Reactivity Initiated Accidents and Loss of Shutdown - 20 Years Later

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ABSTRACT

A review of the safety of Ontario's nuclear power reactors was conducted in 1987 after the Chernobyl accident. As part of this review an analysis was performed of a Loss of Coolant Accident in a Pickering A unit with coincident failure to shutdown. This analysis showed that the power excursion was halted by channel and calandria vessel failures leading to moderator fluid displacement. The containment structure did not fail and, at worst, might suffer minor cracking at the top of the dome of the reactor building. Overall the dose consequences of such an accident were no worse than the limiting design basis dual failure event. In the intervening twenty years following this analysis, significant experimental information has been obtained that relates to power pulse behaviour. This information, together with conservatisms in the original analysis, are reviewed and assessed in this paper. In addition, the issue of reactivity initiated events in other reactor types is reviewed to identify the reactor design characteristics that are of importance in these events. Contrary to popular belief the existence of positive coolant void reactivity is not as significant a factor as it is sometimes stated to be. On balance, with appropriate design measures, no one reactor type can be claimed to be "more safe" than another. The underlying basis for this statement is articulated in this paper.

INTRODUCTION

After the reactivity initiated accident in the Chernobyl Unit 4 reactor a review of the safety of Ontario's nuclear power reactors was conducted during 1987 by Prof. Kenneth Hare at the request of the Ontario Minister of Energy (Ref.1). As part of this review an analysis was performed of a Loss of Coolant Accident in a Pickering A unit with coincident failure to shutdown. This analysis, conducted by Ontario Hydro (Ref.2,3), with independent analysis of channel failures performed by Argonne National Laboratories (Ref. 4), showed that the power excursion was halted by channel and calandria vessel failures leading to moderator fluid displacement. The containment structure did not fail and, at worst, might suffer minor cracking at the top of the dome of the reactor building. Overall the dose consequences of such an accident were no worse than the limiting design basis dual failure event.

This analysis has been judged by some to be speculative and, as a consequence of this judgment, the positive results showing limited consequences have tended to be viewed skeptically, if not negatively. However, in the intervening twenty years following this analysis, a large body of relevant experimental information relating to aspects of fuel and fuel channel behaviour under power pulse conditions has been accumulated. This information includes the collation and systematic assessment of a wide range of experimental fuel behaviour and failures under large power pulse conditions, information on fuel channel failures at high pressure, and information on the energetics of hot fuel-moderator interaction. This information is reviewed and analyzed in this paper to establish its relevance. Additionally, the significant conservatisms incorporated in the 1987 loss of shutdown analysis are reviewed and assessed. Based upon this re-assessment it is demonstrated that the analysis has indeed stood the test of time and remains both relevant and conservative for extremely low frequency reactivity events resulting in early core disassembly in CANDU reactors.

In addition to reassessing the loss of shutdown scenario, the issue of reactivity initiated events in other reactor types is reviewed with a view to identifying the reactor design characteristics that are of importance in these events. The position is advanced that, contrary to popular belief, the existence of positive coolant void reactivity is not as dominant a factor as it is sometimes stated to be. On balance, with appropriate design measures, no one reactor type can be claimed to be "more safe" than another. The underlying basis for this statement is articulated in this paper.

PART I: REASSESSMENT OF THE LOSS OF SHUTDOWN EVENT IN A PICKERING A UNIT

Scope and Purpose of the Reassessment

Since the 1987 analysis was performed a number of significant pieces of information have become available that are of direct relevance to the results of the analysis. Some of the important items are;

- Improved estimates of coolant void reactivity and delayed neutron fractions for CANDU lattices,
- Identification and quantification of fuel string relocation reactivity effects,
- Improved understanding and consolidation of the available international database on oxide fuel behaviour under rapid energy deposition conditions,
- Experimental data on the interaction of molten fuel and moderator liquid following a fuel channel failure,
- Experimental data on failure of fuel channels at high pressure and temperature conditions.

This information is reviewed in this paper and the relevance of this information to key phases of the loss of shutdown scenario is established. It is demonstrated below that this information supports the failure criteria that were employed in the original analysis and that the failure event sequence and consequences remain essentially unchanged by new information.

