CLINCH RIVER

BREEDER REACTOR PROJECT

PRELIMINARY SAFETY ANALYSIS REPORT

VOLUME 16

PROJECT MANAGEMENT CORPORATION

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Amendment 48

List of Responses to NRC Questions

Reference: NRC Letter Dated December 1, 1976

NRC
Ques. No.

020.49

Reference: NRC Letter Dated March 30, 1977

NRC
Ques. No.

001.700
001.701

Amendment 49

List of Responses to NRC Questions

Reference: NRC Letter Dated August 17, 197

Reference: NRC Letter Dated August 17, 1976

NRC Ques. No.
310.44

Amendment 50

List of Responses to NRC Questions

Reference: NRC Letter Dated August 17, 1976

NRC Ques. No.
011.23

Amendment 51

List of Responses to NRC Questions

There are no new NRC questions in Amendment 51.

Amendment 53

<u>List of Responses to NRC Questions</u>

There are no new or updated Responses to Questions in this Amendment.

AMENDMENT 55

LIST OF RESPONSES TO NRC QUESTIONS

There are no new NRC Questions in Amendment 55.

PAGE REPLACEMENT GUIDE FOR

AMENDMENT 55

CLINCH RIVER BREEDER REACTOR PLANT

PRELIMINARY SAFETY ANALYSIS REPORT

(DOCKET NO. 50-537)

Transmitted herein is Amendment 55 to the Clinch River Breeder Reactor Plant Preliminary Safety Analysis Report, Docket 50-537. Amendment 55 consists of new and replacement pages for the PSAR text.

Vertical lines on the right hand side of the page are used to identify question response information and lines on the left hand side are used to identify new or changed design information.

The following attached sheets list Amendment 55 pages and instructions for their incorporation into the Preliminary Safety Analysis Report.

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AMENDMENT 56

LIST OF RESPONSES TO NRC QUESTIONS

There are no new NRC Questions in Amendment 56.

AMENDMENT 56

QUESTION/RESPONSE SUPPLEMENT

This Question/Response Supplement contains an Amendment 56 tab sheet to be inserted following the Q-i (Amendment 55, June 1980) page. Page Q-i (Amendment 56, August 1980) is to follow the Amendment 56 tab.

There are no new or updated Question/Response pages included in this Amendment.

Amendmend 57

LIST OF RESPONSES TO NRC QUESTIONS

There are no new NRC Questions in Amendment 57.

Amendment 57

Question/Response Supplement

This Question/Response Supplement contains an Amendment 57 tab sheet to be inserted following Q-i (Amendment 56, Aug. 1980) page. Page Q-i (Amendment 57, Nov. 1980) is to follow the Amendment 57 tab.

Replacement pages for the Question/Response Supplement are listed below.

Replacement Pages

Remove These Pages	Insert These Pages
Q7-1	Q7-1

Question 7

Provide an overview of the methods used to evaluate the structural integrity of the fuel assembly including a description of all analytical methods used (e.g., PECT2) and all applicable data. The report should be in the form of a summary addressing all calculational limits (e.g., stress and deflection).

Response:

The information requested concerning the CRBRP Fuel Assembly was provided under separate cover in the following topical report:

> "CRBRP Fuel Assembly Structural Analysis in Support of the Final Design Review", CRBRP-ARD-0204

Additional information concerning the CRBRP Fuel Rod will be provided in a topical report at a later date. A Table of Contents for this report
57 was provided to the NRC in December, 1976.

00837

Department of Energy
Clinch River Breeder Reactor
Plant Project Office
P.O. Box U
Oak Ridge, Tennessee 37830

Docket No. 50-537

November 7, 1980

Mr. Darrell G. Eisenhut, Director
Division of Project Management
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Mr. Eisenhut:

AMENDMENT NO. 57 TO THE PRELIMINARY SAFETY ANALYSIS REPORT FOR CLINCH RIVER BREEDER REACTOR PLANT

The application for a Construction Permit and Class 104(b) Operating License for the Clinch River Breeder Reactor Plant, docketed April 10, 1975, in NRC Docket No. 50-537, is hereby amended by the submission of Amendment No. 57 to the Preliminary Safety Analysis Report pursuant to 50.34(a) of 10 CFR Part 50. This Amendment No. 57 includes: updates to Section 7, "Instrumentation and Controls"; Section 11.6, "Offsite Radiological Monitoring Program"; Section 15.A, "CRBRP Radiological Source Term for Assessment of Site Suitability"; and other updates and revisions.

A Certificate of Service, confirming service of Amendment No. 57 to the PSAR upon designated local public officials and representatives of the EPA, will be filed with your office after service has been made. Three signed originals of this letter and 97 copies of this amendment, each with a copy of the submittal letter, are hereby submitted.

Sincerely,

Raymond L. Copeland

Raymond L. Copeland
Acting Assistant Director
for Public Safety

PS:80:332

Enclosure

cc: Service List
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SUBSCRIBED and SWORN to before me
this 30th day of October, 1980.

Inez K. Minor

Notary Public

My Commission Expires April 28, 1984

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10/30/80

PAGE REPLACEMENT GUIDE FOR

AMENDMENT 57

CLINCH RIVER BREEDER REACTOR PLANT

PRELIMINARY SAFETY ANALYSIS REPORT

(DOCKET NO. 50-537)

Transmitted herein is Amendment 57 to the Clinch River Breeder Reactor Plant Preliminary Safety Analysis Report, Docket 50-537. Amendment 57 consists of new and replacement pages for the PSAR text.

Vertical lines on the right hand side of the page are used to identify question response information and lines on the left hand side are used to identify new or changed design information.

The following attached sheets list Amendment 57 pages and instructions for their incorporation into the Preliminary Safety Analysis Report.

AMENDMENT 59

LIST OF RESPONSES TO NRC QUESTIONS

There are no new NRC Questions in Amendment 59.

00806

Department of Energy
Clinch River Breeder Reactor
Plant Project Office
P.O. Box U
Oak Ridge, Tennessee 37830
Docket No. 50-537

February 13, 1981

Mr. Darrell G. Eisenhut, Director
Division of Project Management
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Mr. Eisenhut:

AMENDMENT NO. 60 TO THE PRELIMINARY SAFETY ANALYSIS REPORT FOR CLINCH RIVER BREEDER REACTOR PLANT

References:

(1) Letter, PS:79:041, R. L. Copeland to R. S. Boyd, "Topical Report [CRBRP-3, Volume 1] on Structural Margin Beyond the Design Base (SMBDB)," dated February 9, 1979.

(2) Letter, PS:80:092, R. L. Copeland to D. B. Vassallo, "Topical Report [CRBRP-3, Volume 2] on Thermal Margin Beyond the Design Base (TMBDB)," dated March 25, 1980.

(3) Letter, PS:78:317, R. L. Copeland to R. S. Boyd, "Topical Reports 'Structural Response to CRBRP Scale Models to a Simulated Hypothetical Core Disruptive Accident' (WARD-D-0218) and 'Closure Head Capability for Structural Margin Beyond Design Base Loading' (WARD-D-0178)," dated December 13, 1978.

(4) Letter, PS:79:047, R. L. Copeland to R. S. Boyd, "Topical Report on HCDA's," dated February 16, 1979.

The application for a Construction Permit and Class 104(b) Operating License for the Clinch River Breeder Reactor Plant, docketed April 10, 1975, in NRC Docket No. 50-537, is hereby amended by the submission of Amendment No. 60 to the Preliminary Safety Analysis Report pursuant to 50.34(a) of 10 CFR Part 50.

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The CRBRP Project has done extensive work in the area of Hypothetical Core Disruptive Accidents (HCDA) and has documented this effort in a single comprehensive report, CRBRP-3, Volumes 1 and 2, which was provided to the NRC in References (1) and (2). Additional documents that serve as significant references for this work were also provided to the NRC by References (3) and (4). It is now appropriate to withdraw Appendix F, "Core Disruptive Accident Accomodation," from the PSAR, since all of the relevant material relating to HCDA's is provided in CRBRP-3 and its associated references.

Accordingly, this Amendment No. 60 withdraws Appendix F from the PSAR and includes: responses to NRC's requests for additional information contained in a letter dated March 30, 1977, and revised responses to previously answered NRC questions concerning structural margins associated with an HCDA.

The CRBRP Project is confident that the above documentation is responsive to NRC Requests for Additional Information regarding HCDA energetics, associated structural margins, and degraded core considerations, and that this information will lead to resolution of outstanding questions regarding these subjects.

A Certificate of Service, confirming service of Amendment No. 60 to the PSAR upon designated local public officials and representatives of the EPA, will be filed with your office after service has been made. Three signed originals of this letter and 97 copies of this amendment, each with a copy of the submittal letter, are hereby submitted.

Sincerely,

Raymond L. Copeland

Raymond L. Copeland
Acting Assistant Director
for Public Safety

PS:81:056

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SUBSCRIBED and SWORN to before me
this 6th day of February, 1981.

Ruby L. Williams

Notary Public

My Commission Expires April 28, 1984

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PAGE REPLACEMENT GUIDE FOR

AMENDMENT 60

CLINCH RIVER BREEDER REACTOR PLANT

PRELIMINARY SAFETY ANALYSIS REPORT

(DOCKET NO. 50-537)

Transmitted herein is Amendment 60 to Clinch River Breeder Reactor Plant Preliminary Safety Analysis Report, Docket 50-537. Amendment 60 consists of new and replacement pages for the PSAR text and Question/Response supplement pages.

Vertical lines on the right hand side of the page are used to identify question response information and lines on the left hand side are used to identify new or changed design information.

The following attached sheets list Amendment 60 pages and instructions for their incorporation into the Preliminary Safety Analysis Report.

8102180227

AMENDMENT 60

List of Responses to NRC Questions

REFERENCE: NRC Letter Dated March 30, 1977

NRC QUESTION NUMBER

Q001.615	Q001.639	Q001.663
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Q001.619	Q001.643	Q001.667
Q001.620	Q001.644	Q001.668
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AMENDMENT 62

LIST OF RESPONSES TO NRC QUESTIONS

There are no new NRC Questions in Amendment 62.

AMENDMENT 63

LIST OF RESPONSES TO NRC QUESTIONS

There are no new NRC Questions in Amendment 63.

AMENDMENT 64

LIST OF RESPONSES TO NRC QUESTIONS

There are no new NRC Questions in Amendment 64.

AMENDMENT 65

LIST OF RESPONSES TO NRC QUESTIONS

There are no new NRC Questions in Amendment 65.

AMENDMENT 66

LIST OF RESPONSES TO NRC QUESTIONS

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AMENDMENT 67

LIST OF RESPONSES TO NRC QUESTIONS

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Q760.3-1
Q760.4-1
Q760.5-1

AMENDMENT 69

QUESTION/RESPONSE SUPPLEMENT

This Question/Response Supplement contains an Amendment 69 tab sheet to be inserted following Qi page Amendment 68, May 1982. Page Qi Amendment 69 is to be inserted following the Amendment 69 tab sheet.

Following new Question/Response pages will be inserted in PSAR Volume 25 behind the appropriate numbered tabs located within that volume. Replacement numbered tabs 270, 281, 410, 430 and 451 are also provided.

The parenthesis beside each Question/Response shown indicates the number of pages associated with each Question/Response.

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AMENDMENT 69

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QCS421.5 (1)
QCS421.7 (3)
QCS421.8 (3)
QCS421.9 (1)
QCS421.10 (1)
QCS421.11 (1)
QCS421.12 (1)
QCS421.13 (1)
QCS421.14 (3)
QCS421.15 (1)
QCS421.16 (2)
QCS421.17 (1)
QCS421.18 (1)
QCS421.19 (2)
QCS421.20 (1)
QCS421.23 (1)
QCS490.1 (6)
QCS490.2 (1)
QCS490.3 (1)
QCS490.4 (2)
QCS490.5 (2)
QCS490.6 (1)
QCS490.7 (1)
QCS490.8 (2)
QCS490.9 (2)
QCS490.10 (4)
QCS490.11 (14)
QCS490.12 (1)
QCS490.13 (5)
QCS490.14 (1)
QCS490.15 (1)
QCS490.16 (1)
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QCS490.18 (3)
QCS490.19 (2)
QCS490.20 (1)
QCS490.21 (1)
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QCS490.23 (3)
QCS490.24 (3)
QCS490.25 (2)
QCS490.26 (1)
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QCS490.35 (5)
QCS490.36 (1)
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QCS491.11 (2)
QCS491.12 (2)
QCS491.13 (1)
QCS491.14 (2)
QCS491.15 (1)
QCS491.16 (1)
QCS491.17 (2)
QCS491.18 (2)
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QCS810.1 (1)
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QCS810.11 (1)
QCS810.12 (1)
QCS810.13 (1)
QCS810.14 (1)
QCS810.15 (1)
QCS810.16 (1)

AMENDMENT 70

List of Responses to NRC Questions
Received Since the Fall of 1981 and
Located Chronologically in Volume 25

QCS220.1
QCS220.2
QCS220.3
QCS220.4
QCS220.5
QCS220.6
QCS220.7
QCS220.8
QCS220.9
QCS220.10
QCS220.11
QCS220.12
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QCS220.31
QCS220.32
QCS220.33
QCS220.34
QCS220.35
QCS220.36
QCS220.37
QCS220.38
QCS220.39
QCS220.40
QCS220.41
QCS220.42
QCS220.43
QCS220.44
QCS220.45

AMENDMENT 71

List of Responses to NRC Questions
Received Since the Fall of 1981 and
Located Chronologically in Volume 25 and 26

QCS231.1
QCS421.6
QCS421.21
QCS421.24
QCS421.25
QCS421.26
QCS421.28
QCS421.29
QCS421.32
QCS421.33
QCS421.35
QCS421.36
QCS421.38
QCS421.39
QCS421.40
QCS421.41
QCS421.43
QCS421.44
QCS421.45
QCS421.46
QCS421.49
QCS421.50
QCS421.51
QCS421.52
QCS421.53
QCS421.54
QCS421.55
QCS421.56
QCS421.57
QCS421.59
QCS491.5
QCS760.6
QCS760.7
QCS760.8
QCS760.9
QCS760.10
QCS760.11
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QCS760.162
QCS760.163
QCS760.164
QCS760.165
QCS760.167
QCS760.168
QCS760.169
QCS760.170
QCS760.171
QCS760.173
QCS760.174
QCS760.179

AMENDMENT 71

QUESTION/RESPONSE SUPPLEMENT

This Question/Response Supplement contains an Amendment 71 tab sheet to be inserted following Qi page Amendment 70, August 1982. Page Qi Amendment 71 is to be inserted following the Amendment 71 tab sheet.

This Amendment 71 provides both OLD and NEW Question/Response pages which are to be inserted in numerical order behind the appropriate numbered tabs in the PSAR. The parenthesis beside each Question/Response shown indicates the number of pages associated with each Question/Response.

The following Question/Response pages are revised pages to OLD QUESTIONS and should be inserted in the appropriate PSAR Volumes other than Volumes 25 and 26.

REPLACEMENT PAGES

REMOVE THESE PAGES	INSERT THESE PAGES
Q222.76-1, 2	Q222.76-1, 2
Q241.83-1	Q241.83-1

With the issue of this PSAR Amendment 71, an additional PSAR Binder (Volume 26) is being provided. In order to accommodate the volume of Question/Response pages being issued with this amendment and the expected volume of Question/Response pages to be issued with future amendments, a shifting of pages from PSAR Volume 25 to Volume 26 is suggested. This could be accomplished as follows:

a) Remove from PSAR Volume 25 numbered tabs 490, 491, 760, 810 and all pages behind each of these tabs and insert them into the new PSAR Volume 26.

b) Question/Response pages provided with this Amendment for NRC Questions Received Since the Fall of 1981 will be inserted behind the appropriate numbered tabs in numerical sequence either in PSAR Volume 25 or 26 as appropriate. Additionally, new numbered tab 721 provided with this Amendment will be inserted following page QCS491.22-1.

The following Question/Response pages are to be inserted in numerical order in the PSAR Volumes 25 or 26 as appropriate.

*QCS231.1	(1)	QCS760.23	(1)	QCS760.74	(1)	QCS760.125	(2)
QCS421.6	(1)	QCS760.24	(6)	QCS760.75	(1)	QCS760.126	(2)
QCS421.21	(1)	QCS760.25	(1)	QCS760.76	(1)	QCS760.127	(1)
QCS421.24	(1)	QCS760.26	(1)	QCS760.77	(2)	QCS760.128	(1)
QCS421.25	(1)	QCS760.27	(3)	QCS760.78	(1)	QCS760.129	(1)
QCS421.26	(1)	QCS760.29	(7)	QCS760.79	(3)	QCS760.130	(2)
QCS421.28	(1)	QCS760.31	(2)	QCS760.80	(1)	QCS760.132	(1)
QCS421.29	(1)	QCS760.32	(1)	QCS760.81	(1)	QCS760.133	(1)
QCS421.32	(1)	QCS760.33	(1)	QCS760.82	(1)	QCS760.134	(1)
QCS421.33	(1)	QCS760.34	(2)	QCS760.83	(2)	QCS760.135	(1)
QCS421.35	(1)	QCS760.35	(1)	QCS760.84	(1)	QCS760.136	(1)
QCS421.36	(1)	QCS760.37	(1)	QCS760.85	(1)	QCS760.137	(4)
QCS421.38	(1)	QCS760.38	(1)	QCS760.86	(3)	QCS760.138	(1)
QCS421.39	(1)	QCS760.39	(4)	QCS760.87	(1)	QCS760.139	(1)
QCS421.40	(1)	QCS760.40	(2)	QCS760.88	(1)	QCS760.140	(1)
QCS421.41	(1)	QCS760.41	(1)	QCS760.89	(1)	QCS760.141	(2)
QCS421.43	(1)	QCS760.42	(1)	QCS760.90	(2)	QCS760.142	(1)
QCS421.44	(1)	QCS760.43	(2)	QCS760.91	(1)	QCS760.143	(1)
QCS421.45	(1)	QCS760.44	(1)	QCS760.92	(2)	QCS760.144	(1)
QCS421.46	(4)	QCS760.45	(1)	QCS760.93	(1)	QCS760.145	(1)
QCS421.49	(1)	QCS760.46	(1)	QCS760.94	(1)	QCS760.146	(1)
QCS421.50	(1)	QCS760.47	(1)	QCS760.95	(1)	QCS760.147	(1)
QCS421.51	(2)	QCS760.48	(1)	QCS760.96	(2)	QCS760.148	(1)
QCS421.52	(1)	QCS760.49	(1)	QCS760.97	(1)	QCS760.149	(1)
QCS421.53	(1)	QCS760.50	(1)	QCS760.98	(1)	QCS760.150	(1)
QCS421.54	(1)	QCS760.51	(1)	QCS760.99	(3)	QCS760.151	(1)
QCS421.55	(2)	QCS760.52	(1)	QCS760.100	(1)	QCS760.152	(1)
QCS421.56	(1)	QCS760.53	(1)	QCS760.101	(1)	QCS760.153	(1)
QCS421.57	(2)	QCS760.54	(1)	QCS760.102	(2)	QCS760.154	(1)
QCS421.59	(9)	QCS760.55	(1)	QCS760.103	(1)	QCS760.155	(1)
*QCS491.5	(1)	QCS760.56	(1)	QCS760.104	(1)	QCS760.156	(1)
QCS760.6	(54)	QCS760.57	(3)	QCS760.106	(2)	QCS760.157	(2)
QCS760.7	(1)	QCS760.58	(1)	QCS760.107	(1)	QCS760.158	(1)
QCS760.8	(1)	QCS760.59	(1)	QCS760.108	(1)	QCS760.159	(1)
QCS760.9	(1)	QCS760.60	(1)	QCS760.109	(1)	QCS760.160	(2)
QCS760.10	(99)	QCS760.61	(1)	QCS760.111	(1)	QCS760.161	(1)
QCS760.11	(1)	QCS760.62	(1)	QCS760.112	(1)	QCS760.162	(1)
QCS760.12	(1)	QCS760.63	(1)	QCS760.113	(1)	QCS760.163	(1)
QCS760.13	(1)	QCS760.64	(1)	QCS760.114	(1)	QCS760.164	(1)
QCS760.14	(2)	QCS760.65	(1)	QCS760.115	(1)	QCS760.165	(1)
QCS760.15	(1)	QCS760.66	(1)	QCS760.117	(2)	QCS760.167	(2)
QCS760.16	(1)	QCS760.67	(1)	QCS760.118	(1)	QCS760.168	(1)
QCS760.17	(1)	QCS760.68	(1)	QCS760.119	(1)	QCS760.169	(1)
QCS760.18	(1)	QCS760.69	(1)	QCS760.120	(1)	QCS760.170	(1)
QCS760.19	(1)	QCS760.70	(1)	QCS760.121	(2)	QCS760.171	(2)
QCS760.20	(1)	QCS760.71	(1)	QCS760.122	(1)	QCS760.173	(1)
QCS760.21	(1)	QCS760.72	(1)	QCS760.123	(1)	QCS760.174	(2)
QCS760.22	(1)	QCS760.73	(1)	QCS760.124	(1)	QCS760.179	(2)

*These are replacement pages.

AMENDMENT 72

List of Responses to NRC Questions
Received Since the Fall of 1981 and
Located Chronologically in PSAR Volumes 25 and 26

QCS220.25
QCS421.9
QCS421.17
QCS421.22
QCS421.27
QCS421.30
QCS421.31
QCS421.34
QCS421.36
QCS421.37
QCS421.42
QCS421.47
QCS421.48
QCS421.58
QCS721.1

QCS760.13
QCS760.28
QCS760.30
QCS760.36
QCS760.105
QCS760.110
QCS760.116
QCS760.131
QCS760.166
QCS760.172
QCS760.175
QCS760.176
QCS760.177
QCS760.178

AMENDMENT 73

LIST OF RESPONSES TO NRC QUESTIONS

There are no new NRC Questions in Amendment 73.

AMENDMENT 74

LIST OF RESPONSES TO NRC QUESTIONS

RECEIVED SINCE THE FALL OF 1981

QCS421.26
QCS421.45
QCS430.1
QCS430.2
QCS430.3
QCS430.5
QCS430.6
QCS430.7
QCS430.8
QCS430.9
QCS430.10
QCS430.11
QCS430.12
QCS430.13
QCS430.14
QCS430.15
QCS430.16
QCS430.17
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QCS430.30
QCS430.31
QCS430.32
QCS430.33
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QCS430.100
QCS430.101
QCS430.102
QCS430.103
QCS430.104
QCS760.144

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Department of Energy
Washington, D.C. 20545
Docket No. 50-537
HQ:S:83:197

February 8, 1983

Mr. Paul S. Check, Director
CRBR Program Office
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Check:

AMENDMENT 75 TO THE PRELIMINARY SAFETY ANALYSIS REPORT (PSAR) FOR CLINCH RIVER BREEDER REACTOR PLANT (CRBRP)

The application for a Construction Permit and Class 104(b) Operating License for the CRBRP, docketed April 10, 1975, in NRC Docket No. 50-537, is hereby amended by the submission of Amendment 75 to the PSAR pursuant to 50.34(a) of 10 CFR, Part 50.

