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UNCERTAINTY REDUCTION REQUIREMENTS IN CORES DESIGNED FOR PASSIVE REACTIVITY SHUTDOWN*

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ABSTRACT

The first purpose of this paper is to describe the changed focus of neutronics accuracy requirements existing in the current US advanced LMR development program where passive shutdown is a major design goal. The second purpose is to provide the background and rationale which supports the selection of a formal data fitting methodology as the means for the application of critical experiment measurements to meet these accuracy needs.

I. US ADVANCED LMR PROGRAM NEEDS FOR CRITICAL EXPERIMENTS

A) Passive Reactivity Shutdown Core Design Goal

A recent focus for advanced LMR design in the US has been on achieving passive reactivity shutdown in response to unprotected whole core accident initiators such as unprotected loss of flow (LOF), loss of heat sink (LOHS), rod runout transient overpower (TOP), and chilled inlet coolant. The coolant mixed mean outlet temperature reached asymptotically upon passive shutdown in each of these unprotected events is a useful figure of merit for assessing passive reactivity shutdown effectiveness. These asymptotic temperatures are found to depend on ratios of reactivity feedbacks and (for the TOP) on a ratio of burnup control swing to reactivity feedbacks.[1] The relevant reactivity feedbacks are identified in Table I. where also are shown the typical sizes of the reactivity coefficients in ¢/°C.

One is struck by the small sizes of the numbers in Table I. In contrast to the multiple tens of dollars of shutdown reactivity vested in control rod scram, the passive shutdowns bring the core to zero power by balancing off reactivities in the range of cents or several tens of cents. As an example, Figure 1 shows the calculated results for passive shutdown of an unprotected LOHS accident in a 900 MW_{th} metal-fueled modular LMR. As the core inlet temperature rises in response to the loss of heat sink, radial core expansion introduces a negative reactivity of several tens of cents, causing the power level to be reduced to near zero. The coolant temperature rise, ΔT_c collapses to a small value, and the final asymptotic state is achieved when the positive reactivity introduced by bringing power to zero, (A+B), is balanced by the negative reactivity introduced by raising the core average (nearly isothermal) temperature, δT_{in} C:

$$\delta T_{in} C = (A+B) \tag{1}$$

Here C is the inlet temperature coefficient of reactivity (¢/°C) and (A+B) is the decrement in reactivity, (¢) which occurs upon taking the core to full power and flow from isothermal at the normal coolant inlet temperature. The asymptotic core outlet temperature is equal, in the LOHS, to the asymptotic core inlet temperature, and its change relative to its normal full power value (of $T_{in} + \Delta T_c$) is given by:

$$\delta T_{out}\ (\text{LOHS}) = - \left(1 - \frac{A+B}{C\Delta T_c}\right)\Delta T_c \qquad (2)$$

Table II summarizes the corresponding results -- in terms of ratios of reactivity parameters -- for the asymptotic core outlet temperature change resulting in each of the passively-shutdown ATWS events.

B) Impact of Neutronics Uncertainties on Passive Shutdown Performance

The reactivities involved in passive shutdown are numerically extremely small -- i.e. several cents -- and moreover they derive not only from the traditionally-considered temperature dependencies of densities and of Doppler broadening but also from very subtle geometrical displacements. For example a 10°C temperature rise at the grid plate dilutes a 2 meter diameter core by only 4 millimeters and yet it comprises one of the important reactivity feedbacks for passive shutdown. In a generic way:

$$\Delta\rho = \Delta T^{*} \left(\frac{\partial\rho}{\partial T}\right) \qquad \text{Na density \& Doppler}$$

$$+ \Delta T^{*} \left(\frac{\Delta\ \text{Position}}{\Delta\ \text{Temperature}}\right) * \left(\frac{\partial\rho}{\partial\ \text{Position}}\right) \qquad \text{radial and axial expansion \& bowing} \qquad (3)$$

In view of the small sizes of the reactivities involved and of the subtleness of the thermo/structural processes on which major components of the reactivity feedbacks depend, one might anticipate that unavoidable uncertainties in the values of nuclear and thermo/structural properties present a hopeless situation as regards reliability of passive shutdown. That this is not so is one of the amazing aspects of the effort to design for passive shutdown. There are two principal features which mitigate the impact that uncertainties and variability of key neutronics, thermohydraulic, and structural properties impute to passive shutdown performance.