Additionally, a number of significant conservatisms were applied to key phases in the failure sequence. They were intentionally imposed on the analysis to maximize the energy release to the containment atmosphere. These conservatisms are revisited and their physical reasonableness is re-assessed in order to establish their impact on the calculated consequential challenge to containment integrity. It is shown in this paper that the challenge to containment integrity was conservatively over-predicted in the original analysis – the implication of this finding is that there is greater margin to impairment of the containment envelope than originally predicted.

The reassessment is structured to focus upon the following key phases of the short-term accident progression:

- Neutron kinetics excursion
- Fuel Response during the Power Pulse
- Fuel channel Failure
- Moderator displacement, and
- Containment Response

Each of these areas is discussed below.

Neutron Kinetics Excursion

The variables and parameters governing the reactor kinetics excursion are the void reactivity, delayed neutron fraction and the prompt neutron generation time. Other influencing factors are the reactivity feedbacks, such as fuel temperature (Doppler), fuel string relocation and Xenon burnout.

Of these parameters, the void reactivity has been subjected to the most intense scrutiny over the past decade due to issues relating to the magnitude of the void reactivity uncertainty allowance. This has resulted in a more definitive statement of the void reactivity uncertainty allowance and, by application, in the void reactivity itself. In the 1987 analysis the assumed value of full-core void reactivity was assumed to be +13.4 mk. This corresponded to the upper bound of the uncertainty range at that time. With the revised void reactivity uncertainty allowance the best estimate of full-core void reactivity for a Pickering A equilibrium fuelled core is +15.6 – 2.0 mk = +13.6 mk, which is marginally different from the value used in the original analysis. The relevant coolant void reactivity during the early power excursion is the half core void reactivity associated with coolant voiding in the broken heat transport loop. The half-core void dynamic reactivity of 8.5 mk, which accounted for the effect of the side-to-side flux tilt induced by the coolant voiding, is not significantly different from the current best estimate. Finally, and most importantly, the transient void reactivity during the early part of the transient is governed by the voiding in the affected pass of the broken loop (i.e. quarter core voiding) since coolant void in the other pass develops much more slowly.

The total delayed neutron fraction assumed in the analysis was 0.005849 (β = 5.849 mk). This value is marginally higher than the current best estimate value of 0.00550 (β = 5.550 mk) which is 5% lower. The effect of variation in the value of β is addressed by the dynamic sensitivity analysis discussed below.

Fuel string relocation reactivity was not included in the original analysis. The assumed break was a guillotine rupture of a reactor inlet header and, therefore, a rapid insertion of approximately -0.7 mk from fuel string relation is assured. On account of this incremental and rapidly inserted negative reactivity, the original initial positive reactivity transient is over-predicted and the power excursion will develop over a slightly longer time period. The difference in the transient power excursion does alter the energy deposition transient and does alter the timing of the first fuel channel failures – failures delayed by approximately 0.3 seconds, as shown below.

The effects of variations in the different neutron kinetics parameters on the energy deposition in the fuel are quantified by evaluating the dynamic sensitivity variables as follows. The energy deposition sensitivity variables are defined with respect to the following variables and parameters:

- The neutron flux $\eta(t)$ in normalized full-power units,
- The normalized energy deposition $E_d(t)$ in units of full-power-seconds deposited from time $t = 0$ to time $t = \tau$.
- The net reactivity transient, $\rho(t)$
- The transient void reactivity $\rho_{void}(t)$ and
- The transient fuel string relocation reactivity, $\rho_{fuel\_string}(t)$
- The total delayed neutron fraction, β

$$SE_1(\tau) = \frac{\partial E_d(\tau)}{\partial \rho}$$

$$SE_2(\tau) = \frac{\partial E_d(\tau)}{\partial \rho_{void}} = SE_1(\tau)$$

$$SE_3(\tau) = \frac{\partial E_d(\tau)}{\partial \rho_{fuel\_string}} = SE_1(\tau)$$

$$SE_4(\tau) = \frac{\partial E_d(\tau)}{\partial \beta}$$

$$E_d(\tau) = \int_0^{\tau} \eta(t)dt$$

The change in energy deposition up to a time t = τ is evaluated using these dynamic sensitivity variables and the following perturbation equation:

$$\Delta E_d(\tau) = SE_2 \Delta\rho_{void} + SE_3 \Delta\rho_{fuel\_string} + SE_4 \Delta\beta$$

The sensitivity variables, SE_1 and SE_4 associated with the original analysis assumptions are shown in Figure 1 and the change in energy deposition relative to the original assumptions is shown in Figure 2 and Table 1. These results show that the original analysis remains conservative, primarily because of the effect of fuel string relocation negative reactivity that was not included in the original analysis.

