1 end state will be revised to include MODE 4.

15) d. The Required Actions and Completion Times for Conditions C and D are not acceptable. When both accumulators are inoperable, due to boron concentration outside limits or other reasons, the plant should enter into LCO 3.0.3 immediately, consistent with STS.

15) LCO 3.5.1, Accumulators (continued)

d. Response

- Condition C Both accumulators inoperable due to boron, pressure or volume

The NRC comment 15.d recommends that Condition C be revised to specify entry into LCO 3.0.3 immediately. This approach assumes that both accumulators have no capability to mitigate accidents.

However, with minor deviations in the boron concentration, nitrogen pressure or volume, the accumulators (although not fully operable) are capable of mitigating all DBAs, based on a conservative analysis but without a single failure. The reduction in effectiveness of the accumulators, due to slight deviations in the specified parameters, is more than offset by the less severe conditions assumed without a single failure.

The Condition C, 72-hour restoration time is consistent with STS precedent such as LCO 3.5.2, Condition A in which all DBAs can be mitigated without a single failure.

- Condition D Actions not met, or both accumulators inoperable

The NRC comment 15.d recommends that Condition D be revised to specify entry into LCO 3.0.3 immediately. This recommendation implies that the Condition D end state is actually different from LCO 3.0.3.

With the correction discussed in the Response to Comment 15.c, above, there is no difference in the end state required by Required Action D.1 and LCO 3.0.3. Both D.1 and LCO 3.0.3 require that the plant be put in a MODE outside the Applicability:

MODE 3 or 4 with pressurizer pressure ≤ 1000 psig

The Condition D specified Completion Time takes into account that after reaching the MODE 3 temperature, additional time is required to reduce the pressure to less than 1000 psig.

LCO 3.0.3 does not address Completion Time requirements to reduce pressure to less than 1000 psig and is, therefore, an inappropriate action for this case.

16) LCO 3.5.4, Passive Residual Heat Removal Heat Exchanger - Operating

LCO 3.5.4 specifies operability of the PRHR heat exchanger during MODEs 1, 2, 3, and 4. When the LCO is not met for reasons other than conditions listed in the specification, the required end state is MODE 4. This end state and LCO MODES of applicability are not consistent. Failure to meet this LCO should result in immediate entry into LCO 3.0.3 immediately.

Response

MODE 4 has been established as the AP600 safe shutdown end state (SECY-96-128, June 12, 1996). The AP600 passive, safety-related plant cool down systems are capable of shutting the plant down from MODE 1 to MODE 4 (420°F). The AP600 Technical Specification shutdown requirements have been specified so as to permit shutdown compliance using only the passive, safety-related systems. This is accomplished by specifying MODE 4 as the shutdown end state, even when the LCO Applicability includes MODE 4. This deviation from Standard Technical Specifications (NUREG-1431) practice is based on the passive cooldown capability of the AP600 plant design.

LCO 3.5.4 specifies OPERABILITY requirements for the PRHR HX in MODES 1 through 4, with the RCS not cooled by RNS. LCO 3.5.5 specify PRHR HX requirements applicable in MODES 4 with RNS cooling and in MODE 5 with the RCS intact. Considering these LCOs, there are PRHR HX Technical Specification requirements in all MODES except the part of MODE 5 with the RCS pressure boundary open and MODE 6.

MODE 4 is specified as the LCO 3.5.4, Required Action F.2 end state. Since the PRHR HX LCO Applicabilities cover all MODES except the part of MODE 5 with the RCS pressure boundary open and MODE 6, there are no practical end states available outside the collective Applicability. (In the event of a PRHR HX inoperability, it is not considered practical to send the plant to MODE 5 with the RCS pressure boundary open from MODE 1 through 4.)

Considering that for AP600, MODE 4 is the preferred end state and that for PRHR HX there are no special circumstances which indicate that further cooldown to MODE 5 is necessary, the optimum end state for LCO 3.5.4, Required Action F.2, is MODE 4, as currently specified.

17) LCO 3.5.6, In-Containment Refueling Water Storage Tank - Operating

LCO 3.5.6 specifies operability of the IRWST during MODEs 1, 2, 3, and 4. When the LCO is not met for reasons other than conditions listed in the specification, the required end state is MODE 4. This is not acceptable. Failure to meet this LCO should result in immediate entry into LCO 3.0.'

Response

MODE 4 has been established as the AP600 safe shutdown end state (SECY-96-128, June 12, 1996). The AP600 passive, safety-related plant cool down systems are capable of shutting the plant down from MODE 1 to MODE 4 (420°F). The AP600 Technical Specification shutdown requirements have been specified so as to permit shutdown compliance using only the passive, safety-related systems. This is accomplished by specifying MODE 4 as the shutdown end state, even when the LCO Applicability includes MODE 4. This deviation from Standard Technical Specifications (NUREG-1431) practice is based on the passive cooldown capability of the AP600 plant design.

LCO 3.5.6 specifies OPERABILITY requirements for the IRWST in MODES 1 through 4. LCOs 3.5.7 and 3.5.8 specify IRWST requirements applicable in MODE 5 and in MODE 6 with the internals removed or the cavity full. Considering these LCOs, there are IRWST Technical Specification requirements in all MODES except the part of MODE 6 with the RCS pressure boundary open and MODE 6.

MODE 4 is specified as the LCO 3.5.4, Required Action E.2 end state. Since the IRWST LCO Applicabilities cover all MODES except the part of MODE 6 with the internals removed or the cavity full, there are no practical end states available outside the collective Applicability. (In the event of a IRWST inoperability, it is not considered practical to send the plant to MODE 6 with the internals removed or the cavity full from MODE 1 through 4.)

Considering that for AP600, MODE 4 is the preferred end state and that for IRWST there are no special circumstances which indicate that further cooldown to MODE 5 is necessary, the optimum end state for LCO 3.5.6, Required Action E.2, is MODE 4, as currently specified.

18) LCO 3.5.7, IRWST - Shutdown, RCS Inventory High
LCO 3.5.8, IRWST - Shutdown, RCS Inventory Low

LCO 3.5.7 and 3.5.8 specify that the IRWST, with one injection flow path and one containment recirculation flow path, shall be operable during MODE 5 and 6 operation. Since Westinghouse has not submitted the Shutdown Evaluation Report, the staff has not made a determination as to whether this configuration (compared to operability of both injection flow paths and both recirculation paths) is sufficient. Westinghouse should ensure this configuration is consistent with that assumed in the shutdown PRA.

The bases for LCO 3.5.8 states that two injection and recirculation flow paths must be operable. The is inconsistent with the actual LCO and presumably in error.

Response

The Technical Specifications specify the following IRWST requirements during MODE 5 and 6 shutdown operation:

LCO 3.5.7	One injection flow path and one containment recirculation flow path shall be OPERABLE
Applicability	MODE 5 with the RCS pressure boundary intact; MODE 5 with the RCS pressure boundary open and a visible level in the pressurizer
LCO 3.5.8	One injection flow path and one containment recirculation flow path shall be OPERABLE
Applicability	MODE 5 with the RCS pressure boundary open and level not visible in the pressurizer; MODE 6 with the upper internals in place and the refueling cavity less than full

In MODE 6, with the upper internals removed and the cavity full, IRWST injection is not needed to borate the RCS or to provide heat removal for the RCS.

While it is not the purpose of Technical Specifications to ensure that PRA assumptions are met during plant operations, the IRWST LCOs 3.5.7 and 3.5.8 are consistent with the shutdown PRA.

The editorial correction to the LCO 3.5.8 Bases will be included in the next revision to the AP600 Technical Specifications.

20) 5.6.5, Core Operating Limits Report (COLR)

Item b. lists the approved analytical methods used to determine the core operating limits. The small break LOCA analysis code, NOTRUMP, is not listed. Both large and small break LOCA should be listed because the small break LOCA analysis must be performed to confirm that large break LOCAs is limiting.

The approved methodology for LCO 3.2.5 OPDMS-monitored power distribution parameters is not listed in this section. Is the methodology the same as WCAP-12473, BEACON? If so, WCAP-12473 should be listed.

Response

The following approved methodology references will be added to the next revision of the AP600 Technical Specifications, Section 5.6.5:

5. WCAP-14807, "NOTRUMP Final Validation for AP600", R. L. Fittante et. al., January, 1997 (Westinghouse Proprietary).

   (Methodology for Specification: 3.2.1, Heat Flux Hot Channel Factor)

6. WCAP-12472-P-A, "BEACON Core Monitoring and Operations Support System", August 1994 and Addendum 1, May 1996 (Westinghouse Proprietary).

   (Methodology for Specification: 3.2.5, OPDMS - Monitored Power Distribution Parameters)

21) LCO 3.4.12, Automatic Depressurization System
LCO 3.5.6, In-containment Refueling Water Storage Tank

The squib valves are important components of these systems, however, no operability determination criteria or surveillance requirements are stated for these valves. These valves should be specifically addressed in the AP600 TS.

Response

The requirements specified in the Inservice Testing Program (Section 5.5.4) provide control for inservice testing of ASME Code Class 1, 2, and 3 components including applicable supports. The subject squib valves will be tested in accordance with ASME Section XI which specifies valve testing in accordance with the ASME OM Code. The applicable ASME OM Code squib valve requirements are specified in paragraph 4.6, Inservice Tests for Category D Explosively Actuated Valves. The requirements include actuation of a sample of the installed valves each 2 years and periodic replacement of charges.

21) LCO 3.4.12, Automatic Depressurization System
LCO 3.5.6, In-containment Refueling Water Storage Tank

Response (continued)

The Inservice Testing Program requirements applicable to squib valves are comprehensive and verify the operability of the squib valves. Specification of any additional Technical Specification requirements would be redundant to the existing Inservice Testing Program requirements.

22) LCO 3.4.12, Automatic Depressurization System

It is the staff's understanding that ADS 1, 2, and 3, discharge piping have vacuum breakers to reduce possible hydrodynamic loads from water slugs which may form in the sparger piping. The function of these vacuum breakers is similar to those found on the safety relief valves in BWRs. The BWR vacuum breakers are subject to technical specification operability requirements. Westinghouse should determine if the ADS line vacuum breakers should be subject to similar TS requirements.

Response

This question was resolved in a phone conversation (Friday, 2/21/97) between the NRC (W. Huffman) and Westinghouse (M. Corletti). It was agreed that the ADS discharge piping vacuum breakers do not meet the 10CFR50.36 criteria and, therefore, do not need to be added to the AP600 Technical Specifications.

This determination is based on the ability of the ADS to perform its safety related function independent of the vacuum breakers. The vacuum breakers are subject to testing and inspection in accordance with the ASME XI Inservice Test Program.

26) Section 5.5.6, Secondary Water Chemistry Program

Revise the first sentence to read: "This program provides controls for monitoring secondary water chemistry to inhibit SG tube degradation, low pressure turbine disc stress corrosion cracking, and flow accelerated corrosion of all carbon steel components."

26) Section 5.5.6, Secondary Water Chemistry Program (continued)

Response

The AP600 Technical Specification sentence currently reads exactly as stated in NUREG-1431, Rev. 1:

> "This program provides controls for monitoring secondary water chemistry to inhibit SG tube degradation and low pressure turbine disc stress corrosion cracking."

The proposed sentence adds:

> "and flow accelerated corrosion of all carbon steel components"

The NRC requested change will be included in the next revision to the AP600 Technical Specifications.

27) B 3.7.1, Main Steam Safety Valves

TS bases B 3.7.1, Main Steam Safety Valve, states that in MODES 4 (with normal residual heat removal system in service) and 5, there are "no credible transients requiring the MSSVs." Therefore, the applicability modes for LCO 3.7.1, Main Steam Safety Valve, do not include Mode 4 with RCS cooled by RNS.

Justify the statement that "no credible transients requiring the MSSVs." Assuming that RNS fails during Mode 4 (with RCS cooled by RNS) and that MSSVs are not operable, explain how the secondary system overpressure can be protected.

Response

If RNS becomes unavailable in MODE 4 with the RCS cooled by RNS, the following Technical Specification and defense in depth systems are available to provide decay heat removal, thus preventing overpressurization of the Main Steam System:

Core Makeup Tank (LCO 3.5.2)
Passive Residual Heat Removal (LCO 3.5.4)
Automatic Depressurization System (LCO 3.4.12) and Incontainment Refueling Water Storage Tank (LCO 3.5.6)
Feedwater System and Power Operated Relief Valves

27) B 3.7.1, Main Steam Safety Valves

Response (continued)

With these heat removal systems available, in MODE 4 with RNS cooling, there are no credible transients which would heat the main steam system enough to produce pressure at or above the lowest MSSV setpoint (1085 psig).

Therefore, the Applicability specified for LCO 3.7.1 is correct as is.

28) LCO 3.7.1, Main Steam Safety Valves - MODE 4

In LCO 3.7.1 Action B.2, Westinghouse uses Mode 4 as the safe end state. This is not consistent with the specified applicability modes. It should be "Mode 4 with RCS cooled by RNS" even if the justification provided in the above question concerning B 3.7.1 is found acceptable.

Response

MODE 4 has been established as the AP600 safe shutdown end state (SECY-96-128, June 12, 1996). The AP600 passive, safety-related plant cool down systems are capable of shutting the plant down from MODE 1 to MODE 4 (420°F). The AP600 Technical Specification shutdown requirements have been specified so as to permit shutdown compliance using only the passive, safety-related systems. This is accomplished by specifying MODE 4 as the shutdown end state, even when the LCO Applicability includes MODE 4. This deviation from Standard Technical Specifications (NUREG-1431) practice is based on the passive cooldown capability of the AP600 plant design.

The MSSV LCO, Table 3.7-1, specifies the number of operable valves per steam generator:

100% RTP	3 valves
67% RTP	2 valves

With the plant in MODE 4 at 0.0% RTP and cooled to less than 420°F, the need for secondary system overpressure protection (>1085 psig) is greatly diminished. Considering that for AP600, MODE 4 is the preferred end state and that for the MSSVs there are no special circumstances which indicate that further cooldown to MODE 5 is necessary, the optimum end state for LCO 3.7.1, Required Action B.2, is MODE 4, as currently specified.

29) Draft TS 3.7.8, Secondary Coolant Leakage

The Plant Systems Branch has reviewed Westinghouse's letter, dated July 26, 1996, regarding "AP600 LBB QUESTIONS" and found the position described in the letter regarding steamline leakage control unacceptable. In the letter, Westinghouse revised its previous response to Q410.145 withdrawing its TS commitment for steamline leakage detection without withdrawing LBB for the steamline application. The staff found that the alternative method proposed by Westinghouse, using administrative procedures, did not provide sufficient measures to justify LBB application for steamlines. By letter dated September 5, 1996, the staff indicated to Westinghouse that application of LBB to the steamlines is not acceptable without a TS for mainsteam leakage detection because the technology of LBB relies on the detection of leaks prior to pipe breaks. Westinghouse was requested to provide a proper TS for steamline leakage detection.

In response, Westinghouse provided informal draft TS 3.7.8, Secondary Coolant Leakage on October 24, 1996. The NRC staff (SPLB, TSB, and ECGB) are reviewing it. A telecon between Westinghouse and the staff was held on November 5, 1996. The staff asked justification from Westinghouse as related to (1) the adequacy and margin of 5.0 gpm as the leakage limit and (2) the leakage reducing time of 8 hours before entering into Action B.1.

Response

(1) The leakage limit has been conservatively selected to correspond to a leakage crack size that is detectable, structurally stable even under seismic loads, and permits sufficient time for operator corrective action to preclude pipe rupture.

The leakage limit (5.0 gpm) is 10 times the minimum leak detection capability of the instrumentation. This large detection capability margin ensures that the leaks can be detected as assumed in the LBB analysis.

The acceptability of the 5.0 gpm leakage limit is established by doubling the size of a 5.0 gpm leakage crack and verifying that the 2X crack is structurally stable and thus not liable to increase in size. Therefore, cracks twice as big as those allowed by the 5.0 gpm limit are stable and will not increase in size.

(2) The Required Action A.1 8 hour Completion Time was selected based on the stability of leakage cracks twice as long as those corresponding to the 5.0 gpm limit. If the leakage can be restored to within the 5.0 gpm limit within 8 hours, there is no technical basis for shutting the plant down to MODE 4 with RNS cooling, since the crack will not increase in size in any amount of time. If the leakage can not be restored to within the limit within 8 hours, it is considered that some difficulty has been encountered in attempting to control the leak and that shutdown should be initiated so that repairs may be pursued.

31) LCO 3.7.6, Main Control Room Habitability System (VES)

a. APPLICABILITY

It is stated that "APPLICABILITY" applies for MODES 1, 2, 3 and 4 and during movement of irradiated fuel assemblies. The bases states that VES is not required for Modes 5 or 6 (when irradiated fuel is not being moved) because accidents involving fission product release are not postulated. Where have Mode 5 and 6 events been evaluated to justify this conclusion.

a. Response

The requested reference to justify the VES Applicability in MODE 5 and 6 is not available. The VES LCO Applicability was established to be consistent with the mitigation function assumed in accident analyses. None of the accidents discussed in SSAR Chapter 15 which are postulated for MODES 5 or 6, except a fuel handling accident, assume credit for operation of the VES.

The basis for the LCO Applicability is documented in the LCO 3.7.6 Bases with the same level of detail as the STS Bases.

b. SURVEILLANCE FREQUENCY

The staff believes that the surveillance requirements for SR 3.7.6.3, SR 3.7.6.4, SR 3.7.6.5, SR 3.7.6.6, SR 3.7.6.7, and SR 3.7.6.8 should all be "7 Days" instead of the frequencies specified unless specific justification can be provided.

The surveillance frequency for SR 3.7.6.9 should be every refueling outage in accordance with SECY-95-132.

b. Response

Valve and Damper Surveillances

The NRC comment states that each of the following Surveillance Frequencies should be 7 days. No reasons have been provided for this recommendation. Since there is no STS LCO corresponding to a bottled air control room habitability system, STS precedents from similar Surveillances were used to establish the Frequency for these AP600 Surveillances.

	VES Surveillances	Freq.	STS Precedent
SR 3.7.6.3	Isolation valve operability	IST	SR 3.6.3.8 - [18 m]
SR 3.7.6.4	Manual valve position	31 d	SR 3.6.3.3 - 31 d

31) LCO 3.7.6, Main Control Room Habitability System (VES)

b. Response (continued)

SR 3.7.6.5	Isolation damper actuation	24 m	SR 3.7.10.3 - [18 m] SR 3.6.11.4 - [18 m]
SR 3.7.6.6	Relief valve operability	IST	SR 3.6.12.1 - IST
SR 3.7.6.7	Relief damper operability	24 m	SR 3.6.11.4 - [18 m]
SR 3.7.6.8	Regulating valve operability	IST	SR 3.6.12.1 - IST

No changes to the specified Surveillance Frequencies are needed.

System Pressurization Test Frequency

SR 3.7.6.9 (control room pressurization test) specifies a Surveillance Frequency in accordance with the Inservice Testing Program. AP600 Technical Specification section 5.5.4, Inservice Testing Program, when revised per resolution of comment number 25, will reference SSAR Table 3.9-7, System Level Inservice Testing Requirements, which specifies the required test at a Frequency of 24 months, consistent with SECY-95-132. No additional changes are necessary.

c. SURVEILLANCE DESCRIPTION

SR 3.7.6.1 Verification of air temperature of main control room should be "78° F" and not "80° F."

SR 3.7.6.2 should state that "Verify that the compressed air storage tanks are pressurized with no storage tanks isolated from the system to [≥ 3400 psig but ≤ 3600 psig]."

SR 3.7.6.6 should state that "Verify that each VES pressure relief isolation valve within the MCR pressure boundary is OPERABLE upon receipt of an actual or simulated actuation signal."

The surveillance description in SR 3.7.6.9 should be consistent with SECY-95-132. Both a minimum and maximum flow addition rate should be included in the surveillance description.

31) LCO 3.7.6, Main Control Room Habitability System (VES) (continued)

c. Response

SR 3.7.6.1

Agree, the control room surveillance temperature will be changed from 80°F to 78° F. The corresponding Bases section will be revised to note that 78°F is the nominal temperature and that 80°F was used in the safety analysis and includes a 2°F measurement uncertainty.

SR 3.7.6.2

The suggested addition of "with no storage tanks isolated from the system" would require that the tank pressure be verified while the isolation valves are open and air is discharging. There is no need to measure the air storage tank pressure during discharge. The proposed test could deplete the pressure such that the lower limit might not be met at the end of the test, thus requiring tank restoration and retest. Verification of static tank pressure along with valve and damper position and operability requirements are adequate to ensure the system's capability to perform its safety function.

No changes to SR 3.7.6.2 are needed.

SR 3.7.6.6

The suggested addition to the surveillance would restrict the relief valve operability requirement to only "upon receipt of an actual or simulated actuation signal." The verification currently specified requires the relief valve to be OPERABLE at all times covered by the Applicability (MODES 1-4 and during fuel movement), not just upon receipt of an actuation signal.

The currently specified surveillance is correct as is. No changes to SR 3.7.6.6 are needed.

SR 3.7.6.9

Since the AP600 Technical Specification Inservice Test Program (Section 5.5.4) refers to the SSAR requirements, SSAR Table 3.9-17, Note 8, will be revised to specify the VES pressure and flow rate in accordance with SECY-95-132.

No changes to SR 3.7.6.9 are needed.

31) LCO 3.7.6, Main Control Room Habitability System (VES)

31) d. BASES

The "BACKGROUND" description (Page B 3.7-24, Third Paragraph, Fourth line) should state MCR initial temperature as "78° F" and not "80° F."

APPLICABLE SAFETY ANALYSES (Pages B 3.7-25) Explain why VBS does not isolate and initiate the VES on containment isolation and include the rationale in Third Paragraph.

The Third Paragraph, First Sentence should state that "In the event of high level of gaseous radioactivity outside of the MCR, the VBS....filtration

functions." The Third and Fourth Sentences should state that "Upon exceeding predetermined setpoint for high particulate or iodine radioactivity or all ac power loss, a safety related signal......VES storage tanks. Isolation of the VBS consists of closing safety related VBS isolation dampers in the supply....MCR pressure boundary."

d. Response

BACKGROUND

Agree, the temperature will be changed to 78°F nominal temperature in the next revision to the AP600 Technical Specifications.

APPLICABLE SAFETY ANALYSES

The existing information provided in the Applicable Safety Analyses section is correct as is. The additional details requested are not needed to explain the basis for the LCO. No changes are needed.

32) LCO 3.7.3, Main Feedwater Isolation and Control Valves

In LCO 3.7.3, Action C, Main Feedwater Isolation and Control Valves, the required action is not possible for two flow paths with two valves in the same flow path inoperable, in Modes 1 or 2, because the system will be not functional with both flow paths isolated. The plant should be placed in Mode 3 and at least one flow path should be restored to operable status.

32) LCO 3.7.3, Main Feedwater Isolation and Control Valves (continued)

Response

The specified actions are based on STS requirements specified in LCO 3.7.3, which for each case of inoperable valves, specifies isolation of the flow path. While it is unlikely that the plant could be operated in MODE 1 or 2 with all main feedwater isolated, the specified action places the main feedwater flow paths in the condition required to perform the safety function. This approach allows the greatest operational flexibility while ensuring the safety function is accomplished. With the flow path isolated, the required safety function is met, Mode change and valve restoration are not required.

No changes to Action C are needed.

33) B 3.7.3, Main Feedwater Isolation and Control Valves

In Bases C.1 of TS 3.7.3, with two inoperable valves in the same flow path, there will be no redundant system to operate automatically and perform the required safety function. Under these conditions, at least one valve in each affected flow path should be restored to operable status and the affected flow path should be isolated within 8 hours.

Response

Required Action C.1 specifies isolation of the affected flow path.

As discussed in the Applicable Safety Analyses section of the LCO 3.7.3 Bases, the safety function of the Main Feedwater Isolation and Control Valves is to automatically isolate the startup feedwater flow path to the steam generator in the event of a steam line break or feedwater line break. This isolation function limits the mass and energy delivered to the steam generators, affecting cooldown and mass and energy release to containment. When a main feedwater flow path is isolated, the automatic isolation capability is no longer needed, since the safety analysis isolation assumption is met.

Once Required Action C.1 is met and the flow paths with inoperable isolation valves are isolated, no additional action is needed. With the flow path(s) isolated, the required safety function is met, Mode change and valve restoration are not required to satisfy the safety analysis assumptions.

Since valve restoration is permitted in accordance with LCO 3.0.2, a specific Required Action to restore the inoperable valves is not needed.

No changes to Action C are needed.

34) LCO 3.7.7, Startup Feedwater Isolation and Control Valves

In LCO 3.7.7, Action B, Startup Feedwater and Control Valves, the required action is not acceptable for more than one flow paths with two inoperable valves. Because the system become inoperable with two inoperable flow paths isolated. The plant should be in Mode 3 and at least one flow path should be restored to operable status.

Response

Required Action B.1 specifies isolation of the affected flow path(s).

As discussed in the Applicable Safety Analyses section of the LCO 3.7.7 Bases, the safety function of the Startup Feedwater Isolation and Control Valves is to automatically isolate the startup feedwater flow path to the steam generator in the event of a steam line break or feedwater line break. This isolation function limits the mass and energy delivered to the steam generators, affecting cooldown and mass and energy release to containment. When the a startup feedwater flow path is isolated, the automatic isolation capability is no longer needed, since the safety analysis isolation assumption is met.

Once Required Action B.1 is met and the flow paths with inoperable isolation valves are isolated, no additional action is needed. With the flow path(s) isolated, the required safety function is met, Mode change and valve restoration are not required to satisfy the safety analysis assumptions.

Periodic verification that the flow path remains isolated in accordance with Required Action B.2, ensures that the safety function continues to be met.

No changes to Action B are needed.

35) B 3.7.7, Startup Feedwater Isolation and Control Valves

Third paragraph of the BACKGROUND in Bases B 3.7.7 (startup feedwater isolation and control valves), the statements does not correctly describe the system design and the paragraph should be revised to conform to the design changes.

35) B 3.7.7, Startup Feedwater Isolation and Control Valves (continued)

Response

The third paragraph of the BACKGROUND in Bases B 3.7.7 will be revised as shown below in the next revision of the AP600 Technical Specifications. The changes shown underlined clarify the description to ensure the system design is correctly understood.

> The subsystem consists of two series startup feedwater valves within a startup feedwater line which bypasses the main feedwater line and provides feedwater control for low feedwater demand conditions. Feedwater can be supplied to the startup feedwater line via either the main or startup feedwater pumps. The feedwater is delivered directly to the SG independent of the main feedwater line. Each startup feedwater line contains one control valve and one isolation valve.

36) B 3.7.7, Startup Feedwater Isolation and Control Valves

In Bases B 3.7.7, Actions B.1 and B.2, for one or more flow paths with both the isolation and control valves inoperable, isolation of the affected flow paths within 1 hour is acceptable. However, at least one flow path should be restored to operable status within 8 hours.

Response

Required Action B.1 specifies isolation of the affected flow path(s).

As discussed in the Applicable Safety Analyses section of the LCO 3.7.7 Bases, the safety function of the Startup Feedwater Isolation and Control Valves is to automatically isolate the startup feedwater flow path to the steam generator in the event of a steam line break or feedwater line break. This isolation function limits the mass and energy delivered to the steam generators, affecting cooldown and mass and energy release to containment. When the a startup feedwater flow path is isolated, the automatic isolation capability is no longer needed, since the safety analysis isolation assumption is met.

Once Required Action B.1 is met and the flow paths with inoperable isolation valves are isolated, no additional action is needed. With the flow path(s) isolated, the required safety function is met, flow path restoration is not required to satisfy the safety analysis assumptions.

Periodic verification that the flow path remains isolated in accordance with Required Action B.2, ensures that the safety function continues to be met.

No Bases changes are needed.

37) B 3.7.7, Startup Feedwater Isolation and Control Valves

TS 3.7.7 and its bases are written without references. The TS should refer to AP600 SSAR Section 10.4.9 and should be conform with the system design and functioning.

Response

A reference to SSAR Section 10.4.9 will be added to the Background section of the Bases in the next revision of the AP600 Technical Specifications.

48) PORV Block Valves

Westinghouse should determine if the main steam line PORV block valves should be included in the AP600 TS since credit is taken for the closure of these valves in SSAR Chapter 15 accident analyses.

Response

The accident analyses take credit for closure of the PORV block valves on a containment isolation signal.

Operability of the PORV block valves is specified by existing requirements in LCO 3.6.3, Containment Isolation Valves. The LCO section of the Bases specifies that the Containment Isolation Valves are listed in SSAR section 6.2. The Containment Isolation Valves covered by this LCO are listed in SSAR Figure 6.2.3-1, Containment Mechanical Penetrations and Isolation Valves. Figure 6.2.3-1, sheet 2 of 4, lists the subject PORV block valves as SGS-PL-V027A and B which (along with several other valves) isolate the SGS mainsteam lines 01 and 02.

No AP600 Technical Specification change is needed, since the PORV block valves are included in existing LCO 3.6.3 requirements.

WESTINGHOUSE ELECTRIC CORPORATION
PO BOX 355
PITTSBURGH, PA 15230-0355

AP600

ADVANCED PLANT SAFETY AND LICENSING

FAX NO: 412-374-4887 (WIN 284)
CONFIRMATION NO: 412-374-4237

DATE: 3/21

TO: Diane Jackson

LOCATION: NRC/DRPM

FAX NUMBER:

FROM: **Brian A. McIntyre**

PHONE: WIN: 284-4334
BELL: 412-374-4334

NUMBER OF PAGES (INCLUDING COVER SHEET): 6

COMMENTS / MESSAGE:

Diane,

Attached are our notes on your DRAFT minutes from the 3/3 Senior Management Meeting on Structural Issues.

Please call if you have any questions.

B

NRC FORM 366 (12-89) U.S. NUCLEAR REGULATORY COMMISSION

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

DRAFT DRAFT DRAFT

APPLICANT: Westinghouse Electric Corporation

FACILITY: AP600

SUBJECT: SUMMARY OF MARCH 3, 1997, SENIOR MANAGEMENT MEETING

The subject meeting was held at the Nuclear Regulatory Commission (NRC) office in Rockville, Maryland, on March 3, 1997, between representatives of Westinghouse and its consultants and the NRC staff. The main purpose of this meeting was to discuss the siting of the AP600 on shallow soil sites, and the ~~thickness~~ DESIGN of the basemat. Attachment 1 is a list of meeting participants. Attachment 2 is the meeting agenda and staff handouts. Attachment 3 is the Westinghouse handouts.

Mr. Tim Martin opened the meeting stating the objectives and emphasizing the importance to the resolution of the key issues. Earlier Westinghouse had questioned whether the staff had considered the Westinghouse January 28, 1997, letter in its January 31, 1997, letter. The staff informed Westinghouse that the staff did review the letter and concluded it did not change ~~the letter~~ Their content. However, the staff will formally respond to the January 28, and February 10, 1997, Westinghouse letters.

Mr. Jerry Wilson of the staff opened the technical issue discussion with a presentation on the regulatory definitions. Focus was placed on Westinghouse's use of the terms "site parameter" and "interface requirement" and the regulatory differences between the terms. The staff stated that Westinghouse is not using these terms consistent with the regulatory definition. The staff pointed out that all standard plants must meet the defined site parameters and that interface requirements were for those portions of the plant outside the scope of design certification. It was also stated that the scope of design certification was for an essentially complete design except for site-specific elements such as the ultimate heat sink. Westinghouse stated that they believe they have used a consistent Westinghouse definition that complies with the Regulations. They are willing to discuss the differences and proper usage of the terms with the staff.

Vulnerability of Nuclear Island Foundation Basemat to Soil Variability

Mr. Goutam Bagchi of the staff presented the technical and regulatory position regarding the thickness of the basemat and its vulnerability to variance in soil stiffness. In the presentation, it was stated that the AP600 basemat is an irregular shape. Clarification of this statement was requested. Mr. Bagchi explained the shape of the basemat is not the typical rectangular shape and therefore, the location of stresses may be different.

The staff stated that its concern was that the six foot basemat would require a more detailed geotechnical investigation and a more specific construction sequence than used in any operating or advanced plant design. The basemat

DRAFT

will not have the benefit of the added stiffness from the walls during construction. Additionally, verification by Inspection, Tests, Analyses, and Acceptance Criteria would be necessary.

The staff has expressed its position in letters and suggested two options to Westinghouse for the resolution of this key issue: (1) Westinghouse should demonstrate that the basemat can accommodate the effects of soil variability and without reliance on a detailed construction sequence, or (2) provide an alternate design or subdesign with a different basemat thickness for non-uniform sites. The staff also stated that this issue could be presented to the Commission, at least for information, since it differs from past practices and may not meet the Commission's intent for the design of advanced plants.

apply to a full Range of site Condi-

2 To ...

Mr. Richard Orr presented the structural Westinghouse position. Westinghouse stated that due to the "egg-crate" design of the basemat and lower elevation walls that the "integrated" basemat was equivalent to the thicker basemat of the ~~other advanced~~ evolutionary designs. ~~Mr. Orr stated that due to NRC staff review, and subsequent design changes, the basemat has additional margin and Westinghouse has more confidence in the design.~~ Westinghouse believes the staff's confirmatory analysis was useful in identifying this issue, however, they do not agree with the results because they believe the assumptions were unduly conservative. Mr. Paul Rizzo, Westinghouse consultant, also presented information on the basemat for Westinghouse. He provided information on the construction sequence, including building height limitations ~~on~~ during construction. Due To Time Limitations Not all of the presentation material was discussed. Westinghouse stated that the AP600 could be sited on 85 percent of all eastern US (east of the Rocky Mountains) soil sites and could tolerate soil stiffness variations, within defined limits, ~~in~~ the construction sequence. Westinghouse provided the staff a draft markup of the standard safety analysis report (SSAR) via facsimile on Thursday, February 27, 1997, (Attachment 4) that described its construction sequence.

and With Broad Limits On

For resolution of this issue, Mr. Ed Cummins of Westinghouse stated that Westinghouse is willing to define ~~additional~~ Appropriate site parameters that would apply to the AP600 standard plant, but ~~has decided not to~~ does Not Believe It Is Necessary To change the basemat design. Mr. Howard Bruschi of Westinghouse agreed with the staff's statement that the AP600 must be able to be sited at the majority of US sites. The staff agreed that, given the two statements by Mr. Cummins and Mr. Bruschi, technical review could continue.

Actions for this issue: Westinghouse will review its draft SSAR markup sent by facsimile for any necessary changes ~~due to the~~ Resulting From this meeting and formally submit the information. The staff will begin its review the draft SSAR markup. The OGC ~~staff~~ and Westinghouse will schedule a meeting to discuss the ~~terminology of~~ used of site parameter and site interface. The staff and Westinghouse will schedule a meeting to resolve the structural issues.

Siting

DRAFT

Sitting on Shallow Soil Sites

Mr. Goutam Bagchi presented the staff's position on sitting the AP600 on shallow soil sites. The staff stated that 0.3 g (with its response spectrum) is a design parameter and must be met by all AP600 standard plants and that for the standardized plant, design parameter analyses can not be delayed for the COL. Mr. Cummins stated that Westinghouse did not design the AP600 for shallow soil sites, however, they are confident it has the capacity to handle some shallow soil sites. Westinghouse stated that they wanted to establish a means for the COL to demonstrate that a shallow soil site was acceptable. The staff explained that the regulations already allow the COL to apply for a 10 CFR Part 52 license with an exemption to a site parameter. Westinghouse believes that excluding shallow soil sites, the AP600 could be sited at approximately 67 percent of US sites.

Application

APPLICANT

Mr. Bob Vijuk reiterated the resolution options that the staff had put forth in its letters: Westinghouse can either (1) exclude shallow soil sites or (2) include shallow soil sites and perform the analysis as part of the SSAR. ~~The staff agreed~~ that these were the two most viable and expedient options. Option 1 still allows the COL to apply for an exemption to a Part 52 license.

Believes

BASED ON THE CESAR AND STAFF FSER ON THE SYSTEM 80+,

Westinghouse stated that they believed Combustion Engineering (CE) System 80+ was allowed to use site-specific analysis if the site was outside of the design envelope. ~~Westinghouse cited a staff safety evaluation report.~~ The staff clarified that CE, in its design control document, used its design parameters, not a site-specific seismic hazard for COL site-specific analysis.

Actions: Westinghouse will review the CE design certification document to verify the above statement. Westinghouse will review its approach and decide on a resolution path. Westinghouse will provide a response and draft SSAR markups on the revised approach.

Schedule

and

ON Westinghouse

Mr. Tim Martin of the staff provided an overview of the staff review status. Current challenges to the review schedule discussed include late submittals by Westinghouse, design changes for post 72-hour actions, lack of resolution of key issues, lack of staff feedback ~~of~~ submittals. Review progress was also highlighted. Four of the 27 key issues identified in January have been technically resolved. ~~In~~ many review areas, the staff is preparing its final safety evaluation reports. THE STAFF IS IN THE PROCESS OF PROVIDING WESTINGHOUSE WITH THEIR POSITIONS ON A NUMBER OF THE REMAINING KEY ISSUES

Mr. Howard Bruschi stated that communication was good between the staff and Westinghouse. He requested further management ~~involvement in~~ issuing future requests for additional information. He also acknowledged that to meet the schedule would be a challenge to both the staff and Westinghouse.

NRC

REVIEW PRIOR TO THE STAFF

DRAFT

A draft of this meeting summary was provided to Westinghouse to allow them the opportunity to ensure that the representations of their comments and discussions were correct.

Diane T. Jackson, Project Manager
Standardization Project Directorate
Division of Reactor Program Management
Office Of Nuclear Reactor Regulation

Docket No. 52-003

Attachments: As stated

cc w/attachments:
See next page

March 20, 1997

Subject: Informal Transmittal of Information on
- WCAP-14407 Ch 9 informal questions 67 - 132, and A1 - A11

To: Ed Throm Fax: 301-415-3577 (14 pages)

cc: Jim Gresham Dick Haessler
Brian McIntyre Tim Andreychek

Attached are proposed approaches to respond to the subject NRC review questions. The ones marked with a "T" are those which Westinghouse recommends to have technical discussions during the 3/25/97 PCS DBA meeting.

Unfortunately, the reviews of the proposed responses to Ch 9 informal questions 1-66, 72, and 91, and the Ch 7 RAIs were not completed by the end of today. We will forward those to you as soon as possible tomorrow.

It should be noted that WCAP-14407 Section 4, WCAP-14812, and WCAP-14845 have been issued subsequent to the issuance of NRC review comments on the preliminary draft Section 9. The attached responses assume the reviewers have access to all three WCAPs at this time.

Joel Woodcock

prelim.wp

C = Clarification at meeting w/ NRC
S: Straight forward
T = Technical discussion @ meeting w/ NRC

Responses to NRC Questions on Section 9 of the WCAP-14407

Several preliminary reports on AP600 mixing and stratification effects have been issued. The knowledge base was consolidated into NTD-NRC-96-4763 (Reference 1) which was reviewed by the NRC. Subsequent to the NRC review (Reference 2) of NTD-NRC-96-4763, the report was incorporated into WCAP-14407 (Reference 3) as Section 9. Throughout the responses, section number references (i.e., Section 9) refer to the section in WCAP-14407. In addition, Appendices A, B, and C refer to new appendices which will be added to Section 9 in WCAP-14407 in response to NRC requests for additional detail. Also note that the Figure numbers given in the responses are those from WCAP-14407.

Page 9 General Comment

In general, Section 9 will be revised in the following way to respond to NRC review comments:

Appendices A, B, and C will be added to Section 9 of WCAP-14407. Appendix A will provide a description of the CMT calculations performed.

Appendix B will provide a description of the effects of stratification on heat sink utilization within a volume.

Appendix C will provide a summary of test data available from which to judge acceptability of the circulation predicted by lumped parameter modeling.

Additional information will be added to figures for clarification.

Section 9 sensitivity cases will be rerun using the evaluation model in Section 4 Revision 1.

More references will be added to the text to support the observations and conclusions, as discussed below for specific questions.

S #67 The axial gradient in the bulk of the CMT room is not indicated on Figure 9-5. Appendix A will show the basis for the mixing time constant. Circulation rates and the resultant small mixing time constant support the statement that a small axial gradient is expected. Reference to Appendix A will be added to the text in Section 9.3.1.3. A jet cannot pass into the CMT room undissipated as discussed in the response to #68. A buoyant plume will rise from the CMT room drawing flow in from above and through the steam generator (SG) compartment. Therefore, Figure 9-5 is valid for all LOCA post-blowdown phases.

T #68 A jet cannot pass directly from the cold leg piping through the CMT room due to the physical layout. Assuming the source goes entirely into the CMT room is one of the extreme cases examined. To enter the CMT room, the source comes in from the steam generator (SG) into the vertical access tunnel (stairwell), then into the CMT room. Given the equipment/structures in the vertical access tunnel and the CMT room, and the location of the CMT room ceiling openings, there is no credible path for a jet to go from the cold leg piping through the CMT room undissipated. A

sketch will be provided to support this. Therefore, the source will be dissipated as it moves to the CMT room and will rise as a plume up to the CMT room ceiling. Therefore, the general pattern shown in Figure 9-5 is valid for all post-blowdown phases. This discussion will be included in Appendix A.

S #69 Appendix B will provide the methods used for the hand calculation to assess stratification effects on integrated heat removal.

T #70 The subdivision of the CMT into three equal sections is a realistic approach based on the physical layout of the CMT room. The floor and the ceiling define two regions, and the grating and associated structural steel at the mid-plane of the room define the third region. This discussion will be included in Appendix B.

S #71 The outer surfaces of the floor and ceiling were modeled as an adiabatic boundary condition. The thermal boundary conditions used in the calculations were the lumped parameter CMT room steam concentrations (Figure 9-6) from a typical evaluation model calculation.

