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— Topical Report —

1000-MWe LMFBR Accident Analysis
And
Safety System Design Study
Accident Initiating Conditions
Part 2, Reactivity Insertions

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1. INTRODUCTION

Babcock & Wilcox is performing work on the Accident Analysis and Safety System Design Study under ANL Contract No. 31-109-38-2339. Various utility companies are participating in the work on a cost-sharing basis, under separate agreements with B&W. The study is being conducted primarily to gain a better understanding of how safety requirements affect the design of large, liquid-metal, breeder reactors. This objective will be met by analyzing accidents that determine the design bases of the protective systems and safety features and by developing conceptual designs for these protective systems and engineered safety features. The reference design¹ of B&W's 1000-MWe Follow-On Study is the basis of this study.

The accurate analysis of accidents generally requires as precise a definition of the initiating events as is possible, since even relatively small uncertainties in the definition of initiating conditions could lead to large uncertainties in the accident analyses. Activities 210 and 211 of the study involve the definition of such initiating conditions. Several broad categories of accident-initiating events were identified from the malfunction survey performed during Phase I². Briefly, these categories are as follows:

1. Accident-initiating conditions arising from primary coolant abnormalities.
2. Accident-initiating conditions arising from secondary coolant loop abnormalities.
3. Accident-initiating conditions arising from reactivity insertions.

Studies involving the first two categories have been completed, and the results are reported in BAW-1350, Initiating Conditions, Part 1, Flow Abnormalities. The models, methods, calculations, and results of the third category of analyses are discussed here. In this report, initiating conditions are defined as proposed in BAW-1350; that is, a

transient is considered an initiating event until irreversible damage occurs. The consequences of irreversible damage will be investigated under Activity 220, Accident Analysis and DBA selection.

A review of the Phase I fault trees and malfunction catalog revealed a number of events that could cause accidental reactivity insertions. Among these are some that are patently incredible; others, while credible, are of little significance to this study. A comprehensive list of such malfunctions or events follows:

1. Single rod withdrawal at rundown speed.
2. Continuous withdrawal of four regulating rods at shim speed.
3. Gas bubble(s) entrained in the coolant.
4. Loading accidents.
5. Moderator entrainment.
6. Sudden fuel movements.
7. Sudden drop in inlet temperature.

In addition, several arbitrary reactivity insertions were investigated. These calculations were designed to provide data for the design of the protection system. Flow blockages were excluded from this study because blockage events are being investigated in Activity 220, Accident Analysis.

The results of the analyses are as follows:

1. The classic loading accidents are incredible in the reference design.
2. No large sources of moderator are available to the primary coolant.
3. The gas-bubble entrainment event will lead to some fuel melting if a sufficiently large fraction of the core participates.
4. The maximum unbounded ramp that the reactor can accommodate without cladding failure before shutdown is about 5\$/s. Cladding failure occurs for bounded ramps when 1.25\$ is inserted in 0.1 second.

2. SUMMARY

The consequences of reactivity insertions arising from definable malfunctions or events have been investigated. The specific events that were identified, without regard to credibility, from the Phase I work are as follows:

1. Single rod withdrawn at 30 in. /s.
2. Continuous withdrawal of four regulating rods at shim speed.
3. Gas bubble(s) entrained in the coolant.
4. Loading accidents.
5. Moderator entrainment.
6. Sudden fuel movement.
7. Drop in inlet temperature.

All of these accidents were investigated to some extent; the last four were eventually dropped when the investigations indicated that the events required to produce dangerous conditions were extremely improbable.

In addition to the mechanistic study outlined above, a series of calculations was performed to determine the dynamic response of the reactor to arbitrary ramp reactivity insertions. This permitted us to define the parameters related to the protection system design, to determine the intrinsic stability of the reactor, and to provide a larger scope of investigation than was possible with mechanistically determined reactivity insertions.

The results of these investigations are summarized as follows:

1. No protected event leads to fuel melting, coolant boiling, or cladding failure.
2. The gas-bubble accident, if unprotected, will lead to severe melting. At least 8.0 ft³ of entrained gas must flow through the central seven assemblies at 20 fps in order to produce fuel melting.

3. The loading accident, the moderator entrainment accident, the fuel movement accident, and the inlet temperature drop accident are so improbable that they are incredible in the reference design.

4. The reference design can accommodate an unbounded ramp of up to 5\$/s or an insertion of 1.25\$ in 0.1 second without fuel pin failure.

3. DESCRIPTION OF WORK

3.1. Objective

The analyses presented here were performed to provide data and initial conditions for the accident analysis activity and to aid in selecting the DBA(s). In addition, information on the relative importance of various core parameters generated during this activity provides guidance for the protective system design effort.