This Amendment 75 includes: Revisions to Section 1.4, "Identification of Project Participants;" Section 5.0, "Heat Transport and Connected Systems;" Section 5.7, "Overall Heat Transport System Evaluation;" Chapter 7, "Instrumentation and Controls;" Chapter 9, "Auxiliary Systems;" Section 17D, "A Description of the Westinghouse Quality Assurance Program;" and other updates and revisions.

A Certificate of Service, confirming service of Amendment 75 to the PSAR upon designated local public officials and representatives of the Environmental Protection Agency, will be filed with your office after service has been made. Three signed originals of this letter and 97 copies of this amendment, each with a copy of the submittal letter, are hereby submitted.

Sincerely,

John R. Longenecker

John R. Longenecker
Acting Director, Office of
Breeder Demonstration Projects
Office of Nuclear Energy

exact copy of original

SUBSCRIBED AND SWORN to before me
this 26 day of January 1983

Enclosure

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2/17/82

PAGE REPLACEMENT GUIDE FOR

AMENDMENT 75

CLINCH RIVER BREEDER REACTOR PLANT

PRELIMINARY SAFETY ANALYSIS REPORT

(DOCKET NO. 50-537)

Transmitted herein is Amendment 75 to Clinch River Breeder Reactor Plant Preliminary Safety Analysis Report, Docket 50-537. Amendment 75 consists of new and replacement pages for the PSAR text and Responses to NRC Questions.

Vertical margin lines on the right hand side of the page are used to identify changes resulting from NRC Questions and margin lines on the left hand side are used to identify new or changed design information.

The following attached sheets list Amendment 75 pages and instructions for their incorporation into the Preliminary Safety Analysis Report.

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PDR ADOCK 05000537
B PDR

AMENDMENT 75

LIST OF RESPONSES TO NRC QUESTIONS

There are no new NRC Questions in Amendment 75.

817

Department of Energy
Washington, D.C. 20545

March 2, 1983

Docket No. 50-537
HQ:S:83:223

Dr. J. Nelson Grace, Director
CRBR Program Office
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Dr. Grace:

AMENDMENT 76 TO THE PRELIMINARY SAFETY ANALYSIS REPORT (PSAR) FOR CLINCH RIVER BREEDER REACTOR PLANT (CRBRP)

The application for a Construction Permit and Class 104(b) Operating License for the CRBRP, docketed April 10, 1975, in NRC Docket No. 50-537, is hereby amended by the submission of Amendment 76 to the PSAR pursuant to 50.34(a) of 10 CFR, Part 50.

This Amendment 76 includes: Revisions to Section 3.1, "Conformance with General Design Criteria;" Section 4.2, "Reactor Mechanical Design;" Chapter 7, "Instrumentation and Controls;" Chapter 9, "Auxiliary Systems;" Appendix A, "Computer Codes;" and other updates and revisions.

A Certificate of Service, confirming service of Amendment 76 to the PSAR upon designated local public officials and representatives of the Environmental Protection Agency, will be filed with your office after service has been made. Three signed originals of this letter and 97 copies of this amendment, each with a copy of the submittal letter, are hereby submitted.

Sincerely,

John R. Longenecker

John R. Longenecker
Acting Director, Office of
Breeder Demonstration Projects
Office of Nuclear Energy

Enclosure

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AMENDMENT 76

LIST OF RESPONSE TO NRC QUESTIONS

REFERENCE NRC LETTER DATED DECEMBER 1, 1976

NRC QUESTION NO.
110.78

AMENDMENT 77

LIST OF RESPONSES TO NRC QUESTIONS

There are no new NRC Questions in Amendment 77.

Question 001.1 (1.1.2.2)

You have established the goal that "The probability of exceeding 10 CFR 100 guidelines shall be less than one chance in one million per reactor year." Later in setting up reliability allocations (Section C.1 of Appendix C) you become quite rigorous in apportioning the goal of 10^{-6}/yr by subdividing it into:

Element	Goal
Shutdown System Failure	10^{-7}
Shutdown Heat Removal Failure	8×10^{-7}
Other Faults Leading to Loss of Coolable Geometry	10^{-7}

The assignment of 80% of the goal to a single element, shutdown heat removal, raises several questions.

a. Can one make a meaningful distinction between 8×10^{-7} and 1×10^{-6}?
b. Should the bulk of the goal be allocated to a single failure mechanism?
c. Is the design approach for shutdown heat removal such that there is practical limit to the goal that can be attained? For example, does the need for AC power to transfer heat to the steam system constitute such a limitation?
d. Two of the elements in this tripartite allocation are singular elements, the shutdown system failure and the shutdown heat removal failure. The third element is a sum of all others. Does this imply that the individual elements in this sum are allocated a probability substantially less than the 10^{-7}/yr allocated to the sum? For example, would you allocate a probability of the order of 10^{-8}/yr to an event in the sum such as a serious airplane crash?

Present your responses to these questions.

Response:

The information requested is in revised Section C.1.3.3. 25

Question 001.2

Is Appendix D considered part of the Reference Design Submittal?

Response:

With the deletion of Appendix D in Amendment 24 this question is no longer applicable.

Question 001.3 (Table 1.1-1)

Table 1.1-3 is referred to, this table is missing.

Response:

Table 1.1-3, List of Extremely Unlikely Faults Used as Design Bases, was included in the 100 copies of the PSAR transmitted to NRC by PMC letter TS-75-242 dated June 13, 1975.

Question 001.4 (Table 1.1-2)

Discuss the bases for the assigned levels and ranges given for the occurrence probabilities of the various events and accidents.

Response:

A revised Table 1.1-2 is provided. Engineering judgement was used to group the various design basis events and predict the numerical values associated with the groups previously indicated in the Table.

Question 001.5 (1.2-38)

Provide elevation views of the Control Building and the Diesel Generator Building.

Response

These elevation views are provided on new Figure 1.2-44, Control Building Section A-A and Section B-B, and new Figure 1.2-46, Diesel Generator Building, Section A-A and Section B-B.

Question 6

Provide a description of pertinent experimental data, analytical models and plans for obtaining future data regarding fuel rod-wire wrap interactions.

Response:

This information requested has been provided under separate cover.

Question 001.7 (3.1.2)

Provide a complete list of all components or parts of components which comprise the reactor coolant boundary.

Response:

The list of Components or Parts of Components which comprise the Reactor Coolant Boundary can be found in new Table 3.1-1. 25

Question 001.8 (3.1.2)

For Systems whose failure could not result in fuel design limits being exceeded, justify the proposed identification of nozzles as the outermost part of a component or its associated piping which comprises a part of the reactor coolant boundary.

Response:

The justification for identifying the nozzles connecting the system to the primary heat transport system as being the reactor coolant boundary for those systems whose failure could not result in the fuel design limits is as follows:

The only systems which fall into this category are the Argon cover gas pressurization and equalization systems and the primary sodium makeup and overflow systems. These low pressure systems are connected to the primary cooling system and are located in inerted inner cells. Failure of these systems would only result in minor consequences, such as limited sodium spill effects or release of reactor cover gas.

On the basis described above, it is therefore justifiable that, for these particular small branch lines, the connecting nozzle be considered as the end of the reactor coolant boundary.

Question 001.9 (3.1.2)

Provide a complete list of all components or parts of components which comprise the intermediate coolant boundary.

Response:

Components which comprise the intermediate coolant boundary are listed in new Table 3.1-2. 25

Question 9

Provide a discussion of the methods used for the analysis of fuel bundle response to a seismic event including the forcing functions and pertinent bundle properties. Applicable experimental data should be presented and discussed; derivations of all equations should be given or appropriately referenced.

Response:

The information requested has been provided under separate cover in the topical report, WARD-D-0158, "Seismic Evaluation Methods and Criteria for CRBRP Fuel Assembly Duct Structure."

Question 001.10 (3.1.2)

Identify any systems, components or parts of components connected to the intermediate coolant system, not penetrating reactor containment, whose failure would impair the capability of the intermediate coolant system to perform its safety function.

Response:

Revised Section 3.1.2 identifies esstential portions of the intermediate coolant boundary.

Question 001.11 (3.1.2)

For systems and components whose failure could not impair the capability of the intermediate coolant system to perform its safety function, justify the proposed identification of nozzles as the outermost part of a component or its associated piping which comprises part of the intermediate coolant boundary.

Response:

The justification for identifying the nozzles connecting the system to the intermediate heat transport system as being the intermediate coolant boundary for those systems whose failure could not impair the capability of the intermediate coolant system to perform its safety function is as follows:

The only systems which fall into this category are the cover gas pressurization and equalization system and the sodium fill and drain system. Failure of these low pressure systems would only result in minor consequences such as loss of cover gas and minor sodium spills. While such spills could lead to sodium fires, the safety function of the IHTS (viz: to remove heat) is not impared. The consequences of a large sodium spill are presented in Chapter 15 (Section 15.6.1.5) wherein it is indicated that even for a large spill there would be no propagation of the event and thus no loss of safety function. On the basis described above, it is therefore justifiable that for these particular small branch lines, the connecting nozzle may be considered as the end of the intermediate coolant boundary.

Question 001.12 (3.1.2)

Confirm that anticipated operational occurrences in the definition of Fuel Design Limits means all those events identified as Anticipated Faults in the definition of off-normal conditions.

Response:

In the definition of Fuel Design Limits, "anticipated operational occurrences" was meant to imply that those events identified as Off-Normal Conditions, namely: both Anticipated Faults and Unlikely Faults Faults (see response to Question 001.13 and also Table 15.1.2-3 of the PSAR).

In conjunction with this response, the definition of Fuel Design Limits in Section 3.1.2 is revised to read:

> "Fuel Design Limits. Fuel design limits means those limits such as temperature, burnup, fluence, and cladding strain which are specified by the designer for normal operation and off-normal conditions."

Question 001.13 (3.1.2)

The text of this section should compare your definitions of Normal Operation, Off-Normal Conditions, and Extremely Unlikely Faults to the spectrum of operational conditions and events used in 10 CFR 50 Appendix A.

Response:

This comparison can be found in new Section 3.1.2.1 and new Table 3.1-3. 25

Question 001.14 (3.1.3.3)

The response to Criterion 23 does not indicate the means by which components are protected from fallout of sodium vapor or reaction products. Provide a description of the means to do so.

Response:

The information requested is incorporated in the changed PSAR page 3.1-21.

Question 001.15 (3.1.3.4)

Justify the omission of a mandatory leak identification requirement in Criterion 32.

Response:

For CRBRP, detection and confirmation of a leak in the primary system leads to shutdown of the plant regardless of the precise location therefore, location of the leak is an economic consideration and a mandatory requirement in the Criterion is not warranted.

Question 001.16 (3.1.3.4)

Justify the omission of a mandatory leak identification requirement in Criterion 36.

Response:

The CRBRP Criterion 36 is the same criterion, verbatim, as set forth in the "Interim General Design Criteria for the Clinch River Breeder Reactor Nuclear Power Plant," issued by the Commission on July 19, 1974. The criterion number in the Commission's Interim GDC is Criterion 34.

Through the discussions during a series of pre-application meetings with the Commission, the applicant agreed to use the Commission's Criterion verbatim as specified.

Question 001.17 (3.1.3.5)

Confirm that Criterion 52 applies to all structures where a specified leak rate must be demonstrated.

Response:

The information requested is in revised Section 3.1.3.5. 25

Question 001.18 (3.1.3.5)

Justify the basis for the statement that "the design of the IHTS lines meet the requirements of GDC 57".

Response:

The system boundary of the IHTS main lines is classified as ASME-III Code Class 2. It will be protected against accidents, extreme environmental conditions and natural phenomena by a Seismic Category I structure. Therefore, the design of these lines meets the requirements of GDC 57 set forth in Section 3.1.3 of this PSAR.

Question 001.19 (3.1.3.6)

Provide the general criteria to be applied to assure that systems which handle or contain irradiated fuel shall be designed, fabricated, installed, and tested to the highest quality standards practical.

Response:

The systems which handle or contain irradiated fuel assemblies will conform specifically to the following CRBRP General Design Criteria: 61, 62 and 63 which are essentially identical to the related criteria in 10 CFR 50. Revised Section 3.1.3.6 includes this information.

In addition, GDC 1, 2, 3, 3a and 4 will also apply, as appropriate, to these systems.

Quality standards for these systems are determined in accordance with the procedures provided in Section 3.2, which is consistent with the basic intent of Regulatory Guide 1.26. 25

Question 001.20 (3A.1.1)

The inner cell system is intended to mitigate the consequences of sodium spills, to preclude detrimental sodium/material reaction, and to maintain structural integrity following any of the design basis accidents. Explain your claim then that "the inner cell system of the CRBRP is not an Engineered Safety Feature". In particular, explain what design or quality differences in the system derive from your choice and exclude it from the list of Engineered Safety Features. | 27

Response:

The inner cell system is designed to have a nominal leak tightness for the purpose of maintaining the purity of the inerted atmosphere in the inner cells during normal plant operation. The safety analyses, which show compliance with the 10 CFR 100 radiological guidelines for all the postulated sodium spill accidents have not taken credit for any leak tightness of the in-containment cells.

The analysis of sodium spills assumes, however, that the cell liners prevent sodium from contacting the concrete. This assumption does establish a functional requirement for cell liners. It is expected that, even if the liner does fail and sodium does contact with concrete, the reaction is self-limiting. Cell liners are being designed to remain intact for the life of the plant for both normal operating and accident conditions. Fabrication and inspection requirements will insure that the as-built liner is installed as specified in the design.

Development programs are planned to provide additional assurance that the cell liner will remain intact during accident conditions. Tests are also planned to show that even if there is large local liner failure under accident conditions the sodium concrete reaction will be self limiting. A more detailed discussion of cell liner design criteria and development programs is contained in the response to Q130.37.

Since the liners are a passive component of the inner cell system and are not required to serve as a containment barrier against the accidental release of radioactivity, liners are not considered Engineering Safety Features. It will be shown that the inner cell system will function as required in the event of a sodium spill. No credit will be taken in the safety analyses for design features that cannot be proven to be functional by test and/or analysis.

The single exception is the RAPS surge and delay tank cell, located in the Reactor Service Building. This cell does not contain sodium or NaK. This cell will be considered as an Engineered Safety Feature due to leak tightness requirements. Design bases for this cell are as described in Section 3A.1.2 of the PSAR.

Question 001.21 (3A.1.2)

This section does not provide the information requested in Section 6.3.1.1 of the Standard Format. Revise it to provide this information. In particular, note that none of the five items presented in this section are responsive to the five items of 6.3.1.1 in the Standard Format.

Response:

For the Reference Design, as described in the CRBRP PSAR, the only inner cell specifically designed to act as a containment barrier against the accidental release of radioactivity is the RAPS Surge and Delay Tank Cell located in the Reactor Service Building. No credit for retention, holdup, plate-out, or settling of accidentally released radioactive material is taken for any other inner cell in the Reference Design evaluations.

The five information items requested in Section 6.3.1.1 of the Standard Format, for the RAPS Surge and Relay Tank Cell, are provided on new pages 3A.1-2a and 3A.1-2b.

Question 001.22 (3A.1.3)

Table 3A.1-3 does not provide a sufficient description of the inert cell system. Provide a design description of the system. Refer to Section 6.3.1.2 of the Standard Format for the scope and nature of the design information required. In particular, present simple sketches of the cells showing the principal equipment in each cell and the extent of cell sealing required for piping and electrical penetrations and access openings. Include a summary of your design calculations which established the cell heat loads listed in Table 3A.1-1.

Response:

The information requested is provided in revised Section 3A.1.3 and Tables 3A.1-1 and 3A.1-3.

Question 001.23 (3A.1.3)

Describe the design criteria to be used to establish the location of the transition from "hot" cell liner design to "cold" cell liner design. Describe how you will account for spray and splash effects and in-cell heat transfer.

Response:

The current cell liner design concept is that of a fixed liner which will be used in all areas of lined cells (ie there will be no differentiation made between "hot" and "cold" liners). Therefore, there will be no transition point between "hot" cell liner design and "cold" cell liner design. PSAR Section 3.8.3.1 and Appendix E Supplement Section 15.6.1.6.2 have been revised accordingly.

Sodium spray and splash effects and in-cell heat transfer are discussed in the responses to questions 130.38 and 130.39. 24

Question 001.24 (3A.1.4)

This section does not provide a sufficient design evaluation. Revise this section to include the information requested in Section 6.3.1.3 of the Standard Format. Include in the revised section more detailed discussion of system response following a sodium spill.

Response:

As stated in Section 3A.1.1 of the PSAR, the inner cells within the containment of the CRBRP are designed and required only to have a nominal leaktightness for the purpose of maintaining the purity of the inerted atmosphere in these cells during normal plant operation. Note that the inerted cells are operated at a slight negative pressure to provide the desired normal operating leakage path. For the radiological safety analyses of all the related accidents such as sodium spills, no credit is taken for any leaktightness or pressure control of these cells. It is assumed that all airborne reaction products are released directly to the containment.

The analyses used to set structural design requirements are based on the maximum pressure associated with the events used in the design basis for the cells assuming no leakage.

Because of this selection of functional design, the inner cells of the CRBRP are not Engineered Safety Features. Consequently, the information presented in 3A.1.4 provides the bounding design evaluations for these cell designs for both radiological and structural evaluations recognizing that the cells are not engineered safety features.

Analysis with respect to the consequences of sodium spills is provided in Section 15.6 of the PSAR.

Further details relating to the question have been provided in the response to Q.001.25, Amendment 1 to the PSAR.

Question 001.25 (3A.1.5)

Justify your conclusion that periodic testing of the cells is not required. Discuss your basis for continuing confidence in cell liner integrity throughout the life of the plant. Include in your discussion the predictability of cell-to-cell leakage as well as cell-to-upper containment leakage.

Response:

The information requested is discussed in revised Section 3A.1.5. 25

Question 001.26 (3A.1.7)

Revise this section to provide a complete identification of the cell materials, especially those that would be exposed to spilled or spray sodium at high temperature.

Response:

Section 3A.1.7 has been revised in response to this question. 31

Question 001.27

Provide at least the preliminary values for the TBD items in Table 3A.1-2.

Response:

The preliminary values for the TBD items have been included in the amended PSAR Table 3A.1-2.

Question 001.28 (3A.2.2)

Provide an analysis of the safety implications of loss of heat removal for the Head Access Area.

Response:

The requested analysis is in revised Section 3A.2.2. 25

Question 001.29 (4.2.3.3.1.5)

(Section 4.2.3.3.1.5) As of 5/6/75, the only reference for B_4C Control Material Data, i.e., HEDL-TME 75-19, had not been cleared for release by HEDL. Provide the referenced document if it has not been made available in the interim. (Ref. 44)

Response:

The referenced (Ref. 44) document, HEDL-TME 75-19, should currently be available in NRC libraries. However, to assure a timely review of Section 4.2.3.3.1.5, the required document is being provided to NRC under separate cover.

Question 001.30 (4.3.2.1)

Provide the secondary control rod clad OD and thickness even if such values are preliminary and subject to change. The referenced table indicated that these are under development.

Response:

Table 4.3-1 has been revised to provide the information requested.

Question 001.31 (4.3.2.1.5)

Justify your proposed use of the Source Range Flux Monitoring System. In particular, provide the experimental and analytical basis for its neutron-flux monitoring capability when the reactor is shutdown and fuel is stored in the in-vessel storage.

Response:

The response to this question has been incorporated into the PSAR as revised section 4.3.2.1.5.

Question 001.32 (4.3.2.3.1)

Justify your omitting consideration of any plutonium (fissile) Doppler effect, as it affects the magnitude of the M/O Doppler reactivity and its temperature dependence. Provide a more detailed description of the experimental validation of the CRBRP Doppler Coefficients. Are the SEFOR experiments your only data base?

Response:

The response to this question has been incorporated in the first and sixth paragraph of Section 4.3.2.3.1 and additional data over and above the SEFOR data are also provided.

Question 001.33 (4.3.2.7.1)

You comment that the major discrepancy between the ZPR criticals and the CRBRP is in the use of plates in a square lattice instead of cylindrical pins in an hexagonal array. Another discrepancy is that the criticals operate at room temperature. Provide a justification for not considering this effect of temperature when comparing ZPR reactivity measurements and corresponding CRBRP analysis.

Response:

The response to this question has been incorporated into Section 4.3.2.7.1.

Question 001.34 (4.3.3.5)

It is not clear how Figure 4.3-40 provides information on the ZPPR-3 experimental program. Provide clarifying information.

Response:

The response to this question is contained in revised Section 4.3.3.5 and an amended version of Figure 4.3-40.

Question 001.35 (15.1.2)

The design limits for fuel pin integrity shown on page 15.1-75 are "preliminary" and are referred to as conservative estimates which "..could change or perhaps even the method of evaluating performance could change." Provide further justification for the preliminary nature of these limits in light of their importance in assessing the margin of safety provided by the transient performance capability of the fuel rod and the PPS.

Response:

The response to this question has been incorporated into the revised section 15.1.2.

Question 001.36 (15.1.2.1)

Justify the use of results from unirradiated tests (Table 15.1.2-4) in demonstrating the failure predictive capability of the CDF procedure.

Response:

The response to this question has been incorporated into the PSAR by the addition of revised section titled, "Experimental Verification of the CDF Procedure," to follow a new section titled, "Discussion or Mechanical Properties Used in the CDF Procedure." (See new PSAR Page 15.1-63).

Question 001.38 (15.2)

Provide a mechanistic description of fuel-pin transient behavior for the several cases considered. In particular, describe the primary and secondary loading on the fuel pin cladding over the full range of burnups and power ratings.