First, as summarized in Table II, the passive shutdown performance -- as characterized by asymptotic change in core outlet temperature -- depends not on individual reactivity coefficients but rather on groupings of feedbacks, A, B, C, which are measurable on the operating reactor. Thus, irrespective of the current level of uncertainty in individual reactivity coefficients, and irrespective of the core-to-core variability of manufactured equipment and of aging effects which change incore equipment, it will always be possible to monitor the actual values of the inlet temperature coefficient, C, the power reactivity decrement, (A+B), and the flow coefficient of reactivity, B, on any operating power reactor. Given the measured values of A, B, and C (and the measurement precision), one can ascertain from the formulas in Table II, whether or not the passive shutdown performance will be capable of maintaining the core in a safe condition. A "Tech Spec" requirement on the frequency of measuring the integral parameters and on the allowed bands into which their values must fall will provide a means to assure that either the feedbacks do, in fact, provide for safety, or that the reactor must be shut down or derated until safety can be assured.

Second, the asymptotic core outlet temperature changes required to passively shut down ATWS events are insensitive to variations in the values of individual reactivity coefficients which comprise the overall integral parameters, A, B, and C defined in Table I. This gratifying result comes about because the same reactivity effects contribute both to the reactivity addition which accompanies power reduction and to the reactivity subtraction which accompanies core isothermal temperature rise; and these reactivities cancel by definition in passive shutdown. Consider, for example, core radial expansion coefficient in the LOHS accident. As the inlet temperature goes up, the grid plate dilates, causing a negative reactivity insertion from the radial expansion coefficient of reactivity. On the other hand, as the power is reduced, the coolant ΔT rise across the core is reduced, the above core load pads cool causing the top of the core to contract relative to the bottom of the core, and this leads to a positive reactivity input from the radial expansion coefficient. Since the radial expansion coefficient of reactivity contributes to both the reactivity addition and the reactivity subtraction processes -- which asymptotically cancel -- uncertainties or variations in this reactivity coefficient tend to self cancel also. In a mathematical sense, the uncertainties in the components of the numerator and denominator of the formulas for δT_{out} in Table II are positively correlated, tending to reduce their impact on the variance of δT_{out}. This will happen not only for the neutronics reactivity coefficients, but also for the thermo/structural components (see Eq. 3) of the reactivity feedbacks comprising the global reactivity parameters A, B, and C.

This serendipitous partial self cancellation of individual reactivity coefficient uncertainties has two payoffs. First, the actual power reactor will experience less variation in its passive shutdown performance in response to unavoidable variations in composition and geometry which derive from manufacturing tolerances and aging effects. Thus, the Tech Spec monitoring will assure safety and the self cancellation property will enhance plant availability in the face of the Tech Spec, when the actual power reactor is in place. Second, during the design and licensing phase, when A, B, and C cannot be measured on the operating plant but must be calculated based on computed values of individual components and when design and licensing activities must rely on calculational prediction, then this insensitivity of passive shutdown performance to uncertainties in individual reactivity coefficients raises the confidence level ascribed to the calculations by designers and licensers alike.

C) Specific Focus For Uncertainty Reduction Via Critical Experiments

There are two key places where the partial self cancellation of uncertainties as they affect passive shutdown consequences fail to take place. The first is for the TOP ATWS event where the uncertainties of the components comprising the BOEC hot, all-rods-out, reactivity excess are poorly correlated with those of the reactivity feedback coefficients. Here, the initial* core outlet temperature increase in response to an unprotected single rod runout is given by:

$$\delta T_{out} = \Delta T_c \left(\frac{\text{BOEC Excess}}{\text{A+B}}\right) \left(\frac{\text{First Rod Out Interaction Factor}}{\text{No of primary rods}}\right) \quad (4)$$

with

$$\begin{aligned}(\text{BOEC Excess}) = {} & (\text{Burnup Control Swing}) \\ & + (\text{Excess to Cover Uncertainties}) \end{aligned} \quad (5)$$

Here the (Excess to Cover Uncertainties) is provided for by over-enrichment of the manufactured assemblies to cover contingencies such as:

$$\begin{pmatrix}\text{Excess to Cover} \\ \text{Uncertainties}\end{pmatrix} = \left\{ \begin{array}{l} (\text{uncertainty in burnup control swing})^2 \\ + (\text{uncertainty in cold to hot reactivity defect})^2 \\ + (\text{uncertainty in cold critical mass})^2 \\ + (\text{uncertainty due to fuel manufacturing tolerance})^2 \end{array} \right\}^{1/2}$$

