QUALITATIVE ASSESSMENT OF THE FISSION PRODUCT RELEASE CAPABILITY OF ELOCA.MK5

M.E. KLEIN, L.N. CARLUCCI and V.I. ARIMESCU

Fuel Safety Branch
Reactor Safety Research Division
AECL, Chalk River Laboratories
Chalk River, Ontario, Canada K0J 1J0

ABSTRACT

A qualitative assessment of the fission product release capability of the ELOCA.Mk5 computer code was performed by simulating two transients from the sweep-gas experiment, FIO-133. Improved agreement between calculated and experimental trends in release was obtained by applying an interface pressure stress component to the pellet center. As well, results show that the current system for defining the reference temperature distribution for the thermal stress component is not always realistic. These results are being used in the development of a new, mechanistic pellet stress model.

INTRODUCTION

For many postulated accident conditions, high fuel temperatures can result in failure of the sheath and the subsequent release of fission products (FP) into the coolant. A fuel code possessing the capability to calculate FP release during transients is of potential use in the licensing of CANDU® reactors.

ELOCA.Mk5 is a FORTRAN-77 computer code developed to model the thermo-mechanical response and associated fission-product release behaviour of CANDU fuel elements during high-temperature LOCA-type transients[1,2]. It is an integration of the FREEDOM[3] gaseous fission-product release model into the ELOCA[1,4] fuel element thermo-mechanical code that allows the feedback mechanisms between fission-product release and thermo-mechanical response to be modelled. In the thermo-mechanical mode, Mk4S, the code is capable of performing multi-segment thermo-mechanical analysis of a CANDU fuel element, accounting for axial variations in thermalhydraulic conditions, element power, Zircaloy microstructure, fuel physical state, fuel-to-sheath heat transfer, and sheath oxidation and deformation.

For a single segment representation of a fuel element, the Mk5 mode of ELOCA.Mk5 couples

---

® CANada Deuterium Uranium. Registered Trademark.

the thermo-mechanical processes with calculations for grain growth, fuel swelling and the formation, diffusion and release of active and stable gaseous fission products. Gaseous FP release is calculated by FREEDOM, accounting for the time-dependence of both the temperature and hydrostatic stress state of the fuel pellet[3].

This paper describes the ELOCA.Mk5 pellet stress model and presents results from a qualitative assessment of the effects of its interface pressure and thermal stress components on FP release. To that end, ELOCA.Mk5 is applied in the Mk5 mode to simulate two transients from the sweep-gas experiment, FIO-133, conducted in the NRX reactor at Chalk River Laboratories in 1981[5]. The assessment of short-lived gaseous FP release trends is qualitative in nature as ELOCA.Mk5 models only Zircaloy sheath deformation and does not have the ability to account quantitatively for stainless steel cladding behaviour. Hence the calculated interface contact pressure is only approximate.

ELOCA.Mk5 FUEL PELLET STRESS MODEL

ELOCA.Mk5 models the fuel pellet as a number of concentric annuli whose height is adjusted to account for dishes, chamfers and tapers that are present for every fuel pellet. A set of unique thermo-physical properties is associated with each annulus. The fuel pellet is simulated by assuming that above a certain temperature the UO_2 deforms plastically, whereas below that temperature it cracks radially and behaves as an elastic solid. The pellet stress model assumes that the pellet consists of four zones[6,7]: a plastic core, a once-plastic zone that is cracked by thermal contraction, a bridging annulus, and a zone of wedge fragments cracked by thermal expansion.

The size of each zone is dependent on the previous power history, and is assumed to remain constant during the transient. The radius of the once-plastic zone corresponds to the plasticity-temperature isotherm at the maximum linear power applied to the pellet prior to the transient. The wedge fragments contact each other to form a bridging annulus around the once-plastic zone. The bridging annulus is the only non-plastic annulus in which all fragments contact each other; at both greater and smaller radii the fragments are assumed to be separated by cracks.

In FREEDOM, the release of fission product gases to the open void is caused not only by temperature changes but also by changes in the pellet stress distribution. The three components of fuel pellet stress used in the FREEDOM calculations are: the fuel-to-sheath interfacial pressure, the element internal gas pressure, and the transient thermal stress.