It is important to note that the sensitivity of energy deposition does not change significantly when the reactor becomes prompt super-critical – in fact the sensitivity shows a marked increase only at times greater than 1 second after achieving prompt criticality. This behaviour is very different from reactors, such as PWR or fast breeders with much shorter prompt neutron generation times, for which there is a very rapid escalation of neutron flux immediately following prompt criticality.

TABLE 1
EFFECT OF CHANGES IN REACTOR KINETICS PARAMETERS ON TIMING OF FUEL MELTING AND FUEL CHANNEL FAILURE

	ORIGINAL ANALYSIS ASSUMPTIONS	REVISED ANALYSIS ASSUMPTIONS
Time of first fuel melt relocation [seconds]	3.3	3.6
Time of first fuel channel failure [seconds]	3.7	4.0

Fuel Behaviour

A large body of experimental data on uranium dioxide fuel behaviour under Reactivity Initiated Accident (RIA) conditions has been accumulated since 1987. This data encompasses many different fuel designs including PWR, VVER, BWR and prototypical CANDU fuel. The tests were performed predominately in test reactors with very short pulse widths (half-power pulse widths in the region of 4 ms), although one set of test data from the Russian IGR reactor involved much wider pulse widths similar to those in CANDU.

This experimental data base has been reviewed and assessed relative to CANDU fuel and power pulse conditions (References 5,6) and relative to LWR fuel (References 7,8). The results of this work reported in References 5 and 6 establish that this data is a) relevant and b) can be utilized, with appropriate analytical corrections, to assess CANDU fuel response. Specifically, with regard to fuel behaviour under loss of shutdown conditions, the experimental data supports the predicted fuel failure mechanisms and the impact on subsequent fuel channel failures. The dominant failure mode is associated with the formation and relocation of relatively limited and localized quantities of molten fuel sheath and UO_2 material.

Consider now the sensitivity of molten fuel relocation to energy deposition. Energy deposition is highest in the outer elements of the bundles experiencing the highest transient powers during the power excursion. This localizes the number and location of channels, as well as the bundle locations and element positions in bundles that will experience earliest fuel melt formation. None of these considerations are significantly changed by variations in the kinetics parameters controlling the power excursion that was discussed in the previous section. However, what does change somewhat is the time over which the melt formation and relocation occurs. As noted above, the effect of fuel string relocation reactivity is to delay the transient power escalation. As the local energy deposition increases, the central melt region in the fuel pellets in the outer fuel elements of the highest powered bundles will increase. Increasing the amount of fuel melting increases the internal driving force for melt relocation due to the volumetric expansion when solid UO_2 liquefies. This volumetric expansion will increase the interfacial contact pressure between the outer surface of the fuel pellet and sheath, thereby increasing the heat transfer rate between the two fuel element components. Additionally, the surface region of the fuel pellet receives a larger fraction of energy deposition than the central region because of the "self-shielding" flux depression within an element. Rapid escalation of fuel sheath temperatures in the affected pass of the broken loop will occur and relocated molten fuel cannot re-solidify at the outer edges of a pellet.

In the unbroken pass of the broken loop the coolant flow remains high and there is significantly less void than in the affected pass of the broken loop. Therefore, both the fuel elements and the pressure tubes in this pass will experience high convective cooling for a longer period of time. This will delay fuel channel failures in this pass relative to the affected pass – an effect which was not credited in the original analysis, where for conservatism it was assumed that channels in the two passes would behave identically based upon their initial element ratings and energy deposition in the fuel.

