The RETRAN Newsletter

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INSIDE

NRC Review News

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The NRC Review of RETRAN-3D, which began in August 1998, continues. The most recent meeting was May 8, 2000, in Washington, D.C. and the objective was to obtain a detailed list of the remaining NRC review comments. The intent is to respond with the required information and to obtain as favorable of an SER (Safety Evaluation Report) as possible.

In the May meeting, the status of the responses to the previous two RAIs (Request for Additional Information) were discussed. While no formal conclusions were given at the meeting, the NRC staff will be providing specific information on the acceptability of the RETRAN-3D validation work and the responses to the second RAI within the next few weeks. EPRI/CSA will be responding to any resultant action items.

The NRC is currently drafting the SER, with completion planned this summer and more meetings with the NRC staff, the ACRS Subcommittee, and the full ACRS are expected. It has become apparent that the NRC staff is conducting a very thorough review, more typical of the scope and depth of review that has been associated with LOCA codes.

This approach can be expected on future reviews, and in particular on plant-specific applications of RETRAN-3D. The NRC has requested that EPRI, through the RETRAN Maintenance Group, guide RETRAN users on the transition from RETRAN-02 to RETRAN-3D and communicate the NRC's expectations. This will be communicated to the RETRAN community upon issuance of the SER later this year.

Previous to the May meeting, discussions were held with the NRC staff on December 16, 1999, and a submittal dated March 6, 2000, focused on providing responses to the remaining known NRC issues. The RETRAN review was also on the agenda for the March 15, 2000, ACRS Subcommittee meeting on Thermal-Hydraulic Phenomena.

From the Editor

This issue of the Newsletter features two technical articles on the use of RETRAN-3D and applications to multidimensional kinetics models. Both tasks were undertaken to demonstrate the ability of the code to perform analysis that were not possible with RETRAN-02 and to add to the code validation database.

We would like to encourage all members of the RETRAN community to contribute articles for the newsletter that show how the code is used in your group. We have had good support in the past from DUKE Energy, PSI, TEPCO, IBERINCO, UITESA, INVAP, KEPRI/KEPCO, and GPUN to name a few, and we invite all of you to submit any articles that might be interesting to the user community.

If you have any ideas or suggestions about how we might improve the newsletter, please give us a call or send an e-mail. If you have been considering an article, remember that there is a reward. We are offering the world famous RETRAN polo shirt. It has been shown to lower your golf score, make you look ten years younger, and makes a great conversation starter at class reunions and wedding receptions.

RETRAN-3D Multidimensional Kinetics Calculations for SPERT III E Tests 81 and 86

G. C. Gose, CSA

Two RETRAN-3D calculations were performed to simulate SPERT (McCardell, 1969) experimental reactor tests. The project was undertaken to expand the validation base for the RETRAN-3D multidimensional kinetics model and to demonstrate the ability of the code to reproduce experimental data. The calculated results and experimental data were compared for transient power and energy release. Even though it can be difficult to interpret the calculated results within the context of the experimental uncertainties, measured responses from significant and rapid reactivity changes in a nuclear core are rare. It is important to expand the validation base of the codes using these experiments when possible.

The SPERT reactor was a small oxide-fueled, PWR that was characterized as generally having characteristics of commercial PWRs at that time (1969). During the 1960s, reactivity accident tests were performed in the SPERT III E-Core Reactor under the SPERT experimental program. The program was designed to obtain the kinetics response data of reactivity accidents and evaluate computer codes that were used to predict reactor kinetics behavior. The SPERT experimental program is unique because it is one of the few facilities where prompt critical tests have been performed. The two tests that were analyzed, Tests 81 and 86, were initiated by rapid reactivity insertions from hot-standby and operating-power initial conditions, respectively.

These two tests were selected because they represented conditions from the SPERT high-initial-power test series. The high-initial-power tests were considered the most severe of the SPERT tests because the steady-state fuel temperatures were nearer the melting point, the reactor core would contain more energy, and the power burst energy release would be considerably large.

SPERT III Reactor Description

The SPERT III E-Core Reactor had the characteristics of an unborated commercial PWR, except for its small size. The E-core fuel is comprised of 4.8% enriched UO_2 fuel rods contained within stainless steel assembly cans. The fuel is in the form of 0.42-inch (0.010668-m) diameter pellets contained in stainless steel tubes. The core characteristics are summarized in Table 1.

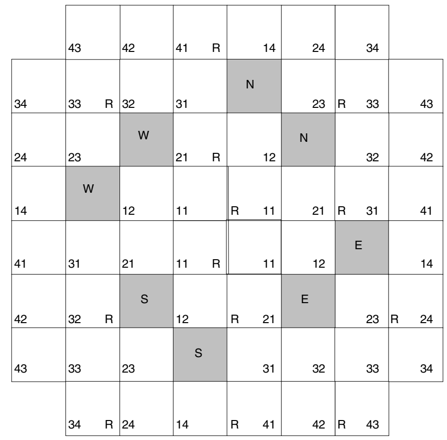


Figure 1. SPERT III E Core Configuration

The SPERT core coordinate system is given in Figure 1. The majority of the fuel rods are contained in 48, 3.0-inch (7.62-cm) square canisters each containing 25 fuel rods in a 5 by 5 rectangular array. To provide control, there are 12 smaller 2.5-inch (0.0635-m) square fuel assembly cans each containing 16 fuel rods in a 4 by 4 rectangular array. Four of these assemblies surround the central transient rod and the remaining eight form the fuel followers of the eight control rods. The cruciform shaped transient rod that is located at the core center is used for initiating the reactivity accidents.