The fuel-to-sheath interfacial pressure is assumed to be focused through the radially cracked fuel fragments onto the bridging annulus of the pellet. The interfacial pressure hydrostatic stress component (σ_j^I) in a particular annulus is given:

$$\sigma_j^I = (P_I\, r_p)/(3\, r_j) \qquad \text{with} \quad r_j \geq r_b$$

where P_I is the interface pressure, r_p is the pellet radius, r_j is the radius of the jth annulus and r_b

is the radius of the bridging annulus. All annuli within the bridging annulus are assumed to be free of interfacial pressure stress.

The internal gas pressure hydrostatic stress component in a particular annulus is the gas pressure (ie. $\sigma_j^G = P_G$).

The ELOCA.Mk5 thermal stress model considers only elastic deformation and therefore does not directly model creep relaxation of the plastic region. Hence, the model does not account for the conversion of elastic strain into plastic strain. To compensate for this, the model requires specification of a reference pre-transient temperature distribution at which all thermal stresses are relaxed. In addition, thermal stresses are calculated only if transient temperatures exceed the reference temperatures. The reference temperature distribution used by ELOCA.Mk5 is the maximum temperatures reached during the pre-transient irradiation.

GENERAL DESCRIPTION OF THE FIO-133 SWEEP-GAS EXPERIMENT

Experiment FIO-133 consisted of an instrumented, stainless-steel-clad UO_2 fuel element, with upper and lower gas lines, irradiated at a linear power of about 55 kW/m to a burnup of approximately 80 MWh/(kg U) in the X-4 loop of the NRX reactor. The gaseous FP from the fuel were continuously removed from the element by a He-2% H_2 stream and collected in a sample chamber outside the loop, where they were measured by gamma spectrometry. A full description of the experiment, and a discussion of the experimental results, can be obtained elsewhere[5].

The experiment operated normally for two months, during which time the fuel assembly was cooled with pressurized water. Three dryout transients were conducted in the third month of operation. Stable steam-water (fog) cooling was established before and immediately after each dryout transient. The transients can be divided into three stages: heatup, steady high-temperature and re-wet. Heatup refers to the time spent changing loop conditions to put the element into dryout. Steady high-temperature is the time the element is in dryout, while re-wet consists of the cooling portion of the transient. For the third transient, a reactor trip occurred during the re-wet stage. Reactor power was maintained during the first two transients.

PREPARATION OF INPUT FOR ELESIM AND ELOCA

ELESIM Input

A version of ELESIM II Mod 10 containing the FREEDOM FP release model[3,8] was used to calculate the pre-transient steady-state condition of the fuel. The code simulates the irradiation history of the fuel and calculates the thermal-mechanical state, the pre-transient internal gas pressure and the inventory of fission products in the grain, on the grain boundaries and in the fuel-to-sheath gap. These calculated parameters are written to a data file, which specifies the initial fuel conditions for the ELOCA.Mk5 simulations.

The axial slots in the fuel pellet, cut to allow the passage of sweep-gas and the peripheral thermocouples, are modelled by ELESIM as circumferential grooves of equivalent volume.

The sweep-gas pressure of 3 MPa was set and maintained in ELESIM by specifying an arbitrarily high filling gas volume of 2.4×10^7 mm^3 at STP and a plenum void of 2770 mm^3/K. As a result, effects from changes in volume were minimized and the calculated gas pressure remained constant.

In the simulations, the power was reduced by 7.4% from 61.2 kW/m immediately prior to the second transient to provide a better match between the calculated and measured central fuel temperatures; and to improve the agreement with the reported average power of 55 kW/m[5]. This change was also made to the power history for the third transient.

ELOCA Input

For the second transient, the reactor power was kept constant with re-wet occurring during the last 15 minutes of the transient. The reactor was tripped at re-wet during the last 4 minutes of the third transient. These effects were accounted for in the ELOCA.Mk5 specification of relative power for each transient.

RESULTS AND DISCUSSION

For the second and third transients, the bulk of the measured FP release occurred at re-wet, with smaller amounts being released during the heatup and steady high-temperature stages of the transient (Table 1)[5]. Trends in calculated release for stable gases, ^{133}I and ^{138}Xe are compared to the measured values below.

