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MASTER

WHOLE-CORE THERMAL-HYDRAULIC TRANSIENT CODE DEVELOPMENT AND VERIFICATION FOR LMFBR ANALYSIS

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Predicted performance during both steady state and transient reactor operation determines the steady state operating limits on LMFBRs. Unnecessary conservatism in performance predictions will not contribute to safety, but will restrict the reactor to more conservative, less economical steady state operation. The most general method for reducing analytical conservatism in LMFBR's without compromising safety is to develop, validate and apply more sophisticated computer models to the limiting performance analyses. The purpose of the on-going Natural Circulation Verification Program (NCVP) is to develop and validate computer codes to analyze natural circulation transients in LMFBRs, and thus, replace unnecessary analytical conservatism with demonstrated calculational capability.

The computer codes currently under development for LMFBR natural circulation calculations include whole-core, plant and hot channel analysis models. The whole-core thermal-hydraulic transient analysis code (whole-core code) will be used to predict peak reactor coolant temperatures, taking into account inter- and intra-assembly flow and heat redistribution. The development and validation requirements for the whole-core code are summarized in Table 1.

The requirements for the development of the whole-core code evolved from a combination of experimental and analytical investigations, due to a lack of capability to perform such analyses without excessive analytical conservatism. The most important requirements are listed in Table 1.

The temperature changes due to modeling the various phenomena are not independent, so the inclusion of one may lessen the effect of modeling another. Furthermore, the time required for heat transfer and flow redistribution to occur may delay the potential temperature reductions until late in the transient after the peak temperature has occurred. Detailed transient modeling is essential for predicting such behavior.

The validation of the whole-core code will provide confidence that the code is an acceptable tool for predicting the performance of an LMFBR during a transient and that the code predictions are sufficiently accurate to be used for licensing an LMFBR. Validation will be accomplished by comparing code predictions to a data base sufficiently broad to demonstrate all of the important code capabilities. The data base is defined by the list of relevant test programs in Table 1. The most rigorous comparisons are to the EBR-II and FFTF Acceptance Test Phase (ATP) tests, which include temperature and flow rate measurements of LMFBR performance during both steady state and natural circulation transients.

The transient flow rate, pressure drop and outlet temperature of a large number of assemblies, subchannels and other flow paths will be predicted by the whole-core code given the transient total flow rate, inlet temperature, heating rates, hydraulics, etc. as input. A "plant code" which models the primary loop with much less reactor detail than the whole-core code is required to predict the time-dependent reactor flow rate which is input to the whole-core code. The coupling between the core and primary loop calculations is expected to be sufficiently weak, that iteration between the whole-core and plant codes will not be required. The validity of this approach will be demonstrated by the FFTF ATP pre-test predictions.

After a satisfactory whole-core code calculation is achieved, the limiting fuel and/or blanket pins will be identified and additional design temperatures calculated with the pin design analysis code. Given the time dependent coolant temperatures from the whole-core code, a hot channel analysis code such as FØRE-2M[11] quickly performs detailed design temperature calculations, including such local rod phenomena as fuel restructuring, with uncertainty factors at selected levels of confidence. The resulting temperatures can then be evaluated for safety margins such as the temperature margin-to-coolant boiling or fuel melting at some level of confidence, or included in cladding damage analyses.

In summary, a coordinated program is underway to demonstrate the capability to predict the natural circulation performance of LMFBRs during transients. The important calculational capabilities were identified and specified as whole-core code development requirements and a data base was identified which is considered adequate for code validation. Pre-test predictions for the FFTF Acceptance Test Phase Four steady state and transient measurements to be made with both plant and in-core assembly instrumentation will provide a comprehensive test of both the calculational sequence and the accuracy of the natural circulation codes.

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TABLE 1

WHOLE-CORE THERMAL-HYDRAULIC TRANSIENT ANALYSIS

CODE DEVELOPMENT AND VALIDATION REQUIREMENTS

MODELING DEVELOPMENT	CODE VALIDATION DATA
Intra-Assembly Heat Transfer and Flow Redistribution	HEDL 217-Pin Low Flow Heat Transfer Test (Ref. 1)
Inter-Assembly Heat Transfer and Flow Redistribution	ARD 61-Pin Blanket Heat Transfer Test (Ref. 2)
Form and Friction Factor Correlations	ORNL 19-Pin FFM Test (Ref. 3)
Inboard/Side Subchannel Flow Split	ORNL 61-Pin THORS Test
Fuel/Blanket Decay Heat Generation Rate	EBR-II Natural Circulation Tests with XX08 (Ref. 5)
Variable Gap Conductance	FFTF Acceptance Test Phase 4 (Ref. 4)
Pressure Drop Boundary Condition	FFTF-Type Fuel Assembly Hydraulic Tests (Refs. 6, 7, 8, 9)
Dissimilar Fuel and Structural Materials	MIT 61 and 217-Pin Tests and Flow Split Models (Ref. 10)
Bypass and Interassembly Flow	
Flow Reversals	
Computer Time Less Than One Hour	

REFERENCES

1. M. L. Millburg, J. A. Hassberger, C. J. Boasso, "Natural Circulation Heat Transfer Testing with a Simulated Full-Scale LMFBR 217-Pin Electrically Heated Fuel Assembly", HEDL-TME-77-3, June 1977. (Availability: US DOE Technical Information Center).
2. F. C. Engel, R. A. Markley, B. Minushkin, "Buoyancy Effects on Sodium Coolant Temperature Profiles Measured in an Electrically Heated Mockup of a 61-Rod Breeder Reactor Blanket Assembly", ASME Paper 78-WA/HT-25.
3. M. H. Fontana, et al., "Temperature Distribution in a 19-Rod Simulated LMFBR Fuel Assembly in a Hexagonal Duct (Fuel Failure Mockup Bundle 2A): Record of Experimental Data", ORNL-TM-4113, September 1973. (Availability: US DOE Technical Information Center).
4. R. D. Coffield, H. P. Planchon, "LMFBR Natural Circulation Verification Program (NCVP) - Review of Experimental Facilities and Test Recommendations", WARD-NC-3045-1, July 1977. (Availability: US DOE Technical Information Center).
5. J. L. Gillette, J. V. Tokar, R. M. Singer, "Experimental Observations of Inter-Subassembly Heat Transfer Effects During Natural Circulation", Trans. Am. Nucl. Soc., 30, pp. 537-538 (1978).
6. "Covered Pressure Drop Flow Test/Crossflow Mixing Test", HEDL-TI-76049, November 1976. (Availability: US DOE Technical Information Center).
7. W. L. Thorne, "Pressure Drop Measurements in FFTF Fuel Vibration Tests", HEDL-TC-812, April 1977. (Availability: US DOE Technical Information Center).
8. W. L. Thorne, "Pressure Drop Measurements from Fuel Assembly Vibration Test II", HEDL-TC-824, April 1977. (Availability: US DOE Technical Information Center).
9. P. M. McConnell, "Clinch River Breeder Reactor Fuel Assembly Inlet/Outlet Nozzle Flow Tests", HEDL-TME-77-8, February 1977. (Availability: US DOE Technical Information Center).
10. C. Chiu, W. M. Rohsenow, N. E. Todreas, "Flow Split Model for LMFBR Wire Wrapped Assemblies", COO-2245-56TR, April 1978.
11. J. V. Miller, R. D. Coffield, "FØRE-2M: A Modified Version of the FØRE-II Computer Program for the Analysis of LMFBR Transients", CRBRP-ARD-0142, May 1976. (Availability: US DOE Technical Information Center).