Response:

The requested description is provided in revised Section 15.1.2.1. 25

Question 001.39 (D5.1.4.1)

Provide documentation which supports the statement: "Knowledgeable project physics groups have currently placed a ±50% uncertainty on . . . sodium void worths." (Pg D5-6, 7)

Response:

This question requests clarification of information which is no longer a part of the current documentation. The Project has since consolidated all considerations given Hypothetical Core Disruptive Accidents into report CRBRP-3 (References 10a and 10b, PSAR Section 1.6) and its associated references; consequently, PSAR Appendices D and F have been withdrawn in Amendments 24 and 60 respectively. A discussion of sodium void worth uncertainty is provided in PSAR Section 4.3.2.3.2 and Section 6 and 7 of Reference 15, PSAR Section 1.6. 60

Question 001.40 (4.2.1.1.2.1)

Provide substantiating data and analysis which you made use of in your "...review of FFTF fuel assembly design evaluations and EBR-II reactor operating experience and information from the US and foreign LMFBR programs..." (P4.2-3) where these "loadings form the minimum basis for conservative evaluation of the design adequacy of the fuel and radial blanket assembly fuel rods". (P4.2-4)

Response:

The response to this question is provided in revised PSAR Section 4.2.1.1.2.1 and References 59 thru 69.

Question 001.41 (4.2.1.3.1.3)

Provide data or reference to the data that is used "...in the formulation of analytic models to describe fuel rod performance up to the point of cladding failure...".

Response:

The response to this question is provided in references 53 through 58 in Section 4.2. Section 4.2.1.3.1.3 has been modified to make specific reference to these.

Question 001.42 (A.82)

Explain the following apparent discrepancy. The only reference for the accident analysis code "SAS-2B" is a report describing a SAS-2B subroutine "SAS FCI".

Response:

The SAS-2B accident analysis code is a developmental version of the SAS-3A code and is virtually identical in content. The proper reference for this code is now available as:

> F. E. Dunn, "The SAS3A LMFBR Accident Analysis Computer Code," ANL/RAS 75-17, April 1975. (Availability: USERDA Technical Information Center).

Question 001.43 (A.49)

Explain the following apparent discrepancy. The only reference for the "LIFE" code is a report describing LIFE-II, not LIFE-III which we understand is currently being used.

Response:

The requested information on LIFE-III code has been incorporated in Revised Section A.49. 25

Question 001.44 (4.4.2.5)

Provide hydraulic loads for both normal and accident conditions (as opposed to the pressure drop given, only for normal conditions).

Response:

See revised Section 4.4.2.5 and revised Figure 4.4-2.

Question 001.45 (4.4.2.6.5)

Provide the cladding to fuel gap characteristics as a function of burnup (at rated power, design overpower and during transient).

Response

The response to this question, addressing gap characteristics at rated power, design overpower and during transients is given in revised Section 4.4.2.6.5 9

Question 001.46 (4.4.2.6)

Discuss the uncertainties associated with estimating the peak or limiting conditions for thermal and hydraulic analysis.

Response:

The response to this question is contained in revised section 4.4.3.3.

Question 001.47 (4.4.2.8)

Provide the information requested in Section 4.4.2.8 of the Standard Format.

Response:

See revised Section 4.4.2.8, "Plant Configuration Data".

Question 001.48 (4.4.3.3)

Evaluate the thermal response of the core for design overpower and for expected transient conditions.

Response:

The thermal response of the core for design overpower is treated in revised Section 4.4.3.3.1, and the transient conditions are covered in revised Section 4.4.3.7.

Question 001.49 (4.4.3.4)

Provide a comprehensive discussion of the analytical techniques used in evaluating the core thermal-hydraulics including estimates of uncertainties.

Response:

See revised Section 4.4.3.4, Analytical Techniques and Figures 4.4-38 and 4.4-39.

Question 001.50 (4.4.4)

Provide a discussion of any plans to conduct full-scale measurements and testing techniques on the reactor core.

Response:

PSAR Section 4.4.4 was amended in response to Question 001.298 to provide further information on tests planned for verification of reactor characteristics. Pre-operational tests will also be performed. Details of these tests will be provided in the FSAR, in Chapter 14. The schedule for performance of these tests is shown on Figure 14.1-1 of the PSAR.

Question 001.51 (4.4.5)

Provide a discussion of the instrumentation to be provided to conform predicted power densities and other design features developed under Feature Model Test Program (Section 4.4.4.1).

Response:

The CRBRP instrumentation provided for design verification consists of thermocouples located at the core assemblies exit. Further discussion of the instrumentation may be found in Revised Section 4.4.5. 25

Question 001.52 (5.1)

Provide an overall system heat balance including a breakdown which shows core power, pump heat input, and losses to ambient from piping and components.

Response:

The requested information is included in revised Section 5.0 of the PSAR, including new Figure 5.1-1A.

Question 001.53 (5.1)

Provide a preliminary design description and thermal analysis of the thermal insulation to be used on the sodium components and piping.

Response:

The information requested has been added. See revised section 5.3.2.3.6 and Figures 5.3-16A thru D.

Question 001.54 (5.1)

Provide a preliminary design description of the trace heating system to be used on sodium components and piping.

Response:

A preliminary design description is provided in PSAR Section 9.4.

Question 001.55 (5.1.2)

Section 5.1 has complied with the Standard Format in a general way. It lacks depth. Provide orientation drawings such as plans and elevations of the heat transport system piping. The single line flow schematics and hydraulic profiles are insufficient to provide a comprehensive understanding of the system.

Response:

The information requested is contained in revised Section 5.1 of the PSAR, including new Figures 5.1-1B and 5.1-1C. Further, plans and elevations of the Reactor Containment Building and Steam Generator Building showing heat transport system components and piping are provided in Section 1.2 of the PSAR.

Question 001.56 (5.1.2)

It is stated that piping is sized to limit sodium velocities to 30 ft/sec. and comply with a pump head limitation of 450 ft. Provide criteria used in selecting these design parameters. Where will sodium velocities of 30 ft/sec. exist in the heat transfer system? What erosion rate is used as a function of temperature, velocity and oxide concentration?

Response:

The requested criteria for these design parameters is provided in revised Section 5.2.3.2.4. 25

Question 001.57 (5.1.2)

It is stated that welded pipe is being used in the hot and cold leg piping; presumably this includes fittings (elbows). Provide the experience basis for this selection in liquid metal systems. What criteria are used for stress conditions in which seamless pipe is used? Will additional in-service inspection requirements be imposed on welded pipe and fittings?

Response:

The experience basis requested is provided in revised Section 5.3.2.3.4. 25

Question 001.58 (5.1.2)

What is the justification for providing only visual inservice inspection of the reactor vessel and nozzles? How would internal cracks be detected if they were to occur?

Response:

Visual inspection of the exterior surface of all sodium-retaining boundaries is the only type of in-service inspection which is practical and within the state-of-the-art today. Visual inspection capability is being provided throughout the primary system of CRBRP via television cameras, periscopes, and/or contact means. However, the need for additional assurance of viable hardware is recognized. For this purpose, multiple, redundant, leak detection systems (Section 7.5.5 of the PSAR) are provided to detect very small leaks, if any should develop, and permit plant shutdown before any significant loss of sodium can occur. Additionally, a considerable amount of development is underway in the industry to provide viable volumetric examination techniques for stainless steel welds and base metal under conditions similar to those in CRBRP.

If any of these systems are sufficiently developed and compatible with the space provided around each surface in the plant likely to require inspection, it will be included. However, it is believed that visual inspection combined with the redundant, sensitive leak detection equipment provides assurance of safe operation.

Question 001.59 (5.1.2)

In Section 5.1.2 the description of standpipe bubbler system is not clear, and it is only referenced in Figure 5.1-2. Provide a description of this system and show how it could not introduce gas into the reactor cooling system through a malfunction in venting. Include in your discussion the interaction of venting to the top of the IHX to the pump tank during rapid flow changes such as scram.

Response:

Revised Section 5.3.2.3.1 provides a description of the standpipe bubbler system. 25

Question 001.60 (5.1.2)

On page 5.1-4 deinerting of individual cells is discussed for independent access. Please describe your anticipated operational mode such as 2 loop operation and access to a down loop if planned on, including means of maintaining an effective barrier between the down loop cell and other structures, earliest access time permitted, etc. Please describe how leakage of a check valve is accommodated in a down loop and how the down loop would be refilled and restarted.

Response:

New Section 5.3.4.5 has been added to include a discussion of the maintenance of a PHTS loop. 25

Question 001.61 (5.1.2)

On page 5.1-5 it is stated that natural circulation will provide sufficient heat transport for safe decay heat removal. Discuss how this will be demonstrated prior to full power operation.

Response:

Revised Section 5.3.3.2 discusses safe decay heat removal via natural circulation. 25

Question 001.62 (5.1.3)

Describe how sodium flow is balanced between evaporator A and B particularly since it is stated that the loop piping differs in length and configuration.

Response:

The loop piping within the steam generator cells is identical (mirror image) for the two evaporators as explained in revised Section 5.1.3.

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Question 001.63 (5.1.4)

Since the sodium side of the steam generators are vented to the expansion tank, which is on the suction side of the IHTS pump, full Δ P would exist across the vent system. Describe the dynamics of fluid movement following scram and any impact it may have on the initiation of natural circulation.

Response:

Revised Section 5.5.3.2 provides a discussion of the dynamics of fluid movement following scram. 25

Question 001.64 (5.2.1.1)

Section 5.2.1.1 discusses codes and standards. No reference is made to Regulatory Guide 1.87, Construction Criteria for Class 1 Components in Elevated Temperature Reactors, supplement to ASME Section III Code Cases 1592, 1593, 1594, 1595 and 1596. Provide the use of, or proposed exceptions to Regulatory Guide 1.87.

Response:

Section 5.2.1 discusses codes and standards applied to the design and manufacture of the reactor vessel, closure head, and guard vessel through the respective equipment specifications. Relevant portions of the Regulatory Guides are applied to equipment design and manufacture through interpretation into the equipment specifications. As such, Regulatory Guides are not imposed directly on the equipment subcontractors, and hence, are not listed in Table 5.2-1, "Summary of Code, Code Cases, and RDT Standards Applicable to Design and Manufacture of Reactor Vessel, Closure Head, and Guard Vessel." With regard to Regulatory Guide 1.87, the discussion is provided in revised Section 5.2.1.1. 25

Question 001.65 (5.2.1.1)

It is not clear how it will be demonstrated that the quantity of accident products reaching the lower head will be insufficient to cause melt-through. Explain the basis for this statement.

Response:

This statement was made based on the expectation that the core support structure and inlet modules would be designed to enhance post accident cooling capability to the extent that the accident products from major core damage could be cooled within the reactor vessel.

Design features including an extended inlet module design were evaluated to determine if they would be effective in enhancing the post accident cooling capability. These evaluations showed that practical design features would not substantially augment the third level thermal margins, due to uncertainties related to postulated meltfront uniformity, boil-up, and the determination of the capability of a thin stainless steel container to contain debris. Consequently, the sentence referred to in Section 5.2.1.1 is no longer considered accurate, and is being deleted. The Project is now engaged in an assessment of the thermal capbility of the plant for downward debris retention.

Question 001.66 (5.2.1.1)

Section 5.2.1.1 makes the following claim "where adherence to such codes or standards does not suffice to assure a quality product in keeping with the safety function, they shall be supplemented or modified as necessary". What criterion is used to make this judgement?

Response:

Where generally recognized codes and standards do not cover conditions that will be imposed on the equipment, they will be supplemented as needed. Examples are the effects of sodium and cover gas on the structural properties of the materials used, and special requirements for protection of the equipment from contamination during fabrication, shipment, and storage. Other examples of equipment requirements not covered by generally recognized codes and standards are surveillance and in-service inspection for liquid metal plants, lifting and handling, methods of selecting materials, surface finish, spare parts, packaging, handling, shipping, operating manuals, design documentation, and guidelines for elevated temperature design.

Section 5.2.1.1 has been revised to reflect the Project position on components not covered by codes and standards.

Question 001.67 (5.2.1.1)

Describe the material surveillance program and the provisions for in-service inspection associated with reactor vessel and closure head.

Response:

The material surveillance program is described in revised Section 5.2.4.5. 25

Question 001.68 (5.2.1.1)

What is meant by the statement that "In all cases the expected or hypothesized condition shall not be more severe than the selected design criteria and transients".

Response:

The statement in question means that conditions imposed on the equipment in actual operation will be less severe than the design conditions which will be used in the thermal and structural analyses; these designs will be shown to satisfy selected criteria for the respective conditions.

Question 001.69 (5.2.1.1)

In the general discussion on stress analysis page 5.2-1 the method of combining loads is not discussed. Is it your plan to generate a histogram and separate stress analysis for segments of the system which can be characterized in a similar manner? Describe and discuss your approach to the resolution of this problem.

Response:

Section 5.2.1.1 of the PSAR provides for an explanation of the methods of combining loads through the following statement:

"The reactor vessel, closure head, and guard vessel are designed and manufactured in accordance with the ASME Boiler and Pressure Vessel Code, Section III, Division 1, Nuclear Power Plant Components, 1974 Edition, with all addenda up to and including applicable ASME Code Cases for elevated temperature components and certain RDT Standards are also imposed. See Table 5.2-1."

The method of combining loads is directed by the acceptance criteria of the design sections of the ASME Code. In the design evaluation, the combining and ordering of the duty cycle events within the loading histograms to be used is first checked to verify the proposed histogram will produce a conservative cumulative damage assessment. (See Appendix B for discussion of development of umbrella transients.)

In addition, histograms for the Reactor Vessel, Guard Vessel, and Closure Head which define the normal, upset and emergency operating history for those components for their design of time from initial start-up are provided in the equipment specifications.

Question 001.70 (5.2.1.1)

What is the basis of picking 1250°F and 300 hours as a post-accident heat removal condition and will this become an ASME design requirement as a faulted condition?

Response

The thermal requirement on the vessel, nozzles and core support structure provides a capability for higher than normal temperatures that may result if natural circulation is the only means to remove decay heat following an event resulting in core damage. The specific temperature and time included in the requirement are not derived from a specific core damage condition. These third level thermal requirements were specified prior to having completed natural circulation analyses for an extensively damaged CRBR core. Guidance was obtained from such analyses for FFTF as reported in Reference Q001.70-1. From those analyses it was calculated that maximum outlet temperatures cloud approach 1200°F if the core inlet was completely blocked. The 1250°F temperature was chosen for CRBR to allow for differences in response from FFTF. Although the analyses predicted that temperatures would decrease at a substantial rate (∿100°F in eight hours), the requirement conservatively assumed that the high temperatures continued for 300 hours (nearly 2 weeks). Since creep rupture is the physical phenomenon of interest and the vessel, nozzles and supports are made of stainless steel which possesses a significant creep rupture strength, there does exist a capability for accommodating higher temperatures at longer times than 300 hours.

Subsequent to the specification of these third level thermal requirements, scoping analyses were made to determine the outlet plenum temperature histories resulting from a range of impedances associated with a damaged CRBR core. The model used in these assessments was an extension of that provided in Reference Q001.70-1. These scoping analyses did not treat in-vessel temperature distributions nor the dynamics of the Intermediate Heat Transport System. A pump coastdown, system pressure losses, a representative power burst and subsequent decay heat were included in the overall gross heat balance model. Thermal mixing in the outlet plenum was assumed and the thermal inertia of one-half of the structural steel in the reactor upper plenum was assumed to be effective in determining the plenum temperature. The pressure drop associated with the reduction in vessel flow area was accounted for by a flow impedance multiplier, normalized to the flow impedance associated with the normal design configuration. For conservatism, the thermal center in the vessel was assumed to be at the elevation of the horizontal baffle and the thermal center in the IHX was assumed to be at its normal elevation. This results in a conservative thermal center height difference of only 8 feet. The resulting outlet temperatures are shown in Figure Q001.70-1. It is noted that the outlet temperature does not exceed 1250°F unless the flow impedance is more than about 35 times the normal impedance. Furthermore, even if the impedance is 75 times the normal

impedance, the time at which the temperature is above 1250°F is only a
few hours (less than 10). Table Q001.70-1 relates these impedances to
flow areas through various components. For example, if no flow area in
the core is available, but the peripheral areas including the radial
blanket, removable radial shield and thermal liner are available, the
appropriate flow impedance multiplier is 27. The third level thermal
requirement accommodates this case and even more severe cases in which
16| the core and part of the radial blanket are blocked. Based on these
evaluations the temperature-time requirement is judged to provide margin
for a wide range of post accident conditions even if natural circulation flow is the only means of heat transport.

These thermal requirements are used to assess components using the criteria of the ASME Code, but they are not included in the Code Stress Report for the Code Stamp. The stainless steel components can be shown to meet Code Faulted Condition limits, but there are no Code rules or material allowables for SA 508 (Closure Head and Reactor Vessel Flange) or Inconel 600 (Reactor Vessel transition region) at 1250°F. Analysis to rules developed via Code philosophy and criteria indicates that the performance of the SA 508 and Inconel 600 components would be acceptable to the Code if the data used were incorporated in the Code via a Code Case.

Reference Q001.70-1. L. Baker et al., "Postaccident Heat Removal Technology", ANL/RAS 74-12, July 1974.

Table Q001.70-1

Flow Impedance Multiplier* Associated with Various Flow Areas

Flow Component	% Total Flow through Components during Normal Full Power Operation	Flow Impedance Multiplier
Thermal Liner & Removable Radial Shield	3.5	625.0
Item 1 & Row 3 Radial Blanket	5.0	306.0
Item 1 & Rows 2 and 3 Radial Blanket	9.6	83.0
Radial Blanket Only	13.5	42.2
Thermal Liner & Removable Radial Shield and Radial Blanket	17.0	27.0
All Normal Flow Areas	100.0	1.0

*Impedance normalized to the normal design configuration.

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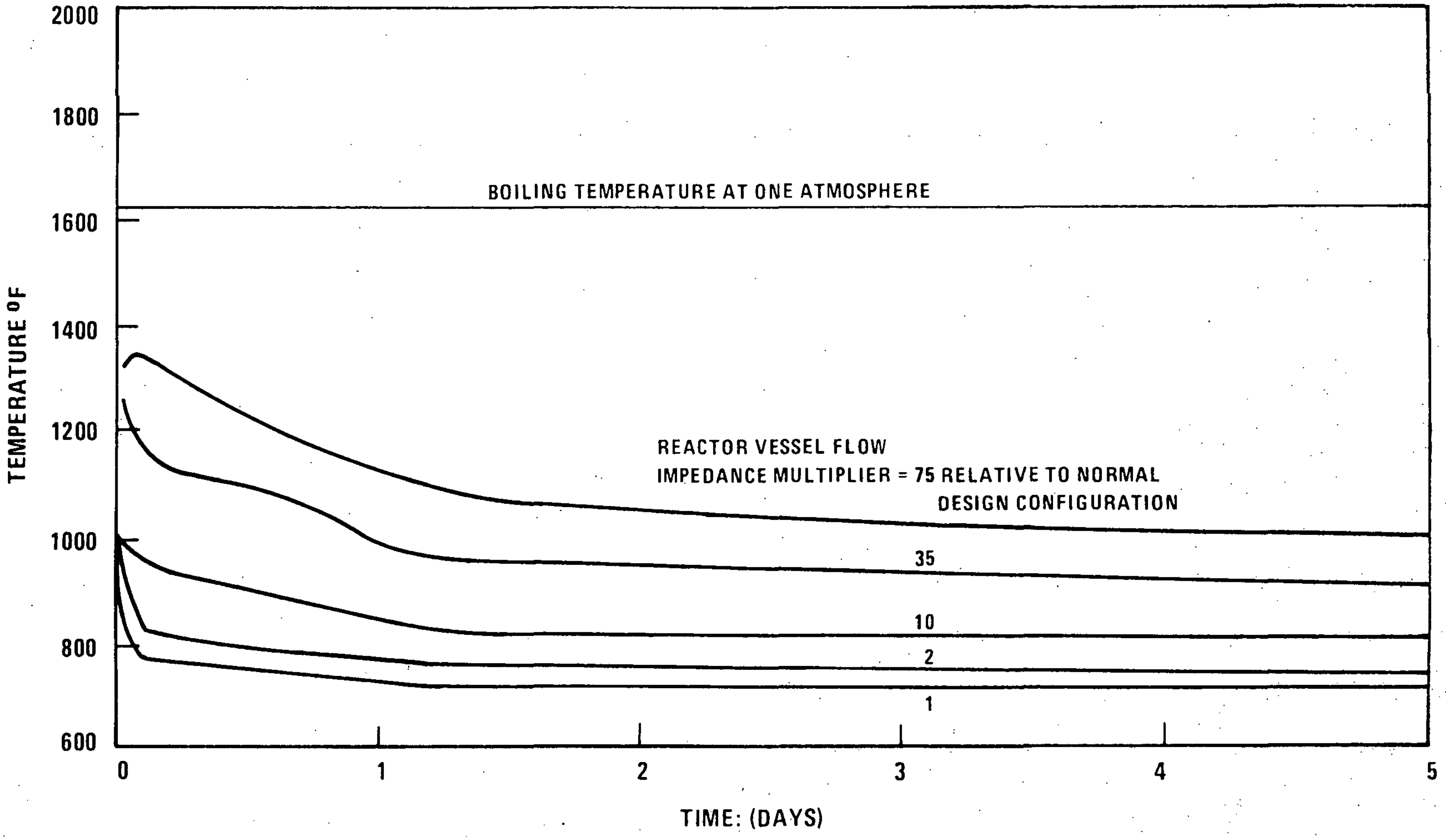

Figure Q001. 70 -1 Vessel Outlet Temperature After LOF Event

Question 001.71 (5.2.1.2-1)

What is the meaning of core replacement without structural modification given in Section 5.2.1.2.

Response:

The statement in question means that no structural components of the reactor enclosure or internals must be modified or removed to accomplish refueling.

Question 001.72 (5.2.1.2-6)

Discuss how these upward and downward 50×10^6 # loads will be accomodated by the concrete ledge and provide your analysis.

Response:

The information requested is provided in revised Section 5.2.2.1. 25

Question 001.73 (5.2.1.3)

Please provide details of the potential sodium leak paths under CDA conditions; describe what provisions exist for deflection of sodium. How much would be predicted to escape?

Response:

Under SMBDB loading conditions, an upward pressure pulse would cause the large rotating plug, intermediate rotating plug, and small rotating plug to impact sequentially against the margin shear rings. This loading would make the head/riser assembly deflect with maximum deflection occurring at the small rotating plug. The riser structures would maintain their concentricity at the upper end and provide passage for sodium through the riser annulus. Leakage paths to the head access area could develop only through the bearing assemblies.

The leakage requirements to be met by the head/riser assemblies are given in Section 5.3.2 of CRBRP-3, Volume 1 (Reference 10a of PSAR Section 1.6).

Question 001.74 (5.2.1.4)

How will the guard vessel be periodically tested to verify its integrity and availability? Will it be periodically inspected?

