For presentation to the May 1988 ANS Topical Meeting
on Safety of Next Generation Power Reactors in Seattle, WA

CONF-880506--3

PROBABILITIES OF INHERENT SHUTDOWN OF UNPROTECTED EVENTS IN INNOVATIVE LIQUID METAL REACTORS

C. J. Mueller & D. C. Wade

CONF-880506--3

DE88 009946

Argonne National Laboratory
9700 South Cass Avenue
Argonne, IL 60439
(312) 972-8164

*Work performed under the auspices of the U.S. Department of Energy.

Work supported by the U.S. Department of Energy "Technology Support Program" under Contract W-31-109-Eng-38.

PROBABILITIES OF INHERENT SHUTDOWN OF UNPROTECTED EVENTS IN INNOVATIVE LIQUID METAL REACTORS

by

C. J. Mueller & D. C. Wade

Abstract

The uncertainty in predicting the effectiveness of inherent shutdown in innovative liquid metal cooled reactors with metallic fuel results from three broad contributing areas of uncertainty: (1) the inability to exactly predict the frequency of ATWS events with potential to challenge the safety systems and require inherent shutdown; (2) the approximation of representing all such events by a selected set of "generic scenarios"; and (3) the inability to exactly calculate the core response to the selected generic scenarios. This paper discusses the work being done to address each of these contributing areas, identifies the design and research approaches being used at Argonne National Laboratory to reducing the key contributions to uncertainties in inherent shutdown, and presents results. The conditional probabilities (given ATWS initiation) of achieving temperatures capable of defeating inherent shutdown are shown to range from ~0.1% to negligible for current designs.

Introduction

The uncertainty in predicting the effectiveness of inherent shutdown in preventing severe core damage in innovative reactors designed for passive shutdown results from three broad contributing areas of uncertainty: (1) the inability to exactly predict the frequency of initiating events with potential to challenge the safety systems and require inherent shutdown; (2) the approximation of representing all such events by a selected set of "generic scenarios"; and (3) the inability to exactly calculate the core response to the selected generic scenarios.

This paper outlines the approach and methods used to address these contributing areas for the loss of flow (LOF) and transient overpower (TOP) scenarios in support of risk assessments for the innovative design projects featuring metal fuel cores in pool configurations. In particular, this paper focuses on uncertainties in core temperatures due to the inability to precisely characterize the core response to LOF and TOP initiators. From sensitivity studies and estimates of the underlying uncertainties, probability distributions for safety margins can be propagated, and the risk can be calculated. The report also discusses supporting work, cites preliminary results, and identifies initiatives being taken to reduce the key contributions to uncertainties in the effectiveness of inherent shutdown, given that an unprotected scenario has occurred.

Uncertainties in the Frequency of Unprotected Events

The frequency of events requiring inherent shutdown is a function of

(1) the frequencies of those transients or other abnormal events (e.g., earthquake, fire) signalling control rod scram; and

(2) the concomitant probabilities of unsuccessfully terminating fission power production by inserting control rods, or if applicable, by activating alternate neutron poison features.

The data base for estimating frequencies of the challenging transients or abnormal events includes published LWR transient data (e.g. ATWS (anticipated transient without scram) events, which in turn factors in licensee event reports (LERs) and abnormal occurrence reports (AORs) of events requiring scram) and the fast reactor experience of EBR-II and FFTF. This data base will be expanded as data from the French LMRs become available. Work to date[1,2] suggests that transient frequencies for the innovative designs are likely to be significantly less than for LWRs because the forgiving nature of the liquid metal pool systems allows both fewer and less stringent trip settings. Current estimates for pump coastdown and loss-of-heat-sink transients are about ~0.2/reactor-year. The estimate for a single-rod-withdrawal, the reference scenario for TOP evaluation, is ~0.01/reactor-year. Because of the conceptual stage of the innovative designs, uncertainties in the frequency of initiators requiring scram cannot be accurately assessed.

