50-537

Department of Energy
Washington, D.C. 20545
Docket No. 50-537
HQ:S:82:066

JUL 14 1982

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:

RESPONSES TO REQUEST FOR ADDITIONAL INFORMATION

Reference, Letters, P. S. Check to J. R. Longenecker, "CRBRP Request for Additional Information," dated April 30 and May 14, 1982

This letter formally responds to your request for additional information contained in the reference letters.

Enclosed are responses to Questions 760.40, 105, 143, 145, 147, 150, 154, 158, 159, 160, and 161 which will also be incorporated into the PSAR Amendment 69; scheduled for submittal later in July.

Sincerely,

John R. Longenecker

John R. Longenecker
Acting Director, Office of the
Clinch River Breeder Reactor
Plant Project
Office of Nuclear Energy

Enclosures

cc: Service List
Standard Distribution
Licensing Distribution

D001

Question CS760.40

In the design evaluations presented in PSAR Section 4.2 and in the assessment of stochastic failure effects presented in Section 15.4.1.1, the overall predictions of the mathematical models and failure criteria were verified by comparison with results of in-pile tests. Such verification was not presented for the control and blanket assembly evaluations in Sections 15.4.2 and 15.4.3, nor for the assessments of the effects of molten fuel ejection or flow blockages in the fuel assemblies in Sections 15.4.1.2 and 15.4.1.3. Utilizing the in-pile experimental results that have been obtained to date, and considering the pertinent experiments that are in progress or are in the planning stages, provide the following information:

1. What is the experimental evidence that the heat-generating blockage configurations assumed in the Section 15.4 analyses occur in an actual situation?

2. Based on experimental evidence, what is the connection between the amount of molten fuel ejection, size of the resulting flow blockage that is formed, and the initiation of multi-rod failure for CRBR fuel conditions?

3. Are there any plans to use analytical methods and criteria which were verified with in-pile safety test results to evaluate the effcts of flow blockage and molten fuel ejection in the CRBR core assemblies? Alternatively, have the analytical methods and criteria utilized in the PSAR analyses of molten fuel ejection and flow blockage effects been verified against in-pile safety test results?

4. Is there any available experimental evidence (in-pile or out-of-pile) that molten fuel ejection and flow blockage effects in CRBR blanket assemblies are no worse than in fuel assemblies?

Response

1. Experimental and operational evidence to date indicate that heat generating blockages of any configuration do not form under normal operating conditions or the design basis events. The analyses presented in Section 15.4 consider postulated blockages to estimate the safety margin available in the design.

2. No molten fuel ejection is predicted during normal operating conditions or any design basis events. Nevertheless, if molten fuel ejection is postulated, any significant fuel exposure to the coolant would be annunciated on a time scale which is short compared with that required to impact other pins in the assembly. Experiments simulating unterminated overpower conditions have been reported as showing no indication of blockages what would suggest propagation to other assemblies (Reference QCS760.40-1).

3. The CRBRP is maintaining continuing cognizance of the tests such as the SLSF P-4 and the Mol-7C. Analytical techniques have been applied to obtain a physical understanding of the controlling process in such tests. Based upon the results from such safety tests as those mentioned above it is unlikely that such information would be required to conservatively predict in-core behavior under normal operating and design basis accidents.

4. A detailed consideration of the relative characteristics of molten fuel ejection in blanket assemblies as compared with fuel assemblies is not necessary because significant quantities of molten fuel are not predicted during normal operating conditions and design basis events for either the fuel or blanket assemblies. Experimental data have been obtained regarding the flow characteristics associated with passive planar blockages in blanket pin geometry (Reference QCS760.40-2). These have been compared with existing data on fuel assemblies to determine that the blanket assemblies have a similar ability to tolerate blockages of a given number of sub-channels.

References: QCS760.40-1 B. W. Spencer, D. R. Armstrong, L. Bova, et al., "Fuel-Sodium Thermal Interactions in the CAMEL TOP Safety Tests," FCI 4/P24, Fourth CSNI Specialist Meeting on Fuel-Coolant Interaction in Nuclear Reactor Safety, CSNI Report No. 37, Bournemouth, U.K., April 1979, pp. 551-569.

QCS760.40-2 B. J. Vegter, B. Minushkin, "Radial Blanket Assembly Flow Blockage Tests," WARD-SR-94000-6.

Question CS760.105

In Section 5.5.3.6 (Evaluation of Steam Generator Leaks), the design basis leak (DBL) appears to be based on limited operational experience, (with different steam generators) a number of non-prototypical tests and a few prototypical tests. Thus, it appears rather optimistic to conclude that "The conservatism of this postulated DBL will be confirmed through the LLTR test program."

a. Assuming this test program does not progress as anticipated, and that a larger design basis leak must be considered, identify the largest leak which can be tolerated by the currently proposed design and discuss the feasibility of design changes to accommodate even larger leaks.

b. Are the systems (particularly the pressure relief system) capable of being modified to accommodate a larger leak if further testing makes it advisable?

Response

a. The completed LLTR Test Program has confirmed the conservatism in the design basis leak. PSAR Section 5.5.3.1.5.1 has been updated to reflect the results.

b. The IHTS piping and IHX are adequately protected by the pressure relief system as presently designed. If a change is made to increase the leak size, changes may be required in the IHTS piping installation, as well as changes in the pressure relief system.