Response:

There is no planned periodic testing of the reactor guard vessel. However, the inner surface of the guard vessel is accessible for in-service inspection by the same television camera equipment that is used for in-service inspection of the external surface of the reactor vessel. This is described in more detail in Section 5.2.4.5 of the PSAR.

The guard vessel will be inspected periodically.

Question 001.75 (5.2.1.3-G)

Please discuss how one performs maintenance of CRDM, seals, etc. at an average plug temperature of $400^{o}F$.

Response:

A discussion of maintenance performance is provided in revised Section 5.2.1.3. 25

Question 001.76 (5.2.1.4)

Discuss how the annular space between the guard vessel and the reactor vessel is uniformly maintained during plant life to provide access for remote in-service inspection devices.

Response:

A discussion of the requested information is provided in revised Section 5.2.1.4. 25

Question 001.77 (5.2.1.4-1)

Describe the reactor vessel preheating system and the average metal temperatures between the guard vessel and reactor vessels as a function of time during preheat, startup and shutdown. Are the relative movements different during preheat and operation?

Response:

Reactor vessel preheat Section 5.2.1.5 and Figures 5.2-4 through 5.2-6 has been added to provide the information requested.

Question 001.78 (5.2.2)

The design sketches shown in Figures 5.2-1 and 5.2-2 have no dimensions. Provide dimensioned drawings.

Response:

Revised drawings in Section 5.2 provide dimensions for the reactor enclosure system.

Question 001.79 (5.2.2.2)

Describe how the gas suppressor plate is supported and its behavior under a CDA in the reactor.

Response:

The gas entrainment suppressor plate assembly is suspended from the lower radiological shield plate into the vessel outlet plenum to a depth of 110.65 inches below the top of each rotating plug. This assembly protects the head shielding from being contacted by the core coolant and minimizes the amount of cover gas entrained within the core coolant.

The assembly is designed to accommodate all operating, upset, emergency, and faulted events. See also revised Section 5.2.2.2. 25

Question 001.80 (5.2.3.2)

Discuss how FFTF experience will be incorporated in CRBRP. The experience gained on FFTF will be scattered among various individuals and not necessarily in existing documents. What feedback methods will be employed? Describe and discuss the procedures you intend to implement.

Response

This question is based on a section (5.2.3.2) of the PSAR dealing with controlled welding to maintain alignments between the core support structure and the upper end of the reactor vessel. The specific way in which this particular FFTF experience will be factored into is described in Section 5.2.3.2 as revised.

The question is asked in a more general sense; and a broader response is that FFTF experience will be carried-over to CRBRP in the following ways: 25

1. A large number of applicable technical decisions, procedures and processes were developed and documented within the FFTF Project. Cognizant CRBRP engineering personnel are encouraged to contact their 25 FFTF counterparts who provide them with the written information described above as well as oral discussion of items, ideas, and areas not documented. Information exchange occurs freely within the ARD-RM organization, and within the AI organization. The ARD-LRM serves the function of enhancing information exchange between the three RM's and the AE. The Construction Liason organization has been especially established to plan ahead to take advantage of FFTF experience to avoid construction and sequencing problems.

2. Many first-level and upper level CRBRP managers, cognizant engineers, 25 and key technical people have had directly applicable FFTF experience. As one specific example, in the area of the reactor vessel and closure head, at the time of writing of this response, the first level managers of head and vessel design, the vessel cognizant engineer, the assigned manufacturing engineer, and the assigned materials personnel were people who had responsibilities in the FFTF program which included those they now have on the CRBRP.

3. A formal system of design reviews, under procedures governed by RDT Standard F2-2, is in place and functioning on CRBRP. Wherever 25 practicable, FFTF personnel participate in the review or even chair the review. This has proven to be of exceptional benefit in technology transfer from FFTF to CRBRP. 25

Question 001.81 (5.2.4.4)

What will the Xe, Kr, and argon diffusion rates be through the elastomer seals postulating loss of the sodium filled trough? What activity level will exist at the operating floor?

Response:

The dip seals are designed to seal during normal reactor operation when the cover gas pressure is $15.05^{+0.25}_{-1.15}$ psia. The present capability is a pressure difference of 5 psi without leakage, and no known conditions can cause such a differential. However, if the dip seal would allow Xe, Kr and Ar to leak, two types of radiation exposure would occur. The first type is radiation due to stagnant noble gases in the riser space above the shielded head, and second type is radiation due to permeation through and leakage around the elastomer seals. At zero time after shutdown the stagnant noble gases would contribute 13 rem/hr to the operating floor, while a normal diffusion and leakage rate of 0.112 scc/sec. by the seals would contribute 0.042 rem/hr, totaling a dosage of 13.042 rem/hr. After 30 minutes the stagnant gases would contribute 4.4 rem/hr. while the seal leakage would contribute 0.040 rem/hr. to the operating floor, totaling a dosage of 4.404 rem/hr.

In the event of a loss of sodium in the dip seal trough, the sodium dip seal feed system would automatically refill the troughs. The riser annulus above the dip seal would then be purged with clean argon. All of these operations would be done without manned access in the head access area.

Question 001.82 (5.2.4.4)

What criteria will be used as a basis of decision as to whether or not backup seals are required for the CDA?

Response:

The design of the Riser Assemblies incorporates a backup seal (margin seal) on the assumption that the inflated elastomer dynamic seals may not have the capability of containing the mass of liquid sodium propelled upward during HCDA loadings.

The margin seals are designed to stop sodium flow out of the riser assemblies and bearing races. The seals would maintain an elastomer-to-metal contact against the races and risers, closing the leakage paths. The seals are self-energized so as to provide sealing under no pressure difference. A pressure difference across the seals causes them to seal more tightly with a force proportional to the pressure difference.

The details of the leakage requirements on the seals are given in Section 5.3.2 of CRBRP-3, Volume 1 (Reference 10a of PSAR Section 1.6).

Question 001.83 (5.2.4.5)

What provisions are available to detect failure in the omega seal? What provisions are incorporated in the design to repair an omega seal should it fail? Would you propose to continue to run the reactor in the event of omega seal failure? Discuss your plans to cope with this type of problem.

Response:

The design of the reactor vessel, closure head, and support system has been revised to eliminate the omega seal, Section 5.2.4.5, Reactor Vessel Cover Seal, has been deleted as indicated on revised page 5.2-10.

Question 001.84 (5.2.4.5)

The rationale given for not complying with Appendix H, 10CFR50, simply because the material is austenitic and the shutdown temperature is 400°F, is inadequate. Provide comprehensive technical justification for your position including applicable test data.

Response:

Appendix H is directed specifically toward reactor vessels fabricated from ferritic steels. As the CRBR vessel is to be made from an austenitic stainless steel,certain sections of Appendix H (in particular, those related to fracture toughness testing) are not considered applicable. Although both types of material suffer a loss of impact resistance as a consequence of neutron irradiation, that undergone by the ferritic steels is accompanied by a significant upward change in the ductile-to-brittle transition temperature. In austenitic stainless steels, there is no comparable effect, and the steels retain adequate impact resistance even after exposure to fluences as high as 10^{22} n/cm^2 (E_{Total}). (see Reference 1 below). However, as covered in subsection 5.2.4.5, the intent of Appendix H and the necessity for a material surveillance program is recognized in the design of CRBRP.

The fact that the shutdown temperature is 400°F is not one of the reasons for not complying with Appendix H. The temperature, 400°F, is mentioned in connection with the operation of the surveillance and in-service inspection (SISI) equipment, and is determined by essentially two factors:

(a) The inability of the SISI equipment to operate at temperatures higher than 400°F.

(b) 400°F will be the lowest temperature achieved by the reactor vessel while containing sodium.

References:

(1) J. R. Hawthorne, H. E. Watson, "Notch Toughness of Austenitic Stainless Steel Weldments with Nuclear Irradiation", Welding Research Supplement, June 1973, pp. 255-S - 260-S.

Question 001.85 (5.2.6)

Elaborate on your plans to assure continuous quality control during shipping, storage, and field erection to protect components. Identify the supplemental requirements in RDT standards that provide control during these time intervals after leaving the fabricators shop.

Describe and discuss what protective measures will be taken in the field during storage and construction to protect metal surfaces and sensitized weld regions from harmful materials and contaminants.

Response:

The specifications will require that packaging and packing be adequate to protect items while at the suppliers' facilities, during transportation to the delivery point and during storage at the site. The applicable 26
requirements will be in the respective equipment specifications rather than in RDT Standards.

The specifications will where appropriate provide requirements for sealing 26
the openings in the components, purging the components and/or their containers, selecting and using desiccants, selecting and using materials contacting the components which are suitably free of chlorides, fluorides, lead, copper, zinc, cadmium, sulfur, mercury, etc.

During storage at the Plant site, the equipment will be maintained in an inert gas environment when appropriate to protect it from contamination. 26
The purge gas, container integrity, etc., will be monitored to assure compliance with previously prepared procedures.

The quality assurance program to be implemented during field erection is described in Appendix F, A description of the Constructor Quality Assurance Program, Chapter 17-Quality Assurance. 26

Question 001.86 (Table 5.2-1)

Where RDT standards are identified to be employed, provide a description of supplemental requirements to existing codes and NRC guides. Identify which RDT Standards are unique. Why is RDT F9-5T identified as non-mandatory?

Response:

Each RDT standard is formatted to contain a table of contents and a scope of applications. The scope generally provides a description of the supplemental requirements to existing codes. None of the standards designated E or M are unique in that each is based upon an existing standard such as the ASME Code. The uniqueness of standards designated F is specified in the scope and applicable documents section, if any, of each standard. An index of RDT Standards is issued on a quarterly basis by Oak Ridge National Laboratory for the Energy Research and Development Administration.

RDT Standard F9-5T is identified as being non-mandatory in paragraph 1.2 of F9-5T and in paragraph 0.2 of RDT F9-4T as it addresses guidelines for the stress analysis aspects of design and supplies procedures for consideration and use by the manufacturer in meeting the requirements of RDT F9-4T, used in conjunction with ASME Code Cases 1592, 1593, 1594, 1595, and 1596.

Question 001.87 (5.3.1.1-6)

How will natural convection decay heat removal be demonstrated prior to plant operation for 1 and 2 loop operations? What contribution to system pressure drop do the check valves have?

Response:

Verification of natural convection decay heat removal is discussed in revised Section 5.3.3 for CRBRP. Revised Section 5.3.1.1 subsection check valve provides the information requested regarding system pressure drop due to check valves. 25

Question 001.88 (5.3.1.1-9)

Discuss the hot standby condition and the meaning of 600°F isothermal conditions. Describe how a ΔT is reestablished across the core and in the various loops.

Response:

The hot standby condition is a temperature condition that will usually exist following a normal shutdown or a plant trip. There are times such as a shutdown for the purpose of refueling or when HTS or steam generator system maintenance is indicated when the sodium temperature will be brought down to 400°F. Ordinarily, however, the sodium temperatures will be brought to the hot standby temperature of approximately 600°F. 600°F is specified as the hot standby temperature because both the PHTS and IHTS cold leg temperatures will approach the saturation temperature (593°F) of the water in the steam drum which will normally be controlled to 1450 psig following a trip or normal shutdown. The precise hot leg temperature will be a function of the PHTS flow established by the pumps at pony motor speed (7 1/2 to 10% of rated flow) and the decay power level. In time, the hot and cold leg temperatures will approach each other thus establishing essentially an isothermal condition. Restart of the primary and intermediate sodium pumps is performed at this essentially isothermal condition. Reestablishment of a ΔT across the core and in the various loops is discussed in Section 5.7.1.1.

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October 1975

Question 001.89 (5.3.1.1-2b)

How far from the frequency generated by the pumps between 40% and 100% flow will the natural frequency of the tubes in the IHX and the fuel subassemblies be?

Response:

The information requested is provided in revised Section 5.3.2.3.1. 25

Question 001.90 (5.3.1.1-2d)

What, if any, piping problems, arise if the dry empty pipe is heated and the top of the pipe is significantly hotter than the bottom of the pipe prior to filling?

Response:

The information requested is provided in revised Section 5.3.2.3.6, Structural Performance. 25

Question 001.91 (5.3.1.1-d)

How will it be demonstrated that the heat transport system can withstand a faulted condition and still perform its function of decay heat removal?

Response:

The information requested is provided in revised Section 5.3.1.1, Transients. 25

Question 001.92 (5.3.1.1-f)

Does the 300 MW sec reference design loading for a CDA establish limiting values for any of the heat transport system.

Response:

The 300 MW sec loading referred to in the question is obsolete. The SMBDB loading requirements are identified in Section 5.2 of CRBRP-3, Volume 1 (Reference 10a of PSAR Section 1.6).

Based on the preliminary analyses performed to date, these SMBDB loadings do not establish limiting conditions on any component of the heat transport system.

Question 001.93 (5.3.1.4.3)

Provide more description of your plans for hydrostatic and pneumatic testing of heat transfer system components. What test pressures are contemplated and what will be measured?

Response:

Refer to revised Section 5.3.1.4.3, Strength Tests For HTS Components.

Question 001.94 (5.3.2.1.3)

The inservice inspection plans are not clear; provide greater description of your planned program. It appears that weld inspection is a matter of convenience to be arranged during extended plant shutdown. Justify this position. Discuss how provisions can be incorporated later in the design of the facility if the need for surveillance programs are not established now.

Response:

Inservice inspections will be provided in accordance with the appropriate requirements of ASME XI, Division 3 (presently under development). Planned techniques for inservice inspections are presented in Section 5.3.2.1.3. For a discussion of incorporation of future inservice inspection techniques, see the response to NRC question 001.58 (5.1.2).

Question 001.95 (5.3.2.1.4)

Describe how you intend to implement the statement made in this section during field construction. Will special fabrication shops be set up?

Response:

This section recognizes care which must be exercised throughout the fabrication, shipping and construction of the plant. The specific implementation for guarding against these phenonmena is part of each component procurement process. No special fabrication shops at the site have been specified at this time. Any fabrication facilities required will be consistent with these precautions.

Question 001.96 (5.3.2.2.1)

Provide the test data to support the derivation of equations for short term tensile properties and over what portion of the plant life they are applicable.

Response:

The basis for the derivations of these equations including supporting test data is given in WARD-NA-3045-2, "The Effect of Carbon and Nitrogen on the Short-Term Tensile Behavior of Solution-Treated Types 304 and 316 Stainless Steels", dated July 1973. These properties are valid for a short term loading any time in plant life.

Question 001.97 (5.3.2.2.3)

Describe and discuss any ongoing programs to determine fatigue properties in the 650°F to 1000°F range if they are not currently available.

Response:

Fatigue properties in the 650°F to 1000°F range are currently available and are given in Code Case 1592. In addition, revised Section 5.3.2.2.3 of the PSAR discusses programs in place to further define fatigue properties of stainless steel.

Question 001.98 (5.3.2.3.1)

What are the pump hydraulic impedance characteristics during natural convection?

Response:

The primary pump locked rotor impedance values are presented in Section 5.3.3.3, page 5.3-42.

The intermediate pump locked rotor impedance values are to be inserted in revised Section 5.4.3.3.

Question 001.99 (5.3.2.3.1)

How long following scram does it take for oscillations to subside between various free surfaces and compressed gas spaces in the reactor coolant system?

Response:

The information requested is provided in revised Section 5.7.3. 25

Question 001.100 (5.3.2.3.1)

What pump speed is required to generate sufficient hydrostatic forces to center the shaft in the hydrostatic bearing?

Response:

Vendor information shows that the CRBRP pump hydrostatic bearing will have a load capacity at pony motor speed of about 165 lbs, ample for the expected total side load of about 50 lbs. Because the pump rotor is coupled to the drive motor by a rigid spool type coupling and the upper rotor support is in the lower motor bearing, a 50 mils anticipated maximum radial misalignment of the hydrostatic bearing will cause an 18 lbs radial side load on the bearing at start-up. Based on vendor bearing analysis, the journal under this load condition will be supported by a 5 mils thick fluid film at a shaft speed of approximately 30 rpm. If the bearing misalignment is the extreme (unlikely) value of 150 mils, the bearing side load at start-up is 53 lbs and the journal will be supported by a 5 mils thick fluid film at a shaft speed of approximately 59 rpm.

Question 001.101(5.3.2.3.1)

Provide detail of the pump internals showing shaft bearings, impeller seals, etc. at least to the same degree as that provided for the IHX.

Response :

Sectioned elevation showing the details requested have been added to Figure 5.3-14.

Question 001.102 (5.3.2.3.2)

In the event of an IHX leak, describe the procedure for identifying and plugging a defective tube.

Response:

At present, the details for locating and plugging a leaking tube have not been developed. However, plans are underway to develop details for both. This data will be provided when it becomes available.

Question 001.103 (5.3.2.3.2)

Describe how a radioactively contaminated IHX tube bundle would be removed.

Response

Based on a design change since PSAR submittal, the IHX tube bundle will not be removed separately. The entire IHX would be removed if necessary. As described in the answer to Question 001.102, tube leak detection and tube plugging will be handled in situ. Also, bellows replacement will be handled in situ. Section 5.3.2.3.2 has been revised to reflect this change.

Question 001.104 (5.3.2.3.3)

What forces are required to overcome the dashpot friction in the check valve during natural convection? Do you have test plans to verify its performance in sodium since it is not a direct scale up of the FFTF design?

Response:

Refer to the 2nd and 3rd paragraphs of revised Section 5.3.2.3.3.

Question 001.105 (5.3.2.3.4)

It is stated that commercially available pipe hangers and snubbers will be used. Does this also apply to the reactor cavity or any other area in which maintenance may be impractical due to radiation or temperature.

Response:

Commercially available pipe hangers and seismic snubbers will be used to provide pipe support and restraints, including in the reactor cavity. In the reactor cavity, removable access hatches are located to allow inservice inspection and component maintenance for the pipe support/ restraint components. Access provisions in the other areas for maintenance and inservice inspection are described in PSAR Section 5.3.2.1.3.

Question 001.106 (5.3.2.3.4)

The description given for the pipe clamps requires further clarification. Provide drawings or sketches of the pipe clamps.

Response:

The information requested is supplied in the revision of 5.3.2.3.4 and in Figures 5.3-36, 5.3-37A, 5.3-37B and 5.3-38.

Question 001.107 (5.3.3.1)

"Describe your Monte Carlo technique for randomly selecting heat transfer coefficients, process variables, and uncertainty ranges. Discuss how you establish the confidence level you use for design parameters".

Response:

The Monte Carlo technique is discussed in revised Section 5.3.3.1. 25

Question 001.108 (5.3.3.1.1)

What organization will review the structural evaluation plan (SEP) submitted by various component manufacturers and assure that it is in conformance with the plant requirements and that the analysis proposed by the vendor is responsive and adequate?

Response:

SEPs are invoked by RDT F9-4 which is a mandatory RDT Standard for all designs to the ASME Code for temperatures above 800°F. When a single SEP involves two or more reactor manufacturers (RMs), the lead reactor manufacturer (LRM) shall review and approve. When a single SEP involves an RM and is prepared by them for their design, it shall be reviewed and approved by the LRM. When a single SEP is created for an RM by one of its suppliers, the RM shall review and approve.

Question 001.109 (5.3.3.1.5)

Discuss the key modeling features and inherent simplifying assumptions in the basic computer codes identified in your stress analysis discussion. What programs have been verified by experimental work and what programs remain to be verified?

Response:

The key features of the codes used for structural analysis of PHTS components are identified in Appendix A. The information has been augmented in response to question 110.27. The specific modelling of the components is input for the code analysis and is determined by the analyst. 23

In general, verification of high temperature structural analysis methods is being conducted as a national program directed by ORNL; "Validation of High Temperature Design Methods and Criteria". This program is applicable to CRBRP.

Specific verification of the computer codes is addressed in response to question 110.27. 23

Question 001.110(5.3.3.1.5)

How will the dynamic load resulting from a steam generator failure be treated with respect to the IHX as an emergency condition?

Response:

In evaluating the structural adequacy of the IHX with respect to the Sodium Water Reaction, the dynamic nature of the intermediate sodium pressure history is being accounted for by using dynamic load factors. The factor will be applied to the maximum intermediate pressure which in turn is used to determine the pressure-induced primary stresses. These primary stresses are limited by the emergency condition allowables of Code Case 1592, Paragraph 3224, as modified by RDT F9-4T. The fatigue damage associated with the cyclic nature of the pressure history will be accounted for per Paragraph T-1400 of Code Case 1592. This damage will be essentially zero.

Question 001.111 (5.3.3.1.6)

Describe how overpressure loads from a CDA will affect the primary sodium pumps. Will seal failure occur? Will the pump integrity be maintained?

Response:

The HCDA is not part of the design basis for the CRBRP Reference Design as explained in Sections 1.1 and 15.1. As indicated in Section 5.3.1.1, Item F under "Transients", SMBDB loadings must be accommodated by the primary coolant boundary. SMBDB design requirements appropriate to the pumps appear in CRBRP 3, Volume 1 (Reference 10a, Section 1.6).

The SMBDB loadings on the primary pump of 550-575 psi will result in high primary pressure loads at the pump tank nozzles--the most critical area in the pump in terms of stress. These pressure loadings in the primary pump will cause stresses in the discharge nozzle to approach the yield point. In the pump tank suction nozzle, the yield stress may be exceeded, but the resulting stress will be within the allowable stress for ASME Section III faulted conditions. The pump procurement specification requires that the primary pump tank be capable of sustaining one occurrence of SMBDB loadings at the end of plant life without loss of ability to contain the sodium.

It is expected that the SMBDB loadings will not adversely effect the impeller/diffuser.

Shaft seal failure is not expected to occur under SMBDB loadings. The pressure pulse at the sodium argon interface in the tank will be attenuated as it passes through the annular space between the shaft and the shield plug and as it passes through the labyrinth bushing below the shaft seal. Above the bushing, there are two paths open to the cover gas; one is a low impedance path through the leakage oil reservoir to the RAPS system, and the second is through the seal oil cavity. If the pressure pulse which reaches the shaft seal exceeds the seal oil cavity pressure of 14.5 psig, it may produce a pressure unbalance which would cause the seal surfaces to separate, thereby allowing some gas to escape. As soon as the unbalance pressure is dissipated, the seal springs would cause the sealing surfaces to reseat, and the seal would then continue to function properly. Because of the low pulse pressure in the gas and the low impedance path to the RAPS system, opening of the shaft seal faces is not expected to occur.

Question 001.112 (5.3.3.1.6)

If the primary sodium pump seal fails, what will be the maximum radioactivity release to the reactor containment building?