All of these analyses are based on the Phase I results, particularly the malfunction catalog. The technical approach to the analysis of reactivity insertions, as well as the methods, models, and calculations for these insertions, is discussed in detail in this section. The results of analysis, including appropriate sensitivity studies, are also presented. Section 4 presents an evaluation of this work in relation to the Accident Analysis and Safety System Design Study.

3.2. Technical Approach

Since a limited number of events lead to reactivity insertions in the reference design, it was possible to examine each event without having to analyze bounding events. This is in direct contrast to the analysis of flow abnormalities, where it was convenient to study accidents that bounded the severity of all possible accidents.

Owing to their nature, certain of the events examined here were dismissed on the basis of credibility considerations. For the remainder of the events examined, it was necessary to determine the time sequence of the events along a fault path.

To determine the causal sequence of events, the Phase I fault trees and malfunction catalog were reviewed. The malfunctions were then analyzed to determine their effect on the system. The effects were then examined by various numerical techniques to determine the core's dynamic response. Next, the most sensitive and/or uncertain parameters were varied so that their influence on the accident could be evaluated.

The transients arising from the identifiable malfunctions were recorded for parameters of interest, such as fuel, cladding, and coolant temperatures, power level, and components of the reactivity feedback. The objective of the analyses was to determine the time and conditions of incipient cladding failure. The analyses performed do not consider the course of the transient beyond cladding failure; this transient is being examined under Activity 220, Accident Analysis.

3.3. Methods and Analytical Models

The results of a safety analysis are likely to be very sensitive to the calculational models used. Therefore, the methods or models used in the analyses of reactivity-insertion initiating conditions are discussed in some detail in this section.

3.3.1. General

The point kinetics version of the TART³ program was used to analyze the transient response of the reactor in all of the studies. The reactor was represented by seven thermal-hydraulic regions: one region represented the hot pin; three regions represented the first enrichment zone, including the peak or maximum powered pin; one region each represented the second and third enrichment zones, and a single region represented the radial blanket. The power and sodium worth distributions were representative of the end-of-cycle (EOC) reactor, and values were taken from the B&W 1000-MWe Follow-On Study.

An approximate model for sodium voiding was used in the TART program. In the event that any initiating condition leads to sodium temperatures above 1700F, voiding is assumed to begin. The sodium-sodium vapor interfaces are conservatively assumed to move from the point of voiding inception at a rate of 50 ft/s in both directions. This rate is somewhat high; however, in the presence of the large sodium void coefficient of B&W's reference case, this high rate leads to larger and hence more conservative reactivity additions.

The EOC fuel pin was modeled using nine axial segments with 10 radial nodes. Again, axial conduction was neglected. A central void in the pin of 0.003-foot radius was assumed. The conductivity function was taken from the TAMPA program. By this means the effect of fuel densification in the EOC pin was approximated. The results

obtained with this pin model agree very well with the conditions predicted by the TAMPA program; the centerline temperature agreed to within 50F.

The fuel conductivity function utilized in the TART program is shown in Figure 3-1. Note that, contrary to the TAMPA formulation, the fuel conductivity function in TART has no true discontinuities. Finiteness of slope was preserved everywhere in order to improve convergence characteristics in the TART program.

3.3.2. Protection System Model

The protection system contains 25 rods; seven of these are safety rods and are thus out of the core during operation. The remaining rods are used as shim rods, which are inserted at various positions throughout the cycle. The distribution of rods among the five TART thermal-hydraulic regions was varied for these analyses depending on which point in time during the cycle produced the most unfavorable configuration for the accident in question. For beginning-of-life conditions the configuration was as follows:

<u>Region No.</u>	<u>No. of rods</u>	<u>Configuration</u>	<u>Reactivity available for scram</u>
1	1	All out	-0.00335
2	3	75% in	-0.0021
	3	All out	-0.0084
3	6	All out	-0.0168
4	4	All out	-0.0112
	2	75% in	-0.0042
5	<u>6</u>	All out	<u>-0.0168</u>
Total	25		-0.06285

This distribution approximates the minimum rod worth available at scram. The central rod is worth 92¢ and all other rods are worth 77¢ each. The rod worth curve, fractional worth versus fraction inserted, is presented in Figure 3-2.

Figure 3-1. Fuel Conductivity Vs Temperature

Figure 3-2. Rod Worth Vs Fraction Inserted

The scram setpoints for the core instrumentation were 2621 MWt for the power monitor (107% power) and 1139F for the bulk outlet temperature monitor. The bulk outlet thermocouples were located just above the upper axial blanket outlet in the outlet plenum. A 5-second time constant was used for this thermocouple.

The scram delay time of 300 ms is defined as the elapsed time between the detection of an out-of-limits condition and the first motion of the rods. The nominal rod speed for shim motion is 15 inches per minute; the scram acceleration of 0.7g is backed up by a secondary motor-driven scram at 30 inches per second.