- o The natural frequency of the steam tubes in air and water
- o The damping rate of the tubes in air and water
- o Tube, support plate, shroud, liner, baffle and shell response to prototypic flow-induced forces
- o Vortex shedding frequencies in the cross flow region

Flow distribution tests were conducted for sodium flow rates from 10% to 100% of rated flow for the superheater and at 40 and 100% for the evaporator mode. Pressure measurements were made using two and three dimensional pitot probes; approximately 30 penetrations were made in the model. Flow measurements were made with a magnetic flowmeter having a calibrated accuracy of ± 260 gpm.

Schedule of Tests

Testing with the HTM was conducted between July 1975 and June 1976.

Summary of Results

Vibration tests indicated that the vibration levels are small and that excessive stress levels should not occur. Preliminary analysis determined the maximum vibration induced tube stress to be about 2000 psi (13.8 MD). This would occur in the cross flow region. The maximum tube peak-to-peak displacement amplitude was measured to be 6 mils and occurred in the static region. The corresponding tube stress was determined to be 400 psi by preliminary analysis. Vortex shedding was found to be not a dominant source of tube excitation.

The flow tests showed that the flow became uniform at an L/D of 21 (109 cm below the bottom of the inlet window) and remained uniform until about 15 cm above the outlet window. Relatively strong mixing occurred between the main body flow and the lower stagnant region. However, this condition did not appear to persist in the region of the tubesheet where near-stagration conditions are desirable.

b. Large Leak Tests

Objective

The objective of the large leak tests was to support establishment of adequate design and operational methods to accommodate large sodium-water reactions within the steam generator system of a LMFBR.

The large leak tests provided data in support of efforts to validate interim and advanced computer codes for large leak SWR analysis programs.

Data were obtained from the large leak tests for assessing the potential of secondary tube failures.

Data were also obtained on relief system performance and on cleanup and recovery techniques employed to return the system to operation following the major tube leaks.

Program Description

The Series I tests utilized, as a test article, the Atomics International modular steam generator (MSG) thermal hydraulic model previously tested in the sodium components test installation (SCTI) at Santa Susana, California. The model was converted to a large leak test article by replacing selected tubes with tubes designed for controlled rupture upon signal command. The large leak tests were conducted in the Large Leak Test Rig (LLTR), located at the Energy Technology Engineering Center (ETEC) at Santa Susana, California. The water injection system of the LLTR provided water under the desired conditions for the large leak injection device (LLID). The LLID for these tests consisted of a cylinder with a pneumatic piston which applies an axial load to the circumferentially weakened tube. The gas pressure is applied to the piston, the tube is pulled apart at the weakened spot, creating a guillotine type failure.

The MSG test article is an approximately 70' Lg. x 16" I.D. hockey stick steam generator containing 158 steam tubes each of which is 0.625 inches O.D. with a wall thickness of 0.109 inches. The tube material is 2-1/2 Cr-1 Mo steel and the tubing is spaced on centers 1.042 inches apart. The intentional rupture tube for a given test is pressurized by water/steam from the LLTR water injection system tanks. The remaining tubes are supplied by common headers at each end of the steam generator. The steam generator was assembled with four (4) inplace pre-weakened rupture tubes, and a capability for an additional four (4), using replaceable rupture tubes in the horizontal short leg of the AI-MSG.

The AI-MSG instrumentation consisted of approximately 130 thermocouples, 13 high-frequency pressure transducers, and 18 strain gauges. These sensors were provided to monitor bubble growth and pressure wave propagation, mechanical effects deformation, and to define the potential for secondary tube failures.

The test data capability provided for the modified AI-MSG includes:

1) Pressure wave magnitude and propagation mapping.
2) Bubble growth mapping.
3) Visual damage inspection.
4) Results of pneumatic and helium leak tests.

5) Ultrasonic signature comparison of tube wall thickness with original (full treatment depends on technique development in the bend regions).
6) Measurements of water/steam injection flow conditions.
7) Piping and vessel transient pressure, temperatures, and strain.
8) Relief system transients including pressure, temperature, flow, and strain responses at representative positions in the system out to and including the stack.
9) Transient pressures and temperatures in reaction products tank resulting from the entry of unreacted steam into the tank.
10) Vent system flow vs. time, and particulate size, distribution, and chemical composition in the vent stack.

Six large leak tests were conducted in the MSG during the Series I LLTR tests. The first three investigated double-ended guillotine (DEG) breaks in the evaporator. The first test was of a break near the sodium outlet window. The second test, a break about twenty feet above the lower tube sheet. The third test, was of a break in the hockey stick region. The remaining three tests investigated a DEG in a superheater, DEG with inert gas and a large break (equivalent to 3 DEGS) in the superheater.

The results and data obtained from the LLTR Series I tests demonstrate that the TRANSWRAP Code predictions of pressures and velocities resulting from large SWR events are conservative (Reference 25). The methodology which validates the TRANSWRAP code for use in the design of the CRBRP SWRPRS utilized the following procedures:

1) The injection flow transients on the water side were computed with the RELAP/MOD5 code (Reference 9).

2) These flow injection rates were input to the TRANSWRAP code to compute pressures throughout the sodium side of the system. The calculations were based on a sodium water reaction with an assumed hydrogen yield of 65% of the injected water to hydrogen gas and a resulting bubble temperature of 1700°F.