Response:

The detailed design of the shaft seal system for the primary sodium pump has not been completed. Therefore, the maximum release to the containment building resulting from design basis faults in the shaft seal system cannot be precisely quantified now. However, the consequences of pump shaft seal failures are bounded by the hypothetical event analyzed in Chapter 15, Appendix A (Instantanesus Release of Radioactive Cover Gas) and are therefore orders of magnitude less than the guidelines valves of 10CFR100.

Question 001.113 (5.3.3.2)

With respect to natural convection, how will the preceding effects resulting from scram including flow surging between free surfaces and compressed gas spaces be included in your natural convection analysis to assure cladding temperatures in hot channels are not exceeded?

Response:

As discussed in the response to Question 001.99, there are no oscillations in free surface levels nor oscillations in loop flows which may be associated with oscillating free surface levels. The DEMO code used in the natural circulation analysis includes a model of the cover gas system and computes sodium levels in the pumps and reactor vessel as a function of time. Including these effects in the calculation does not significantly affect the resulting cladding temperatures for natural circulation events.

Question 001.114 (5.3.3.6)

It is stated that all computer codes that will be used have already been verified on the FFTF project. What plans do you have to support that statement? FFTF has not submitted an FSAR or completed its stress analysis.

Response:

The three computer codes identified for performing analyses of the PHTS piping are ELTEMP, WECAN and WESTDYN. The past and future verification efforts for these codes are discussed and references identified in Appendix A of the PSAR as revised in response to question 110.27.

Question 001.115 (5.3.3.6)

It is stated that the potential for longitudinal and circumferential crack growth will be investigated. What criteria will be used for judging acceptability and when will the results of this investigation be made available.

Response:

The criteria for judging acceptability is that no failures due to crack growth lead to loss of core coolable geometry. Loss of core coolable geometry is conservatively defined as sodium boiling. Refer to Table 1.1-6 for the guidance to the PSAR sections treating the leak before break philosophy. The results of the investigation are to be available in July 1976 as cited in PSAR Appendix E, Table E4-2.

Question 001.116 (5.3.3.6.2)

This section on fatigue crack growth is informative, and will have to be studied in detail. Provide details of the experimental work performed on straight pipe, elbows, and nozzles. Include any work that may be applicable to interior surface defects, or in the heat effected zones of welds, or regions of structural discontinuity.

Response:

This information has been included in the "CRBRP Primary Pipe Integrity Status Report", submitted to NRC on December 19, 1975. This report
25 has been incorporated by reference in Section 1.6.

Question 001.117 (5.3.3.10)

Most of the test data on creep is from uniaxial loading. How appropriate is this for triaxial loading.

Response:

Normal practice is to resolve the stress field into an equivalent stress for comparison with the uniaxial based material data. This approach is being used in LMFBR component structural analysis. This approach is consistent with multi-axial stress evaluation methods in the ASME code Section III.

Question 001.118 (5.3.3.10.1.2)

Since you are describing the opposing effects of sodium exposure with strain controlled conditions as opposed to higher temperature creep controlled conditions, how will you establish conservative design conditions?

Response:

As described in PSAR Section 5.3.3.10.1.1, the creep strength of the material will be degraded to account for the effect of the sodium environment. By also ignoring the fatigue strength improvement due to the sodium environment, the net result is a conservative estimate of the cumulative creep-fatigue damage.

Question 001.119 (5.3.3.10.1.4)

It is not clear from this paragraph just what the confidence level is regarding the current state of the art. Discuss what elevated temperature tests are required for CRBR.

Response:

The current data incorporating the environmental effects of sodium or argon upon 316 stainless steel are limited. The data from the specimens listed in Table 5.3-19 are required for CRBR. The need for data from the specimens listed in Table 5.3-21 is less certain and depends upon the results of tests scheduled on Table 5.3-19.

Question 001.120 (5.3.4.4)

Will the planned provisions for periscope examination in equipment spaces allow observation during plant operation?

Response:

The information requested is provided in revised Section 5.3.4. 25

Question 001.121 (Fig. 5.3-22)

Explain why the pump mass flow rate does not diminish with time. At what time are the pony motors turned on?

Response:

The information requested is provided in revised Section 5.3.3. 25

Question 001.122 (Fig. 5.3-29)

If all other loads such as the OBE, SSE and CDA were added, would the problem evaluation change.

Response:

It is anticipated that the addition of OBE, SSE and third level design margin loadings will not change significantly the crack growth as determined using the load cycle given in Fig. 5.3-29. However, all of the loading cycles on the piping will be used in the detailed assessment of crack growth as discussed in PSAR Section 5.3.3.6.

Question 001.123 (5.4.1.1)

Discuss the meaning of "following a faulted condition, the intermediae heat transport system must remain sufficiently intact to be capable of performing its decay heat removal function, including maintenance of intermediate coolant pump only motor flow". Does this mean a design condition for the pump to remain running is an emergency condition?

Response:

The data requested is provided in revised Section 5.4.2.3.1. 25

Question 001.124 (5.4.1.3)

Since the Intermediate Coolant System is accessible, discuss the choice of words "intent of ASME Section XI". Discuss what plans you have for inservice inspection of the intermediate coolant system.

Response:

The NRC position with respect to preservice and inservice inspection for CRBRP has been provided in Round 2 Question (RSP) 120.66. The Project discussed its plans with the NRC Staff in a meeting on September 8, 1976, and is subsequently revising its Inservice and Inspection Plan to more fully accommodate the NRC position. The plan and any necessary supporting discussion have been provided in response to Question 120.66. 33

Question 001.125 (5.4.1.4.2)

Discuss the selection of austenitic material for the intermediate coolant system when the selection of the material for the steam generator is CrMo. Would CrMo be more stable in an air atmosphere than sensitized stainless?

Response:

The selection of Types 316 and 304 austenitic stainless steel for the intermediate heat transport system (IHTS) hot and cold piping was based upon the same reasoning as put forth in the primary heat transport system (PHTS) materials selection (see Section 5.3.1.4.2). The IHTS piping material between the superheaters and evaporators has been chosen to be 2 1/4 Cr - 1 Mo to eliminate the need for six 18" and six 26" stainless to 2 1/4 Cr - 1 Mo transition welds. Stress analysis comparing 2 1/4 Cr - 1 Mo with stainless for this run indicates either material would be acceptable. In the hot and cold legs, there was no chance to eliminate the transition welds which are required since the pumps and IHX, which are the piping run end points, are austenitic stainless steel.

Identification of 2 1/4 Cr - 1 Mo as the piping material between the superheaters and the evaporators represents a change in the IHTS design from that submitted in the PSAR in April 1975. Sections 5.4 and 5.5, as required, have been revised to include this change. 40

Regarding the stability of Cr-Mo compared to stainless steel in an air atmosphere the general corrosion rate in air is considerably higher for the 2 1/4 Cr - 1 Mo than for austenitic stainless steels. The corrosion 40 allowances used in the piping design reflect this difference in corrosion rates. Sensitization of the austenitic stainless steel does not affect the general corrosion rate in air.

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Identification of 2 1/4 Cr - 1 Mo, as the piping material between the superheaters and the evaporators represents a change in the IHTS design from that submitted in the PSAR in April 1975. Modifications to Sections 5.4 and 5.5 as required to include this change will be submitted in a future amendment.

Regarding the stability of Cr-Mo compared to stainless steel in an air atmosphere the general corrosion rate in air is considerably higher for the 2 1/4 Cr - 1 mo than for austenitic stainless steels. The corrosion allowances used in the piping design reflect this difference in corrosion rates. Sensitization of the austenitic stainless steel does not affect the general corrosion rate in air.

Question 001.126 (5.4.1.5)

Please describe and discuss the differences in the response of the leak detection devices when operated in an air atmosphere as opposed to an inert atmosphere.

Response:

The information requested is provided in revised Section 7.5.5. 25

Question 001.127 (5.4.2.1.3):

Since welds are accessible in the IHTS loop, why is only surface examination for inservice inspection proposed? This does not provide volumetric examination of welds.

Response:

The NRC position with respect to preservice and inservice inspection for CRBRP has been provided in Round 2 Question (RSP) 120.66. The Project discussed its plans with the NRC Staff in a meeting on September 8, 1976, and is subsequently revising its Inservice and Inspection Plan to more fully accommodate the NRC position. The plan and any necessary supporting discussion have been provided in response to Question 120.66. 33

Question 001.128 (5.4.2.3.4)

Describe and discuss the pressure pulse from the stem generator design basis leak as seen by the IHTS pump, IHX, piping, and adjacent steam generators.

Response:

Section 5.5.3.6 "Results" has been modified and Figure 5.5-4A has been added in response to this question. The results are shown for the IHX since this is the only component listed above which is part of the primary boundary.

Question 001.129 (5.4.3.2)

At the start of natural convection, since there is an excess of heat transfer surface in the heat exchangers, the thermal center will move towards the hot end of the heat exchanger. Will this cause a delay in maintaining coolant circulation in the reactor coolant system?

Response:

The requested information is provided in revised Section 5.4.3.2. 25

Question 001.130 (5.4.3.6.1.2)

Provide a description and discussion of the mixing tee.

Response:

The information requested has been added to Section 5.4.3.6.1.2.

Question 001.131 (Fig. 5.4-36)

Provide a sectional elevation of the Intermediate Sodium Pump.

Response:

Sectional elevations have been added to Figure 5.4-1.

Question 001.132 (5.5.1.1)

Justify the design basis as to number and sequence for guillotine rupture of steam generator tubes.

Response:

Revised Sections 5.5.1.1 and 5.5.3.6 discuss the design basis for steam generator leaks.

Question 001.133 (5.5.1.1)

How rapidly will the sodium dump system remove sodium from the steam generators, and will the IHX remain wetted with sodium following dumping of the sodium?

Response:

Section 5.5.1.1 has been modified to respond to this question.

Question 001.134 (5.5.1.5)

Describe and discuss the early leak detection system for the steam generators. What are the limits of detectability?

Response:

Section 5.5.1.5 indicates that further details of the steam generator leak detection system are provided in Section 7.5.5.3. The limits of detectability and descriptions of the leak detection system, system operation, and instrumentation are included in revised Section 7.5.5.3.

Question 001.135 (5.5.2.3.4)

Describe how the wastage baffles work in the lower tube sheet region, and what provisions made to assure that they are not blown out of place or cause failures of adjacent tubes.

Response:

Section 5.5.2.3.4b has been revised to incorporate a discussion of the wastage baffle.

Question 001.136 (5.5.2.4)

How will the reaction forces from the power relief and safety valves be accommodated, and what is their magnitude.

Response:

The information requested is provided in revised Section 5.5.2.4. 25

Question 001.137(5.5.2.7)

It is apparently intended to reuse the sodium dump subsystem. Describe and discuss the effects of hot caustic in the system. What is the anticipated life of the system?

Response

Section 5.5.2.7 (pg 5.5-16) has been modified to provide more details regarding anticipated life of the system.

Question 001.138 (5.5.3.3)

Does the steam drum have internal separators to keep water droplets out of the superheater?

Response:

The steam drum has internal centrifugal separators as well as plate-type dryers.

Section 5.5.2.3.5 and Figure 5.5-4 have been added to provide a description of the steam drum.

Question 001.139 (5.5.3.5)

Of the 4000 hours of testing of the AI Test Steam Generator, how much was at CRBRP operating temperatures?

Response:

Revised Section 5.5.3.5 contains the information requested.

Question 001.140 (5.5.3.6)

Provide your analysis of tube whip in a steam generator showing that adjacent tubes are not failed.

Response;

Section 5.5.3.6 has been modified to add the analysis of tube whip in the event of a steam generator module tube failure.

Question 001.141 (5.5.3.6)

Discuss the method used in TRANSWRAP to treat the attenuation of acoustic waves in piping and components following a tube failure.

Response:

In response to this question, PSAR Section 5.5.3.6 has been expanded.

Question 001.142 (5.5.3.6)

Describe and discuss the modeling used in the region of tube failure such as rate of formation of reaction products, energy partitioning, local hot spot temperatures and resultant pressure source term.

Response:

See additional information incorporated into Section 5.5.3.6.

Question 001.143(5.5.3.6)

What criteria are used in arriving at the conclusion that the integrity barrier will withstand a large sodium water reaction?

Response:

See response to question 001.110.

Question 001.144 (5.6.1.3.7)

Will the Auxiliary Feedpump turbine be kept hot by bleed steam for quick start operation?

Response:

Revised Section 5.6.1.2.3.2 provides the requested information. 25

Question 001.145 (5.7.4)

Clarify Table 5.7-1. The numbers shown for pump trips from Loss of Offsite Power appear inconsistent and it is not obvious why there are no Scram cycles on the Intermediate Heat Transfer System pumps from the overpower transients. It is also not clear why there are no trip cycles from the steam generator transients.

Response

Section 5.7.4 and Section B.2 of Appendix B discuss the bases for the heat transport system design transients. As pointed out in those sections, duty cycle events resulting in similar transients on a particular component were grouped together with the most severe transient in each group selected as the umbrella transient for that group. The frequency specified for each transient is the sum of all duty cycle events assigned to that group. Since individual duty cycle events have different effects on different components, the grouping of duty cycle events results in differing umbrella transients and frequencies for each component.

Because of differences in grouping events, the frequency of plant trips assigned to the various components due to loss of offsite power supplies differs. The number of trips due to loss of offsite power for the primary pump is 11 compared to 19 for the intermediate pump. For the primary pump, two events were grouped together with duty cycle event U-18 selected as the umbrella transient; U-18 (6 events) and U-5b (5 events). For the intermediate pump, three events were grouped together with duty cycle event U-18 also selected as the umbrella transient; U-18 (6 events), U-12 (3 events), and U-17 (10 events). Similar differences occur with other components.

There are no trips from full power with normal decay heat assigned to the intermediate pump or steam generator components because these duty cycle events (U-1a) have been grouped under event U-1b, reactor trip from full power with minimum decay heat. The U-1b event results in a somewhat more severe transient on the intermediate pump and steam generator components and thus conservatively provides an umbrella for the U-1a event (as well as other events).

In order to provide a clear cut reference to the duty cycle event descriptions, a column has been added to Table 5.7-1 showing the duty cycle event designation in Appendix B for each event listed.

Question 001.146 (5.3-11)

Provide a discussion of the potential for cell atmosphere ingress into the coolant system in the event of a leak.

Response:

The potential for leaks in the piping is extremely remote. However, this improbable occurrence is evaluated in revised PSAR Section 5.3.2.1.1. 25

Question 001.147 (5.3-28)

Clarify what is meant by the statement that the "piping supports will be designed to fail if the loads are well beyond normal operating conditions".

Response:

If the SMBDB loadings were to cause rapid and extreme pipe motion the seismic restraint could cause excessive pipe loads. This situation is avoided by use of shear pins in the snubber shafts which are set to fail only under such a sevene event.

Question 001.148 (5.3-31)

Discuss the effect on the drained loop of any of the design thermal transients in the two loop operating mode.

Response:

As provided in Section 16.3.2, the plant will not be operated on two-loops with one loop drained, thus this question is not applicable.

Question 001.149 (5.3-31)

Provide the design bases and related design details included to reduce any splash effect.

Response:

The requested design bases and related details are provided in revised Section 5.3.2.5.6. 25

Question 001.150 (5.3-32)

Provide the results of your analysis of the rate of back diffusion.

Response:

The results of analysis of back diffusion show the following relationship:

1) $C(IHTS) = C(PHTS) \exp [-1.6 \times 10^{+7}]$

where:

C(PHTS) is the primary heat transport system volumetric concentration of ^{22}Na at the defect source

C(IHTS) is the resultant ^{22}Na concentration in the intermediate heat transport system.

As can be seen from the above relationship, the activity that can diffuse upstream into the intermediate sodium is insignificant.

Question 001.151 (5.3-34)

Provide the schedule for submitting the results of the evaluation identified in 5.3.3.1.2.

Response:

The structural evaluation of PHTS pressure-containing components listed in the referenced section will be contained in their final design reports. The current schedule for completion of final design reports is given below. Each of these reports will be prepared by the component vendor with the exception of the PHTS piping and the flowmeter. At the flowmeter location,the sodium pressure is contained by the piping. Design and analysis of the PHTS piping is being done by Westinghouse. The final PHTS piping design report will include analysis of all associated parts under Westinghouse design cognizance such as thermowells and connections to pressure sensors.

PHTS Component	Final Design Report Due
Coolant Pump	November 1982
Intermediate Heat Exchanger	presently available
Cold Leg Check Valve	presently available
Piping	October 1983
Reactor Vessel	April 1982

Question 001.152 (Tables 5.3-18, 19) (Figures 5.3-1 thru 13)

Identify the publications containing the information in these tables and figures.

Response:

The publications requested are listed in Section 5.3 references 52-57. 25

Question 001.153 (5.4.2.5.6)

Describe the means of directing IHTS leaks in the SGB to splash pans.

Response:

There are no special means provided to direct the sodium from a leak in the intermediate system to the sodium catch pans. The catch pans cover the entire floor area in regions where potential sodium leaks could occur, and the sodium will flow directly to the pans or drain off the cell walls or equipment into the pans. As the SGB design progresses, if splash shields are required to protect certain components or structures, they will be designed such that they do not hold up sodium but allow the leakage to flow to the catch pans. At present, no items have been identified which require use of splash shields.

Question 001.154 (5.4.3.1.7)

What is the design basis for seal leakage in the dump valves.

Response:

A discussion of seal leakage is presented in revised Section 5.4.3.1.7.

Question 001.155 (5.4.3.3)

What "similar pump designs" were used to arrive at the IHTS pump characteristics?

Response:

The primary and intermediate pump performance curves shown in Figures 5.3-19 and 5.4-3, respectively, are based on the performance of the Byron Jackson 20 x 20 x 18 B-HDR commercial boiler feed pump which serves as a model for the CRBRP pump. The CRBRP pump and the B-HDR pump have the same specific speed with the CRBRP pump hydraulics being scaled up from the B-HDR pump hydraulics by a factor of 2.1. The head and capacity of the model B-HDR pump as follows:

$$H_2 = H_1 \, (2.1)^2 \, (n_2/n_1)^2$$

$$Q_2 = Q_1 \, (2.1)^3 \, (n_2/n_1)$$

where the subscripts 1 and 2 refer to the model B-HDR pump and the CRBRP pump, respectively.

Question 001.156 (5.4.3.6.1.1)

Provide the bases and criteria for the design of the IHTS piping system.

Response:

The design bases for the IHTS, including the piping are contained in Sections 5.1.3 and Section 5.4.1. In addition general plant requirements are given in Chapter 1.0. Those requirements which reflect on the IHTS piping are summarized below.

A. Normal Operation:

(1) Each of the three circuits of IHTS piping system shall be thermal-hydraulically designed on the basis of removing 325 MWt from the IHX to the Steam Generator System. The maximum coolant (sodium) velocity shall be less than 30 fps to minimize the potential for erosion.

(2) All IHTS sodium piping components (pipes, tees, elbows, mixing tee, etc.) shall be drainable. Positive means of preventing accidental drainage shall be provided.

(3) There shall be no direct piping connections between the primary and intermediate systems; physical mixing of primary and intermediate coolants shall be completely avoided. The intermediate loop pressure shall in all circumstances be 10 psi higher than the primary loop pressure in the IHX barrier.

(4) To minimize the chance of a sodium-water reaction in the event of a sodium leak, the design shall preclude the possibility that water can collect underneath sodium containing equipment or piping. Water lines will not be routed in the vicinity of sodium containing equipment or piping, except as absolutely necessary.

B. Decay Heat Removal:

The IHTS piping shall be arranged such that the reactor decay heat removal can be effected by the method of pony motors operation or by natural circulation, utilizing the normal heat removal train.

C. Service Life: A service life of 30 years shall be used as a basis for the IHTS piping system. The design criteria for the IHTS piping system are summarized as follows:

(1) Code Classification: Based on the functional requirements, the IHTS piping system is classified Safety Class 2. The system is therefore required to be designated ASME Code Class 2. However,

to assure added quality and reliability of the system, the system will be designed and constructed to ASME Code Class 1 requirements.

(2) Structural Design Criteria: The structural design criteria for the piping system are specified in Section 5.4.1.2 of the PSAR.

Question 001.157 (5.5.2.1.2)

Provide the maximum allowable valve leak rate.

Response:

Section 5.5.2.1.1 has been expanded including addition of Table 5.5-12 in response to this question.

Question 001.158 (5.5.2.7)

Provide the design basis for the rupture disc.

Response:

Revised Section 5.5.2.7 provides the requested information.

Question 001.159 (5.3.3.2)

Provide more detail on the analysis of natural circulation. Provide the analysis and calculations to demonstrate the adequacy of natural circulation to comply with the fuel cladding criteria.

Response:

Natural circulation analyses for the preliminary design are currently being completed. Reference 37 of Section 5.3 of the PSAR gives the details requested and Section 5.3.3.2 has been revised to summarize that report.

As indicated in the reference above for a postulated event of this low probability, it is not appropriate to apply fuel cladding criteria. The criterion for success is that sufficient decay heat removal shall be provided, by natural circulation in the main heat transport loops, to prevent loss of core coolable geometry following shutdown from full power operation with three HTS loops in service.

Question 001.160 (5.6.1.3.2)

Provide an analysis and calculation to demonstrate the adequacy of natural circulation for the PACC.

Response:

Revised Section 5.6.1.3.2 provides the PACC natural circulation analysis requested. 25

Question 001.161 (5.3.3.5)

Provide the information required in the Standard Format Section 5.3.5.

Response:

The information requested is provided in revised Section 5.3.3.5 of the PSAR.

Question 001.162 (5.7.1)

Provide a complete discussion of startup and shutdown in accordance with the STANDARD FORMAT Section 5.3.7 and address the specific items identified in the format.

Response:

The information requested is supplied in the revised version of Sections 5.7.1.1 and 5.7.1.2.

Question 001.163 (5.3.9)

Discuss the transient effects on the heat tranpsort system at full power with scram, of the following events:

a. Loss of an Intermediate Coolant Pump
b. Loss of a Feedwater Pump
c. Loss of Reactor Coolant Flow
d. Loss of Intermediate Coolant Flow
e. Loss of Feedwater Flow
f. Turbine Trip
g. Loss of Station Electrical Power

Response:

The accident analyses discussed in Chapter 15 which relate to the reactor core effects due to the items a, b, d, and e above have been expanded to include the transient effects on the Heat Transport System. The revised sections are listed below. Items b and e are both covered by the modifications to Chapter 15 dealing with SGAHRS initiation as an umbrella event for the two events. In addition, the response to NRC question 001.169 contains additional supportive information relating to items b and e. The following modifications constitute the response for the specific items identified.

a. Revised Sections 15.3.1.3 and 15.3.2.2.
b. See answer to question 001.169, Also, revised Section 15.3.1.6.3 + 15.3.1.6.3.
d. Revised Section 15.3.3.5.1, 15.3.3.5.2 + 15.3.3.5.3.
e. See answer to question 001.169. Also, revised Section 15.3.1.6.2 + 15.3.1.6.3.