It is evident that the uncertainties in Doppler, sodium density, and radial and axial expansion temperature coefficients of reactivity which comprise the (A+B) factor in the denominator of Eq. 4 are but loosely correlated with the uncertainties in the factors determining the burnup control swing in the numerator of Eq. 4. Thus, one does not expect a partial uncertainty self cancellation as was enjoyed for the other ATWS events. It must be noted that, in fact, the advanced LMR cores are designed to achieve a nominally zero burnup control swing so that -- to the degree that the design goal is achieved -- it is the (Excess to Cover Uncertainties) which controls the size of the (BOEC Excess) and of the TOP initiator. Nonetheless, the loose correlation between the contributors to uncertainty in the denominator and numerator of Eq. 4 persists with the exception of shared components in the cold to hot reactivity defect.

The second instance where uncertainties relevant to passive shutdown performance are poorly correlated with those of the reactivity feedback coefficients is for the local power peaking factor which is a necessary factor for converting the global core mixed mean coolant outlet temperature rise to a local, hot channel, value required in an actual assessment of margin to core damage in response to the passive shutdown of ATWS events.

$$\begin{aligned}\delta T_{out}\,(\text{local}) = {} & \delta T_{out}\,(\text{core mixed mean}) * \\ & * [\text{Local Peaking Factor (Burnup State, Rod Position)}] \\ & * [\text{Local/Ave Flow Redistribution}] \end{aligned} \quad (7)$$

From the results of the above discussions it is seen that at our current state of knowledge prior to power reactor construction, in order to reduce the neutronics uncertainties which importantly impact calculational predictions of

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* If the initial power rise corresponding to Eq. 4 is large enough to boil dry the steam generator, then a loss of heat sink on top of the TOP will determine the asymptotic state.

passive shutdown performance we must address the criticals measurements to:

- the reactivity coefficient components of A, B, and C. i.e.
  Doppler
  Na density
  Axial and radial expansion
  Control rod differential worth

- the burnup control swing

- The components of the BOEC (Excess to cover Uncertainties) i.e.
  - cold critical mass
  - cold to hot reactivity defect
  - fuel worth

and

- local power peaking factor
  - vs rod position and burnup state

This list is seen to encompass not only the traditional focus of previous criticals measurements programs, but additional ones which are not ameanable to direct measurement on a critical facility such as burnup control swing and cold to hot defect.

D) Institutional Environment

To summarize the discussions of the previous section, we find that in a regime of cores designed for passive shutdown, the need for criticals experiments to both correct calculational predictions of reactor quantities and to reduce their uncertainties encompasses all of the neutronics quantities stressed in prior programs and more as well. Particular stress in the current US program must be put on reduced uncertainty in burnup control swing because the rod runout TOP is unique among the ATWS events in that the uncertainty of the outcome of the event does not benefit from a partial self cancellation of the uncertainties in the underlying parameters which control the outcome.

Not only have the design goals shifted so as to modify the focus of the ZPPR criticals program, but the current US institutional environment imposes additional boundary conditions on how it is to be conducted. First, while the NRC staff has informally indicated a willingness to "give credit" for passive shutdown of Beyond Design Basis Events in the licensing of advanced LMR's, "receiving credit" will require the establishment with the licensing bodies of a high degree of credibility for the calculational predictions of passive shutdown effectiveness and for the provision of margins which will comfortably accommodate the current level of uncertainty. One might protest that the ATWS events which depend on passive shutdown are Beyond the Design Base and therefore their consequences are to be computed based on best-estimate values and that uncertainties are irrelevant. But such an objection ignores the reality of how human judgments concerning acceptable protection from risk are made in the face of uncertainty. Beyond Design Base or not, sensitivity studies of passive shutdown scenarios to quantify the dependence of consequences on input variations and the establishment of large margins between nominial consequence and initiation of massive core disruption are a prerequisite to the use of passive shutdown features in licensing. Since the licensing interactions are based in part on calculated performance, this implies the needs to:

a)- validate the calculational predictions of core response to ATWS events

b)- place realistic and defensible bounds on the ATWS event consequences when uncertainties in the underlying parameters are propagated

and beyond that, to

c)- provide substantial additional margins to cover the undefined phenomena and/or scenarios unaccounted for in the calculations.