3) A dynamic rupture disc model was incorporated into the TRANSWRAP code to conservatively predict pressures within the LLTR.

4) A static rupture disc model was used in TRANSWRAP to conservatively predict velocities throughout the system and pressures within the LLTR SWRPRS

The Series I Test Article was disassembled and examined following the sodium-water reaction tests. This examination (reported in Reference 28) showed no evidence of secondary tube failures. Tube deformation and localized wastage was found in the regions of the tube rupture sites. The maximum wastage found was 0.019 inch adjacent to the test no. 2 site. Wastage at other tube rupture sites was in the order of 0.004-0.005 inch.

Five large leak Sodium/Water Reaction (SWR) tests (two non-reactive and three reactive), and three intermediate size (reactive) leak tests have been conducted in the LLTR Series II Program. These tests were conducted in a half length, full-diameter prototypic cross-section steam generator with prototypic rupture disk assembly. These tests involved nitrogen gas as the non-reactive fluid and both subcooled water and superheated steam as reactive fluids that were injected into the sodium. A comparison between the key fluid parameters in the reactive tests and those in CRBRP at normal on-load conditions is tabulated below.

	CRBRP Range	Test Range
Initial Sodium Pressure, psig	110-220	80-255
Initial Sodium Temperature, ^{o}F	400-935	580-900
Water/Steam Pressure, psig	1500-2024	1550-2000
Water/Steam Temperature, ^{o}F	548-906	543-700

The test results are reported in detail in References 26, 27, 30, 31, and 32, and are summarized in Table 5.5-13. The three reactive large leak tests A2, A6, and A7 each had rapid DEG tube failure with the injection rate averaging 5 lb/sec of subcooled water for 36-40 seconds, in half-length, full size cross-section, prototypic model of the CRBRP steam generator. These tests were fitted with a prototypic, full-size, reverse buckling, double rupture disk assembly. These tests resulted in no subsequent tube failures.

Three reactive, intermediate size leak tests have been conducted in the LLTR Series II Program. A short duration superheater leak test (A8) resulted in no secondary tube failures (Ref. 32). The most recent leak test (A5), conducted in April 1982, is being evaluated. The A3 leak progression test (Ref. 27) produced a number of secondary tube failures. This test was initiated by rapidly pulling apart a pre-notched tube to expose an injection tube containing a pre-drilled apart a pre-notched tube to expose an injection tube containing a pre-drilled 0.040 inch diameter hole. This hole, representing the self-wastage leak depicted in Step 5 of Figure 15.3.3.3-1, was aimed at a target tube two rows away. The aiming and spacing had been previously determined by bench scale experiments to yield the maximum wastage rate on the target tube. Observed secondary failure sequence is tabulated below. For reference, an EDEG failure area (two cross sections) is 0.26 sq. in.

TIME-SECONDS	SECONDARY FAILURE	FAILURE AREA-SQUARE INCHES
0	Injection Tube	Pre-drilled 0.0013 sq. in.
60	1	
72	2 (Same Tube)	0.017 (total area of two holes)
97	3 (Thin Wall)	0.125
108	4	0.029
113.7	Rupture Disks Opened	
114	5	0.109
114-120	6	0.200
114-120	7	0.170

The first secondary leak required one minute to develop and was about 7% of an EDEG. These results are a conservative representation of how an actual leak would progress because: (1) Both the sodium and the water were static, (2) the initial leak was aimed and spaced to produce maximum wastage on the target tube (wastage rate is observed in bench scale tests to be sensitive to configuration), (3) the third secondary failure occurred at 97 seconds in the injection tube itself which was of non-prototypic (thin) wall thickness (injection tube original thickness was 0.025" compared to 0.109" prototypic) and the size of this failure was sufficient to cause significant wastage/overheating of other tubes in the leaksite region, and (4) the tubes contained initially subcooled water which was static; as the test progressed, the water flashed and was expelled into the supply system. The pressure in the secondary tubes rose from 1700 psi at 97 seconds to 2600 psi at 119 seconds. The tubes were thus both undercooled and overpressurized as compared to CRBRP conditions. Secondary leaks 5,6 and 7 occurred after rupture disk burst had further increased the ΔP across the tube walls. The three wastage/overheating/overpressure failures occurred approximately 17 to 23 seconds after the causative event (secondary failure number 3) occurred and together totaled less than two EDEG. The CRBRP Design Basis Leak is seen to be conservative in both the magnitude of and timing of secondary failures.

As a final level of protection against tube leaks in a steam generator, the steam generators and the IHTS are being designed to withstand the effects of a large sodium water reaction (SWR). The ASME Code categories being applied in the design of the steam generators and IHTS piping and components for the large SWR event are given in Table 5.5-10.

The design basis leak (DBL) for the CRBRP was selected based upon examination of the physical processes which exist for leak initiation and growth. Two types of tests have been reported which provide information on the leak growth mechanism - small scale tests which model effects of a SWR on materials, and large scale tests which model a large water leak in a model of a steam generator. Smaller scale sodium-water reaction tests have been done to develop an understanding of the effect of a SWR on neighboring tubes in a steam generator. Three mechanisms have been identified for leak growth: self-wastage, impingement, and overheating (mechanical damage from pipe whip, although extremely unlikely, could be considered another mechanism, as discussed later in this section). Self-wastage has been shown to occur for very small leaks in the range of 10^{-6} to 10^{-3} lb/sec (Ref. 13). The process is depicted in Figure 15.3.3.3-1. The result of this process is a leak size of the order of 10^{-3} to 10^{-2} lb/sec. which can produce wastage on another tube in the vicinity of the leaking tube.