3.4. Description of Calculations

This section describes the calculations performed in this segment of Activity 210; pertinent input variables, initial conditions, and assumptions are discussed for each major initiating event.

3.4.1. Rod Withdrawal Accidents

During normal operation the reactivity control system maintains the position and velocity of all control rods as directed by the reactor protection and control systems and by the operator through manual commands. The reactivity control system is designed so that the number of potential faults is extremely limited; however, if they did occur, the system would provide a potential source of reactivity addition. Although the probability of these faults is small, it is important to be able to predict the response of the reactor system as a result of these failures.

3.4.1.1. Scope of Calculations

Two malfunctions are considered here:

1. The withdrawal of a control rod at 30 in./s due to failure of a secondary drive motor.
2. The continuous withdrawal of four "ganged" shim rods at 15 in./min.

The B&W control rod drive unit comprises two synchronous motors that operate through an epicyclic gear train consisting of a combination ring gear and fly wheel, three planet gears, a planet cage, and a sun gear. For convenience, the upper drive motor is called

the primary motor and the lower one is called the secondary motor. Whenever the reactor is critical, both of the motors in a drive operate continuously while linked to the gear train. No clutches or mechanical disconnectors are used.

A noninterruptible vital bus provides power for the secondary drive motors. Power for the primary motors is obtained from two separate nonvital buses, either of which can power all primary motors. Synchronism is maintained between the vital and non-vital buses. The rod-hold electromagnetic latch and primary motor brake for each rod receive power from the same feed as the primary motor for that rod. Setback, rundown, and control operations are accomplished by switching primary motor power to one of three motor windings corresponding to no-motion, plus 15 in./min, or minus 15 in./min. Scram is accomplished by interrupting primary motor power so that the secondary motor, acting through the gear train, drives the rod into the core. Conversely, if the secondary motor failed, the primary motor would drive the rod out of the core at 30 in/s. The secondary motor cannot fail through loss of power since it derives its power from a vital bus. However, faults could occur through bearing failure or flywheel seizure.

The "ganged rods" could be withdrawn because of operator error. It is possible to conceive of situations in which the three ganged rods could be continuously withdrawn; in addition, if a jumpered condition existed in a particular control switch, one additional rod could be withdrawn at the same rate, 15 in./min.

3.4.1.2. Reference Calculations

The single-rod withdrawal event is discussed first. Since the rod of maximum worth (the central rod) is a safety rod, it is in the core only when the core is subcritical; hence, its ejection by secondary motor seizure is not possible. The rod used for this analysis is one of six in the first radial ring of control rods. A single such rod is conservatively assumed to be worth 0.82\$. For automatic shim control at the beginning of life, such rods may be ganged and inserted as much as 75% of their length. In this study it was assumed that the secondary motor on one of these rods failed. It should be noted that

even for ganged rods no more than a single rod may be ejected. Even if a vital bus failed, which is incredible, no more than one rod would be ejected since each of the three ganged rods is on a separate vital bus. The event was analyzed with the TART model described in the previous section. The protection system was modeled according to that description and was assumed operable. The reactivity insertion function for a rod worth 0.82\$ being withdrawn at a rate of 76.2 cm/s is shown in Figure 3-3. This figure displays the relationship between percentage worth, time, and reactivity.

In the second event under consideration, four rods are continuously withdrawn at a shim speed of 15 in./min. The four rods were located in the first radial ring of control assemblies; their worth was conservatively assumed to be 0.76\$ each. The distribution of the rod reactivities among the core regions is identical to that tabulated in section 3.3.2 except that the reactivity available in region 2 was adjusted to account for the insertion of an additional rod. All rods were assumed to be fully inserted in violation of procedural limits at the inception of the failure.

In both cases the reactivity feedback coefficients and nuclear parameters used in this analysis are as follows:

Doppler coefficient, $\left(T \frac{dk}{dT}\right)$	-0.00533
Sodium density coefficient, $\left(\rho \frac{dk}{d\rho}\right)$	+0.0253
Axial expansion coefficient, $\left(H \frac{dk}{dH}\right)$	-0.158
Linear expansion coefficient, $\left(\frac{1}{H} \frac{\delta H}{\delta T}\right)$	0.0000104
β_{eff}	0.00364
$\ell^*, \text{ s}$	0.29×10^{-6}

3.4.1.3. Results

The power trace following the failure of a secondary motor on a single rod drive unit is shown in Figure 3-4.

Drive failure was assumed at time zero; the power rises sharply, crossing the 107% overpower trip at 0.14 second. It was conservatively assumed that the magnetic clutch on the drive unit did not function as designed and therefore did not disengage. The scram began after a 300 ms delay, and the power peaks at 3310 MWt, 0.481 second after the accident began. The maximum cladding temperature was 1186.4F (7.4F rise), and the maximum fuel temperature rise was 48.6F in the hot channel. The excess integrated energy was 188 MW-s.

