CONF-851115--28

CONF-851115--28

DE86 002190

ADJOINT-BASED SENSITIVITY ANALYSIS FOR REACTOR ACCIDENT CODES

C. V. Parks

COMPUTING AND TELECOMMUNICATIONS DIVISION
at
Oak Ridge National Laboratory*
Post Office Box X
Oak Ridge, Tennessee 37831

Paper to be
Presented at the Winter Meeting of the
American Nuclear Society
San Francisco, California
November 10-14, 1985

*Operated by Martin Marietta Energy Systems, Inc., under Contract DE-AC05-840R21400 for the U.S. Department of Energy

ADJOINT-BASED SENSITIVITY ANALYSIS FOR REACTOR ACCIDENT CODES

C. V. Parks

This paper summarizes a recently completed study[1] that identified and investigated the difficulties and limitations of applying first-order adjoint sensitivity methods to reactor accident codes. The work extends earlier adjoint sensitivity formulations and applications[2-4] to consider problem/model discontinuities in a general fashion, provide for response (R) formulations required by reactor safety applications, and provide a scheme for accurately handling extremely time-sensitive reactor accident responses. The scheme involves partitioning (dividing) the model into submodels (with separate defining equations and initial conditions) at the location of the parameter($\vec{\alpha}$)-dependent discontinuity. Successful partitioning moves the problem dependence on the discontinuity location from the whole model system equations to the initial conditions of the second submodel.

The adjoint system and sensitivity expressions developed were initially tested with a simple model of a loss-of-decay heat-removal accident in a GCFR.[5] The accident code (FASTGAS) models the reactor kinetics, adiabatic heating of fuel pins, and core slumping with ordinary differential equations. Model discontinuities occurred at initiation of fuel melting (t_o) and at initiation (t_1) and termination (t_2) of melt-front controlled slumping. Sensitivity coefficients from the adjoint code were obtained using the base case of Ref. 5 for the threshold times--$t_o(\vec{\alpha})$, $t_1(\vec{\alpha})$, and $t_2(\vec{\alpha})$--, the peak power, and the energy at peak power. Since the accident transient evolved from a very slowly changing (order of minutes) transient for $t<t_o$ to a very rapidly changing (order of milliseconds) transient for $t>t_2$, the power and energy responses (at $t>t_2$) were extremely sensitive to

movements of the discontinuities and provided severe tests of the usefulness of the partitioning scheme. By comparing with direct recalculations, it was discovered that the partitioned adjoint code version (ADJGAS-P) produced the best sensitivity results because of numerical problems with the unpartitioned version (ADJGAS). For example, ADJGAS-P provided an accurate slumping worth sensitivity coefficient of 0.195 for the energy response, while ADJGAS provided an erroneous value of 1.512. The error is primarily a result of cancellation between two very large ($\sim 10^4$) integral contributions to the sensitivity, i.e., 5534.147 - 5532.635 = 1.512.

FASTGAS was also used to investigate various methods for evaluating function responses,[4] e.g., time- or space-dependent responses, that have significance in reactor safety applications. The methods studied involved (1) expanding the responses in a Fourier or Taylor series and (2) generating a spline fit based on adjoint results at fixed (functional) locations in the response profile. The energy profile from $t=t_2$ to $t=t_p$ (time at peak power) was used for the investigation. The spline fit and Legendre expansions proved to be the best methods in terms of both accuracy and ease of use. The economic advantage is with the spline fit because it required fewer adjoint calculations and the adjoint source was easier to formulate and implement.

Upon completion of the work with FASTGAS, an adjoint version (VENUS-ADJ) of the existing VENUS-II fast reactor disassembly code[6] was developed. VENUS-II is a two-dimensional coupled neutronic-Lagrangian hydrodynamics program for modeling core disruption from the onset of prompt-critical conditions to the termination of the power burst. The VENUS-ADJ code was validated with direct recalculations using a variety of end-of-excursion responses that are most closely linked to the damage potential--total energy, pressure at core center, energy in the molten fuel, and work