Table 1. SPERT III E-Core Characteristics

Core Configuration	~ 26-inch (0.6602-m) Diameter	Cylinder
Assembly Types	48 25-Rod Assemblies 16 16-Rod Assemblies	
Fuel Rod Length	40.8 in. (1.03632 m)	
Active Fuel Length	38.3 in. (0.97282 m)	
Assembly Pitch	0.585 in. (0.014859 m)	
Fuel Rod Outer Diameter	0.466 in. (0.011836 m)	
Clad Thickness	0.020 in. (0.000508 m)	
Control Rods	8 Total – 2 per Quadrant Poison Section: 1.1684 m Fuel Follower: 1.1593 m	Poison Section: Type 18-8 Stainless Steel with 1.35 Wt% B-10
Transient Rod	1 Central Cruciform Shape Rod Poison Section: 0.9652 m	Upper Section: 18-8 Stainless Steel; Poison Section: 1.35 Wt% B-10

The four control rod pairs, indicated on Figure 1, by shading, are placed in the core in rotational symmetric locations. The transient rod is bottom inserted and the control rods are top inserted.

The RETRAN-3D thermal-hydraulic model consists of the active core region and thermal-hydraulic boundaries. The core consists of 48 active channels and a single bypass path. Within each channel, there are 30 axial nodes consisting of thermal-hydraulic control volumes and core conductors. There are 31 flow junctions per channel. Figure 2 shows the thermal-hydraulic model.

Reactor Modeling
Cross-section files that represent the core conditions for Tests 81 and 86 were supplied from a previous calculation. These files were in the format used by the NESTLE code (Turinsky, 1996). The NESTLE cross-section model is based upon polynomials for both controlled and uncontrolled states, and the RETRAN-3D cross-section routines were modified in order to use this format.

The RETRAN-3D core model for SPERT is very simple, consisting of three assembly types (Types 1-3) for the active fuel and one (Type 4) for the reflector. Each fuel assembly has 30 axial nodes for the active portion, and a single node for both the top and bottom reflectors. The active core is surrounded by one row of assemblies representing the radial reflector. The layout of assembly types is illustrated in Figure 3. For thermal-hydraulic calculations, the fuel assemblies are modeled by 48 flow channels, and the reflector region is model by a single bypass path. The RETRAN-3D channel map is shown in Figure 4.

Transient Modeling
The modeling of the control rods and the transient rod is crucial to the simulation of the reactivity accident tests. The excursions were initiated by dropping the transient rod poison section from the core.

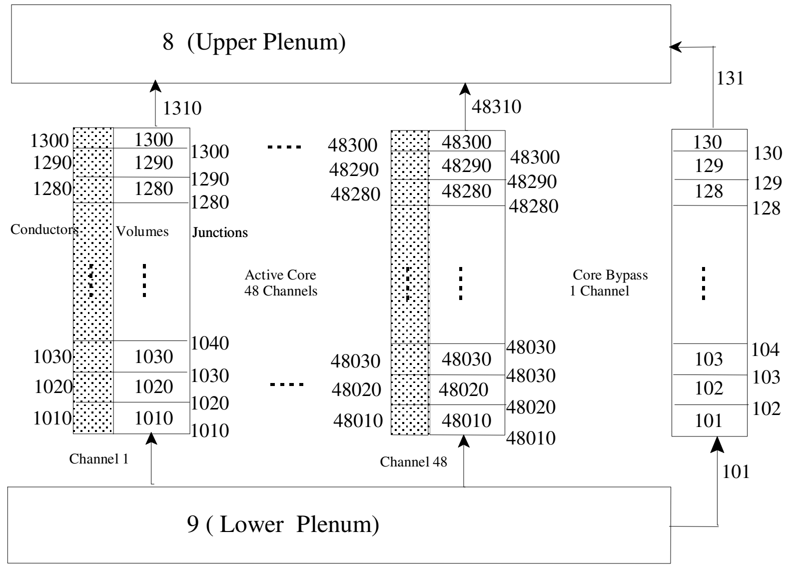


Figure 2. SPERT RETRAN-3D Model

To simulate the procedure, the first task is finding the initial transient rod position required for the power excursion. The transient rod tip was first positioned to the top of the bottom reflector, and then a criticality search was performed manually by moving the control rod positions until a critical condition was obtained. For Tests 81 and 86 the required reactivity insertion was $1.17. The transient rod was then inserted into the bottom of the core with the thermal-hydraulic feedback frozen at the initial conditions until the desired rod worth was obtained.

The transient was modeled by moving the transient control rod to the final position determined from the above procedure, and the core was maintained at critical by adjusting the position of the control rods. The transient rod was then dropped out of the core with the designed acceleration of 2000 in./sec^2 (50.8 m/sec^2) to simulate the reactivity insertion rate.