Wastage can occur on the outside of a steam generator tube from a leak in another tube in the vicinity. Tests of this mechanism have typically been done by using a water jet directed through sodium to a target material sample. Water injection rates of approximately 10^{-4} lb/sec to 1 lb/sec have been tested. The wastage mechanism results in erosion of the target material at maximum rates of 0.001 to 0.007 inches per second (Ref. 14, 29). The wastage rate is found to be a function of the water injection rate, tube spacing, sodium temperature and leak geometry. Wastage occurring on the surface of a CRBRP steam generator tube at these rates could cause a secondary water leak from tube penetration. However, this would require at least 20 seconds to penetrate the 0.109 inch thick tube wall assuming an initiating leak of the proper characteristics to produce maximum wastage.

The size of a secondary water leak resulting from wastage is difficult to quantify since wastage tests are typically done on materials samples rather than pressurized tubes. The wastage areas observed in tests have ranged from 0.1 in^2 to 1.5 in^2. Failure areas corresponding to the highest observed wastage areas would result in water leak rates corresponding to that of a double-ended guillotine tube failure. However, the entire wastage area would not be expected to blow out. The wasted areas are typically pit-shaped with the area of the pit decreasing with depth. It would be expected that the small area at the bottom of the pit would fail, yielding a return water leak which halts the wastage. Therefore, while the size of a secondary failure caused by wastage is difficult to predict, it is expected to be smaller than the leak rate corresponding to a double-ended guillotine failure.

water injection rates and sodium temperatures. Japanese, German and US large leak SWR tests have produced no secondary failures.

The Japanese have conducted seven large leak SWR tests ranging from 7 to 10 seconds. The Germans have conducted five large leak tests of durations 4 to 9 seconds. Six large leak tests (in near-prototype configurations) have been conducted in the U.S. The U.S. test have ranged from 2 to 40 seconds in duration. Significant wastage was observed in only one U.S. test in which one tube in the leaksite region exhibited a 0.016 inch reduction in wall thickness. This corresponded to a wastage rate of 0.016 inch/sec. The U.S. large leak tests are described in Section 5.5.3.1.5.1.b.

References to Section 5.5

* 1. W.H. Yunker, "Standard FFTF Values for the Physical and Thermophysical Properties of Sodium" WHAM-D-3, July 6, 1970.

* 2. J.A. Bray, "Some Notes on Sodium/Water Reaction Work," CONF-710548, pp. 187-205, July 1972.

* 3. Nuclear System Materials Handbook, Hanford Engineering Development Laboratory, TID-26666, Volume 1, Section 2-2 1/4 Cr-Mo, pp. 1.0-1.2, Rev. 0, August 14, 1974.

* 4. R.B. Harty, "Modular Steam Generator Final Project Report," Atomics International, TR-097-330-010, September 1974.

* 5. Nuclear Systems Materials Handbook TID 26666, 1974.

* 6. V.L. Streeter and E.B. Wylie, Hydraulic Transients, McGraw-Hill, New York, 1967, Ch. 2 and 3.

* 7. John Pickford, Analysis of Surge, MacMillan, London, 1969, pp. 32-37.

* 8. D.J. Cagliostro, S.J. Wiersman, A.L. Florence, Stanford Research Institute Final Report P.O. 190-C1H88GX, "Pressure Pulse Propagation in a Simple Model of the Intermediate Heat Transport System of a Liquid Metal Fast Breeder Reactor," June 1975.

* 9. "RELAP4/MOD5 a Computer Program for Transient Thermal-Hydraulic Analysis of Nuclear Reactors and Related Systems," prepared by Aerojet Nuclear Company for U.S. Nuclear Regulatory Commission and Energy Research and Development Administration under Contract E (10-1) - 1375, ANCR-NUREG-1335, September 1976.

* 10. J.N. Fox, R. Salvatori, H.J. Thailar (W NES), "Experimental Bending Tests on Pressurized Piping Under Static and Simulated Accident Conditions" TRANSACTIONS, ANS Power Division Conference on Power Reactor Systems and Components, September 1-3, 1970.

11. "Draft Design Basis for Protection Against Pipe Whip," ANSI N176, June, 1974.

12. Deleted

* 13. Gudahl, J.A. and Magee, P.M., "Microleak Wastage Test Results," GEFR-00352, March 1978.

* References annotated with an asterisk support conclusions in the Section. Other references are provided as background information.

27. J. C. Amos, et.al, "Evaluation of LLTR Series II Test A-3 Results, Revision I," General Electric Advanced Reactor Systems Department, May 1982, Prepared for U.S. Department of Energy under Contract No. DE-AT03-76SF70030, Work Package AF 15 10 05, WPT No. SG037.

28. J. O. Sterns, "Metallurgical Evaluation of the Modular Steam Generator (MSG) after LLTR Testing," ETEC-78-12, Sept. 1978.