potentials from isentropic fuel vapor expansion to 1 atm (W_1) and to the cover gas volume. The base case excursion used for validation is from an earlier sensitivity study[7] and involved a \$100/s ramp rate in a sodium-voided FFTF core. Some results from VENUS-ADJ and direct recalculations are provided in Table 1 for the W_1. Sensitivity coefficients are presented for scale factor parameter changes while response differentials (i.e., perturbations) are presented for new program functions (e.g., vapor pressure functions) where the changes are time- and space-dependent. Also, response differentials are shown for the fuel melting temperature and initial fuel temperature distribution because scale factor changes in these parameters cause nonlinear changes in the initial energy distribution which are accounted for in VENUS-ADJ. The response changes from VENUS-ADJ for the vapor pressure functions only provide correct trends as opposed to accurate results because the base case adjoint solution is only a crude approximation to the importance value that would be obtained if the new vapor pressure function were used.

Results of this study indicate that adjoint methods are a viable alternative for reactor accident sensitivity analysis. Current efforts are directed towards using the exact response derivatives as part of an overall procedure for generating nonlinear response surfaces covering the large range of uncertainty found in postulated reactor accident models.

Table 1. Major Sensitivity Results for the Work Potential From Fuel Vapor Expansion to 1 atm, $W_i = 2.235 \cdot 10^8 J$

Parameter[a] α_j	$S^b_{\alpha_j}$	Average Response Change Error for $\pm$ 1% α_j, %	$\delta\alpha_j/\alpha_j$, %	Exact Response Change, % of Base Case Value	VENUS-ADJ Predicted Response Change, % of Base Case Value
Initial reactivity	7.603	2.2			
β	-7.012	2.3			
β_4	-2.493				
β_2	-1.469				
β_3	-1.329				
Doppler coefficient	-1.309	1.6			
Initial fuel density (total core)	1.239	3.0			
β_5	-1.140				
Reactivity insertion rate	0.945	0.3			
Initial fuel density (inner core)	0.869	5.1			
Fuel heat capacity	0.848				
β_6	-0.394				
Initial fuel density (outer core)	0.370	1.3			
Vapor pressure function #1				-27.2	-44.1
Vapor pressure function #2				-16.2	-23.6
Vapor pressure function #3				-31.4	-65.9
Vapor pressure function #4				-44.0	-132.8
Fuel melting temperature			0.1	7.3	8.9
			-0.1	-6.7	-8.9
			1.0	6.4	7.9
			-1.0	-5.7	-7.9
Initial fuel temperature distribution			0.1	-6.6	-8.7
			-0.1	7.1	8.7
			1.0	-4.7	-6.6
			-1.0	4.8	6.6
			10.0	10.7	10.6
			-10.0	-14.2	-10.6

[a] Several "parameters" with S_{α_j} values are actually scale factors for the noted input spatial distribution or function.

[b] $S_{\alpha_j} = \frac{dR}{d\alpha_j} \cdot \frac{\alpha_j}{R}$

References

1. C. V. Parks, "Adjoint-Based Sensitivity Analysis for Reactor Safety Applications," ORNL/CSD/TM-231, Oak Ridge National Laboratory (1985).

2. D. G. Cacuci, C. F. Weber, E. M. Oblow, and J. H. Marable, "Sensitivity Theory for General Systems of Nonlinear Equations," Nucl. Sci. Eng. 75, 88 (1980).

3. C. V. Parks and P. J. Maudlin, "Application of Differential Sensitivity Theory to a Neutronic/Thermal-Hydraulic Reactor Safety Code," Nucl. Tech. 54, 38 (1981).

4. D. G. Cacuci, "Sensitivity Theory for Nonlinear Systems. II. Extensions to Additional Classes of Responses," J. Math. Phys. 22, 2803 (1981).

5. E. E. Morris et al., "A Comparison Study of Response Surface Methodology and Differential Sensitivity Theory in Core Disruptive Accident Analysis," Proc. Int. ANS/ENS Topical Meeting on Probabilistic Risk Assessment, Vol. III, 953, American Nuclear Society (1981).

6. J. F. Jackson and R. B. Nicholson, "VENUS-II: An LMFBR Disassembly Program," ANL-7951, Argonne National Laboratory (1972).

7. R. B. Nicholson and J. F. Jackson, "A Sensitivity Study for Fast-Reactor Disassembly Calculations," ANL-7952, Argonne National Laboratory (1974).