The transient following the continuous withdrawal of four rods is somewhat different (Figure 3-5). The power is slowly varying although there is an inflection in the power trace due to the increasing slope of the rod worth curve. A scram was produced by the power monitor at 4.90 seconds. The maximum power of 2638 MWt occurred 5.2 seconds following initiation of rod withdrawal. The maximum cladding temperature was 1196F, and the maximum fuel temperature rise was 91.4F. The excess integrated energy for this transient was 398 MW-s.

The larger integrated energy associated with the slower four-rod withdrawal ramp is to be expected. This phenomenon demonstrates one of the principal axioms of protective system design—that the further one removes the scram setpoint from operating conditions, the more vulnerable one becomes to very slow insertions of reactivity. Although the integrated energy in this accident is relatively high, fuel melting did not occur, and the cladding temperatures were acceptable.

3.4.1.4. Sensitivity Studies

A number of parameters may affect the course of the reactivity-induced transient following malfunctions involving control rods. Unfortunately, there are a number of uncertainties in these parameters, and the influence of certain of the most sensitive of these parameters on the course of the excursions requires investigation. Two parameters were investigated in connection with the withdrawal of a single rod at 30 in./s: (1) the reactivity coefficients and (2) the degree of insertion of the rod at the inception of the transient. The variation of rod worth was not investigated since this variable was adequately investigated in the series of arbitrary reactivity insertions that were studied.

Figure 3-3. Reactivity Trace — Single Rod Withdrawn at Rundown Speed

Figure 3-4. Initiating Conditions — One Rod Withdrawn at Rundown Speed

Figure 3-5. Initiating Conditions — Four Rods Withdrawn,
Each Worth \$0.76

The reactivity coefficients and the initial insertion length were varied independently to assess the relative importance of each. The protection system was assumed to be functional for each of the transients studied. The values of the reactivity coefficients used were (1) 0.5 nominal Doppler with all other coefficients nominal and (2) 1.5 times the nominal sodium density coefficient with all other coefficients nominal.

In one case it was assumed that the procedural controls on a maximum rod insertion of 75% were violated and that the rod was fully inserted when the secondary drive motor failed. The results of this transient are similar to those for the reference case, since in any event the scram is produced at 2621 MWt. However, since the fully inserted rod is initially positioned on a segment of the rod worth curve where the slope is small, the power overshoot is less severe than in the reference case. For the fully inserted rod, the maximum power of 3251 MWt occurred 0.568 second after the initiating malfunction. The fuel did not melt, and the cladding temperature rise in the hot channel was only 1.1F. Therefore, it is reasonable to assume that the cladding did not fail.

The effect of the variation of feedback parameters on the single rod withdrawal transient can be seen in Figure 3-6, which displays the power traces corresponding to the three conditions of the feedback parameters. It is evident that in this case, as in the case of transients arising from flow abnormalities,⁵ that any "worsening" of the feedback parameters leads to a power rise that is more rapid than nominal with a consequent earlier-than-nominal scram. Thus, for larger positive or lower negative feedback parameters, the energy yield of the excursion is reduced.

In contrast with the flow abnormality initiating conditions, the Doppler coefficient has the greater effect in these calculations. This is certainly to be expected since the effect of a reactivity transient is more immediately felt as an increase in fuel temperature, which is the driving effect for Doppler feedback. The most immediate effect of a flow abnormality is the change in coolant temperature; hence, for flow abnormalities the sodium density coefficient has the most pronounced effect.

Figure 3-6. Power Traces — Single-Rod
Withdrawal Accident

The results of these calculations are summarized as follows:

<u>Case</u>	<u>Max. fuel centerline rise, F</u>	<u>Max. cladding temp rise, F</u>	<u>Excess integrated energy, MW-s</u>
Nominal	48.6	7.4	187.9
1.5 Na density	41.3	6.0	154.6
0.5 doppler	40.0	5.0	147.6

It is immediately evident that the consequences of this transient are trivial regardless of the values of the feedback coefficients.

For the continuous withdrawal of four shim rods, the parameter investigated was the variation of reactivity feedback coefficients. The values used were the same as those in the previous case: (1) nominal values, (2) 0.5 of the nominal Doppler, and (3) 1.5 times the nominal sodium density coefficient. Again, the protection system was assumed operable.

The power traces resulting from these transients are shown in Figure 3-7. The behavior of the power and temperature is predictably similar to those for the previous case. Once again the worst coefficients lead to the smallest transient energy generation. The results are summarized as follows:

<u>Case</u>	<u>Max. centerline fuel temp rise, F</u>	<u>Max. cladding temp rise, F</u>	<u>Excess integrated energy</u>
Nominal	91	24	398
1.5 Na density	66	16	259
0.5 doppler	59	15	238

Figure 3-7. Power Traces — Four-Rod Withdrawal Accident

3.4.2. Gas Bubble Accident

Because of the rather large positive sodium density coefficient of reactivity any event resulting in the removal of sodium from the core is potentially serious. Apart from void formation due to sodium, the event that leads to the most severe conditions is the entrainment of gas bubbles in the sodium coolant.