29. D. A. Greene, J. A. Gudahl and P. M. Magee, "Recent Experimental Results on Small Leak Behavior and Interpretation for Leak Detection," CONF-780201, Vol. 1, paper No. 12 (First Joint U.S./ Japan LMFBR Steam Generator Seminar), February 1978.

30. J. C. Amos, et al, "Evaluation of LLTR Series II Test A6 Results," prepared for U. S. DOE under Contract DE-AT03-76SF0030, June 1981.

31. D. E. Knittle, et al, "Evaluation of LLTR Series II Test A7 Results, prepared for U. S. DOE under Contract DE-AT03-76SF70030, September 1981.

32. J. J. Regimbal, et al, "Evaluation of LLTR Series II Test A-8 Results," prepared for U. S. DOE under contract DE-AT03-76SF70030, February 1982.

*References annotated with an asterisk support conclusions in the Section. Other references are provided as background information.

TABLE 5.5-13 SUMMARY OF U.S. LARGE SODIUM/WATER REACTION TESTS

COUNTRY	TEST DESIGNATION/ OBJECTIVE	TEST VESSEL	TEST BUNDLE	INITIAL PRESS/TEMP. SODIUM PSIG	SODIUM °F	WATER PSIG	WATER °F	WATER INJECTION METHOD	DURATION SEC	WEIGHT LB	SIGNIFICANT RESULTS
U.S.	LLTR Series-1, SWR-1/One Double ended Guillotine (DEG) Failure near lower nozzle, sub-cooled H_2O	AI-MSG 16- inch ID Vessel Proto-typic Height	158 tubes of prototypic material and dimensions, prototypic-ally spaced	122	600	1900	543	Rapid DEG of pre-weakened tube	10	80	No Secondary failures. Maxi-mum wastage on one tube near leaksite = 0.016 inches. Only significant wastage in all 6 Series 1 tests.
	LLTR Series 1, SWR-2/Same as SWR-1 @ mid-span	Same as SWR-1	Same as SWR-1	81	628	1900	543	Same as SWR-1	10	60	No secondary failures
	LLTR Series 1, SWR-3 One DEG @ 1.75 in from upper tube sheet, Two-Phase H_2O	Same as SWR-1	Same as SWR-1	116	800	1900	700	Same as SWR-1	5	40	No secondary failures
	LLTR Series 1, SWR-4 Same as SWR-3 with superheated steam	Same as SWR-1	Same as SWR-1	80	800	1900	700	Same as SWR-1	3	8	No secondary failures
	LLTP Series 1, SWR-5 Same as SWR-4 with 700-F Nitrogen injected	Same as SWR-1	Same as SWR-1	90	800	1900	700	Same as SWR-1	3	zero	Served to cali-brate RELAP code
	LLTR Series 1, SWR-5 Same as SWR-4 with Three Equivalent DEG	Same as SWR-1	Same as SWR-1	90	800	1900	700	Same as SWR-1	3	8	No secondary failures. Series 1 served to validate the TRANSWRAP Code.

TABLE 5.5-13 SUMMARY OF U.S. LARGE SODIUM/WATER REACTION TESTS

COUNTRY	TEST DESIGNATION/ OBJECTIVE	TEST VESSEL	TEST BUNDLE	INITIAL PRESS/TEMP. SODIUM PSIG	SODIUM °F	WATER PSIG	WATER °F	WATER INJECTION METHOD	DURATION SEC	WEIGHT LB	SIGNIFICANT RESULTS
U.S.	LLTR Series II, Test A1a, One DEG @ Lower Midspan, Injected Nitrogen, Prototypic Rupture Disk Assembly used on all Series II Tests	Same as A2	Prototypic	125	580	2000	580	Same as SWR-1	30	0	Prototypic rupture disk assembly used on all Series II tests. Served to verify RELAP calibration
	LLTR Series II, Test A1b, Same as A1a except A1b used double disk and minor difference in leak location.	Same as A2	Same as A2	125	580	2000	580	Same as SWR-1	43	0	Served to verify RELAB calibration
	LLTR Series II, Test A2, One DEG @ Lower Midspan, subcooled H_2O	Prototypic Cross-Section 1/2 Length	Prototypic	125	580	1700	580	Same as SWR-1	40	200	No secondary failures. Maximum measured secondary wastage equals 4 mils. Prototypic double disc assembly served to calibrate TRANSWRAP rupture disc model

TABLE 5.5-13 SUMMARY OF U.S. LARGE SODIUM/WATER REACTION TESTS

COUNTRY	TEST DESIGNATION/ OBJECTIVE	TEST VESSEL	TEST BUNDLE	INITIAL PRESS/TEMP. SODIUM PSIG	INITIAL PRESS/TEMP. SODIUM °F	INITIAL PRESS/TEMP. WATER PSIG	INITIAL PRESS/TEMP. WATER °F	WATER INJECTION METHOD	WATER INJECTION DURATION SEC	WATER INJECTION WEIGHT LB	SIGNIFICANT RESULTS
U.S.	LLTR Series II, Test A-3, One self-Wastage Leak Simulation @ sub-cooled H_2O @ 0.1 lbm/sec aimed for maximum secondary damage.	Same as A2	Same as A2	145	580	1700	580	Rapid pull-apart of prenotched tube to expose 0.040" dia. hole.	145	144 plus	Secondary failures (less than an EDEG) after long delays (one minute and longer).
	LLTR Series II, Test A6, One DEG @ Lower Midspan Periphery, subcooled H_2O.	Same as A2	Same as A2	125	580	1700	580	Same as SWR-1	36	200	No secondary failures.
											System modified as gas-free Actual test contained large gas space to S.G. TRANSWRAP over-predicted measured pressures where comparable.

