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COMPARISON OF SCDAP/RELAP5/MOD3 TO TRAC-PF1/MOD1 FOR TIMING ANALYSIS OF PWR FUEL PIN FAILURES

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COMPARISON OF SCDAP/RELAP5/MOD3 TO TRAC-PF1/MOD1 FOR TIMING ANALYSIS OF PWR FUEL PIN FAILURES[a]

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ABSTRACT

A comparison has been made of SCDAP/RELAP5/MOD3- and TRAC-PF1/MOD1-based calculations of the fuel pin failure timing (time from containment isolation signal to first fuel pin failure) in a loss-of-coolant accident (LOCA). The two codes were used to calculate the thermal-hydraulic boundary conditions for a complete, double-ended, offset-shear break of a cold leg in a Westinghouse 4-loop pressurized water reactor. Both calculations used the FRAPCON-2 code to calculate the steady-state fuel rod behavior and the FRAP-T6 code to calculate the transient fuel rod behavior. The analysis was performed for 16 combinations of fuel burnups and power peaking factors extending up to the Technical Specifications limits. While all calculations were made on a best-estimate basis, the SCDAP/RELAP5/MOD3 code has not yet been fully assessed for large-break LOCA analysis.

The results indicate that SCDAP/RELAP5/MOD3 yields conservative fuel pin failure timing results in comparison to those generated using TRAC-PF1/MOD1.

1. INTRODUCTION

A licensing basis for nuclear reactors has been the loss-of-coolant accident (LOCA), with an assumed instantaneous release of fission products from the fuel into the containment. Certain equipment performance requirements, such as rapid closure of containment isolation valves, have been required to facilitate compliance with 10 CFR Part 100[1] regarding offsite radiological consequences. These fast closure times have placed a burden on valve design and maintenance.

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The objective of this research is to develop a viable methodology for calculation of the timing of the earliest fuel pin cladding failure, relative to the containment isolation signal, for LOCAs. The calculation was expected to show that certain isolation valves do not have to be closed as rapidly as now required, thus permitting more realistic licensing requirements.

In order to meet this objective, a calculational methodology was developed employing the FRAPCON-2[2], SCDAP/RELAP5/MOD3[3], and FRAP-T6[4] computer codes. Demonstration calculations were performed, applying this methodology to two plant designs, a Westinghouse 4-loop design analyzed using a Seabrook plant model and a Babcock and Wilcox (B&W) design analyzed using an Oconee plant model.

These calculations represent the first application of SCDAP/RELAP5/MOD3 and were performed using a preliminary version of the code, prior to completion of the code assessment efforts. In order to evaluate its adequacy, a single TRAC-PF1/MOD1[5] calculation was performed, duplicating the worst-case SCDAP/RELAP5/MOD3 calculation for the Seabrook analysis. This worst-case calculation consisted of a complete, double-ended, offset shear break of a cold leg, without pumped emergency core cooling systems (ECCS), and assuming that the main coolant pumps continued operating.

This paper discusses the methodology, assumptions, and a comparison of the results obtained for the worst-case calculations using SCDAP/RELAP5/MOD3 and TRAC-PF1/MOD1.

2. METHODOLOGY AND MODELING ASSUMPTIONS

A four-code approach, utilizing FRAPCON-2, SCDAP/RELAP5/MOD3, TRAC-PF1/MOD1, and FRAP-T6, was adopted for the analysis. This four-code approach provided a defensible calculational methodology for performing the calculation, incorporating a fully assessed calculational path, using FRAPCON-2, TRAC-PF1/MOD1, and FRAP-T6, and a parallel path, utilizing FRAPCON-2, SCDAP/RELAP5/MOD3, and FRAP-T6.

The FRAPCON-2[2] code was developed to calculate the steady-state response of light water reactor (LWR) fuel rods during long-term burnup. It calculates the temperature, pressure, deformation, and failure histories of a fuel rod as functions of time-dependent fuel rod power and coolant boundary conditions.

The FRAP-T6[4] code was developed to predict the performance of LWR fuel rods during operational transients and hypothetical accidents. It obtains initial fuel rod conditions by reading a file created by the FRAPCON-2 code and calculates all of the phenomena that influence the transient performance of fuel rods, with particular emphasis on temperature and deformation of the cladding.

The SCDAP/RELAP5/MOD3[3] code was developed for best-estimate transient simulation of LWR coolant systems under severe accident conditions as well as large- and small-break LOCAs. It is currently under development, and a preliminary version (cycle 7B) was used for the analyses.

The TRAC-PF1/MOD1 code[5] was developed for transient simulation of LWR coolant systems during large-break LOCAs. Version 14.3U5Q.LG was used for this analysis. This version was frozen in 1987 by the U.S. Nuclear Regulatory Commission (NRC) for use in the code scaling, applicability, and uncertainty evaluation (CSAU) study.[6] A broad assessment effort has been completed, which has demonstrated that the code is capable of addressing the entire large-break LOCA scenario (blowdown, refill, and reflood). Appendix III of the CSAU report[6] provides an extensive list of assessment reports applicable to this code.

SCDAP/RELAP5/MOD3 provides a considerable cost savings over TRAC-PF1/MOD1 for calculation of system thermal-hydraulic response under LOCA conditions. SCDAP/RELAP5/MOD3 is a relatively fast-running code that can execute from a UNIX workstation platform, as opposed to TRAC-PF1/MOD1, which requires a mainframe platform. Thus, SCDAP/RELAP5/MOD3 was chosen as the primary thermal-hydraulic code for the analysis.