ELOCA.Mk5

Figures 1 and 2 show that the ELOCA.Mk5-calculated FP release did not capture the trends in measured FP release for either transient. Most of the calculated FP release occurred during the heatup and steady high-temperature phases, with only a small burst in release seen during re-wet. In contrast, most of the measured FP release occurred during re-wet for both transients. Modelling the stainless steel sheath as Zircaloy would likely affect the magnitude of the calculated release but not the trends in release during the different stages of the transient.

Pellet Stress Model

As the release of FP to the gap depends not only on temperatures but also on the stress distribution in the pellet, further simulations were conducted to investigate the thermal stress and interface pressure stress components of the ELOCA.Mk5 pellet stress model.

Thermal Stresses Suppressed. Currently, ELOCA uses the maximum temperatures calculated by ELESIM during the steady-state irradiation as reference temperatures. The shortcomings to this approach are clearly evident for the third transient, where the irradiation history includes the second transient. The maximum ELESIM-calculated temperatures of the second transient are used as the reference temperatures in the ELOCA simulation of the third transient.

In addition, stress-relaxation calculations indicate that the stress in the fuel will be halved in approximately 200 seconds at 2500 K, and in about 2000 seconds at 2300 K. For experiment FIO-133, the calculated fuel temperatures exceeded 2500 K for 1600 seconds of the second transient, and for 1370 seconds of the third transient. This suggests that for either transient, the thermal stresses in the fuel would be relaxed by the time re-wet is reached, indicating that the fuel temperatures at re-wet are a more logical reference for these transients.

To investigate the effect of thermal stresses on FP release, ELOCA.Mk5 was modified to suppress the thermal stress calculation, leaving the interface pressure stress as the only time-dependent stress component acting on the fuel. For the second transient, suppressing the thermal stresses results in more FP release earlier in the transient (compare Figure 1 and 3). In addition, an increase in the final release of short-lived isotopes (^{138}Xe) was observed. It appears that for this case, the thermal stresses exert a compressive force on the pellet, which limits FP bubble growth on the grain boundaries, and hence, the interlinkage and subsequent release of FP to the gap.

Suppression of the thermal stress calculation in ELOCA.Mk5 had little effect on the release of fission products in the third transient (Figures 2 and 4). The insensitivity of the third transient to thermal stresses was due to the system of reference temperatures used in the thermal stress calculations. The reference temperature distribution used by ELOCA corresponds to the maximum ELESIM-calculated temperatures reached during the second transient. As a result, thermal stresses are calculated only for the few annuli whose temperatures exceed the reference temperatures.

Interface Pressure. The fuel-to-sheath interfacial pressure is assumed to be focused through the radially cracked fuel fragments onto the bridging annulus of the pellet. All annuli within the bridging annulus are assumed to be free of interfacial pressure stress. However, during the heat-up phase of a transient the pellet center will swell against the bridging annulus and create a stress field within it. In addition, the position of the bridging annulus is fixed and, therefore, cannot account for changes in the size of the plastic core as temperatures increase during the transient.

The effect of non-zero interface pressure stresses to the pellet center and relocation of the bridging annulus on FP release was studied by simulating the second transient with a modified version of ELOCA.Mk5. Simulation (i) applied the stress component, due to interface pressure at the bridging annulus, to all annuli lying within the bridging annulus. Since this stress component is always compressive, it will reduce the growth and interlinkage of fission-gas bubbles along the grain boundaries in these annuli, and will result in a reduction of FP release to the open void. Simulation (ii) is identical to simulation (i) except that the bridging annulus was relocated from the ELESIM-specified annulus 42 to annulus 75 to account for a larger plastic core resulting from the higher temperatures reached during the transient. For both simulations, the thermal stress calculation was suppressed.

Figure 5 shows that applying the interface pressure stress to the pellet center resulted in similar trends in calculated and measured FP release through the heatup and steady high-temperature stages and a burst in release during re-wet. The reintroduction of liquid coolant to the fuel element causes the fuel-to-sheath interface pressure to drop, resulting in a decrease in the stress field in the pellet center. This in turn allows some of the accumulated fission-gas bubbles on the grain boundaries to grow, interlink and be released.

Compared to simulation (i), relocating the bridging annulus towards the outer pellet edge (simulation (ii)) resulted in an increased FP release (Figure 6). Similar trends in calculated and measured release were observed. Relocating the bridging annulus towards the outer edge of the pellet focuses a lower interface pressure stress on the bridging annulus and results in a reduced compressive stress field in the pellet core, relative to simulation (i). This reduced stress field allows more fission-gas bubble growth and interlinkage, thereby resulting in increased FP release.