Concomitant probabilities of failing to terminate fission power by the engineered (i.e. noninherent) features of the reactor protection system are based on reliability analyses of the protection system. If the protection system is limited to traditional control rod scram action, the scram failure probability should be comparable to that for LWR scram systems. A failure probability of 3 x 10^{-5}/demand was used by NRC as a guideline in resolution of the ATWS issue. Current evaluations yield about 1 x 10^{-6}/demand for the fa/demand for the failure probability of a single system for the innovative designs. Redundant and diverse neutron poisoning systems are calculated to further reduce the failure probability of automatic shutdown by up to several orders of magnitude, depending on initiator-dependent common-cause failure effects.

In summary, current evaluations indicate that the protection systems for the innovative designs will be more reliable than those for existing LWR systems. Forthcoming LMR operational data as well as design-specific assessments will serve to improve the estimates of transient frequencies as described above. However, no information is expected to change significantly the relative uncertainties in estimates of frequency of challenge to inherent shutdown over the design and testing phases of the innovative designs. That is not to say that design choices cannot be made to reduce the best estimates of these frequencies. For example, one class of initiators for the LOF involve signals sent to the protection system that fail to initiate scram but successfully trip the pumps. Clearly this class of initiator can be effectively eliminated by requiring confirmation of successful scram before the signal is issued to trip the pumps.

Uncertainties in Representing all Unprotected Events by Selected Generic Scenarios

Nominal responses of the innovative metal core reactors to unprotected events are characterized by transient temperatures with large safety margins to short term boiling during coastdown (LOF) or reactivity insertion (TOP); long term temperatures with large margins to fuel temperature regimes capable of leading to fuel-clad eutectic penetration and/or creep rupture (LOF, TOP); and long term temperatures with large margins to creep of the reactor vessel or its internals (all events including the LOHS). The risk of irreversible core disruption or severe structural failures from these events comes about as a result of anomalous behavior or unexpected design deviations such as

-- accelerated flow coastdowns due to multiple pump lockup or, depending on design, flow extension failure (LOF)

-- multiple rod withdrawals that through some combination of structural, electrical, or design failures proceed at speeds and to levels beyond those which are nominally limited by design (TOP)

-- hypothetical structural failure or core distortion events that place the core in a more reactive configuration or retard radial core or axial control rod driveline expansion feedback (catastropic events)

-- the composite of core response effects, of which reactivity feedback is dominant, failing to perform as expected and predicted (all events)

For all but extremely improbable, massive seismic or similarly catastrophic initiators, the top three items can be addressed by careful design and should have slight importance to risk relative to the fourth. Accordingly, the three generic unprotected scenarios, the LOF, TOP, and LOHS events, have traditionally been analyzed, both individually and in combination, to provide an envelope for the consequences of all unprotected sequences. Key parameters are then varied to explore the sensitivities of core temperature fields and concomitant safety margins for these reference cases. Combination scenarios, e.g. a TOP that trips the main coolant pumps resulting in a TOP-driven-LOF, have been calculated to gain insight into core response. However, it appears that the likelihoods of these combination scenarios can be made so remote that their contribution to risk is negligible. Accordingly, these combination scenarios are not treated in this report. What results then is the dominance of risk due to the core failing to inherently shut down as predicted.

Sources of Uncertainty in Predicting Core Response to Generic Scenarios

The uncertainty in core response predictions comes about as a result of (1) uncertainties and limitations in transient modeling; (2) uncertainties in steady-state characterizations of neutronic (e.g. rod worths and positions, nominal power ratings) and thermal-hydraulic (e.g. inlet and outlet temperatures, flow distributions) parameters; (3) variations from the reference reactor due to manufacturing, installation, and operating tolerances; and (4) uncertainties in the conditions needed to result in core disruption or vessel failure, i.e. the "criteria" to signal the defeat of inherent shutdown. Sources of uncertainty (2) and (3) are often lumped together and referred to as "hot channel effects".[3,4] The most dominant uncertainties with respect to inherent shutdown prediction appears to come from calculating the various feedback effects. Table 1 presents an update of previously published[5] estimates of feedback uncertainties used in ANL risk assessments.[6]

Many of the modeling uncertainties associated with both steady state characterization and transient response will be reduced as the designs evolve and R&D efforts are completed. These uncertainties will be further reduced and some sources effectively removed when testing of the first demonstration plant is completed. Thus, as the design and testing progress, the means (or best estimates) of the probability distributions of core temperature responses to unprotected events will probably be held relatively constant by choice of design parameters, and the spreads of these distributions will be reduced.