Recent sessions of the U.S. Congress have reflected both the general public's disenchantment with nuclear power and it's concern over the federal deficit by allocating funding for but a small and at best non-expanding advanced reactor program. Since public perceptions comprise a significant factor in influencing public policy and public spending, notwithstanding the NRC staff's encouragement regarding acceptability of passive shutdown as a component of licensing, it is essential to establish a widespread perception that passive shutdown has technical credibility and that, as a result, an R&D program to pursue its potential is in the public interest. This institutional need imposes both a timeliness and a low-cost boundary condition on the measurements program to reduce uncertainties. Results are needed early to favorably influence funding of a continuing R&D program while at the same time these measurements must be conducted under the existing level of funding.

Thus, in view of the institutional boundary conditions:

d)- the establishment of a widespread perception of credibility for passive shutdown must occur early in the program in order to favorably influence the continuing flow of development funds,

e)- this requires not only demonstrating acceptable performance on a best estimate basis but also requires that a defensible quantification of the uncertainty levels be provided

f)- and that the entire process be "explainable" to a general audience who are not technical specialists in critical experiments or in uncertainty propagation,

and finally,

g)- the program to achieve these goals cannot depend upon massively expensive testing programs.

II. APPROACH TO APPLYING ZPPR CRITICALS TO DESIGN

A) Traditional Methodology

The technical and institutional requirements discussed above for the application of critical experiments to the US Advanced LMR Program present a dilemma for the ZPPR critical experiments program in that the past 15 year's worth of high quality criticals measurements data from ZPPR have been focused on the oxide fuel form, whereas the metal fuel form is now the centerpiece of the US LMR program because, among others, of properties advantageous for passive shutdown. Moreover, some of the key parameters influencing passive shutdown performance are not amenable to direct measurement in a critical experiment. And finally, with a heightened focus not only on the nominal calculated value but also on the uncertainty of the calculated prediction, a means to address ZPPR critical experiments to not only the traditional best estimate value, but also to a defensible quantification of its uncertainty is needed.

In previous US LMFBR programs the critical experiments have been applied to the design process through the use of bias factors. At a relatively late stage of the reactor design process (which is conducted using the evaluated ENDF data library) an Engineering Mockup Critical (EMC) is assembled, and as many design-related quantities as feasible are measured. The reactor design team models this EMC using their design-level modeling rules and codes to establish the calculated to experimentally measured C/E ratio for the quantities of design interest which are measurable. Then, the best estimate power reactor prediction is determined by:

$$(C')_{power} = (C)_{power} * \left(\frac{1}{C/E}\right)_{ZPPR} \qquad (8)$$

for those quantities which are measurable on the critical. For those quantities which are not measurable, ad hoc corrections are made. Finally, uncertainties are estimated based on historical trends of variation of C/E's for similar EMC's.

But vis-a-vis the current US advanced LMR program's set of technical requirements and institutional boundary conditions this traditional bias factor approach is inadequate in a number of its facets:

- First, it produces licensing-related results rather late in the project's life cycle as a result of resting on measurements from an EMC; but the current need is to establish credibility of the veracity of passive shutdown early in the program to favorably influence R&D funding.

- Second, it is not possible to develop other than an ad hoc estimate for the uncertainties in the calculational predictions based on bias factors from an EMC program; but the current need is for a quantitative bound on the impact of uncertainties -- which is defensible in a licensing arena considering, for the first time, whether to give credit for passive shutdown.

- Third, some of the key reactor performance quantities important to passive shutdown are not amenable to direct measurement in a critical experiment; an example is the burnup control swing which strongly influences the feasibility of passive shutdown of a rod runout TOP.

B) Formal Data Fitting Methodology

In confronting the disparity between the capabilities of the traditional bias factor methodology for applying ZPPR critical experiments results to design and the US LMR Program's current needs in an regime of metallic fuel, design focus on passive shutdown, funding uncertainty, and need for timely, inexpensive, and credible reduction and quantification of neutronics uncertainties, the formal data fitting methodology appears to offer a number of advantages. As extensively developed in the 1970's this methodology updates a multigroup cross section vector, $\underline{T}$, having covariance matrix, $\underline{M}$, to a revised set, $\underline{T}'$ and $\underline{M}'$, by using least squares fitting to find that set of cross section revisions ($\underline{T}'-\underline{T}$) which minimizes the square of the deviation between an ensemble of criticals measurements, $\underline{R}$, and calculations of those measurements, $\underline{C}(\underline{T})$, based on the original cross sections, $\underline{T}$. The formal results are given by:

$$\underline{T}' - \underline{T} = \underline{MG}^T \, \underline{W} \, (\underline{C} - \underline{R}) \tag{9}$$

$$\underline{M}' = \underline{M} - \underline{AWA}^T \tag{10}$$

where

$$\underline{W} = [\underline{GMG}^T + \underline{V}]^{-1} \tag{11}$$

$$\underline{A} = \underline{MG}^T \tag{12}$$

and

$\underline{M}$ = covariance matrix for the cross section, $\underline{T}$

$\underline{V}$ = covariance matrix for the criticals measurements, $\underline{R}$

$\underline{G}$ = matrix of sensitivity coefficients = $\dfrac{\%\text{ change in R}}{\%\text{ change in }\sigma}$

(computed using unadjusted cross sections, $\underline{T}$).