CONCLUSIONS

ELOCA.Mk5 is a FORTRAN-77 computer code that calculates the thermo-mechanical and gaseous fission-product release behaviour of CANDU fuel elements under transient conditions. This paper presents results from a qualitative assessment of the fission product release capability of the ELOCA.Mk5 code.

ELOCA.Mk5-calculated releases were compared with measured data from the sweep-gas experiment FIO-133. Calculated trends in release were not in good agreement with the experimental observation that most of the measured release occurred during re-wet. ELOCA calculated that most of the release would take place during the heatup and steady high-temperature stages.

These differences can, in part, be attributed to the modelling of the stainless steel-clad element, used in the experiment, as a Zircaloy-clad element in the ELOCA.Mk5 simulation. Stainless steel is stiffer and has a larger coefficient of thermal expansion than Zircaloy, in addition to a different high-temperature creep behaviour. As a result, the interface pressure resulting from deformation and differential fuel/sheath expansion was not an accurate representation of the experimental conditions and hence only a qualitative assessment of its effects was possible.

The qualitative assessment has demonstrated that better agreement of calculated releases with experimental trends can be obtained when the interface pressure component of the pellet stress is applied to the plastic core. In contrast, the effects of suppressing the thermal stress component of the pellet stress had an insignificant effect on the calculated trends in release. The assessment has identified possible areas of improvement in the treatment of the thermal stress and interface pressure components to make the stress model more general and consistent. In addition, the results have shown that the current criterion for defining the reference temperature distribution for the thermal stress component, which is based on elastic theory, is not always realistic.

ACKNOWLEDGEMENTS

This work was funded by the CANDU Owners Group (COG) consisting of the Canadian nuclear utilities (Ontario Hydro, Hydro Quebec and the New Brunswick Electric Power Commission) and AECL.

REFERENCES

(1) J.R. WALKER, J.W. DE VAAL, V.I. ARIMESCU, T.G. MCGRADY and C. WONG, "Use of ELOCA.Mk5 to Calculate Transient Fission Product Release From CANDU Fuel Elements", AECL Report, AECL-10591, 1992.

(2) M.E. KLEIN, L.N. CARLUCCI and V.I ARIMESCU, "Simulation of In-Reactor Experiments with the ELOCA.Mk5 Code", AECL Report, AECL-11133, 1994.

(3) L.D. MACDONALD, D.B. DUNCAN, B.J. LEWIS and F.C. IGLESIAS, "FREEDOM: A Transient Fission Product Release Model for Radioactive and Stable Species", AECL Report, AECL-9810, 1988.

(4) V.I. ARIMESCU, "An Elasto-Plastic Model for Mechanical Contact Between the Pellets and Sheath in CANDU Nuclear Fuel Elements", AECL Report, AECL-10426.

(5) I.J. HASTINGS, C.E.L. HUNT, J.J. LIPSETT and R.G. GRAY, "Transient Fission Product Release During Dryout In Operating UO_2 Fuel", AECL Report AECL-7832, 1982.

(6) M.F. NOTLEY, "Calculation of Fission Gas Pressures in Operating UO_2 Fuel Elements", Nuclear Applications, **3** (1967) 334-342.

(7) J.H. GITTUS, "Theoretical Analysis of the Strains Produced in Nuclear Fuel Cladding Tubes by the Expansion of Cracked Cylindrical Fuel Pellets", Nucl. Eng. Des., **18** (1972) 69-82.

(8) M.F. NOTLEY, "A Computer Code for Predicting the Performance of Nuclear Fuel Elements", Nuclear Technology, **44** (1979) 445-450.