Test 81
SPERT Test 81 is one of the high-initial-power test cases. The steady-state core power is about 1 MW, which represents the hot-standby condition. The amount of reactivity insertion is $1.17. The comparison of the RETRAN-3D results with experimental data are plotted in

4	4	4	4	4	4	4	4	4	4
4	4	3	3	3	3	3	3	4	4
4	3	3	3	3	2	3	3	3	4
4	3	3	2	3	3	2	3	3	4
4	3	2	3	1	1	3	3	3	4
4	3	3	3	1	1	3	2	3	4
4	3	3	2	3	3	2	3	3	4
4	3	3	3	2	3	3	3	3	4
4	4	3	3	3	3	3	3	4	4
4	4	4	4	4	4	4	4	4	4

Figure 3. RETRAN-3D Assembly Layout

		40	40	40	39	38	38		
	41	24	23	22	21	20	19	37	
	41	25	9	8	7	6	18	37	
	42	26	10	2	1	5	17	37	
	43	27	11	3	4	16	36	48	
	43	28	12	13	14	15	35	47	
	43	29	30	31	32	33	34	47	
		44	44	45	46	46	46		

Figure 4. RETRAN-3D Channel Layout

Figures 5 and 6 which include transient power and total reactivity.

Given the experimental uncertainties, the agreement between the RETRAN-3D results and the experimental data are excellent. RETRAN-3D captured both the time and the magnitude of the peak power, also the pre- and post-peak power agreed very well with the experimental data. The added energy and the net reactivity (calculated based on the measured power and point kinetics equations) were correctly predicted.

Test 86

SPERT Test 86 is a full power test case. The initial core power is 19 MW. The amount of reactivity insertion is $1.17. The comparison of the RETRAN-3D results with experimental data were plotted in Figures 7 and 8 which include transient power and total reactivity.

In general, the results are similar to those obtained in Test 81 but the magnitude and the time of the calculated power peak (710 MW at 0.114 sec) are different from the experimental data. It is noted that the experimental power peak that was shown in Table VII of the SPERT report (McCardell, 1969) was 610 MW at 0.110 sec. The power peak plotted in Fig. D-70 of the report, however, was 598 MW at 0.122 sec. The reason for this discrepancy is unknown.

Conclusions

The RETRAN-3D code has been applied to the simulation of SPERT Tests 81 and 86, which are reactivity accident tests in a small experimental PWR. The transients were initiated by rapid reactivity insertions and experimental data included the subsequent time-dependent power and energy release. The results from RETRAN-3D indicate that the fundamental trends in both tests are correctly reproduced. The transient power and added energy were accurately predicted in Test 81, suggesting that the reactivity insertion rate be correctly modeled. The uncertainties in the fuel conduction model were not pronounced in the hot-standby case.

A significant factor in the interpretation of the results presented here is the degree of experimental uncertainty. For example, the documented standard deviation in reactor power for Tests 81 and 86 was ±10% (as much as 60 MW for Test 86), the time of the peak power, ±5 msec, and the uncertainty in the inferred reactivity (not a measured parameter) was about ±4%, translating to about $0.05 uncertainty in the reactivity insertion for the $1.17 cases.

To summarize the RETRAN-3D analysis work, the results show that the peak values and timing of the two cases are reasonably captured with the best comparison for the hot-standby initial power Test 81. More detailed sensitivity studies involving the fuel pin conductance may identify the significant parameters that affect the higher initial power case performance. It can be concluded that RETRAN-3D can produce good comparisons with experimental data from transient kinetics systems.