The strength of the data fitting methodology is its ability to both improve the calculational predictions of the critical experiment results to which the fitting is done

$$\underline{C}' = \underline{C}(\underline{T}) + \underline{G}^T(\underline{T}'-\underline{T}) \tag{13}$$

and to reduce their variance to a value which is near that of the criticals measurements -- which are generally of a higher precision than can be calculated:

$$\begin{aligned}(\overline{\underline{C}'^2} - \overline{\underline{C}'}^2) &= \underline{G} \, \underline{M}'\underline{G}^T \\ &= \underline{GMG}^T - (\underline{GMG}^T)^T \, \underline{W} \, (\underline{G}^T\underline{MG})\end{aligned} \tag{14}$$

$$= \underline{GMG}^T \{\underline{I}-\underline{I} + (\underline{GMG}^T + \underline{V})^{-1}\underline{V}\}$$

$$\approx \underline{V}$$

if $\underline{V} << \underline{G}^T\underline{MG}$.

It is noted that the cross sections per se are not necessarily improved either by a movement of their values closer to physical truth or by a reduction of their uncertainties. It is the design predictions as calculated using a specified modeling and computer code set which improve.

Moreover, if the dependences on cross sections of the power reactor are "the same" as those of the ensemble of critical experiments, these advantages carry over to the power reactor as well. For example, for a power reactor having a sensitivity matrix, $\underline{S}$, relating quantities of interest to cross sections, the adjusted cross sections yield corrected power reactor calculational predictions:

$$\underline{C}'_{power} = \underline{C}'(\underline{T})_{power} + \underline{S}^T(\underline{T}'-\underline{T}) \tag{15}$$

and revised uncertainty levels:

$$\left(\overline{\underline{C}'^2}_{power} - \overline{\underline{C}'}^2_{power}\right) = \underline{S}^T\underline{M}'\underline{S} \tag{16}$$
$$= \underline{S}^T\underline{MS} - (\underline{GM}^T\underline{S})^T\underline{W}(\underline{GM}^T\underline{S})$$

If the projection of $\underline{G}$ on $\underline{S}$ is large, the data fitting will lead to a reduction in uncertainty of power reactor quantities -- even for quantities which cannot be directly measured in the critical experiments.

Finally, among the potential institutional-related advantages of this approach are first, that use of the method allows us to benefit from the very substantial historical accumulation of high-quality criticals measurements on a variety of fast spectrum cores (albeit not metal-fueled EMC's) and to thereby produce results for the metal-fueled conceptual core designs prior to the buildup of a comparable multi-year data base for design-specific metal-fueled EMC's. This property of the data fitting methodology facilitates the effort to establish a credibility for passive shutdown early in the program and at low cost.

Second, use of the method permits us to establish a formal, mathematically-well-founded procedure for quantifying the uncertainties in key neutronics quantities important to passive shutdown performance. For a power reactor having sensitivity matrix, $\underline{S}$, whereas the original uncertainties in calculationally-predicted quantities are

$$\underline{S}^T\underline{MS}$$

after data adjustment they are reduced to

$$\underline{S}^T\underline{M}'\underline{S} = \underline{S}^T\underline{MS} - (\underline{GM}^T\underline{S})^T \underline{W}(\underline{GM}^T\underline{S}).$$

This property of the data fitting methodology to replace the ad hoc uncertainty estimates of the bias factor method with estimates which are rigorously founded in the mathematics of least squares fitting is uniquely important in a licensing regime where, for the first time the petitioner will ask that credit be given for passive shutdown properties.

Third, the formal data fitting methodology permits us to reduce uncertainties on even those reactor performance quantities which are not directly measurable on a critical experiment. To the degree that the projection of the power reactor's sensitivity vectors, $\underline{S}$, onto those of the critical, $\underline{G}$, is large, the criticals help to reduce uncertainty in unmeasurable quantities. The salient example here is burnup control swing. Even though a burnup control swing does not occur and therefore cannot be measured on a critical assembly, the uncertainty in its value can be reduced by measurements such as c^{28}/f^{49}, $(1 + \alpha^{49})$, and small sample worth, ρ^{28}/ρ^{49}, measurements on a critical.