A wide range of sensitivity cases were analyzed using SCDAP/RELAP5/MOD3 to assess the impact of break size, ECCS availability, and main coolant pump trip on the fuel failure timing. Due to the lack of code assessment for SCDAP/RELAP5/MOD3, a supplemental TRAC-PF1/MOD1 calculation, duplicating the case resulting in the shortest time to pin failure, was run to provide an evaluation of its accuracy.

The calculational methodology using SCDAP/RELAP5/MOD3 is illustrated in Figure 1. In these calculations, FRAPCON-2 was used to calculate the burnup-dependent fuel pin initial conditions for FRAP-T6; FRAP-T6 was used to calculate the initial steady-state fuel pin conditions for SCDAP/RELAP5/MOD3; SCDAP/RELAP5/MOD3 was run to obtain the system thermal-hydraulic boundary conditions, consisting of the fuel pin power distribution and thermodynamic conditions of the coolant channel; and FRAP-T6 was used to calculate the transient fuel pin behavior.

The supplemental calculation utilizes a similar methodology with the exception that SCDAP/RELAP5/MOD3 is replaced by TRAC-PF1/MOD1, as illustrated in Figure 2. Initialization of burnup-dependent variables for the TRAC-PF1/MOD1 fuel components is not necessary, since the code does not have a fuel performance model. However, a comparison of initial stored energy calculated by

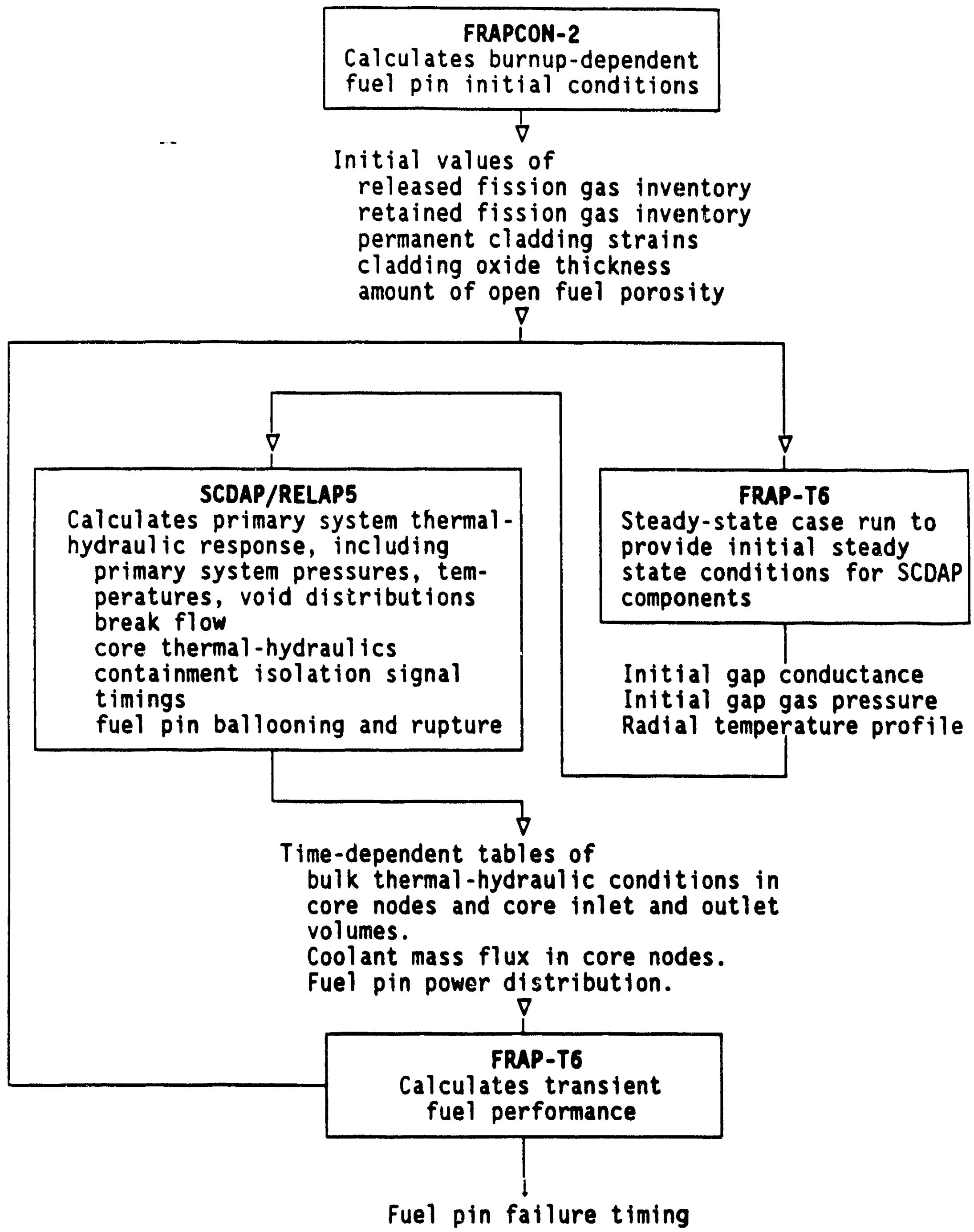

Figure 1. Flow chart of methodology using SCDAP/RELAP5/MOD3 thermal-hydraulic data.