The main issue in addressing the probability of inherent shutdown of unprotected events is how the uncertainties in individual feedback components

Table I.

Uncertainty[a] Assignments in Reactivity Coefficients
Used in ANL Risk Assessments of Advanced LMR Concepts

Reactivity Feedback Mechanism	Metal	Oxide
Doppler	20%	15%
Na Density	20	20
Fuel Axial Expansion/Contraction	30	25
-- neutronic	20	15
-- thermo-mechanical	20	20
Net radial expansion (P/F >0.8) (including bowing)	20	20
Neutronic	15	15
Thermal hydraulic	10	10
Structural	10	10
(P/F <0.8)	50	50
Neutronic	15	15
Thermal hydraulic	15	15
Structural	50	50
Control Rod Expansion	20	20
-- neutronic	10	10
-- thermal-hydraulic	<20	<20
Pre-clad failure, in-pin, molten fuel relocation	Not evaluated	Not evaluated
Vessel Axial Expansion	Not evaluated	Not evaluated
Core Support Structure Expansion	Not evaluated	Not evaluated

a Values shown represent 1σ deviations from the mean of a normal distribution expressed as percentages of the best estimate reactivity coefficient. "Sub-effect" contributions are statistically combined to develop the five major short term reactivity feedback uncertainties, which are rounded to the nearest 5%.

combine to yield uncertainties in the global power, power-to-flow, and inlet temperature reactivity coefficients. Direct testing of these global coefficients will, of course, reduce their uncertainties to levels below that obtained through statistical combination of the individual feedback component uncertainties as shown here.

There are several main areas of physics R&D that will contribute to reducing the uncertainties in inherent neutronic feedback:

(1) upgrading the physics cross-section data base to factor in the results of criticals testing thereby improving the calculational tools used to estimate individual feedback coefficients;

(2) actual ZPPR criticals testing of metal fuel assemblies to provide a stronger experimental data base for individual neutronics feedback components as well as a calibration for the physics codes;

(3) EBR-II shutdown heat removal tests (SHRT) to validate transient modeling codes; and

(4) validating the codes predicting radial expansion effects as described previously.

All uncertainties, as well as stochastic variations in actual operations, can be propagated through accident analysis codes or models to produce a probability distribution for maximum temperatures capable of being reached in a specific reactor accident by fuel pins, flow channels, or key structural parts. The probability of failure of inherent shutdown is the fraction of this distribution of maximum temperatures that exceed safety limits keyed to the onset of core disruption or severe irreversible damage to the vessel or its internals. The uncertainties in the safety limits per se (uncertainty source (4) listed above) have not been treated explicitly - i.e. probabilistically-in the studies reported herein because of the lack of relevant failure data. Rather, single-value definition of these limits has been used to implicitly but conservatively include the uncertainties in failure criteria.

Conditional Probability of Inherent Shutdown

The true conditional probability of inherent shutdown failure (CPF) for an initiator class j (e.g. the TOP) is given by

$$CPF_j = \sum_i \text{Prob}(F/UI)_{ij} \text{ Freq } (UI)_{ij} / \sum_i \text{ Freq } (UI)_{ij}$$

where the summations are over all unprotected initiators (UI) in that class. The methodology used here assumes that this conditional probability can be approximated by that area of the uncertainty distribution for selected core response temperatures in reference accident scenarios lying above preestablished core disruption indicator temperatures. Thus, a key assumption of the modeling approach here is that the frequencies of nonmechanistic accidents (first three categories delineated in Section 3) can be made so low relative to those enveloped by the reference scenarios that the CPF can be approximated or bounded by evaluating only uncertainties in response to the reference scenario, that is

$$CPF_j \cong \text{Prob } (F/UI)_{j\text{-reference}}.$$

In other words, design assurances can be made sufficiently strong so that frequencies of initiators much more severe than the reference scenarios can be rendered negligible.