The argon cover gas is a potential source of gas that may be entrained in the primary coolant. Argon is somewhat soluble in sodium at the temperatures characteristic of the primary system; additionally, a large interfacial area exists between the primary sodium and the cover gas. It is therefore conceivable that a certain amount of argon can be dissolved in the primary system under certain circumstances.

Another potential source of gas in the primary system is the gaseous fission products released from the fuel vents. Certain of the gaseous fission products have a chemistry similar to argon and are soluble in hot sodium.

The concern over dissolution of gases in the primary system arises because the temperature differences in the system may cause some of the gas to precipitate from the coolant and to be swept into the core region. Such an event would lead to momentary voiding of regions of the core and initiate a reactivity transient. The severity of the transient depends, of course, on the location of the voided channel or channels since the magnitude of sodium density feedback reactivity is spatially dependent.

There are two aspects to the gas-bubble problem: reactivity effects and thermal-hydraulic effects. The reactivity effects arise solely as a result of removing sodium from regions of the core in which entrained gas is flowing. As associated phenomenon occurs during bubble passage, however, and its effect is principally thermal-hydraulic. As the bubble passes a particular axial location, it in effect removes cooling at that location. Depending on whether or not a sodium film is left on the pins, the cladding materials may experience a precipitous rise in temperature. After the bubble passes, the returning sodium is then exposed to elevated temperatures, which could precipitate coolant boiling.

To cause any difficulty, a relatively large volume of gas must flow through some portion of the envelope of positive sodium worth in the core. The gas precipitated in the intermediate heat exchange is likely to be in the form of a very few small, discrete bubbles. In view of this, it appears that these small bubbles would have to be collected and coalesced in order to produce any significant effect. In the absence of a collection location the probability of entrained gas leading to an accident is very low.

3.4.2.1. Scope of Calculations

Traditionally, the analysis of fast reactors has evaluated the consequence of the coherent passage of a large bubble (enveloping the entire positive worth region of the core) through the reactor even though such passage is physically unrealistic. The undetected entrainment of large volumes of gas is improbable, and even if postulated, the large volume would surely be broken up and dispersed by the flow turbulence in the primary system.

The objective of this analysis was to determine how much gas must flow in the central seven assemblies to produce incipient fuel failure in the unprotected reactor. It was assumed that a large bubble of undetermined size enters a primary pump and is broken up by the coolant flow. After breakup it was assumed that the smaller bubbles remain very closely packed in a random array. The void fraction for this foam is about 61% based on values in Perry's Chemical Engineering Handbook⁶. This representation was used in all of the analyses. It was further assumed that the velocity of bubble passage is the same as the steady-state coolant velocity in the core, 20 fps.

3.4.2.2. Reference Calculations

The reactivity trace produced by the passage of a 61% void foam through the reactor is shown in Figure 3-8. The most important feature of this trace is the negative effect produced as gas flows through the axial blankets. The control system was assumed to be inoperable.

The resultant power trace is shown in Figure 3-9. The power rises quite rapidly as the gas flows into the core, but since the flow was postulated to be continuous, the power quickly reaches

a near-equilibrium value of 5480 MWt in 0.2 second. The maximum cladding temperature as a function of time is shown in Figure 3-10. Fuel failure will probably occur between 0.8 and 1.0 second after the beginning of the transient because of the molten fuel expansion mechanism. Fuel melting began 0.4 second into the transient. At this time approximately 8 ft³ of gas has flowed into the core.

3.4.2.3. Evaluation of Gas-Bubble Accident

Although 8 ft³ of gas could possibly be entrained in the system over an extended period of time, it is extremely unlikely that that amount would collect at some point and be suddenly released into the primary system. Design features may be included in the primary design to disentrain the gas and/or to prevent its collection. In particular, the intermediate heat exchangers could be equipped with gas bleed lines to disentrain gases. In addition, the primary pump outlets could be equipped with suitable electromagnetic flowmeters to detect the passage of significant volumes of gas into the inlet plenum.

However, the most important aspect of this accident can be seen by inspecting Figure 3-9. This trace assumes no scram; it may be seen that the reactor power passes the overpower trip at 0.03 second. If the protective system operates as designed, the transient may be terminated before fuel melting begins, even with a 300-ms scram delay time. Since there is no reason to believe that the protection system will fail to produce a scram, this accident, viewed realistically, will not produce fuel melting or cladding failure.