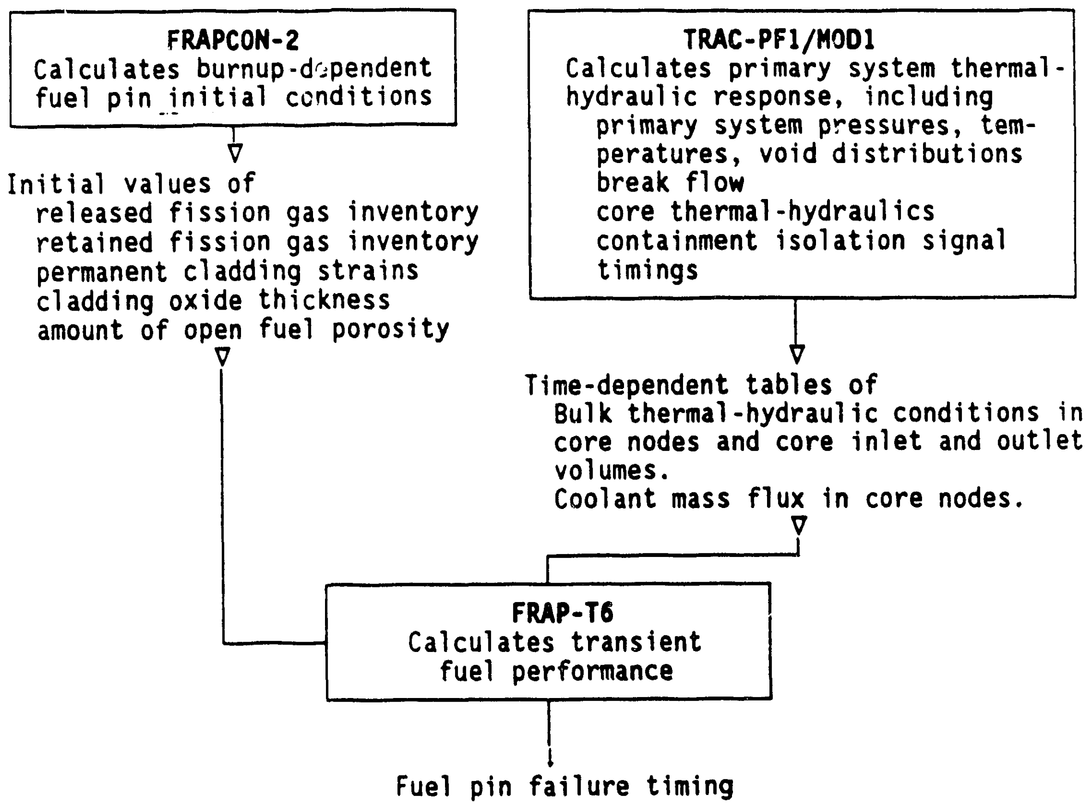

Figure 2. Flow chart of methodology using TRAC-PF1/MOD1 thermal-hydraulic data.

TRAC-PF1/MOD1 to that calculated by FRAP-T6 indicated reasonable agreement.

The calculations were performed assuming an equilibrium core operating at 102% core thermal power. Identical core nodalization was used for the SCDAP/RELAP5/MOD3 and TRAC-PF1/MOD1 core models, with the exception that the core bypass was lumped into the outer core region in the TRAC-PF1/MOD1 model. This nodalization consisted of a detailed three-channel core model with nine axial nodes. The hot channel included four fuel assemblies. The total power generated in the hot channel was assumed to be governed by the technical specification enthalpy rise hot channel factor.

The SCDAP/RELAP5/MOD3 model used for this analysis was adapted from a RELAP5/MOD2 deck created for station blackout transient analysis of the Seabrook nuclear power plant.[7] Several modifications were required to produce the model needed for this analysis. These included: addition of a detailed 3-channel, 9-axial-node core model, describing the hot channel and the central and outer core region; point kinetics modeling; SCDAP modeling; a simplified containment model; and a more detailed downcomer model.

A simplified containment model, consisting of a single RELAP5 volume with heat conductors representing steel and concrete surfaces, provided a fairly rough estimate of containment response. A more detailed treatment of containment response would require the use of a containment analysis code; however, results indicate that the containment isolation signal from the pressurizer low pressure trip trails the signal received from high containment pressure by only about 3 s. Due to the approximate nature of the containment pressure calculation, the pressurizer low pressure trip time was used to determine the containment isolation signal time.

The Seabrook TRAC-PF1/MOD1 model used for this analysis was derived from a TRAC-PF1/MOD1 model utilized for the CSAU study.[6] The modifications for this analysis included renodalization of the core region from five to nine axial nodes, describing the hot channel and the central and outer core region, removal of pumped ECCS, modification of the core power distribution, and replacement of containment pressure and decay heat boundary conditions. Boundary conditions for containment pressure and total core power history were obtained from the corresponding SCDAP/RELAP5/MOD3 calculation.

For each set of transient thermal-hydraulic conditions generated by either SCDAP/RELAP5/MOD3 or TRAC-PF1/MOD1, a series of 16 FRAP-T6 cases were run to determine fuel pin failure times for a range of fuel pin peak burnups (50, 35, 20, and 5 GWd/MTU) and axial peaking factors (2.32, 2.2, 2.0, and 1.8). A fundamental assumption governing this methodology is that the hot channel thermal-hydraulic conditions generated by SCDAP/RELAP5/MOD3 do not vary significantly for changes in hot pin axial power profile. In each case, the total fuel pin power, integrated over the length of the pin, is governed by the enthalpy rise hot channel factor and is therefore independent of the axial peaking factor applied.