#72 later

S #73 a. The conclusions drawn from Figure 9-8 are not used to justify the volume being well mixed. The CMT room calculation shows that the mixing time constant is small so steam density gradients are not expected to be large. The steam stratification calculation shows that an extreme assumed gradient does not significantly affect the total integrated heat removal in the CMT room relative to the average steam concentration. Therefore, the lumped parameter model is reasonable to use. The figure also shows that stratification has a weak effect by the time of maximum containment pressure because the CMT heat sinks have reached thermal saturation. Although the results show that the benefit of enriched steam at the top is offset by the penalty of enriched non-condensibles at the bottom, a conservative approach is taken by neglecting upward facing surfaces. Thus, the evaluation model uses the lumped parameter model in a conservative way to bound potential penalties on floor heat transfer due to concentration of non-condensibles.

b. Table 9-1 will taken to the next level of detail to show the break scenarios evaluated.

S #74 During the blowdown, the high break mass flow pressurizes the SG compartment and flow exits based on the relative loss coefficients. Such pressure-driven flows are well represented by the lumped parameter node-network formulation. The first paragraph in Section 9.3.2 will be revised to more clearly describe the evaluation model with respect to the blowdown phase. During the blowdown phase, the high velocity jets exiting the openings of the SG compartment provide excellent circulation so it is reasonable to assume that the individual volumes are well mixed. A lumped parameter node is well suited for a well mixed volume.

S #75 Containment pressure can be well predicted by lumped parameter modeling for buoyancy dominated flows from breaks in lower containment regions, where

distributions within rooms are small. This is based on comparisons of lumped parameter predictions to HDR. The relevant conclusions from publicly available HDR code validation will be described in Appendix C.

T #76 An attempt was being made to distinguish between flows driven by high static pressure due to high mass flow rates, such as the SG compartment during blowdown, and flows within a room driven by local momentum effects, such as jet and wall entrainment. The paragraph will be clarified.

S #77 Based on the circulation rates present in AP600, gradients would be small within compartments and within the above-deck region during post-blowdown LOCAs. One may conclude that stratification is a second order effect which may be superimposed on a lumped parameter circulation calculation. Therefore, a reasonable solution can be based on an initial assumption of uniform properties within compartments, and the perturbation of stratification within a compartment can be studied separately.

S #78 All phases of the LOCA event are covered in Section 9.3.2 and its subsections. Section 9.3.2 will be revised to clarify which LOCA phase is being addressed throughout the discussion.

S #79 The distribution of volumes and heat sinks is provided in WCAP-14812 (Reference 4). It should be noted that only a small fraction of the volume exists below the break initially, and the lower compartments fill as primary water exits the break. Table 3-1 of WCAP-14812 will be added to Section 9. In addition, the relative elevation of the break source will be added to Figure 9-2, and the initial volume existing below the break elevation will be added to the text along with the volume fill height as a function of time.

S #80 The node-network solution is the WGOTHIC AP600 evaluation model described in Section 4. The sensitivity cases of Section 9 will be performed using the base model in Section 4, Revision 1, because of the potential for an effect of changing flow paths on predicted circulation.

S #81 The comparison shows that, for the LOCA blowdown phase, the details of the flow connections for the multi-node evaluation model are not important with respect to the pressure results. This is consistent with scaling results which show the volume compliance is the dominant pressure mitigator during blowdown. Clarification will be added to this paragraph in Section 9.3.2.1.

S #82 The evaluation model uses clime free convection correlations for the inner containment surface as described in Section 4.4.1 and 4.4.2 for wet stacks and dry stacks, respectively (note the first conductor inner surface forced convection heat transfer multiplier of 10^{-10}). The evaluation model uses the Uchida correlation on internal heat sinks as described in Section 4.2.x.3 for each volume.

S #83 The heat sinks were eliminated entirely from the model.

S #84 a. "blowdown" will be revised to "blowdown pressure history".

b. The major conclusions of Section 9.3.2.1 are directed towards showing the relative insensitivity of the pressure results to the flow paths and heat sinks during blowdown. This section is redundant with Section 8, and the text will be revised to clarify conclusions which can be drawn from Section 8 relative to Section 9. Section 8 sensitivities will be run using the Section 4, Revision 1 evaluation model as a basis for consistency.

S #85 A new figure will be added to Section 9.3.1.1 which shows SG compartment and adjacent compartment pressures and containment pressure versus time.

S #86 a. The last sentence in Section 9.3.2.1 will be revised to "Thus, uncertainties in heat and mass transfer or stratification, and flow path effects, do not significantly impact the AP600 LOCA blowdown pressure history and the evaluation model adequately models the LOCA blowdown phase."

b. A sensitivity to the resistance at the SG exits will be provided to show the effects of different end-of-blowdown distributions on the pressure history during subsequent time phases. Since internal heat sinks saturate well before the maximum pressure, the effects of the post blowdown distributions are expected to be negligible. Figure 9-13 will be revised to be similar to WCAP-14845 Figure 10-4 to more clearly illustrate the effects.

T #87 For the effects of stratification on heat sink utilization, the most significant compartments (heat sinks) are the above-deck region (containment shell) and the CMT room (steel and jacketed concrete). A sentence to this effect will be added to the paragraph discussing stratification in Section 9.3.2.1. A summary of the compartment features is provided in Table 3-1 of WCAP-14812 (Reference 4) and will be added to Section 9. The relative locations of circulating compartments to a LOCA with the jet dissipated in the steam generator compartment will be shown in Figure 9-2.

T #88 Eliminating heat sinks from consideration produces a conservatively high pressure response since the heat sinks remove mass from the atmosphere by condensation. The maximum fill height inside containment is the [illegible] elevation. This corresponds to the bottom of the CMT room. The evaluation model includes the effects of containment filling in nodes which receive water and accounts for submerging walls.

S #89 See the response to #72 which addresses the first part of this question. The CMT room circulation calculations were performed at the time of maximum containment pressure as stated in Section 9.3.1.3. Similar circulation calculations will be included in Appendix A to address the different phases of the LOCA. Section 9.3.2.2 will be revised to summarize the CMT calculations presented in Appendix B.

S #90 As stated in Section 9.3.1.3, the CMT calculations were performed for a time near maximum containment pressure. Related to questions #68 and #89, the CMT

calculations will be included in Appendix A. The text in Section 9.3.1.3 will be revised to summarize and refer to Appendix A for CMT results for each of the post blowdown LOCA phases.

#91 later

S #92 Figure 9-14 shows the steam concentration distributions for the time covering the three phases of the transient. Figure 9-14 is being revised as noted in response to question #94 and #100.

C #93 Discuss at meeting with NRC.

S #94 Figures 9-9 and 9-14 will be updated with the Section 4 Revision 1 evaluation model results and clarified to address the question.

T #95 a. A single node model does not allow for the evaluation of the effects of circulation for a LOCA, or evaluation of the effects of different main steam line break elevations. Therefore, the evaluation model is based on a multi-node network.

b. A summary of experimental data used to determine that a buoyant plume will rise from the break compartment will be provided in Appendix C. It should be noted that the lumped parameter model is not being used to predict a particular realistic scenario on a best estimate basis. Rather, the lumped parameter model is used to perform sensitivities to various circulation patterns (and thus, time-wise development of steam concentrations) for the extreme cases of complete dissipation of momentum. The lumped parameter formulation is consistent with dissipated momentum.

S #96 The title for Figure 9-15 will be revised to indicate that the steam pressure ratios are at 24 hrs. Note that this figure will also be revised in response to question #100.

T #97 The predicted steam concentrations for various representative compartments are shown in Figures 9-9 and 9-14. The focus of 9.3.2.3 is on determining the steam concentrations at 24 hours. The trend towards increased mixing over time in the lumped parameter model leads to a very small density gradient between above-deck and the CMT room at 24 hours. This will be clarified in the text in Section 9.3.2.3.

S #98 The well mixed result is a steam concentration calculation where the same steam/air content from the evaluation model is used to calculate the steam concentration if it were well mixed. The CMT room and above-deck steam concentrations are from the evaluation model which has the penalties on upward facing floors and dead ended compartments. The text will be clarified.

S #99 Refer to the responses to questions #79 and #87.

T #100 a. The evaluation model has a multi-node model of the above-deck region to better

steam concentrations in

allow the code to predict reasonable circulation patterns. There is a predicted deviation between nodes at the operating deck level and dome apex. Figures 9-9 and 9-14 will be revised to more clearly indicate the nodes plotted. The long term results are clearly conservative for the extreme case of well mixed due to the dominance of the PCS surfaces. The evaluation model noding is presented in Section 4. As shown in Section 4, there are 7 horizontal planes between the operating deck and the dome apex. Appendix B will provide the assessment of stratification effects currently contained in Section 9.3.1.3, extended to more clearly address potential stratification in the above-deck region, using similar methods.

T #101 Figures 9-16, 9-18, and 9-20 will be extended to 24 hours. Long term results are conservatively biased with increased mixing. Evaluation of the effects of increased mixing at 24 hours, relative to the evaluation model, is given in Section 9.3.2.3.

S #102 A single node model for the long term phase is not being used. Figures 9-16, 9-18, and 9-20 are from the results of the evaluation model described in Section 4 and the plots will be expanded to 24 hours as discussed in the response to #101.

T Figures 17, 19, and 21

Figures 9-17, 9-19, and 9-21 are snapshots at the time near maximum containment pressure. Figure 9-17 showed flow patterns and steam concentrations for a LOCA with the break momentum dissipated in the broken loop steam generator cavity. This sensitivity case resulted in the highest containment pressure, so additional figures will be added to show how the flow patterns and steam concentrations evolve for three representative times during the event.

Figures 9-17, 9-19, and 9-21 will be revised to show:

a. The location of the break
b. The time of the snapshot
c. A description of the time of the snapshot in the figure title.

The above-deck regions in the figures are simplified representations of the multi-node above-deck model. The evaluation model with the noding described in Section 4 was used. The sketched region representing the above deck region will be revised to more clearly show this.

S #103 Evolution of the flow patterns at various times during the LOCA will be added as discussed in the response to "Figures 17, 19, and 21."

S #104 See the response to question #103.

S #105 The time on Figures 17, 19, and 21 corresponds to the time near maximum containment pressure. Evolution of the flow patterns at various times during the LOCA will be added as discussed in the response to question #103.

T #106 Section 9.3.1.2 showed that the extreme case of an undissipated jet is not limiting.

The lumped parameter model is used to study the hypothetical extreme cases of all momentum dissipated, for which the lumped parameter model is a reasonable tool. Because of plume and wall entrainment rates, which will be discussed in Appendix A for the CMT room and Appendix C for the above-deck region, the axial steam density gradients within the above-deck region and CMT compartment are shallow. After incorporating stratification penalties discussed in Section 9.3.2.4, it is reasonable to use the lumped parameter model to perform sensitivities. Such reasonableness is judged based on comparing expected circulation patterns to those predicted by the model shown in Figures 9-17, 9-19, and 9-21 (see also the response to #107). The results discussed in Section 9 show that a wide range of possible circulation patterns and resulting evolution of steam distributions with time were examined to select the limiting scenario. This discussion will be added to the beginning of section 9.3.2.4 to clarify its purpose.

T #107 The lumped parameter evaluation model over-mixes between adjacent open volume nodes where entrainment dominates, due to numerical diffusion. As discussed in RAI 480.390, which relates to specific test data, the multi-node lumped parameter model over-predicts the amount of mixing between the LST below deck and above-deck regions. This is seen by comparing the slow, long-duration pressure rise predicted by the multi-node lumped parameter model to the observed longer term pressure rise in the test. In other words, the LST shows a continual gradual pressure increase due to longer term non-condensible mixing, and the lumped parameter shows the longer term increase but at a higher rate. It is believed that the lumped parameter rate of pressure increase is due to numerical diffusion, based on the better agreement when the distributed parameter model is used with its higher noding resolution. The time frame for the pressure increase is on the order of hours (both observed and predicted). The segregation between above-deck and below deck concentrations is much less for AP600 due to the presence of flow paths for circulation into the steam generator compartment, so the effect of over-mixing on non-condensible distributions should be less for AP600. (The effect of over-mixing on predicted velocities is eliminated by the use of free convection correlations internal to containment.) Thus, it is concluded that the multi-node lumped parameter model is a reasonable tool to examine circulation and concentration sensitivity to assumed circulation patterns for low momentum (low Fr number) scenarios through the time of maximum pressure at about 20 minutes.

The longer term (through 24 hours and beyond) prediction of the multi-node lumped parameter model tends toward a homogeneous condition as evaluated and discussed in Section 9.3.2.3.

S #108 The case in which the buoyant plume is placed in the CMT room is one extreme case of the sensitivity cases examined with the node network (lumped parameter evaluation) model. As described in Sections 9.3.1.3 and 9.3.2.4, another sensitivity case examined jet dissipation in the vertical access tunnel and the plume is predicted to split between the CMT room and the unaffected SG compartment according to the relative losses in the evaluation model. Note that the node network connections are the same for all of the sensitivities performed. Due to the potential concern raised for flow connections to affect the circulation results in previous

questions, the sensitivities in Section 9 will be rerun using the Section 4 Revision 1 evaluation model.

T #109 The small sensitivity is physically based, and can be expected because the sensitivities show that circulation affects the rate at which the internal heat sinks saturate, but they saturate well before the time of maximum pressure (WCAP-14845, Figure 10-4, Reference 8). Therefore, it is shown that even a rather wide range of transient steam concentrations in the CMT room (Figures 9-17, 9-19, and 9-21) do not have a significant impact on the calculated maximum pressure. Thus, the results are not sensitive to circulation pattern, and the most limiting case has been selected.

S #110 As stated in the response to #108, the flow resistance coefficients were the same for all vents for all three scenarios, and the values are specified in Section 4.

S #111 In the context of Section 9.2.2, a low Fr number is defined as one in which buoyancy dominates and a high Fr is defined as one in which the kinetic energy of the source dominates. The AP600 containment is buoyancy dominated during post-blowdown LOCA, and kinetic energy dominated during LOCA blowdown and MSLB. As discussed in Section 9.2.2, the LST configuration is represented in both buoyancy and kinetic energy dominated situations. The AP600 does not have a significant time frame during which intermediate Fr occurs (WCAP-14845, Section 6.5.2); therefore, the transition from buoyancy to kinetic energy dominated flows has not been specifically studied.

S #112 Mixing as a function of Fr is used to qualitatively assess stratification gradients and to ascertain the relation of LST internal mass transfer data to AP600. No functional relationship between mixing and Fr is required for this approach.

T #113 Experimental data has been discussed in Section 9.2.2. The large scale test (LST) results discussed in Section 9.2.2 and Section 6.5.2 of WCAP-14845 show the density gradient for the above-deck region and between the above-deck and below-deck regions. The LST results indicate some degree of kinetic energy driven circulation below the operating deck grating would occur in the AP600 design for the high Froude number of a main steam line break. This circulation would be difficult to quantify, so no credit is taken for it. A bounding lumped parameter model was used which dissipates kinetic energy in the above-deck nodes and minimizes the predicted ingress of steam into the below deck region as discussed in Section 9.4.2 and shown in Figure 9-25 for the MSLB evaluation model..

S #114 Based on the LST results in Section 9.2.2, kinetic energy is sufficient to drive circulation in the regions below the break and even below the operating deck for a break at the top of the steam generator. See also the response to #103.

S #115 Circulation figures for two additional times during the MSLB transient, which include steam concentrations, will be added.

S #116 It is reasonable to expect that, for the MSLB where internal heat sinks dominate

pressure mitigation, a break in the CMT room which contains the majority of internal heat sinks (WCAP-14812, Table 3-1) would be less limiting than a MSLB in the above-deck region. This was confirmed by calculation as described in Section 9.4.1.2.

S #117 The LST WGOTHIC model was used to gain insight on the noding structure's effects on steam mixing. The evaluation of steam concentration biases in Section 9.4.2 is based on an examination of what the model predicted, recognizing that the lumped parameter model inherently cannot predict kinetic energy effects. Therefore, the objective is to compare MSLB lumped parameter results to each other for cases with various assumed break elevations. The resulting conservative predicted stratification between above- and below-deck is shown in Figure 9-25. Due to the lack of a simulated steam generator flow path in the LST model, Westinghouse will replace the LST model studies in Section 9.4.2 with equivalent studies using the more appropriate AP600 evaluation model.

S #118 The lumped parameter model dissipates momentum in a node. Therefore, buoyancy is the only driving force, and buoyancy promotes upward flow of the steam. The dissipation of momentum in a node, therefore, does not promote over-mixing in a downward direction.

S #119 As discussed in the response to question #117, the study in Section 9.4.2 will be replaced with an equivalent study using the AP600 evaluation model.

S Figure 23
Figure 9-23 is not clear. However, as discussed in response to #117, an equivalent study based on the evaluation model will be performed and the text and figures will be clarified.

S #120 The evaluation model, described in Section 4, will be used to rerun the sensitivity study in Section 9.4.2 (see response to question # 117). The locations for the steam concentrations plotted will be provided.

S #121 The text will be clarified relative to the assumed break location.

T #122 Mixing is not a phenomena itself, but is a qualitative term. Westinghouse is revising Section 9 to more clearly focus on circulation and stratification as they affect steam concentrations and resulting mass transfer. Appendix C will be provided giving a summary of test data available from which to judge acceptability of the circulation predicted by lumped parameter modeling.

T #123 The multi-node lumped parameter evaluation model introduces biases in two regions: the multiple-node above-deck region and the one-node-per-compartment below deck region. The biases due to use of lumped parameter can be summarized as follows:

<u>Above-deck region lumped parameter biases</u>

Jet/plume entrainment is over-predicted, leading to under-prediction of the axial steam gradient and over-prediction of velocities.

Because of the short-time constant for circulation and large volumetric entrainment rates in the above-deck region, the bias on concentration gradient is not large. However, an extreme gradient, well beyond what would be expected based on LST results has been studied and has shown a weak sensitivity of total heat removal to the assumed gradient.

The lumped parameter velocity bias is eliminated by the use of free convection heat and mass transfer on all internal surfaces.

Below deck compartment region

Each compartment below deck is modeled with a single homogeneous fluid node. This represents a bias relative to axial concentration gradients which may exist in a compartment. When the plume is assumed to enter the CMT room, which has the largest group of heat sinks in the below deck region, the CMT room has been shown to have a short time constant for circulation due to high volumetric flows entrained into the plume so that the CMT room stratification gradients are shallow. Thus, the gradients within the CMT room deviate from well mixed by a small amount. An extreme case of stratification was studied for the effects on heat sink utilization, as discussed in Section 9.3.1.3, and the CMT room heat sink utilization was relatively insensitive to the two stratification scenarios.

Upward-facing floors

Although not related to the use of lumped parameter, a bias in the evaluation model exists relative to the liquid film thickness on upward facing floors that do not accumulate deep pools (surfaces that are designed to drain to lower elevations). As discussed in Section 9.3.2 and WCAP-14812, Section 4.4.3C, horizontal surfaces facing up may build up films thick enough to degrade heat transfer into the solid surface. Therefore, a bias has been introduced into the evaluation model by modeling upward facing surfaces as insulated to conservatively bound this effect.

Other DBA model biases

The DBA lumped parameter model contains additional biases due to input values specifically targeted to bound the significant uncertainties. These input biases are described or examined in WCAP-14407, Tables 4-102, 4-103, 4-104, 4-105, 4-106, 4-107, 4-108, 5-1, 10-1, 14-1, 14-2, 14-3, 14-4, and 14-5, and Section 7.4.2.

S #124 By demonstrating that the model predicts less steam circulation below the break location than would be expected, the model is conservatively limiting the access of steam to internal heat sinks located below the operating deck. This results in a conservative containment pressure for the transient which bounds the effects of

stratification. As discussed in the response to question #117, the LST studies will be replaced by more appropriate studies using the evaluation model.

S #125 The conclusion is based on the steam concentrations in the above-deck nodes, as predicted by the model, being nearly uniform as compared to LST data (Figure 9-1) which shows mixing below the operating deck. This will be addressed more clearly in the revision described in the response to question #117.

S #126 The elevation of the evaluation model assumed break node will be added to the text and the node will be identified relative to the model described in Section 4. The results of the evaluation model study will show that predicted mixing occurs throughout the AP600 above-deck region based on Froude number comparisons, but limits access of steam below-deck.

S #127 The LST results show that the above-deck region and regions below-deck are well mixed for a MSLB. Placing the break at the node above the operating deck in the evaluation model bounds the physics of the AP600 design. This break location provides conservative results as compared to the LST data which shows mixing below the deck. Locating the break at a higher elevation in the model would increase the maximum predicted containment pressure, however, the data shows that the operating deck location sufficiently bounds the circulation and stratification expected for a high kinetic energy break.

S #128 Refer to the responses to questions #117 and #118. The phrase "conservatively limits steam access" will be revised to state that the noding structure conservatively reduces steam access to the below deck heat sinks by dissipating momentum which reduces the circulation to nodes below the break.

S #129 All upward facing surfaces, including the operating deck are neglected. When applicable for a given node, this is discussed in the "Special Modeling Assumptions" subsections in Section 4 for each compartment. See also the response to #91.

S #130 For Figure 9-25, the figure title will be corrected and the 4 curves will be identified.

S #131 The MSLB evaluation model differs from the LOCA model, described in Section 4, only in the mass and energy boundary conditions and the assumed break location node.

S #132 a. Refer to the response to question #126.

b. Refer to the response for question #127 for a discussion on the selection of the assumed break location in the evaluation model.

S #A1 We concur that the LST does not cover the blowdown phase of a LOCA. Section 9.2.19 will be revised to clarify which LOCA phases are relevant to the LST.

S #A2 The last sentence in Section 9.2.2 will be revised to "The high degree of mixing for test 222.4 compared to test 222.2 is due to the high kinetic energy of the injected

fluid because that is the only significant difference between the two tests."

S #A3 Appendix C will be added to Section 9 to summarize applicable conclusions from the HDR studies.

S #A4 Section 9.2.1 will be revised to refer to Section 6.5.3 of WCAP-14845, and the specific LOCA tests will be listed in Section 9.2.1.

S #A5 The text in Section 9.3.1.2 has already been corrected from 4% to 8%. The text will be clarified as follows: "From the point of view of pressure mitigation, Section 8.5 of WCAP-14845 shows that volume compliance is the most significant factor during blowdown. Figure 9-11 shows that, during blowdown, the internal heat sinks absorb only 8% of the total integrated break energy released during the first 3000 seconds of the transient."

S #A6 Three references will be added to the text in Section 9.3.1.3. They are References 5, 6, and 7 at the end of these responses.

S #A7 Refer to the response to question #70.

S #A8 Refer to the response to question #83.

S #A9 Yes, the evaluation model uses the Uchida correlation which has been shown to be conservative by approximately a factor of 2 or more. The last sentence in Section 9.3.2.1 will be revised as follows: "Thus, uncertainties in heat and mass transfer or stratification and flow path effects do not significantly impact the AP600 LOCA blowdown pressure history."

S #A10 Refer to the response to question #121.

S #A11 The sentence referred to in Section 9.4.2 will be reworded as suggested for clarification.

<u>References</u>

1. NSD-NRC-96-4763, Docket No.: STN-52-003, July 1, 1996,"Assessment of Mixing and Stratification Effects on AP600 Containment."

2. NRC informal review questions - Part 1 from D. Jackson (NRC) to J. Butler (Westinghouse), 8-22-96; Part 2 from D. Jackson (NRC) to J. Butler (Westinghouse), 11-8-96.

3. WCAP-14407, "WGOTHIC Application to AP600," September 1996.

4. WCAP-14812, "Accident Specification and Phenomena Evaluation for AP600 Passive Containment Cooling System," December 1996.

5. Jaluria, Y. "Buoyancy Driven Wall Flows in Enclosure Fires," Twenty-first Symposium (International) on Combustion, The Combustion Institute, 151-157 (1986).

6. Goldman, D., Jaluria, Y., "Effect of Opposing buoyancy on the Flow in Free and Wall Jets," Journal of Fluid Mechanics, 166, 41-56 (1986).

7. Jaluria, Y., Cooper, L.Y., "Negatively Buoyant Wall Flows Generated in Enclosure Fires," Progress in Energy and Combustion Science, 15, 159-182 (1989).

8. WCAP-14845, "Scaling Analysis for AP600 Containment Pressure During Design Basis Accidents," February 1997.

Westinghouse

FAX COVER SHEET

RECIPIENT INFORMATIO:		SENDER INFORMATION	
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Cover + Pages 1 + 5

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COMMENTS:

DIANE

HERE ARE THE MARKUPS THAT SHOULD RESOLVE ITEMS 472 AND 1172 FROM OUR 3/19/97 PHONE CALL. THEY WILL GO INTO REVISION 12 UNLESS WE HEAR FROM YOU. WE WILL CHANGE THEIR "W STATUS" TO "CLOSED." PLEASE CALL WITH THE NEW "NRC STATUS."

Jim Winters

cc: LINDGREN
MCINTYRE
CUMMINS
RONVIJUK
ISRAELSON
WINTERS
JEANNE EVANS

Criteria Section	Referenced Criteria	AP600 Position	Clarification/Summary Description of Exceptions
C.1.1.3		Conforms	The auxiliary building ~~that~~ a contains the liquid radwaste system ~~tanks~~ is designed to Seismic Category I criteria. The Seismic Category I structure will retain the maximum liquid inventory of the ~~tanks~~ system. The lowest level of the auxiliary building, elevation 66'6", contains the liquid radwaste system effluent holdup tanks, waste holdup tanks, a monitor tank and chemical waste tank within a common flood zone. This flood zone has watertight floors and walls. The enclosed volume within this flood zone is sufficient to contain the contents of the ~~tanks.~~ system ¶ ~~Within the flood zone, the effluent holdup tanks, waste holdup tanks, chemical waste tank, and monitor tank are each in a separate room, which is accessed by climbing down a ladder. The only point in the auxiliary building with a lower elevation is the auxiliary building sump. The tank rooms each have one or two floor drains that lead to the sump. Tank overflows or spills will be collected in the auxiliary building sump. The sump is automatically pumped to a waste holdup tank.~~ Two liquid radwaste system monitor tanks are three levels up at elevation 100"-0". Overflows or spills from these monitor tanks drain by gravity down through the drain system to a waste holdup tank. ~~These monitor tanks are each in separate rooms, which are accessed by climbing down a ladder. The tank rooms each have one floor drain that leads to a waste holdup tank.~~ # The Seismic Category I criteria exceed the operating basis earthquake required by regulatory position C.5 of Regulatory Guide 1.143.
C.1.1.4		Conforms	Components in the liquid radwaste systems are nonseismic. They are not required to be designed for seismic loads.
C.1.2.1		Conforms	Atmospheric tanks in the liquid radwaste system have level sensors, transmitters, and alarms. Local alarm is not provided because the tanks are located in shielded areas that are not normally occupied by people.

Criteria Section	Referenced Criteria	AP600 Position	Clarification/Summary Description of Exceptions
			The areas do not contain any radioactive materials. The potential for the transfer of radioactive contamination from adjacent areas is prevented through interfaces with only clean areas and through general pressurization of the annex building. No potentially contaminated ductwork is contained in the areas. The ventilation system serving the radioactive waste areas is designed to maintain a slightly negative pressure. The exhaust from this system is through the monitored plant vent.
C.1.2.5		Conforms	This guideline does not apply because the liquid radwaste treatment system has no outdoor tanks. No other outside tanks store radioactive fluids.
C.2.1.1	Regulatory Guide 1.143, Table 1	Conforms	Components in the gaseous radwaste systems are designed and tested to the requirements set forth in the codes and standards listed in Table 1 of Regulatory Guide 1.143. Heat exchangers are designed and built according to ASME, Section VIII, Div. 1 and TEMA (for shell and tube). Piping and valves are per ANSI B31.1. Pumps are according to manufacturer's standards.
C.2.1.2	ASME Code, Section II	Conforms	Materials, except elastomers for gaskets, seals, seats, diaphragms, and packing, are provided in accordance with the ASME Code Section II when the ASME Code is the design and fabrication standard. Piping and valves materials are per ASTM specifications consistent with ANSI B31.1. Pump materials are provided according to manufacturer's standards.
C.2.1.3		Conforms	The guard bed and the delay beds, including supports, in the gaseous radwaste system are designed for seismic loads according to the requirements of Regulatory Guide 1.143. These are the only AP600 components used to store or delay the release of gaseous radioactive waste. The beds are located in the seismic Category I auxiliary building at elevation 66'-6". Seismic loads for this equipment will be established using one-half of the safe shutdown earthquake (SSE) floor response spectra. The loads due to this seismic response spectra are equivalent or greater than those due to an operating basis earthquake (OBE). ~~The~~ Other equipment and supports will be

Criteria Section	Referenced Criteria	AP600 Position	Clarification/Summary Description of Exceptions
			designed in accordance with the ~~commercial~~ codes ~~(ANSI B31.1 and AISC)~~ indicated in Table 3.2-3.
C.3.1		Conforms	The regulatory guidance applies to the AP600 solid waste processing system except for components and subsystems used to solidify or concentrate liquid waste. The AP600 solid waste processing system does not have these components/subsystems. These functions are provided by contractors who process these wastes using mobile systems.
C.3.1.1	Regulatory Guide 1.143, Table 1	Conforms	The solid radwaste system is designed and tested to the requirements set forth in the codes and standards listed in Table 1 of Regulatory Guide 1.143. The spent resin tanks are designed and tested in accordance with ASME Code, Section VIII, Div. 1. Piping and valves are designed and tested according to ANSI B31.1. The pumps are designed to manufacturers' standards and tested in accordance with the Hydraulic Institute standards.
C.3.1.2	ASME Code, Section II	Conforms	Materials, except elastomers for gaskets, seals, seats, diaphragms, and packing, are provided in accordance with the ASME Code, Section II when the ASME Code is the design and fabrication standard. Piping and valves materials are per ASTM specifications consistent with ANSI B31.1. Pump materials are provided according to manufacturer's standards.
C.3.1.3		Conforms	The Seismic Category I auxiliary building will retain the maximum liquid and spent resin inventory of the spent resin tanks. The Seismic Category I criteria exceed the operating basis earthquake required by regulatory position C.5 of Regulatory Guide 1.143.
C.3.1.4		Conforms	The equipment and components used to collect, process, and store solid radwaste are nonseismic as permitted by this paragraph.
C.4.1	Regulatory Guide 8.8	Conforms	Standard safety analysis report section 12, "Radiation Protection," discusses the measures taken to maintain the radiation exposure to personnel as low as reasonably achievable.

Criteria Section	Referenced Criteria	AP600 Position	Clarification/Summary Description of Exceptions
C.4.2		Conforms	The quality assurance program applied to the radwaste systems is discussed in Chapter 17. This quality program meets the requirements of Regulatory Guide 1.143.
C.4.3	ASME Code, Section IX	Conforms	Pressure-containing components in the radwaste systems are of welded construction to the maximum practical extent. Flanged joints and quick connect fittings are used only where maintenance or operational requirements indicate that they are preferable. Screwed connections are not used except for some instrumentation and vents and drains where welded construction is not suitable. Process lines are 1 in. or larger. Butt welds are used in process lines, which contain radioactive fluids. Nonconsumable backing rings are not used in process piping welds. Process pipe welding is performed as required by ANSI B31.1. Component welding is performed as required by the applicable construction code.
C.4.4		Conforms	Hydrostatic testing is performed as required by the applicable construction codes.
C.4.5		Conforms	In-service testing of the containment penetrations and isolation valves is performed as described in standard safety analysis report section 3.9.6. Other tests, on nonsafety equipment, are performed on an item-by-item basis as judged necessary to confirm proper operation of the systems.
C.5.1.1	Regulatory Guide 1.60	Exception	The operating basis earthquake has been eliminated from the AP600 design basis. See the discussion of conformance to C.2.1.3.
C.5.1.2	AISC-1969, Section 1.5.6, ACI 318-77	Exception	Regulatory Guide 1.143 endorses AISC-1969 (Reference 47) and Reference 45 that have been superseded by AISC-1989 (Reference 48) and Reference 46. The AP600 uses the latest version of the industry standards (as of 1/90). These versions are not endorsed by a regulatory guide but their use should not result in deviation from the design philosophy otherwise stated in Regulatory Guide 1.143.

Replace with copy of ¶ in C.6 on next page

Criteria Section	Referenced Criteria	AP600 Position	Clarification/Summary Description of Exceptions
C.5.1.3		Conforms	The construction and inspection requirements of AISC-1989 and ACI 318-89 are followed as appropriate.
C.5.2	Regulatory Guides 1.60 & 1.61 Table 1	Exception	Those portions of the radwaste systems that require seismic design by Regulatory Guide 1.143 are housed in the auxiliary building that is Seismic Category I. Certain portions that do not require seismic design (for example, dry solid radwaste storage) are housed in the radwaste building, which is nonseismic.
C.5.3		Conforms	Shield structures, if used, will comply with Regulatory Guide 1.143, position C.5.2.
C.6	ANSI N199-1976/ ANS-55.2	Conforms	The quality assurance program, as outlined in Chapter 17 of the standard safety analysis report and applied to the radwaste systems, meets the requirements of Regulatory Guide 1.143, position C.6.

Copy this into C.4.2 on previous page as indicated.

Reg. Guide 1.144 - Withdrawn

Reg. Guide 1.145, Rev. 1, 11/82 - Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants

Criteria Section	Referenced Criteria	AP600 Position	Clarification/Summary Description of Exceptions
General		N/A	The atmospheric dispersion factors for use in determining potential accident consequences are selected to be representative of existing nuclear power plant sites and to bound the majority of them. Chapter 2 provides the interface criteria. Therefore, this regulatory guide is not applicable to AP600 design certification.

Reg. Guide 1.146 - Withdrawn

Reg. Guide 1.147, Rev. 8, 11/90 - Inservice Inspection Code Case Acceptability ASME Section XI Division 1

Criteria Section	Referenced Criteria	AP600 Position	Clarification/Summary Description of Exceptions
General	ASME Code, Section XI	Conforms	

** TX CONFIRMATION REPORT ** AS OF MAR 21 '97 13:32 PAGE.01

AP600 DESIGN CERT

	DATE	TIME	TO/FROM	MODE	MIN/SEC	PGS	STATUS
01	3/21	13:29	#23:NRC	G3--S	02'52	06	OK

Fax to: **Bill Huffman**

Subject: Response to AP600 SSAR Chapter 18 Minimum Inventory NRC Review Comments

cc: Steve Kerch
Chuck Brockhoff
Mike Corletti
Ken Deutsch

Bill - Here is the Westinghouse response to those two questions regarding AP600 SSAR Chapter 18, Minimum Inventory.

1) whether the steam generator wide range level variable category should be D1 or D2 and/or a B

2) meeting R.G. 1.97 on the letdown radiation monitor

Please review the attached write-up and let me know if it's ok for formal transmittal or if another telecon is required. Note that because it is written as the text would appear in a formal letter, no mention is made of Vogtle or Comanche Peak (i.e., reference is made to "recent plants"), but I wanted you to know they were the model plants.

I'll talk with you soon,

Robin Nydes

Robin 3/18

1. **NRC Question:**

"Should steam generator wide range level be a D1 or a D2 variable? What is the basis for leaving it as a D2 or making it a D1? And, should it also be a B variable (it is noted as an A1 in PAMS for Vogtle); why or why not?"

Westinghouse Response:

The Westinghouse position for post-accident monitoring (PAMS) variables is that there are no Category 1, Types D variables. This position has been accepted by the NRC and licensed for Westinghouse PWRs.

The basis for this approach is that when considering all PAMS variables, Type D variables fundamentally provide indirect or backup indication related to the satisfaction of true design safety goals, identified as critical safety functions and fission product barrier challenges that are monitored by Type B and C variables. From this overall perspective, the Type D variables provide backup indication to the Type B and C variables, by monitoring the operation of safety-related systems and equipment that are used to achieve the critical safety functions and protect the fission product barriers.

If no Type D variables were available following an event at a Westinghouse PWR, the operators would still have the total inventory of Category 1 and 2, Type B and C variables to directly monitor the status of the critical safety functions and the fission product barriers. In addition, the Type B and C variables indirectly confirm the satisfactory operation of the safety-related systems (that are used to achieve the critical safety functions and protect the fission product barriers) monitored by the Type D variables.

The inventory of Type D variables in Westinghouse PWRs essentially serve as "backup indications" for the more important Type B and C variables. Therefore, the Westinghouse approach, which has been accepted by the NRC, is not to mandate the imposition of Category 1 design requirements for Type D variables which only monitor the operation of the safety-related systems and equipment. (The primary differences between Category 1 and 2 variables are related to classification of power supplies, seismic qualification, and single failure response.) Therefore, there have been no D1 variables for the Westinghouse plants.

Consistent with the Westinghouse approach accepted for current Westinghouse PWRs, the AP600 steam generator (SG) wide range level is a Type D, Category 2 (D2) variable instead of Category 1 (D1). This is consistent with SSAR Table 7.5-1 (sheet 1 of 12, Revision 10).

SG wide range level is not a Type B variable for two reasons. First, the design of the AP600 is fundamentally different from current plants where SGs and auxiliary feedwater provide the core heat sink and SG level is used to monitor the status of this heat sink.

For AP600, the Passive Residual Heat Removal heat exchanger (PRHR Hx), not the SGs, is the safety-related heat sink. There is no safety-related Auxiliary Feedwater System in the AP600. The startup feedwater system that provides steam generator feedwater is nonsafety-related and does not provide the design basis heat sink monitored by the Type B variables for the AP600. The PRHR Hx parameters (such as flow and outlet temperature) and IRWST level are the AP600 B1 variables used to monitor the status of the safety-related heat sink. These are listed in SSAR Table 7.5-5 for heat sink, equivalent to the SG level and associated variables (auxiliary feedwater flow, etc.) in current plant PAMS.

Secondly, SG wide range level could be considered a Type B variable for monitoring RCS inventory (e.g., to monitor for SG overfill following a SG tube rupture event with the SG safety valves sticking open). Some older plants have included SG wide range level as a Type B variable under RCS inventory. However, the current licensing approach approved by the NRC for recent plants does NOT list SG wide range level as a backup variable for RCS inventory.

Westinghouse has adopted the most current licensing approach for AP600 and, therefore, SG wide range level is NOT included as a Type B variable for RCS inventory control.

2. **NRC Question:**

"How does Westinghouse meet R.G. 1.97 on the letdown radiation monitor?"

Westinghouse Response:

Table 2 of RG 1.97 identifies the following variables to monitor for a fuel cladding fission product barrier breach during and following an accident:

- Core Exit Temperature (Type C, Category 1)
- Radioactivity Concentration or Radiation Level in Circulating Primary Coolant (Type C, Category 1)
- Analysis of Primary Coolant (Type C, Category 3)

The AP600 specifies the following post-accident monitoring variables for fuel cladding barrier integrity:

- Core Exit Temperatures (Type C, Category 1)
- Analysis of Primary Coolant (manual sample) (Type C, Category 3)

The AP600 approach, to take credit for measurement of primary coolant radioactivity levels in the analysis of the primary coolant and not to specifically include this second item in the RG 1.97 PAMS variables, is consistent with the approach licensed in other recent Westinghouse plants.

Traditionally, nuclear power plants designed by Westinghouse have taken exception to the second item in RG 1.97 as a PAMS variable and this has been accepted by the NRC in the most recent licensing submittals. In particular, recent plants have not included the "radioactivity concentration or radiation level in circulating primary coolant" as a separate PAMS variable. Instead, they have taken credit for the primary coolant analysis to determine the "radioactivity concentration in circulating primary coolant."

These recent plants include another backup variable, Reactor Vessel Level Indication (RVLIS) as a backup indication to monitor for fuel cladding breach. Comparably, the AP600 has reactor vessel - hot leg water level indication that is used as a Type B, Category 2 variable for Reactor Core Cooling and a Type B, Category 3 for Reactor Coolant Inventory Control. This variable is not identified as a Type C, Category 2 variable for monitoring Fuel Clad Breach since it cannot be used to satisfy the monitoring requirement associated with radioactivity

levels in the coolant. While hot leg level can function as a backup to indicate potential to establish conditions where a cladding breach could develop, as in current plants that use RVLIS, hot leg level need not be included as a Type C variable in the AP600 PAMS since primary sampling is considered an adequate backup indication to core exit thermocouples.

AP600 does not have a radiation monitor in the purification loop (equivalent to the chemical and volume control system letdown line in current plants). While some Westinghouse plants have a radiation monitor installed in the letdown line for the chemical and volume control system, this monitor is unavailable to provide post-accident indication for current plants since the letdown line has safety-related automatic isolation following certain design basis plant events. Therefore, this monitor cannot be credited for post-accident monitoring of circulating reactor coolant radioactivity levels.

Westinghouse

FAX COVER SHEET

RECIPIENT INFORMATION		SENDER INFORMATION	
DATE:	MARCH 21 1997	NAME:	JIM WINTERS
TO:	TOM KENYON	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

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

COMMENTS:

TOM

THE ATTACHED MARKUP OF THE PMS ITAAC SECTION SHOULD RESOLVE OPEN ITEM 1044 AGAINST CHAPTER 7. WE ARE CHANGING "W STATUS" TO "CONFIRM W". PLEASE CALL AND TELL US WE CAN CHANGE "NRC STATUS" TO "ACTION N" OR "CONFIRM W"

Jim Winters

cc: SCHREIBER
CUMMINS
MCINTYRE
NYDES
DEUTSCH
WINTERS

2.5.2 Protection and Safety Monitoring System

Design Description

The protection and safety monitoring system (PMS) initiates reactor trip and actuation of engineered safety features in response to plant conditions monitored by process instrumentation and provides safety-related displays. The functional arrangement of the PMS is depicted in Figure 2.5.2-1.