And last, once the computational and data management machinery is set in place to implement the formal data fitting methodology, all future relevant experimentation can be added to the cumulative integral data base and will influence an evolutionary, monotonic improvement of design predictions. Not only critical experiments can be brought to bear on design in the existing framework, but so also can operating power reactor data which addresses those phenomena such as fission product poisoning, temperature coefficients and burnout which are not ameanable to direct study in a critical assembly.

D) Interface Between ZPPR Criticals and Core Designers

The formal data fitting methodology possesses a number of features which meet the needs of the current US advanced LMR program. However, unlike the situation in European programs where a uniform methodology is employed nationwide -- permitting the use of a National adjusted cross section set -- the US program must accommodate to the presence of a plethora of modeling rules, unit cell codes, and full-core analysis codes in use among the various industrial contractors and government laboratories. Since the formal data fitting methodology improves the predictions and reduces the uncertainties of reactor integral quantities as calculated using a specified set of modeling rules and computer codes, but does not necessarily improve the cross sections, per se, the production of an adjusted cross section set for widespread use is not appropriate to the US situation of nonuniform design codes and modeling rules.

We do not produce an adjusted cross section set. Instead, the interface between the design team and the ZPPR criticals staff is placed beyond the EMC and beyond the data set, $\underline{T}'$, onto a design-specific calculational "Secondary Standard" which has the properties that:

- it is relevant to the reactor designer's design activity (it is probably a conceptual design which will be refined later)

- its performance quantities of specific design interest are identified and are calculated by the designer using his normal, design-level methods with the unadjusted ENDF cross sections.

Then

- The ZPPR staff calculates the sensitivity coefficients, $\underline{S}$, for this Secondary Standard, using ENDF data and the ZPPR modeling rules and computer codes,

- selects the relevant critical experiments data base,

- performs the formal cross section adjustment for this data base to determine ($\underline{T}'-\underline{T}$) and $\underline{M}'$, using the formal data adjustment methodology, ZPPR modeling rules, and ZPPR calculational codes and methods which correct for those higher order effects which are not well treated by design-level methods.

- and at the same time generates the best-estimate prediction of the Secondary Standard physical performance for the quantities of interest

  $$\underline{C}' = \underline{C}(\underline{T}) + \underline{S}^T(\underline{T}'-\underline{T})$$

- The ZPPR staff also produces the quantified uncertainty estimates

  $$\left(\overline{\underline{C}'^2} - \overline{\underline{C}'}^2\right) = \underline{S}^T\underline{M}'\underline{S}$$

  for these design quantities.

The designer can then note the difference between his design-level predictions of this Secondary Standard and the ZPPR staff's "best estimate" of its actual properties and their current level of uncertainties which resulted from the formal data fitting methodology and can use this information in his design activities in any way that is convenient.

As the project progresses, an EMC critical configuration will be specified based in part on an evaluation of its potential to further reduce uncertainties -- as indicated by an evaluation of the projection of the EMC's sensitivity vectors on those of the power reactor. The Secondary Standard procedure can then be repeated based on the final power reactor design and an extended integral data base which includes the new EMC measurement data.

III. STATUS AND THE FUTURE

In the US advanced LMR program, the passive reactivity shutdown goal has been added to the traditional core neutronics design goals, and the previous focus on breeding performance has been replaced by a focus on high internal conversion ratio to minimize burnup control swing and thence TOP initiator. These design goals have placed increased focus upon use of ZPPR criticals for reduction of the current level of calculational uncertainties in reactivity coefficients and in burnup control swing as compared with earlier US designs where reactivity shutdown relied on control rod scram with rod banks possessing substantial shutdown margin to cover uncertainties, and Beyond Design Basis accidents consequences were mitigated by traditional containment structures. Institutional boundary conditions imposed on the use of the critical experiments in design include quantification and reduction of uncertainties in a timely low-cost way as a means to help establish the credibility of passive shutdown. The traditional methods of applying ZPPR criticals data to design via bias factors from an EMC are ill-suited to the new set of technical and institutional needs.