Question 001.164 (5.7.4)

Submit temperature response curves for the components in the heat transport system corresponding to the flow transients discussed with the Standard Format, Section 5.3.10.

Response:

Section 5.3.10 of the Standard Format, Evaluation of Thermal Hydraulic Characteristics and Plant Design, states, "Summarize the anticipated Plant Transients and their estimated number of occurrences and submit temperature response curves for the components in the heat transport system corresponding to the calculated flow transients."

Section 5.7.4 of the PSAR provides, in conjunction with Appendix B of the PSAR, a summary of the transient events considered for the basis of structural evaluation of plant components. Table 5.7-1 provides a preliminary summary of the design transients for the major components and Figures 5.7-3 through 5.7-11 have been provided to illustrate the temperature response, at selected locations in the system, for several of the most severe transients.

There are approximately 450 individual temperature and flow transient curves (excluding most pressure transients) for the IHX, Primary Pump, Intermediate Pump, Steam Generator Module, Check Valve, and Reactor Vessel. These curves in themselves, provide no verification of the structural adequacy of individual components. For this reason they have not been submitted with the PSAR. They furnish inlet transients, (for flow, temperature and pressure) to the component manufacturers. These input conditions are then used to compute transient temperatures in the structural parts of the components. These time/temperature histories for individual nodes in a component then are used to perform a structural evaluation according to Section III of the ASME Code and Code Case 1592 as modified by RDT F9-4T where appropriate. The thermal/hydraulic reports and stress reports by component manufacturers, then, provide verification of the structural adequacy of plant components.

An evaluation of the adequacy of the transient requirements specified in equipment design specifications may be made on the basis of a review of the duty cycle (which provides a description of the events along with the number of occurrences assumed for each event) and the DEMO Code(previously furnished to the NRC separately) used to compute transients.

Question 001.165 (5.6.1.1.1)

Provide a summary description of the design rationale for the various decay heat removal systems presenting the logic by which the heat removal burden is transferred from the Balance of Plant to the Auxiliary Feedwater System and then to the Protected Air Cooled Condensers. Include a discussion of the steam pressures and temperatures at which the various systems phase into and out of service.

Response:

A summary description is provided as an introduction to Section 5.6.1. 17

Question 001.166 (5.6.1.2)

Provide suitable references to the system schematics presented in Section 5.1.

Response:

Section 5.6.1.2.1.1 has been revised to include the reference.

Question 001.167 (5.6.1.2.3)

Provide the reactor decay heat curve which is used for your component sizing and system response calculations.

Response:

The requested Reactor decay heat curve is in new Figure 5.6-4 and described in revised Section 5.6.1.2.3. 25

Question 001.168 (5.6.1.3.9)

The use of 115% of rated power, 1121 MWt, as a design basis suggests that the SGAHRS may overcool the system when actuated. What consideration has been given to overcooling? In your response discuss the effects on the potential for natural circulation in the sodium systems.

Response:

The concerns expressed in this question are addressed in revised Section 5.6.1.3.9. 17

Question 001.169 (5.6.1.3.9)

Explain the notes used in Figure 5.6-1. Amplify the discussion of the results presented in this figure to include the resulting sodium system temperatures.

Response:

The revised Section 5.6.1.3.9, "Operational Characteristics" and new Figures 5.6-2 and 5.6-3 provide the answer to the above question and give additional clarity on the operation of SGAHRS. Revised Figure 5.6-1 should clarify the notation indicated above.

Question 001.170 (5.6.1.2.3.1)

Provide a preliminary design drawing of the PACC with its steam/condensate piping and air cooling system.

Response:

A preliminary design drawing of the PACC is not available. In lieu of this drawing, the following description and data in Section 5.6.1.2.3.1 has been expanded.

Question 001.171 (5.6.2.3.2)

Estimate the heat removal capability of the OHRS based on estimated natural circulation flow with no heat removal through the IHX's.

Response:

The OHRS has no natural circulation capability as stated in revised Section 5.6.2.3.2.

Question 001.172 (Table 5.6-1)

Confirm that the minimum NPSH figure given for the AFP motor drives applies to the motor driven AFP's.

Response:

The NPSH given in the PSAR under "AFP motor drive" is listed erroneously. It has been removed and placed under "Motor driven AFP's" in the revised Table 5.6-1. An error showing a design pressure of 25 psig for the turbine driven pump has also been corrected to read 2200 psig. 17

Question 001.173 (5.6.2)

Provide an equipment list and materials summary for the DHRS as you have for the SGAHRS. Include valve and pump classifications.

Response:

Revised PSAR Section 5.6.2 and Tables 5.6-7 and 5.6-8 provide the information requested.

Question 001.174 (Section 5 and others)

On all P&I diagrams furnished in the PSAR identify the boundaries between safety classes.

Response:

As indicated in the response to Question 001.274, CRBRP will be designed and constructed in general agreement with the Regulatory Staff Position on Safety Classifications. An integrated P&ID is being prepared and will be included in a future amendment.

Question 001.175 (5.6.2.3.7)

a. Estimate the extent of degradation of cooling ability, etc., quantitatively, as a result of inadvertent operation of valves.

b. What other malfunction or inadvertent function of components can interfere with or degrade the expected OHRS performance, and to what extent will the performance be affected?

c. How much time is available for operator action in the event of a mispositioned valve, and how is this calculated?

Response:

See revised section 5.6.2.3.7.

Question 001.176 (6.2.1.2)

Provide a more complete system design description for the containment including the principal dimensions, free volumes, etc.

Response:

The information requested is provided in revised Section 6.2.1.2.

Question 001.177 (6.2.1.4)

For containment testing, exceptions are taken to various sections of 10CFR50, Appendix J. Justify these exceptions.

Response:

All exceptions in Section 6.2.1.4 have been withdrawn except III.D.2. See the justification to this exception in the response to Question 040.17 (6.2.1.4).

Question 001.178 (6.2.1.3)

The transients presented for the containment response to the sodium fire are not calculated out to steady-state. Does your analysis indicate that pressure and temperature will asymptotically approach ambient conditions. What provisions, if any, will be made for vacuum relief?

Response:

The Information requested is provided in revised Sections 6.2.1.2 and 6.2.1.3. 25

Question 001.179 (6.2.1.3)

The containment is not fitted with an emergency cooling system. Describe your model and calculations for possive post accident heat rejection by the containment. Discuss the heat sink conditions assumed.

Response:

Revised Section 6.2.1.3 describes in detail the design basis pressure and temperature loadings imposed on the containment as a result of the most limiting in-containment sodium fire and has been expanded to discuss passive post accident heat rejection by the containment.

As described in Section 6.2.1.3, the sodium pool fire is located on the floor of the Overflow Vessel/Storage Tank Cell, located below the containment building operating floor. The fire is postulated to occur during maintenance when the cell and upper containment atmospheres communicate freely through an open 21 Ft^2 access hatch. 25

Question 001.180 (6.2.1.3)

Describe the structures within the containment which are not seismic Category I. Evaluate the consequences of their failure in a SSE.

Response:

The information requested is provided in revised section 3.8.3.1. 25

Question 001.181 (6.2.4.1)

Explain your reasons for selecting a continuous purge type ventilation system for the containment.

Response:

This explanation is provided in revised Section 6.2.4.3. As described in PSAR Section 6.2, (supplemented by response to Question 310.18) the RCB will be equipped with automatic vent closure following detection of high activity in the exhaust. The design basis accident for the RCB is described in PSAR Section 6.2 and analyzed in Section 15.6.1. The resulting doses are decades below appropriate 10CFR100 guideline values. 25

Question 001.182 (Table 6.2-5)

Amend or supplement this table to list isolation valve sizes, their loss of power position, and their required post-accident position. Also enumerate all lines penetrating containment which have no isolation valves.

Response:

The information requested has been provided in Table 6.2-5 in response to Question 040.11.

Question 001.183 (6.2.4.1)

The automatic closure of isolation valves in lines connecting directly to the containment atmosphere appears to be effected only by a high radiation signal from the Head Access Area. Discuss the possibility of using high temperature or high pressure signals there and in other locations as well to control these isolation valves.

Response:

A discussion of using high temperature or pressure signals for the initiation of the Containment Isolation Systems is provided in Revised Section 6.2.4.3. 25

Question 001.184 (9.1.2.1.1)

Section 9.1.2.1.1 describes the top of the EVST as being designed to absorb the load of the heaviest (equipment) dropping onto it. Section 9.1.2.1.2 states that the EVST top is designed for the accidental load of 13.5 tons (the NFTM weight) being lowered onto it at 8 fpm. Resolve this discrepancy.

Response:

The design basis for the EVST, is stated correctly in Section 9.1.2.1.1.

Section 9.1.2.1.2 has been revised. The EVST head design and the potential impact load on the head have also been revised. This is discussed in the response to Question 001.390.

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Question 001.185 (9.1.2.1.3)

What is the calculated k_{eff} for the EVST loaded with 650 new fuel assemblies of the highest reactivity but drained of sodium?

Response:

The information requested is provided in revised Section 9.1.2.1.3. 25

Question 001.186 (9.1.2.2)

Provide the results of criticality calculations for the FHC storage tank containing ten spent fuel elements of maximum reactivity without sodium in the tank.

Response:

Criticality calculations for the FHC storage tank containing 10 fuel assemblies with the highest reactivity (outer core equilibrium feed assemblies) have been performed using the one-dimensional multigroup Sn transport code ANISN. The problem was run in cylindrical geometry with vertical buckling set equal to zero and smeared number densities corresponding to the fueled horizontal plane of the tank. There were 126 energy groups and S-8 quadrature was employed. The analysis assumed a uniform tank temperature of 450^{o}F. The calculated k_{eff} is 0.455 with the tank drained of sodium and 0.648 with the tank at normal operating sodium level.

Question 001.187(9.1.2.2.1)

Provide the results of analysis of the accidental dropping of the maximum piece of equipment handled by the RSB crane from maximum height onto the weakest point of the FHC roof. In your analysis, assume a concurrent spent fuel handling operation is taking place in the FHC.

Response

The analysis is provided in revised Section 9.1.2.2.2. 20

Question 001.188 (9.1.3.1.1)

Provide the design basis for EVST cooling by relating the heat loads given to the number and history of the spent fuel assemblies involved.

Response:

Revised Section 9.1.3.1.1 and new Table 9.1-4 provide the requested design basis. 25

Question 001.189 (9.1.3.2.1)

Provide the design basis for FHC spent fuel storage tank cooling by relating the heat loads given to the number and history of the spent fuel assemblies involved.

Response:

The requested design basis is provided in revised Section 9.1.3.2.1 and new Table 9.1-4. 25

Question 001.190 (9.1.4.1)

Provide a complete discussion of the use of the simulated core assemblies in the refueling procedure. Include a discussion of the storage locations for these assemblies and the possible mis-location of real fuel assemblies there.

Response:

The simulated core assemblies in the refueling procedure are discussed in revised Section 9.1.4.1. 25

Question 001.191 (9.1.4.4.1)

Discuss the capability provided for removing a jammed fuel assembly from the core.

Response:

The capability provided to remove jammed fuel assemblies is described in the last paragraph of Section 9.1.4.4.2.

Question 001.192 (9.0 and others)

The attached branch position, APCSB 9-1 has been developed to reflect the NRC staff position on overhead handling systems for nuclear power plants. Provide an evaluation of all of the principal CRBRP overhead handling systems against APCSB 9-1-

Response:

Additional information on design of overhead handling systems is contained in revised Section 9.2.1.2.2. Branch position APCSB 9-1 is being evaluated for technical feasibility and applicability to CRBRP. 25

Question 001.193 (9.2.1.1)

Provide the criteria for determining whether hands-on, semi-remote, or fully remote operations are to be performed.

Response

Maintenance operations will be of the hands-on and semi-remote type. There are no fully remote operations. Revised PSAR Section 9.2.1.1 provides the criteria for determining which operation is to be performed.

Question 001.194 (9.2.1.1)

Provide a preliminary list of components where semi-remote maintenance may be required, and the "conceptual design basis" for related maintenance equipment.

Response:

See revised PSAR section 9.2.1.1

Question 001.195 (9.2.1.2.2)

Provide a description of the large component floor valves, including their design bases.

Response:

See rewritten Paragraph 5 of Section 9.2.1.2.2.

Question 001.196 (9.2.1.2.2)

Provide a description of the special floor valve for removing reactor internals.

Response:

For description of reactor internals design package see expanded Section 9.2.1.2.2.

Question 001.197 (9.2.1.2.3)

Provide a description of, and design bases, for the periscope systems, ports and gas lock valves.

Response:

See revised paragraph 4 of Section 9.2.1.2.3.

Question 001.198 (9.2.2.1)

Provide an analysis of normal and accidental process fluid reactions, for both Na and NaK operations.

Response:

The information requested is provided in new PSAR Section 15.7.3.7.

Question 001.199 (9.2.2.2)

Provide a description of, and design bases for, the Intermediate Sodium-Removal System.

Response:

The Intermediate Sodium Removal System (ISRS) is located in the Steam Generator Building (SGB) of the CRBRP.

The funciton of the Intermediate Sodium Removal System is to remove the residual sodium, which is contaminated with tritium, from the components of the Intermediate Heat Transport System. The major components to be cleaned in the Intermediate System are the Steam Generator Module and the IHTS Pump Assembly. The ISRS provides the capability to clean these components 12 and 6 times in 30 years, respectively.

The Intermediate Sodium Removal System consists of a 12,000 gal. cleaning vessel, and associated process equipment. (See Figure 9.2-3 for Equipment Arrangement, and Figure 9.2-4 sheets 1 and 2 for P&I Diagram.) The cleaning process will be identical to that for the Primary System (see Section 9.2.2.2) with the exception that there will be no decontamination capability provided in the intermediate system, and the Steam Generator Modules, because of their size, will be cleaned outside of the Intermediate System Cleaning Vessel.

As noted in Section 9.2.2.2 of the PSAR, this system will be designed and built after reactor start up. Thus, complete details are not available at this time.

Question 001.200 (9.2.2.3)

Provide a description of the cold trap removal operation, including an analysis of the quantities of wastes processed.

Response:

A discussion of cold trap handling procedure has been added to Section 9.2.1.3; reference is made to Section 11.5.3 for analysis of quantities of wastes processed.

Question 001.201 (9.2.2.3)

Provide a detailed description of the equipment and procedures used to contaol hydrogen gas concentrations, including an identification of the safety class of such equipment, and the means of controlling water vapor introduction.

Response:

The detail description requested is found in revised Section 9.2.2.3. 25

Question 001.202 (9.3.1.3)

Provide a description of the "normal precautions and fire protection used in handling sodium and NaK".

Response:

See revised Section 9.3.1.3.

Question 001.203 (9.3.1.5)

Identify the planned location of leak detection sensors, and their required sensitivity.

Response:

The planned locations of leak detection sensors for the Auxiliary Liquid Metal System are contained in Table 9.3-4. This table shows leak detectors for the sodium and NaK receiving system as well as for all of the Auxiliary Liquid Metal System throughout the CRBRP. Since the Sodium to Gas Leak Detection System is still in Preliminary Design, this table is subject to change.

The sensitivity requirements for the leak detectors for the auxiliary systems are based on maintenance and availability requirements. Design criteria and bases for the Sodium Leak Detection System are described in PSAR Section 7.5.5.1 as revised by the response to Question 222.75.

During preliminary design of the Sodium to Gas Leak Detection System (Nov 75 to Oct. 76) location of leak detectors as well as required sensitivity will be finalized. 20

Question 001.204(9.3.2.1)

Identify the tritium content limits and plant radiological release criteria.

Response:

As discussed in PSAR section 11.2 and 11.3, the design objectives of CRBR plant operation include levels of radioactive materials in the plant effluents to the environment being kept as low as practicable. Plant radiological release criteria include conformance within the requirements of 10CFR20.

Analysis of CRBR plant design, including accident conditions, are based on conservation estimates of tritium in the primary sodium. Results of the analyses confirm that a stipulation of a tritium content limit is not required.

Analyiis of plant releases in the PSAR section 11.2 and 11,3 show the design tritium releases to be less than 10% of 10CFR20. Results of analysis presented in response to Q 310.11 shows that the activity acsociated with the entire water inventory of the steam water system would result in doses of 20% of 10CFR20 limits.

Question 001.205 (9.3.2.2.2)

Describe the measures available to prevent inadvertent sodium transfer from the primary sodium storage vessels to the EVST and reactor coolant system.

Response:

The measures available to prevent inadvertent sodium transfer are discussed in amended Section 9.3.2.2.2.

Question 001.206 (9.3.4.3)

Provide a description of the measures available to detect and control the in-leakage of NaK into the primary coolant.

Response:

The answer to this question is provided in the revised PSAR Page 9.3-11, Section 9.3.4.3.

Question 001.207 (9.3.5.5)

Provide a complete list of all leak detectors, their location and required sensitivity.

Response:

The information requested is provided in the response to Question 001.203. 20

Question 001.208 (15.1.1.2)

In this Section (15.1.1.2) a maximum probability of failure of 10^{-6}/yr. is assigned to the shutdown system. In Appendix C, arguments are made to show that the maximum tolerable failure probability for the Shutdown System is 1×10^{-7}/yr. Resolve this discrepancy.

Response:

The response to this question has been incorporated into the PSAR by rewriting the last paragraph of Section 15.1.1.2.

Question 001.209 (15.1.1.3.14)

The PSAR states that the Overflow Heat Removal System (OHRS) is not required to function following the third level dynamic-loadings. It appears that the third level margins are simply mechanical and provide no level of assurance that the reactor would be maintained in any safe configuration following a third level event. Explain what means of cooling would be available to the reactor core after a third level event.

Response:

The Direct Heat Removal Service, DHRS (formerly OHRS) has been included in the design to provide a diverse means of heat removal. It provides capability to protect the plant in the highly unlikely event that heat removal through all of the steam generators is impossible as a result of a sequence of failures. For that highly unlikely situation, the DHRS is capable of removing the full decay heat assuming no other heat removal from the time of reactor trip.

The SMBDB margins are provided as additional assurance that the health and safety of the public is protected even in the event of the occurrence of unforseen circumstances. These margins are not associated with a specific event. The specification of SMBDB requirements provides capability for a spectrum of highly improbable circumstances and involves judgment based on test and calculational results together with a knowledge of system behavior. The dynamic loading requirements within the reactor vessel and in other parts of the Primary Heat Transport System assure that the system will remain intact. Since the system elevations have been specified to be consistent with natural circulation, this capability would exist following the dynamic loads. The heat removed through the normal cooling systems would be a function of the availability of flow paths through the reactor. In the postulated event of failure of heat removal through the PHTS following a hypothetical core disruptive accident, the heat is removed through the TMBDB process described in CRBRP-3, Volume 2 (Reference 10b of PSAR Section 1.6).

Question 001.210 (15.6.1.1.2)

Discuss the possibility for liner failure due to steam pressure and/or concrete spalling as a consequence of a sodium spill. Is it possible that this might require the containment to be purged? If so, what are the consequences of the purging on the radiological dose calculations? This same question applies to other accidents in Section 15.6 involving sodium spills in containment structures. Even with a steel liner, the heat of liquid sodium may drive water out of concrete, the water then reacting with sodium to generate hydrogen.

Response:

The possibility of liner failure is discussed in new PSAR Section 3A.8.3.5 Radiological consequences of liner failure are discussed in new PSAR Section 3A.8.3.6. 37

Question 001.211 (15.7.2.5.2)

Since the tritium involved in this event is presented as HTO (liquid) include the degree of contamination of surface and ground waters resulting from a spill. Also include a discussion of the consequences of tritium and HTO leakage from any other sources. Cold traps are a principal reservoir of tritium, as shown in Table 11.1.9. Discuss the possibilities of leakage during operations, servicing, or disposal of their contents.

Response:

New PSAR Section 15.7.2.7 discusses "Leakage From Sodium Cold Traps".

Question 001.212 (15.5.2.3.2)

Provide the basis for limiting the potential source to isotopes of Kr, Xe, I.

Response

The event described in Section 15.5.2.3 of the PSAR considers radioactivity in the gas phase from the following sources:

1. Reactor cover gas

2. Primary coolant vapor

3. Volatile fission products of a reactor core fuel assembly

1. Reactor Cover Gas

The gaseous radionuclide inventory in the reactor cover gas at the design base condition is given in Table 11.3-2 of the PSAR. The following isotopes are listed in this table in addition to Kr and Xe isotopes: Ar^{39}, Ar^{41}, Ne^{23}, and H^3. The activity of these four isotopes at reactor shutdown and 36 hours later are shown in Table 1.

TABLE 1

Isotope	Half-Life	Activity (Ci)	
		0-Hr Decay Time*	36-Hr Decay Time
Ar^{39}	269 yrs.	1.36	1.36
Ar^{41}	110 min.	26.6	3.19×10^{-5}
H^3	12.5 yrs.	3.04×10^{-3}	3.04×10^{-3}
Ne^{23}	38 sec.	8.98×10^5	6.9×10^{-7}

* From Table 11.3-2 of the PSAR

The activities listed above at 36 hour decay time are negligibly small as compared to the activities of the Kr, Xe, and I isotopes shown in Table 15.5.2.3-1. In spite of their small activities, the isotopes Ar^{39} and H^3 were considered in the analysis, and are listed in PSAR Tables 15.5.2.3-2 and 15.5.2.3-3. (They are not listed in Table 15.5.2.3-1 which lists only fuel assembly fission products.) The isotopes Ar^{41} and Ne^{23} were excluded from the analysis because of their negligible contribution to the total EVTM fission gas inventory.

2. Primary Coolant Vapor

The EVTM seals and adjacent surfaces will not reach steadv state temperatures higher than 200°F with either forced or natural convection air cooling. Such low temperature surfaces will cause primary coolant vapor to plate out from the EVTM internal atmosphere before significant diffusion through the seals can occur. (The melting point of sodium is 208°F.)

3. Core Fuel Assembly Fission Products

The entire isotopic content of the equilibrium at the end of cycle is given in Table 12.1-35. Fission products with melting points above 200°F were not considered in the event discussed in Section 15.5.2.3.2 for the same reason as discussed above in Item 2.

The disposition of the volatile fission products with melting points below 200°F is as follows:

(a) Kr, Xe, and I isotopes were all considered in the analysis of the event described in 15.5.2.3.2.