1. The PMS has the equipment identified in Table 2.5.2-1.

2. The seismic Category I equipment, identified in Table 2.5.2-1, can withstand seismic design basis dynamic loads without loss of safety function. [Revise item 3 and add new item 4 as shown on change attachment]

3. The Class 1E equipment, identified in Table 2.5.2-1, can withstand the electromagnetic interference (EMI) and radio frequency interference (RFI) conditions that would exist before, during, and following a design basis accident without loss of safety function for the time required to perform the safety function.

5. ~~4.~~ a) The Class 1E equipment, identified in Table 2.5.2-1, is powered from their respective Class 1E division.

   b) Separation is provided between PMS Class 1E divisions, and between Class 1E divisions and non-Class 1E cable.

6. ~~5.~~ The PMS provides the following safety-related functions:

   a) The PMS initiates an automatic reactor trip, as identified in Table 2.5.2-2, when plant process signals reach specified limits.

   b) The PMS initiates automatic actuation of engineered safety features, as identified in Table 2.5.2-3, when plant process signals reach specified limits.

   c) The PMS provides manual initiation of reactor trip and selected engineered safety features as identified in Table 2.5.2-4.

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

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

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

CHANGE ATTACHMENT TO ITAAC FOR THE
PROTECTION AND SAFETY MONITORING SYSTEM
SECTION 2.5.2

Design Description

REVISE ITEM NUMBER 3 TO BE AS FOLLOWS:

3. The Class 1E equipment, identified in Table 2.5.2-1, has electrical surge withstand capability (SWC), and can withstand the electromagnetic interference (EMI), radio frequency interference (RFI), and electrostatic discharge (ESD) conditions that would exist before, during, and following a design basis accident without loss of safety function for the time required to perform the safety function.

ADD THE FOLLOWING NEW ITEM NUMBER 4:

4. The Class 1E equipment, identified in Table 2.5.2-1, can withstand the room ambient temperature, humidity, and mechanical vibration conditions that would exist before, during, and following a design basis accident without loss of safety function for the time required to perform the safety function.

REVISE ITEM NUMBER 9b (formally item 8b) TO BE AS FOLLOWS:

9. b) The PMS two-out-of-four initiation logic reverts to two-out-of-three coincidence logic if one of the four channels is bypassed. The PMS automatically produces a reactor trip or engineered safety feature initiation upon an attempt to bypass more than two channels of a function that uses two-out-of-four initiation logic.

REVISE AND EXPAND ITEM NUMBER 11 (formally item 10) AS FOLLOWS:

11. The PMS hardware and software is developed using a planned design process which provides for specific design documentation and reviews during the following life cycle stages:

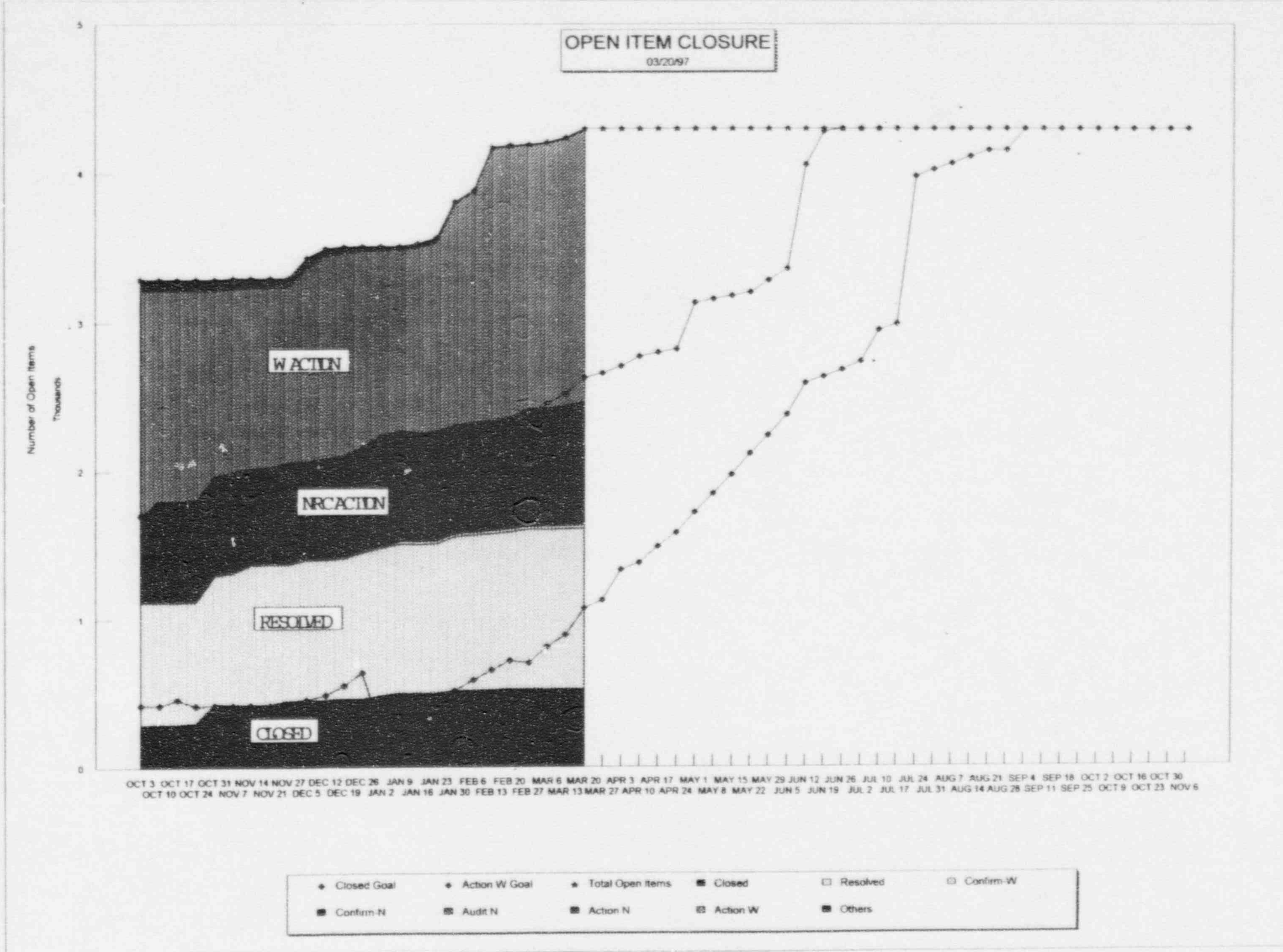
    a) Design requirement phase
    b) Definition phase
    c) Development phase
    d) Test phase
    e) Installation phase
    f) Operation and maintenance phase.

12. The PMS software is designed, tested, installed, and maintained using a process which incorporates a graded approach according to the software's relative importance to safety and specifies requirements for:

    a) Software management including documentation requirements, standards, review requirements, and procedures for problem reporting and corrective action

    b) Software configuration management including historical records of software and control of software changes

    c) Verification and validation including requirements for reviewer independence.



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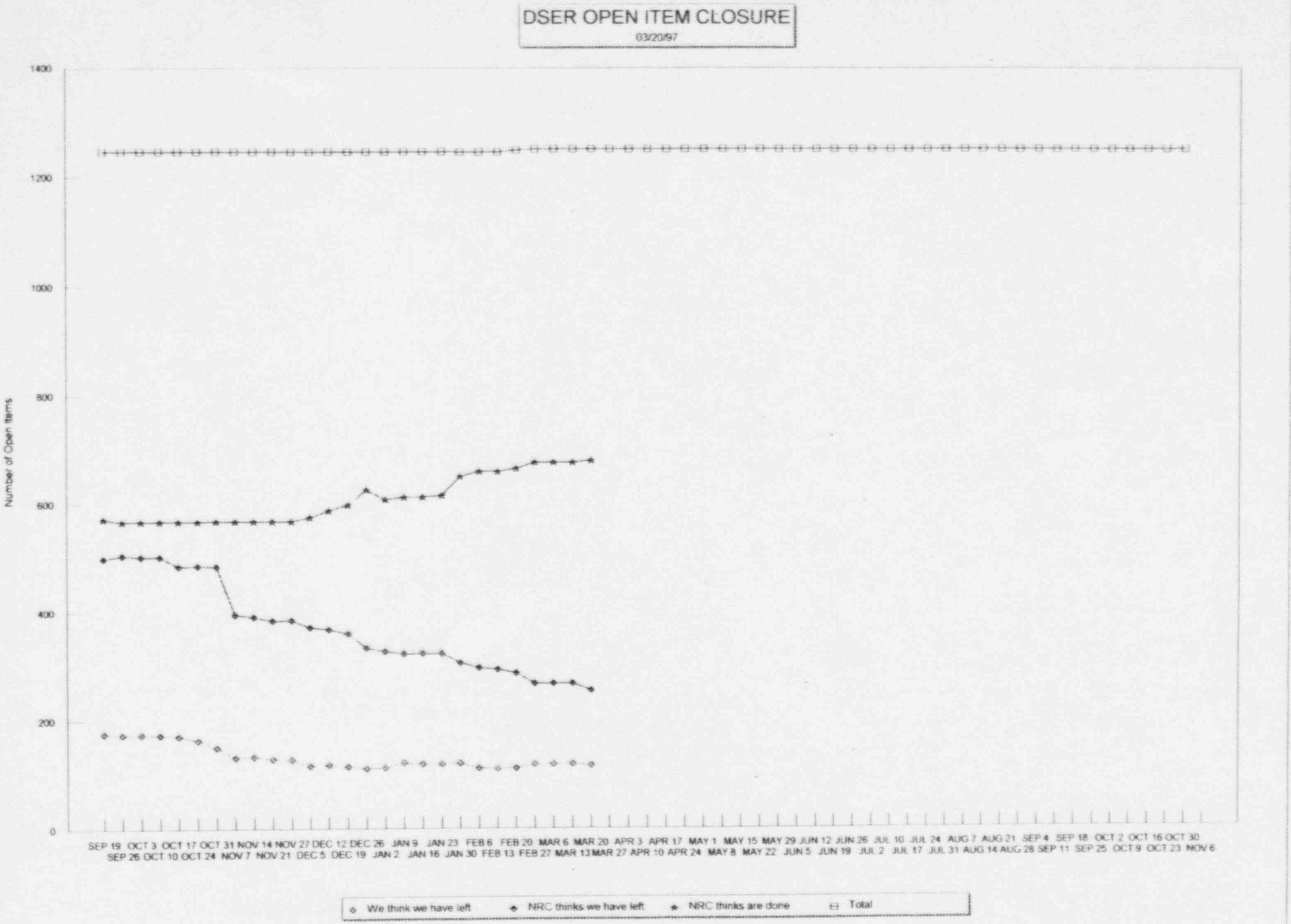
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We think we have left
NRC thinks we have left
NRC thinks are done
Total

Westinghouse

FAX COVER SHEET

RECIPIENT INFORMATION		SENDER INFORMATION	
DATE:	MARCH 20, 1997	NAME:	Jim Winters
TO:	DIANE JACKSON	LOCATION:	ENERGY CENTER - EAST
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LOCATION:	301-415-2002		

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

COMMENTS:

DIANE,

HERE IS INFORMATION ON THE ITEMS WE TOOK AS ACTIONS FROM OUR 3/7/97 MEETING. UNLESS WE HEAR FROM YOU SOON, WE WILL ATTACH THESE TO A LETTER STATING WE HAVE COMPLETED ALL ACTIONS ON FIRE PROTECTION.

Jim Winters

cc: McINTYRE
CUMMINS
HUTCHINGS
VIJUK (BOB)
WINTERS

OI-306. NRC's concerns regarding AP600 safe shutdown capabilities during and/or after a design basis fire.

I. AP600 Compliance with Regulations for Passive ALWRs

A. Compliance with BTP CMEB 9.5-1

SSAR Table 9.5.1-1, note the following:

Item 16 AP600 uses two levels of damage limits: safe shutdown and design basis accidents *(in contrast to three in App. R; hot shutdown, cold shutdown, and DBAs).* Safe shutdown capability is protected from damage caused by a single fire.

(AP600 "safe shutdown" is deemed equivalent to cold shutdown. In SECY-94-084, it is recognized that Passive ALWR designs are limited by the inherent ability of the passive heat removal processes. EPRI's position is that safe stable shutdown condition is at 420 °F, and that passive safety systems need not be capable of achieving cold shutdown, based on the belief that the passive decay heat removal systems have inherently high long-term reliability.

The NRC Staff position is that an RHR system be able to bring the plant to cold shutdown conditions (with reference to GDC 34 and RG 1.139) was established to enable the licensee to perform inspection and repair at the plant. The Staff believes that other plant conditions may constitute a safe shutdown state as long as reactor subcriticality, decay heat removal, and radioactive materials containment are properly maintained for the long term.

<u>The Staff recommends that the Commission approve the EPRI's proposed 420 degrees F or below, rather than the cold shutdown condition required by RG 1.139, as a safe stable condition which the passive decay heat removal systems must be capable of achieving and maintaining following non-LOCA events. This recommendation is predicated on an acceptable passive safety system performance and an acceptable resolution of the issue of regulatory treatment of non-safety systems.)</u>

The Commission has made determinations on the Staff's recommendations (SECY-94-084) regarding Safe Shutdown Requirements (item C) in the memorandum of June 30, 1994 from John C. Hoyle to James M. Taylor.

Item 25 Safe shutdown systems are protected such that reliance on alternative or dedicated shutdown capability is not necessary *(in contrast with App. R allowed Alternative or Dedicated Shutdown).*

(AP600 safe shutdown capabilities include methods for using safety-related systems only, safety-related and nonsafety-related systems, or nonsafety-related systems only. The safe shutdown capabilities using safety-related systems are fully protected to ensure that at least one safe shutdown capability is available in the event of a single fire, without taking any credit for repairs or operator actions in the fire-affected area, and all equipment within the fire area is rendered inoperable by the fire. AP600 does not rely on an alternative or dedicated shutdown capability.)

B. SECY-90-016 Evolutionary LWR Certification Issues (Jan. 12, 1990)
(re-iterated in SECY-93-087)

"D. Fire Protection"

The evolutionary ALWR designers must ensure that safe shutdown can be achieved, assuming that all equipment in any one fire area will be rendered inoperable by the fire and that re-entry into the fire area for repairs and operator actions is not possible. The control room is excluded from this approach, provided an independent alternative shutdown capability that is physically and electrically independent of the control room is included in the design.	AP600 complies. At least one of safe shutdown capability using safety-related systems (SSAR 7.4.1.1) is available in the event of a single fire, without requiring repairs or operator actions in the fire-affected area.
Evolutionary ALWRs must provide fire protection for redundant shutdown systems in the reactor containment building that will ensure, to the extent practicable, that one shutdown division will be free of fire damage.	AP600 complies. SSAR 9A.3.1.1 provides the FHA and Safe Shutdown Analysis.
Additionally, the evolutionary ALWR designers must ensure that smoke, hot gases, or the fire suppressant will not migrate into other fire areas to the extent that they could adversely affect safe-shutdown capabilities, including operator actions.	AP600 complies. Fire-smoke dampers are utilized to minimize migration of the effects of fire through the shared HVAC. (See attached sketch and response to OI-323 located elsewhere in this letter.)

C. April 26, 1990 Staff Letter to the Commission, Re. Evolutionary LWR Certification Issues and Their Relationship to Current Regulatory Requirements

"6. Fire Protection"

Proposed enhancements that represent a significant improvement in physical separation requirements and in the need to consider the effects of smoke, heat, and fire suppressant migration into other areas. In particular, redundant train separation is likely to be the most significant feature leading to reduced fire risk.	AP600 complies. Fire-smoke dampers are utilized to minimize migration of the effects of fire through the shared HVAC.

D. Additionally, in the **NRC Request for Additional Information**, RAI 280.12, it was stated that ...."based on its review of Section 7.4, the staff has determined that safe shutdown as defined above can be achieved within 36 hours following a fire event using only safety-related equipment and can be maintained long-term (i.e., beyond 36 hours after it has been achieved) provided: (1) the safety-related passive systems used for safe shutdown perform their intended function; (2) nonsafety-related equipment are available for long-term maintenance of safe-shutdown; and (3) all staff's concerns identified in the following sections are resolved. For the above reasons, consistent with SECY-94-084 (approved by Commission, see SRMs dated June 30, 1994 and June 28, 1995) position on safe shutdown requirements for passive plant designs, the staff accepts safe shutdown as defined above as a safe stable condition for AP600, subject to an acceptable passive system performance and an acceptable resolution of the issue of regulatory treatment of non-safety systems (RTNSS). ...."

"<u>Response</u>:
Westinghouse concurs with the definition of safe shutdown presented in this Request for Additional Information."

For discussion purposes only, the following AP600 comparison with App. R and other regulations/guidelines for LWRs is presented. No regulations require such comparison as App. R is not applicable to AP600.

A. AP600 short-term safe shutdown capability (to be initiated following a design basis fire-event, when using safety-related systems only) includes maintaining the reactor subcritical, the reactor coolant average temperature less than or equal to no load temperature, and adequate coolant inventory and core cooling. The long-term safe shutdown conditions are the same as the short-term safe shutdown conditions except that the coolant temperature shall be less than 420 °F. This long-term condition must be achieved (using safety related equipment) within 36 hours and maintained indefinitely. (SSAR 7.4)

Based on the above:

(1) AP600 long-term safe shutdown condition shall be deemed equivalent to cold shutdown discussed in Appendix R of 10CFR50. SECY-94-084 confirms this equivalency.

(2) App.R III.G.1.b. "Systems necessary to achieve and maintain cold shutdown from either the control room or emergency control station(s) can be repaired within 72 hours" is not applicable to AP600.

(3) App.R III.G.3 that provides "Alternate or dedicated shutdown capability" is also not needed for AP600, because on AP600 none of the following App.R conditions exist; a) protection of systems whose function is required for hot shutdown does not satisfy the requirement of III.G.2, and b) where redundant trains of systems required for hot shutdown located in the same fire area may be subject to damage from fire suppression activities or from the rupture or inadvertent operation of fire suppression systems. However, the App.R requirement of "In addition, fire detection and a fixed fire suppression system shall be installed in the area, room, or zone under consideration" has been implemented for AP600 Containment fire area. Fire detectors and standpipe and hose coverage are provided.

See also AP600 SSAR Table 9.5.1-1 (Comparison with BTP CMEB 9.5-1 Guidelines) Item 25. Remarks: "Safe shutdown systems are protected such that reliance on alternative or dedicated shutdown capability is not necessary".

GL 86-10 that provides interpretations of App. R regarding the Alternative or Dedicated Shutdown is also not applicable on AP600. And the interpretation of App. R "free of fire damage" is not applicable, because on AP600, all equipment within the fire area are rendered inoperable by the fire, in compliance with SECY-90-016.

(4) App.R.III.L. detailing the requirements of Alternative and dedicated shutdown capability provided for a specific fire area is not applicable to AP600. Item (3) above confirms that AP600 does not require an Alternative or dedicated shutdown capability.

B. App.R Fire Damage Limits for hot shutdown safety function is "One train of equipment necessary to achieve hot shutdown from either the control room or emergency control station(s) must be maintained free of fire damage by a single fire, including an exposure fire", and for cold shutdown is "Both trains of equipment necessary to achieve cold shutdown may be damaged by a single fire, including an exposure fire, but damage must be limited so that at least one train can be repaired or made operable within 72 hours using onsite capability".

AP600 complies with the fire damage limits for hot shutdown safety function, and even better for cold shutdown function, because AP600 safe shutdown systems (using safety-related systems) are fully protected such that reliance on repairs of fire-damaged equipment within 72 hours is not necessary. See also AP600 SSAR Table 9.5.1-1 (Comparison with BTP CMEB 9.5-1 Guidelines) Item 16, Remarks: "AP600 uses two levels of damage limits: safe shutdown and design basis accidents. Safe shutdown capability is protected from damage caused by a single fire."

Also, GL 86-10 that provides interpretations of App. R regarding the allowed repair of cold shutdown equipment is also not applicable on AP600.

C. Based on the discussions above, a major fire involving the Turbine Building with its potential consequences of loss of the nonsafety-related RHR systems should not be a nuclear safety issue on AP600. It may become a property loss prevention issue, especially from the plant underwriter's perspective. However, AP600 turbine building fire protection is consistent with the current industry standards, such as, NFPA 803, 804, 850 and EPRI NP-4144 (July 1985).

OI-314. **AP600 Fire Detection and Alarm System**

1. Compliance with BTP CMEB 9.5-1 (see SSAR Table 9.5.1-1)

112. Fire detection systems should be provided for areas that contain or present a fire exposure to safety-related equipment.	AP600 complies.
113. Fire detection systems should comply with the requirements of Class A systems as defined in NFPA 72D and Class I circuits as defined in NFPA 70.	AP600 complies, no exception to NFPA 70 & 72D is identified.
114. Fire detectors should be selected and installed in accordance with NFPA 72E.	AP600 complies, no exception to NFPA 72 is identified.
115. Testing of pulsed line-type heat detectors should demonstrate that the frequencies used will not affect the actuation of protective relays in other plant systems.	AP600 complies.
116. Fire detection systems should give audible and visual alarm and annunciation in the main control room.	AP600 complies.
117. Where zoned detection systems are used in a given area, local means should be provided to identify which zone has actuated.	AP600 complies.
118. Local audible alarms should sound in the fire area.	AP600 complies.
119. Fire alarms should be distinctive and unique so they will not be confused with any other plant system alarms.	AP600 complies.
120. Primary and secondary power supplies, which satisfy the provisions of section 2220 of NFPA 72D, should be provided for the fire detection system and for electrically operated control valves for automatic suppression systems.	AP600 complies, no exception to NFPA 72D is identified.

2. Disposition of NRC concerns

1. Fire detection in safety related areas is seismically qualified.	AP600 does not intend to have a seismically qualified fire detection system. There is no requirement to make the fire detection system to withstand design basis earthquake, or to remain functional following a seismic event. However, with microprocessor-based equipment and components the fire detection system can be expected to be unaffected by earthquakes, and the potential of spurious actuation should be minimal.

OI-321. AP600 Fire Pumps

I. Compliance with BTP CMEB 9.5-1 (see SSAR Table 9.5.1-1)

128. A sufficient number of pumps should be provided so that 100 percent capacity will be available assuming failure of the largest pump or loss of offsite power.	AP600 complies. one electric motor-driven and one diesel-driven fire pump, 100% capacity each, are provided.
129. Individual fire pump connections to the yard fire main loop should be separated with sectionalizing valves between connections.	AP600 complies. Fire pump discharge lines are re-arranged so they are individual connections to the yard fire main, with sectionalizing valves between connections.
130. Each pump and its driver and controls should be separated from the remaining fire pumps by a 3-hour rated fire wall.	AP600 complies. Each fire pump and its controller are located in a room enclosed by 3-hour fire barrier.
131. The fuel for the diesel fire pump should be separated so that it does not provide a fire source exposing safety-related equipment.	AP600 complies. Fuel oil day tank is in the diesel engine-driven fire pump room that is enclosed by 3-hour fire barrier. The fuel oil storage tank is located outdoor.
132. Alarms indicating pump running, driver availability, failure to start, and low fire main pressure should be provided in the main control room.	AP600 complies. Refer to P&ID FPS M6-001.
133. The fire pump installation should conform to NFPA 20.	AP600 complies. No exception to NFPA20 is identified. (See Table 9.5.1-3.)

2. Disposition of NRC concerns

1. Can the yard fire main be supplied by the fire pumps if there is fire damage to the yard main extension inside the Turbine Building?	Yes. The fire pumps can still supply the yard main and the yard main extension can be isolated if it is damaged by the fire.
2.Will the air intake of the diesel fire pump be affected by the turbine building fire?	No. The fire pump diesel-engine driver outside air intake is located at the same side of the building with the other HVAC outside air intakes, remote from the discharge points including smoke relief through the roof. Additionally, the intake is located within the envelope of the fire pump fire area.
3. Will the power supply of the electrical motor-driven fire pump survive a fire in the Turbine Building?	Fire pump motor supply is designed and routed in conformance with NFPA 20 and NFPA 70. Since routing from the non-diesel bus is less susceptible to a turbine building fire and a diesel back fire pump is not required if a diesel fire pump is installed, the present SSAR 9.5.1.2.3 wording "The motor-driven fire pump is supplied with power from the diesel-backed non-Class 1E switchgear", will be revised in SSAR Rev. 12 to indicate the motor-driven fire pump is not on the plant diesels.
4. Are the fire tanks too close to the Turbine Building? Could they be damaged by a turbine building fire?	4. As noted in meetings with the NRC, this is not a licensing issue but an insurance issue. Westinghouse is presently reviewing its design in this area to determine its insurance liabilites.

OI-322. AP600 Basis for Selecting NFPA 14, Class II, Standpipe and Hose Stations

BTP 9.5-1 guidelines recommend installation of standpipe and hose stations that meets the requirements of NFPA 14, however, it does not call for a specific class of standpipe system per NFPA 14 to be provided. NFPA 14 provides three classes for a standpipe system based on its intended use (for the manual firefighting efforts), however, it too does not specify specific applications, buildings or facilities where such classes of standpipe systems should be provided.

As stated in SSAR 9.5.1.2.1.5, the AP600 fire protection standpipe and hose systems are provided for each building, for Class II service in accordance with NFPA 14, i.e., primarily intended for use primarily by the building occupants or by the fire department (plant fire brigade) during initial response. Each hose reel or rack contains up to 100 ft. of 1-1/2 in. fire hose.

AP600 fire hazard and protection analyses (SSAR section 9A) showed that in the nuclear island the postulated fires are primarily fires involving electrical equipment and cables or ordinary class A combustibles such as paper or trash. There are no insitu flammable liquids or gases expected to be present within the nuclear island. Consistent with the postulated fire characteristics and manual extinguishing techniques, plus the AP600 fire areas compartmentalization and configurations, a Class II standpipe with 1-1/2" hoses is deemed most practical as it can be safely used by the plant fire brigade without undue damages to nonfire-affected facilities and equipment.

In the Turbine Building, recognizing that the postulated fires may involve flammable liquids or gases, such as, lubricating fluid, hydraulic fluid, hydrogen, etc., the hose stations are provided with a 2-1/2 in. angle valve. A 2-1/2 to 1-1/2 hose coupling is installed at the hose rack, together with the up to 100 ft. of 1-12 in. fire hose. Hence the fire brigade has the option to breakaway the hose coupling, and attached their portable 2-1/2 in. hoseline in order to obtain a greater flow rate.

In either case, the fire brigade can also supplement the interior hosestreams by using the additional 2-1/2 inch hoses that are connected to the nearest hydrant(s), should it become necessary.

OI-323. AP600 Protection from Smoke Spread

1. Compliance with BTP CMEB 9.5-1 (see SSAR Table 9.5.1-1)

99. Smoke and corrosive gases should be discharged directly outside to an area that will not affect safety-related plant areas.	AP600 complies, smoke exhaust outlets are located remote from outside air intake openings to preclude recirculation of smoke into the buildings.
100. To facilitate manual firefighting, separate smoke and heat vents should be provided in certain areas.	AP600 complies, smoke and heat venting capability is provided as described in App.9A, Fire Protection Analysis.
101. Release of smoke and gases containing radioactive materials to the environment should be monitored.	AP600 complies.

2. Compliance with SECY-90-16, Evolutionary LWR Certification Issues (Jan.12, 1990)

Additionally, the evolutionary ALWR designers must ensure that smoke, hot gases, or the fire suppressant will not migrate into other fire areas to the extent that they could adversely affect safe-shutdown capabilities, including operator actions.	AP600 complies. Fire-smoke dampers are utilized to minimize migration of the effects of fire (smoke and hot gases) through the shared HVAC ductwork that serves a single train of safety-related equipment rooms (Div. A&C, or Div. B&D).

3. Disposition of NRC concerns

1. Define the detector to smoke damper logic.	Opening and closing of the combination fire/smoke dampers will be controlled by the fire detection system that comprises of area detectors and in-duct detectors. The area smoke detectors will initiate the closing of the fire/smoke dampers. A high temperature override will close the damper when the in-duct secondary high temperature sensor senses a temperature higher than the fire damper's fusible link rating.
2. Define the location of smoke detectors used for damper control.	Area smoke detectors will be used for controlling the opening/closing of the combination fire/smoke dampers. The concept is to close the smoke damper and to isolate the fire-affected room at the early stage of the fire, and as soon as smoke is developed and detected by the area detector(s), while allowing the HVAC system to continue running and providing pressurization of the non-affected rooms.
3. Describe overall smoke control philosophy, logic and implementation.	Smoke control logic will be integrated with the corresponding HVAC control logic. Upon detection of smoke in a room, that room will be immediately isolated by using the combination fire/smoke dampers. Meanwhile, the HVAC is designed to continue running to serve the other non-affected rooms and to provide ambient pressurization that will help confine smoke and hot gases within the fire-affected room. For post-fire recovery the fire/smoke damper will be reopened and smoke removal will be accomplished by running the HVAC system in a once-through mode.

4. Describe in details the re-opening of fire and smoke dampers.	Reopening the combination fire/smoke dampers can be accomplished from a remote location, i.e. from the fire alarm and control panel(s). However, when the damper is closed due to high temperature, resetting the high temperature sensor needs to be made at the damper. Therefore, on AP600 the damper actuators and controller will be located outside of the fire-affected areas, either in the corridor ceiling or in the mechanical equipment room

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Westinghouse

FAX COVER SHEET

RECIPIENT INFORMATION		SENDER INFORMATION	
DATE:	MARCH 20, 1997	NAME:	TIM WINTERS
TO:	DIANE JACKSON	LOCATION:	ENERGY CENTER - EAST
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COMPANY:	US NRC	Facsimile:	win: 284-4887 outside: (412)374-4887
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DATE:	MARCH 20, 1997	NAME:	Jim Winters
TO:	Tom Kenyon	LOCATION:	ENERGY CENTER - EAST
PHONE:	FACSIMILE:	PHONE:	Office: 412-374-5290
COMPANY:	USNRC	Facsimile:	win: 284-4887 outside: (412)374-4887
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COMMENTS:

Tom

HERE IS THE INFORMATION WE THINK C. HINSON ASKED FOR TO RESOLVE ITEM 5011 ON THE FUEL TRANSFER PORT WATER BLADDER. PLEASE CALL IF YOU OR CHARLEY HAVE ANY MORE QUESTIONS.

Jim Winters

cc: LINDGREN
MCINTYRE
CUMMINS
MAHLAO
SEJUAR
MEUSBERKE
WINTERS

Open Item No. 5011 requests that Westinghouse provide drawings to show the design of the water-filled bladders that will provide shielding in the 2 inch seismic gap between the end of the fuel transfer tube shielding and the steel containment.

The water-filled bladders will provide shielding in this area during fuel transfer and will accommodate relative movement between the containment and the concrete transfer tube shielding during reactor operation. The location of the water-filled bladders is shown on Sketch 970312, and outline Sketches 970313 and 970317 shows the size and configuration of the bladders.

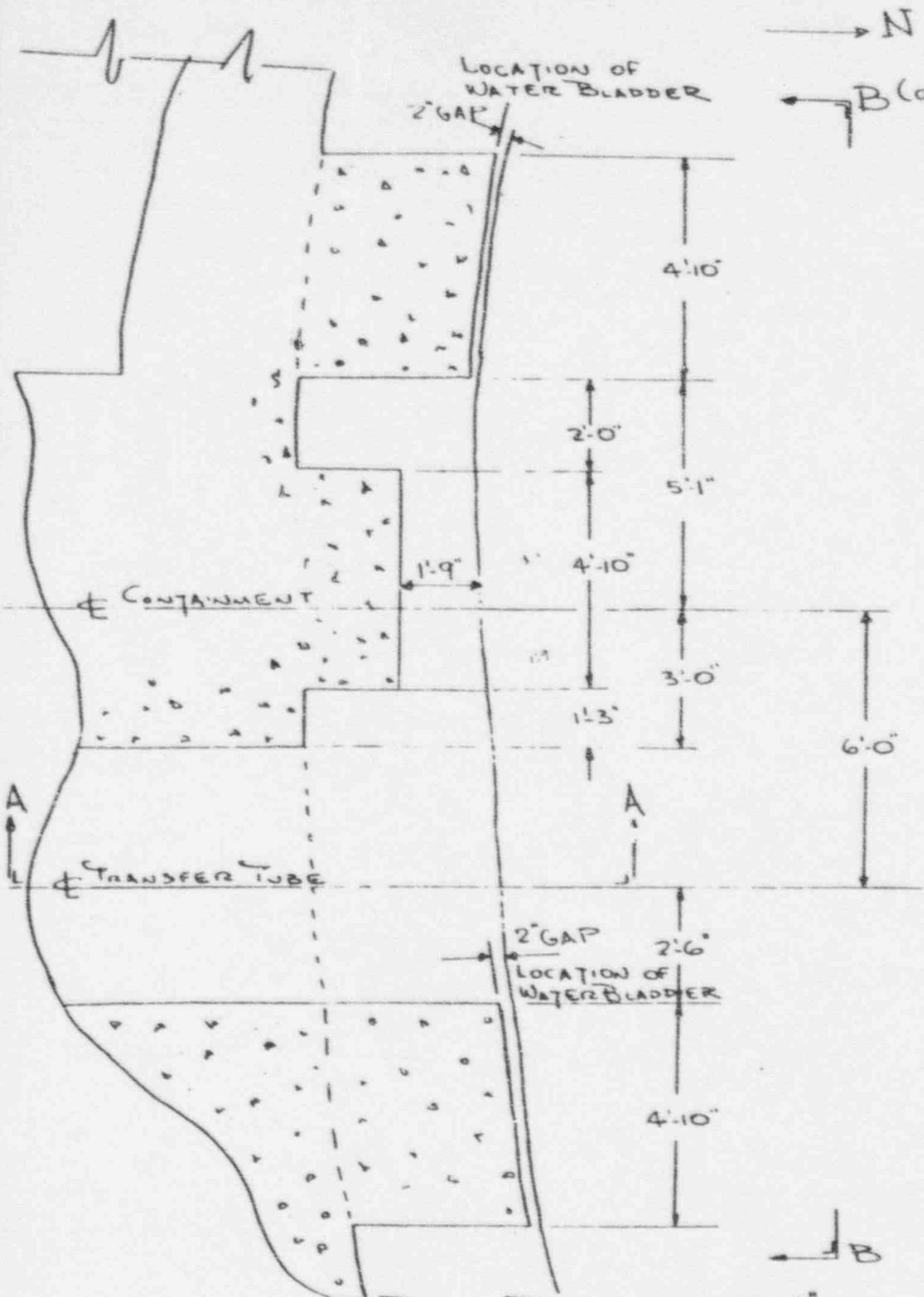
The shields are totally passive and are flexible to adapt to varying gaps. The shields will be fabricated using a 2-ply bladder with an inner rubberized bladder for long term fluid containment and an outer bladder for puncture and abrasion resistance. The bladders will be supported by a structural angle which will sit on top of the concrete shielding as illustrated on the sketches. The shields will withstand the radiation exposure expected during the transfer of spent fuel assemblies(i.e. approximately 10^7 rads) during the plant design life.

Dose rates outside of the fuel transfer tube shield walls are expected to be less than 15 millirem/hr during fuel transfer operations. Similar levels are expected outside of the gap between the shield walls and the containment liner with the water-filled shield bladder in place.

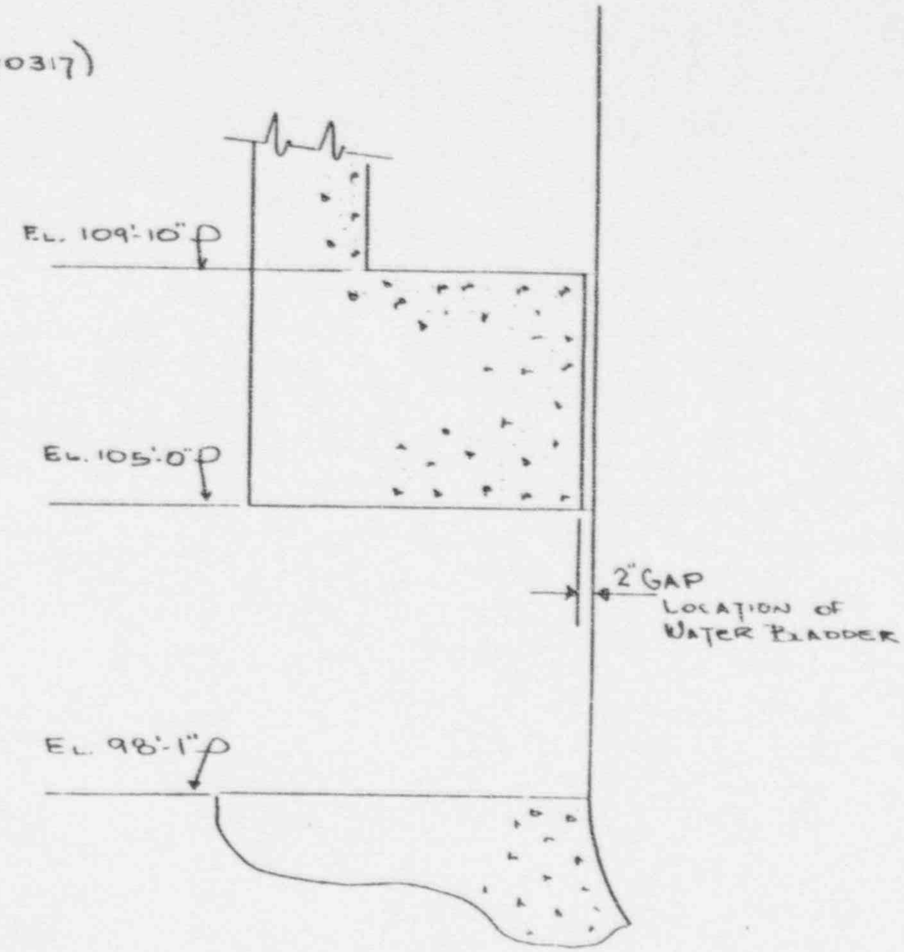
Personnel access to the middle annulus area, where the transfer tube is located, will be minimal with highly restrictive personnel access controls. As shown on SSAR Figure 12.3-3 (Sheet 6 of 16), there is a locked entrance at the only access point to the middle annulus area (room no. 12354) outside the annulus. Further, there is a fire door (always closed) at the entrance to the middle annulus at elevation 107'-2".

The fuel transfer tube shield has a removable shield plug/hatch that allows for periodic inspection of the fuel transfer tube's welds. The opening of this hatch is near the west wall of the shield at elevation 109'-10" and will always be locked except during transfer tube inspection activities when fuel transfer operations are not in progress. The hatch will be administratively controlled and treated as an entrance to a very-high radiation area under 10 CFR 20. Also, the shield design includes a shadow shield between the ladder below the hatch and the transfer tube. This serves as a labyrinth shield and avoids a direct line of sight to the transfer tube during egress from the area around the transfer tube.