TABLE 5.5-13 SUMMARY OF U.S. LARGE SODIUM/WATER REACTION TESTS

COUNTRY	TEST DESIGNATION/ OBJECTIVE	TEST VESSEL	TEST BUNDLE	INITIAL PRESS/TEMP. SODIUM PSIG	SODIUM °F	WATER PSIG	WATER °F	WATER INJECTION METHOD	DURATION SEC	WEIGHT LB	SIGNIFICANT RESULTS
U.S.	LLTR Series II, Test A7, One DEG @ Lower Midspan, sub-cooled H_2O higher initial sodium pressure.	Same as A2	Same as A2	255	580	2000	580	Same as SWR-1	2	15	Secondary tubes filled with nitrogen @ 400 PSIG.
	LLTR Series II, Test A8, intermed-iate-sized super-heated steam injection.	Same as A2	Same as A2	180	900	1550	700	Rapid pull-apart of prenotched tube to ex-pose 0.054" dia. hole.	40		No secondary failures deduced from instrum-entation and post test helium leak checks. Final confirm-ation awaits post test destructive examination.
	LLTR Series II, Test A5, inter-mediate-sized superheat in-jection	Same as A2	Same as A2	50	625	1450	625	Rapid pull apart of tube to ex-pose 0.25" dia. hole	58	TBD	Test Report not available. Examination of of test article in progress.

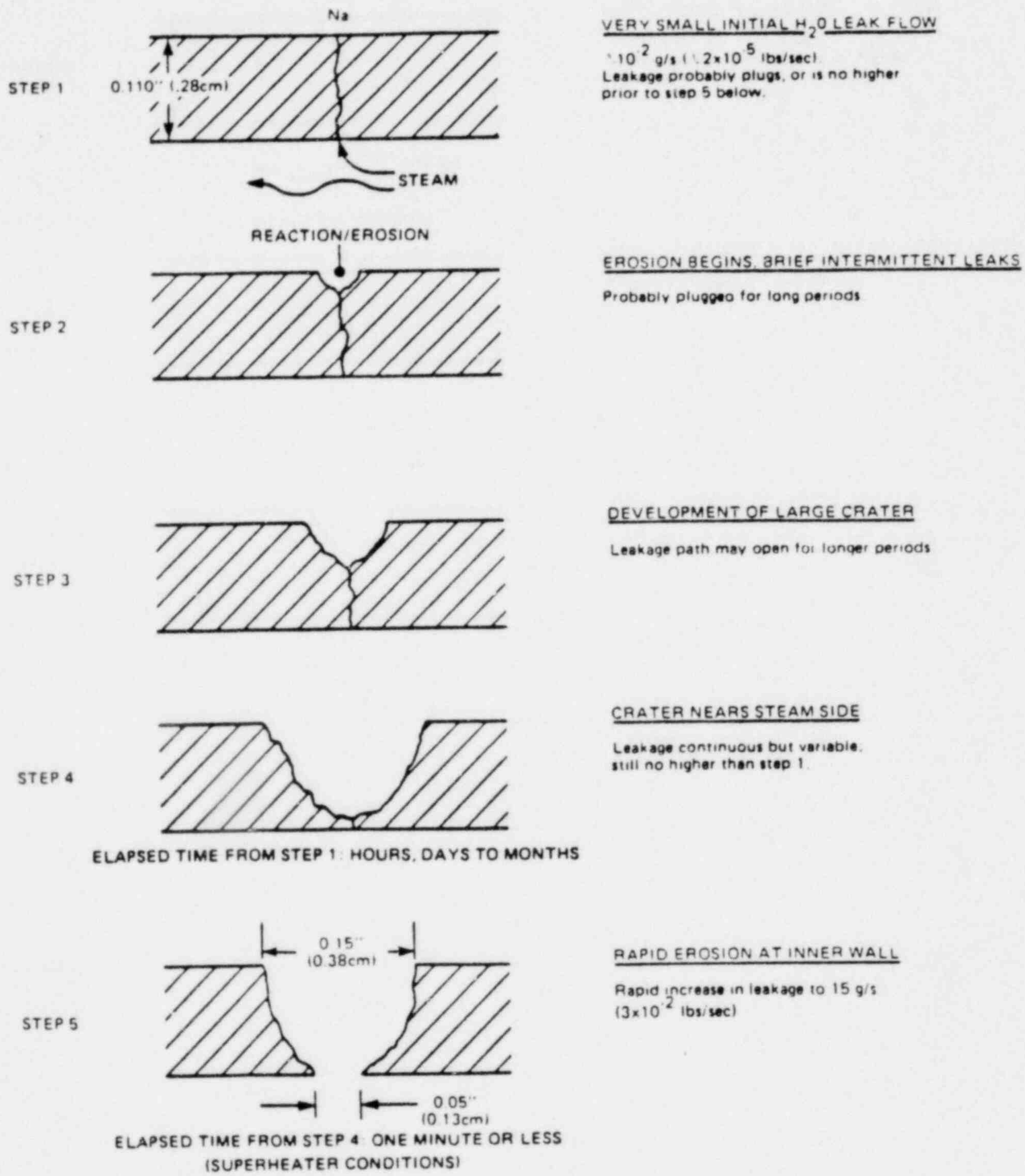

Figure 15.3.3.3-1. Development of a Large Leak From a Small Steam Leak in 2 1/4Cr-1Mo Tubing Exposed to Sodium