Rapid relocation of molten Zircaloy and UO_2 onto the pressure tube occurs from the outer elements at the bottom of the fuel channel. The molten UO_2 relocation is expected to be mildly forcible – in the nature of a "squirting" flow driven by the large increase in UO_2 linear expansion that occurs during the solidus-to-liquidus transition, as has been observed in separate effect rapid direct electrical heating (DEH) tests performed on fuel element segments at AECL Chalk River. However, should there be any appreciable delay in the localized relocation of molten material onto the pressure tube, then the superheat of the molten material at the pellet centre will increase while the molten fraction of the pellet will rapidly increase until most of the pellet is melted. This will occur for the outer elements of the highest powered bundles. Fuel melting in the other rings of elements in the bundle will not yet have started while this melt escalation occurs in the high powered outer elements. This dependency on energy deposition in a fuel pellet is shown in Figure 3. Additionally, expansion of fission gas in the melt will assist the process of forced ejection of molten material from the pellet onto the pressure tube.

The important point is that, irrespective of whether a lesser or greater forced ejection of molten fuel material occurs, the molten material ejection will be localized to outer elements of high powered bundles in a localized group of channels in the broken loop. Fuel channel failures are driven by this localized overheating and limited melt relocation and not by widespread gross melting of fuel in an entire channel – conditions that the fuel channel could not develop without prior rapid failure. This failure behaviour in a CANDU loss of shutdown accident should not be confused with the formation of large quantities of molten material in an LWR vessel and the subsequent discharge of this molten material into containment – the so-called high pressure melt ejection scenario in LWR severe accidents that can lead to direct containment heating phenomena.

Fuel Channel Failure

Failure of a fuel channel was predicted in the 1987 analysis to occur shortly following relocation of some molten fuel material. This conclusion remains valid and is supported by a series of separate effects test data from various experiments performed in the COG R&D program – albeit that the tests were not specifically designed to address loss of shutdown conditions. Nevertheless, these test results do demonstrate very clearly the fact that channel failures will occur very rapidly once heat loads and deformation conditions are pushed to extremes on pressure tubes and calandria tubes.

First, from the results of contact boiling tests it is observed that at high channel pressures, typically greater than 6.5 MPa, with relatively high heatup rates at power levels representative of decay heat, rapid localized deformation of pressure tubes occurs which invariably results in pressure tube failure due to localized thermal creep strain. In LOCA/loss of shutdown conditions the fuel channels experience high pressures in the range of 5 to >14 MPa when molten fuel relocation starts – depending upon the heat transport pass and loop. These pressures in the channels, together with the very high heat flux onto the surface of the pressure tube wall ensures both rapid thermal creep strain failure of the pressure tube and the calandria tube following melt relocation.

Second, experiments involving molten Zircaloy interaction with a ballooned pressure tube contacting a calandria tube, as well as separate effects experiments investigating fuel channel rupture under conditions simulating extreme flow blockage in a channel, have been performed subsequent to the 1987 analysis. The results from these tests indicate that channel failure occurs very shortly after melt relocation for pressures above 5 MPa and for relatively small amounts of relocated molten Zircaloy-4 (approximately 120g). As discussed above, the molten material undergoing relocation will include significant amounts of UO_2 in addition to Zircaloy-4 and will have mass of the order > 1kg. The UO_2 melt has higher thermal capacitance and higher temperature and will experience significant increase in volumetric fission heat generation rate once it relocates out of the element (due to the higher thermal neutron flux at the pressure tube relative to the outer elements of the fuel bundle). Additionally, the mass of locally relocated material from outer elements onto the bottom of the pressure tube will increase over a short period of time.

The experimental information regarding rapid channel failures is consistent with the predicted results of the analysis, which indicated channel failure occurring within about 0.22 to 0.5 s of molten material contacting the inner wall of the pressure tube, depending upon the local pressure and melt conditions. Increasing the local temperature, amount of melt or melt temperature reduced the time to channel failure. This is consistent with observations from the channel flow blockage experimental tests. This conclusion can be further demonstrated by the data plotted in Figure 4 which shows the relevant conditions of the experiments and the conditions associated with the loss of shutdown scenario. As is clearly apparent from this figure the loss of shutdown conditions are significantly more

severe than the experimental tests and projected failure times are consistent with those calculated in the 1987 analysis.
The consequences of channel failure on the moderator displacement response and calandria vessel failure remain valid. In particular, the interaction of fuel debris, including the limited amount of molten fuel will behave according to the forced interaction model used in the original analysis. This is supported by the results of recent CANDU Owners Group (COG) experiments on molten fuel-moderator interaction (MFMI) conducted at AECL's Chalk River Laboratories.