J. Sejvar / R. Meuschke 3/17/97



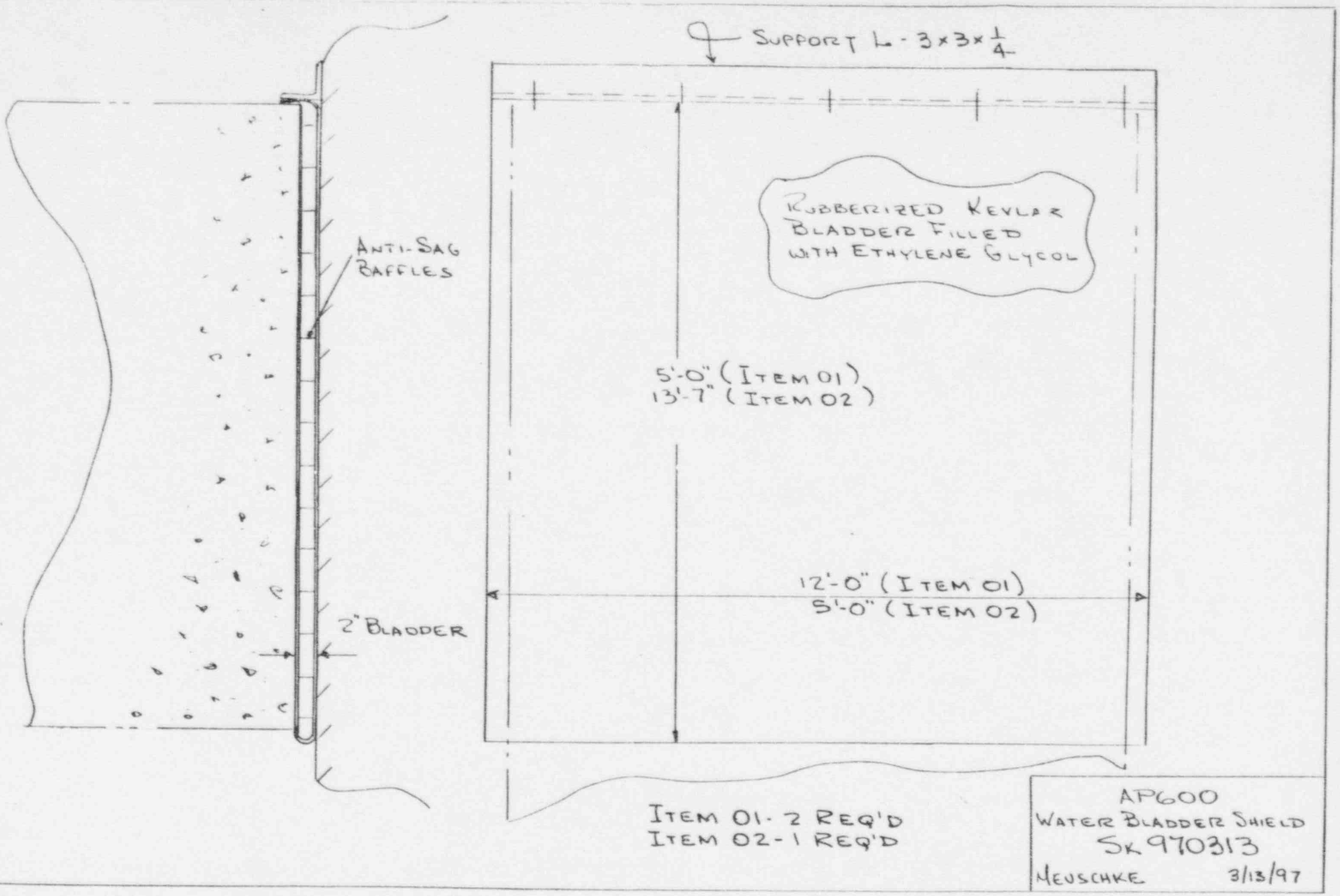
PARTIAL PLAN @ EL. 104'-1 1/2"



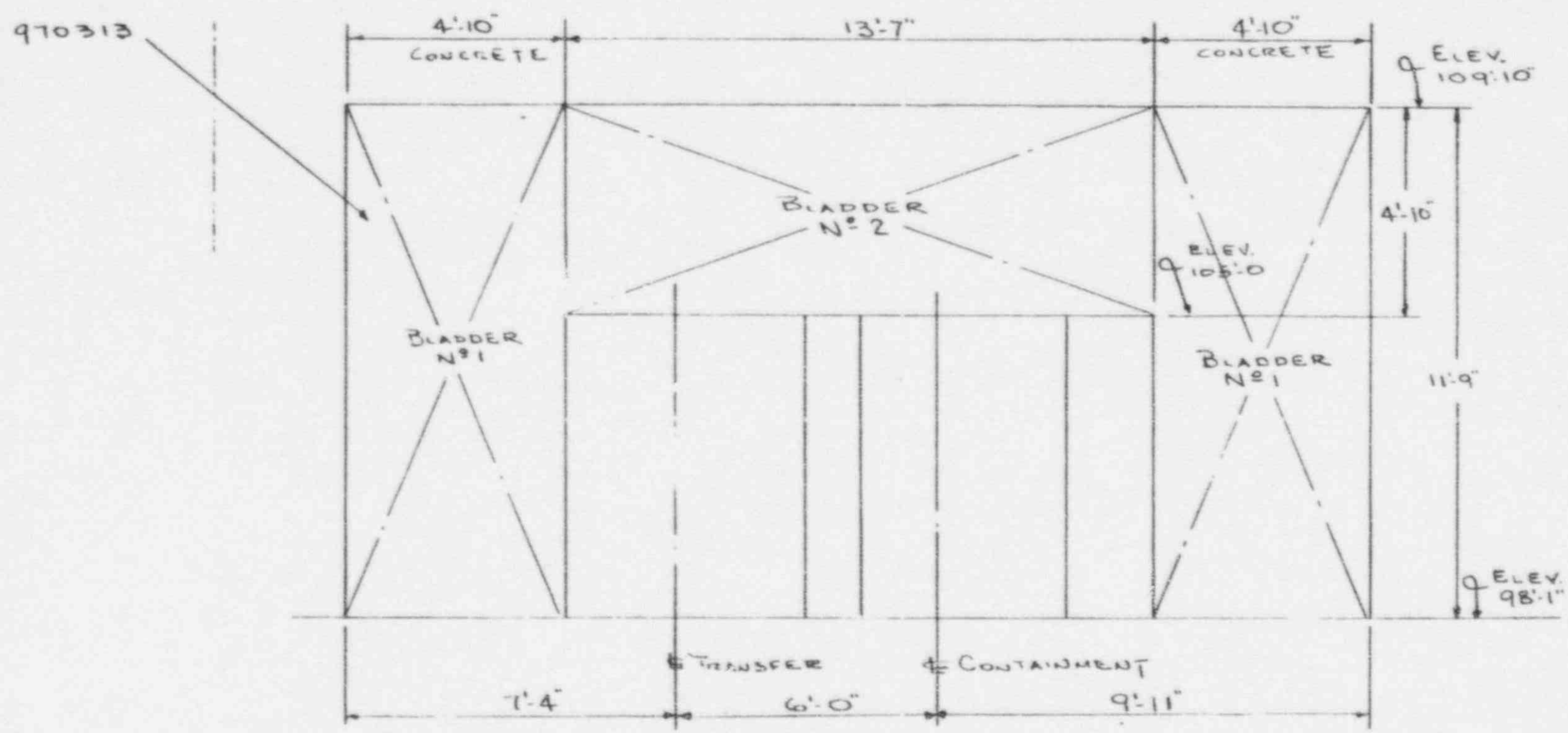
SECTION A-A

SCALE 1/2" = 1'-0"

AP600
FUEL TRANSFER TUBE SHIELDING
LOCATION OF WATER BLADDERS
SK 970312
HEUSCHKE 3/12/97



Support L - 3×3×1/4
Rubberized Kevlar Bladder Filled with Ethylene Glycol
Anti-Sag Baffles
5'-0" (Item 01)
13'-7" (Item 02)
12'-0" (Item 01)
5'-0" (Item 02)
2" Bladder
Item 01 - 2 Req'd
Item 02 - 1 Req'd
AP600
Water Bladder Shield
SK 970313
Meuschke 3/13/97



View B-B (from 970312)

Scale $\frac{1}{2}$" = 1'-0"

AP600
Fuel Transfer Tube Shielding
Water Bladder Arrg't
SK 970317

Meuschke 3/17/97

To: Cliff Fineman (INEEL) fax 208-526-2930

Subject: Use of Quench Model in NOTRUMP Final Validation Report
Date: March 19, 1997
Pages: 30 counting this page

During the Westinghouse/NRC meeting on 3/13/97, we discussed the use of the "quench" model in the NOTRUMP Final Validation Report. Ralph Landry asked that we fax a copy of the key overheads related to the quench model to you.

Attached are the key overheads.

Included is:

1) (page 2) a brief history of the use of the quench model
2) (page 3) a list of the tests analyzed with NOTRUMP and the test conditions (3 runs were done for each case - base, high leakage, low leakage)
3) (pages 4 - 8) five sample plots from the Final Validation Report showing how our calculations for G2 compare to the data
4) (page 9) summary table of simulation results with final code version and quench model off compared to preliminary code version with quench model on
5) (pages 10 - 20) sample comparison plots for cases with final code version and quench model off versus cases with preliminary code version with quench model on
6) (pages 21 - 28) comparison plots including cases with final code version and quench model on
7) (pages 29 and 30) summary slides

Feel free to call me to discuss this information. I can be reached at 412-374-4255.

Bob Osterrieder

Bob Osterrieder
Westinghouse Electric

3/19/97

Brian,
Carl said to give you this copy incase you wanted to include it in your informal NRC file. I faxed this to Fineman this morning.

Bob Osterrieder
x 4255

USE OF QUENCH MODEL

- o Quench model developed for cases with core nodes that uncover then have some recovery
- o Quench model originally used for all reported G2 runs
- o Quench model originally used for all reported ACHILLES runs
- o Code changes during project along with test cases indicated no need for use of quench model in final calculations therefore, quench model description not included in report
- o In closing out documentation, repeated calculations with final code version, final options (quench model off, birthing off)
- o Repeat of base ACHILLES calculations with quench model off showed no difference in results
- o Repeat of ACHILLES noding study Section 4.3.4 (4, 12, 24 & 48 nodes) showed no difference in results except for 4 node case only (mixture level spikes - 4 node case uses quench model to eliminate spikes)

  Section 4.3.4 concludes that we will use approximately 1 foot axial noding for NOTRUMP simulations of heated bundles or cores. Therefore, 4 node case not used to justify bigger nodes and not important to support final model.
- o Quench model NOT USED for any reported OSU or SPES-2 runs
- o Repeat of all G2 calculations indicated need for quench model for a few cases

TABLE 4.4-3 G2 LOOP CORE UNCOVERY TEST PARAMETERS				
Run Number	Pressure (psia)	Bundle Power (MW)	Initial Bundle Water Level (in.)	Tests Analyzed with NOTRUMP
715	779	0.603	114	✓
716	775	0.252	138	✓
717	796	0.905	102	
718	799	1.258	90	
719	394	0.267	138	✓
720	395	0.615	114	✓
721	394	0.914	102	
722	395	1.264	84	
723	395	0.614	114	
724	96	0.252	126	✓
725	96	0.599	96	✓
726	96	0.857	84	
727	97	1.247	78	
728	50	0.596	84	✓
729	50	0.250	114	✓
730	50	0.894	66	
731	50	1.244	54	
732	15.1	0.254	102	✓
733	15.8	0.600	72	
734	16.1	0.900	60	
735	16.7	1.249	54	
736	15.3	0.253	102	

4

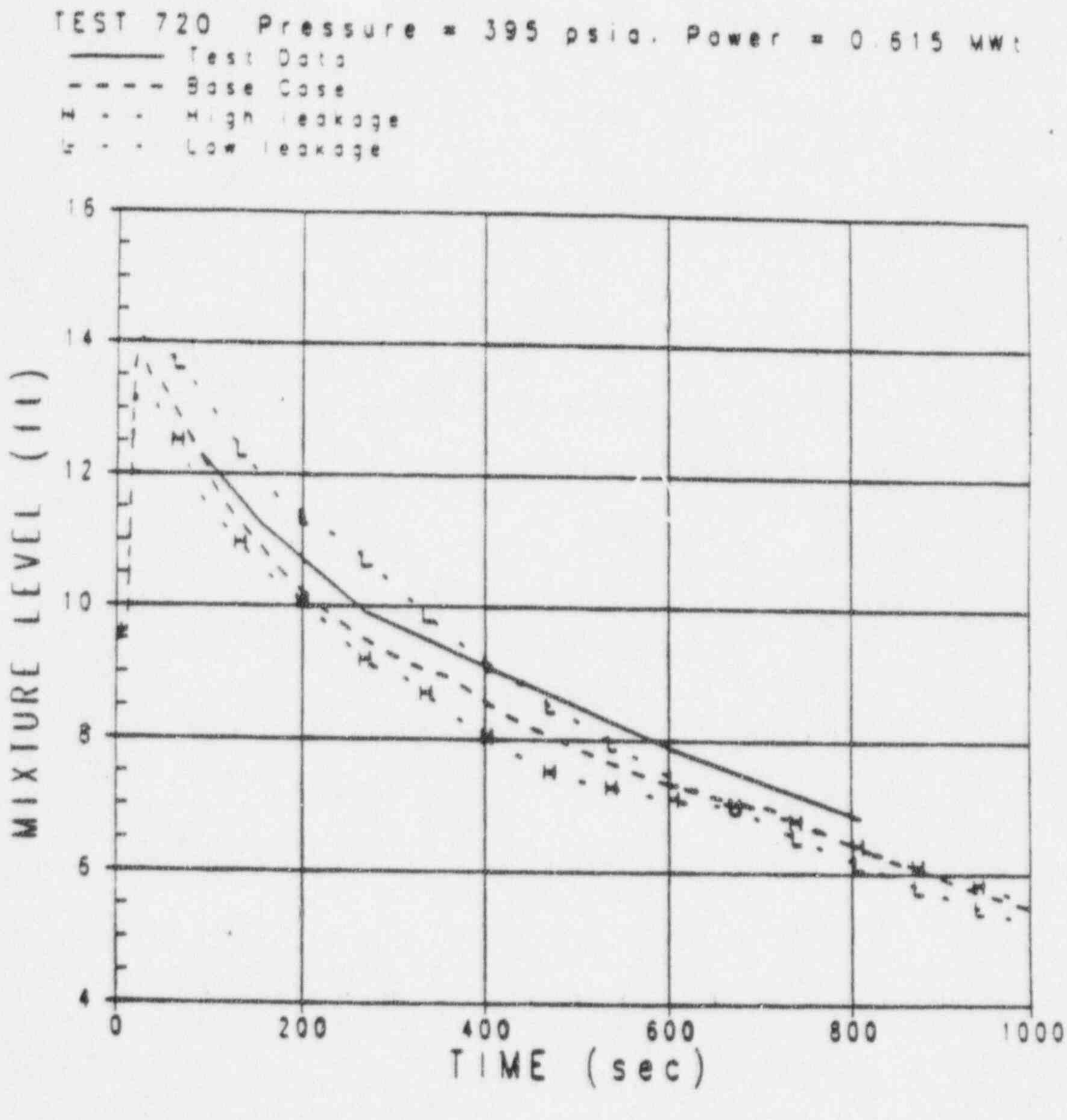


Figure 4.4-26 NOTRUMP Comparisons to G2 Test 720 Mixture Height with Uncertainties

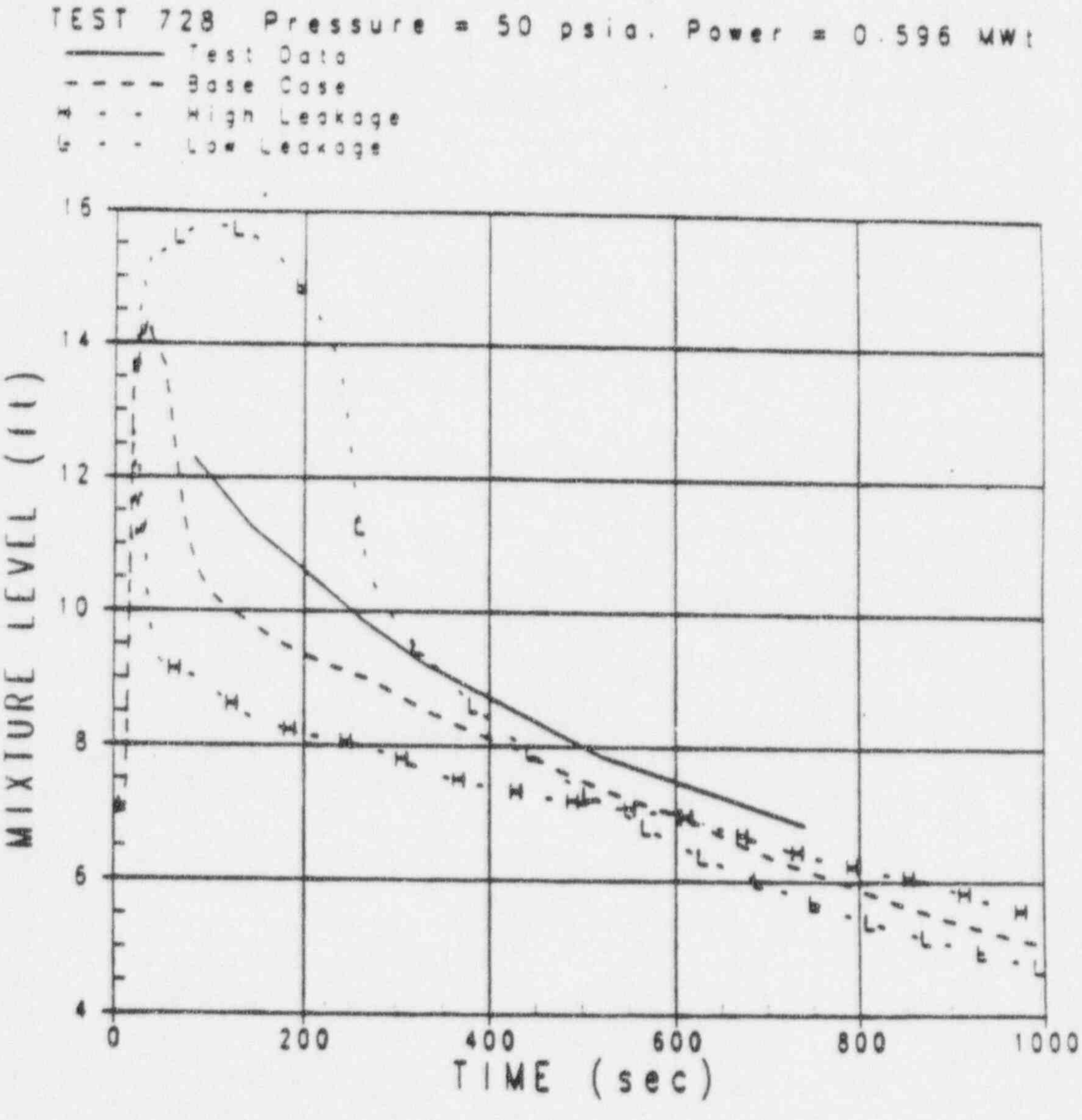


Figure 4.4-29 NOTRUMP Comparisons to G2 Test 728 Mixture Height with Uncertainties

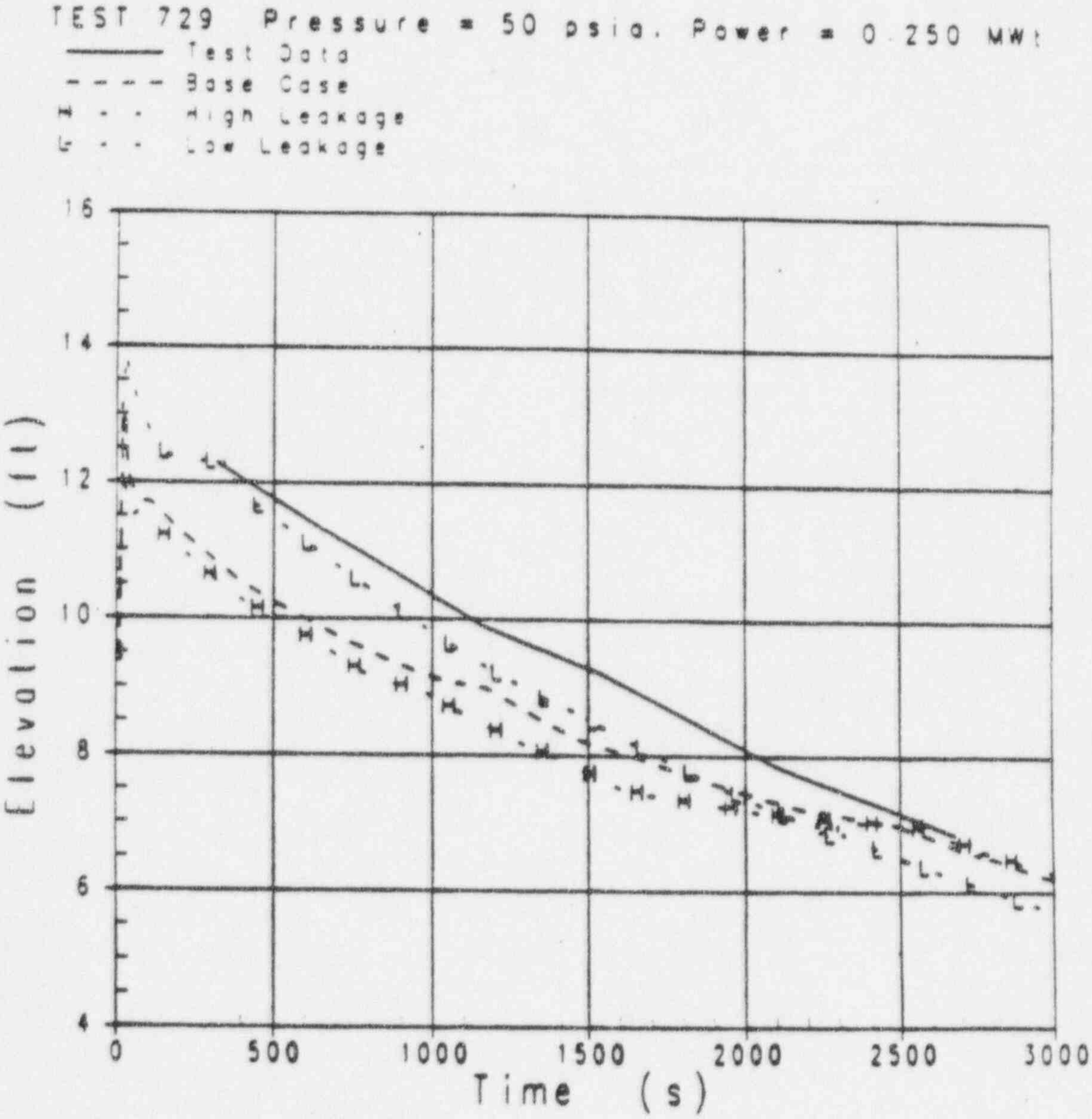


Figure 4.4-30 NOTRUMP Comparisons to G2 Test 729 Mixture Height with Uncertainties

7

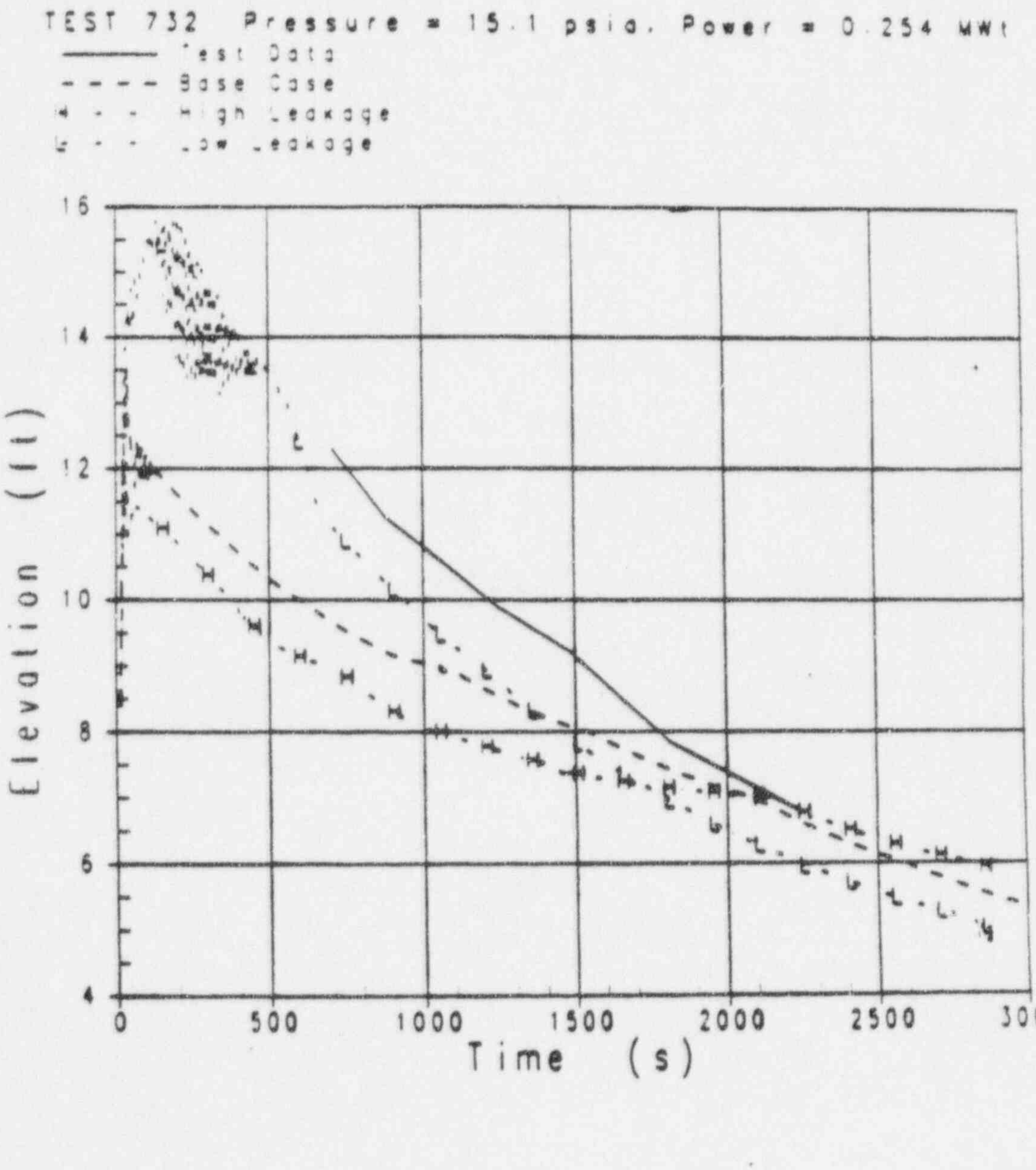


Figure 4.4-31 NOTRUMP Comparisons to G2 Test 732 Mixture Height with Uncertainties

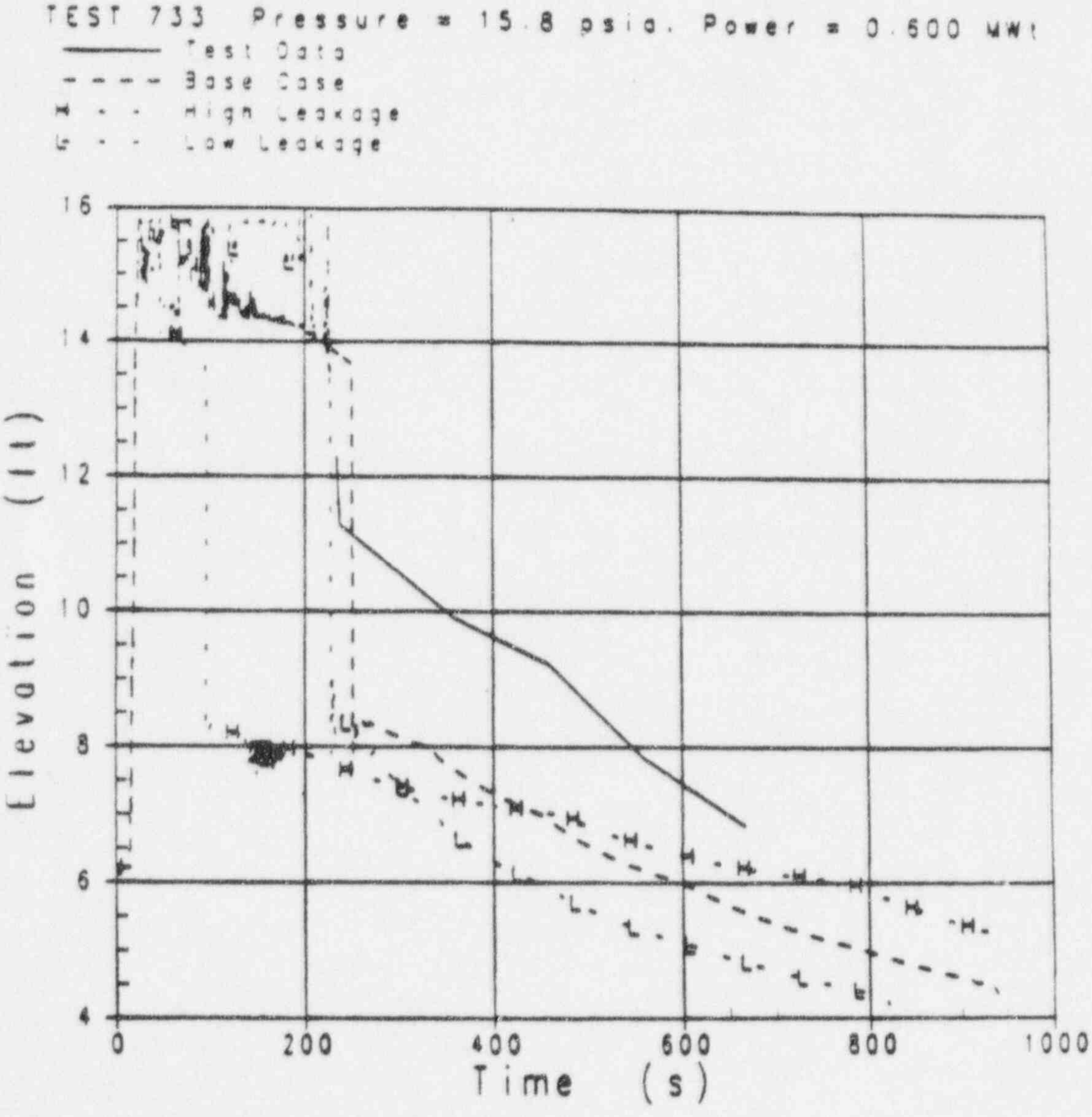


Figure 4.4-32 NOTRUMP Comparisons to G2 Test 733 Mixture Height with Uncertainties

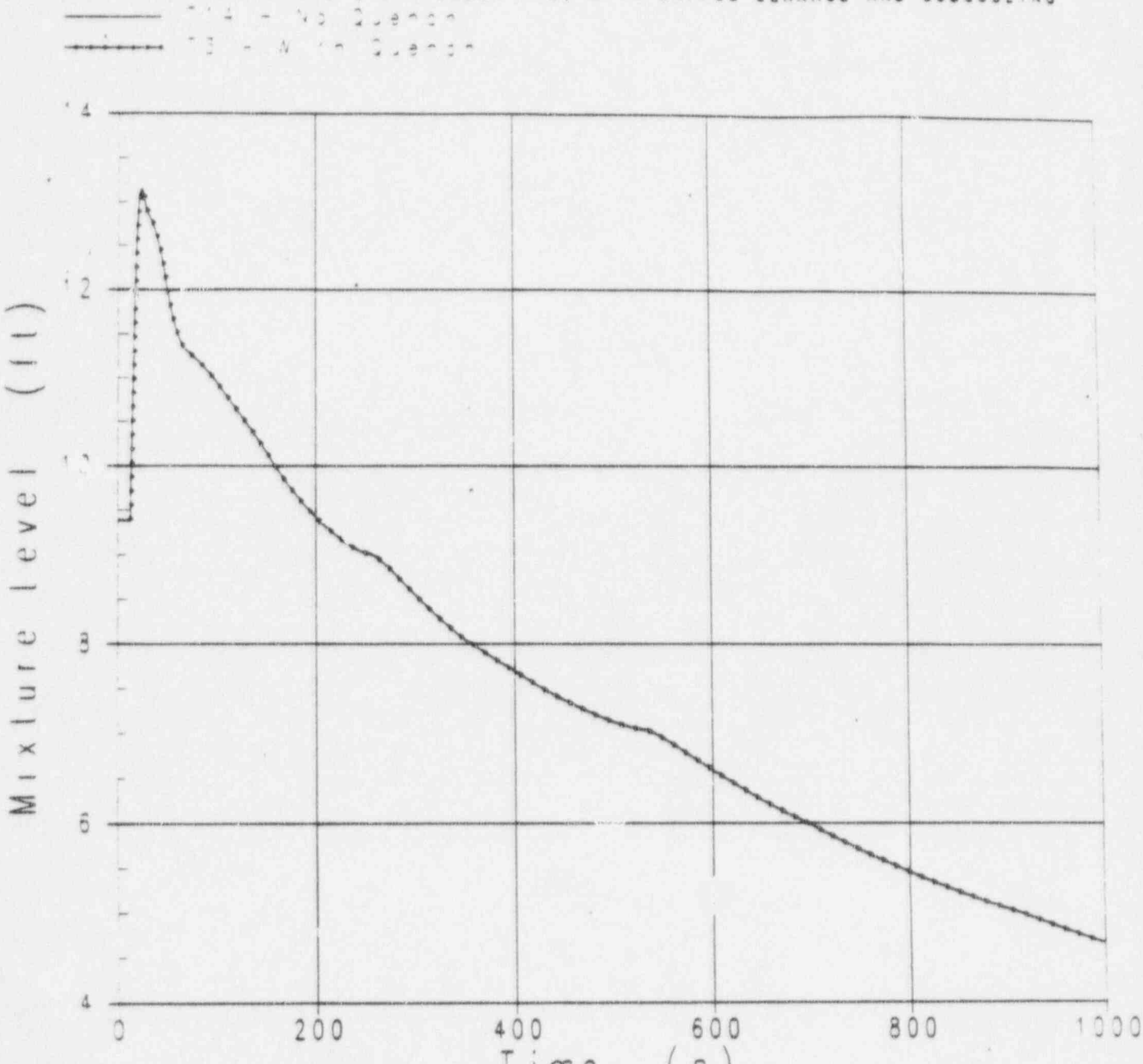
SUMMARY OF G2 CASES WITH FINAL CODE VERSION AND OPTIONS
(Quench Model Off)

NSD = No significant difference

Case	Base	High Leakage	Low Leakage
715	NSD	NSD	NSD
716	NSD	NSD	NSD
719	NSD	NSD	NSD
720	NSD	NSD	NSD
724	NSD	NSD	NSD
725	NSD	NSD	NSD
728	NSD	NSD	mixture level spike
729	mixture level spikes	NSD	NSD
732	mixture level spike	mixture level spikes	mixture level spike
733	run aborted ~ 200 sec	run aborted ~ 220 sec	some early differences, level spikes

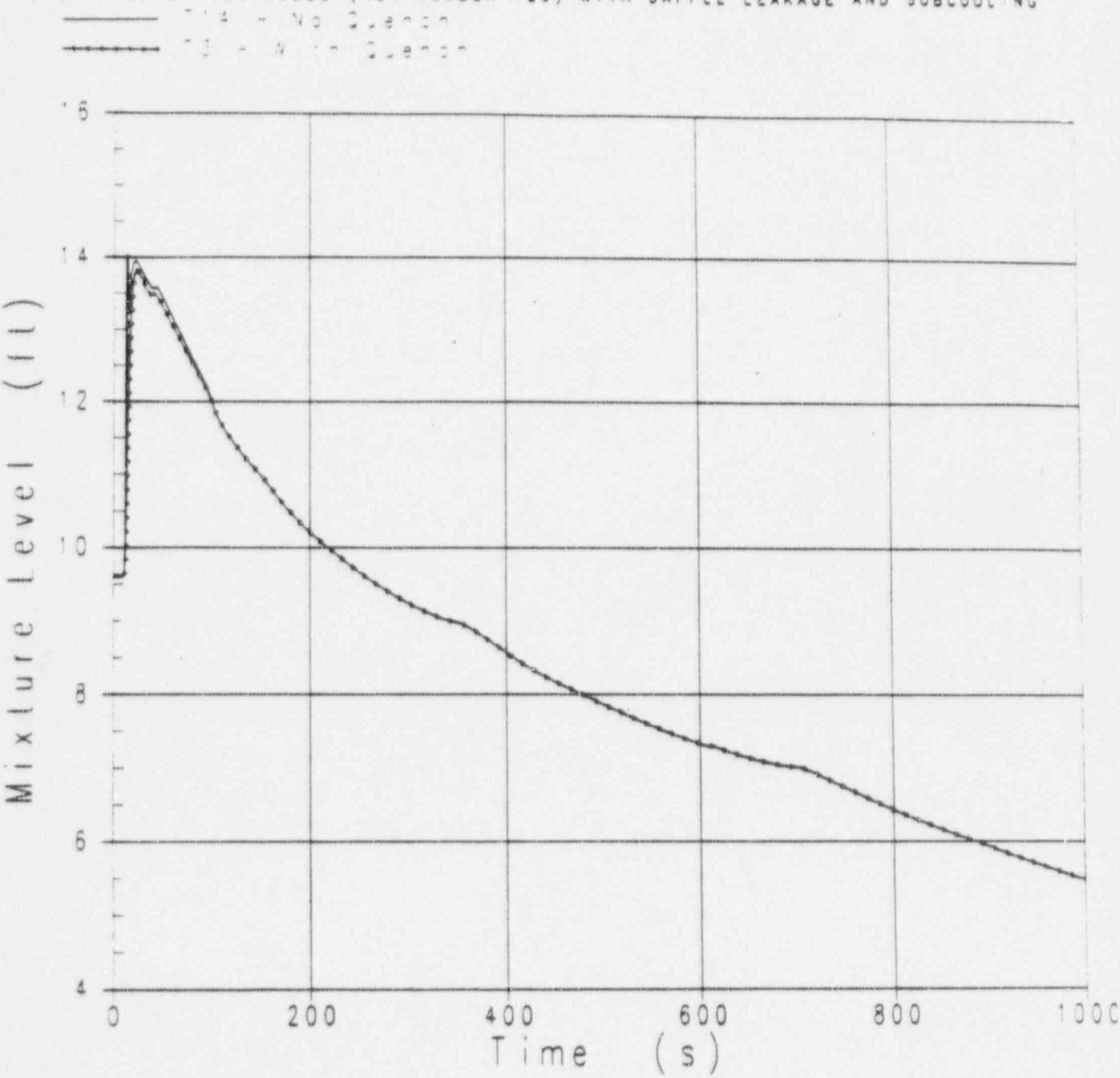
15

C-2 14-CORE NODE MODEL (RUN NUMBER 715) WITH BAFFLE LEAKAGE AND SUBCOOLING



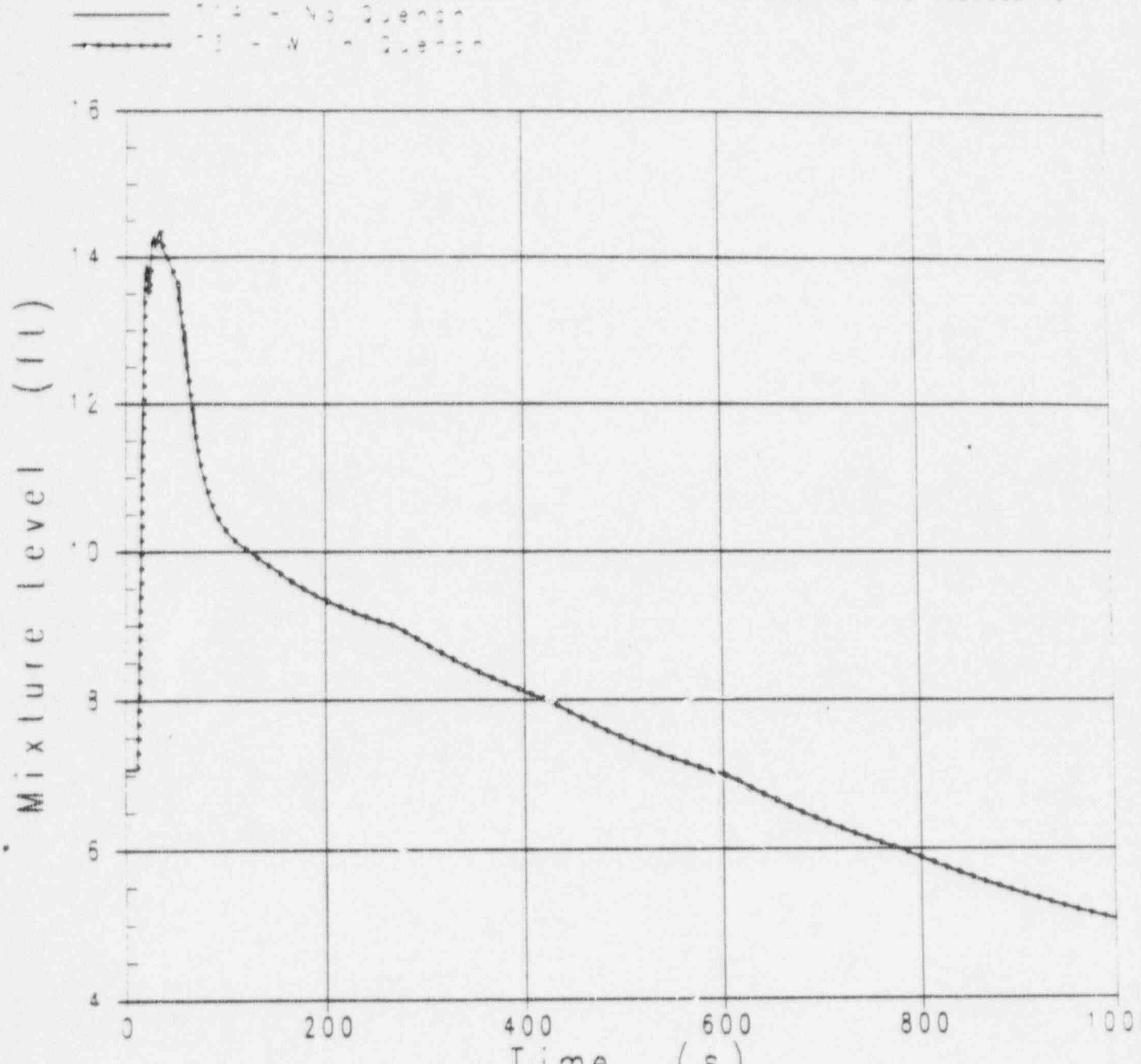
(11)