Question CS760.143

At several locations in CRBRP-3, Volume 2, Revision 0, the applicant states that the concrete structures being analyzed can withstand the imposed loads with additional reinforcement. One example is in Section 3.2.2.5.1.2, at the bottom of page 3-48. To make the statement relevent, the applicant needs to indicate what the basis for the statement is (i.e., additional to what?). Does the current design include this additional reinforcement?

Response

In CRBRP-3, Volume 2 "base design" is considered to include the requirements for operating conditions and all accident conditions other than TMBDB. Wherever there is reference to "additional reinforcement", it means that it is additional over the base design. The current design does include the additional reinforcement required for TMBDB.

Question CS760.145

In Table 3-10 in CRBRP-3, Volume 2, Revision 0, the applicant presents containment capability in terms of pressure for a range of temperatures. How were the stresses calculated to compare with Sy and Su? Were penetrations and discontinuities considered?

Response

The stresses were calculated by the ASME Code formula setting the allowable stress at yield for the Sy case and using the ASME Code formula for faulted conditions for the Su case. Penetrations and discontinuities were considered in accordance with the ASME Code Rules. Deformations and local yielding were considered acceptable because for a one cycle highly unlikely event, this could not cause gross failure except by a possible buckling mode, which is considered in Sections 3.2.2.5.2 and 3.2.3.3.1.3 of CRBRP-3, Volume 2.

Question CS760.147

In CRBRP-3, Volume 2, Revision 0, Sections 3.2.3.5.2 and 3.2.3.3.1.3, the applicant refers to a 240°F critical containment vessel buckling temperature. Where does this come from? Are the buckling criteria presented in the PSAR used? If not, what criteria are used? Possible buckling at points other than the base of the cylinder should be considered and any appropriate assymmetries should be included.

Response

Section 3.2.3.5.2 does not exist. Section 3.2.3.3.1.3 refers to "critical containment vessel buckling temperature" and 3.2.2.5.2 and 3.2.2.5.2.2 discuss this general subject.

Buckling stresses in the entire shell were examined and it was determined that the controlling area was the discontinuity at El 816'. This controlling area was therefore, analyzed further.

The PSAR design criteria were used to determine that the interacting stresses were 92% of the allowable value at 240°F. (The safety factor was set to 1.0 to determine this parameter.) This 240°F temperature is referred to as the "critical containment vessel buckling temperature" in CRBRP-3, Volume 2.

Question CS760.150

How is the temperature of a new fuel element in an EVST preheat tube determined? What is the maximum ΔT allowed when a new fuel element is put in a sodium-filled CCP? Where is the temperature of the fuel determined by the operator of the fuel handling equipment?

Response

The temperature and rate of temperature rise of new fuel assemblies are inherently and passively determined by the preheating procedure. The assembly is heated by placement in a preheating station in the EVST which is argon-filled but surrounded by EVST sodium. Heat transfer to the assembly in this oven is sufficiently slow to avoid excessive thermal transients but gradually raises the assembly to the temperature of the EVST sodium. The assembly resides in the preheat station for at least the predetermined time required to raise the temperature of each region to be sufficiently close to the sodium temperature to avoid excessive thermal stresses. Conservative one-dimensional calculations (considering heat transfer by radial radiation and convection, but not axial conduction) show that sufficient preheating will be obtained in a period of 8 hours. This time is reasonable in terms of impact on overall fuel handling time. At the end of the specified time period, the EVTM operator will transfer the assembly to a sodium-filled core component pot. There is no measurement of the assembly temperature since, with the passive heating, temperature will depend only on the length of heating time.

Question CS760.154

Describe the "emergency cooling" process instituted in case of electrical power failure to the fuel transfer port cooling insert blower during CCP transfer. For each case, what is the maximum time allowed without heat removal for the hottest fuel subassembly?

Response

PSAR Section 9.1.4.7.3 has been revised in response to NRC Question CS410.4 (9.1.4).

Question CS760.158

In air-filled cells, the PSAR states that the catch pan sides extend up the wall to a height sufficient to prevent spilled liquid metal from flowing over the edge of the plate between the plate and the wall. Additionally, a continuous lip plate is provided at the top of the catch pan side walls to prevent sodium or NaK from running down the structural concrete walls into the region behind the catch pan plate sidewalls. Also, in the event of a liquid metal spill, the catch pan contains the liquid metal and prevents contact between the liquid metal and the concrete structure. If liquid metal can run down the structural concrete walls, what prevents liquid metal-concrete reactions on the vertical structural concrete wall areas above the catch pans? What penetration or degradation of the fire wall between equipment spaces would be expected? Discuss your acceptance criteria for this event.

Response

Postulated liquid metal spill events in air-filled cells may result in impingement on vertical concrete surfaces depending on break size characteristics and hydraulic head effects. The Project is investigating techniques to accommodate liquid metal jet impingement on vertical concrete walls whereby any degradation to the vertical concrete walls does not result in loss of wall structural integrity or propagation of the event to the operable decay heat removal loops.