Sample Results

Figure 1 illustrates the results of an LOF analysis[7] using the SASSYS code[8] which demonstrate several aspects of LMR innovative designs pertinent to the themes discussed herein: (1) The margin to boiling, one measure of the failure of inherent shutdown for an LOF, is several hundred degrees K; (2) the dominant contributor to inherent shutdown is negative reactivity feedback from radial expansion of the core - thus uncertainty in this feedback mechanism is particularly important to core response; (3) the other feedback mechanisms taken individually are relatively unimportant.

Using SAS as the reference code, analyses[9] of various designs have provided one sigma (1σ) uncertainties of 25-45K in overall safety margin, depending on particular design, type of transient, and choice of transient

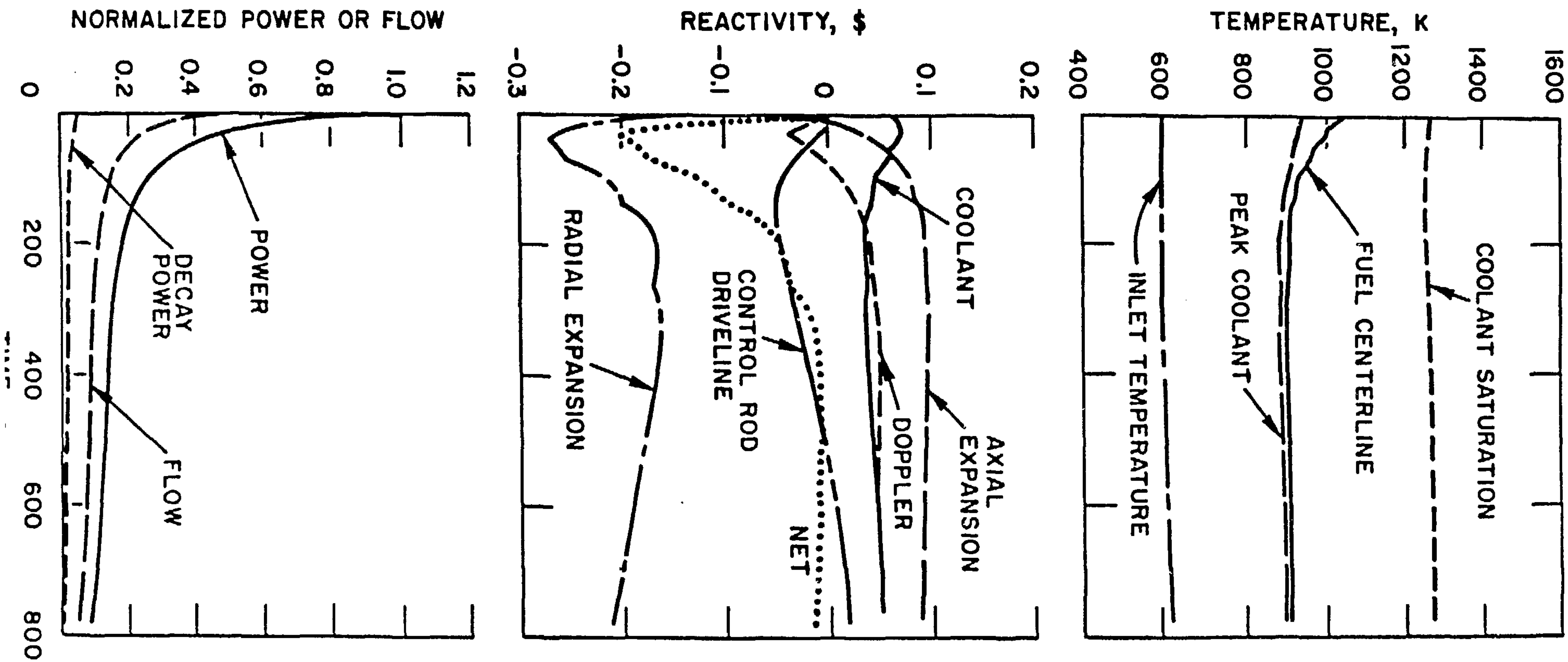

Figure 1. Early Innovative Design Example of Metallic Core Response of LOF.

parameters (e.g. flow coastdown characteristic in an LOF). As a result, assuming normal distributions for those core temperatures used to signal failure of inherent shutdown, the probabilities of achieving temperatures capable of defeating inherent shutdown are generally predicted to range from ~0.1% to negligible for current designs.