D-2 14-CORE NODE MODEL (RUN NUMBER 720) WITH BAFFLE LEAKAGE AND SUBCOOLING

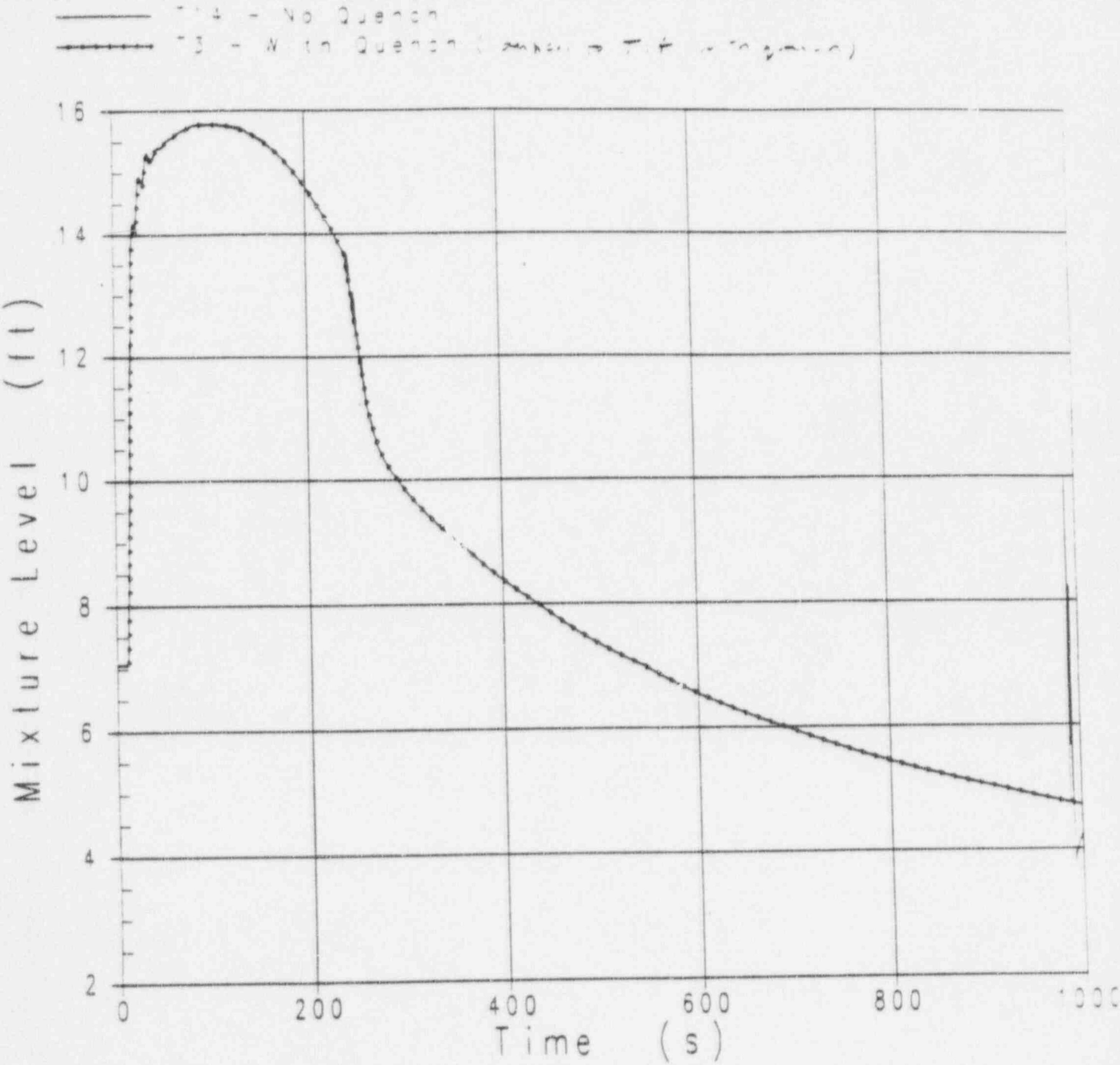


(12)

D-2 14-CORE NODE MODEL (RUN NUMBER 728) WITH BAFFLE LEAKAGE AND SUBCOOLING

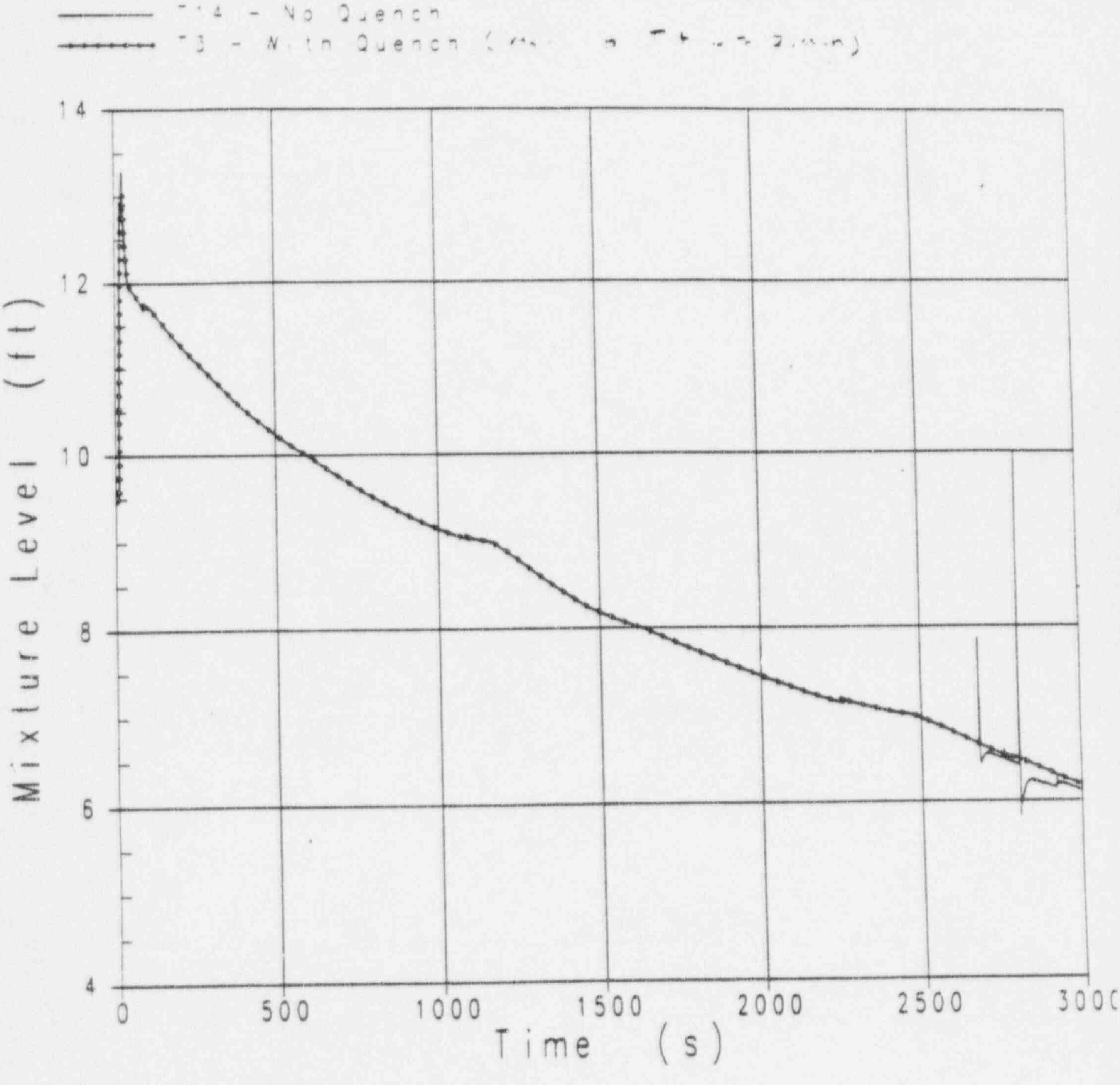


C-2 14-CORE NODE MODEL (RUN # 728) WITH LOW BAFFLE LEAKAGE AND SUBCOOLING



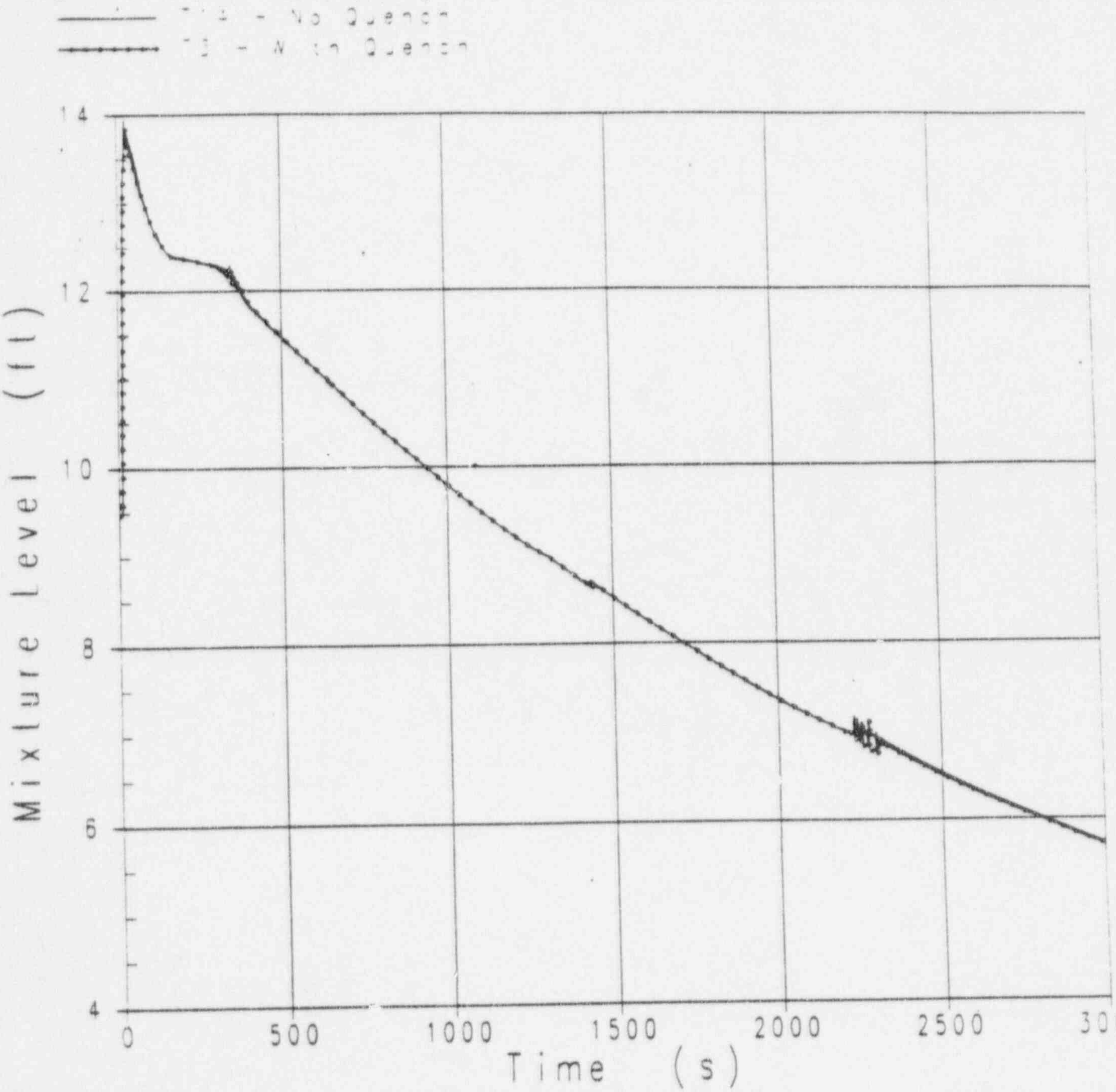
(14)

G-2 14-CORE NODE MODEL (RUN NUMBER 729) WITH BAFFLE LEAKAGE AND SUBCOOLING



15

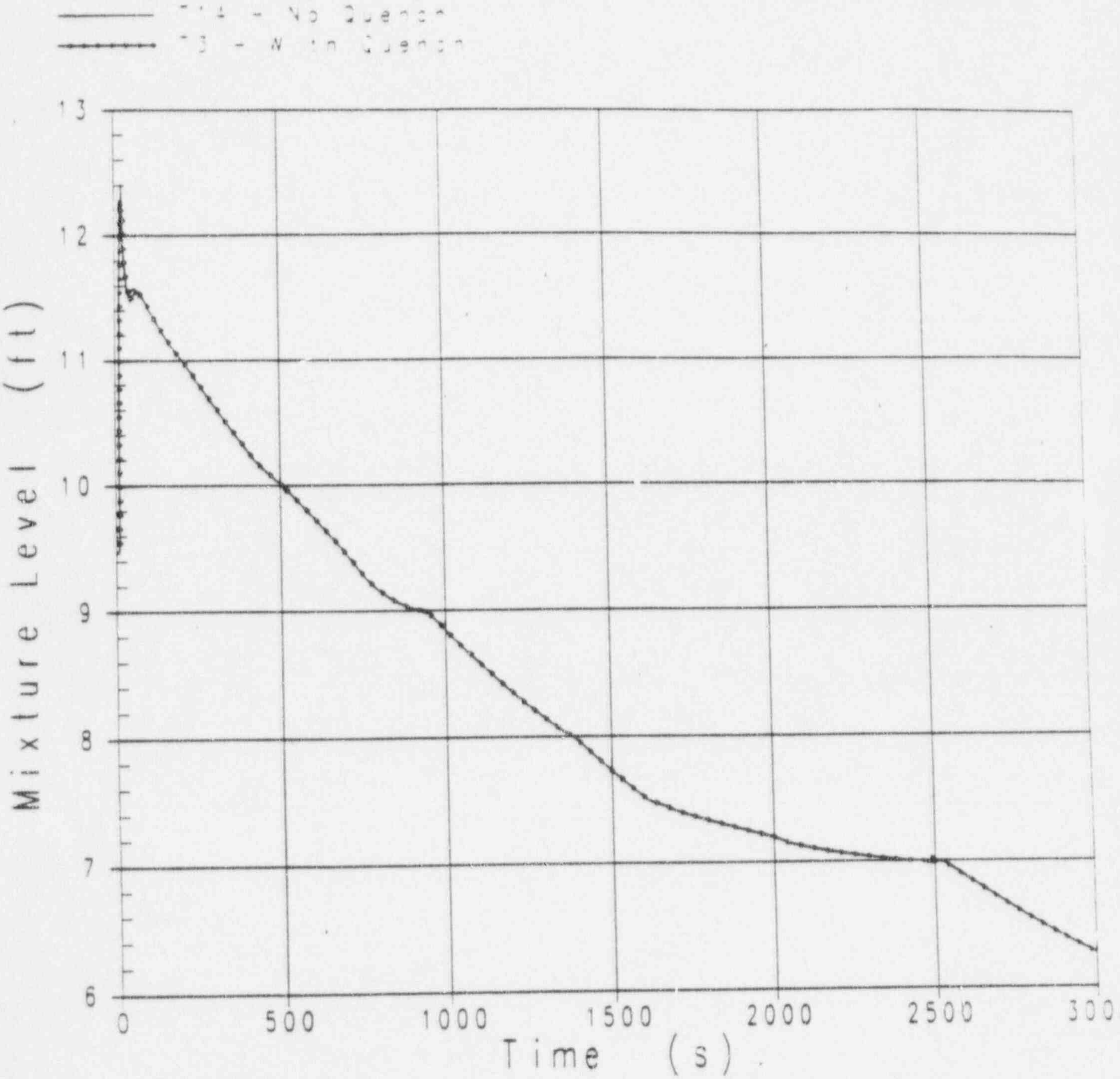
C-2 14-CORE NODE MODEL (RUN # 729) WITH LOW BAFFLE LEAKAGE AND SUBCOOLING



17

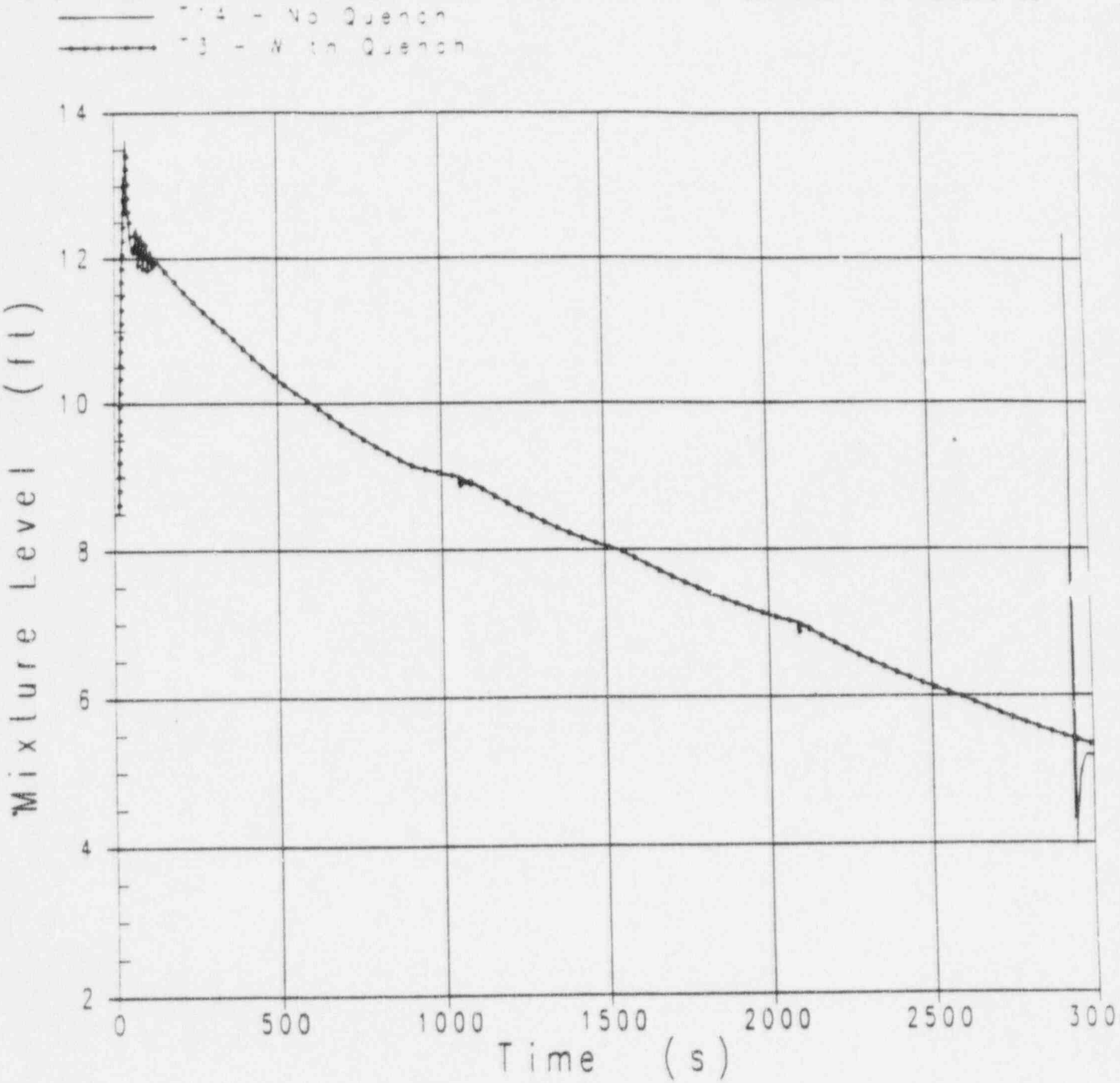
(16)

G-2 14-CORE NODE MODEL (RUN # 729) WITH HIGH BAFFLE LEAKAGE AND SUBCOOLING

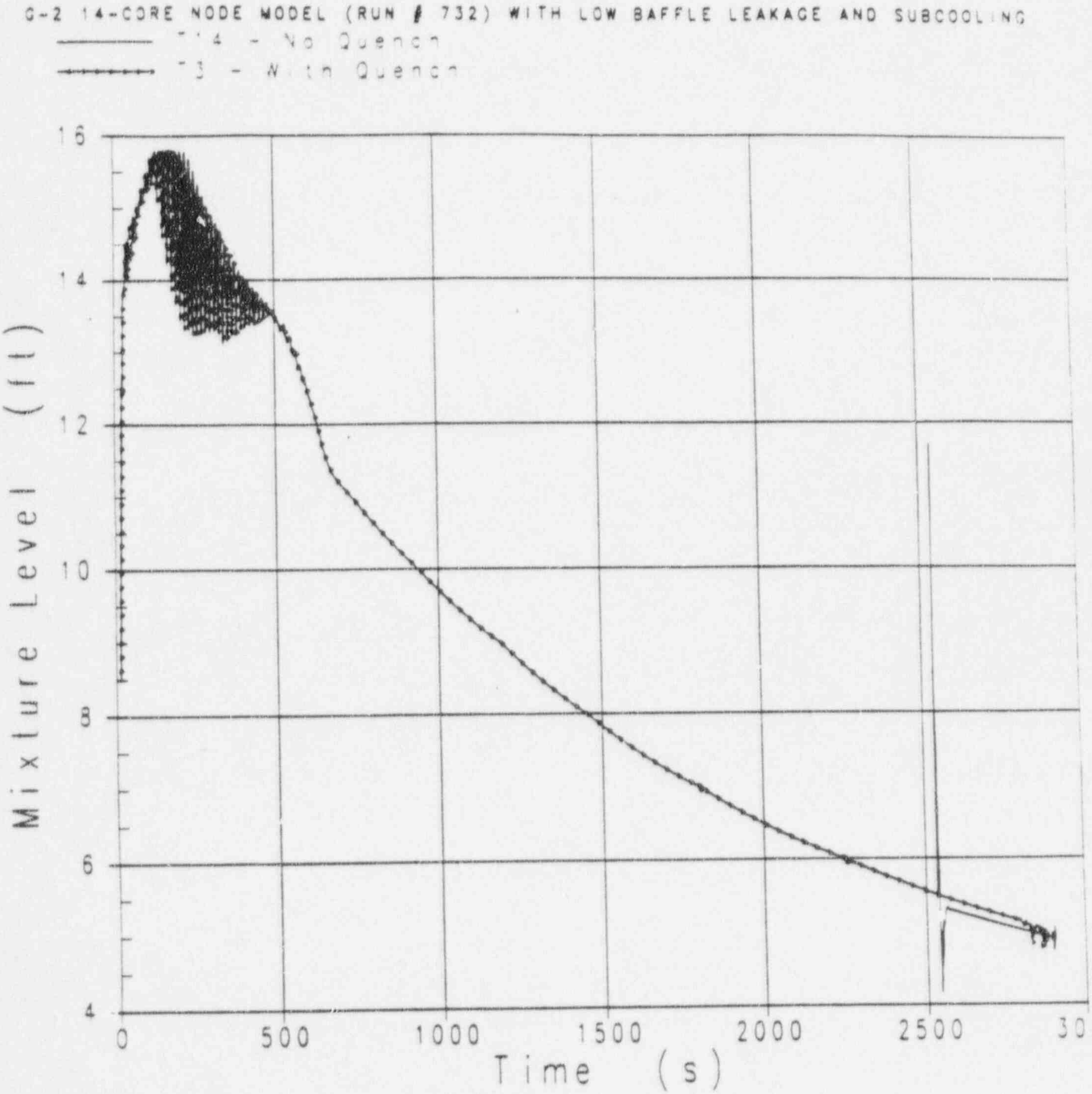


17

G-2 14-CORE NODE MODEL (RUN NUMBER 732) WITH BAFFLE LEAKAGE AND SUBCOOLING

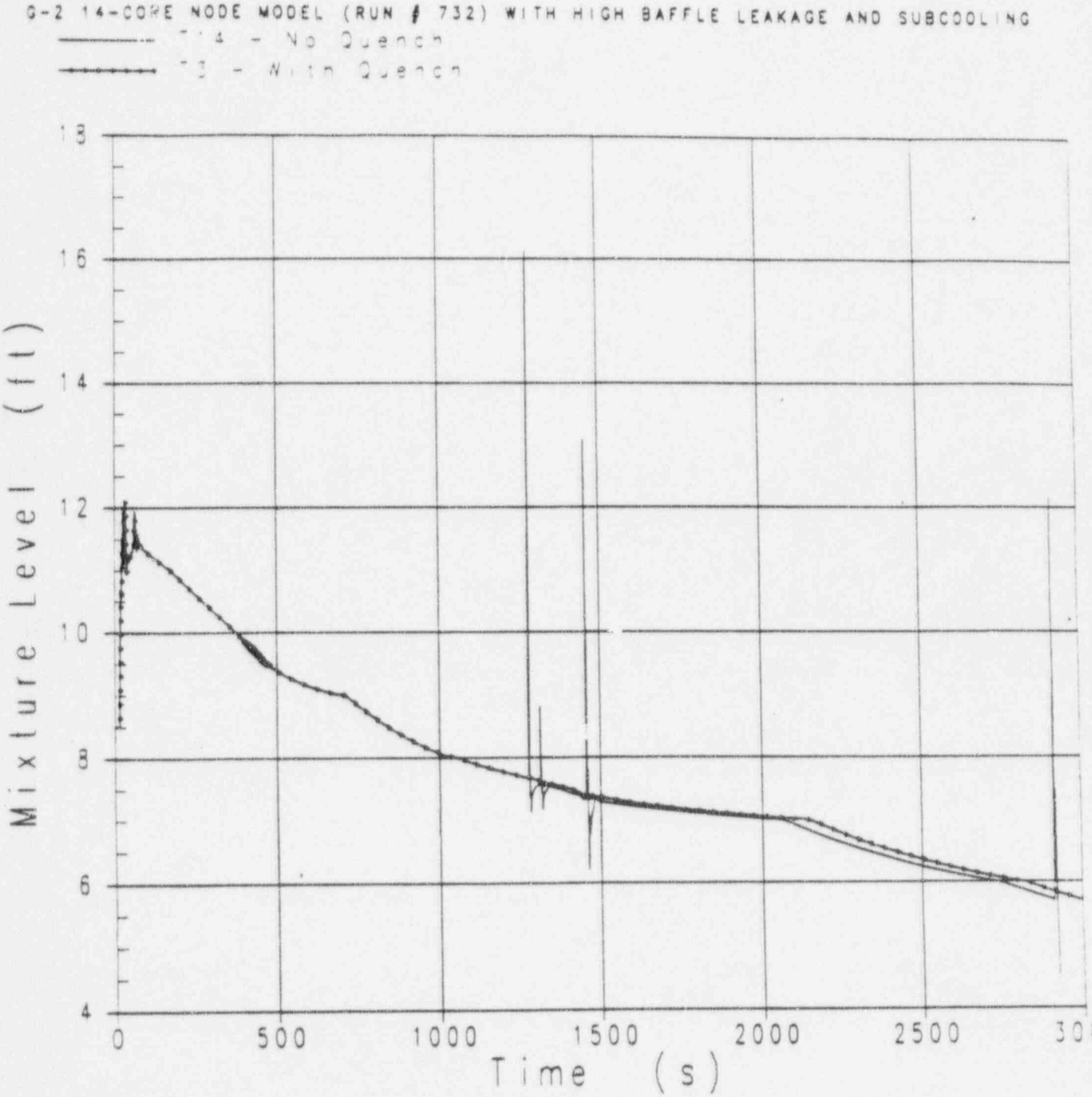


(18)


G-2 14-CORE NODE MODEL (RUN # 732) WITH LOW BAFFLE LEAKAGE AND SUBCOOLING
T14 - No Quench
T3 - With Quench
16
14
12
10
8
6
4
Mixture Level (ft)
0
500
1000
1500
2000
2500
300
Time (s)

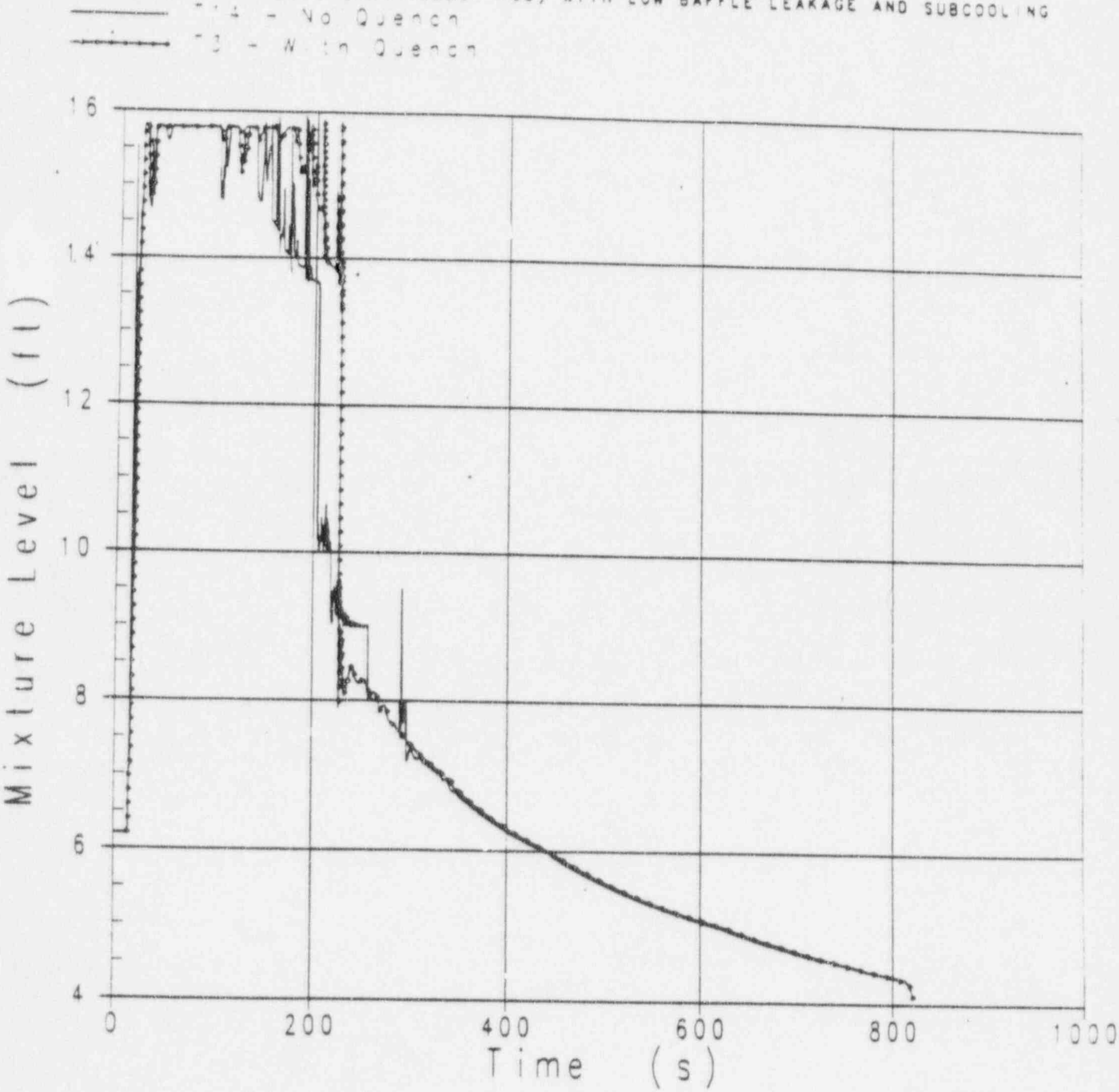
20

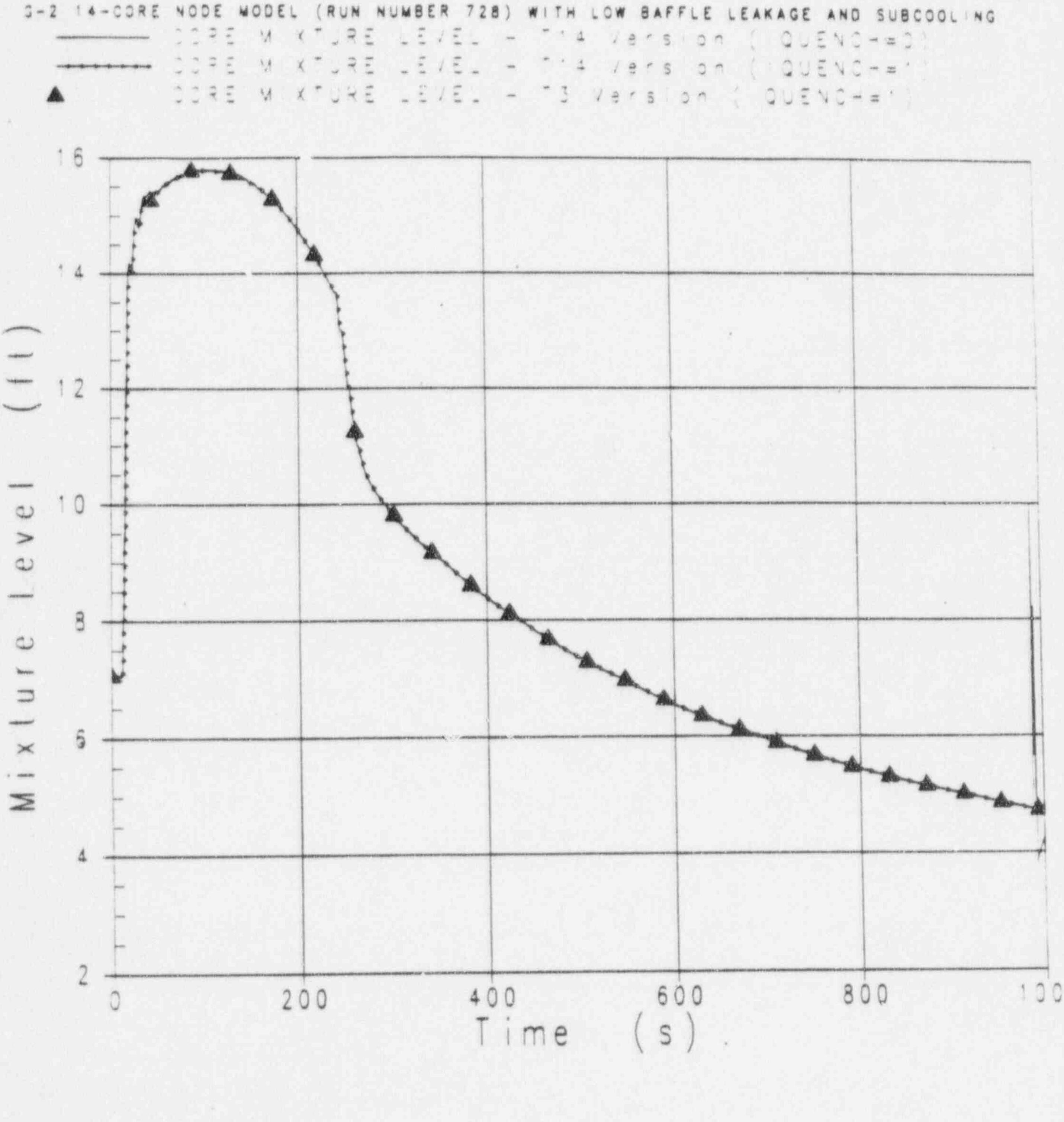
(19)

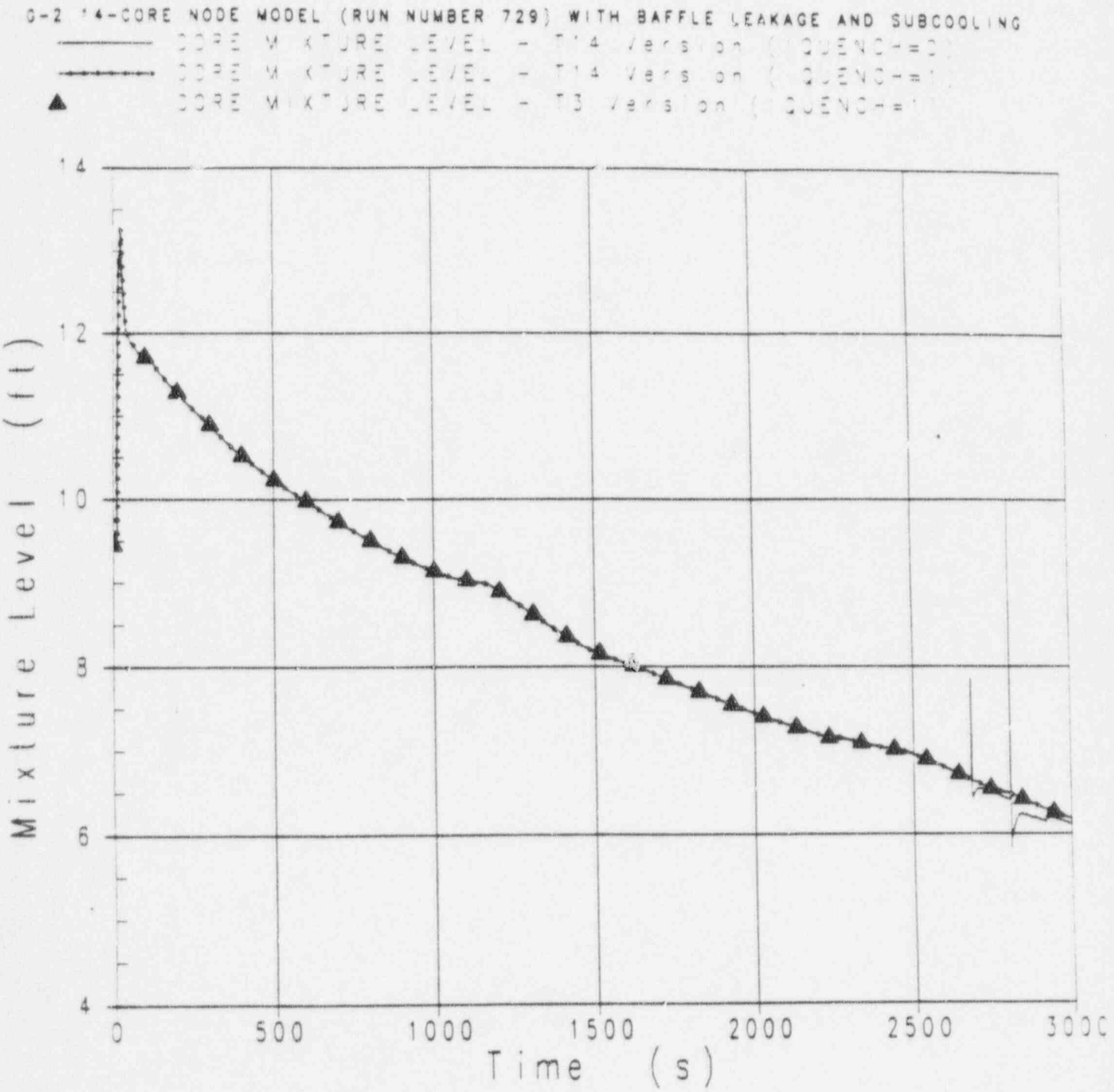


G-2 14-CORE NODE MODEL (RUN # 732) WITH HIGH BAFFLE LEAKAGE AND SUBCOOLING
T14 - No Quench
T3 - With Quench
Mixture Level (ft)
18
16
14
12
10
8
6
4
Time (s)
0
500
1000
1500
2000
2500
300