The FRAPCON-2 and FRAP-T6 codes have not been assessed for analysis of high-burnup fuel (>35 GWd/MTU). However, results obtained for exposures above 35 GWd/MTU are in general agreement with expected trends. In addition, it is not anticipated that high-burnup fuel pins (>35 GWd/MTU) would be operating at power levels that would cause them to fail earlier than lower-burnup pins.

3. RESULTS

Figure 3 compares the transient results generated by SCDAP/RELAP5/MOD3 and TRAC-PF1/MOD1. The plots illustrate that good comparison was obtained for the break flow and resulting system depressurization. The SCDAP/RELAP5/MOD3 calculation reached the low pressurizer pressure setpoint at 3.73 s, only 0.11 s earlier than indicated by the TRAC-PF1/MOD1 calculation.[b] The flows from the accumulator, intact hot leg, and intact cold leg also compare well.

The largest deviation between results occurred after the accumulators emptied and discharged nitrogen into the system. In the SCDAP/RELAP5/MOD3 calculation, the accumulators were isolated as they approached an empty condition, in order to prevent code failure. In the TRAC-PF1/MOD1 calculation, however, as the accumulators emptied, nitrogen gas was discharged into the cold leg and vessel. This surge of noncondensible gas pressurized the upper downcomer, resulting in a surge of fluid into the core region. A surge can be seen as the broken loop accumulator empties at approximately 35 s and again as the intact accumulators empty at about 40 s. This surge of fluid is clearly seen in the hot channel mass flow at the midcore level and in the collapsed reactor water level. The downcomer void fraction plots indicate similar responses for voiding of the downcomer adjacent to the intact loops; however, the TRAC-PF1/MOD1 calculation indicates a quicker and more prolonged voiding for the downcomer quadrant adjacent to the broken cold leg.

The FRAP-T6 fuel pin failure times generated using SCDAP/RELAP5/MOD3 and the times generated using TRAC-PF1/MOD1 are summarized in Tables 1 and 2, respectively. The axial node in which failure occurred is given in parentheses. In cases where no fuel pin failure was predicted, the values given in the tables correspond to the transient time at the end of the calculation, prefixed by a "greater than" symbol (>).

Transient fuel performance results calculated by FRAP-T6 are shown in Figures 4 and 5 for the SCDAP/RELAP5/MOD3 and TRAC-PF1/MOD1 cases, respectively. The fuel cladding surface temperatures rise rapidly during the first few seconds, as the fuel surface heat flux is reduced due to core voiding. Fuel cladding temperatures peak at about 1100 K for the SCDAP/RELAP5/MOD3 calculation and at about 1000 K for the TRAC-PF1/MOD1 calculation. The fuel cladding temperatures then decline over the next few seconds as the fuel gives up its stored energy and fuel pellet temperatures drop, due to the reduced power generation. Eventually, the reduced heat transfer at the cladding surface produces a steady rise in cladding and fuel pellet

b. An additional delay of 2.0 s to account for instrument response is assumed for the analysis.