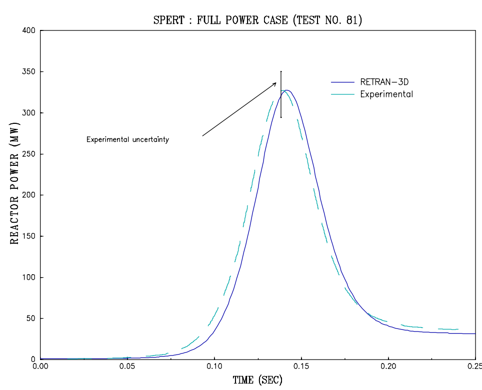


Figure 5. Test 81 - Reactor Power

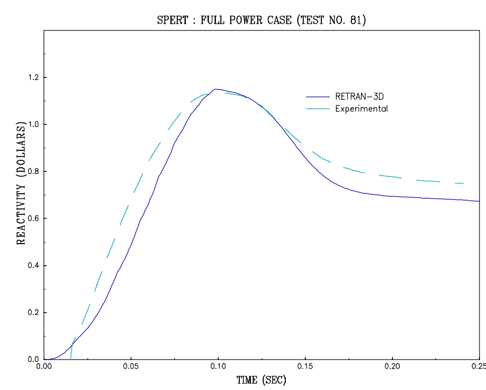


Figure 6. Test 81 - Net Reactivity

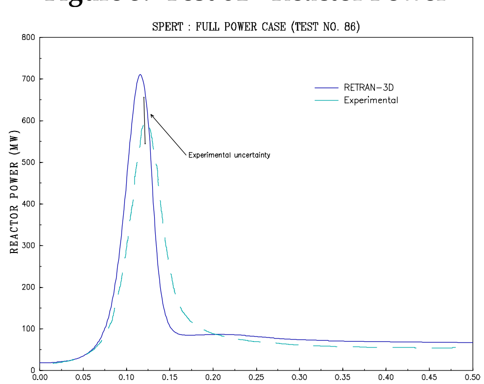


Figure 7. Test 86 - Reactor Power

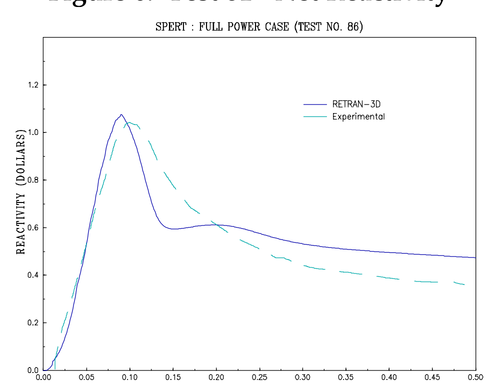


Figure 8. Test 86 - Net Reactivity

Pressurized Water Reactor Main Steam Line Break Benchmark

J. G. Shatford, CSA

A PWR main steam line break (MSLB) benchmark was jointly sponsored by the Organization for Economic Cooperation and Development and the U.S. Nuclear Regulatory Commission. The benchmark was established to demonstrate coupled three-dimensional neutronic/thermal-hydraulic behavior and compare predictions from different codes. The MSLB event is characterized by a significant radial power shift caused by asymmetric cooling and the assumption of a stuck-out control rod. The power shift can only be predicted with three-dimensional kinetics. One concern with the MSLB is a return to power after the scram due to reactivity addition from reactor coolant system temperature decrease.

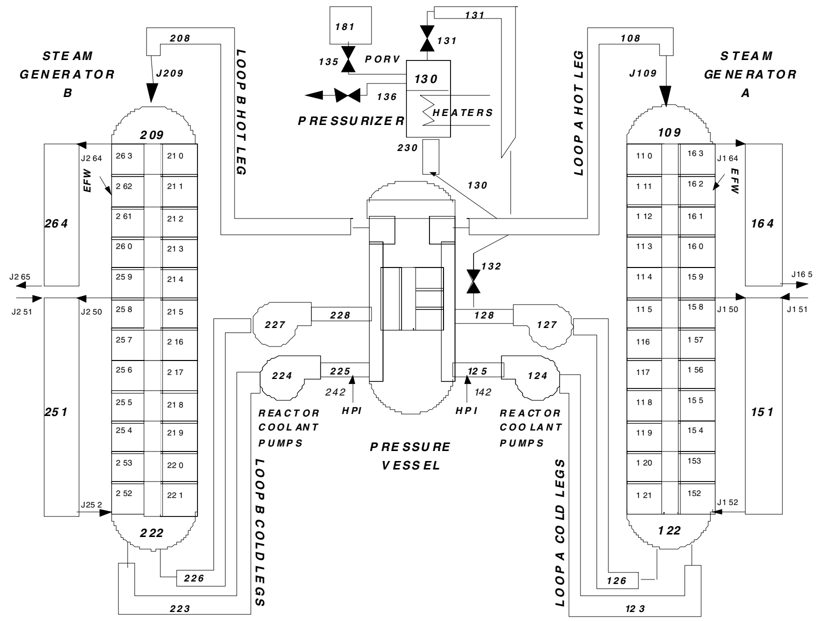


Figure 1. TMI RETRAN-3D Two-Loop Model Nodalization Diagram

Historically, this transient has been analyzed with point kinetics using conservative assumptions that compensate for the inability to simulate the power shift.

The benchmark consists of three phases: Exercise I – a system simulation using point kinetics, Exercise II – a core only simulation using three-dimensional kinetics, and Exercise III – system simulation using coupled three-dimensional neutronics/thermal-hydraulics.

The benchmark was intended to provide enough detail to ensure that all simulations use consistent input. The cross sections for Exercises II and III were provided. The specification defines most key parameters and modeling methods.

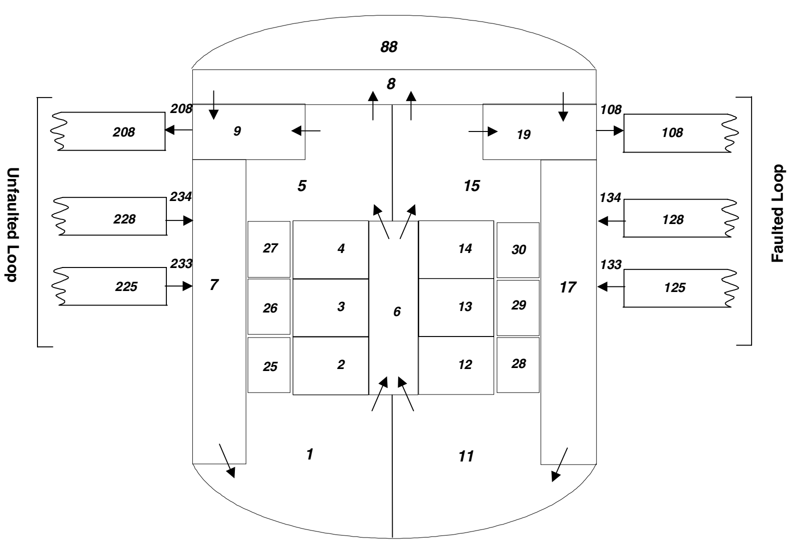


Figure 2. TMI RETRAN-3D Split Vessel Nodalization

The MSLB benchmark analysis was performed jointly by Computer Simulation & Analysis and GPU Nuclear. These analyses were performed with RETRAN-3D. The RETRAN code was modified to allow the cross-section data to be used as supplied.