(b) The isotopes of Br, Ga, Rb, and Cs (except Cs^{134}, Cs^{136}, Cs^{137}, and Rb^{86} - see item (c)) - were not considered in the analysis because of either too short half lives, too small activities, or a combination of both. Table 2 below lists the isotopes, their half lives, their average activities in one fuel assembly at reactor shutdown (0 hour decay time), and the reasons for their elimination from the analysis.

(c) The long lived isotopes of Cs and Rb, their half lives, relative amounts, and activities are shown in Table 3. Since the melting temperature of cesium is 83°F, and of rubidium is 102°F (boiling temperatures: Cs - 1265°F, Rb - 1294°F) some cesium and rubidium in the EVTM will be in vapor form. As a maximum EVTM seal temperature of 200°F, the activities of these Cs and Rb vapors are insignificant as compared to those of Xe, Kr, and I.

TABLE 2

Isotope	Half Life	Average Fuel Assembly Activity at 0-Hr Decay Time (Ci)	Reason for Elimination from Analysis	Isotope	Half Life	Average Fuel Assembly Activity at 0-Hr Decay Time (Ci)	Reason for Elimination from Analysis
Br^{80m}	4.38 Hour	4.53×10^{-5}	1	Cs^{141}	24 Sec	1.32×10^{5}	3
Br^{80}	17.6 Min	1.78×10^{-4}	1	Cs^{142}	2.3 Sec	8.60×10^{4}	3
Br^{82}	36 Hour	6.12×10^{1}	1	Cs^{143}	2.0 Sec	4.24×10^{4}	3
Br^{83}	2.41 Hour	1.34×10^{4}	2	Cs^{144}	Short	1.81×10^{4}	3
Br^{84M}	6 Min	5.31×10^{2}	2	Ga^{76}	32 Sec	8.97	4
Br^{84}	31.8 Min	2.04×10^{4}	2	Rb^{86M}	60 Sec	41.7	4
Br^{85}	3.0 Min	2.68×10^{4}	2	Rb^{88}	17.7 Min	5.61×10^{4}	5
Br^{86}	54 Sec	3.42×10^{4}	2	Rb^{89}	15.2 Min.	7.19×10^{4}	3
Br^{87}	55.6 Sec	4.01×10^{4}	3	Rb^{90}	2.9 Min	8.22×10^{4}	3
Br^{88}	15.5 Sec	3.96×10^{4}	3	Rb^{91M}	72 Sec	9.71×10^{4}	3
Br^{89}	4.5 Sec	3.55×10^{4}	3	Rb^{91}			
Br^{90}	1.6 Sec	2.48×10^{4}	3	Rb^{92}	4.48 Sec	1.02×10^{5}	3
Cs^{134M}	2.9 Hour	5.71×10^{2}	4	Rb^{93}	5.87 Sec	9.12×10^{4}	3
Cs^{138}	32.2 Min	2.12×10^{5}	3	Rb^{94}	2.67 Sec	8.65×10^{4}	3
Cs^{139}	9.5 Min	1.86×10^{5}	3	Rb^{95}	.36 Sec	4.85×10^{4}	3
Cs^{140}	66 Sec	1.62×10^{5}	3	Rb^{97}	.14 Sec	1.98×10^{4}	3

Key to "Reason for Elimination from Analysis":

1. Small activity compared to that of Kr, Xe, I.
2. Short half life; also produced as daughter of Se isotope with very short half life. Activity insignificant at 36-hour decay time.
3. Short half life; activity insignificant at 36-hour decay time.
4. Combination of small activity and short half life; activity insignificant at 36-hour decay time.
5. Short half life; also produced as daughter of Kr^{88} with short half life. Activity insignificant at 36-hour decay time.

TABLE 3

Isotope	Half Life	Average Fuel Assembly Activity at 36-Hour Decay Time (Ci)	Approximate Relative Amount of Each Isotope Present at 36-Hour Decay Time (%)	Activity of Vapor* at 36-Hour Decay Time (Ci)
Cs^{133}	Stable	0	35	0
Cs^{134}	2.05 Yrs	1.40×10^3	29	1.8
Cs^{136}	13 Days	3.66×10^3	1	4.6
Cs^{137}	30 Yrs	7.20×10^3	35	0.15
Rb^{85}, Rb^{87}	Stable	0	99.94	0
Rb^{86}	18.66 Days	3.93×10^2	0.06	0.073

* Based on vapor pressure of Cs and Rb at the maximum EVTM seal temperature of 200°F.

Question 001.213 (15.5.2.3.2)

Provide the calculations leading to the assumed permeabilities.

Response:

The requested calculations are provided in revised Section 15.5.2.3.2. 25

Question 001.214 (15.5.2.3.2)

Provide an analysis of the release of all volatile fission products, assuming failure of the elastomer seals and loss of EVTM cooling.

Response:

This question appears to be motivated by a concern that there might be a common mode failure due to the loss of EVTM cooling that could result in the release of all volatile fission products and failure of the seals. Such a common mode failure is not considered to be a credible accident, as discussed below.

The EVTM cold wall is designed to remove decay heat from 20 Kw spent fuel assemblies. As discussed in detail in PSAR Section 9.1.4.3, heat from the cold wall is removed by forced air flow, provided by an air blower. In the event of a complete loss of forced air cooling, the cooling mode is automatically and inherently changed from forced to natural convection. This change is effected by fail safe controls, not by operator action. The maximum fuel cladding temperature of a 20 kw spent fuel assembly in a CCP in the natural air convection cooling mode was calculated to about 1500^{o}F. At this cladding temperature, only random fuel rods would be expected to release fission gas into the EVTM. The accident discussed in Section 15.5.2.3 conservatively, assumed fission gas release from all 217 fuel rods in the EVTM.

All steady-state seal temperatures in the EVTM during natural convection are below 200F, which is less than the upper limit of 350^{o}F considered as detrimental to the integrity of elastomeric seals. This is achieved by protecting seals from direct heat radiation, placing them in areas benefiting from convective air cooling, and providing a geometry with high thermal resistance and heat capacity between the heat source and the seals.

It is, therefore, concluded that loss of forced air cooling cannot lead to a common mode failure resulting in the sudden release of volatile fission products from the EVTM.

A potential mechanism of fission gas leaving the EVTM is by slow diffusion through the seals. The radiological consequences of such an event were analyzed in Section 15.5.2.3.2 and resulted in dose rates less than the limiting values. As described in Section 9.1.4.3 of the PSAR, all surfaces in the EVTM to be sealed against radioactive gas are provided with at least two seals in series with pressurized gas between them. The seals will be leak tested before reactor refueling operations are initiated. As discussed in the response to Question 310.22, leak testing of the EVTM will be required by a technical specification. This specification will assure that accidental releases of fission gas from the EVTM are below the limiting values.

Question 001.215 (15.5X)

Provide an analysis of the effect of a failure of the periscope/ seals in any of the in-containment cells (EVST cell, FHC, RC, PHTS cells).

Response:

Of the four cells mentioned in the question, only two (RC, PHTS cells) are in-containment cells. However, all of the cells mentioned are addressed below.

General

The EVST cell, the RC cell, and the PHTS cells are all inerted with a nitrogen - 2% oxygen atmosphere which is operated at a slight negative pressure of -2 inches water gauge. The FHC is inerted with continuously purified argon and is operated at a vacuum of -3 inches water gauge. The purpose of the negative pressure operating feature is to ensure that leaks will be inward. Even ignoring this mitigating feature of CRBRP design, the following discussions show that postulated cell leakage would result in acceptable off-site consequences.

EVST Cell (No Periscope)

The nitrogen in the EVST cell is normally not expected to be radioactive. Its purpose is to cool the cell walls and to inert the cell to minimize sodium fire in the event of pipe or tank leakage. As described in Section 9.5.2, a radioactivity sampling system periodically samples each nitrogen-inerted cell and analyzes the cell atmosphere for radioactivity. In the highly improbable event of a failure of the omega seal and the presence of some radioactivity in the nitrogen gas, the resulting release of radioactivity to the outside environment via the RSB/RCB ventilation system would be significantly smaller than the umbrella event analyzed in Section 15.5.2.4 (Reactor Cover Gas Release)

Fuel Handling Cell (FHC)

As described in Section 9.1.2.2, all elastomeric seals in the FHC are double and periodically leak checked with pressurized, clean argon gas. A single seal failure would result in no release of radioactivity.

As a further safety measure, the radioactivity of the FHC atmosphere will be monitored and controlled to a low level, such that its instantaneous release to the RSB operating area by a hypothesized accident will result in a site boundary dose which is less than the 10 CFR 20 guidelines (see Section 16.3.10).

The sealed scanning periscope installed in the FHC will be a commercially available instrument, designed for use in alpha contaminated hot cells. It consists of: (1) a gas-tight sleeve which is inserted in and sealed to a FHC wall penetration, (2) a gas-tight periscope which is inserted into the sleeve from the operating gallery, and (3) a shield plug which is inserted into the sleeve when the periscope has been removed.

The in-cell end of the sleeve contains a glass dome which permits viewing into the cell. The horizontal tube annulus between periscope and sleeve is sealed at both ends and filled with pressurized argon gas. The operating gallery side of the sleeve is provided with double seals buffered with pressurized argon gas. The detachable vertical leg of the periscope in the operating gallery is mechanically attached to the horizontal tube assembly, but the atmospheres of the two parts are separated.

A failure of an inner periscope seal or a breakage of the glass dome would result in argon gas flow into the FHC. The failure of an outer periscope seal would lead to argon discharge into the operating gallery. No release of radioactivity from the FHC would occur in either case.

PHTS Cells/Reactor Cavity (RC)

The atmospheres of the PHTS Cells and RC are periodically bled to CAPS for removal of radioisotopes. Sound design and administrative controls ensure that periscope openings and/or latch seals will not be opened while substantial radioactivity is present in the cell atmosphere. However, in the highly improbable event that a seal leak does occur, the release would be less than that discussed in response to NRC question 001.258 which is shown to result in site boundary doses well below the guidelines of 10CFR100.

Question 001.216 (15.6)

Provide revised analyses in this section using the results of on-site meteorological data, as presented in Section 2 of the PSAR.

Response:

During a telephone conference between the CRBRP Project and NRC on January 6, 1976, agreement was reached on an acceptable interim calculation of χ/Q using available on-site meteorological data. The χ/Q will be based on $\Delta T/\Delta Z$ between elevations of 200 and 75 feet and wind speed and direction measured at 33 feet.

The project will use the calculated interim χ/Q to update all affected sections of the PSAR by 4/15/76.

Question 001.217 (15.6.1.1)

State the maximum sodium temperature permitted in the primary sodium in-containment storage tank during such time when the storage tank cell may be open.

Response:

The response to this question has been incorporated into the PSAR by the revision of the first two paragraphs of Section 15.6.1.1.1.

Question 001.218 (15.6.1.1.2)

Justify the use of the SOFIRE-II pool burning code for the initial stages of the 32,000 gallon spill.

Response

The postulated event evaluated in Section 15.6.1.1 is the Extremely Unlikely failure of the Primary Sodium In-Containment Storage Tank During Maintenance. The potential consequences of this failure are conservatively assessed assuming instantaneous and complete tank failure such that the entire Na inventory of the tank (32,000 gallons) is immediately spilled on the tank cell floor.

The tank is used to store sodium coolant in the event maintenance activities require drainage of a portion of the primary system. The sodium in the tank is essentially a stagnant pool under very low cover gas pressure and thus any foreseeable failure, such as the cracking of a fill or drain line, would result in only a slow leakage of sodium and a spill orders of magnitude less than the assumed 32,000 gallons. Further, the tank is mounted near the lowest level of the Overflow Vessel/Storage Tank Cell, with a floor clearance of approximately 2 feet. The proximity of the tank to the cell floor coupled with the low operating pressure of the tank precludes releases, due to postulated failures, characteristic of sodium sprays. A sodium spray requires the pressurized discharge of a sodium stream, impingement of the stream on a structural surface, stream breakup leading to droplet formation characteristic of sodium sprays, and finally the free-fall of these droplets allowing interaction of the sodium droplets and the atmosphere.

To summarize, because of the design, location, and operating pressure of the tank, no mechanism resulting in the pressurized discharge of sodium, in the event of postulated tank failure, exists. The expected release mode, in the event of tank failure, is a slow gravity drain of the contained sodium to the cell floor, approximately 2 feet beneath the tank. However, the potential consequences of such a tank failure were assessed assuming the immediate release of the total tank inventory to the cell floor and thus the SOFIRE-II pool fire analyses presented in Section 15.6.1.1.2 are judged to conservatively bound the consequences of postulated tank failures.

Question 001.219 (15.6.1.1.2)

Justify the assumptions that, (i) no isotopes would have greater concentrations in the aerosol than in the pool and, (2) that fission products will remain associated with airborne sodium.

Response:

In the sodium fire analyses of Section 15.6 of the PSAR, it is assumed that the concentration of fission products in the sodium aerosol is the same as that in the sodium pool. The assumption is conservative when applied to non-volatiles. Experimental work at Atomics International (AI)(Ref. Q001.219-1) was conducted in which samples of sodium containing uranium were burned in a 7.4 liter test chamber containing air. The release fraction of sodium from the samples was less than 5%, but the release fraction of uranium was 0.1% in one case, and in the other case, was too small to be detected. Regarding plutonium release, it was inferred(Ref. Q001.219-2) that from the results of Chatfield(Ref.Q001.219-3), the combustion of sodium releases only about 10^{-5} of the plutonium in the sodium. Applying these results to non-volatiles in general indicates that the assumed concentrations of radioisotopes in airborne sodium would be considerably higher than the actual concentrations. Reference may also be made in Figure 1 in the article by Castleman (Ref.Q001.219-4), concerning fission-product behavior in sodium. The figure indicates the relative volatilities of several fission products with respect to sodium. It can be seen that for the non-volatiles, strontium and barium, even bulk boiling of the sodium pool would release small percentages of these radioisotopes for large percentages of released sodium.

On the basis of recent experimental evidence, the aerosol concentrations of iodine and, in all likelihood, volatile fission products in general, will be greater than that in the sodium pool. Experimental results by R. Koontz, et.al.(Ref.Q001.219-5), of AI, and S. Kilani (Ref.Q001.219-6) of Japan, indicate aerosol concentrations which are 2-3 times higher than the concentrations in the pool. Assuming comparable behavior of volatile solid fission products and iodine, it was conservatively assumed on the basis of this experimental data, that the radioisotopic concentrations in the aerosol are 3 times the concentration in the sodium pool for iodine and volatile sold fission products. In order to assess the impact of this change on the original dose calculations in the PSAR, additional dose calculations have been performed. The modified thyroid and lung doses, due primarily to the increased release of iodine and cesium, respectively, are given in the response to Question 310.12 to the PSAR. It should be noted that these doses are still well below maximum allowable guideline limits of 300 Rem thyroid and 75 Rem lung.

For these same sodium fires analyses, it was also assumed that the fission products released from the sodium pool remained associated with the sodium aerosol for the duration of the accident. R. Koontz and his associates at AI observed, as a result of their sodium burning experiments with subsequent release of I-131, that "...the I-131 and sodium agglomerated together and remained together throughout the time when plating and settling removed the aerosol from the air".(Ref.Q001.219-5) Keilholtz and Battle, in their paper on fission product transport in liquid metal fast breeder reactors, stated that although fission product gases, primarily the noble gases, would remain independent of the sodium aerosol during a sodium fire, large fractions of most radioactive materials might be reacted with occluded by, absorbed in, or adsorbed on the sodium oxide aerosols (Ref.Q001.219-7) Experiments on fission product release from burning sodium have been performed by C. Descamps, et.al., of Belgium (Ref.Q001.219-8). In their experiments, a capsule containing preheated sodium is broken, and the sodium discharged into a large vessel filled with atmospheric air. At the same time, the released sodium is placed in contact with molten irradiated uranium oxide. The evolved aerosols are then analyzed for determination of radioisotopic content. On the basis of his results, Descamps concluded that the particles of sodium oxide capture the fission products and keep them in the gaseous (aerosol) phase. In Table 3 of his paper (Ref.Q001.219-8) are tabulated the percentage of fission products in the aerosols as a function of time. It can be seen that the fractional decrease of fission products in the gases with time is roughly equivalent to the fractional decrease of sodium aerosol with time. This implies that the fission products tend to adhere to the sodium oxide aerosol and fall out as the aerosol falls out. Therefore, the assumption that the fission products remain associated with airborne sodium is considered appropriate.

References

Q001.219-1 R.L. Koontz and D. Tappen, "Radionuclide Release from Burning Sodium" presented at the Safety Technology Meeting on Radiological Assessment, July 30-31, 1975, p. 9.

Q001.219-2 Ibid., p. 4

Q001.219-3 E.J. Chatfield, Journal of Nuclear Materials, 32, 228 (1969).

Q001.219-4 A.W. Castleman, Jr., "LMFBR Safety, I. Fission Product Behavior in Sodium", Nuclear Safety 11, 379, 1970.

Q001.219-5 R.L. Koontz and D. Toppen, "Radionuclide Release from Burning Sodium" presented at the Safety Technology Meeting on Radiological Assessment, July 30-31, 1975.

Q001.219-6 LMFBR Japanese and American Information Exchange Meeting on LMFBR Safety, held at Beverly Hilton Hotel in Los Angeles, April 1, 1968.

Q001.219-7 G.W. Keilholtz and G.C. Battle, Jr., "Fission-Product Release and Transport in Liquid Metal-Cooled Fast Breeder Reactors", Nuclear Safety, 9(6), November-December, 1968, pp. 494-509.

Q001.219-8 C. Descamps, et.al., "Behavior of Several Fission Products in the Atmosphere Surrounding Sodium-Cooled Reactors in the Event of a Major Accident", in "Proceedings of the International Conference on Sodium Technology and Large Fast Reactor Design", November 7-9, 1968, USAFC Report ANL-7520, Pt. 1, pp. 555-561, Argonne National Laboratory, 1968.

Question 001.220 (15.6.1.1.2)

Justify the use of a different leak rate than that set forth in Regulatory Guide 1.4, staff position 1.e.

Response:

The Regulatory position set forth in Regulatory Guide 1.4 is that leakage from the reactor containment to the environment should be assumed to occur at the design leak rate of the containment for the first 24 hours following an accident and at 50% of the design leak rate for the remaining duration of the accident. Justification for the use of a different leak rate is in revised Section 15.6.1.1.2.

The potential consequences of this accident have been re-evaluated assuming containment leakage at its design rate, 0.1% Vol/Day, for the duration of the accident. Note that this leak assumption is even more conservative than the Regulatory Guide assumption, which allows for reduced leakage beyond 24 hours. The results of this re-evaluation have been previously provided in response to NRC question 310.21. The results of this re-evaluation indicated large margins (greater than a factor of 10^5) between each of the potential off-site doses and the applicable guideline limits. 25

Question 001.221 (15.6.1.3.2)

Justify the use of SOFIRE-II in lieu of SPRAY-1 for this accident.

Response:

(Because Q001.222 refers to the same PSAR Accident, 15.6.1.3.2 Failure of the Ex-Containment Primary Na Storage Tank, and requests the same information, i.e., justification of SOFIRE-II, this question has been interpreted as requesting justification of SOFIRE-II vs SPRAY-1 for PSAR Accident 15.6.1.2.2, EVST Na Cooling System Failure). This information is found in revised Section 15.6.1.2.2. 25

Question 001.222 (15.6.1.3.2)

Justify the use of SOFIRE-II for the initial stages of the 90,000 gallon spill.

Response:

Revised Section 15.6.1.3.2 contains the justification for the use of SOFIRE-II for the initial stages of this event. 25

Question 001.223 (15.6.1.4.2)

Provide the value for assumed mass release per mass burned.

Response:

Section 15.6.1.3.2 was modified to answer this question.

Question 001.224 (15.6.1.4.2)

Provide a detailed description of the SPRAY 1 Code.

Response:

The description of the SPRAY 1 Code is found in references 4-6, Section 15.6.1.4. 25

Question 001.225 (15.6.1.4.2)

Provide the calculations used to derive the estimates of 1500 and 18,000 gallon spills.

Response:

The requested calculations are in Revised Section 15.6.1.4.2. 25

Question 001.226 (15.6.1.4.2)

Justify the use of SOFIRE II to estimate the consequences of sodium draining from elevated piping onto a floor.

Response:

The sodium fire analysis presented in Section 15.6.1.4 consisted of both a spray and pool component. As pointed out in Section 15.6.1.4, a sodium spray requires the pressurized discharge of a sodium stream, impingement of the stream on a structural surface, stream breakup leading to droplet formation characteristic of sodium sprays, and finally the free-fall of these droplets allowing interaction of the sodium droplets and the cell atmosphere.

The spray fire component analyzed (SPRAY-1) in Section 15.6.1.4 was evaluated as the pressurized discharge of 30 gpm (100% of which was assumed to react as a sodium spray) for a 10 minute duration. Section 15.6.1.4.2 has been revised to justify the use of SOFIRE II. 25

Question 001.227 (15.6.1.4.2)

Provide a list of the principal parameters used in the HAA-3 calculations.

Response:

The response to this question has been incorporated into the PSAR by the addition of new material to Section 15.6.1.4.2

Question 001.228 (15.6.1.4.2)

Provide the analyses of system pressures using the DEMO and SPRAY codes identified on page 15.6-23.

Response:

The analyses referenced on page 15.6-23 of the PSAR were based on preliminary assessments of 1) PHTS cell maximum pressures versus sodium leak rate and 2) Reactor Vessel inlet plenum minimum pressures versus sodium leak rate. The cell pressures were computed with the SPRAY-1 computer code and the plenum pressures with the DEMO computer code. New Figure 15.6.1.4-10 presents the results of these analyses. The two cell pressure curves which are included in the Figure are one for hot leg (1015°F) leaks and another for cold leg (730°F) leaks. Each pressure curve was obtained based on a leak duration of 10 minutes and the assumption that the total leakage reacts as a sodium spray. The curves represent the maximum cell pressure resulting from the indicated leak rate, for hot and cold leg leaks.

The inlet plenum pressure curves provide the minimum inlet plenum pressure during the pipe leak transient. The minimum inlet plenum pressure is a function of the leak rate and the leak location. The dependence of the minimum plenum pressure on these parameters is shown in the Figure.

Question 001.229 (15.6.1.4.2)

Provide the analyses supporting the claimed 24 psig capability of the cells.

Response:

The information requested is provided in revised section 15.6.1.4.2. 25

Question 001.230 (15.6.1.4.2)

Provide the estimates of sodium vapor assumed to be generated as a result of the PHTS leaks.