Containment Response

In the 1987 analysis a number of significant conservative assumptions were applied to the analysis of the containment response. Since containment integrity is a major governing factor for off-site releases it was considered prudent to apply these conservatisms in order to bound uncertainties. This was necessary given a) the very short time period over which the analysis was performed, and b) the very high profile of off-site releases from the Chernobyl Unit 4 accident. The two most important assumptions were:

- Although the pipe break was assumed to be a RIH guillotine rupture (in order to maximize the coolant voiding), the break discharge was assumed to occur directly in the reactor building (RB) to maximize pressure and hence maximize the loading on the containment envelope.

- The coolant discharge from the ruptured channels in the calandria was assumed to discharge directly into the calandria vault. This leads to the earliest and largest steam discharge from the calandria vault into the reactor building.

In reality, an inlet header break discharges into the fuelling machine vault and from there into the reactor building volume. Such a flow path will reduce the steam discharge into the RB and limit the rate of pressure rise in the RB. Secondly, when the fuel channels fail and initiate displacement of the moderator fluid, a significant amount of the initial steam discharge will be condensed in the cold moderator fluid during the displacement transient. Additionally, a significant fraction of the energy of the ejected fuel debris will be transferred to metal structures in the calandria which will result in delayed steam generation from this source. Removal of discharge energy by these two heat transfer mechanisms will reduce the rate and amount of steam discharged into the calandria vault, thereby reducing the steam discharge rate into the RB. In addition, discharge from failed channels is initially into the calandria vessel and the flow rate from the vessel is limited by the area openings – part of the discharge being directed into the fuelling machine vault via the calandria relief ducts and part into the calandria vault via the rupture area in the vessel.

The effect of reduced steam discharge rate and the short term integrated mass discharge into the RB was clearly apparent in the original sensitivity results performed for three different steam discharge scenarios – the base case, early termination and worst case scenarios.

If the following assumptions are made:

- heat transport system break occurs in the fuelling machine vault (consistent with the postulated break location)
- the steam discharge rate is limited by the opening areas on the calandria vessel with half the flow going into the FM vault and half into the calandria vault, and
- energy removal by direct contact condensation of steam discharged into the calandria vessel prior to and after vessel failure is taken into account,

then the peak reactor building pressure for the three different cases is estimated to be reduced as follows:

1. From 160 kPa(a) to approximately 140 kPa(a) for the base case (190 channels fail within 2.5s of calandria vessel failure)
2. From 157 kPa(a) to approximately 137 kPa(a) for the early termination case (90 channels fail within 2.5s of calandria vessel failure), and
3. From 180 kPa(a) to approximately 160 kPa(a) for the upper bound case (390 channels fail within 2.5s of calandria vessel failure)

Since the design pressure of the containment structure is 142 kPa(a), then only for the upper bound case does the reactor building pressure exceed design pressure. The most likely channel failure process will involve a limited number of channels (2 to 3) failing initially in the affected pass of the broken loop followed by a significantly more gradual failure of additional channels in the broken loop over the next 20 seconds - approximately 40 based upon the energy depositions experienced in the various fuel channels. However, retaining some conservatism and assuming that the total number of failed channels remain the same for the three cases, but occur over a period of 20 seconds, then the peak RB pressures for the three cases are further reduced such that in all cases the pressure is below the design value. Therefore, based upon these more realistic assumptions, containment integrity is assured with significantly greater margin than originally predicted in the 1987 analysis. Not even minor cracking of the dome concrete will occur.

Findings and Conclusions of the Reassessment

Reassessment of the 1987 analysis of a large break LOCA in a Pickering NGS A unit accompanied by a loss of shutdown has shown the following:

1. The impacts of findings regarding reactor kinetics parameters over the last 20 years have had a small net effect on the calculated power excursion in such an event. The effect is primarily to reduce the rate of power escalation because of the negative fuel string relocation reactivity effect.
2. Experimental data on fuel behaviour under power pulse conditions and rapid heating leading to melting do not contradict the governing phenomena, behaviour and key assumptions made in the 1987 analysis.