The plant selected for the MSLB benchmark was the Three Mile Island Unit 1 (TMI-1). This is a two-loop plant with a once through steam generator on each loop. A nodalization diagram is shown in Figures 1 and 2. The core is split into faulted and unfaulted halves (Figure 2). The transient was initiated by a guillotine rupture of a steam generator steam line.

The MSLB results in the blowdown of a single steam generator, causing extreme cooling of the faulted loop but minimal cooling from the unfaulted steam generator. Flow from the two loops mix to some degree. However, the temperature in the half core on the faulted loop side decreases much more than the unfaulted side due to incomplete loop flow mixing. The benchmark defines how the loop flow mixing and asymmetric core cooling is to be modeled. The split plenums and core shown in Figure 2 allow simulation of incomplete loop flow mixing.

The moderator temperature coefficient is negative; consequently, the power will increase and tilt to the core side with the colder fluid. After the scram, there is the possibility of the net reactivity becoming positive if reactivity addition due to cooldown (from fuel and moderator) becomes larger than the scram worth.

The RETRAN-3D system power and component reactivities from Exercise I are shown in Figures 3 and 4. The power drops rapidly after the reactor trip and then at 20 seconds the power begins to increase and reaches a maximum around 60 seconds. As can be seen from the component reactivities in Figure 4, shortly after 50 seconds the reactivity contributions from the fuel and moderator exceed the control reactivity and the total reactivity momentarily becomes positive.

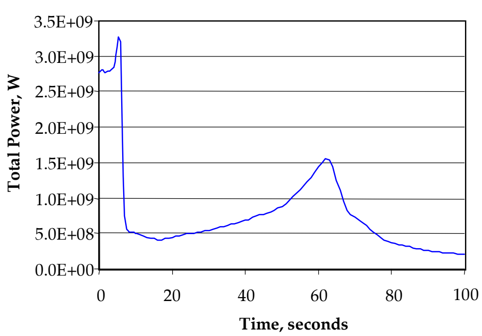


Figure 3. Exercise I Total Core Power

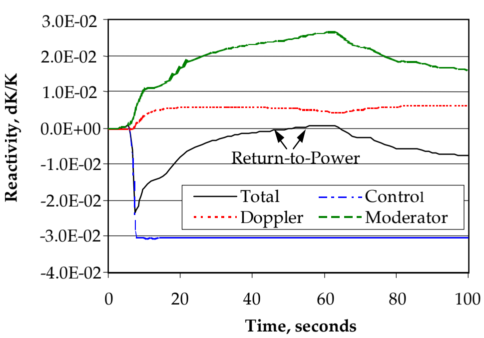


Figure 4. Exercise I Reactivity Components

Exercise II is a plenum-to-plenum model. The thermal-hydraulic conditions are completely defined by the benchmark, so Exercise II shows the effect of different three-dimensional kinetics modeling.

The benchmark provided cross sections for 24 axial nodes for each of the 177 fuel assemblies that are a function of fuel temperature and moderator density. Consequently, there are 4248 neutronic nodes within the reactor core. The benchmark defined thermal-hydraulic nodalization that was less detailed than the neutronic nodalization. The core was divided into 18 parallel “flow channels” between the lower and upper plenums. The split plenums were retained from Exercise I, so there are nine flow channels between the two plenum volumes on each side. The 18 flow channels are distributed in a symmetric radial manner. Since there are less flow channels than fuel assemblies, several fuel assemblies are placed within a single flow channel. The flow channel/fuel assembly map is shown in Figure 5 where each square represents a fuel assembly and the numbers (1 to 18) represent the flow channel.

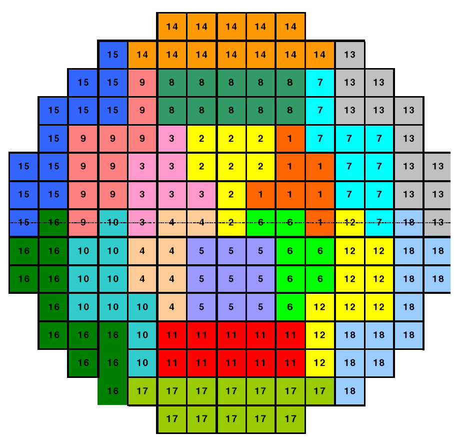


Figure 5. Exercise III 3-D Kinetics Core Fuel Flow Bundle/Flow Channel Map

The benchmark provided two separate sets of cross-section data. The first set, based on current licensing practices, uses a very conservative tripped rod worth. For these conservative assumptions, point kinetics models typically predict a return-to-power while three-dimensional kinetics models do not. A second set of cross sections based on a less conservative tripped rod worth were supplied as a better test of the coupled three-dimensional kinetics/ thermal-hydraulic codes.