In summary, the reference protection system is capable of terminating the gas bubble transient without damage to the system. Even in the absence of protection an exceptionally large volume, 8 ft³, of gas is required to produce fuel failure. The results also indicate that the cladding temperatures remain low enough to prevent sodium boiling in the two-component mixture flowing through the channel.

Since the consequences are not unduly severe for the accidents analyzed, and since there are potential design features that could obviate the accident, this event can be removed from the category of potential DBAs.

Figure 3-8. Reactivity Trace — Gas-Bubble Accident

Figure 3-9. Power Trace — Gas-Bubble Accident

Figure 3-10. Maximum Cladding Temperature Vs Time — Gas-Bubble Accident

3.4.3. Other Initiating Events

A number of other initiating events were identified from the malfunction catalog or fault trees. These devices, however, were constructed without regard to probability or, equivalently, credibility. As a consequence, many of the events thus identified have a very low probability of occurring and may be eliminated on grounds of credibility. Several events fall into this category: (1) loading accidents, (2) moderator entrainment, (3) sudden fuel movements due to thermal ratcheting, and (4) sudden drops in the coolant inlet temperature. Credibility considerations are emphasized in discussing these events.

3.4.3.1. Loading Accidents

The classic loading accident is postulated based on the initial assumption that the core has been misloaded to the extent that the reactor is just critical at zero power. At this point the assembly of maximum worth is dropped into the location of maximum worth. Criticality during loading is assumed to arise as a result of (1) rods being inadvertently withdrawn, and this condition remaining undetected, and/or (2) gross misloading of fuel.

In B&W's reference design it is physically impossible for the rods to be out of the core during refueling. The rods must be in the core with the drives disconnected before refueling can take place. Because of the weight of an individual rod, the rod cannot be lifted from the core by hydraulic pressure even under full-flow conditions. With all rods in the reactor the system is subcritical by several percent.

A study was undertaken to determine whether the core could be misloaded so as to be just critical with all rods in the reactor. A criticality search for the reference core with all rods in was performed with the one-dimensional diffusion theory code FARED⁷. The results indicate that a uniform enrichment >14% fissile is required to achieve criticality with rods in. Since this enrichment is significantly greater than the maximum enrichment in the reference design, criticality with rods in is deemed impossible. Since criticality cannot even be approached during refueling in B&W's reference design, the loading accident, in the classic sense, is clearly impossible and was not considered further.

3.4.3.2. Moderator Entrainment

If a substance having substantially different moderating properties from the sodium coolant were to be introduced into the reactor primary system, a reactivity insertion could take place. This effect would be pronounced if the foreign substance were either hydrogen or carbon.

The introduction of hydrogenous material into the reactor results in two competing reactivity effects: a gain in reactivity due to decreased leakage in the lower energy groups, and a loss of reactivity due to the decrease in k_{∞} at low energy. The positive component predominates in general.

For the Fermi reactor⁸ it has been estimated that a substance with the hydrogen density of water flowing at 20 ft/s through the core would result in a reactivity ramp of 800\$/s. A concentration of 0.01 Vol % of hydrogenous material, such as oil, will cause a reactivity increase of several cents.

Further, the inleakage of oil to sodium systems is not entirely without precedent. In May 1959 a thermocouple well failure in the SRE (Sodium Reactor Experiment) led to between 2 and 10 gallons of Letralin being admitted to the primary sodium.⁹

Thus, it is seen that the exclusion of moderating material from the primary system should concern the designer of fast reactors. In B&W's reference design only two sources of moderator are associated with the primary system. The first is the oil-filled eddy current couplings between the primary motors and their shafts, and the second is the carbon-filled cans of the radial neutron shields. A schematic diagram of the primary pump arrangement is shown in Figure 3-11. It can be seen that the immediate consequence of a multiple seal failure that releases oil is the leakage of oil onto the charge face of the reactor. From here there are two additional seals between the oil and the primary coolant. In any event, such leakage is most likely to be confined to the charge face. Thus, this mechanism does not seem very important as a potential source of introducing moderator to the primary coolant.

Figure 3-11. Primary Pump Schematic Drawing

The graphite logs of the radial neutron shields are solid and canned in stainless steel. If failure of several shield cans (an extremely low probability event) is postulated, it is unlikely that significant quantities of graphite would be released. In all likelihood the graphite would rapidly log with sodium, as it did at Hallam, and release a negligible amount of moderator to the coolant stream. Because of these considerations the rapid release of sufficient quantities of moderator to cause reactivity effects is deemed extremely remote. Slow leaks of these substances into the primary system have little effect on safety, and they can be detected rather easily. Hydrocarbon vapor is readily sensed in the argon cover gas, and carbon is easily cold-trapped.