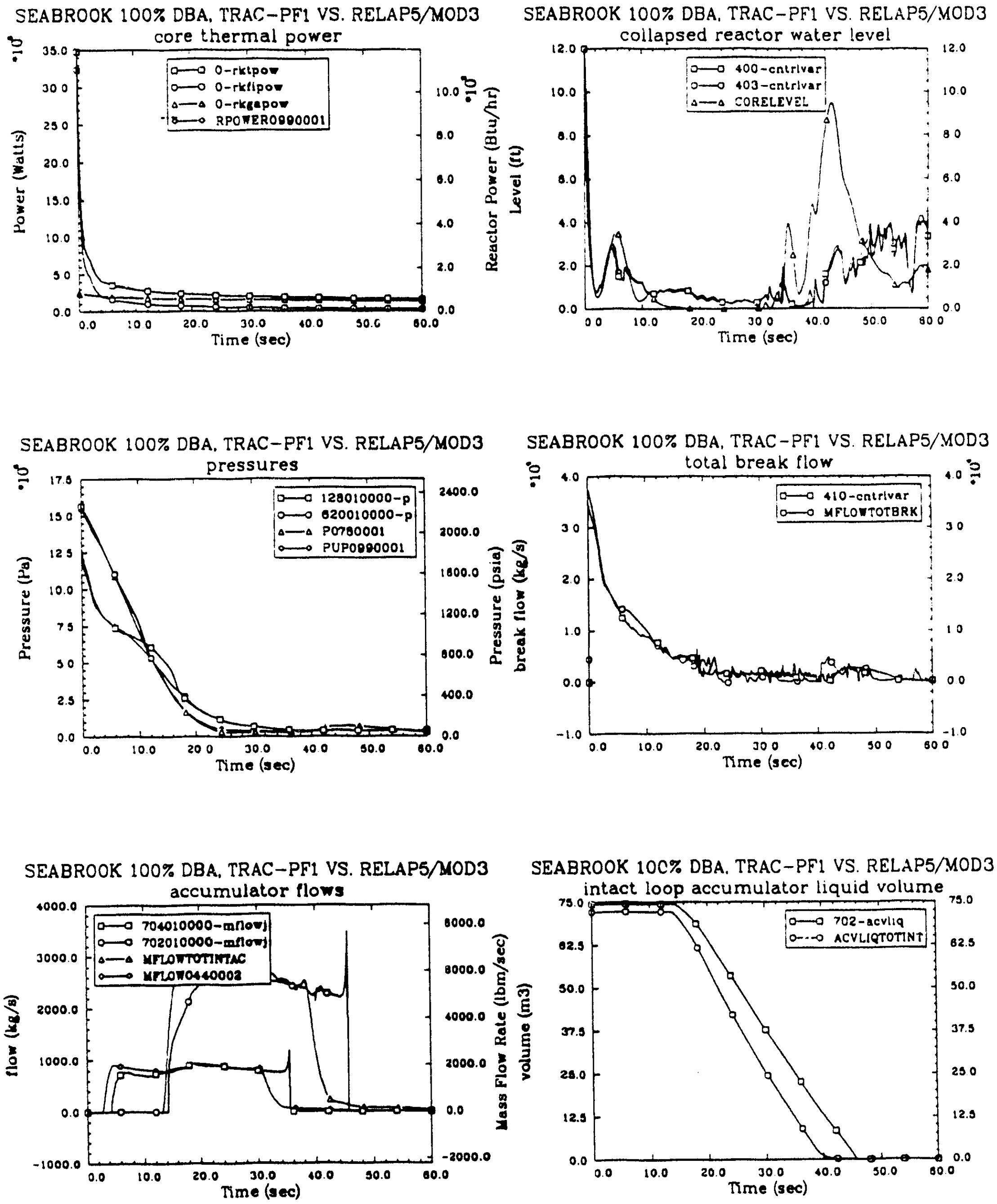

Figure 3. Plots of the transient results generated by SCDAP/RELAP5/MOD3 and TRAC-PF1/MOD1.

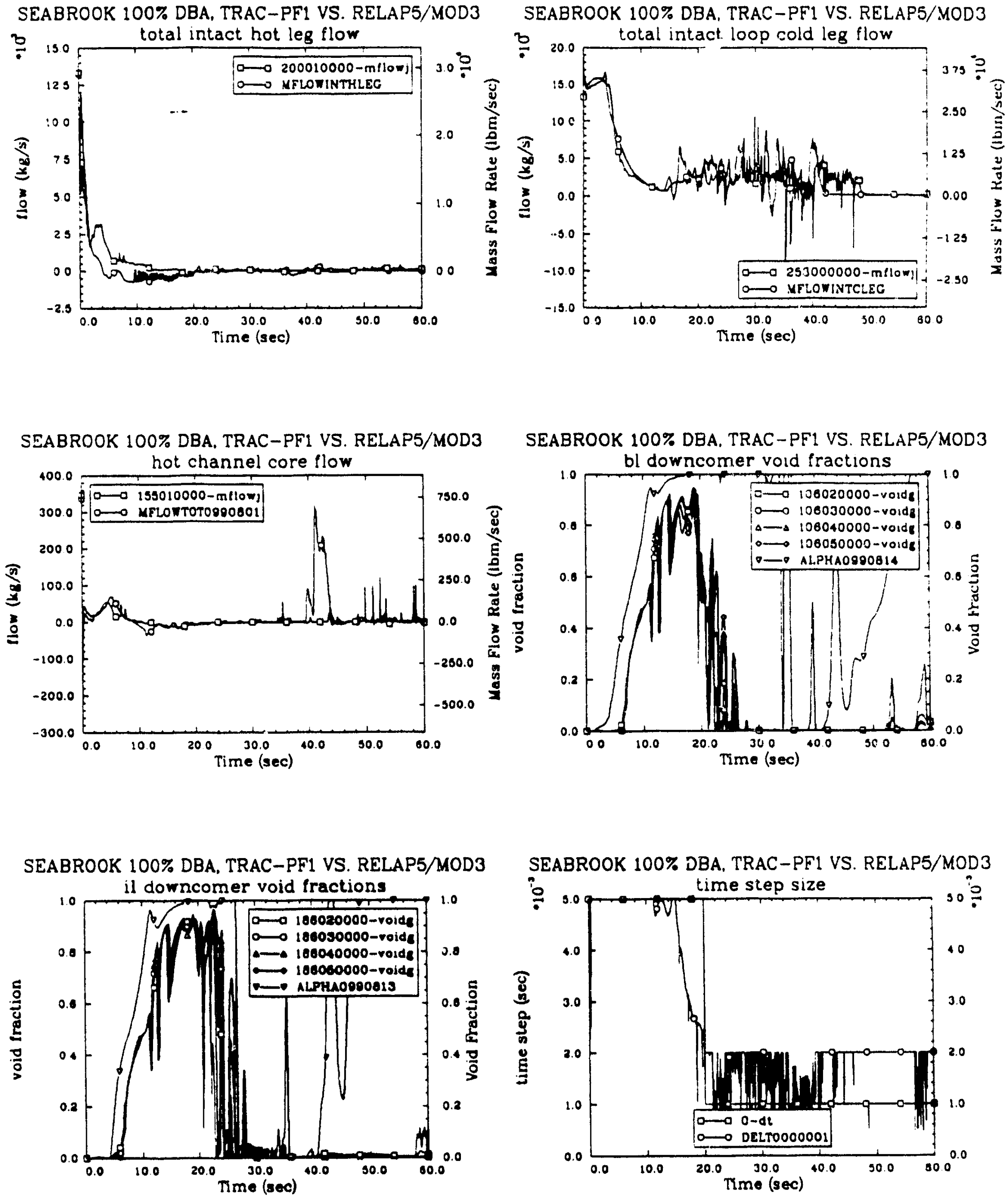

Figure 3. (continued)