The formal data fitting methodology has been implemented over the past four years at ANL as a means to bring the ZPPR criticals experiments to bear on neutronics design issues in the US advanced LMR program -- with a stress on uncertainty reduction in calculations of passive shutdown performance. The paper by Collins, et al. describes the criticals data base which has been assembled and regularized. The paper by Poenitz and Collins describes the specifics of the methodology, validation of the data base for internal consistency, and displays a number of relevant examples of the methodology's effaciousness.

In the near term, the application of the methodology to the FFTF metal core reload design and concommitant FSAR revision will provide its first full scale utilization in the US. The Poenitz and Collins paper addresses FFTF design quantities which are measurable on a critical, while the paper by Khalil and Downar addresses the methodology to the reduction of uncertainty in burnup control swing. The formal data fitting methodology will also be applied in support of the industrial sector's advanced LMR licensing interactions with the NRC over the next several years, and in support of the SP-100 space reactor ground test design, fabrication and test programs.

As for the continuing refinement of the methodology and extension of its applications, the paper by Hwang addresses interpretive methodologies for relating the power reactor's dependence on cross sections to that of the ensemble of critical assemblies and also discusses work in progress to incorporate a rigorous treatment of the differences in space and energy shelf shielding which exist in the criticals vis-a-vis the power reactor. The paper by Orechwa initiates a broader view of the advanced LMR design accuracy requirements than has been taken in the present work where the focus was on uncertainty reduction in passive shutdown performance; the closed, fissile self sufficient fuel cycle employed in the US advanced LMR program brings depletion dependences and nuclear properties of minor actinides, fission products, and waste streams into stronger focus than in past US LMFBR cycles. Over the next five years, burnup measurements data from EBR-II will be added to the data base as a part of the program to address these depletion-dependent issues.

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Table 1. Components of Power and Power/Flow Reactivity Decrements and of T_{inlet} Coefficient of Reactivity

A (¢) =	$\{\ \alpha_D$ +	$\left[\begin{array}{ll} 0 & \text{bounded to clad} \\ \alpha_\ell & \text{free of clad} \end{array}\right.$			$\}\ * \overline{\Delta T_f}$ (ave. fuel - ave. coolant) temperature
B (¢) =	$\{\ \alpha_D$ +	α_ℓ	+ α_{Na} +	$2\alpha_R$	$\}\ * \dfrac{\Delta T_c}{2}$ (ave. coolant-T_{inlet}) temperature
C (¢/°C) =	$\{\ \alpha_D$ +	α_ℓ	+ α_{Na} +	α_R	$\}$
	Doppler	Fuel axial expansion	Na density	Radial expansion	
Typical Size ¢/°C	-0.05 to -0.1	-0.1	+0.15 to 0.2	-0.2 to 0.3	
	Associated with fuel temperature	?	Associated with coolant temperature		

Table 2. Quasi Static Reactivity Balance Results for Unprotected Accidents

	Asymptotic State				Intermediate State	Indicated Trend for Inherent Shutdown*
	P	F	δT_{in}	δT_{out}		
LOHS	≈ 0	1	$\frac{A+B}{C}$	$\left(\frac{1+A/B}{\frac{C\Delta T_c}{B}} - 1\right)\Delta T_c$	Monotonic transition to asymptotic	• A+B small • C large
TOP	≈ 1 (after rise in T_{in} due to BOP heat removal limit)	1	$\frac{\Delta\rho_{TOP}}{-C}$	$\delta T_{out} = \delta T_{in}$ $= \left(\frac{\Delta\rho_{TOP}/B}{-C\Delta T_c/B}\right)\Delta T_c$	Initial rise at constant T_{in} $P = 1 + \frac{-\Delta\rho_{TOP}/B}{1 + A/B}$ $\delta T_{out} = \left(\frac{-\Delta\rho_{TOP}/B}{1 + A/B}\right)\Delta T_c$	• A+B large • C large • $\Delta\rho_{TOP}$ small
LOF	≈ 0	Natural Circulation	0	$(A/B)\Delta T_c$	overshoot relative to delayed neutron hold-back of power decay minimized if $\Delta\tau(1+A/B)^2\|B\| >> 1$	• A small • B large • τ long
Chilled Inlet	$1 - \frac{C\delta T_{in}}{A+B}$	1	$\|-\delta T_{in}\| \le (T_{inlet} - T_{Na\ freeze}) \approx 1.5\ \Delta T_c$	$\left(\frac{C\Delta T_c/B}{1 + A/B} - 1\right)(-\delta T_{in})$	monotonic transition	• C small • A+B large
Pump Overspeed	$\frac{1 + A/B}{1/F + A/B}$ (always > 1)	F>1	0	$\left(\frac{1}{1 + \frac{A/B}{1+A/B}(F-1)}\right)\Delta T_c$ (always < 0)	monotonic transition	• A negative • B negative

*Conflicts are seen to exist between desirable trends for different ATWS events. Resolution of these conflicts is discussed in the text.