The transient power response and total reactivity for Exercise II for both cross-section sets are shown in Figures 6 and 7.

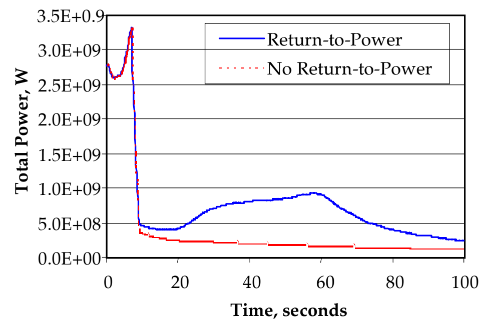


Figure 6. Exercise II Total Core Power

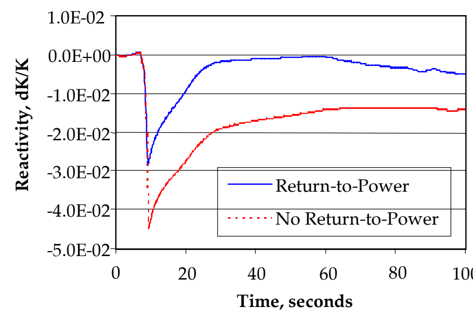


Figure 7. Exercise II Total Reactivity

For RETRAN-3D, Exercise II results varied significantly from the full system model (Exercise III) since Exercise II uses fixed flow rates, specified temperature and pressure boundaries and the scram time is predetermined. Exercise II was used mainly to verify code modifications allowing use of the specified cross-section data.

Exercise III uses the Exercise I system model with the three-dimensional kinetics core model from Exercise II. Consequently, this provides a direct comparison of the core behavior using the two reactor kinetics models.

There is a power increase after scram in the three-dimensional kinetics case but less than point kinetics (Figure 8) and there was not a return to power as can be seen in the total reactivity response (Figure 9). The response of the rest of the system was quite similar in the two cases. The different power response in the two cases is due to the ability to simulate the extreme radial flux shift to the faulted half of the core in addition to the more detail in the local feedback in the region of power increase.

Additional margin relative to a return-to-power concern can be achieved by using three-dimensional kinetics and more detailed thermal-hydraulic nodalization in the regions where the power excursion is expected to be the most significant.

More accurate thermal-hydraulic reactivity feedback can be achieved when several assemblies are not placed within a single flow channel. Further from the stuck rod, the number of fuel assemblies per flow channel should become less important. To optimize the number and placement of flow channels, sensitivity studies should be performed to define the point where more detail does not change the power response.

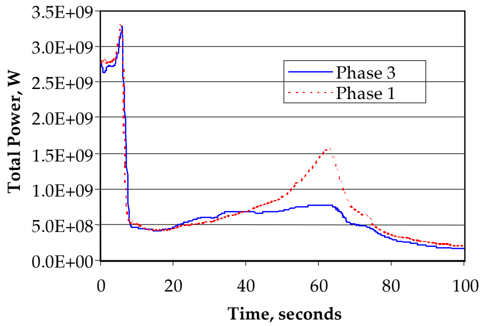


Figure 8. Exercise III/Exercise I Total Core Power Comparison

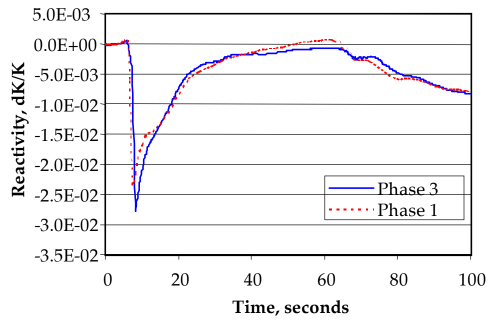


Figure 9. Exercise III/Exercise I Total Reactivity Comparison

Summary of RETRAN-02 Trouble Reports

The following is a summary of RETRAN-02 Trouble Report/Code Maintenance Activity as of April 30, 2000. There are 12 outstanding trouble reports. A list of trouble reports and the status can be obtained directly from the EPSC (1-800-763-3772). Additional information is available from the RETRAN-02 Trouble Report Page at http://www.csai.com/retran/r02trpt/index.html.

NO.	TROUBLE REPORT TYPE OF PROBLEM	CORRECTION NO.	IDENT	COMMENTS
354	Large Step Change in PHIR	***	*******	
376	Control Reactivity, No Motion	***	*******	
394	Anomalous Heat Trans. Behavior	***	*******	
408	OTSG Heat Transfer Problems	***	*******	
431	Failure in JN Properties	406	MOD005P3	
436	Prandtl Number is Discontinuous	405	MOD005P3	
437	Heat Transfer Logic/CHF	---	-------	Not a Code Error
438	Restart Failure/Pipe Transport	407	MOD005P3	
439	Decay Heat Input	***	*******	
440	Kinetic Energy/Time Dep Area	***	*******	
441	Anomalous Power Increase	---	-------	Not a Code Error
442	Poor Diagnostics	***	*******	
443	Liquid Region Work Term	***	*******	
444	Positive Slip Velocity	***	*******	
445	Boron Transport Inconsistency	***	*******	
446	Theory Manual Problem in Bubble Rise	***	*******	TH Manual Mod.
447	Smoothing Algorithm in SVOID	***	*******	
448	Decay Heat Input	---	-------	User's Manual Mod.
449	LOFW Transient Behavior	---	-------	Code Limitation
450	Momentum Flux Error Non Right Angles	***	*******	