TABLE 1 PERCENTAGE RELEASE OF ^{138}Xe MEASURED DURING THE SECOND AND THIRD TRANSIENTS OF THE FIO-133 EXPERIMENT[5]

	Heatup	Steady High-Temperature	Re-wet
Second Transient	0.03	0.09	0.44
Third Transient	0.10	0.09	1.30

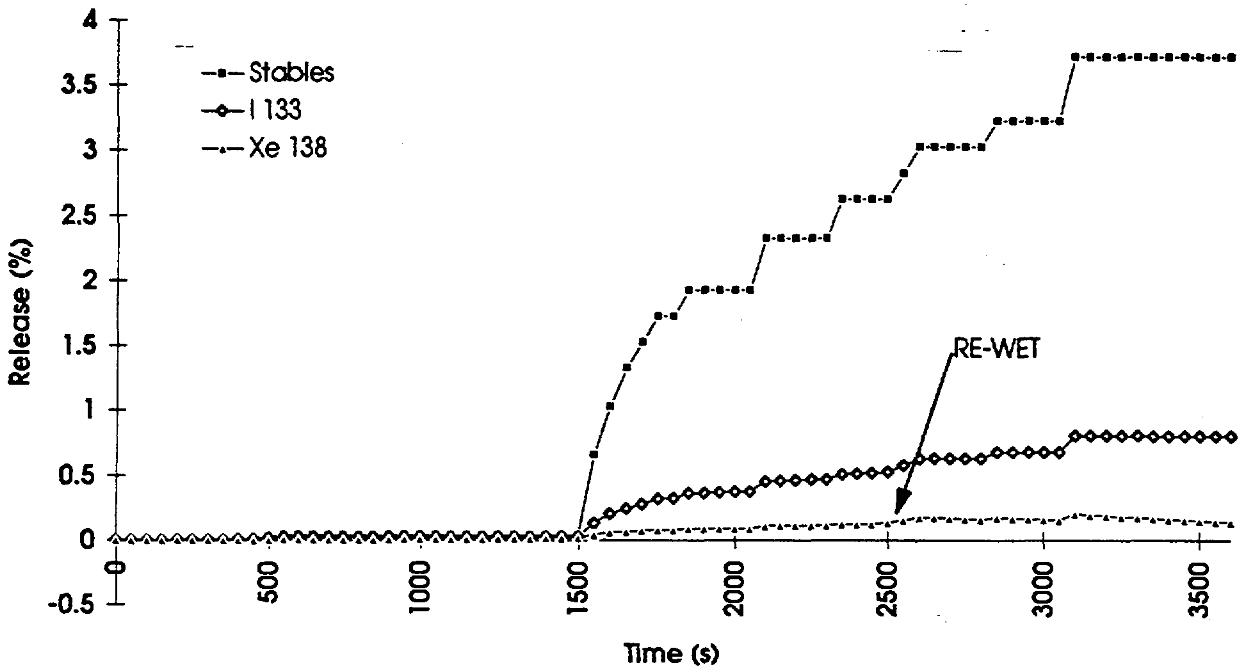