Variable	Description
SCDAP/RELAP5/MOD3 Variables:	
0-rktpow	Total core thermal power (W)
0-rkfipow	Total core fission power (W)
0-rkgapow	Total core decay heat (W)
400-cntrlvar	Hot channel collapsed reactor water level (m)
403-cntrlvar	Core-average collapsed reactor water level (m)
128010000-p	Reactor upper head pressure (Pa)
620010000-p	Pressurizer dome pressure (Pa)
410-cntrlvar	Total break flow (kg/s)
704010000-mflowj	Accumulator flow for the broken loop (kg/s)
702010000-mflowj	Total accumulator flow for the intact loop (kg/s)
702-acvlig	Accumulator liquid volume for the intact loop (m^3)
200010000-mflowj	Total hot leg flow for the intact loop (kg/s)
253010000-mflowj	Total cold leg flow for the intact loop (kg/s)
155010000-mflowj	Hot channel flows at the core midplane (kg/s)
1060n0000-voidg	Broken loop downcomer void fraction for node n at the core midplane elevation
1860n0000-voidg	Intact loop downcomer void fraction for node n at the core midplane elevation
0-dt	Time step size (s)
TRAC-PF1/MOD1 Variables:	
RPOWER0990001	Total core thermal power (W)
CORELEVEL	Core-average collapsed reactor water level (m)
PUP0990001	Reactor upper head pressure (Pa)
P078001	Pressurizer dome pressure (Pa)
MFLOWTOTBRK	Total break flow (kg/s)
MFLOW0440002	Accumulator flow for the broken loop (kg/s)
MFLOWTOTINTAC	Total accumulator flow for the intact loop (kg/s)
ACQLIQTOTINT	Accumulator liquid volume for the intact loop (m^3)
MFLOWINTHLEG	Total hot leg flow for the intact loop (kg/s)
MFLOWINTCLEG	Total cold leg flow for the intact loop (kg/s)
MFLOWTOT990801	Hot channel flows at the core midplane (kg/s)
ALPHA0990814	Broken loop downcomer void fraction for node n at the core midplane elevation
ALPHA0990813	Intact loop downcomer void fraction for node n at the core midplane elevation
DELT0000001	Time step size (s)

Figure 3. (continued)

Table 1. Fuel pin failure times (s) calculated by FRAP-T6 using thermal-hydraulic conditions generated by SCDAP/RELAP5/MOD3.

Burnup/pf	5 GWd/MTU	20 GWd/MTU	35 GWd/MTU	50 GWd/MTU
2.32	29.1 (5)	29.7 (5)	27.7 (5)	24.8 (4)
2.2	34.4 (5)	36.7 (5)	35.8 (5)	32.5 (4)
2.0	44.5 (4)	48.4 (4)	43.6 (4)	43.6 (4)
1.8	> 60.0	> 60.0	> 60.0	> 60.0

Table 2. Fuel pin failure times (s) calculated by FRAP-T6 using thermal-hydraulic conditions generated by TRAC-PF1/MOD1.

Burnup/pf	5 GWd/MTU	20 GWd/MTU	35 GWd/MTU	50 GWd/MTU
2.32	> 60.0	41.4 (5)	41.3 (6)	34.9 (6)
2.2	> 60.0	> 60.0	41.4 (5)	41.2 (6)
2.0	> 60.0	> 60.0	> 60.0	> 60.0
1.8	> 60.0	> 60.0	> 60.0	> 60.0

temperatures. This temperature rise continues until water from the accumulators makes its way into the core region.

Cladding surface temperatures calculated by FRAP-T6 using SCDAP/RELAP5/MOD3 data are higher than those calculated using TRAC-PF1/MOD1 data. As shown in Figure 5, this deviation becomes even more apparent after about 40 s, due to the nitrogen induced flow surge that results in a quenching of the cladding for the TRAC-PF1/MOD1 calculation. The cladding surface temperatures calculated by SCDAP/RELAP5/MOD3 also begin to decrease after about 40 s, as flow from the accumulators reaches the core. However, the rapid quenching of the core was not predicted by the SCDAP/RELAP5/MOD3 calculation.

The zircaloy cladding undergoes a phase change starting at about 1050-1090 K and ending at about 1250 K. As a result of this phase change, the material properties of the cladding change rapidly over this temperature range. In the SCDAP/RELAP5/MOD3 case, pin failures was calculated to occur during this phase transition. In the TRAC-PF1/MOD1 case, pin failure occurred during the initial coolant surge, prior to reaching the phase transition temperature.