Summary of RETRAN-3D Code Trouble Reports

A total of 209 trouble reports had been filed as of April 30, 2000. Of these, 177 reports have been resolved, while 32 remain unresolved. A summary of the unresolved trouble reports is shown below. Additional information for RETRAN-3D trouble reports is available at http://www.csai.com/retran/r3dtrpt/index.html.

NO.	TROUBLE REPORT TYPE OF PROBLEM	CORRECTION NO.	IDENT	COMMENTS
22	Problem using Wilson bubble rise model &	***	*******	
	error when using low power initialization		MOD001	(partial fix)
30	2-loop Oconee w/5-eq. fails in steady state	***	*******	
40	Results do not agree with data	***	*******	
48	Steady state fails after 6 iterations	***	*******	
		006	MOD001g	(partial fix)
52	MOC does not return to the initial temp.	***	*******	
54	MOC solution; no null transient for two-phase	***	*******	
60	Anomalous countercurrent flooding	***	*******	
70	Fails in subroutine DERIVS	***	*******	
81	Steady-state failure at iteration #6	***	*******	
116	Fails in steady-state initialization	***	*******	
122	Problems with EOS convergence	***	*******	(water packing)
142	Timestep selection causes 3-D kin to fail	***	*******	
144	TAUGL model doesn't apply for horiz. flow	***	*******	
145	SS fails to converge for low press. and flow	***	*******	
150	SS solution void fraction oscillation	***	*******	
152	Junct pressure lags vol pressure 1 time step	***	*******	Model limitation
164	3-D kinetics causes floating point exceptions	***	*******	
165	3-D kinetics unable to specify profile fit	***	*******	
	for subcooled boiling model			
168	Incorrect null trans w/3d Kin., mod ht & 5eq	***	*******	
170	PARCS numerics will not hold a null transient	***	*******	
174	5-EQ error in steam lines	***	*******	
181	No rod cusping treatment in 3D kinetics	020	*******	
182	Kinetics problem type is fixed at 3	***	*******	Model limitation 3D kinetics
190	Error when reversing from/to junc. w/ angle	***	*******	
197	>1 geometry data set is supplied on the CDI	***	*******	
198	Momentum flux error – if junction angles are	***	*******	
	not 0, 90, 180, 270			
200	SS failure for NCG (WAT0 error maybe WAT17)	***	*******	
201	SS failure when flow split option used	***	*******	
202	Error when pcrit reached during tran – 5-Eq	***	*******	
203	Pressurizer time step selectn when Przr solid	***	*******	
204	Impl Przr – Int reg HT and spray mdl errors	***	*******	
205	Channel model doesn't allow dyn gap cond mdl	***	*******	
206	PARCS inner iteration BICGSTAB fails	022	*******	ARROTTA Source
207	Xsec Extrapolation on DM is not supported	023	*******	ARROTTA Source
208	PARCS BC=2 (no return flux) is not allowed	024	*******	ARROTTA Source
209	SLB sample problem using direct mod. heating	***	*******	

Simulating a Second-Order Control System in RETRAN-02

G. C. Gose, CSA

Sometimes it is necessary to simulate the response from a second-order control system but the elements for this task are not directly available in RETRAN-02. A recent update to RETRAN-3D has added a second-order control block, STF, allowing the user to directly model the elements of a second-order system, given by the Laplace transform

$$1/(1 + a_2 s + a_1 s^2)\ .$$

The RETRAN-3D input allows the user to specify the block gain and the characteristic constants, a_1 and a_2.

But, what about the RETRAN-02 user? Can the behavior of a second-order system be simulated? Well, the answer is yes, with a little algebra. The Laplace transform of a LAG block looks like

$$1/(1 + t_1 s)\ .$$

By combining two LAG blocks together (that is, feeding the output of the first into the input of the second) an equivalent Laplace transform results

$$1/(1 + (t_1 + t_2)\, s + t_1 t_2 s^2)\ .$$

This looks very similar to the transform for the second-order system. An equivalent behavior to the second-order block can be obtained if the following constraints are followed

$$a1 = t_1 * t_2$$

and

$$a2 = t_1 + t_2\ .$$

Thus, subject to this constraint, users can combine LAG-LAG blocks to achieve a second-order system response.

As an example, let's look at a comparison of the method for a second-order system with the following constants,

$$4y'' + 4y' + y = x(t)$$

where

> x(t) is some arbitrary function, say a step or a ramp.