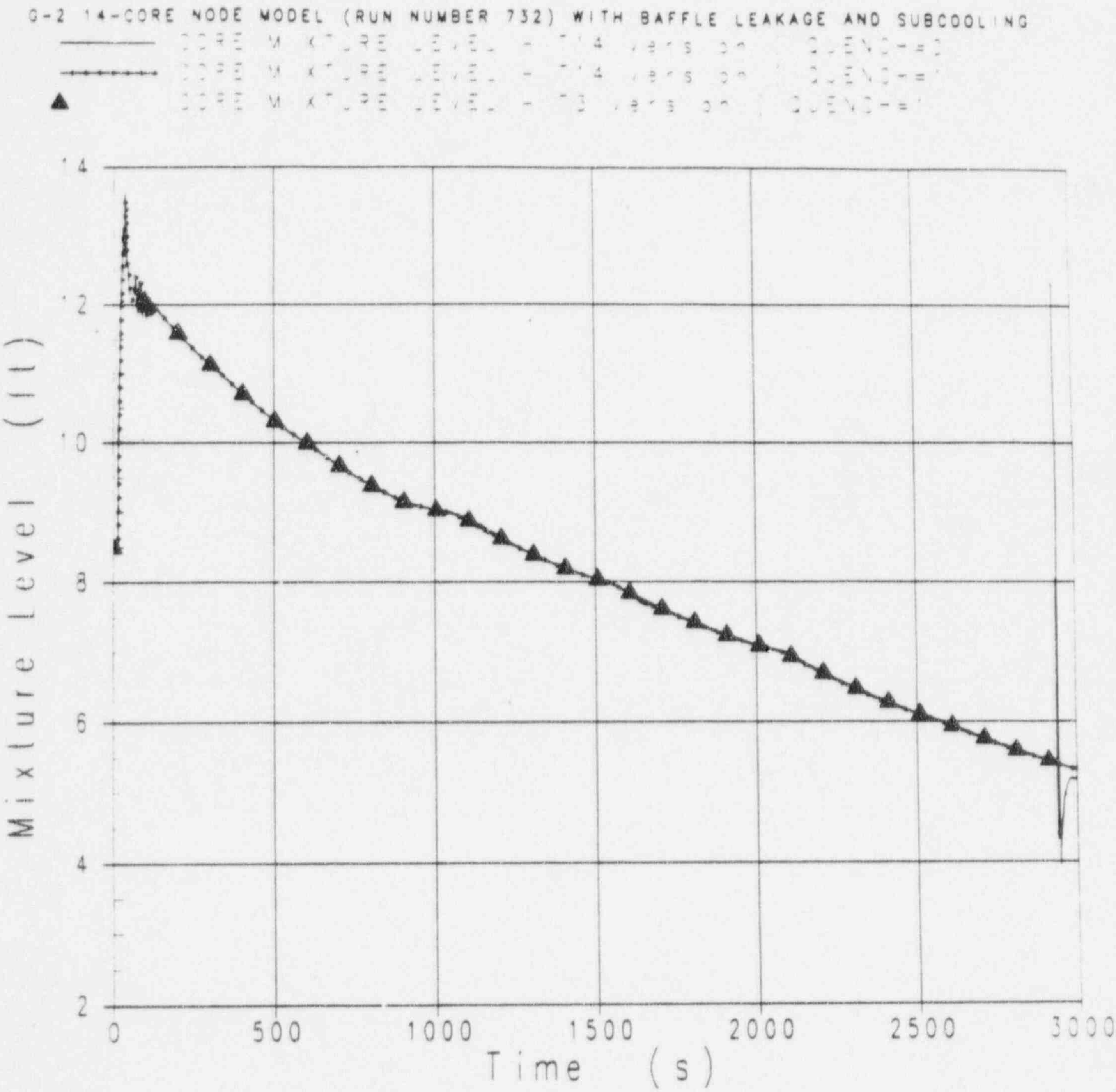
G-2 14-CORE NODE MODEL (RUN NUMBER 733) WITH LOW BAFFLE LEAKAGE AND SUBCOOLING



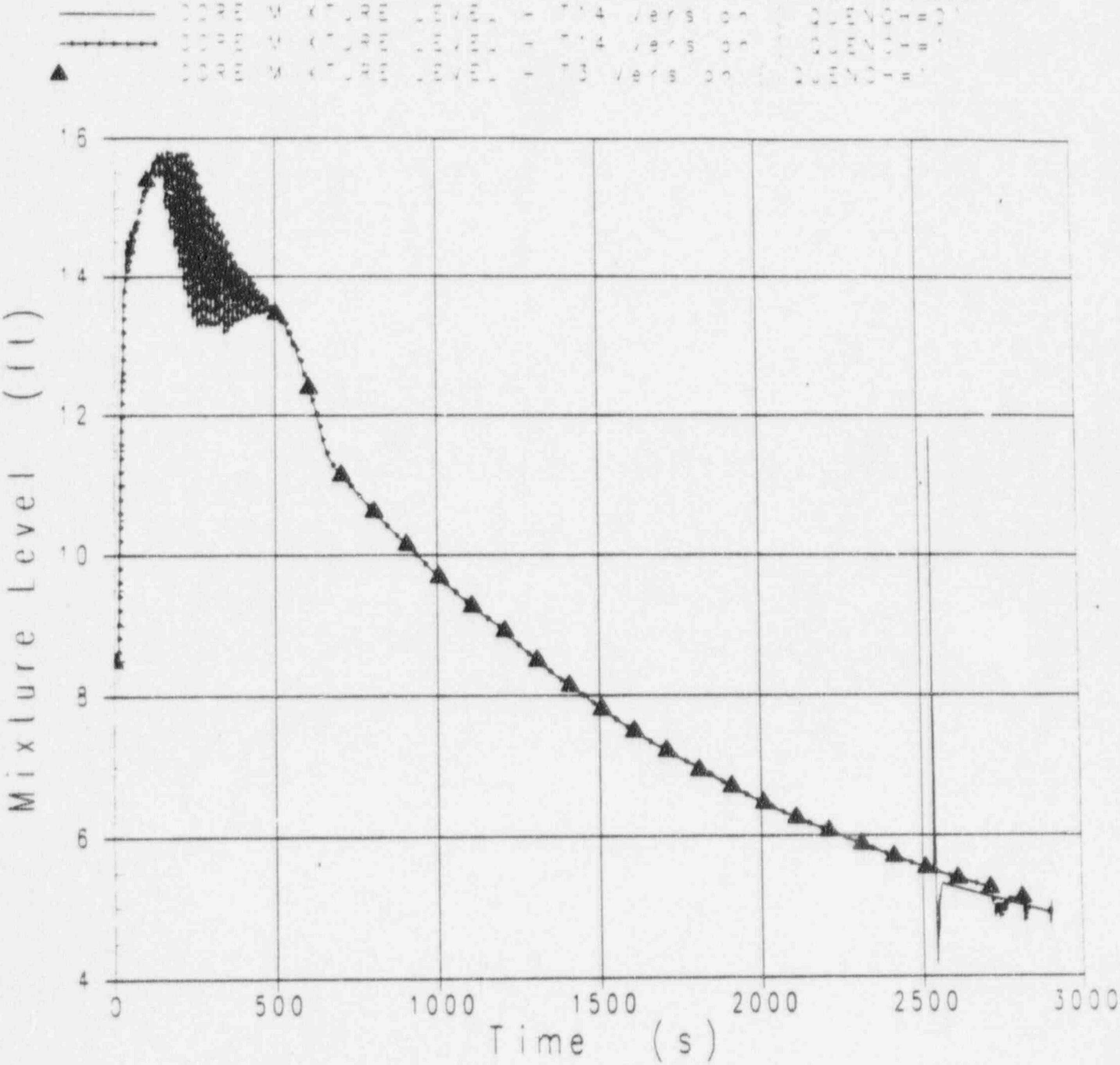

G-2 14-CORE NODE MODEL (RUN NUMBER 728) WITH LOW BAFFLE LEAKAGE AND SUBCOOLING
Mixture Level (ft)
16
14
12
10
8
6
4
2
0
200
400
600
800
1000
Time (s)


0-2 14-CORE NODE MODEL (RUN NUMBER 729) WITH BAFFLE LEAKAGE AND SUBCOOLING
CORE MIXTURE LEVEL - T14 Version (IQUENCH=0)
CORE MIXTURE LEVEL - T14 Version (IQUENCH=1)
CORE MIXTURE LEVEL - T3 Version (IQUENCH=1)
14
12
10
8
6
4
Mixture Level (ft)
0
500
1000
1500
2000
2500
3000
Time (s)

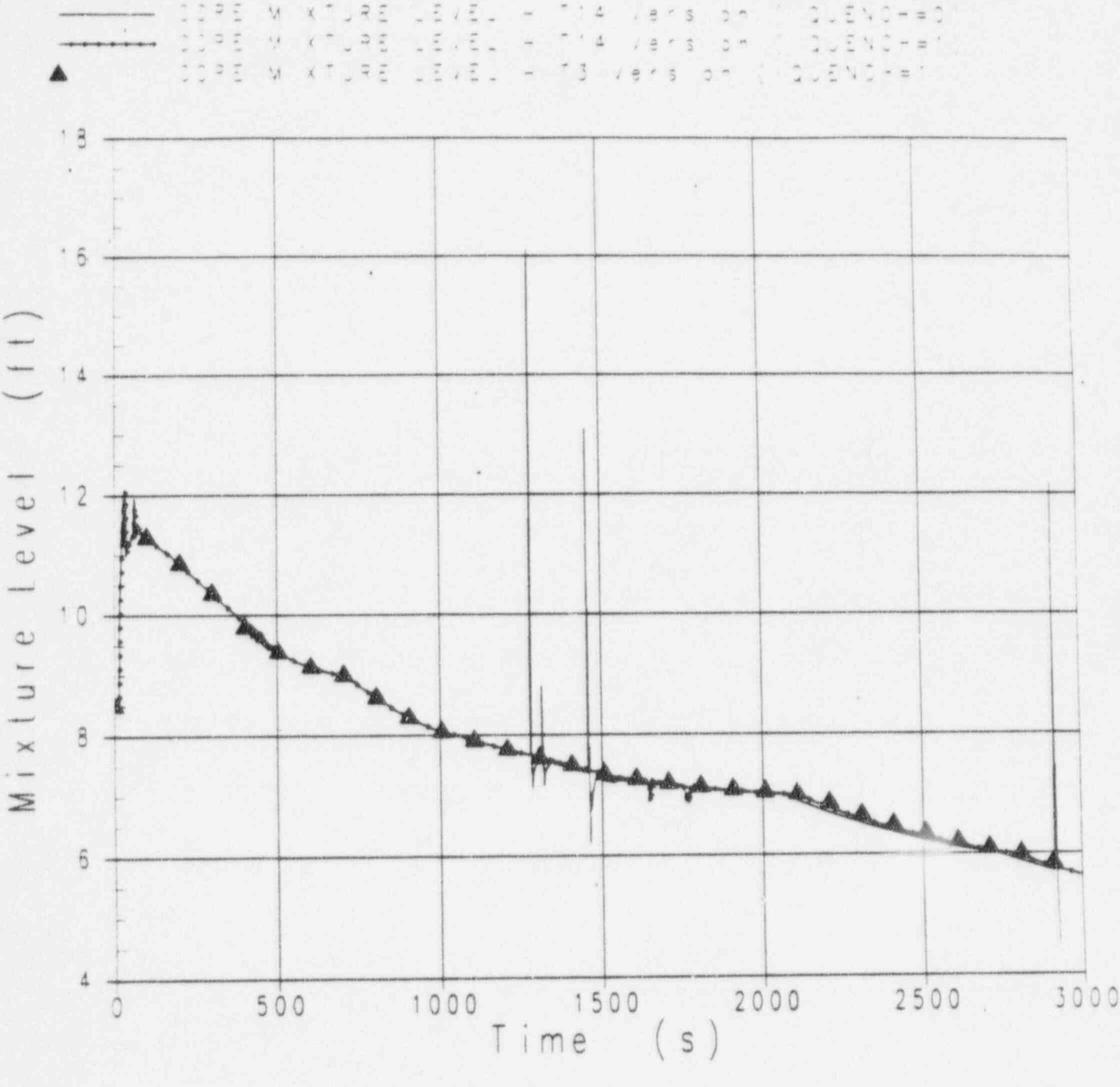
(23)


G-2 14-CORE NODE MODEL (RUN NUMBER 732) WITH BAFFLE LEAKAGE AND SUBCOOLING
Mixture Level (ft)
Time (s)
14
12
10
8
6
4
2
0
500
1000
1500
2000
2500
3000

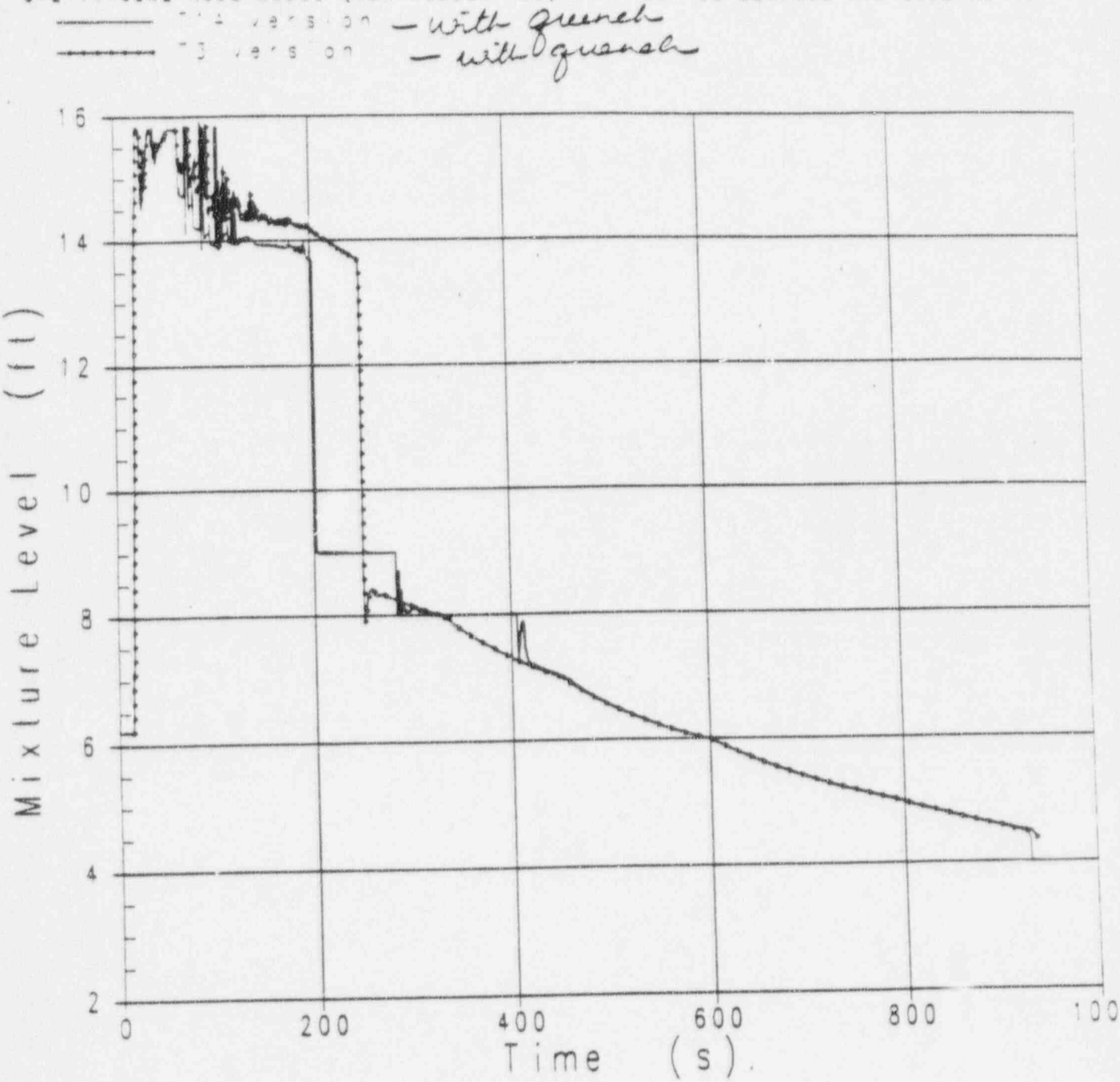
G-2 14-CORE NODE MODEL (RUN NUMBER 732) WITH LOW BAFFLE LEAKAGE AND SUBCOOLING



G-2 14-CORE NODE MODEL (RUN NUMBER 732) WITH HIGH BAFFLE LEAKAGE AND SUBCOOLING"

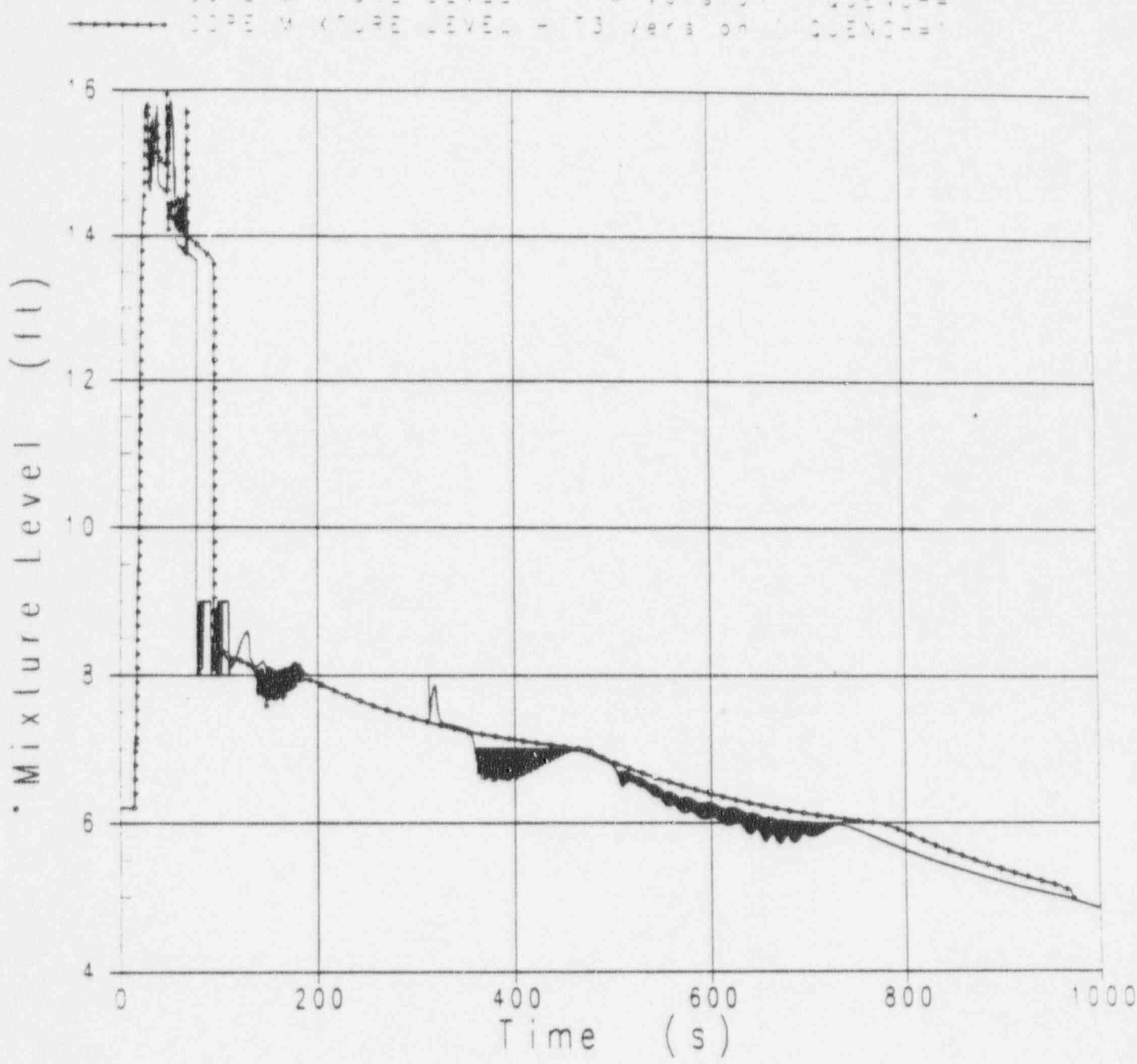


G-2 14-CORE NODE MODEL (RUN NUMBER 733) WITH BAFFLE LEAKAGE AND SUBCOOLING



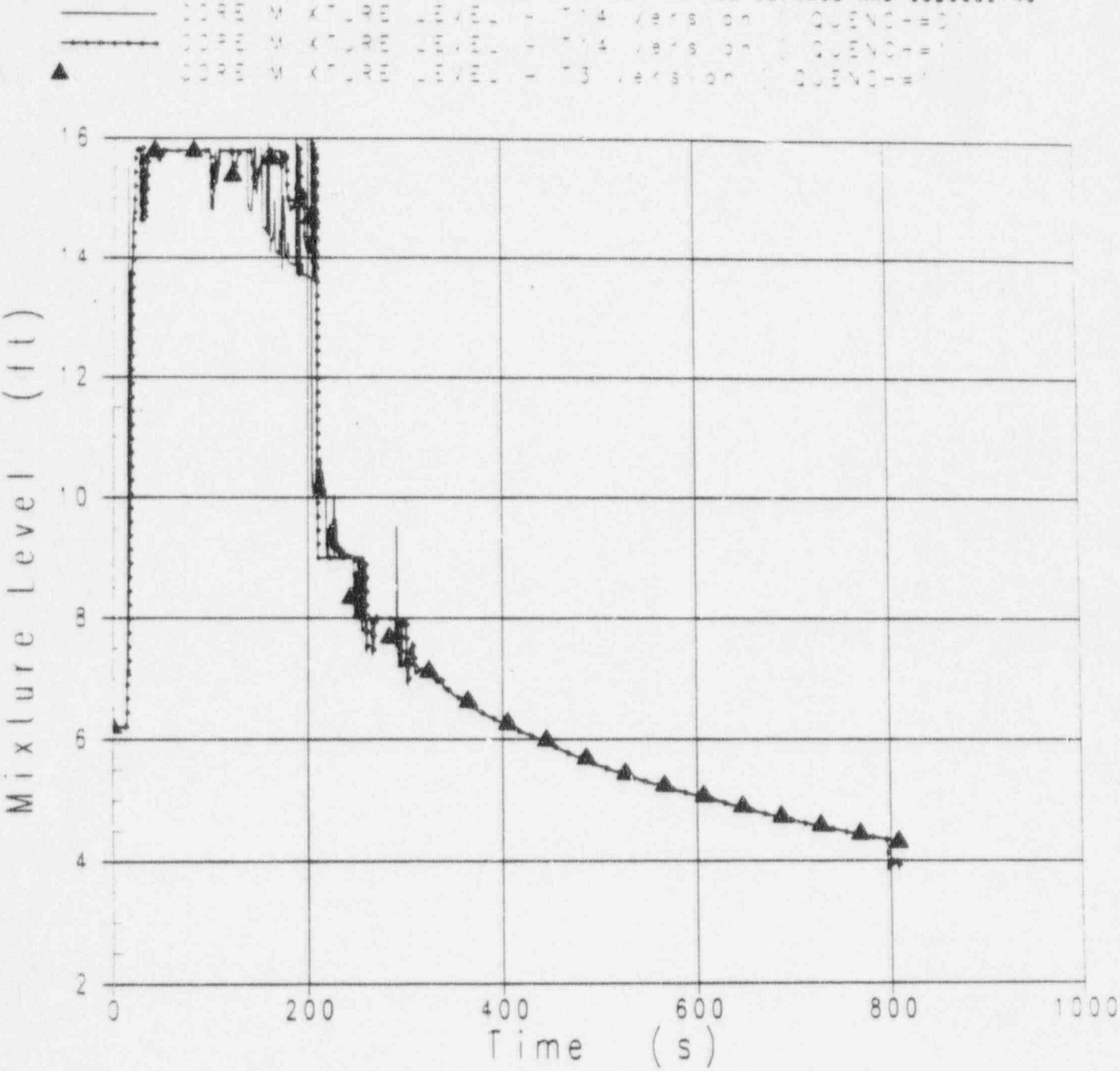
(27)

G-2 14-CORE NODE MODEL (RUN NUMBER 733) WITH HIGH BAFFLE LEAKAGE AND SUBCOOLING"



23

C-2 14-CORE NODE MODEL (RUN NUMBER 733) WITH LOW BAFFLE LEAKAGE AND SUBCOOLING



RESULTS OF USE OF QUENCH MODEL FOR G2 CALCULATIONS

- o Most cases don't need quench model
- o Quench model eliminates mixture level spiking
- o Mixture levels trends similar with or without spikes

36

CONCLUSIONS

- o Demonstration of code prediction of level swell based on GE, ACHILLES, and G2
- o Report already indicates that G2 calculations have much uncertainty
- o Not planning to use quench model for plant design basis calculations since no uncovery in analyzed cases
- o Plan to

    1) Indicate in G2 section of report that quench model exists and which cases use it

    2) Include new plots for cases with quench model included in G2 section of report

    3) Include reference to T/H Uncertainty report for description of quench model

3/18/97 doc file

1) Safe Shutdown End State, LCO 3.0.3:

LCO 3.0.3 specifies that when an LCO is not met and the associated Actions are not met, the plant shall be placed in MODE 5 within 48 hours. However, both LCOs 3.0.2 and 3.0.3 add a statement that "when plant conditions or configuration prevent the unit from being brought to the required MODE within the time limits specified using normal plant procedures, expedited actions shall be taken to establish and maintain the required plant conditions..." LCO 3.0.2 further states that "entry into LCO 3.0.3 is not required in this situation" (Underline added). This essentially amounts to a waiver for LCO 3.0.3 when the mode reduction cannot be accomplished within the required completion time as long as "expedited actions" are taken.

The situation arises because the MODE reduction completion times specified in LCO 3.0.3, when an LCO is not met, are based on the availability of non-safety, active systems, such as startup feedwater system (SFW) and normal residual heat removal system (RNS). If these active systems are not available, the AP600 passive systems alone cannot achieve MODE reduction within the specified times. Therefore, if the active heat removal systems needed for cool-down are not available, LCO 3.0.2 and 3.0.3 would not only permit longer MODE reduction times, but also exempt the MODE reduction requirement altogether as long as expedited actions are taken to accomplish unit shutdown as soon as practical. There are no further specifications regarding the "expedited actions" as well as the extended completion time.

While the SFW and RNS are necessary systems for MODE reduction within the specified completion times, they are not included in the TS because, Westinghouse contends, they do not meet any of the criteria for inclusion in the TS. Therefore, there is no TS requirement to control the reliability and availability of an important non-safety active system that is needed to complete the MODE reduction mission; and, when an LCO is not met, the compliance for LCO 3.0.3 MODE reduction is waived when this active system is unavailable. Without any control over the reliability/availability of the important RNS, there will be no control to minimize the exemption of MODE reduction requirement when an LCO is not met, and LCO 3.0.3 becomes meaningless because it is exempt when the RNS is unavailable.

Therefore, the proposed LCOs 3.0.2 and 3.0.3, as well as their BASES, are not acceptable. The TS should be revised by either deleting the statement of compliance exemption (due to nonsafety system unavailability) from LCOs 3.0.2 and 3.0.3, or adding the reliability/availability requirements of important nonsafety systems that are needed for completion of MODE reduction within the specified times. In SECY-94-084, Item A, Regulatory Treatment of Non-safety Systems, the Commission indicated its acceptance of "simple technical specifications" as an availability control mechanism for the important non-safety systems. If Westinghouse proposes to include in LCOs 3.0.2 and 3.0.3 the statement allowing compliance exemption, it should also propose a proper TS LCO to assure availability of those active systems which are relied upon to complete the mode reduction requirements.

To: Bill Huffman
From: Robin Nydes

p1/13

W Response (Revision 1 based on informal NRC feedback on previous Response):

The AP600 Technical Specification LCO 3.0.3 was originally written in 1992 to be consistent with the capabilities of the passive systems to place the plant to a safe shut down condition. The LCO 3.0.3 Completion Times were longer than for operating plants and the final shut down mode was MODE 4, consistent with the capabilities of the passive systems. In conjunction with this approach, MODE 4 was defined as 200 to 420°F rather than the standard 200 to 350°F, since passive systems can not cool the plant down to much less than 400°F in ~~a reasonable period of time~~ **less than 37 hours**. The use of nonsafety-related systems was not discussed in LCO 3.0.3. It was expected that the plant would use the systems normally used to perform plant shutdowns, including main feedwater, offsite power, etc.

In the May 1996 Senior Management Meeting, the NRC provided guidance that shutdown to only MODE 4 in LCO 3.0.3 was unacceptable and that if shutdown to MODE 5 were specified assuming the availability of nonsafety-related systems, that Technical Specification requirements on the nonsafety-related systems would not be necessary. Based on the NRC guidance, LCO 3.0.3 was revised (SSAR Rev. 9) , including shutdown to MODE 5~~-and shorter completion times~~. Additionally, LCO 3.0.2 and 3.0.3 were revised to clearly state that the nonsafety-related shutdown systems were not governed by Technical Specifications so that no violations would apply if non-technical specification shutdown systems were unavailable.

The importance of AP600 nonsafety-related systems has been systematically evaluated and an appropriate level of reliability and availability requirements has been identified. These evaluations include the RTNSS evaluation, Technical Specification selection evaluation and the Reliability Assurance Program (RAP).

The RTNSS evaluation (WCAP-13856) did not capture SFW and only captured RNS because it is an important factor in the initiating event frequency of loss of normal cooling during cold shutdown operation with reduced inventory. The RNS was not RTNSS important during other modes of plant operation. WCAP-13856 contains recommendations to the COL for the development of plant operating procedures that will provide short term availability control of the RNS and its necessary support systems during its RTNSS important mission (cold shutdown with reduced inventory).

The Technical Specification selection criteria in 10CFR50.36 was applied to AP600. The selection criteria did not capture the SFW, the RNS or any other nonsafety-related systems. It should be noted that operating plants which have implemented NUREG-1431 have eliminated LCOs from the Standard Technical Specifications using the 10CFR50.36 selection criteria; many of these eliminated LCO's are similar in safety importance to the LCO's suggested by the NRC for the RNS and SFW. LCO's that were eliminated from the Technical Specifications were relocated in other procedures or programs such as the Technical Requirements Manual (TRM). These programs provide an appropriate level of regulatory oversight / availability control for these less important features.

The AP600 RAP (SSAR table 16.2-1) provides for long term reliability of AP600 features. This table includes both safety-related and nonsafety-related defense-in-depth features, such as the SFW and the RNS.

The AP600 LCO 3.0.3 approach is based on specifying the safest course of action, considering potential plant conditions. If nonsafety-related systems used to cool the plant to MODE 5 conditions are unavailable when the plant enters LCO 3.0.3, the operators will **have sufficient time to restore those systems and** ~~still have to~~ place the plant in a safe shutdown condition.

~~The AP600 LCO 3.0.2 and 3.0.3 requirements are similar to the precedent provided by NUREG 1431 requirements which are applicable when the means to cool the plant are not available. For example, the Standard Technical Specifications (STS) LCO 3.7.5 requirements recognize that if the AFW pumps are inoperable that the safest action is to maintain the plant in the current MODE and repair one AFW train. Additionally, the Action D.1 Note suspends compliance with LCO 3.0.3 or other Required Actions which require MODE changes:~~

~~LCO 3.7.5 Condition D.~~

~~"D. [Three] AFW trains inoperable in MODE 1, 2, or 3.~~

~~D.1 NOTE~~

~~LCO 3.0.3 and all other LCO Required Actions requiring MODE changes are suspended until one AFW train is restored to OPERABLE status.~~

~~Initiate action to restore one AFW train to OPERABLE status.~~

~~Immediately"~~

~~STS BASES 3.7.5 explanation:~~

~~"D.1~~

~~If all [three] AFW trains are inoperable in MODE 1, 2, or 3, the unit is in a seriously degraded condition with no safety related means for conducting a cooldown, and only limited means for conducting a cooldown with nonsafety related equipment. In such a condition, the unit should not be perturbed by any action, including a power change, that might result in a trip. The seriousness of this condition requires that action be started immediately to restore one AFW train to OPERABLE status.~~

~~Required Action D.1 is modified by a Note indicating that all required MODE changes or power reductions are suspended until one AFW train is restored to OPERABLE status. In this case, LCO 3.0.3 is not applicable because it could force the unit into a less safe condition".~~

~~The AP600 situation follows the precedent set in STS LCO 3.7.5. When the principle means of plant cooldown are unavailable, cooldown should not be attempted, rather the cooldown system(s) should be restored. The AP600 additions to LCOs 3.0.2 and 3.0.3 should be retained to avoid forcing the plant into a less safe condition.~~

Conclusions

In summary, Westinghouse has attached a suggested modification to modified LCO 3.0.3 for NRC review. This remains consistent with ~~at~~ the agreement at ~~request of~~ the NRC ~~in a~~ Senior Management Meeting ~~based on a agreement~~ that this action would not lead to Technical Specifications on the nonsafety-related systems which are needed to achieve MODE 5, but which are not needed to achieve the AP600 safe stable long term shutdown condition in MODE 4.

The NRC question recommendations two possible resolutions to their concerns:

1) to add technical specification requirements for SFW and RNS or

2) to eliminate the LCO 3.0.2 and 3.0.3 compliance exception

Westinghouse concludes that technical specification requirements for SFW and RNS are not necessary in accordance with the 10CFR50.36 criteria as well as past agreements with the NRC. NRC concurs.

Westinghouse agrees to eliminate the LCO 3.0.2 and 3.0.3 compliance exception as shown in the attached markup. ~~concludes that the LCO 3.0.2 and 3.0.3 compliance exceptions are necessary to provide for coherent requirements consistent with the precedent provided in the STS.~~

Therefore, LCOs 3.0.2 and 3.0.3 ~~should maintain the approach currently specified~~ are revised as attached. Longer completion times and deletion of the compliance exceptions in LCO 3.0.2 are based on the understanding that fines are assessed on utilities when the LCO 3.0.3 conditions are not met, not for entering LCO 3.0.3.

3.0 LIMITING CONDITIONS FOR OPERATION (LCO) APPLICABILITY

LCO 3.0.1 LCOs shall be met during the MODES or other specified conditions in the Applicability, except as provided in LCO 3.0.2.

LCO 3.0.2 Upon discovery of a failure to meet an LCO, the Required Actions of the associated Conditions shall be met, except as provided in LCO 3.0.5 or 3.0.6.

In some cases, the Required Actions specify that the unit be shutdown and placed in a lower MODE. When plant conditions or configuration prevent the unit from being brought to the required MODE within the time limits specified using normal plant procedures, expedited actions shall be taken to establish and maintain the required plant conditions. This should be accomplished in a controlled and orderly manner in accordance with normal plant procedures, well within the specified maximum cooldown rate and within the capabilities of the unit using the available plant systems and components. ~~Entry into LCO 3.0.3 is not required in this situation.~~

If the LCO is met, or is no longer applicable prior to expiration of the specified Completion Time(s), completion of the Required Action(s) is not required, unless otherwise stated.

LCO 3.0.3 When an LCO is not met and the asscciated ACTIONS are not met, an associated ACTION is not provided, or if directed by the associated ACTIONS, the unit shall be placed in a MODE or other specified condition in which the LCO is not applicable. Action shall be initiated within 1 hour to place the unit, as applicable, in:

a. MODE 3 within 8 hours; and

b. MODE 4 within ~~24~~ 48 hours; and

c. MODE 5 within ~~48 hours.~~ 7 days (168 hours)

When plant conditions or configuration prevent the unit from being brought to the required MODE within the time limits specified above using normal plant procedures, expedited actions shall be taken to establish and maintain the required plant conditions. This should be accomplished in a controlled

(continued)

3.0 LCO APPLICABILITY

LCO 3.0.3 (continued)

and orderly manner in accordance with normal plant procedures, ~~well~~ within the specified maximum cooldown rate and within the capabilities of the unit using the available plant systems and components.

Exceptions to this specification are stated in the individual specifications.

Where corrective measures are completed that permit operation in accordance with the LCO or ACTIONS, completion of the actions required by LCO 3.0.3 is not required. LCO 3.0.3 is only applicable in MODES 1, 2, 3, and 4.

LCO 3.0.4

When an LCO is not met, entry into a MODE or other specified condition in the Applicability shall not be made except when the associated ACTIONS to be entered permit continued operation in the MODE or other specified condition in the Applicability for an unlimited period of time. This Specification shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

Exceptions to this specification are stated in the individual Specifications. These exceptions allow entry into MODES or other specified conditions in the Applicability when the associated ACTIONS to be entered allow unit operation in the MODE or other specified condition only for a limited period of time.

LCO 3.0.4 is only applicable for entry into a MODE or other specified condition in the Applicability in MODES 1, 2, 3, and 4.

LCO 3.0.5

Equipment removed from service or declared inoperable to comply with ACTIONS may be returned to service under administrative control solely to perform testing required to demonstrate its OPERABILITY or the OPERABILITY of other equipment. This is an exception to LCO 3.0.2 for the system returned to service under administrative control to perform the test required to demonstrate OPERABILITY.

(continued)

B 3.0 LIMITING CONDITIONS FOR OPERATION (LCO) APPLICABILITY

BASES

LCOs

LCO 3.0.1 through LCO 3.0.6 establish the general requirements applicable to all Specifications and apply at all times, unless otherwise stated.

LCO 3.0.1

LCO 3.0.1 establishes the Applicability statement within each individual Specification as the requirements for when the LCO is required to be met (i.e. when the unit is in the MODES or other specified conditions of the Applicability statement of each Specification.)

LCO 3.0.2

LCO 3.0.2 establishes that upon discovery of a failure to meet an LCO, the associated ACTIONS shall be met. The Completion Time of each Required Action for an ACTIONS Condition is applicable from the point in time that the ACTIONS Condition is entered. The Required Actions establish those remedial measures that must be taken within specified Completion Times when the requirements of an LCO are not met. This specification establishes that:

a. Completion of the Required Actions within the specified Completion Times constitutes compliance with a Specification; and

b. Completion of the Required Actions is not required when an LCO is met within the specified Completion Time, unless otherwise specified.

c. Required Actions which specify unit shutdown require the use of nonsafety related systems which may be unavailable. ~~Compliance with the specified Completion Time is not required provided~~ Restoration of the nonsafety system(s) should be ~~is~~ actively pursued such that the required unit shutdown is accomplished ~~as soon as practical~~ within the specified Completion Time. It ~~However,~~ unit conditions shall ~~also~~ be considered by the operators such that overall safety is optimized. These considerations may include delay of repairs or use of temporary equipment. In these cases, Specification compliance is maintained by completion of the specified shutdown. ~~as soon as practical. In particular, transition from MODE 3 to~~

(continued)

BASES

LCO 3.0.2 (continued)

~~MODE 4 using only the passive, safety related system(s) may be accomplished in a time greater than 24 hours.~~

Although nonsafety related systems are necessary for RCS cooldown to less than 200°F within the Completion Times specified, OPERABILITY of the nonsafety related systems is not required or implied by these Technical Specifications.

There are two basic types of Required Actions. The first type of Required Action specifies a time limit in which the LCO must be met. This time limit is the Completion Time to restore an inoperable system or component to OPERABLE status or to restore variables to within specified limits. If this type of Required Action is not completed within the specified Completion Time, a shutdown may be required to place the unit in a MODE or condition in which the Specification is not applicable. (Whether stated as a Required Action or not, correction of the entered Condition is an action that may always be considered upon entering ACTIONS.) The second type of Required Action specifies the remedial measures that permit continued operation of the unit that is not further restricted by the Completion Time. In this case compliance with to the Required Actions provides an acceptable level of safety for continued operation.

Completing the Required Actions is not required when an LCO is met, or is no longer applicable, unless otherwise stated in the individual Specifications.

The nature of some Required Actions of some Conditions necessitates that, once the Condition is entered, the Required Actions must be completed even though the associated Conditions no longer exist. The individual LCO's ACTIONS specify the Required Actions where this is the case. An example of this is in LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits."

The Completion Times of the Required Actions are also applicable when a system or component is removed from service intentionally. The reasons for intentionally relying on the ACTIONS include, but are not limited to, performance of Surveillances, preventive maintenance, corrective maintenance, or investigation of operational problems. Entering ACTIONS for these reasons must be done in a manner that does not compromise safety. Intentional entry into ACTIONS should not be made for operational

(continued)

BASES

LCO 3.0.2 (continued)

convenience. Alternatives that would not result in redundant equipment being inoperable should be used instead. Doing so limits the time both subsystems/trains of a safety function are inoperable and limits the time other conditions could exist which result in LCO 3.0.3 being entered. Individual Specifications may specify a time limit for performing an SR when equipment is removed from service or bypassed for testing. In this case, the Completion Times of the Required Actions are applicable when this time limit expires, if the equipment remains removed from service or bypassed.

When a change in MODE or other specified condition is required to comply with Required Actions, the unit may enter a MODE or other specified condition in which another Specification becomes applicable. In this case, the Completion Times of the associated Required Actions would apply from the point in time that the new Specification becomes applicable, and the ACTIONS Condition(s) are entered.

LCO 3.0.3

LCO 3.0.3 establishes the actions that must be implemented when an LCO is not met; and:

a. An associated Required Action and Completion Time is not met and no other Condition applies; or

b. The condition of the unit is not specifically addressed by the associated ACTIONS. This means that no combination of Conditions stated in the ACTIONS can be made that exactly corresponds to the actual condition of the unit. Sometimes, possible combinations of Conditions are such that entering LCO 3.0.3 is warranted; in such cases, the ACTIONS specifically state a Condition corresponding to such combinations and also that LCO 3.0.3 be entered immediately.

This Specification delineates the time limits for placing the unit in a safe MODE or other specified condition when operation cannot be maintained within the limits for safe operation as defined by the LCO and its ACTIONS. It is not intended to be used as an operational convenience that permits routine voluntary removal of redundant systems or components from service in lieu of other alternatives that would not result in redundant systems or components being inoperable.

(continued)

BASES

LCO 3.0.3 (continued)

Upon entering into LCO 3.0.3, 1 hour is allowed to prepare for an orderly shutdown before initiating a change in unit operation. This includes time to permit the operator to coordinate the reduction in electrical generation with the load dispatcher to ensure the stability and availability of the electrical grid. The time limits specified to reach lower MODES of operation permit the shutdown to proceed in a controlled and orderly manner that is well within the specified maximum cooldown rate and within the capabilities of the unit. The specified Completion Times assume the availability of nonsafety related systems that are normally used during a unit cooldown. All or some of these systems will be used, as necessary, to establish and maintain the required MODE. Using only the passive, safety related systems, the time to reach MODE 4 is not expected to exceed the specified Completion Time (~~24~~ 48 hours) and ~~to~~ will vary, depending upon the specific systems used and the initial and transitory plant conditions experienced during the MODE change. For example, the unit can be brought to MODE 4 in approximately 37 hours using the Passive Residual Heat Removal System. Regardless of the cooldown systems available, the MODE changes will be completed in a controlled and orderly manner that is well within the specified maximum cooldown rate and within the capabilities of the available cooldown systems. This reduces thermal stresses on components of the Reactor Coolant System and the potential for a plant upset that could challenge safety systems under conditions to which this specification applies. The use and interpretation of specified times to complete the actions of LCO 3.0.3 are consistent with the discussion of Section 1.3, "Completion Times".

Although nonsafety related systems ~~such as startup feedwater (for steam generator heat removal) and normal residual heat removal along with the associated support systems~~ are necessary for RCS cooldown to less than 200°F (MODE 5) within the Completion Times specified, OPERABILITY of ~~these or other~~ nonsafety related systems is not required or implied by these Technical Specifications.

If nonsafety related systems required to complete a shutdown in accordance with LCO 3.0.3 are not available, the systems shall be restored as needed to accomplish the shutdown within the specified completion time. System restoration shall be completed as soon as possible; however, unit conditions shall also be considered by the operators such that overall safety is optimized. These considerations may include delay of repairs or use of temporary equipment.

(continued)

BASES

LCO 3.0.3 (continued)

A unit shutdown required in accordance with LCO 3.0.3 may be terminated, and LCO 3.0.3 exited if any of the following occurs:

a. The LCO is now met.

b. A Condition exists for which the Required Actions have now been performed.

c. ACTIONS exist that do not have expired Completion Times. These Completion Times are applicable from the point in time that the Condition was initially entered and not from the time LCO 3.0.3 is exited.

The time limits of Specification 3.0.3 allow ~~48 hours~~ 7 days (168 hours) for the unit to be in MODE 5 when a shutdown is required during MODE 1 operation. If the unit is in a lower MODE of operation when a shutdown is required, the time limit for reaching the next lower MODE applies. If a lower MODE is reached in less time than allowed, however, the total allowable time to reach MODE 5, or other applicable MODE is not reduced. For example, if MODE 3 is reached in 2 hours, then the time allowed for reaching MODE 5 is the next 166 ~~46~~ hours, because the total time for reaching MODE 5 is not reduced from the allowable limit of ~~48 hours~~ 7days. Therefore, if remedial measures are completed that would permit a return to MODE 1, a penalty is not incurred by having to reach a lower MODE of operation in less than the total time allowed.