3. Experimental data regarding fuel channel failure under high pressure, temperature and heat flux conditions and with molten fuel element material (Zircaloy-4 sheath material) exhibit rapid failures that are consistent with those calculated in the 1987 analysis.
4. Reassessment of the effects of major conservatisms applied in calculating the pressurization of containment shows that the 1987 analysis significantly over-predicted the peak pressure in the reactor building. Therefore, it can be concluded that the margin for integrity of the containment envelope is significantly higher than originally stated in the 1987 analysis. Most probably not even minor cracking of the dome concrete will occur.

The important overall conclusions that can be drawn are as follows:

- The discharge of steam from a failed calandria vessel must consider the available physical heat transfer mechanisms and compartment volumes. This becomes the dominant discharge into containment volumes over and above the discharge from the initiating LOCA pipe rupture and determines the extent of over-pressurization of the containment envelope. Thus, containment integrity margins can be expected to be larger than in Pickering A for designs which have water filled reactor (calandria) vaults (Pickering B, CANDU-6) or shield tanks (Bruce A & B, Darlington) which will further condense steam discharged from a failed calandria vessel, or for plants which have large multi-unit shared containment volumes (Bruce A & B, Darlington). Since Pickering A has acceptable margin it may be inferred that the margins for other CANDU plant will also be acceptable.

- The original 1987 analysis was considered at the time by some, and to this date by others, to be speculative. This reassessment has demonstrated that the analysis was in fact robust and the conclusions remain significantly conservative and essentially unchanged by knowledge gained and discoveries made in the intervening years.

- CANDU plants are capable of withstanding extremely unlikely events causing early core disruption without significant risk to the public.

PART II: REACTIVITY INITIATED ACCIDENTS IN WATER REACTORS

Over the years a lot has been written and said about mitigating the effects of fast reactivity initiated accidents in water reactors. In the early days of nuclear power development there was a divergence in views between the light water reactor (LWR) and heavy water pressure tube reactor proponents. This was reflected in many instances by blanket statements regarding the acceptability of designs having positive reactivity coefficients, for example the position stated in the classic text by T. Thompson, page 622 of volume 1, [Reference 9] with a footnote regarding rebuttal comments by W.B. Lewis and D.G. Hurst. It is worth noting that these issues regarding positive reactivity feedback in water reactors did not deter the development of fast breeder technology in the same period – even though these reactors had large prompt positive reactivity feedback mechanisms.

The occurrence of the Chernobyl Unit 4 accident re-ignited this latent controversy and the same arguments centred about positive coolant void have reappeared, but now with an assumed moral imperative of preventing such an event from occurring again. The competitive nature of a reactor market that essentially disappeared in North America after TMI-2 and Chernobyl has given these arguments a sharper, and some might say nastier edge. They have also caused confusion in regulatory jurisdictions where conformance to international standards has become a necessity – albeit that the international standards may have in-built technology bias.

Given this background it is desirable that some of the inherent features of different designs be re-examined and put into perspective – otherwise invalid conclusions could be drawn by some regarding the relative safety of different reactor types. This will be addressed below in the context of rapidly developing reactivity initiated accidents.

Item 1: Positive reactivity holdup (defect)

Reactivity is like an accountant's double column ledger – there are positive and negative entries (credit and debit columns). For example, in CANDU there is positive coolant void reactivity which on a normalized-beta basis is approximately +$2.5 for full core voiding or +$1.5 for half-core voiding. PWR's have a positive moderator temperature reactivity feedback of the order of +$8 to +$12 for rapid under-cooling events (e.g. steam line break) and a very rapid positive reactivity of up to +$1.5 for a single rod ejection accident.

Item 2: Prompt neutron generation time

CANDU, because of it's distributed fuel lattice, has a prompt neutron generation time of approximately 0.89×10^{-3} s, whereas PWR, with their tight lattice cores have prompt neutron generation times in the order of 0.18×10^{-4} s.

Considering these two items the following observations can be made.

