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Validation of the HOTCHAN Code for Analyzing the EBR-II Driver Following Loss of Flow Without Scram*

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A series of experiments involving unprotected (no scram) loss-of-primary flow (LOF) down to natural convection was successfully conducted in February 1985 on the Experimental Breeder Reactor-II (EBR-II). The predicted and measured behavior of a special instrumented assembly, the XX09 fueled INSAT, is compared for the most severe test in the group to demonstrate the validation of the thermal-hydraulic code HOTCHAN. The particular test of interest in this paper was initiated at full power by tripping the primary and secondary pumps. These tests were part of the Shutdown Heat Removal Tests (SHRT)[1] being conducted in EBR-II. The reactor and balance of plant are extensively instrumented and measurements were recorded by a data acquisition system. The reactor and plant response confirm predictions that the driver fuel cladding can survive temperatures above the eutectic threshold for the transient following a station blackout without scramming the reactor. The in-core data provide an additional basis for validation of the recently developed HOTCHAN code for analyzing the thermal-hydraulic behavior of specific fuel subassemblies. In this paper the analytical model for HOTCHAN will be described as well as its relationship to the NATDEMO code.[2] The predicted behavior of the hottest driver subassembly is also discussed and compared with XX09 results.

The EBR-II reactor contains uranium metallic fuel in the driver and depleted uranium plus built-up plutonium in the outer blanket. An intermediate radial zone consists of a stainless steel reflector which operates at a very low specific power. The EBR-II driver region exhibits a rapid thermal response to power/flow upsets in contrast to a slow response of the radial re-

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flector and blanket. Since the EBR-II is a form of pool-type LMFBR with a large sodium inventory, the dynamic response of the primary pool/tank is very slow causing core inlet temperature changes to be minimal during the usual transient period of interest.

The analytical studies were conducted with a substantially validated systems code, NATDEMO, used in conjunction with the new HOTCHAN code. NATDEMO is used to predict the behavior of the reactor/plant and provides dynamic results of the thermal-hydraulic behavior of the driver region. The input required for the HOTCHAN analysis includes the transient fission and fission product decay power, the pressure drop (ΔP) across the core, the ΔP vs. flow characteristic of the central subassembly, the power and flow of the six surrounding drivers, and uncertainties to be applied to the nominal power generation and flow rate of the central subassembly. HOTCHAN is capable of predicting the temperatures of either a standard driver (91 elements) or a 61-element control subassembly including its outer bypass flow region. The driver assembly is radially divided into three regions to represent the 91 fuel elements; while in the case of the 61-element XX09 INSAT, two regions represent the fuel element bundle and the third region simulates the bypass region. HOTCHAN has the ability to modify the average-channel temperature profile obtained from NATDEMO according to a validated algorithm to produce a regional peaking factor that is dependent upon core loading details and the transient flow rate.[3] The combined temperature profile of adjacent subassemblies thus becomes the dynamic thermal boundary condition for the HOTCHAN intra-assembly analysis. The paper describes the analytical assumptions and structure of the model, the relationships defining the thermal and hydraulic boundary conditions, and the application of the model to a recent SHRT test.

The study demonstrated that a key objective of the SHRT-related model development has been met, namely that the core response to LOF/natural convection events in an LMFBR can be successfully predicted by lumped-parameter codes such as HOTCHAN and NATDEMO. It has also been found that the most important limitations in describing in-core thermal hydraulic behavior do not rest with intra-assembly details, but instead with modeling phenomena such as reactivity feedbacks, low-flow frictional behavior, and pool stratification.

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