In MODES 1, 2, 3, and 4 LCO 3.0.3 provides actions for Conditions not covered in other Specifications. The requirements of LCO 3.0.3 do not apply in MODES 5 and 6 because the unit is already in the most restrictive condition required by LCO 3.0.3. The requirements of LCO 3.0.3 do not apply in other specified conditions of the Applicability (unless in MODES 1, 2, 3, or 4) because the ACTIONS of individual Specifications sufficiently define the remedial measures to be taken.

Examples of the required end states specified for inoperable passive systems while in MODES 5 and 6 are provided in Table B 3.0-1, Passive Systems Shutdown MODE Matrix. These requirements are specified in the individual Specifications. The required end states specified for passive systems, when the unit is in MODE 5 or 6, are selected to ensure that the initial conditions and system and equipment availabilities minimize the likelihood and consequences of potential shutdown events.

(continued)

Table B 3.0-1 (page 1 of 1)
Passive Systems Shutdown MODE Matrix

LCO Applicability	Automatic Depressuration System	Core Makeup Tank	Passive RHR	IRWST	Containment	Containment Cooling
MODE 5 RCS pressure boundary closed	9 of 10 paths OPERABLE All paths closed LCO 3.4.13	One CMT OPERABLE LCO 3.5.3	System OPERABLE LCO 3.5.5	One injection flow path and one recirculation sump flow path OPERABLE LCO 3.5.7	Closure capability LCO 3.6.8	Two water flow paths OPERABLE LCO 3.6.7
Required End State	MODE 5 RCS pressure boundary open, visible level in pressurizer	MODE 5 RCS pressure boundary open, visible level in pressurizer	MODE 5 RCS pressure boundary open, visible level in pressurizer	MODE 5 RCS pressure boundary closed, visible level in pressurizer	MODE 5 RCS pressure boundary closed, visible level in pressurizer	MODE 5 RCS pressure boundary closed, visible level in pressurizer
MODE 5 RCS pressure boundary open	Stages 1, 2, and 3 open 2 stage 4 valves OPERABLE LCO 3.4.14	None	None	One injection flow path and one recirculation sump flow path OPERABLE LCO 3.5.7	Closure capability LCO 3.6.8	Two water flow paths OPERABLE LCO 3.6.7
Required End State	MODE 5 RCS pressure boundary open, visible level in pressurizer			MODE 5 RCS pressure boundary closed, visible level in pressurizer	MODE 5 RCS pressure boundary closed, visible level in pressurizer	MODE 5 RCS pressure boundary closed, visible level in pressurizer
MODE 5 RCS pressure boundary open, reduced RCS inventory	Stages 1, 2, and 3 open 2 stage 4 valves OPERABLE LCO 3.4.14	None	None	One injection flow path and one recirculation sump flow path OPERABLE LCO 3.5.8	Closure capability LCO 3.6.8	Two water flow paths OPERABLE LCO 3.6.7
Required End State	MODE 5 RCS pressure boundary open, visible level in pressurizer			MODE 5 RCS pressure boundary closed, visible level in pressurizer	MODE 5 RCS pressure boundary closed, visible level in pressurizer	MODE 5 RCS pressure boundary closed, visible level in pressurizer
MODE 6 Reactor internals in place, refueling cavity not full	Stages 1, 2, and 3 open 2 stage 4 valves OPERABLE LCO 3.4.14	None	None	One injection flow path and one recirculation sump flow path OPERABLE LCO 3.5.8	Closure capability LCO 3.6.8	Two water flow paths OPERABLE LCO 3.6.7
Required End State	MODE 6 Reactor internals removed, refueling cavity full			MODE 6 Reactor internals removed, refueling cavity full	MODE 6 Reactor internals removed, refueling cavity full	MODE 6 Reactor internals removed, refueling cavity full
MODE 6 Reactor internals removed, refueling cavity full	None	None	None	None	Closure capability LCO 3.6.8	None
Required End State					MODE 6 Reactor internals removed, refueling cavity full	

BASES

LCO 3.0.3 (continued)

Exceptions to 3.0.3 are provided in instances where requiring a unit shutdown in accordance with LCO 3.0.3, would not provide appropriate remedial measures for the associated condition of the unit. An example of this is in LCO 3.7.5, Spent Fuel Pool Water Level. This Specification has an Applicability of "At all times." Therefore, this LCO can be applicable in any or all MODES. If the LCO and the Required Actions of LCO 3.7.5 are not met while in MODES 1, 2, or 3, there is no safety benefit to be gained by placing the unit in a shutdown condition. The Required Action of LCO 3.7.5 of "Suspend movement of irradiated fuel assemblies in the spent fuel pool" is the appropriate Required Action to complete in lieu of the actions of LCO 3.0.3. These exceptions are addressed in the individual Specifications.

LCO 3.0.4

LCO 3.0.4 establishes limitations on changes in MODES or other specified conditions in the Applicability when an LCO is not met. It precludes placing the unit in a MODE or other specified condition stated that Applicability (e.g., Applicability desired to be entered) when the following exist:

a. Unit conditions are such that the requirements of the LCO would not be met in the Applicability desired to be entered; and

b. Continued noncompliance with the LCO requirements, if the Applicability were entered, would result in the unit being required to exit the Applicability desired to be entered to comply with the Required Actions.

Compliance with Required Actions that permit continued operation of the unit for an unlimited period of time in a MODE or other specified condition provides an acceptable level of safety for continued operation. This is without regard to the status of the unit before or after the MODE change. Therefore, in such cases, entry into a MODE or other specified condition in the Applicability may be made in accordance with the provisions of the Required Actions.

(continued)

FAX to DINO SCALETTI

March 17, 1997

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

Don Lindgren
Richard Orr
Ed Cummins
Bob Vijuk
Brian McIntyre

OPEN ITEM #681 (DSER 3.8.2.4-3)

THIS IS AN INFORMAL RESUBMITTAL OF INFORMATION TO OBTAIN NRC ACKNOWLEDGMENT OF RECEIPT.

To meet the SECY-97-051 schedule of "Applicant Submits Final SSAR Revisions & Documentation" by 5/97, we believe that NRC must acknowledge receipt of all Westinghouse submittals by May 30, 1997. This is just 75 calendar days away.

The relevant documentation related to Open Item #681 (DSER 3.8.2.4-3) is attached to today's fax on this item. We provided the original response for this item in NSD-NRC-97-4981 of February 11, 1997 (over a month ago). We provided this information with a request for NRC Status change on February 17, 1997 (a month ago). We revised SSAR subsection 3.8.2.4.1.2 to reflect this information in Revision 11, February 28, 1997. We believe that this information resolves the concerns of item #681. It seems a reasonable request that NRC acknowledge receipt of the information and that NRC has a responsibility to recognize that Westinghouse, as an applicant has submitted the requested 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 681 And 681 Sorted by Item #

Item No.	Branch	DSER Section/ Question	Type	Title/Description Detail Status	Resp Engineer	(W) Status	NRC Status	Letter No. / Date
681	NRR/ECGB	3.8.2.4-3	DSER-OI		Orr /WSL / NRCCV	Closed	Action W	NTD-NRC-95-4464

Westinghouse should demonstrate that calculated stresses in the vicinity of the concentrated masses based on an equivalent static analysis bound the local stresses computed by the dynamic analysis.

Local dynamic analyses are performed for the responses of the local masses using as input the floor response spectra at the appropriate elevation of the containment vessel. The local analyses have been added in SSAR Rev. 3.
Action W - Discussed at meeting at CBI 8/30 - 31/95.
Westinghouse stated that detailed analyses and design of the
containment vessel in the vicinity of concentrated masses are beyond the scope of the AP600 standard design. However, Westinghouse agreed to expand SSAR Section 3.8.2.4.1.2 to include (1) a detailed description of methods to be used for the dynamic analysis of local masses, (2) the approach for analyzing the local buckling potential of the containment shell adjacent to major penetrations, (3) the stress redistribution criteria to be applied for the shell adjacent to local masses, and (4) methods for
evaluating the compressive strength of the containment shell in the vicinity of major penetrations.
Action W The draft SSAR markup is incomplete. Westinghouse does not describe the dynamic evaluation for local masses.
Closed - Response provided by NSD-NRC-97-4981 of 2/11/97.

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Open Item # 681- DSER Open Item 3.8.2.4-3

Based on the staff's review experience of other nuclear power plants, local high stresses may occur in the vicinity of the concentrated masses such as the equipment hatches and personnel airlocks. Westinghouse was requested to demonstrate that calculated stresses in the vicinity of the concentrated masses such as equipment hatches and personnel airlocks based on an equivalent static analysis bound the local stresses computed by the dynamic analysis. This was open item 3.8.2.4-3. In the August 30 through 31, 1995 review meeting, Westinghouse stated that detailed analyses and design of the containment vessel in vicinity of concentrated masses are beyond the scope of the AP600 standard design. However, Westinghouse agreed to expand SSAR Section 3.8.2.4.1.2 to include:

1. A detailed description of methods to be used for the dynamic analysis of local masses
2. The approach for analyzing the local buckling potential of the containment shell adjacent to major penetrations.
3. The stress redistribution criteria to be applied for the shell adjacent to the local masses, and
4. Methods for evaluating the compressive strength of the containment shell in the vicinity of major penetrations.

<u>Westinghouse Response</u>

Replace existing SSAR subsection 3.8.2.4.1.2 with the following:

3.8.2.4.1.2 Local Analyses

The penetrations and penetration reinforcements are designed in accordance with the rules of ASME III, Subsection NE. ~~The dynamic response of the local concentrated mass is considered in local analyses of the shell and is included in the design.~~ The design of the large penetrations for the two equipment hatches and the two airlocks use the results of finite element analyses which consider the effect of the penetration and its dynamic response as follows:

1. The upper airlock and equipment hatch penetrations are modeled in individual finite element models. The lower airlock and equipment hatch are modeled in a combined finite element model (Figure 3.7.2-8) including the boundary conditions representing the embedment. The finite element models include a portion of the shell sufficient that the boundary conditions do not affect the results of the local analyses.

2. Surface loads are applied for pressure and inertia loads on the shell included in the model. Loads corresponding to the stresses in the unpenetrated vessel at the location of the penetration, obtained from the axisymmetric analyses described in the previous subsection, are applied as boundary conditions for the local finite element models.

3. The out-of-plane stiffness of the containment vessel is determined for unit radial loads and moments at the location of the penetration. The frequency of the local radial and rotational modes are calculated using single degree of freedom models with mass and rotational inertias of the penetration. Seismic response accelerations for the radial and rotational modes are determined from the applicable floor response spectra for the containment vessel. Equivalent static radial loads and moments are calculated from these seismic response accelerations

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4. Radial loads and moments due to the local seismic response and due to external loads on the penetration are applied statically at the location of the penetration. These loads are applied individually corresponding to the three directions of input (radial, tangential and vertical). The three directions of seismic input are combined by the square root sum of the squares method or by the 100%, 40%, 40% method as described in subsection 3.7.2.

5. Stresses due to local loads on the penetration (step 4) are combined with those from the global vessel analyses (step 2). Stresses are evaluated against the stress intensity criteria of ASME Section III, Subsection NE. Stability is evaluated against ASME Code Case N-284, Revision 1. Local stresses in the regions adjacent to the major penetrations are evaluated in accordance with paragraph 1711 of the code case. Stability is not evaluated in the reinforced penetration neck and insert plate which are substantially stiffer than the adjacent shell.

The 16 foot diameter equipment hatch located at elevation 112′ 6″ and the personnel airlock located at elevation 110′ 6″ are in close proximity to each other and to the concrete embedment. Design of these penetrations uses the finite element model shown in Figure 3.8.2-7. Static analyses are performed for dead loads and containment pressure. Response spectrum analyses are performed for seismic loads. Stresses are evaluated as described for the single penetrations in step 5 above.

Finite element analyses are performed to confirm that the design of the penetration in accordance with the ASME code provides adequate margin against buckling ~~safety factors~~. A finite element ANSYS model, as shown in Figure 3.8.2-7, represents the portion of the vessel close to the embedment with the lower equipment hatch and personnel airlock. This is analyzed for external pressure and axial loads and demonstrates that the penetration reinforcement is sufficient and precludes buckling close to the penetrations. ~~and that the~~ The lowest buckling mode occurs in the shell away from the penetrations and embedment.

Open Item # 698 - DSER Open Item 3.8.2.4-20

Westinghouse should provide the leakage estimate through penetrations such as equipment hatches and personnel airlocks.

Westinghouse response

The treatment and the description of the leakage modeling in the severe accident fission product source term analysis is found in chapter 45 of the PRA report, Revision 3. The leakage area in the severe accident is equal to that corresponding to the specified containment leakage of 0.12% at design basis conditions. There is no increase in leakage area caused by containment pressurization. The ultimate pressure capacity for containment function is calculated to occur once the general membrane stresses in the shell reach yield. Thus the general membrane shell remains elastic for pressures up to this ultimate capacity and increased leakage area is not expected due to pressure. Leakage after general membrane yield of the containment cylinder is assumed as containment failure as discussed in the response to RAI 220.99 in letter NSD-NRC-96-4904, dated 12/9/96.

The bottom head is embedded in the concrete base at elevation 100 feet. This leads to circumferential compressive stresses at the discontinuity under thermal loading associated with the design basis accident. The containment vessel design includes a Service Level A combination in which the vessel above elevation 100′ is conservatively specified at the design temperature of 280°F and the portion of the embedded vessel (and concrete) is specified at a temperature of 70°F. Containment shell buckling close to the base is evaluated against the criteria of ASME Code, Case N-284, Revision 1, using a BOSOR-5 model of the portion of the shell above elevation 100′ extending up to the horizontal stiffener at elevation 132′ 3″. Material yield and stiffness properties are based on properties at the design temperature of 280°F. Temperature differences are raised by small increments until buckling is predicted. Buckling occurred 20 inches above elevation 100′ for a circumferential wave number, N = 190, at a factor of 6.0 times the design differential temperature condition. The half buckling wave length is less than 0.5 $\sqrt{(rt)}$. This is not a significant buckling issue; buckling did not occur for wave numbers below N = 60, which is the critical range for the cylinder and top head under external and internal pressure.

3.8.2.4.1.2 Local Analyses

The penetrations and penetration reinforcements are designed in accordance with the rules of ASME III, Subsection NE. The design of the large penetrations for the two equipment hatches and the two airlocks use the results of finite element analyses which consider the effect of the penetration and its dynamic response as follows:

1. The upper airlock and equipment hatch penetrations are modeled in individual finite element models. The lower airlock and equipment hatch are modeled in a combined finite element model (Figure 3.7.2-8) including the boundary conditions representing the embedment. The finite element models include a portion of the shell sufficient that the boundary conditions do not affect the results of the local analyses.

2. Surface loads are applied for pressure and inertia loads on the shell included in the model. Loads corresponding to the stresses in the unpenetrated vessel at the location of the penetration, obtained from the axisymmetric analyses described in the previous subsection, are applied as boundary conditions for the local finite element models.

3. The out-of-plane stiffness of the containment vessel is determined for unit radial loads and moments at the location of the penetration. The frequency of the local radial and rotational modes are calculated using single degree of freedom models with mass and rotational inertias of the penetration. Seismic response accelerations for the radial and rotational modes are determined from the applicable floor response spectra for the containment vessel. Equivalent static radial loads and moments are calculated from these seismic response accelerations.

4. Radial loads and moments due to the local seismic response and due to external loads on the penetration are applied statically at the location of the penetration. These loads are applied individually corresponding to the three directions of input (radial, tangential and vertical). The three directions of seismic input are combined by the square root

sum of the squares method or by the 100%, 40%, 40% method as described in subsection 3.7.2.6.

5. Stresses due to local loads on the penetration (step 4) are combined with those from the global vessel analyses (step 2). Stresses are evaluated against the stress intensity criteria of ASME Section III, Subsection NE. Stability is evaluated against ASME Code Case N-284, Revision 1. Local stresses in the regions adjacent to the major penetrations are evaluated in accordance with paragraph 1711 of the code case. Stability is not evaluated in the reinforced penetration neck and insert plate which are substantially stiffer than the adjacent shell.

The 16 foot diameter equipment hatch located at elevation 112′ 6″ and the personnel airlock located at elevation 110′ 6″ are in close proximity to each other and to the concrete embedment. Design of these penetrations uses the finite element model shown in Figure 3.8.2-7. Static analyses are performed for dead loads and containment pressure. Response spectrum analyses are performed for seismic loads. Stresses are evaluated as described for the single penetrations in step 5 above.

Finite element analyses are performed to confirm that the design of the penetration in accordance with the ASME code provides adequate margin against buckling. A finite element ANSYS model, as shown in Figure 3.8.2-7, represents the portion of the vessel close to the embedment with the lower equipment hatch and personnel airlock. This is analyzed for external pressure and axial loads and demonstrates that the penetration reinforcement is sufficient and precludes buckling close to the penetrations. The lowest buckling mode occurs in the shell away from the penetrations and embedment.

3.8.2.4.2 Evaluation of Ultimate Capacity

The capacity of the containment vessel has been calculated for internal pressure loads for use in the probabilistic risk assessment analyses and severe accident evaluations. Each element of the containment vessel boundary was evaluated to estimate the maximum pressure at an ambient temperature of 100°F corresponding to the following stress and buckling criteria:

- Deterministic severe accident pressure capacity corresponding to ASME Service Level C limits on stress intensity, ASME paragraph NE-3222, and ASME Code Case N-284 for buckling of the equipment hatch covers, and 60 percent of critical buckling for the top head. The deterministic severe accident pressure capacity corresponds to the approach in SECY 93-087, to maintain a reliable leak-tight barrier approximately 24 hours following the onset of core damage under the more likely severe accident challenges. This approach was approved by the Nuclear Regulatory Commission as outline in the Staff Requirements Memorandum on SECY-93-087 - Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light Water Reactor (ALWR) Designs, Dated July 21, 1993.

FAX to DINO SCALETTI

March 17, 1997

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

Don Lindgren
Richard Orr
Ed Cummins
Bob Vijuk
Brian McIntyre

OPEN ITEM #706 (DSER 3.8.2.4-28)

THIS IS AN INFORMAL RESUBMITTAL OF INFORMATION TO OBTAIN NRC ACKNOWLEDGMENT OF RECEIPT.

To meet the SECY-97-051 schedule of "Applicant Submits Final SSAR Revisions & Documentation" by 5/97, we believe that NRC must acknowledge receipt of all Westinghouse submittals by May 30, 1997. This is just 75 calendar days away.

The relevant documentation related to Open Item #706 (DSER 3.8.2.4-28) is attached to today's fax on this item. We provided the original response for this item in NSD-NRC-96-4904 of December 9, 1996 (over 3 months ago) as a response to RAI 480.191. We then answered the same question, now as a DSER item, in NSD-NRC-97-4981 of February 11, 1997 (over a month ago). We provided this information with a request for NRC Status change on February 17, 1997 (a month ago). We believe that this information resolves the concerns of item #706. It seems a reasonable request that NRC acknowledge receipt of the information and that NRC has a responsibility to recognize that Westinghouse, as an applicant has submitted the requested 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 5

Item No.	Branch	DSER Section/ Question	Type	Title/Description Detail Status	Resp Engineer	(W) Status	NRC Status	Letter No. /	Date
706	NRR/ECGB	3.8.2.4-28	DSER-OI		Orr	Closed	Action W	NTD-NRC-95-4464	

Westinghouse should provide in the SSAR an assessment of the pressure capability of the main steamline and main feedwater line bellows, a corresponding failure probability distribution curve, and the impact on the overall cumulative failure probability curve.

The effect of containment pressure on the bellows was addressed in the response to RAI 720.206. This response has been incorporated in SSAR Subsection 3.8.2.4.2.6, Rev 3. The bellows remain intact when the containment shell remains elastic and imposed deflections remain close to the design conditions. Failure of the bellows is assumed to occur once the containment cylinder yields. This mechanism is already included in the failure probability curve for the cylinder.

Discussed in meeting at CBI 8/30 - 31/95. Additional discussion was provided in draft SSAR or PRA report that bellows failure mode and fragility is included in the containment vessel cylinder yield failure mode.
Closed: further review is under new RAI 220.99 transmitted by NRC letter of April 4, 1996

NRC Status Update provided in September 5, 1996 letter:
This staff does not agree that this item is closed or resolved due to RAI# 220.99. Both this open item and OI# 3268 (RAI #220.99) should be tracked individually to resolution. Action Westinghouse
Closed - Response provided by NSD-NRC-97-4981 of 2/11/97.

Westinghouse Electric Corporation

Energy Systems

Box 355
Pittsburgh Pennsylvania 15230-0355

NSD-NRC-96-4904
DCP/NRC0674
Docket No.: STN-52-003

December 9, 1996

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

ATTENTION: T. R. QUAY

SUBJECT: AP600 RESPONSE TO REQUESTS FOR ADDITIONAL INFORMATION, AND AP600 PRA PAGE MARKUPS

Dear Mr. Quay:

Enclosure 1 provides Westinghouse responses to NRC requests for additional information pertaining to the conditional containment failure probability distribution used in Chapter 42 of the AP600 Probabilistic Risk Assessment (PRA). Specifically, the responses to the following RAIs are included: 480.190 through 480.192, 220.95 through 220.99. These responses close, from a Westinghouse perspective, the addressed questions. The NRC technical staff should review these responses. The status of these RAIs will be changed to "Action N" in the OITS on January 2, 1997.

Enclosure 2 provides the draft responses to NRC requests for additional information pertaining to the AP600 at-power fire PRA. These responses are being provided to the NRC for discussion purposes at the December 18, 1996 NRC and Westinghouse AP600 fire PRA meeting. The staff is expected to review these draft responses and be prepared to discuss them at the meeting.

Enclosure 3 contains markup page changes to the AP600 PRA. These markups primarily fix typos found in the report since the issuance of Revision 8. These changes will be included in the next revision to the PRA. The staff reviewers should include these markup pages with their PRA report.

A listing of the NRC requests for additional information responded to in this letter is contained in Attachment A.

Please contact Cynthia L. Haag on (412) 374-4277 if you have any questions concerning this transmittal.

for

Brian A. McIntyre, Manager
Advanced Plant Safety and Licensing

3013A

3 of 5

Question: 480.191 Containment Pressure Capacity

Provide an assessment of the pressure capability of the main steam line and main feedline bellows, and a corresponding failure probability distribution curve.

Response:

The pressure capability of the main steam line and main feedline bellows are discussed in SSAR subsection 3.8.2.4.5. The pressure capability exceeds the pressure at which the containment vessel cylinder yields. It is assumed that once general yielding of the cylinder occurs, the bellows may fail due to the large deflection of the cylinder. However,, it is not necessary to provide the probability of failure for bellows separate from the probability of cylinder yield. The probability of failure of the bellows due to large deformation of the cylinder is a part of the probability given in PRA Chapter 42 for vessel failure due to the general yielding of the cylinder.

PRA Revision: None.

4 of 5

Open Item #706 - DSER Open Item # 3.8.2.4-28

Westinghouse should provide in the SSAR an assessment of the pressure capability of the main steamline and main feedwater line bellows, a corresponding failure probability distribution curve, and the impact on the overall cumulative failure probability curve.

Westinghouse response

This is the same question as RAI 480.191. See response to RAI 480.191 provided in letter NSD-NRC-96-4904, dated 12/9/96.

Open Item #1888 - DSER Open Item # 3.8.2.4-1

The COL applicant should demonstrate that EPAs to be used shall be at least as strong as the AP600 SCV.

Westinghouse response

Additional information has been included in SSAR subsection 3.8.2.4.2.5, Revision 7. See also response to RAI 220.102 (OI# 3271).

Open Item #2515 - Telecon June 23, 1995

Westinghouse should address the issue of fatigue and corrosion of containment bellows. The number of thermal cycles and loading information included in the design specification should be addressed. The material requirements and effect of corrosion should also be included.

Westinghouse response

SSAR section 3.8.2.1.5, Rev 7 has been revised to include material and additional information on the displacement cycles. Fatigue is evaluated in accordance with ASME subsection NE as stated in SSAR subsection 3.8.2.1.5. Bellows materials are stainless steel or nickel alloy. Corrosion is not expected; if there is any degradation it would be observed by inservice inspection or testing. The bellows are included in the ISI of the containment vessel as well as the containment leak rate testing.

DSER Open Item # 3269 (NRC letter dated 4/4/96) RAI # 220.100

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

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

FAX to DINO SCALETTI

March 17, 1997

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

Don Hutchings
Don Lindgren
Ed Cummins
Bob Vijuk
Brian McIntyre

OPEN ITEM #302 (M9.4.11-1)

To meet the SECY-97-051 schedule of "**Applicant** Submits **Final SSAR Revisions & Documentation**" by **5/97**, we **believe** that **NRC** must **acknowledge receipt of all Westinghouse submittals** by **May 30, 1997.** This is just **75** calendar days **away.** In my quest to make sure we have provided NRC with everything needed to prepare an FSER, I have been providing background packages for open items that we believe are complete. Relevant documentation related to Open Item #302 (M9.4.11-1) is attached. We agreed to update Table 3.2-3 to include the balance of the components in VHS over three months ago. We provided Revision 11 of the AP600 SSAR on February 28, 1997. A week later we provided an informal redline/strikeout version as requested to ease comparison between SSAR revisions. We believe that this information resolves the concerns of item #302 and that NRC has a responsibility to acknowledge its receipt. 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" or "Closed". I will be sending this message by both fax and E-Mail. Thank you.

Jim

Jim Winters
412-374-5290

AP600 Open Item Tracking System Database: Executive Summary

Date: 3/17/97

Selection: [item no] between 302 And 302 Sorted by Item #

Item No.	Branch	DSER Section/ Question	Type	Title/Description Detail Status	Resp Engineer	(W) Status	NRC Status	Letter No. /	Date
302	NRR/SPLB	9.4.11	MTG OI		3.2-3/Winters	Closed	Action W		

M9.4.11-1 (HEALTH PHYSICS AND HOT MACHINE SHOP HVAC SYSTEM) Update Table 3.2-3 of the SSAR for VHS components including unlisted system dampers and hot machine shop filtration unit subsystem components.

Closed. Revision of SSAR Table 3.2-3 includes safety-related items in the VHS

Action W - same as OITS 285

Closed - Table 3.2-3 revised to include balance of systems components in Revision 11, 2/28/97. jww

2 of 3

Total Records: 1

Table 3.2-3 (Sheet 57 of 64)

AP600 CLASSIFICATION OF MECHANICAL AND FLUID SYSTEMS, COMPONENTS, AND EQUIPMENT

Tag Number	Description	AP600 Class	Seismic Category	Principal Construction Code	Comments
Containment Air Filtration System (Continued)					
VFS-PY-S02	Containment Air Exhaust Strainer	C	I	ASME Sec. III Class 3	
VFS-PL-V001	Containment Isolation Test Connection	B	I	ASME III-2	
VFS-PL-V002	Containment Isolation Test Connection	B	I	ASME III-2	
VFS-PL-V007	Containment Isolation Test Connection	B	I	ASME III-2	
VFS-PL-V008	Containment Isolation Test Connection	B	I	ASME III-2	
VFS-PL-V009	Containment Purge Discharge Containment Isolation Valve	B	I	ASME III-2	
VFS-PL-V010	Containment Purge Discharge Containment Isolation Valve	B	I	ASME III-2	
VFS-PL-V012	Containment Isolation Test Connection	B	I	ASME III-2	
VFS-PL-V015	Containment Isolation Test Connection	B	I	ASME III-2	
n/a	Valves Providing VFS AP600 Equipment Class D Function	D	NS	ANSI 16.34	
n/a	Dampers in lines isolating radioactive contamination	R	NS	ASME-509	
n/a	Shutoff, Isolation, and Balancing Dampers	L	NS	ANSI/AMCA-500	
n/a	Fire Dampers	Note 3	NS	UL-555	
Balance of system components are Class L and Class R					
Health Physics and Hot Machine Shop HVAC System (VHS)				Location: Annex Building	
n/a	Shutoff, Isolation, and Balancing Dampers	L	NS	ANSI/AMCA-500	
n/a	Fire Dampers	Note 3	NS	UL-555	
Balance of system components are Class E or Class L					

FAX to TOM KENYON

Sharon or Tom, please make a copy for C. Hinson 3/24/97

cc: R. Meuschke
G. Israelson
D. Lindgren
B. McIntyre
E. Cummins
M.Mahlab

This is additional information about new fuel handling so that NRC can continue its review of our compliance with 10CFR70.24. It is related to OITS #21 (RAI 471.24) and may be used to revise our answer to that RAI is requested by NRC.

The AP600 fuel storage and handling systems and equipment will, wherever possible, use simple proven technology. The best and most reliable features and equipment from existing nuclear and commercial plant will be used in the design of the AP600. Designs for fuel storage will be conservative. Control systems will use proven reliable industrial control components. Control software will include mistake avoidance techniques to preclude as much, as possible, operators doing the wrong thing. Equipment will include self diagnostic, set up, calibration, and troubleshooting aids to enhance maintainability and system reliability.

New fuel is received in the Spent Fuel Building ground level truck/train bay. The new fuel containers are unloaded, the removable hatch covers are removed from the 135' level deck and the containers are raised to that level. Room is provided at the west end, for temporary storage of these containers if necessary. The containers are moved one at a time using a cart designed for this purpose to the jib crane area at the east end. The container covers are then removed and the fuel is upended and removed from the container using the jib crane. Inspections are completed and the new fuel is stored in the new fuel storage racks until needed.

When needed the new fuel is removed from the new fuel storage racks using the jib crane and placed in the new fuel elevator in the south west corner of the spent fuel pool. The fuel is lowered into the pool to a point where it can be picked up by the spent fuel handling machine. The spent fuel handling machine then places the fuel assembly in the spent fuel rack.

The overhead crane is a heavy lift (150 tons) crane used for lifting new fuel assembly containers and spent fuel shipping casks. The crane is designed according to the requirements of ASME NOG-1. This crane has mechanical stops to prevent its travel, and therefore the travel of heavy loads, over the spent fuel pool, new fuel storage racks and jib crane areas.

The jib crane is a light load device (2000 pounds) used for moving new fuel assemblies from their shipping containers to the new fuel storage racks and from the racks to the new fuel elevator. A short tool is used to attach the crane hook to the new fuel. The jib crane is seismic category II.

New fuel storage racks hold 56 fuel assemblies with sufficient spacing and shielding to prevent criticality under any postulated conditions including flooding with non borinated water. The racks are

seismic category I and are built to withstand dropping a fuel assembly and the lifting tool (1625 pounds) from 3 feet above the racks. This is considered the heaviest load able to be suspended above the rack since the fuel handling machine and the overhead cask loading crane are prevented from traveling over these racks by mechanical stops. The fuel assembly spacing in the rack is a conservative 10.9 inches center to center. Existing PWR plants have as low as 9.0 inch spacing without problems.

The new fuel elevator lowers new fuel assemblies from the fuel handling area operating floor into the spent fuel pool where they can be picked up by the fuel handling machine.

Additional details of some aspects of this design can be found in SSAR subsection 9.1.1. The subsection provides additional details about the mechanical design of the equipment described above and about criticality prevention features. This design preludes criticality of material in new fuel assemblies while in the new fuel handling area. We believe that sufficient justification exists for an exemption from compliance with the radiation monitoring for criticality requirements of 10CFR70.24(a) in accordance with 10CFR70.24 (d).

The radiation monitoring for this area is provided by the fuel handling area exhaust radiation monitor described in SSAR subsection 11.5.2.3.2. This monitor is a general area monitor and may be capable of detecting a criticality as defined in 10CFR70.24(a)(1). Although criticality is precluded by design, we could modify SSAR subsection 9.1.6 to require the COL holder to use a portable radiation monitor on the overhead cask loading crane, the jib crane and either of the fuel handling machines when in use to ensure detection of a higher radiation event.

Please review this and give us a telephone call to agree on the specific changes, if any, required for our exemptions request letter or the SSAR.

Thanks

Jim

Jim Winters
412-374-5290

FAX to DINO SCALETTI

March 17, 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)

To meet the SECY-97-051 schedule of "Applicant Submits Final SSAR Revisions & Documentation" by 5/97, we believe that NRC must acknowledge receipt of all Westinghouse submittals by May 30, 1997. This is just 75 calendar days away. The relevant documentation related to Open Item #184 (M5.4.11-7) is attached to today's fax on this item. We provided the original fax of a markup on January 13, 1997 (over two months ago). ~~We provided~~ this markup again with a request for NRC Status change on February 20, 1997 (Almost a month ago). We revised SSAR subsection 5.2.5.1.3 and 5.4.11.4 to reflect these markups in Revision 11, February 28, 1997. We believe that this information resolves the concerns of item #184. It seems a reasonable request that NRC acknowledge receipt of the information and that NRC has a responsibility to recognize that Westinghouse, as an applicant has submitted the requested 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: 3/17/97

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. Closed - SSAR subsections 5.2.5.1.3 and 5.4.11.4 revised in Revision 11, 2/28/97.	Corletti,M.	Closed	Action W		

4 of 2

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

5.2.5.1.3 Pressurizer Safety Relief and Automatic Depressurization Valves

Temperature is sensed downstream of each pressurizer safety relief valve and each automatic depressurization valve mounted on the pressurizer by a resistance temperature detector on the discharge piping just downstream of each valve. High temperature indications (alarms in the main control room) identify a reduction of coolant inventory as a result of seat leakage through one of the 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 or the in-containment refueling water storage tank during accident conditions. This identified leakage is measured by the change in level of the reactor coolant drain tank.

5.2.5.1.4 Reactor Coolant Pump Drain

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

5.2.5.1.5 Other Leakage Sources

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

5.2.5.2 Intersystem Leakage Detection

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. In accordance with position 4 of Regulatory Guide 1.45, 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.

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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.

5.4.11.4 Instrumentation Requirements

The instrumentation for the safety valve discharge pipe, containment, and in-containment refueling water storage tank are discussed in subsections 5.2.5, 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.

FAX to DINO SCALETTI

March 20, 1997

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

Robin Nydes
Steve Kerch
Ed Cummins
Bob Vijuk
Brian McIntyre

OPEN ITEMS FOR CHAPTER 13

To meet the SECY-97-051 schedule of "Applicant Submits Final SSAR Revisions & Documentation" by 5/97, we believe that NRC must acknowledge receipt of all Westinghouse submittals by May 30, 1997. This is just 72 calendar days (52 business days) away.

This is a background package for the remaining open items for Chapter 13. Chapter 13 is of interest because by our joint NRC/W schedule, the FSER for this chapter should be turned into Projects by the end of February. Attached to today's fax is a copy of the OITS entries for the 4 items showing "Action W" in "NRC Status". They are items 1222, 1225, 1226 and 2033.

Item 1222 was answered by the information submitted by NSD-NRC-97-4988 of February 21, 1997. A copy of the cover letter is attached to today's fax. We recommend that this item be declared "Action N" or "Resolved".

Item 1225 was answered by the addition of a COL item in subsection 13.5 of the SSAR. A copy of this SSAR section is attached to today's SSAR. Although the Status Detail for this item indicates that the item requires resolution of Chapter 18, we have provided all information required to review Chapter 18 and recommend that this item be declared "Action N".

Item 1226 was answered by the addition of a COL item in subsection 13.5 of the SSAR. A copy of this SSAR section is attached to today's SSAR. Although the Status Detail for this item indicates that the item requires resolution of Chapter 18, we have provided all information required to review Chapter 18 and recommend that this item be declared "Action N".

Item 2033 was answered by the information submitted by NSD-NRC-97-4988 of February 21, 1997. A copy of the cover letter is attached to today's fax. We recommend that this item be declared "Action N" or "Resolved".

It seems a reasonable request that NRC acknowledge receipt of this information. Our records show no outstanding Westinghouse actions for Chapter 13 and we request that NRC provide a definitive action for Westinghouse or provide direction to change the status of items (1222, 1225, 1226 and 2033). Thank you.

Jim

Jim Winters
412-374-5290

1 of 5

AP600 Open Item Tracking System Database: Executive Summary **Date: 3/20/97**

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

Item No.	Branch	DSER Section/ Question	Type	Title/Description Detail Status	Resp Engineer	(W) Status	NRC Status	Letter No. /	Date
1222	NRR/PERB	13.3-5	DSER-OI		MMIS/Kerch	Closed	Action W	NSD-NRC-96-4805	

Westinghouse should demonstrate the ability of the main control room to support the appropriate number of staff when the TSC is unavailable.

Action W - The ERG development will provide the task analysis necessary to determine the number of staff required to support main control room poeration and the required TSC functions. The capability of the MRC to support this staffing level will be provided.
Resolved - Per DCP/NRC0589, this item will be closed with submittal of th at-power ERG's.
This is an MMIS item. Westinghouse to write a letter to the NRC explaining when the TSC would be unavailable and that the staff would not be moved to the MCR. Expect letter by 12/30. This will address items 1222 and 2033. rkn 12/3
Letter not complete by 12/30, expect by Jan 31, 1997. rkn 1/15/97.
This letter, which transmits a revised response to RAI 100.10 is in review. rkn 2/18
Letter NSD-NRC-97-4988 (DCP/NRC0742) provides the revised response. rkn 2/26/97

Item No.	Branch	DSER Section/ Question	Type	Title/Description Detail Status	Resp Engineer	(W) Status	NRC Status	Letter No. /	Date
1225	NRR/HHFB	13.5.1-1	DSER-OI		Winters, J.	Closed	Action W		

Westinghouse should add COL Action Item 13.5.1-1 to the SSAR for the COL applicant to address administrative procedures for the plant.

Closed - A COL information item was added to Chapter 13.5, Revision 3, of the SSAR to address administrative proceedures for the plant

6/10/96 - NRC believes issue remains open pending resolution of the procedures section of Chapter 18.

Item No.	Branch	DSER Section/ Question	Type	Title/Description Detail Status	Resp Engineer	(W) Status	NRC Status	Letter No. /	Date
1226	NRR/HHFB	13.5.2-1	DSER-OI		Winters, J.	Closed	Action W		

Westinghouse should add COL Action Item 13.5.2-1 to the SSAR for the COL applicant to address operating and maintenance procedures for the plant.

Closed - A COL information item was added to Section 13.5, Revision 3, of the SSAR to address the development of operating and maintainance procedures for the plant.

6/10/96 - NRC believes issue remains open pending resolution of the procedures section of Chapter 18.

Item No.	Branch	DSER Section/ Question	Type	Title/Description Detail Status	Resp Engineer	(W) Status	NRC Status	Letter No. /	Date
2033	NRR/PERB	13.	DSER-OI50		MMIS/Kerch	Closed	Action W		

37. Habitability of Technical Support Center
The staff is concerned with the acceptability of the habitability requirements specified by Westinghouse for the TSC under accident conditions. Westinghouse has proposed that a detailed task analysis be performed post-certification to determine disposition of TSC staff when the facility is not habitable. The staff believes that this analysis should be performed pre-certification. (See DSER Open Item 13.3-4)

Need to determine what is needed to close this item. rkn 10/16/96
Met on 10/24 (Wills, Schulz, Kerch, Nydes) and developed plan to research this then discuss with NRC. Appears to be a RTNSS-related item.
To close this item, the letter being written for OITS item 1222 will close this (Kerch- author, Wills and Schulz to review). rkn 12/2
A revised response to RAI 100.10 is in review. rkn 2/18
Letter NSD-NRC-97-4988 provides the revised response. rkn 2/26/97

Westinghouse
Electric Corporation

Energy Systems

Box 355
Pittsburgh Pennsylvania 15230-0355

NSD-NRC-97-4988
DCP/NRC0742
Docket No.: STN-52-003

February 21, 1997

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

TO: T. R. QUAY

SUBJECT: REVISED RESPONSE TO RAI 100.10 FOR THE TECHNICAL SUPPORT CENTER (TSC)

Dear Mr. Quay:

Attached is Enclosure 1, the response to RAI 100.10, revised to address NRC concerns related to unavailability of the Technical Support Center. Given this revision of the RAI response, the Westinghouse status for DSER open item tracking system (OITS) items 1222 and 2033 is Closed as shown in Enclosure 2. Please review the attached RAI response and provide Westinghouse with the NRC status so that the OITS can be updated.

If you have any questions regarding the response to RAI 100.10, or this letter, please contact Robin K. Nydes at (412) 374-4125.

Brian A. McIntyre, Manager
Advanced Plant Safety and Licensing

jml

Enclosures

cc: N. J. Liparulo, Westinghouse (w/o Enclosures)
W. C. Huffman, NRC (w/Enclosures)

3-15

Section 9.4 for a description of the HVAC systems for the main control room/technical support center and the annex building. See Section 18.8 for the high level requirements for the technical support center and the operational support center. See Section 7.5 for identification of plant variables that are provided for interface to the emergency planning areas.

Communication interfaces among the main control room, the technical support center and the emergency planning centers are the responsibility of the Combined License applicant.

13.3.1 Combined License Information Item

Combined License applicants referencing the AP600 certified design will address emergency planning and its communication interface.

13.4 Operational Review

This section is the responsibility of the Combined License applicant.

13.4.1 Combined License Information Item

Combined License applicants referencing the AP600 certified design will address each operational review.

13.5 Plant Procedures

Plant procedures are the responsibility of the Combined License applicant. References to applicable combined license information are included in Section 1.8. This includes, for example, reference to guidelines on inservice inspection in Chapters 3 and 6, and initial testing in Chapter 14.

Reference 2 provides input to the Combined License applicant for the development of plant operating procedures, including information on the development and design of the AP600 emergency response guidelines and emergency operating procedures. Also included in Reference 2 is information on the computerized procedure system, which is the human system interface that allows the operators to execute the plant procedures.