3.4.3.3. Sudden Fuel Movements

Only the thermal ratcheting of fuel is of concern here. Other types of fuel movement, such as fuel slumping or melting, are consequences of initiating events and are not initiating events themselves. As such, these events properly belong in the category of accident analysis.

In pellet fuel pins it is known that thermal cycling will cause unrestrained pellets to move axially away from the midplane of the core. Substantial amounts of reactivity may be tied up in such motion, and if at some later time the axial voids in the fuel column collapse under the action of some shock, then this reactivity may be reintroduced rather rapidly. It is not known now whether vipac fuel can be made to ratchet in a manner similar to pellet fuel.

In order to understand the problem, consider the mechanism of thermal ratcheting¹⁰:

The movement of pellets through a tube is conveniently considered by referring to the simple model illustrated in Figure 3-12. When a hot pin begins to cool, the pellet of radius r and length $2b$ is effectively pushed through the tube by the adjacent pellet with a force whose point of application is at a distance y from the axis. The value of y is determined by imperfections in the shapes of the pellets and perhaps by the presence of abraded particles of oxide. If y is small the pellet will slide rather easily through the tube, but if y is large the pellet will tilt and jam.

At a unique value of y between these extremes, the pellet will slide without jamming if adequate force is applied. This condition is illustrated in Figure 3-12, and it can readily be shown that the corresponding value of y is equal to (b/μ) , where μ is the coefficient of friction.

Even from so simple a model as this it is easily seen that thermal ratcheting is complex and depends on several thermo-mechanical effects.

In vipac fuel, where the fuel column is a continuous homogeneous medium, it is unlikely that enough radial cohesion is present in the pin to cause the mass to slip coherently through the tube. In particular, the forces generated axially are likely to be considerably reduced in a column of vipac fuel.

At this time it is felt that thermal ratcheting in vipac fuel is improbable at best. Even if thermal ratcheting were possible the loss of reactivity would probably become apparent during operation owing to unexpected motion of the shim rods. In this event, safety would not be compromised since a reactivity inventory would betray the advent of a dangerous condition. Caution should be used, however, in interpreting reactivity accounting results since some small, unforeseen condition might introduce enough positive reactivity to mask the negative effects of thermal ratcheting.

3.4.3.4. Sudden Drop in Inlet Temperature

In principle, a sudden drop in the inlet coolant temperature could lead to a reactivity-induced transient. This effect arises because the fuel is momentarily overcooled, leading to a fuel temperature drop and a consequent positive Doppler feedback.

In B&W's reference design this accident has a low probability for several reasons:

1. The primary pump suction is taken from the large pool of sodium. The mass of the sodium in this pool is about 1.2×10^6 pounds, so that any temperature transients affecting the bulk temperature of this mass are necessarily slow because of the very large heat capacity.

2. There are no separate closed loops subject to isolation and sudden reactivation.

3. The intermediate system cannot remove enough heat, in an unbalanced condition, to cause a rapid lowering of pool temperatures, even with imperfect mixing.

Since no mechanism leading to a sudden drop in inlet temperature can be defined, we do not believe that analysis of the event is warranted.

3.4.4. Arbitrary Reactivity Insertions

In a series of calculations the reactor was subjected to (1) unbounded ramps of varying rates and (2) total insertion times of varying rates. The objective was to determine the maximum ramp rate or combination of rate and insertion time that could be controlled to ensure that the power returned to the steady-state value before either coolant boiling or cladding failure occurred. The assumptions made in the study are as follows:

1. The control system is operable.
2. A scram is included at the reference setpoints.
3. The cladding failure states are as given in BAW-1350, Appendix A.
4. There are seven thermal-hydraulic regions in the core.

The calculations used the TART code to study core behavior following ramped reactivity insertions. In all cases the core was protected and would scram at several trip points: 107% overpower, 1139F bulk outlet temperature, 1700F sodium void temperature, and 1.8 power-to-flow ratio. The core model was set up with seven regions of composition, one corresponding to the hot fuel pin and associated materials. The input parameters were chosen so that voiding, fuel melting, and other dangerous behavior would always appear in a pin.

Two major types of reactivity insertions were studied. In the first, calculations were made beginning with a reactivity insertion rate of 0.50\$/s and increasing by 0.50\$/s intervals until the cladding failed. Cladding rupture conditions were taken from Figure A-10 of Appendix A, BAW-1350. For an assumed fuel melting point of 4500F, the ramp rate causing cladding failure was about 5\$/s. The melting point has since been re-evaluated and determined to be near 6000F. The 4500F temperature is conservative, however, since it leads to earlier failure at lower reactivity rates. Failure occurred 0.486 second after the transient began.

An additional set of reactivity insertion calculations was made in which specific reactivities were inserted in short time intervals. The calculations began with 0.25\$ intervals for each set of three insertion times. The trip points and cladding rupture conditions were not changed for these calculations. Cladding failure occurred when 1.25\$ was inserted in 0.1 second. The time from the beginning of reactivity insertion to the cladding rupture was 0.2 second.

