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ACRS CRBR WORKING GROUP MEETING MINUTES
OCTOBER 26, 1982
WASHINGTON, DC

Purpose: The purpose of the meeting is to review the thermal hydraulic aspects of the CRBR plant design.

Attendees: Principal meeting attendees include:

ACRS	DOE
M. Plesset, Chairman	J. Longenecker
R. Axtmann, Member	**Westinghouse**
M. Carbon, Member	P. Dickson
C. Mark, Member	G. Clare
W. Kastenberg, Consultant	R. Markley
R. Shumway, Consultant	T. Pitterle
Z. Zudans, Consultant	R. Smith
P. Boehnert, Staff*	J. Winters
	R. Lowrie
	NRDC
	T. Cochran

*Designated Federal Employee

Meeting Higlights, Agreements, and Requests

1. Mr. G. Clare (W) provided a review of the overall plant heat transport systems. He outlined the design details of the reactor, fuel assemblies and heat transport loops (Figure 1). In response to Dr. Plesset, Mr. Clare said that pressure control for the primary system is maintained by use of the cover gas system. Dr. Plesset asked if the system has been tested. No specific tests of the system have been done. Additional discussion on this item and the potential for troublesome sodium oscillations was postponed to the presentation on primary pump level control.

The normal (Figure 2), and off-normal (Figure 3) heat removal paths were described. The normal heat removal system through the steam generators is also used for decay heat removal. If the condenser water system is unavailable, auxiliary water is supplied to the steam drum and heat is removed via the Protected Air Cooled Condenser (PACC). In addition heat can be removed by venting to the atmosphere. In response to Dr. Shumway, it was noted that with one of the three PACC's operating in the natural circulation (NC) mode, it is necessary to steam to the atmosphere for one/hour after shutdown using water from the protected water storage tank (PWST). The PWST inventory is sufficient to allow steaming to the atmosphere for up to eleven hours, if required. Beyond 11 hours additional water supplies can be secured if necessary.

In addition to the above, if none of the three heat transport loops are available, a Direct Heat Removal Service (DHRS) system is available for decay heat removal (Figure 4). Motive power is required for system function however, and emergency AC power is available to the DHRS.

In response to Dr. Kastenberg, Mr. Stark (NRC) said the DHRS was provided for diversity of decay heat removal. It is not redundant to the three primary heat transport loops. In response to Dr. Zudans, Mr. Clare said a scale test was run to check for possible short-circuiting of the DHRS cooling sodium flow across the top of the reactor vessel. There was no significant problem seen.

Mr. Clare indicated that the plant heat removal systems are designed for both high reliability, and diversity (Figure 5). Guard vessels and elevated piping are provided to assure primary coolant inventory is maintained tained. In response to Dr. Zudans, Mr. Dickson said NC capability is still maintained with a leak into the guard vessel. The plant is also designed to accommodate a station blackout by NC decay heat removal (Figure 6). No operator aciton is required for NC operation.

2. Mr. R. Markley (W) detailed the reactor thermal hydraulics for CRBR. Key points/discussion items of his presentation include:
   - The details of the in-vessel flow paths were noted, particularly the lower inlet plenum flow paths to the core assemblies. There are redundant flow paths available to minimize the impact of any potential flow blockage.
   - The status of the thermal hydraulic development testing program was detailed. Extensive out of pile tests have been run on such items/phenomena as: fuel, blanket, and control assemblies, the core inlet and outlet region, thermal striping, and reactor flow tests. Most of the testing has been completed. No major problems have been identified to date. Irradiation testing conducted at EBR-II, TREAT, and FFTF has verified the thermal hydraulic design approach used. Dr. Plesset questioned the use of wire wrap for pin spacing. He also expressed reservations about the closed flow design of the core vis-a-vis blockage effects. The Project noted that use of wire wrap is state-of-the art and is economical as well. The closed flow design is needed to closely control core flow resulting in improved core performance.
   - Key core T/H design bases include: no fuel melt at 115% overpower (including 3σ uncertainties) and no sodium boiling during normal, anticipated, unlikely, and extremely unlikely conditions. Major core T/H performance criteria are given in Figure 7. Figure 8 shows the T/H codes used by the Project for the fuel/blanket design effort. Mr. Markley discussed the validation of a particular T/H code (COTEC) by comparison to W ARD test data to illustrate the extent of the verification work performed. Fuel rod flow induced vibration tests have shown no wear or vibration problems. Tests on-going in FFTF are expected to confirm their design adequacy. In response to Dr. Shumway, Mr. Markley said tests have not shown any wear between the clad and the wire wrap.
   - The worse-case undercooling and overcooling design transients were detailed. The worse case undercooling event is the natural circulation transient. Figure 9 details the major assumptions used in the NC hot rod analysis. The results show that the 3σ worse-case hot rod is well below (150°F) the sodium boiling temperature. For the worse-case overpower event (SSE), which results in an assumed 60c step reactivity insertion there is no sodium boiling or clad melt seen. (There could be some clade damage if only the secondary shutdown system responds.)