FIGURE 1 CALCULATED FP RELEASE DURING SECOND TRANSIENT

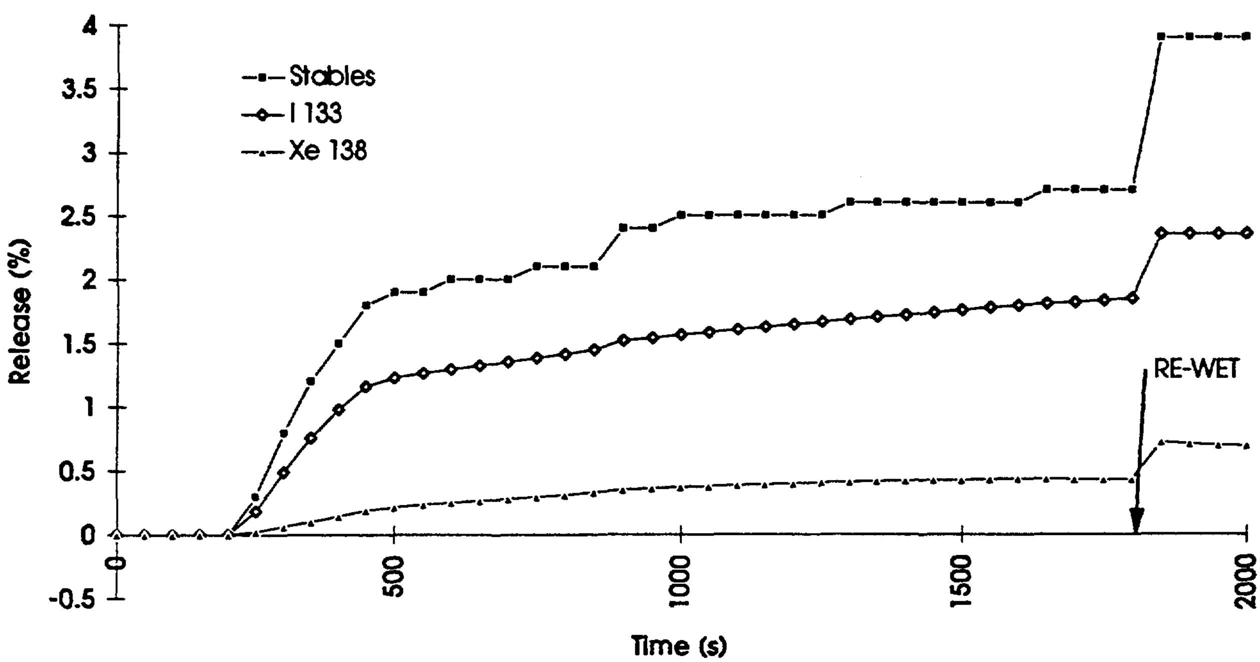

FIGURE 2 CALCULATED FP RELEASE DURING THIRD TRANSIENT

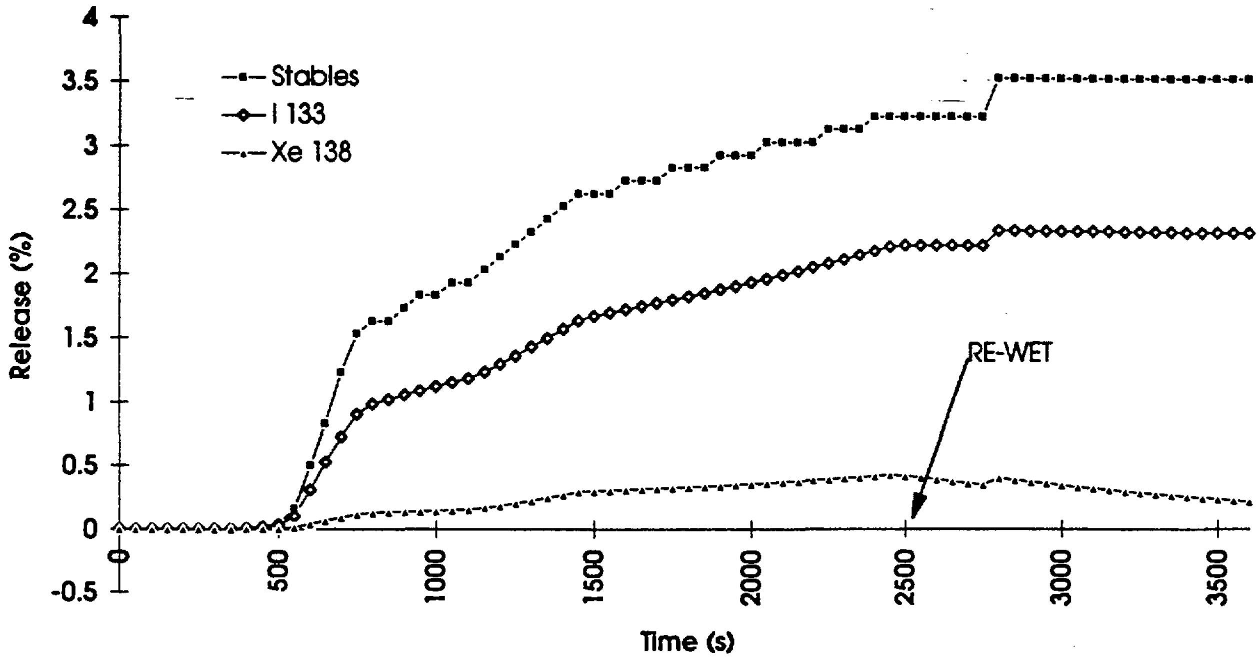