1. A +$1.5 increase in reactivity can be inserted over approximately 2.5 or more seconds as a coolant voiding ramp in a CANDU, whereas the same magnitude of reactivity increase can occur as an approximate step change associated with high pressure ejection of a control rod in a PWR. With this magnitude of reactivity increase taking both reactors super-prompt critical, the time taken for the reactor power level to increase by a factor of 5 is approximately 1800 ms in a CANDU and 6 ms in a PWR. Clearly, the rate of power rise in a PWR is beyond the physical capability of any shutdown system and necessitates that there be prompt large fuel temperature (Doppler) feedback to limit the magnitude of the power excursion. Given the significantly longer time taken for

the power level to escalate to the same level in a CANDU, there is ample time for fast-acting engineered shutdown systems to act.

2. The above observation indicates that positive coolant void can be safely accommodated by fast-acting shutdown systems in reactors with prompt neutron generation times of the order of 1 ms. It also suggests that it was the lack of a fast-acting shutdown system in Chernobyl Unit 4, and not necessarily the positive coolant void reactivity that was the root cause for the extremely large and damaging power excursion. The Chernobyl shutdown system was woefully slow requiring 12 seconds to drive the rods from fully out to fully in the core. A fast acting shutdown system such as either of the two independent systems in CANDU would have terminated the power excursion without any extreme consequences – probably only limited fuel sheath failures at most.
3. The larger the amount of reactivity holdup in a reactor (the reactivity defect) the larger is the required reactivity depth of engineered shutdown systems. Hence, for steam line breaks in a PWR there is an issue of re-criticality which requires operation of two shutdown systems – rods plus boron addition.

The above observations indicate that one should not claim that one well engineered reactor type is "inherently safer" than another well engineered reactor type. Conversely, a poorly engineered reactor system when coupled with poor safety culture can be more susceptible to damaging events, irrespective of the reactor type. This is a lesson we should not forget and most definitely it is a lesson that should not be misused in order to gain a perceived competitive advantage.

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FIGURE 1
DYNAMIC SENSITIVITY OF ENERGY DEPOSITION TO REACTOR KINETICS PARAMETERS

SE_1 = Sensitivity to reactivity [FPS/mk]
SE_4 = Sensitivity to delayed neutron fraction [FPS/mk]

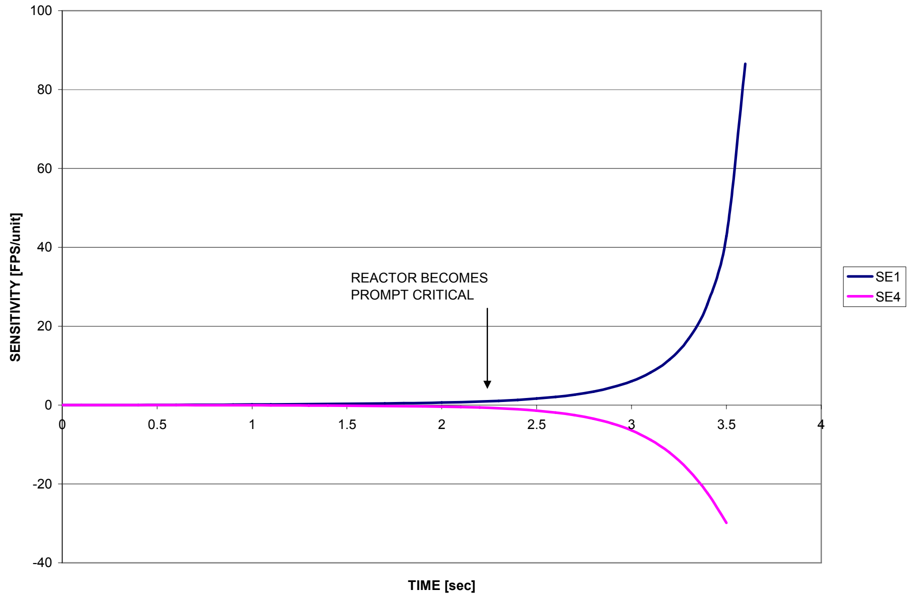

FIGURE 2
CHANGE IN ENERGY DEPOSITION DUE TO REACTOR KINETICS PARAMETERS

Reactivity = Fuel string relocation = -0.7mk
Beta = 5% reduction in total delayed neutron fraction