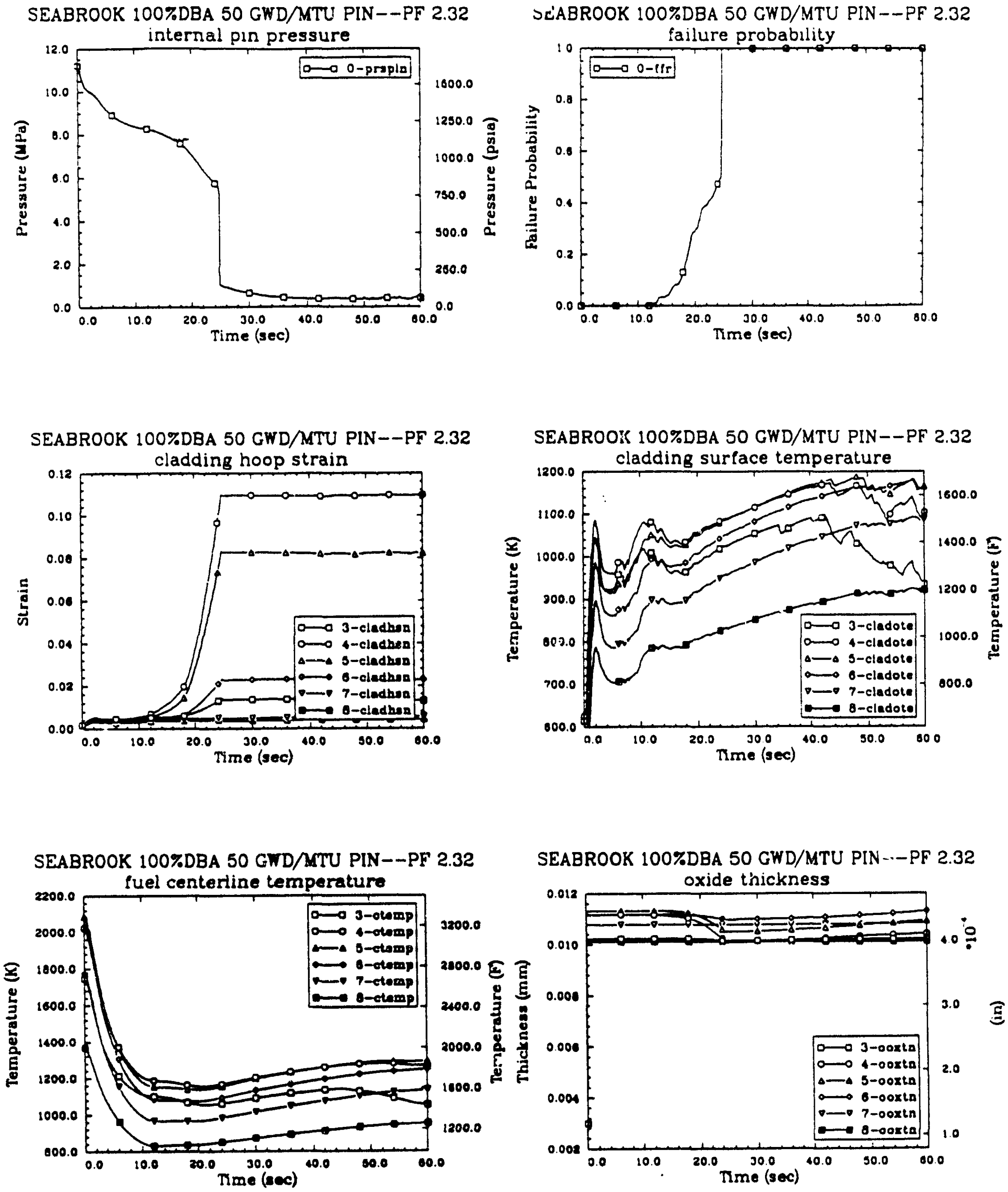

Figure 4. FRAP-T6 transient fuel performance results for the hot channel hot pin, peaking factor 2.32, 50 GWd/MTU burnup, axial nodes three through nine, using SCDAP/RELAP5/MOD3 thermal-hydraulic boundary condition data.