Summary and Conclusions

The work reported here suggests that the risk of current innovative designs suffering severe core damage as a result of unprotected whole core undercooling or overpower accidents is much lower than that of existing commercial power reactors. For example, NUREG-1050[10] reports the frequency range of potential core melt accident sequences initiated by transients with subsequent failure to scram and loss of reactor subcriticality to range from 1-60 x 10^{-6}/reactor-year for PWRs and 0.1-50 x 10^{-6}/reactor-year for BWRs. NUREG-1150[11] estimates of ATWS-induced core melt frequencies for several specific reactors generally are consistent with this range but do show notable exceptions, e.g. Zion is $<10^{-8}$. Nevertheless with comparable assumptions for scram system reliability, the inherent power reduction and large heat sink characteristics of the innovative designs render analogous values for the advanced LMRs,[1,2] several orders of magnitude lower than the ranges reported by NUREG-1050.

Moreover, design options have been identified which can keep risk almost arbitrarily low by design choice. For example, in LOF events the best estimate margin to boiling can be increased by designing the pumps so that the coastdown is extended. The severity of control rod withdrawals can be delimited by core designs that minimize burnup control swing, by control rod schemes that minimize the distance the rods can be withdrawn, or simply by increasing the number of rods. Such design choices may, of course, adversely affect other aspects of performance, and tradeoff studies must be performed. Nevertheless, the point is that best estimate or nominal predictions of safety margins and associated risk due to LOF or TOP sequences can be controlled by design choices. If Tech Spec limits are set on the key accident parameters such as pump coastdown characteristics and control rod withdrawal worths, and also on the global reactivity feedback parameters, public health and safety will be assured - if the limits are not met, the plant will be derated.

Accordingly, the real risk associated with design phase uncertainties in inherent shutdown is financial.

Finally, the work reported herein suggests that the probability of core disruption in the innovative designs from the traditional unprotected LOF and TOP may be so low that other reactor scenarios now become more risk-significant. These include local faults, i.e. local failures or blockages, being propagated into a core disruption accident; and sodium fires causing sufficient loss of the heat removal and/or control system capability to initiate an unprotected (or protected) event that eventually leads to core disruption. Evaluations of these scenarios rendered them risk-insignificant for CRBR and subsequent oxide designs and fuel and plant characteristics would appear to make them even less of a problem for the innovative designs - e.g. metal fuel-sodium compatibility should reduce the risk contribution of local faults relative to that in the oxide case. Nevertheless, these scenarios must be more carefully studied to determine their potential initiation frequencies and consequences. To date no serious effort has been expended along these lines.

References

1. Rockwell International, Private Communication, 1986.

2. General Electric, Private Communication, 1986.

3. Y. S. Tand, R. D. Coffield, Jr., R. A. Markley, "Thermal Analysis of Liquid Metal Fast Breeder Reactors", American Nuclear Society Monograph, 1978.

4. M. C. Carelli and A. J. Friedland, "Hot Channel Factor for Rod Temperature", Nuclear Engineering and Design, 62, 1980.

5. C. J. Mueller, "Impacts of Reactivity Feedback Uncertainties on Inherent Shutdown in Innovative Designs", Trans. Am. Nucl. Soc., 51, 1986.

6. C. J. Mueller et al, Unpublished Information, 1986.

7. D. P. Weber et al, Argonne National Laboratory, personal communication, 1986.

8. L. W. Deitrich and D. P. Weber, "A Review of U.S. Models and Codes for Analysis of Whole-core Transients and Accidents in Fast Reactors", International Conference on the Science and Technology of Fast Reactor Safety, May 12-16, 1986.

9. J. E. Cahalan, Argonne National Laboratory, personal communication, 1986.

10. U.S. Nuclear Regulatory Commission, Probabilistic Risk Assessment (PRA) Reference Document, NUREG-1050, September 1984.

11. U. S. Nuclear Regulatory Commission, Reactor Risk Reference Document [DRAFT for comment], NUREG-1150, February 1987.