FIGURE 3 CALCULATED FP RELEASE DURING SECOND TRANSIENT WITH THERMAL STRESSES SUPPRESSED

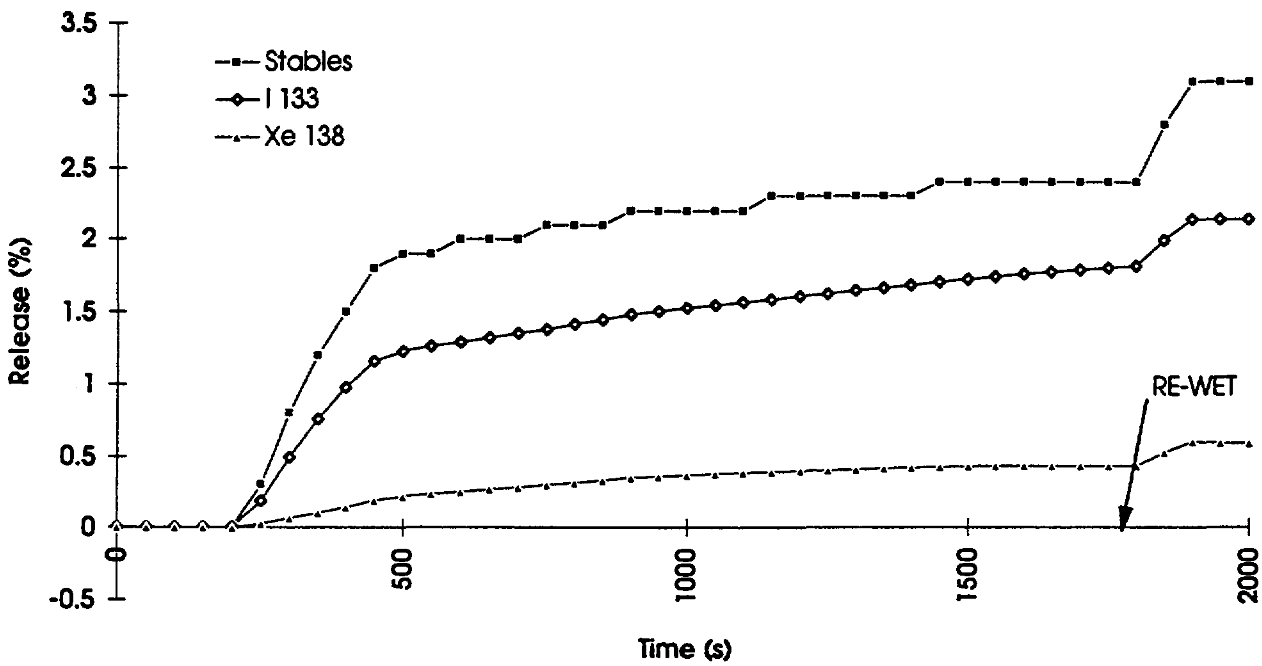

FIGURE 4 CALCULATED FP RELEASE DURING THIRD TRANSIENT WITH THERMAL STRESSES SUPPRESSED

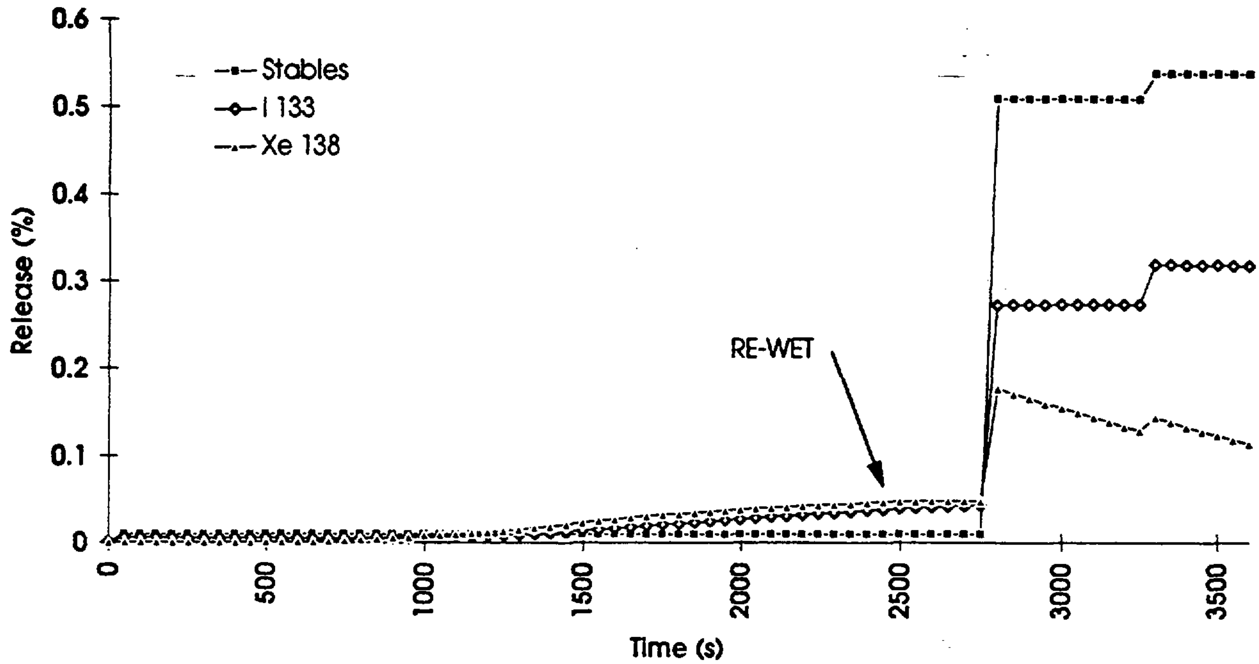