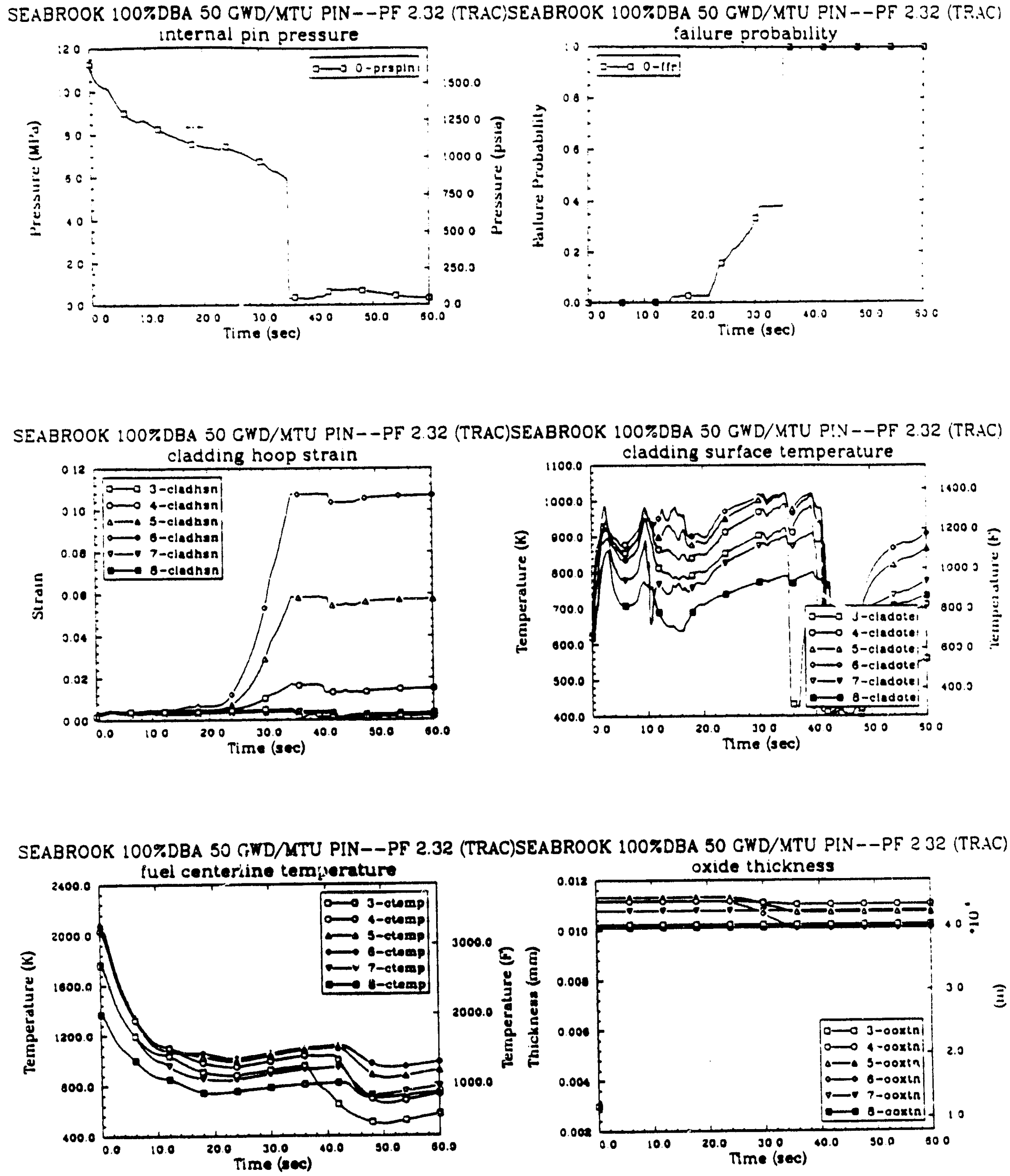

Figure 5. FRAP-T6 transient fuel performance results for the hot channel hot pin, peaking factor 2.32, 50 GWd/MTU burnup, axial nodes three through nine, using TRAC-PF1/MOD1 thermal-hydraulic boundary condition data.

It should be noted that the limitations of point kinetics prevent modeling of the relative differences in decay heat that would be associated with the differences in fuel pin exposure. For this reason, energy deposition in both the low- and high-exposure pins is identical throughout the transient and produces a conservative estimate of pin failure for the lower-burnup fuel pins.

4. CONCLUSIONS

The earliest fuel pin failure times calculated for a complete, double-ended, offset-shear break of a cold leg, without pumped ECCS, assuming the main coolant pumps continued operating were 24.8 s using SCDAP/RELAP5/MOD3 and 34.9 s using TRAC-PF1/MOD1 The corresponding containment isolation signal times were 3.73 s and 3.84 s for the low reactor coolant pressure trip, respectively. Assuming a 2.0-s delay for instrument response, the minimum interval calculated between initiation of containment isolation and failure of the first fuel pin becomes 19.1 s using SCDAP/RELAP5/MOD3 and 29.1 s using TRAC-PF1/MOD1.

These early fuel pin faiure times were obtained for fuel pins with 50 GWD/MTU exposure, operating at the technical specification limits. This represents a conservative result, since fuel pins with such a high exposure would not be operating at such conditions. The fuel pin failure time can increase significantly for both lower burnup and lower peaking factor. Based on this analysis, the methodology using SCDAP/RELAP5/MOD3 to provide thermal-hydraulic boundary conditions for FRAP-T6 appears to produce conservative results. An improved best-estimate approach would require detailed fuel-cycle specific information on the core power and exposure distributions.

5. REFERENCES

1. *Code of Federal Regulations*, 10CFR Part 100, "Reactor Site Criteria," January 1, 1991.

2. G. A. Berna et al., *FRAPCON-2: A Computer Code for the Calculation of Steady State Thermal-Mechanical Behavior of Oxide Fuel Rods*, NUREG/CR-1845, January 1981.

3. C. M. Allison et al. (Eds.), *SCDAP/RELAP5/MOD3 Code Manual*, NUREG/CR-5273, EGG-2555 (Draft), Revision 1, Volumes I-III, June 1990.

4. L. J. Siefken et al., *FRAP-T6: A Computer Code for the Transient Analysis of Oxide Fuel Rods*, NUREG/CR-2148, May 1981.

5. *TRAC-PF1/MOD1: An Advanced Best Estimate Computer Program for Pressurized Water Reactor Thermal-Hydraulic Analysis*, NUREG/CR-3858, April 1987.

6. Technical Program Group, *Quantifying Reactor Safety Margins: Application of Code Scaling, Applicability, and Uncertainty Evaluation Methodology to a Large-Break, Loss-of-Coolant Accident*, EGG-2552, NUREG/CR-5249, December 1989.

7. P. D. Bayless and R. Chambers, *Analysis of A Station Blackout Transient at the Seabrook Nuclear Power Plant*, EGG-NTP-6700, September 1984.

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