Response:

The amount of sodium vapor generated is assumed to be negligible as discussed in revised Section 15.6.1.4.2.

Question

001.231 (15.6.1.4.2) *Provide the cell leak rate assumed in the analyses of the PHTS and RC cell pressures.*

Response

For the calculation of both the primary heat transport system and reactor cavity cell pressure transients, the cells were assumed leak tight; i.e., no escape of cell atmosphere or combustion product heat as a result of leakage. This assumption is conservative and maximizes the pressure transients in the cells.

For radiological considerations, the cells were assumed to have an infinite leak rate; i.e., all sodium aerosol was released instantly from the cells to the upper RCB volume. This assumption is conservative and does not take credit for retention, holdup, settling or plate-out of aerosol in the cells.

Question 001.232 (15.6.1.5.2)

Provide the results of the calculations of cell temperatures and pressures.

Response:

The information requested is provided in revised Section 15.6.1.5.2. 25

Question 001.233 (15.6.1.5.2)

Provide estimates of the airborne concentrations of sodium oxides at the site boundary, and low population zone.

Response:

The non-radiological assessment of the impact of nuclear power plant operations are requested specifically in the content of Environmental Reports (Reg. Guide 4.2 Rev. 1, Jan. 1975). The Project has provided such information in Section 7.2 of the ER for a spectrum of potential sodium releases from the CRBRP. The analyses and results which follow should be interpreted as supplementary in nature to that information and consistent with the approaches therein.

Briefly summarizing, Section 15.6.1.5.2 of the PSAR has conservatively
postulated an IHTS leak in which approximately 2500 pounds of sodium is
burned. It is additionally assumed that 27% of the sodium which burns
becomes airborne in the Steam Generator Building (Ref. Q001.233-1).
39 All generated aerosol is released to the outside environment during
the eight minutes of cell overpressurization.

An assumption of 50% meterology has been used consistent with the general
39 approach of Section 7.2 of the Environmental Report. In addition, a
depletion factor credit of 100 has been applied for sodium hydroxide
in its transit from release to arrival at the site boundary. This
factor will conservatively account for expected (a) significant fallout
of the sodium oxide and (b) conversion of sodium oxide to carbonate
form (Ref. Q001.233-2) well before reaching the site boundary. This
same factor was also applied to the calculated concentration of sodium
39 hydroxide at the LPZ.

39 The resulting concentrations wil- be 2.4 mg/m^3 at the exclusion boundary
and 0.63 mg/m at the LPZ.

References:

Q001.233-1 "Summary Report for Laboratory Experiments on Sodium Fires", I. Baurmash, R. P. Johson, R. L. Koontz, C. T. Nelson, August, 1973, AI-TR-707-130-007.

Q001.233-2 Clough, W. S. and Garland, "Behavior in the Atmosphere of the Aerosol from a Sodium Fire", Journal of Nuclear Energy, Vol. 25, pps 425-435, 1971.

Question 001.234 (15.6.1.5.3)

Summarize the current IHTS cell design bases (including fire suppression systems) and provide those design bases which must be included in consideration of this event.

Response:

Work is currently in progress to define the design basis sodium leak rate for the IHTS cells. A catch pan system is being designed which will accommodate the total available sodium spill volume as summarized in Table 001.234-1. The catch pan system is being designed to allow for overflow to other cells where the spill volume exceeds the pan volume in the cell where the spill originates.

Work is currently in progress to determine where fire suppression systems are required and which type of fire suppression system would be appropriate (i.e., fire suppression decks, nitrogen inerting, etc.).

When these tasks are completed, the IHTS cell design bases can be finalized. The cell design temperatures, pressures, fire suppression capability and other data required to design the cells will be specified.

TABLE Q001.234-1

Summary of Design Requirements For Sodium Spills in IHTS Cells[2]

Cell[1]	Spill, Pounds	Spill Volume, Cu. Ft.
<u>LOOP #1</u>		
227	265,000	5000
231	265,000	5000
251	200,000	3800
207	265,000	5000
224	265,000	5000
<u>LOOP #2</u>		
228	200,000	3800
229	200,000	3800
234	180,000	3400
248	175,000	3300
208	200,000	3800
225	200,000	3800
<u>LOOP #3</u>		
230	275,000	5200
232	275,000	5200
252	200,000	3800
209	275,000	5200
226	275,000	5200

(1) Cell numbers identical to those shown in PSAR Figures 1.2-17 through 1.2-22.

(2) Average sodium temperature assumed to be 800°F in all cases.

Question 001.235 (15.6.1.5.3)

Provide a description of the design changes necessary to adopt either option.

Response:

Paragraph 15.6.1.5.3 of the PSAR states that during the spray phase of the assumed large spill accident and subsequent spray fire, the excessive pressure and temperature buildups can be accommodated by either (1) venting at 700,000 CFM or (2) strengthening the cell walls.

The project is currently defining a design basis leak for the IHTS cells. The resulting cell design pressures and venting rates (if venting is necessary) are expected to be much lower than indicated above. These revised values will be used for design. An evaluation of the cells will be made, however, for a large rupture. The evaluation of the cells for the large rupture will utilize different acceptance criteria than if the large break was the design basis. The cell walls will be evaluated structurally for limiting loads, not design loads. The limit loads for existing cell designs are higher than the design loads. An evaluation of the cells is expected to show that the cells have the capacity to withstand the pressures determined in the evaluation without modification or venting. If necessary, however, the walls can be strengthened by several means. The thickness of the walls can be increased to provide additional strength against pressure loading. A wall thickness increase of approximately $\sqrt{2}$ times the present wall thickness will double the pressure it can withstand. Outside walls can be increased without affecting cell volumes. For inside cells, a slight increase in building dimension will be required if cell volumes are to remain unchanged.

Another method of increasing the strength of the cell walls is to increase the amount of reinforcing bars in the wall. There is a limit to the strength increase gained by this method, but increasing the reinforcing bars may be adequate for inner cells thus avoiding changing building dimensions.

Question 001.236 (15.6)

(a) Provide a brief summary of all the sodium spill cases that have been examined in addition to those reported.

(b) Identify the conservative assumptions used throughout the analyses of sodium spills and quantify these conservatisms to establish the basis and level of confidence in the predicted results.

Response:

(a) A spectrum of postulated sodium spill cases has been investigated. The design basis sodium leak for inerted cells is a maximum of 8 gpm. This leak has been analyzed for a spray fire and shown to cause a pressure rise of less than 2 psi in a PHTS cell and a corresponding temperature increase of 54°F. In order to provide an assessment of the design margin, evaluations are being made considering larger leak rates.

Additional cases, that have been evaluated to determine the margin that exists in the design but were not reported in the PSAR, are outlined in Table Q001.236-1. Minor changes in RCB cell volumes have occurred due to design evolution since the analyses were completed. As a result, the new peak cell pressures will increase by a fraction of a psig which is not significant. The preliminary results are reported in Table Q001.236-2 to provide an interim basis for evaluation of the design.

For each cell, a piping or component leakage point was assumed at the location and of the form which provided the worst case condition. If the operating pressure of the piping or component considered was sufficiently high to result in a spray release mechanism, a spray transient was analyzed in addition to the pool transient. For pool transients, the leak was assumed to occur at the location which maximized sodium spillage. Conservative assumptions were consistently applied in the analysis. For example, 1) the initial sodium spill temperature was assumed equal to the maximum operating sodium temperature in the system, 2) for failures resulting in the pressurized discharge of sodium, 100% of the discharge was assumed to react as a spray during the time that the pressurized conditions existed, and 3) all sodium combustion was assumed to result in the formation of sodium monoxide, which maximizes heat generation per pound of sodium burned. No credit was taken for mitigating the spray effects by the multiple layer steel clad insulation

Table Q001.236-2 summarizes the results of each of these cases. The radiological consequences are enveloped by the cases included in the PSAR.

(b) Two basic types of conservatism are associated with the analyses of sodium spill cases presented in the PSAR. The first type of conservatism is related directly to the analytical models used to evaluate these cases; specifically SOFIRE-II and SPRAY-I (see Appendix A of the PSAR). Experimentally determined pressure and temperature transients resulting from sodium fire tests have been successfully predicted with SOFIRE-II, thus verifying the adequacy of the code. PSAR Section 15.6 provides a detailed discussion of the conservatisms associated with the SPRAY-I analyses as well as quantification of the magnitude of these conservatisms.

The second type of conservatism is associated with the more general assumptions applied to the sodium fire analyses and include the following:

1) Non-gaseous radioactive aerosols generated during sodium combustion are assumed a) to be released directly to the environment if the fire takes place in a cell/building not specifically designed to some leak tightness requirement, or b) to leak from the containment to the environment as gases. In either case, the expected attenuation of the aerosol via plugging and settling in leakage paths is conservatively neglected. This assumption is judged to result in an overestimate of aerosol release.

2) Fallout (cloud depletion) of the aerosol during transfer down-wind is conservatively neglected.

3) Radioactive decay of the radioactive sodium aerosol and associated radionuclides is conservatively neglected.

4) The radioactive content of the sodium is based on continuous plant operation for 30 years with 1% failed fuel (the design basis failed fuel fraction). Fission products and fuel released to the sodium from this failed fuel are assumed present in the sodium for the accident analyses. The expected failed fuel fraction is approximately a factor of 10 less than the design basis fraction. The use of the design basis failed fuel fraction provides a conservative estimate of the potential radionuclide releases associated with sodium fires.

5) Each sodium spill is evaluated based on an initial sodium temperature equal to the maximum temperature anticipated for the system or component whose failure results in the spill considered.

6) Each sodium spill is conservatively evaluated assuming no credit for the Sodium Fire Protection System, which provides the means of detecting, containing, and extinguishing sodium fires.

Based on the consistent application of conservative assumptions, as noted above, and the demonstrated validity of the computer models used for the sodium fire analyses, a sufficient degree of confidence exists that the consequences of the fires are conservatively represented by the predicted results.

Additional spill cases are being performed, including consideration of sodium-concrete reactions. These additional cases will be reported in response to Question 001.581.

Table Q001.236-1

Summary of Additional Sodium Spill Cases

Margin Evaluation

Cell Description	Cell Volume	Volume of Sodium Pool	Sodium Spray Rate	Sodium Temperature
Cell 107 A&B Auxiliary Liquid Sodium System Pipeway and Valve Gallery	41,000 ft^3 (40,967)	13,800 gal	460 gpm	880^{0}F
Cell 157 A,B,D&E Primary Cold Trap and Valve Gallery	11,330 ft^3 (11,329)	1,050 gal	70 gpm	880^{0}F
Cell 104 EM Pump Cell	4,000 ft^3	13,800 gal	460 gpm	880^{0}F
Cell 103 EM Pump Cell	3,400 ft^3	13,800 gal	460 gpm	880^{0}F
Cell 143 Plugging-Temperature Indicator Cell	1,500 ft^3 (1,487)	220 gal	15 gpm	850^{0}F
Cell 141 Plugging-Temperature Indicator Cell	1,220 ft^3 (1,223)	220 gal	15 gpm	850^{0}F

Note: (1) The numbers in parentheses result from a change in cell designation design evolution.

Table Q001.236-2

Summary Results of Sodium Spill Analyses

Margin Evaluation of Events Beyond the Design Basis

	SPRAY		POOL					
Cell Number	Peak Gas Temp. °F	Peak Gas Pressure psig	Maximum Floor Liner Temp. °F	Maximum Floor Concrete Temp. °F	Maximum Wall Liner Temp. °F	Maximum Wall Concrete Temp. °F	Maximum Cell Gas Temp. °F	Maximum Cell Gas Pressure psig
107	689	14.9	750	290	300	187	430	9.0
157	400	5.3	535	149	170	119	205	4.6
104	659	14.3	880	407	435	305	635	13.5
103	719	15.9	880	412	435	310	650	14.0
143	-	-	580	160	180	122	370	7.8
141	-	-	600	177	190	127	385	8.1

Question 001.237 (15.6)

Provide the reference containing the test data on the combustion of sodium, the products to be expected, the rates of peroxide and monoxide formation, etc.

Response:

The following references provide test data on the characteristics of sodium fires. These references have been previously identified in Section 6.2 of the PSAR.

- AI-AEC-13060, "Quarterly Technical Progress Report, LMFBR Safety Program, January-March 1973", May 15, 1973.
- AI-AEC-13055, "SOFIRE II User Report", March 30, 1973.

Question 001.238 (15.6)

Provide the results of calculations on the basis of the maximum possible heat release (no peroxide formation during the combustion of Na but only the formation of the monoxide). Identify the reference.

Response:

With the exception of the "Primary Sodium In-Containment Storage Tank Failure During Maintenance" (Section 15.6.1.1), the sodium pool fire accidents analyzed on Section 15.6 were evaluated based on the assumption of formation of 100% sodium monoxide during combustion.

As explained in Section 15.6.1.1 and more fully in revised Section 6.2.1.3, the consequences of the "Primary Sodium In-Containment Storage Tank Failure During Maintenance" were evaluated based on the formation of 60% sodium monoxide and 40% sodium peroxide during combustion. As pointed out in Section 6.2.1.3 these oxide fractions are based on experimental data which show that during the early stages of sodium pool combustion, sodium monoxide formation predominates, but thereafter, as the surface of the sodium pool becomes less available, due to oxide crusting, peroxide formation predominates. (Reference Q001.238-1 and Q001.238-2). The assumption of no peroxide formation is overly conservative, since test data support peroxide formation.

Neglecting experimental evidence to the contrary, the pressure and temperature transients resulting from the "Primary Sodium In-Containment Storage Tank Failure During Maintenance" have been re-analyzed based on the assumption of only sodium monoxide formation during combustion.

The principal conclusion of this analysis is that, even for the overly conservative assumption of 100% monoxide formation, the pressure and temperature transients imposed on containment are within design capabilities. For example, the peak containment pressure is 1.8 psig, more than a factor of 5 less than the containment design pressure of 10 psig. The maximum containment wall temperature is 194°F, 56°F below the containment wall design temperature of 250°F.

In performing the re-evaluation discussed above, an error in the original (60% monoxide/40% peroxide) PSAR analysis was discovered which resulted in the analysis being overly conservative. SOFIRE-II input parameter XWSE (See Tables 6.2-2 and 6.2-2A) was incorrectly defined as 0.28 feet rather than 0.028 feet due to a key punching error. The effect of this error was to represent the thickness of insulation applied to the containment shell as approximately 4 inches rather than the correct value of 1 inch. The effect of the increased insulation thickness was to severely retard the rate of heat rejection from the containment shell to the environment. With the revised value of 1 inch thick insulation, the peak containment pressure is 1.2 psig and the maximum containment wll temperatiure is 165°F. The analyses presented above are for comparison purposes to assess the differences with and without peroxide formation. The analyses did not model the confinement/containment design. Accordingly, the containment design basis event will be revised to model the confinement/containment design with the results contained in a future amendment to the PSAR.

References

Q001.238-1 AI-AEC-13055, "SOFIRE-II User Report", March 30, 1975

Q001.238-2 AI-AEC-13108, "Annual Technical Progress Report: LMFBR Safety Programs, Government Fiscal Year 1973" August 20, 1973.

Question 001.239 (15.6)

What is the level of confidence in the results obtained with SOFIRE II? Have any verification tests been made for this Code? How sensitive are the results and conclusions to the calculated peak pressure and time in which this peak is reached?

Response:

As reported in Reference Q001.239-1 and explained in Section 6.2.1.3 and Appendix A of the PSAR, an extensive sodium fire test program, used to support SOFIRE II Code development, provided sufficient confidence that the code conservatively predicts pressures and temperatures resulting from postulated sodium pool fires. The code has been verified by using it to analyze the test data as reported in Reference 2, 3.

The design related conclusions of the sodium fire analyses could be sensitive to the calculated peak pressure values, but the time at which this peak is reached is not very significant. The calculated peak pressure values from the postulated sodium spill events are summarized in the Table 15.6-1 of the PSAR. From this, it can be seen that the peak pressure values computed with appropriate conservative assumptions are well within the limits of design pressure values. Further, there is no identifiable mechanism by which these peak pressure values could be increased significantly. Therefore, the sensitivity of the conclusions on design adequacy presented with respect to the peak pressure values is minimal.

Again, as mentioned above, there are no direct design requirements that are related to the time at which the peak pressures are reached. Further, as can be seen from the Table 15.6-1 of the PSAR, the computed maximum off-site doses are well below the guideline limits and any changes in these dose calculations due to the changes in the time history of the pressure pulses would be small. Therefore, the sensitivity of the design related conclusions with respect to the pressure pulse history is not very significant.

Reference 001.239-1: AI-AEC-13060, "Quarterly Technical Progress Report, LMFBR Safety Program, January-March 1973," May 15, 1973.

Reference 001.239-2: AI-AEC-13055, "SOFIRE II User Report", March 30, 1973.

Reference 001.239-3: B.U.B. Sarma, et.al., "Review of Sodium Fire Analytical Models", NEDM-14053, FBRD, General Electric Co., Sunnyvale, Calif., June 1975.

Question 001.240 (15.6)

Justify the differences in calculated results noted for the various sodium spills in terms of pressure and temperature histories, doses, etc.

Response

The pressure and temperature responses of a cell/building to a sodium spill and accompanying combustion are dependent upon several factors, which include: surface area of the sodium pool, initial sodium temperature, sodium pool depth, initial oxygen content of the cell atmosphere, and cell volume. A pool with a small surface area would give off less heat per unit time (from both combustion and sensible heat) than a pool with larger surface area, all other factors equal. The lower heat addition rate to the cell atmosphere gives the cell atmosphere time to transfer heat to the cell walls and results in lower temperature/pressure transients. The factors of oxygen content and cell volume determine how much sodium burns (assuming the fire is not sodium limited) and, therefore, how much heat would be transferred to the cell atmosphere. Also, for a given amount of heat generated, the resulting pressure transient would be less severe as the cell volume increases. The initial sodium pool temperature may also affect the temperature and pressure transient. A higher temperature pool would transfer more sensible heat to the cell atmosphere. The radiological consequences resulting from the various events are dependent on the radiation exposure history of the sodium involved and the amount of sodium that burns. The amount which burns, in turn, is dependent on the various factors noted above.

As indicated in the summary table of sodium fire events evaluated in the PSAR (Table 15.6-1) and as described in detail in the pertinent sections of 15.6.1 for each fire event, the accident conditions (sodium temperature, cell volume, pool area, oxygen content, radioactive content of sodium, etc.) vary considerably for the events analyzed. Consequently, as pointed out above, the pressure and temperature transients and radiological consequences associated with any one sodium fire event are expected to differ from those associated with any other event.

Question 001.241 (15.6.6.1)

How is the pressure decay calculated? What are the values of heat transfer coefficient used in the cooling process of the products of combustion and what is the basis for establishing these values?

Response:

The information on pressure decay calculations is provided in Revised Section 15.6. 25

Question 001.242 (15.6.1.1.2)

How is the containment design leakage rate of 0.1% vol/day, used in computing the leak vs time curve, affected by the temperature of the containment atmosphere.

Response:

This information is found in revised Section 6.2.1.3. 25

Question 001.243 (15.6.1.4.2)

(a) Reconcile the discussions on the experimental verification of the SPRAY-1 Computer Code with the discussion in Section A.86 concerning the experimental verification of this code.

(b) Quantify the conservatism believed to be inherent in the SPRAY-1 code results.

Response:

The discussion requested is provided in revised Section 15.6. 25

Question 001.244

Provide a description and summary of the experimental verification tests for the HAA-3 code.

Response:

Information has been added to Section A.42.

Question 001.245 (15.7.1.2.1)

Identify all safety related valves or instruments which require a compressed air supply.

Response:

See updated Table 15.7.1.2-1

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4. Inert Gas Receiving and Processing

New Valve No.	Valve Name or Function	Fail Position
82RPHV001 (1)	Diversion of RAPS input gas to CAPS on signal (3-way valve)	to RAPS
82RPHV002 (1)	Containment isolation valve, RAPS inlet	Close
82RPUV015A (1)	Flow control - surge tank effluent - lo range	Close
82RPUV015B (1)	Flow control - surge tank effluent - hi range	Close
82RPHV018 (1)	Selection of fill of drain of noble gas storage vessel (3-way)	Fail to isolate vessel
82RPHV019 (1)	Selection of fill or drain of noble gas storage vessel (3-way)	Fail to isolate vessel

(1) See Figure 11.3-4.

4. Inert Gas Receiving and Processing (continued)

Valve No.	Valve Name or Function	Fail Position
82APHV001 (2)	Containment isolation valve, CAPS inlet	Close
82APHV002 (2)	Containment isolation valve, CAPS inlet	Close
82NGHV351A (3)	Containment isolation valve, N_2 supply	Close
82NGHV351B (3)	Containment isolation valve, N_2 supply	Close
82CGHV501 (4)	Containment isolation valve, Ar supply	Close
82CGHV301 (4)	Containment isolation valve, recycle Ar supply	Close

(2) See Figure 11.3-6.
(3) See Figure 9.5-8.
(4) See Figure 9.5-2.

5. Steam Generator/Steam Generator Auxiliary Heat Removal Systems

Valve	Normal Operating Position	Final Fail Position After Loss of Compressed Air*	Function
Auxiliary Feedwater Pump Inlet	Open	Open	Isolation
Alternate Auxiliary Feedwater Pump Inlet	Closed	Closed	Isolation
Auxiliary Feedwater Pump Discharge	Open	In Place	Isolation
Auxiliary Feedwater Pump Recirculation	Closed	Open	Isolation
Auxiliary Feedwater Supply	Closed**	Open	Isolation
Drive Turbine Steam Supply	Closed**	Open	Isolation
Superheater Outlet	Open	Closed	Isolation
Superheater Inlet	Open	Closed	Isolation
Evaporator Inlet	Open	Open	Isolation
Feedwater Inlet	Open	Closed	Isolation
Evaporator Water Dump	Closed	Closed	Isolation
Superheater Outlet	Closed	Closed	Relief (Power Operation)
Evaporator Outlet	Closed	Closed	Relief (Power Operation)
Steam Drum Outlet	Closed	Closed	Relief (Power Operation)

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* Air stored in an accumulator for emergency operation of the valve.

**Open during SGAHRS Heat Removal Operation.

6. Recirculating Gas Cooling System

Air operated safety related valves are shown on Figures 9.16-3 through 9.16-7.

Table 9.16-3 list active air operated safety related valves with their "Preferred Direction".

7. Heating, Ventilating and Air Conditioning

Air operated safety related valves are shown on Figure 9.6-1, 9.6-4, 9.6-5, 9.6-7a.

There are no active air operated safety related valves in the HVAC system. 59