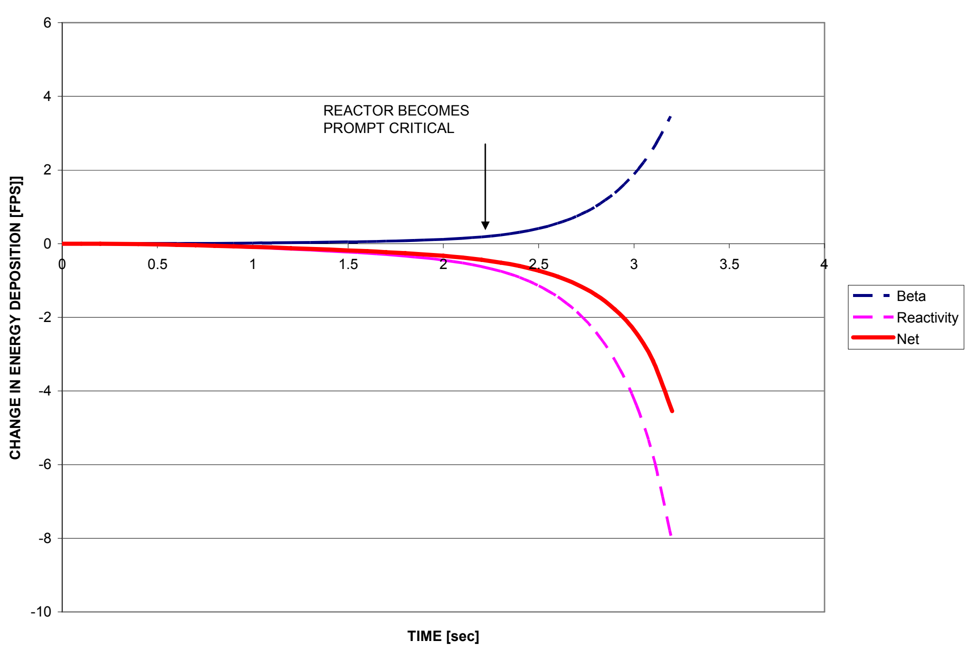

FIGURE 3
ENERGY DEPOSITION IN FULL-POWER-SECONDS REQUIRED FOR ONSET OF FUEL MELTING AS A FUNCTION OF INITIAL ELEMENT LINEAR POWER RATING

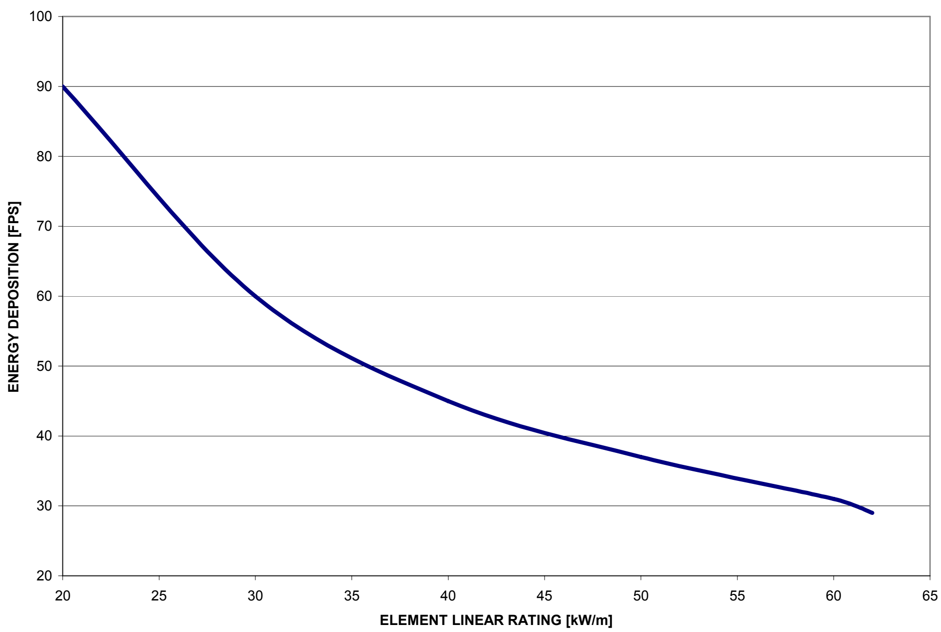

FIGURE 4
TIME TO FUEL CHANNEL FAILURE AFTER MELT RELOCATION AS A FUNCTION OF MELT MASS RELOCATED