13.5.1 Combined License Information Item

Combined License applicants referencing the AP600 certified design will address plant procedures including the following:

- Normal operation
- Abnormal operation
- Emergency operation

- Alarm response
- Maintenance, inspection, test and surveillance
- Administrative

13.6 Industrial Security

13.6.1 Preliminary Planning

Objectives and functional requirements of the AP600 physical protection system and the description of security features are provided in the AP600 Security Design Report, submitted under separate cover in accordance with 10 CFR 2.790(d), Rules of Practice. The report includes the security boundary drawings and the listing of the vital equipment and components. The vulnerability analysis, which demonstrates that the AP600 certified security design adequately protects the AP600 from radiological sabotage, is included in the report.

The AP600 security design features a reduced protected area that does not require a perimeter fence such as found in current plant security plans. This results in a reduced requirement for security staffing. Personnel screening, selection, performance evaluation, and training aspects of the physical security program will be addressed by the Combined License applicant.

13.6.2 Security Plan

The comprehensive physical security program, which is the responsibility of the Combined License applicant, will address the security plan, contingency plan, and guard training plan.

13.6.3 Plant Protection System

13.6.3.1 Introduction

A physical protection system and security organization is provided to protect the AP600 from radiological sabotage, as required by 10 CFR 73.55. To achieve this objective, the physical protection system:

- Includes a security organization

- Locates vital equipment within vital areas

- Controls points of personnel, vehicle, and material access into the protected and vital areas

- Annunciates alarms in a continuously manned central alarm station and at least one other continuously manned alarm station that is physically separated from the central alarm station

- Provides for continuous communications between the security officers and the continuously manned alarm stations

FAX to DINO SCALETTI

March 20, 1997

CC: Sharon or Dino, please make copies for: Tom Kenyon
Ted Quay
Don Lindgren
Jim Grover
Jim Sejvar
Gordon Israelson
Brian McIntyre
Ed Cummins
Bob Vijuk

OPEN ITEMS FOR CHAPTER 12

To meet the SECY-97-051 schedule of "Applicant Submits Final SSAR Revisions & Documentation" by 5/97, we believe that NRC must acknowledge receipt of all Westinghouse submittals by May 30, 1997. This is just 72 calendar days (52 business days) away.

This is a background package for the remaining open items for Chapter 12. Chapter 12 is of interest because by our joint NRC/W schedule, the FSER for this chapter should have been turned into Projects by the end of February. The three remaining items showing "Action W" for NRC Status are 19, 1210 and 5011.

For item 19, Westinghouse still owes some information and an SSAR markup for post accident dose evaluations based upon a telephone conversation with NRC. This information has been generated and is currently undergoing Westinghouse internal review. This is still "Action W".

Item 1210 requests information on the gap between the fuel transfer tube and the containment vessel. The SSAR revision answering this item is attached to the fax of this message. NRC obviously reviewed this information since its information is used in the new item we have numbered 5011. Since this item is a DSER-OI and requires the closure of item 5011, we recommend that this item be declared "Action N".

Item 5011 requires more detailed information on the gap shield addressed in item 1210. Although the SSAR entry may be acceptable, NRC requested additional details on the water filled bladder, the fuel transfer port shield structure and related personnel access restrictions. Information is prepared for this item and is awaiting the return of the shielding engineer from travel. This item is still "Action W".

It seems a reasonable request that NRC acknowledge receipt of the SSAR change for item 1210.

Thank you.

Jim

Jim Winters
412-374-5290

12.3.2.2.9 Spent Fuel Transfer Canal and Tube Shielding

The spent fuel transfer tube is shielded to within adjacent area radiation zone limits. This is primarily achieved through the use of concrete and water. The only removable shielding consists of concrete or steel hatches which reduce radiation in accessible areas to within those levels prescribed in the normal operation radiation zone maps (Figure 12.3-1).

The spent fuel transfer tube is completely enclosed in concrete and there is no unshielded portion of the spent fuel transfer tube during the refueling operation. The only potential radiation streaming path associated with the tube shielding configuration is the 2 inch (5.08 cm) seismic gap between the fuel transfer tube shielding and the steel containment wall. Shielding of this gap is provided by a water-filled bladder. This "expansion gap" radiation shield provides effective reduction of the radiation fields during fuel transfer and accommodates relative movement between the containment and the concrete transfer tube shielding with no loss in shield integrity. A removable hatch in the shield configuration provides access for inspection of the fuel transfer tube welds. The opening of this hatch is administratively controlled; this hatch is in place during the spent fuel transfer operation.

12.3.2.3 Shielding Calculational Methods

The shielding thicknesses provided for compliance with plant radiation zoning and to minimize plant personnel exposure are based on maximum equipment activities under the plant operating conditions described in Chapter 11 and Section 12.2. The thickness of each shield wall surrounding radioactive equipment is determined by approximating as closely as practicable the actual geometry and physical condition of the source or sources. The isotopic concentrations are converted to energy group sources using data from standard references (References 1 through 6).

The geometric model assumed for shielding evaluation of tanks, heat exchangers, filters, ion exchangers, and the containment is a finite cylindrical volume source. For shielding evaluation of piping, the geometric model is a finite shielded cylinder. In cases where radioactive materials are deposited on surfaces such as pipe, the latter is treated as an annular cylindrical surface source.

The computer code SHIELD-SG (Reference 11) is used to calculate dose rates. For complex geometries other computer codes such as QAD (Reference 16) are used. Buildup, calculated using Berger coefficients presented in ORNL-RSIC-10 (Reference 7) and Blizard's Method of Buildup Determination, presented in the Engineering Compendium on Radiation Shielding (Reference 8), is used for laminated shields.

The source activity (Ci) and gamma ray source strengths (MeV/sec) are calculated using one of the following computer codes: ORIGEN (Reference 17), SOURCE2/ACCUM (Reference 12), or RADGAS3 (Reference 13). ACCUM (Reference 12) is an option within SOURCE2 that computes isotope accumulation for several time periods from a given flow of isotopes in curies per second. This accumulated activity may then be decayed for any number of decay times at which gamma energy spectra and isotope Curie activity are computed. The

AP600 Open Item Tracking System Database: Executive Summary

Date: 3/20/97

Selection: [nrc st code]='Action W' And [DSER Section] like '12*' Sorted by Item #

Item No.	Branch	DSER Section/ Question	Type	Title/Description Detail Status	Resp Engineer	(W) Status	NRC Status	Letter No. /	Date
19	NRR/PERB	12.	RAI-OI		Winters/Sejvar	Action W	Action W	NTD-NRC-95-4414	3/13/95
				Question 471.22 (Radiation Protection) This is to formalize a request for additional information discussed during a telecon in June 1994. a. The staff's confirmatory shielding calculations confirmed, for the most part, the radiation zone levels described for the AP600 design during normal and accident conditions. However, these shielding calculations indicated rather high (approximately 95 Rem/hr) post-accident radiation levels in the PASS sample room. Determine the time it will take the operators to perform required post-accident actions in all vital areas (as required by 10 CFR 52.79(b) and described in Item II.B.2 of NUREG-0737), and provide estimated personnel doses for each of these activities for the total length of the accident. b. Justify why the remote shutdown workstation is not considered a vital area.					
				Closed - Response provided via Westinghouse letter NTD-NRC-95-4414, dated 3/13/95. Action W - Westinghouse's response to the staff's RAI on post-accident access to vital areas included a table showing personnel exposures for post-accident actions (Table 471.22-1). This table should be incorporated into Chapter 12 of the SSAR.					
1210	NRR/PERB	12.4.2-2	DSER-OI		Butler/Sejvar/Johnson, F.	Closed	Action W		
				The 5.08 cm (2 in) air gap between the fuel transfer tube shielding and the containment vessel is a possible source for radiation streaming.					
				Closed - Addressed in SSAR section 12.3.2.2.9, Revision 7.					
5011	NRR/PERB	12.	RAI-OI		Winters	Action W	Action W		
				RAI Question: In section 12.3.2.2.9 of the SSAR, Westinghouse states that the AP600 design will utilize a water-filled bladder in the 2 inch seismic gap between the end of the fuel transfer tube shielding and the steel containment wall to provide shielding in this area during fuel transfer (this bladder will accommodate relative movement between the containment and the concrete transfer tube shielding). Provide drawings showing the location of this water-filled shield bladder and provide the dose rates in the vicinity of this portion of the fuel transfer tube during fuel transfer. Discuss how personnel access to this area will be restricted during fuel transfer operations.					

2 of 3

 Total Records: 3

Westinghouse

FAX COVER SHEET

RECIPIENT INFORMATION		SENDER INFORMATION	
DATE:	3/20/97	NAME:	D. LINDGREN
TO:	D. JACKSON	LOCATION:	ENERGY CENTER - EAST
PHONE:	FACSIMILE:	PHONE:	Office: (412) 374-4856
COMPANY:	NRC	Facsimile:	win: 284-4887 outside: (412)374-4887
LOCATION:	ROCKVILLE		

Cover + Pages 1 + 1

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

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

COMMENTS:

DURING A REVIEW OF THE SSAR WE DISCOVERED A TYPO IN SSAR 3.7.4.2.1. THE LOWER LIMIT OF THE DYNAMIC RANGE OF THE TRIAXIAL ACCELEROMETERS SHOULD BE 0.001 g NOT 0.0001 g. THIS CORRECTION WILL BE INCLUDED IN SSAR REV. 12

located in a panel in the nuclear island in a room near the main control room. Seismic event data from these sensors are recorded on a solid-state digital recording system at 200 samples per second per data channel.

This solid-state recording and analysis system has internal batteries and a charger to prevent the loss of data during a power outage, and to allow data collection and analysis in a seismic event during which the power fails. Normally 120 volt alternating current power is supplied from the non-Class 1E dc and uninterruptible power supply system. The system uses triaxial acceleration sensor input signals to initiate the time-history analyzer recording and main control room alarms. The system initiation value is adjustable from 0.002g to 0.02g.

The time-history analyzer starts recording triaxial acceleration data from each of the triaxial acceleration sensors after the initiation value has been exceeded. Pre-event recording time is adjustable from 1.2 to 15.0 seconds, and will be set to record at least 3 seconds of pre-event signal. Post-event run time is adjustable from 10 to 90 seconds. A minimum of 25 minutes of continuous recording is provided. Each recording channel has an associated timing mark record with 2 marks per second, with an accuracy of about 0.02 percent.

The instrumentation components are qualified to IEEE 344-1987 (Reference 16).

The sensor installation anchors are rigid so that the vibratory transmissibility over the design spectra frequency range is essentially unity.

3.7.4.2.1 Triaxial Acceleration Sensors

Each sensor unit contains three accelerometers mounted in a mutually orthogonal array mounted with one horizontal axis parallel to the major axis assumed in the seismic analysis. The triaxial acceleration sensors have a dynamic range of 1000 to 1 (0.0001 to 1.0g) and a frequency range of 0.2 to 50 hertz.

One sensor unit will be located in the free field. Because this location is site-specific, the planned location will be determined by the Combined License applicant. The AP600 seismic monitoring system will provide for signal input from the free field sensor.

A second sensor unit is located on the nuclear island basemat in the spare battery charger room at elevation 66'-6" near column lines 9 and L.

A third sensor unit is located on the shield building structure at elevation 229' near column lines 4-1 and K.

The fourth sensor unit is located on the containment internal structure on the east wall of the east steam generator compartment just above the operating floor at elevation 138' close to column lines 6 and K.

Westinghouse

FAX COVER SHEET

RECIPIENT INFORMATION		SENDER INFORMATION	
DATE:	3/19/97	NAME:	D. LINDGREN
TO:	D. JACKSON	LOCATION:	ENERGY CENTER - EAST
PHONE:	FACSIMILE:	PHONE:	Office: (412) 374-4856
COMPANY:	NRC	Facsimile:	win: 284-4887 outside: (412)374-4887
LOCATION:	ROCKVILLE		

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:

DURING OUR REVIEW OF THE SSAR WE HAVE DISCOVERED AN INCONSISTANCY BETWEEN THE TEXT AND TABLE 3.9-8 OF THE SSAR THE TABLE WAS REVISED IN REV. 9 TO ADDRESS OPEN ITEM 850. THE ATTACHED MARK UP OF 3.9.3.1.1 WILL BE INCLUDED IN REV. 12 TO CLEAR UP THE INCONSISTANCY

DON LINDGREN

assumed to occur at the time the component would be expected to function after the failure of the nonseismic components and structures.

- Nonsafety-related systems are evaluated to confirm that their failure in an earthquake does not jeopardize plant safety.

- A water source is provided for limited fire protection after occurrence of the safe shutdown earthquake. See Section 9.5 for additional information on fire protection.

3.9.3.1.1 Seismic Loads and Combinations Including Seismic Loads

Seismic Category I systems and components, including core support structures, are designed for one occurrence of the safe shutdown earthquake which is evaluated as a Service Level D condition for pressure boundary integrity. In addition, systems and components sensitive to fatigue are evaluated for cyclic motion due to earthquakes smaller than the safe shutdown earthquake. These effects are considered by including 20 full cycles of the maximum safe shutdown earthquake stress range or five seismic events, each resulting in 63 full stress cycles with a magnitude equal to one-third of the calculated safe shutdown earthquake response for structures and components using linear elastic methods.

ASME Class 1, 2, 3 and CS systems, components and supports are analyzed for the safe shutdown earthquake with other dynamic events. See Tables 3.9-5 and 3.9-8 for load combinations.

The safe shutdown earthquake is analyzed in combination with those operating modes that occur more than 10 percent of the time. Plant conditions combined with safe shutdown earthquake include the following:

- Normal 100-percent power operation. Material properties are based on those at operating temperatures. Water inventories are based on normal operating levels. The in-containment refueling water storage tank is full, the refueling canal is empty, the spent fuel pool and transfer canal are full, and the passive containment cooling system tank is full.

- The safe shutdown earthquake, which is postulated to occur with the plant at normal 100-percent power operation, is assumed to cause nonsafety-related systems, including ac power sources, to be unavailable. A single active failure in the safety-related systems is also postulated.

- The timing and causal relationships that exist between the safe shutdown earthquake and transients such as valve discharge are considered and the events ~~combined when the safe shutdown earthquake is the cause of the transient condition.~~ For analysis of piping systems, the timing and causal relationships are not used to exclude load combinations. The safe shutdown earthquake duration is assumed to be 30 seconds. Nonseismically analyzed structures and components are assumed to be unavailable at the beginning of the safe shutdown earthquake. A single active component failure is

FAX to DINO SCALETTI

March 19, 1997

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

Don Lindgren
Don Hutchings
Ed Cummins
Bob Vijuk
Brian McIntyre

OPEN ITEM #305 (M9.4.11-4)

THIS IS AN INFORMAL RESUBMITTAL OF INFORMATION TO OBTAIN NRC ACKNOWLEDGMENT OF RECEIPT.

To meet the SECY-97-051 schedule of "Applicant Submits Final SSAR Revisions & Documentation" by 5/97, we believe that NRC must acknowledge receipt of all Westinghouse submittals by May 30, 1997. This is just 73 calendar days (business days) away.

The relevant documentation related to Open Item #305 (M9.4.11-4) is attached to today's fax on this item. We provided the information on AHU capacity in a SSAR markup on 12/20/96 (3 months ago). We then included information on this item in SSAR subsection 9.4.11.2.1 in Revision 11, February 28, 1997. We believe that this information resolves the concerns of item #305. It seems a reasonable request that NRC acknowledge receipt of the information and that NRC has a responsibility to recognize that Westinghouse, as an applicant has submitted the requested 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

Item No.	Branch	DSER Section/ Question	Type	Title/Description Detail Status	Resp Engineer	(W) Status	NRC Status	Letter No. /	Date
305	NRR/SPLB	9.4.11	MTG-OI		Winters/BRC	Closed	Action W		

M9.4.11-4 (HEALTH PHYSICS AND HOT MACHINE SHOP HVAC SYSTEM) Revise Table 9.4.11-1, Sheet 2 of 2 to state "filter requirements," not "heating coil requirements." Also, the table needs to list the correct number of HEPA filters for VHS or AP600 SSAR section and figure need to be revised, accordingly.

Closed - The health physics and hot machine shop HVAC system is not safety related. As such, SSAR subsection 9.4.3, Revision 7, provides a level of detail that does not include the exact number of filters in the system. The system does not include HEPA filters.

Action W - Include AHU capacities and filter efficiences
AHU information FAXED on 12/20/96, filter efficiency information covered by Open Item 293.

2 of 3

9.4.11.1.2 Power Generation Design Basis

The health physics and hot machine shop HVAC system provides the following functions:

- Provides conditioned air to work areas to maintain acceptable temperatures for equipment and personnel working in the areas

- Provides air movement from clean to potentially contaminated areas to minimize the spread of airborne contaminants

- Collects the vented discharges from potentially contaminated equipment in the area

- Provides for exhaust from welding booths, grinders and other miscellaneous equipment located in the hot machine shop

- Provides for radiation monitoring of exhaust air prior to release to the environment

- Maintains the access control area and hot machine shop at a slight negative pressure with respect to outdoors and the clean areas of the annex building to prevent unmonitored releases of radioactive contaminants

- Provides humidification to maintain a minimum of 35 percent relative humidity

The system maintains the following temperatures based on maximum and minimum normal outside air temperature conditions shown in Chapter 2, Table 2-1:

Room or Area	Temperatures (°F)
Health physics area	73-78
Hot machine shop	65-85

9.4.11.2 System Description

9.4.11.2.1 General Description

The health physics and hot machine shop HVAC system is a once-through ventilation system consisting of two integrated subsystems: a supply air system and an exhaust air system. The systems operate in conjunction with each other to satisfy the functional requirements of maintaining temperatures in the areas served while controlling air flow paths and area negative pressure.

The supply air system consists of two 100 percent capacity air handling units of about 14,000 scfm each with a ducted air distribution system and automatic controls. The air handling units are located in the lower south air handling equipment room on elevation 135'-3" of the annex building. The units draw 100 percent outdoor air through a louvered outdoor air intake plenum and discharge into a duct distribution system which is routed to the health physics and

FAX to DINO SCALETTI

March 19, 1997

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

Don Lindgren
Gordon Israelson
Ed Cummins
Bob Vijuk
Brian McIntyre

OPEN ITEM #205 (M9.1.1-3)

THIS IS AN INFORMAL RESUBMITTAL OF INFORMATION TO OBTAIN NRC ACKNOWLEDGMENT OF RECEIPT.

To meet the SECY-97-051 schedule of "Applicant Submits Final SSAR Revisions & Documentation" by 5/97, we believe that NRC must acknowledge receipt of all Westinghouse submittals by May 30, 1997. This is just 73 calendar days away.

The relevant documentation related to Open Item #205 (M9.1.1-3) is attached to today's fax on this item. We included information on this item in the SSAR in Revision 8, June 19, 1996 (over 10 months ago). We believe that this information resolves the concerns of item #204. It seems a reasonable request that NRC acknowledge receipt of the information and that NRC has a responsibility to recognize that Westinghouse, as an applicant has submitted the requested 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: 3/19/97

Selection: [item no] between 205 And 205 Sorted by Item #

Item No.	Branch	DSER Section/ Question	Type	Title/Description Detail Status	Resp Engineer	(W) Status	NRC Status	Letter No. /	Date
205	NRR/SPLB	9.1.1	MTG-OI	M9.1.1-3 (NEW FUEL STORAGE) As stated in SRP Section 9.1.1, Item III-2-b, the SSAR should discuss provisions in the AP600 design for draining the vault to prevent the accumulation of a fluid moderator. Closed - SSAR Rev. 3 Figure 9.3.5-1 revised to show drain from new fuel pit NRC - Action W - SSAR 9.1 needs pointer to 9.3. Closed - SSAR subsection 9.1.1.2, Revision 8, includes references to the SSAR sections covering the drain path from the new fuel storage area.	Hutchings/BPC	Closed	Action W		

2 of 3

9.1.1.2 Facilities Description

The new fuel storage facility is located within the seismic Category I auxiliary building fuel handling area. The facility is protected from the effects of natural phenomena such as earthquakes, wind, tornados, floods, and external missiles by the external walls of the auxiliary building. See Section 3.5 for additional discussion on protection from missiles. The facility is designed to maintain its structural integrity following a safe shutdown earthquake and to perform its intended function following a postulated event such as fire, internal missiles, or pipe break. The walls surrounding the fuel handling area and new fuel storage pit protect the fuel from missiles generated inside the auxiliary building. The fuel handling area does not contain a credible source of missiles. Refer to subsection 1.2.6 for a discussion of the auxiliary building. Refer to Section 3.8 for a discussion of the structural design of the new fuel storage area. Refer to subsection 3.5.1 for a discussion of missile sources and protection.

The dry, unlined, approximately 15.5-feet deep reinforced concrete pit is designed to provide support for the new fuel storage rack. The rack is supported by the pit floor and laterally supported as required at the rack top grid structure by the pit wall structures. The walls of the new fuel pit are seismic Category I. The new fuel pit is normally covered to prevent foreign objects from entering the new fuel storage rack. Since the only crane that can access the new fuel pit does have the capacity to lift heavy objects, as defined in subsection 9.1.5, the new fuel pit cover is not designed to protect the fuel assemblies from the effects of dropped heavy objects. Figures 1.2-7 through 1.2-10 show the relationship between the new fuel storage facility and other ~~features of the fuel handling area.~~

The new fuel storage pit is drained by gravity drains that are part of the radioactive waste drain system (subsection 9.3.5), draining to the waste holdup tanks which are part of the liquid radwaste system (Section 11.2). These drains preclude flooding of the pit by an accidental release of water.

Nonseismic equipment in the vicinity of the new fuel storage racks is evaluated to confirm that its failure could not result in an increase of K_{eff} beyond the maximum allowable K_{eff}. Refer to subsection 3.7.3.13 for a discussion of the nonseismic equipment evaluation.

A jib crane is used to load new fuel assemblies into the new fuel rack and transfer new fuel assemblies from the new fuel pit into the spent fuel pool. The capacity of the jib crane is limited to 2000 lbs. The new fuel pit is not accessed by the fuel handling machine or by the cask handling crane. This precludes the movement of loads greater than fuel components over stored new fuel assemblies.

During fuel handling operations, a ventilation system removes gaseous radioactivity from the atmosphere above the new fuel pit. Refer to subsection 9.4.3 for a discussion of the fuel handling area HVAC system and Section 11.5 for process radiation monitoring. Security for the new fuel assemblies is described in Section 13.6.

FAX to DINO SCALETTI

March 19, 1997

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

Don Lindgren
Gordon Israelson
Ed Cummins
Bob Vijuk
Brian McIntyre

OPEN ITEM #204 (M9.1.1-2)

THIS IS AN INFORMAL RESUBMITTAL OF INFORMATION TO OBTAIN NRC ACKNOWLEDGMENT OF RECEIPT.

To meet the SECY-97-051 schedule of "Applicant Submits Final SSAR Revisions & Documentation" by 5/97, we believe that NRC must acknowledge receipt of all Westinghouse submittals by May 30, 1997. This is just 73 calendar days away.

The relevant documentation related to Open Item #204 (M9.1.1-2) is attached to today's fax on this item. We included information on this item in the SSAR in Revision 8 (June 19, 1996) and earlier. When we submitted our explicit requests for exemptions last month, we did not include a request for exemption for either GDC 2 or 4. Internal re missiles are evaluated for the Spent Fuel Pool area (see SSAR section 3.5.1). Thei~~r~~ are no missile sources identified. We believe that this information resolves the concerns of item #204. It seems a reasonable request that NRC acknowledge receipt of the information and that NRC has a responsibility to recognize that Westinghouse, as an applicant has submitted the requested 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: 3/18/97

Selection: [item no] between 204 And 204 Sorted by Item #

Item No.	Branch	DSER Section/ Question	Type	Title/Description Detail Status	Resp Engineer	(W) Status	NRC Status	Letter No. /	Date
204	NRR/SPLB	9.1.1	MTG-OI		Lindgren,D.	Closed	Action W		

M9.1.1-2 (NEW FUEL STORAGE) The position outlined response to Q410.233 is not in compliance with general design criteria (GDC) 2 and 4. However, the staff is reviewing it's position on these issues and must complete it's review before a final determination on the acceptability of the above stated position can be reach.
This item wil be split. The resolution to address GDC 2and 4 is provided as resolution for this item. Addressing the issues in RAI 410.233 related to spent fuel cooling is a new item (M9.1.3-4).

NRC - Action W - identify as GDC exception in SSAR 1.9.

NRC - Action N - staff review of acceptability SFP design.

Closed - There are no exceptions to GDC 2 or 4 for the spent fuel pool. THe absence of missile sources near the spent fuel pool is discussed in the SSAR and the pool itself is a Seismic Category 1 structure.

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9.1.1.2 Facilities Description

The new fuel storage facility is located within the seismic Category I auxiliary building fuel handling area. The facility is protected from the effects of natural phenomena such as earthquakes, wind, tornados, floods, and external missiles by the external walls of the auxiliary building. See Section 3.5 for additional discussion on protection from missiles. The facility is designed to maintain its structural integrity following a safe shutdown earthquake and to perform its intended function following a postulated event such as fire, internal missiles, or pipe break. The walls surrounding the fuel handling area and new fuel storage pit protect the fuel from missiles generated inside the auxiliary building. The fuel handling area does not contain a credible source of missiles. Refer to subsection 1.2.6 for a discussion of the auxiliary building. Refer to Section 3.8 for a discussion of the structural design of the new fuel storage area. Refer to subsection 3.5.1 for a discussion of missile sources and protection.

The dry, unlined, approximately 15.5-feet deep reinforced concrete pit is designed to provide support for the new fuel storage rack. The rack is supported by the pit floor and laterally supported as required at the rack top grid structure by the pit wall structures. The walls of the new fuel pit are seismic Category I. The new fuel pit is normally covered to prevent foreign objects from entering the new fuel storage rack. Since the only crane that can access the new fuel pit does have the capacity to lift heavy objects, as defined in subsection 9.1.5, the new fuel pit cover is not designed to protect the fuel assemblies from the effects of dropped heavy objects. Figures 1.2-7 through 1.2-10 show the relationship between the new fuel storage facility and other features of the fuel handling area.

The new fuel storage pit is drained by gravity drains that are part of the radioactive waste drain system (subsection 9.3.5), draining to the waste holdup tanks which are part of the liquid radwaste system (Section 11.2). These drains preclude flooding of the pit by an accidental release of water.

Nonseismic equipment in the vicinity of the new fuel storage racks is evaluated to confirm that its failure could not result in an increase of K_{eff} beyond the maximum allowable K_{eff}. Refer to subsection 3.7.3.13 for a discussion of the nonseismic equipment evaluation.

A jib crane is used to load new fuel assemblies into the new fuel rack and transfer new fuel assemblies from the new fuel pit into the spent fuel pool. The capacity of the jib crane is limited to 2000 lbs. The new fuel pit is not accessed by the fuel handling machine or by the cask handling crane. This precludes the movement of loads greater than fuel components over stored new fuel assemblies.

During fuel handling operations, a ventilation system removes gaseous radioactivity from the atmosphere above the new fuel pit. Refer to subsection 9.4.3 for a discussion of the fuel handling area HVAC system and Section 11.5 for process radiation monitoring. Security for the new fuel assemblies is described in Section 13.6.

The spent fuel storage racks include storage locations for 619 fuel assemblies. The modified 10 x 7 rack module contains integral storage locations for five defective fuel storage containers as shown in Figure 9.1-4. The design of the rack is such that a fuel assembly can not be inserted into a location other than a location designed to receive an assembly. An assembly can not be inserted into a full location.

AP600 equipment, seismic and ASME Code classifications are discussed in Section 3.2. The requirements of ASME Section III, Division I, Article NF3000 are used as the criteria for evaluation of stress analyses. The materials are procured in accordance with ASME Section III, Division I, Article NF2000. Criticality analyses are performed in accordance with the requirements of ANSI N16.1-75, Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors (Reference 1); analysis codes are validated against the requirements of ANSI N16.9-75, Validation of Calculational Methods for Nuclear Criticality Safety (Reference 2); and overall requirements for fuel storage are in accordance with ANSI N210-76, Design Objectives for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Stations (Reference 3).

The spent fuel pool is designed to preclude inadvertent draining of the water from the pool.

9.1.2.2 Facilities Description

The spent fuel storage facility is designed to the guidelines of ANS 57.2 (Reference 4). The spent fuel storage facility is located within the seismic Category I auxiliary building fuel handling area. The walls of the spent fuel pool are an integral part of the seismic Category I auxiliary building structure. The facility is protected from the effects of natural phenomena such as earthquakes, wind, tornados, floods, and external missiles.

The facility is designed to maintain its structural integrity following a safe shutdown earthquake and to perform its intended function following a postulated event such as a fire. Refer to subsection 1.2.6 for further discussions of the auxiliary building fuel handling area.

Nonseismic equipment in the vicinity of the spent fuel storage racks is evaluated to confirm that its failure could not result in an increase of K_{eff} beyond the maximum allowable K_{eff}. Refer to subsection 3.7.3.13 for a discussion of the nonseismic equipment evaluation.

The spent fuel pool provides storage space for spent fuel. The pool is approximately 41 feet deep, constructed of reinforced concrete, and lined with a stainless steel plate. The normal water volume of the pool is about 176,000 gallons of borated water (including racks without fuel at a water level 2 foot 6 inches below the operating deck) with a nominal boron concentration of 2500 ppm. Figures 1.2-7 through 1.2-10 show the spent fuel pool and other features of the fuel handling area.

The connections for the drain and makeup lines are located to preclude the draining of the spent fuel pool due to a break in a line or failure of a pump to stop. The connection for the spent fuel cooling pumps' suction is located below normal water level and above the level needed to provide sufficient water for shielding and for cooling of the fuel if the spent fuel

3.5 Missile Protection

General Design Criterion 4 of Appendix A to 10 CFR 50 requires that structures systems and components important to safety be protected from the effects of missiles. The AP600 criteria for protection from postulated missiles provide the capability to safely shut down the reactor and maintain it in a safe shutdown condition. The AP600 criteria also protect the integrity of the reactor coolant system pressure boundary and maintain offsite radiological dose/concentration levels within the limits defined in 10 CFR 100.

Missiles may be generated by pressurized components, rotating machinery, and explosions within the plant and by tornadoes or transportation accidents external to the plant. Potential missile hazards are eliminated to the extent practical by minimizing the potential sources of missiles through proper selection of equipment, and by arrangement of structures and equipment in a manner to minimize the potential for damage from missiles. Potential missiles due to failures of nonseismic items are addressed in Subsection 3.7.3.13. Heavy load-drop evaluations are described in Subsection 9.1.5.

The following are definitions for missile protection terminology:

Internally Generated Missile - A mass that may be accelerated by energy sources continuously present on site.

Single Active Failure - Malfunction or loss of a component of electrical or fluid systems. The failure of an active component of a fluid system is considered to be a loss of component function as a result of mechanical, hydraulic, pneumatic, or electrical malfunction, but not the loss of component structural integrity.

High-Energy System - Fluid systems that, during normal plant conditions, are operated or maintained pressurized with a maximum operating temperature greater than 200°F and/or a maximum operating pressure greater than 275 psig, as discussed in Subsection 3.6.1.

The following criteria are applied in the identification of missiles and the protection requirements that must be satisfied:

- A missile must not damage structures, systems, or components to the extent that could prevent achieving or maintaining safe shutdown of the plant or result in a significant release of radioactivity.

- A single active component failure is assumed in systems used to mitigate the consequences of the postulated missile and achieve a safe shutdown condition. The single active component failure is assumed to occur in addition to the postulated missile and any direct consequences of the missile. When the postulated missile is generated in one of two or more redundant trains of a dual-purpose safety-related fluid system, which is designed to seismic Category I standards and is capable of being powered from both

The AP600 passive design minimizes the number of safety-related structures, systems, and components required for safe shutdown. Systems required for safe shutdown are identified in Chapter 7. Safety class structures, systems and components, their location, seismic category, and quality group classifications are given in Section 3.2. General arrangement drawings showing locations of the structures, systems, and components are given in Section 1.2. The areas required for safe shutdown, and the major systems and components housed therein that are required to be protected from internally and externally generated missiles for safe shutdown, are summarized below:

- The containment vessel, including the reactor coolant loop, and passive core cooling system inside containment

- The shield building, including the passive containment cooling system

- Containment penetration areas, including containment isolation valves and Class IE cables

- The control complex including the main control room, reactor protection system, batteries, and dc switchgear

- The spent fuel pit

The AP600 relies on safety-related systems and equipment to establish and maintain safe shutdown conditions. There are no systems or components identified as important in the evaluation of regulatory treatment of nonsafety systems (RTNSS) that require protection from missiles. WCAP 13856 (Reference 1) describes the implementation of the RTNSS process on the AP600 including mission statements and recommended regulatory oversight for structures, systems, and components identified as important by the RTNSS process.

Evaluations are performed to demonstrate that the criteria are satisfied in the event a credible missile is produced coincident with a single active component failure. These evaluations include the following:

- For those potential missiles considered to be credible, a realistic assessment is made of the postulated missile size and energy, and its potential trajectories.

- Potentially impacted components associated with systems required to achieve and maintain safe shutdown are identified.

- Loss of these potentially impacted components coincident with an assumed single active component failure is evaluated to determine if sufficient redundancy remains to achieve and maintain a safe shutdown condition. If these criteria are satisfied, no further protection is required for the identified missile. If these conditions are not satisfied, additional protective features are incorporated (for example, plant layout is modified, or barriers are added).

- When valves and other eccentric masses are not considered rigid, the dynamic models are simulated by the lumped masses in discrete locations (that is, center of gravity of valve body and valve operator), coupled by elastic members with properties of the eccentric components.

3.7.3.12 Seismic Category I Buried Piping Systems and Tunnels

There are no seismic Category I buried piping systems and tunnels in the AP600 design.

3.7.3.13 Interaction of Other Systems with Seismic Category I Systems

The safety functions of seismic Category I structures, systems, and components are protected from interaction with nonseismic structures, systems, and components; or their interaction is evaluated. The safety-related systems and components required for safe shutdown are described in Section 7.4. This equipment is located in selected areas of the auxiliary building and inside containment. The primary means of protecting safety-related structures, systems, and components from adverse seismic interactions are discussed in the following paragraphs in the order of preference.

- Separation - separation with the use of physical barriers
- Segregation - routing away from location of seismic Category I systems, structures, and components
- Impact Evaluation - contact with seismic Category I systems, structures, and components may occur, and there is insufficient energy in the impact to cause loss of safety function
- Support as seismic Category II

Interaction of connected systems with seismic Category I piping is considered by including the other piping in the analysis of the seismic Category I system. Interaction of piping systems that are adjacent to Category I structures, systems, and components is also considered. This is discussed in subsection 3.7.3.13.4.

The containment and each room outside containment containing safety-related systems or equipment, as identified in Table 3.7.3-1, are reviewed for potential adverse seismic interactions to demonstrate that systems, structures, and components are not prevented from performing their required safe shutdown functions. In addition, the review identifies the protection features required to mitigate the consequences of seismic interaction in an area that contains safety-related equipment.

The evaluation steps to address seismic interaction taken for each room or building area containing seismic Category I systems, structures, and components are:

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COMMENTS:

Joe / Bill -

Attached are markup page changes to AP600 SSAR subsections 6.3.2.1.2, 6.3.2.2.7.7, and 6.3.2.2.7.9. We are adding a statement to each subsection about valve diversity. This is being done to address PRA insights and providing the place for the disposition within the SSAR. These changes will be included in SSAR Rev. 12 unless we hear differently from you.

Cindy

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Although gas accumulation is not expected, there is a vertical pipe stub on the top of the inlet piping high point that serves as a gas collection chamber. Level detectors indicate when gases have collected in this area. There are provisions to allow the operators to open manual valves to locally vent these gases to the in-containment refueling water storage tank.

The passive residual heat removal heat exchanger, in conjunction with the passive containment cooling system, can provide core cooling for an indefinite period of time. After the in-containment refueling water storage tank water reaches its saturation temperature (in about 2 hours), the process of steaming to the containment initiates.

Condensation occurs on the steel containment vessel, which is cooled by the passive containment cooling system. The condensate is collected in a nonsafety-related gutter arrangement located at the operating deck level which returns the condensate to the in-containment refueling water storage tank. The gutter normally drains to the containment sump, but when the passive residual heat removal heat exchanger actuates, a nonsafety-related isolation valve in each of the two gutter drain lines shut and the gutter overflow returns directly to the in-containment refueling water storage tank.

Recovery of the condensate maintains the passive residual heat removal heat exchanger heat sink for a very long time. Without recovery of the condensate, the in-containment refueling water storage tank inventory is sufficient to provide passive residual heat removal heat exchanger operation for 72 hours.

The passive residual heat removal heat exchanger is also used to maintain a safe shutdown condition. It removes decay heat and sensible heat from the reactor coolant system to the in-containment refueling water storage tank, the containment atmosphere, the containment vessel, and finally to the ultimate heat sink – the atmosphere outside of containment. This occurs after in-containment refueling water storage tank saturation is reached and steaming to containment initiates.

6.3.2.1.2 Reactor Coolant System Emergency Makeup and Boration

The core makeup tanks provide reactor coolant system makeup and boration during events not involving loss of coolant when the normal makeup system is unavailable or insufficient. There are two core makeup tanks located inside the containment at an elevation slightly above the reactor coolant loops. During normal operation, the core makeup tanks are completely full of cold, borated water. The boration capability of these tanks provides adequate core shutdown margin following a steam line break.

The core makeup tanks are connected to the reactor coolant system through a discharge injection line and an inlet pressure balance line connected to a cold leg. The discharge line is blocked by two normally closed, parallel air-operated isolation valves that open on a loss of air pressure or electrical power, or on control signal actuation. Insert (A) here.

- The gravity injection line flow paths from the in-containment refueling water storage tank

- The containment recirculation lines that connect to the gravity injection lines

The check valves selected for these applications incorporate a simple swing-check design with a stainless steel body and hardened valve seats. The passive core cooling system check valves are safety-related, designed with their operating parts contained within the body, and with a low pressure drop across each valve. The valve internals are exposed to low temperature reactor coolant or borated refueling water.

During normal plant operation, these check valves are closed, with essentially no differential pressure across them. Confidence in the check valve operability is provided by operation at no differential pressure clean/cold fluid environment, the simple valve design, and the specified seat materials.

The check valves normally remain closed, except for testing or when called upon to open following an event to initiate passive core cooling system operation. The valves are not subject to the degradation from flow operation or impact loads caused by sudden flow reversal and seating, and they do not experience significant wear of the moving parts.

These check valves are periodically tested during shutdown conditions to demonstrate valve operation. These check valves are equipped with nonintrusive position sensors to indicate when the valves are open or closed.

In current plants, there are many applications of simple swing-check valves that have similar operating conditions to those in the passive core cooling system. The extensive operational history and experience derived from similar check valves used in the safety injection systems of current pressurized water reactors indicate that the design is reliable. Check valve failure to open and common mode failures have not been significant problems.

6.3.2.2.7.7 Accumulator Check Valves

The accumulator check valve design is similar to the accumulator check valves in current pressurized water reactor applications. It is also similar to the low differential pressure opening check valve design described in subsection 6.3.2.2.7.6. Insert B here.

During normal operation, the check valves are in the closed position with a nominal differential pressure across the disc of about 1550 psid. The valves remain in this position, except for testing or when called upon to open following an event. They are not subject to the degradation from flow operation or impact loads caused by sudden flow reversal and seating. They do not experience significant wear of the moving parts and they are expected to function with minimal backleakage.

In the incontainment refueling water storage tank injection lines, the squib valves are in series with normally closed check valves. In the containment recirculation lines, the squib valves are in series with normally closed check valves in two lines and with normally closed motor operated valves in the other two lines. As a result, inadvertent opening of these squib valves will not result in loss of reactor coolant or in draining of the incontainment refueling water storage tank.

The type of squib valve used in these applications provides zero leakage in both directions. It also allows flow in both directions. A valve open position sensor is provided for these valves. Insert C here

Squib valves are also used to isolate the fourth stage automatic depressurization system lines. These squib valves are in series with normally open motor operated gate valves. Redundant-series controllers are provided to prevent spuriously opening of these squib valves. The type of squib valve used in this application provides zero leakage of reactor coolant out of the reactor coolant system. The reactor coolant pressure acts to open the valve. A valve open position sensor is provided for these valves.

6.3.2.3 Applicable Codes and Classifications

Sections 5.2 and 3.2 list the equipment ASME Code and seismic classification for the passive core cooling system. Most of the piping and components of the passive core cooling system within containment are AP600 Equipment Class A, B, or C and are designed to meet seismic Category I requirements. Some system piping and components that do not perform safety-related functions are nonsafety-related.

The requirements for the control, actuation, and Class 1E devices are presented in Chapters 7 and 8.

6.3.2.4 Material Specifications and Compatibility

Materials used for engineered safety feature components are given in Section 6.1. Materials for passive core cooling system components are selected to the meet the applicable material requirements of the codes in Section 5.2, as well as the following additional requirements:

- Parts of components in contact with borated water are fabricated of, or clad with, austenitic stainless steel or an equivalent corrosion-resistant material.
- Internal parts of components in contact with containment emergency sump solution during recirculation are fabricated of austenitic stainless steel or an equivalent corrosion resistant material.
- Valve seating surfaces are hard-faced to prevent failure and to reduce wear.

INSERT A

The core makeup tank discharge isolation valves are diverse from the passive residual heat removal heat exchanger outlet isolation valves because they use different globe valve body styles and different air operator types.

INSERT B

The accumulator check valves are diverse from the core makeup tank valves because they use different check valve types.

INSERT C

The IRWST injection squib valves are diverse from the containment recirculation squib valves because they are designed to different design pressures.