Question CS760.159

Along with question 2 above, has any allowance been made on the height of the catch pan walls to allow for thermal expansion of the liquid metal and for addition of any fire extinguishment? Can the catch pans be expected to perform their functions under all anticipated events?

Response

Thermal expansion of the liquid metal was considered in the sizing of catch pans. However, the effect is minimal. No fire extinguishment is required since fire suppression decks are provided for this function.

The catch pans will perform their design function for design basis liquid metal spill events.

Question CS760.160

It is not clear from CRBRP-3, Volume 1, Revision 2, what criteria have been used in developing the component margin requirements presented in Section 5.2. Section 5.1.1.4 indicates that the REXCO-HEP Code has been used to generate these loads and in Section 5.1.1.3, the applicant presents several reasons why the REXCO-HEP calculations are conservative approximations to the loads that would actually be experienced by the structure. The applicant is expected to give some experimental basis for the general assumption of how the REXCO-HEP calculations were compared with the SM-4 and SM-5 scale model test results. The comparison should include peak pressures, total impulse delivered to the component in question and a discussion of frequency content where dominant frequencies in the loading function may possibly be in tune with natural frequencies of vibration for structural components. For any component margin requirements that are not taken directly from REXCO-HEP predictions at the obvious point of application, such as the load to be applied to the UIS given in Figure 5-19, a full description is needed of how the requirements are derived.

Response

The overall criterion used in developing the SMBDB requirements in Section 5.2 of CRBRP-3, Volume 1 was:

The reactor vessel, closure head, PHTS and other piping systems connected to the reactor vessel shall continue to function as limited leakage barriers following dynamic loads that would result from bubble expansion as defined by the pressure-volume curve of Table 5-1 of CRBRP-3, Volume 1. (The expansion of this bubble to the point of sodium impact with the head would release 101 MJ.)

The dynamic loads for the various components and systems were calculated using the methods and computer codes described in Section 5.1 of CRBRP-3, Volume 1. Additional information is as follows:

Upper Internals Structure

Since the REXCO-HEP Code cannot model the fluid response of the upper internals structure, it was not included in the system model. However, because of the dissipative nature of this component (principally through fluid turbulence), a model which excludes the upper internals structure would be expected to provide generally conservative loads on other components such as the closure head. This was confirmed in scale model experiments (e.g., compare the experimental results in Figures 4.1-16 and 4.1-17 of Reference 14, PSAR Section 1.6, "Structural Response of CRBRP Scale Models to a Simulated Hypothetical Core Disruptive Accident", October 1978). To assess the response of the upper internals structure and the load on the closure head transmitted by the upper internals structure columns, a dynamic load was defined for the upper internals structure. This load was taken from the REXCO-HEP calculation at the location of the top of the core barrel. This location approximates the elevation of the underside of the upper internals structure.

Head Mounted Components

To assess the response of head mounted components to the slug impact load, the SMBDB slug load, as defined in Figure 5-18 of CRBRP-3, is used. The load is applied to a detailed three-dimensional finite element model of the head and associated shielding. This dynamic model developed with the ANSYS Code allows plastic yielding of the head plates to occur, where appropriate. Additional loads resulting from vessel movement, upper internals structure response and direct under-head gas pressurization are also applied. The translational and rotational motions defined at locations appropriate to each of the head mounted components are then used to evaluate component responses to head motion.

An experimental program was performed using scale models to simulate the response of the reactor vessel system to the expansion of a bubble as defined by the pressure-volume curve of Table 5-1 of CRBRP-3, Volume 1. The experimental program and analyses are provided in Reference QCS760.160-1. REXCO-HEP and ANSYS calculations are compared to the scale model experimental results in Section 4.1 and Appendix A of the reference.

In addition to a generally higher level of loading in the analytical simulation, there is also a significantly greater higher frequency excitation in the analytical cases. While detailed frequency response assessments are not performed in all cases, it is judged that considerable conservativeness is inherent in the loadings as specified.

Question CS760.161

In CRBRP-3, Volume 1, Revision 2, it is unclear how the component margin requirements are to be applied. Are any to be applied simultaneously? Where the requirements are given in terms of pressure histories, how are the loads to be distributed? What boundary conditions will be used or what will be the criteria for choosing boundary conditions when separate components are analyzed?

Response

Discussion of the application of component margin requirements and associated boundary conditions is provided in Section 5.4 of CRBRP-3, Volume 1. This discussion, together with the requirements of Section 5.2, generally furnish sufficient information for component analysis to be performed. The requirements indicate cases in which simultaneous loads are to be applied (e.g., see Section 5.2.1B).

In some cases (e.g., the vessel wall), CRBRP-3 does not show loadings at all axial intervals simply because of the large quantity of loading zones involved. Therefore, only representative ones are shown in CRBRP-3. In the actual analyses of the components, however, additional loading curves at points other than those shown are used in addition to the representative ones shown in CRBRP-3.

With respect to the vessel head loading, the load, as shown in Figure 5-18 of CRBRP-3, is applied to the impact surface in a uniform manner. The uniform impact assumption is justified by the fact that the slug surface remains nearly flat at the time impact takes place (see Figure 5-2a of CRBRP-3).