FIGURE 5 CALCULATED FP RELEASE DURING SECOND TRANSIENT, WITH INTERFACE PRESSURE TO PELLET CENTER

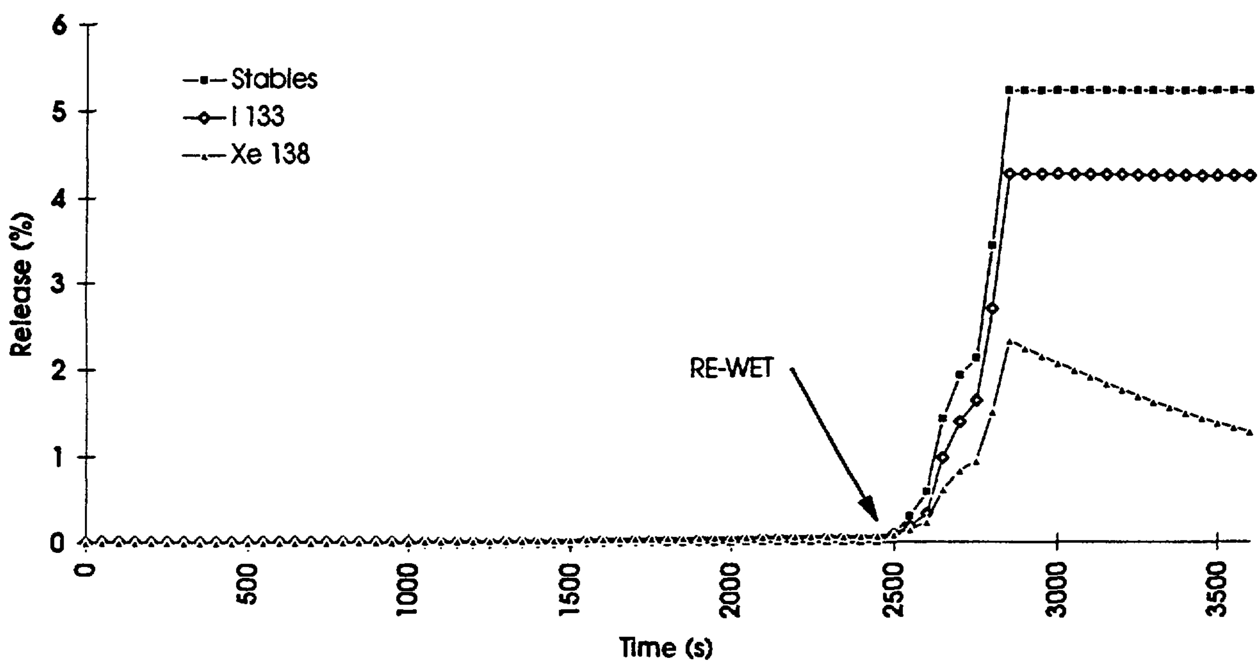

FIGURE 6 CALCULATED FP RELEASE DURING SECOND TRANSIENT, WITH INTERFACE PRESSURE TO PELLET CENTER AND RELOCATED BRIDGING ANNULUS