° Assembly inlet and module blockages are highly unlikely due to the built-in flow blockage features (Figure 10). In-core passive blockage is not believed possible; however, an ORNL test has shown rod-to-rod failure propagation will not occur for a postulated six-channel blockage. Heat generating blockage can only be formed by fuel debris buildup and would be detected by the delayed neutron detector.

3. T. Pitterle discussed the program established to address flow induced vibration. The program emphasis has been on testing, due to the complexity of the in-vessel design. Both 1/4- and 1/3-scale as well as full-scale tests have been conducted (Figure 11). In addition, the FFTF test experience will be factored into this Program. CRBR preoperational tests will also be factored into this Program. Finally, DOE will install accelerometers in the Upper Internal Structure for preoperational vibration monitoring.

   Dr. Zudans asked if there is a possibility that the primary pump vibration frequency is at or near the resonance frequency of the core support cone structure. The Project cited figures to demonstrate that there is a significant separation in the structure forcing function vis-a-vis the pump frequency.

4. R. Smith (W) discussed the design details of the primary and intermediate sodium pumps. Details of the pump and hydraulic design were noted (Figures 12 and 13). In response to Dr. Mark, Mr. Smith said that the free surface hydraulic design was chosen because of its tolerance to thermal transients. Functional requirements are noted in Figure 14. A 75 H.P. pony motor is included on each pump to aid decay heat removal during and after design basis events. One-half scale hydraulic model testing and full scale prototype tests in water have been run. Currently prototype sodium tests are underway. These tests have shown a need to install baffles to mitigate thermal convection currents seen near the internal thermal shield.

5. The CRBR steam generator design was reviewed by Mr. Winters. Details of the design description are given in Figures 15 and 16. All design tests are now complete and confirmatory tests are presently underway (Figure 17).

Dr. Mark inquired about potential corrosion problems. Mr. Winters said the material used (2-1/4 CR-1 Mo stainless steel) should preclude corrosion problems. In addition, ultrasonic inspection of sample tubes will be performed every three years. Dr. Mark asked how tube ruptures are detected. Mr. Winters noted that there are continuously operable hydrogen and oxygen monitors, to detect either of the above sodium/water reaction products.

Dr. Plesset inquired as to the potential for flow induced vibration (FIV). The Project indicated that elimination of FIV has been carefully considered in the design and this consideration should be confirmed in planned FIV tests.

6. Mr. Lowrie discussed the heat transport system thermal-hydraulics. Details of the intermediate heat exchanger and cold leg check valve were noted. In response to Dr. Carbon, Mr. Dickson said all accident analyses assume the check valve function is lost (fails open). Also detailed were the PHTS and IHTS nominal and maximum piping and component pressure drops. For all decay heat calculations, the minimum possible flow is assumed.

   The most severe/limiting transients for the plant were noted. These are: pump seizure, failure of pony motor and check valve, NC, and use of the DHRS. Design features to mitigate these transients are extended pump coastdown, use of pony motors, and the cold leg check valve. Figure 18 shows the conservatisms in the analysis. The NC and use of DHRS was highlighted. For a NC event given loss of flow in two of the three heat transport loops, maximum clad temperature is maintained at ~1000°F.

   Dr. Plesset asked if ATWS had been analyzed.. Mr. Clare said that common cause failures will dominate the cause of this event and W will perform a common cause analysis as part of their plant PRA. Dr. Carbon asked if power fluctuations could cause a problem here. The Project said they would look into this and address this item at the next Subcommittee/Working Group meeting on the plant protection system.

Analysis of the use of the DHRS was discussed (Figure 19). The Project analysis assumes full DHRS operation 30 minutes after scram. Peak primary system temperature seen in the analyses is 1055°F occurring about seven hours into the transient. All DHRS pumps and fans have access to emergency AC power. The DHRS is seismic Class I qualified.

For the NC transient analysis, the Project assumed loss of all power to motor driven pumps, thus relying on use of one heat transport loop. The acceptance criterion is that core hot channel temperatures shall be below sodium boiling. The Project methodology used to analyze CRBR NC was verified against FFTF NC tests. Results showed conservative predictions of the FFTF core temperatures. The CRBR plant was shown to be able to adequately remove decay heat via one heat transport loop with substantial margin to sodium boiling.

Further discussion brought to light the fact that there is at least one hour available to get a diesel generator started for use of the DHRS.

7. The status of the NRC CRBR T/H review was described by T. King. The scope of review, criteria/standards applied, and the adequacy of the heat transport system were detailed. Some of the items under active review include: potential for floatation of primary control rods during refueling; methods employed for selecting, categorizing, and applying hot channel factors; degree of margin for natural circulation capability; and PACC design and performance. In response to Dr. Mark, Mr. King said the Staff would like to see the NC analysis carried out to ~24 hours from the current analysis that now stops at 10 minutes. NRC review of the T/H topic is scheduled to be completed in January 1983.

8. Dr. T. Cochran (NRDC) provided comments on the July 13, 1982 ACRS letter commenting on site suitaiblity for CRBR. He noted that the letter does not state that CRBR is suitable for this site. He said ACRS should so state one way or another, persuant with its statutory responsbility. He decried the allowing of site work to progress before site suitability has been determined, and said that the issue of whether a CDA should be a DBA must be addressed.