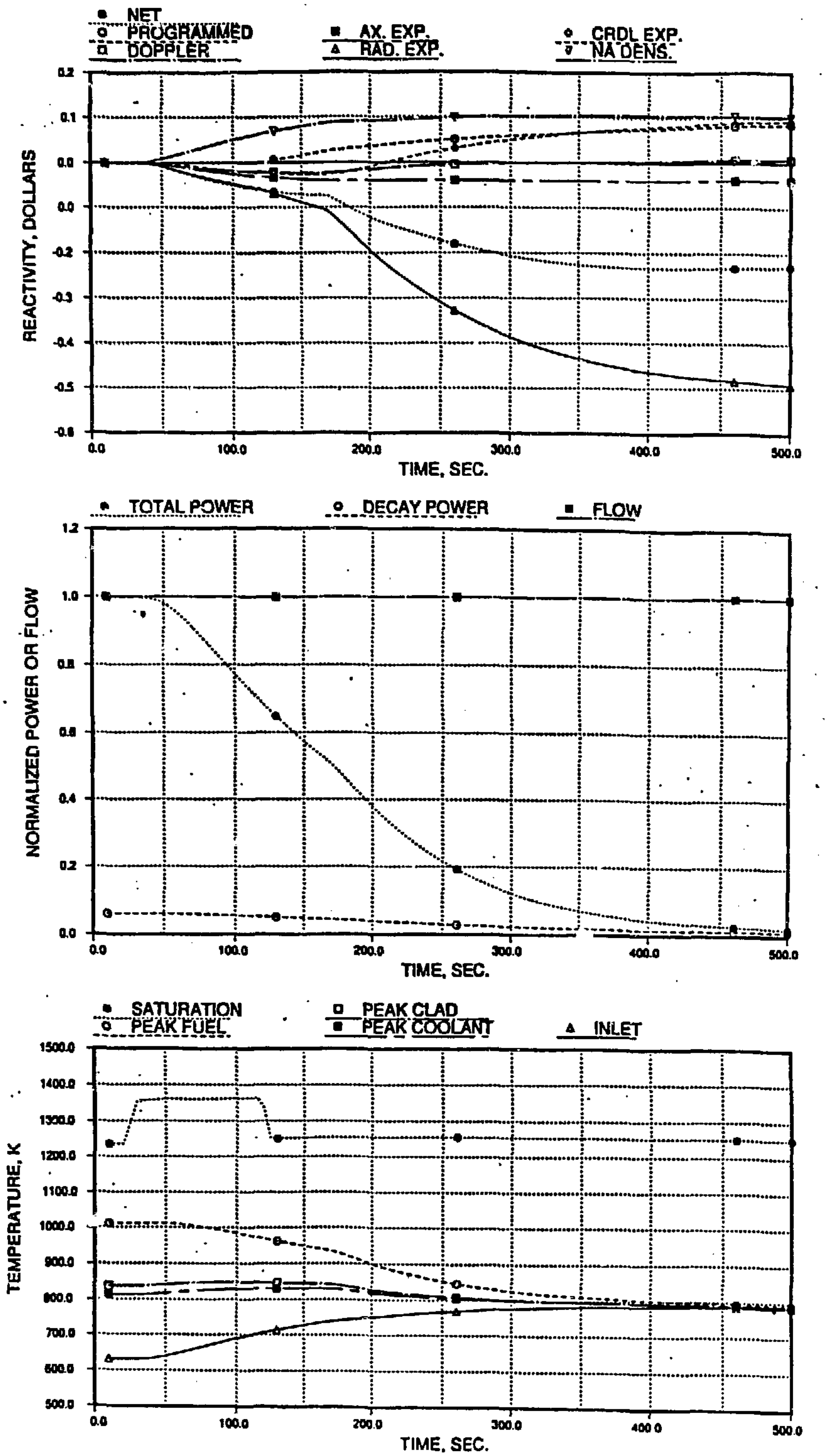

Fig. 1. Passive Shutdown of Loss of Heat Sink ATWS Event in a 900 MW_{th} metal-fueled LMR ("Programmed" reactivity denotes vessel axial expansion)