In Laplace transform space, this is equivalent to

$$1/(1 + 4s + 4s^2)\ .$$

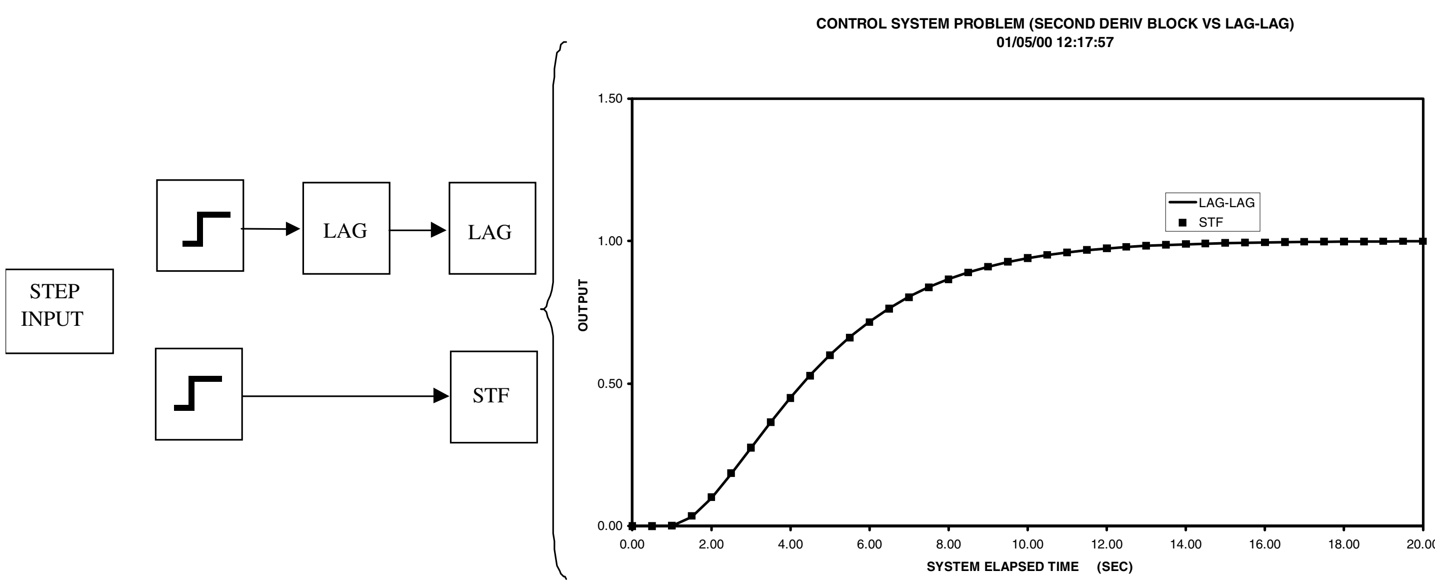
A little algebra shows that this is equivalent to

$$1/((1 + 2s)*(1 + 2s))$$

which is the same as the combination of two LAG blocks, each with time constants of 2.0.

For illustration, we have selected the response of a second-order block (STF) and the LAG-LAG block to a simple unitary step change at four seconds. The accompanying figure shows that for the chosen function (a unit step function) the two control system responses appear to be nearly identical.

Thus in many cases, one can model a second-order control system response in RETRAN-02 if a set of equivalent constants can be found that match the 'transformed' second-order system.



About This Newsletter

RETRAN Maintenance Program

The RETRAN Maintenance Program is part of a program undertaken by EPRI to provide for the support of the software developed in the Nuclear Power Division. The main features of the Subscription Service include:

- the code maintenance activities for reporting and resolving possible code errors,
- providing information to users through the User Group Meetings and this newsletter, and
- preparing new versions of RETRAN.

The RETRAN Maintenance Program now has 26 organizations participating in the program, including 22 U.S. utilities and 4 organizations from outside of the U.S. A Steering Committee, composed of representatives from the participating organizations, advises EPRI on various activities including possible enhancements for the code and the scheduling of future code releases. Information regarding the Maintenance Program can be obtained from

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EPRI
P. O. Box 10412
Palo Alto, CA 94303
lagee@epri.com or (650) 855-2106

Newsletter Contributions

The RETRAN Newsletter is published for members of the Subscription Service program. We want to use the newsletter as a means of communication, not only from EPRI to the code users, but also between code users. If this concept is to be successful, contributions are needed from the code users. The next newsletter is scheduled for August 2000 and we would like to include a brief summary of your RETRAN activities. Please provide your contribution to CSA, P. O. Box 51596, Idaho Falls, ID 83405, or to the E-mail addresses below by August 4, 2000. ***Contributors of a feature article will receive a RETRAN polo shirt.*** We are looking forward to hearing from all RETRAN licensees.

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Garry Gose	gcg@csai.com	
Pam Richardson	pam@csai.com	

The RETRAN Web Page is located at http://www.csai.com/retran/index.html.

Previous issues of the RETRAN Newsletter are available from the RETRAN Web Pages at http://www.csai.com/retran.

EPSC Contacts

EPSC Hours: 7 a.m. to 8 p.m. EST
EPSC Hotline: (800) 763-3772
EPSC Fax: (619) 453-4495

Calendar of Events

June 12-16	RETRAN Training Idaho Falls, ID
Oct. 17-19	User Group Meeting TU Electric

Please supply us with technical tips for our Tech Tips section and you will receive a RETRAN mouse pad.

Your contributions are greatly appreciated. We, EPRI and CSA, encourage everyone to participate in this newsletter.