Figure 3-12. Ratcheting Model

4. EVALUATION

The ultimate goal of the Accident Analysis and Safety System Design Study is the selection of a design basis accident and a conceptual design for protective systems and safety features. To this end, certain malfunctions resulting in reactivity insertions have been investigated. The results indicate that the conditions arising from definable malfunctions at full power will not lead to irreversible damage of the fuel, the core, or components unless at least partial failure of the protection system is postulated. The implications are that reactivity insertions are effectively removed from the category of initiating conditions likely to lead to a DBA.

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- ⁸ Soodak, H., "The Reactivity Effect of Hydrogen Addition in Fast Reactors," Proceedings of International Atomic Energy Agency Symposium on Physics of Fast and Intermediate Reactors, Vienna, August 1961.
- ⁹ Thompson and Beckerly, The Technology of Fast Reactor Safety, Vol I, p 639, MIT Press (1964).
- ¹⁰ Rose, K. M. and Jenkins, J., Ratcheting Caused by Temperature Changes in PFR Fuel Pins, UKAEA, TRG-Memo-5077, November 1969.

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October 20, 1970

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We have been asked by L. W. Fromm, Manager, 1000-MWe Studies LMFBR Program Office, ANL, to print and distribute subject reports as a logical continuation of our involvement in the Follow-on Study Program. A total of 12 reports have been generated by BAW. There may also be a single report from each of 3 contractors, AI, GE and Westinghouse. Each report will be cleared for publication before being sent to DTIE.

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February 19, 1971

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Mr. Robert L. Shannon, Director
Division of Technical Information Extension
U.S. Atomic Energy Commission
P.O. Box 62
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Subject: 1000-MWe LMFBR Safety Studies -
Publication of Babcock & Wilcox Topical Reports

Reference: Letter, L. W. Fromm to R. L. Shannon, "1000-MWe LMFBR
Safety Studies - Publication of Contractors' Phase and
Topical Reports," October 16, 1970

Dear Mr. Shannon:

In the reference letter I advised you that we would be transmitting to you for publication a total of twelve Babcock & Wilcox Company Phase and Topical Reports, and possibly three reports from other contractors, generated under the AEC-sponsored 1000-MWe LMFBR Safety Analysis Studies program. With that letter I enclosed one B&W report (BAW-1344, which you have since published), and advised that the remainder would be transmitted to you for publication when received and patent-cleared.

I am enclosing herewith one copy of each of the eight B&W Topical Reports listed below, all of which are now patent-cleared and ready for publication. The covers for these reports should be that used for the previously-issued BAW-1344, except for the changes noted in the table below. The letters heading the columns of the table are keyed to the markings on the attached xerox copy of the cover for BAW-1344.

Mr. Robert L. Shannon, Director
February 19, 1971

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BAW-1342	TOPICAL REPORT	Accident Analysis Methods
BAW-1349	TOPICAL REPORT	Candidate Secondary Containment Support Systems
BAW-1350	TOPICAL REPORT	Accident Initiating Conditions Part 1 - Flow Abnormalities
BAW-1351	TOPICAL REPORT	Candidate Emergency Decay Heat Removal Systems
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BAW-1355	TOPICAL REPORT	Effects of Irradiation-Induced Metal Swelling on the Reference Design
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All other parts of the front covers for these reports should remain the same as the cover for BAW-1344.

Binding edge captions for the reports should read:

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BAW-1360	1000-MWe LMFBR Safety Studies	B&W	Init. Cond. - 2. Reactivity	USAEC

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February 19, 1971

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The distribution of all of these reports should be our "Distribution A" plus Category UC-80, Reactor Technology, as before. For your convenience I am enclosing another copy of the "Distribution A" list previously supplied to you.

In the reference letter I stated that there would be twelve B&W reports, and possibly three from other contractors. This has now been revised downward to eleven B&W reports and one report from Atomics International. The single remaining B&W report and the AI report will be transmitted to you when received and patent-cleared.

Thank you again for your excellent cooperation in publishing these reports. If there are any questions, please contact me on FTS extension 312/739-2971 or 312/739-4844.

Very truly yours,

L. W. Fromm, Manager
1000-MWe Studies
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March 4, 1971

Mr. Philip Rosser
Division of Technical Information Extension
U.S. Atomic Energy Commission
P.O. Box 62
Oak Ridge, Tennessee 27830

Dear Mr. Rosser:

In response to your telephone call to Mr. Fromm's office I am transmitting herewith one copy of the Topical Report BAW-1360 entitled Accident Initiating Conditions Part 2 - Reactivity Insertions.

Thank you for calling this oversight to our attention.

Very truly yours,

Evelyn Lauer
Evelyn Lauer
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cc: L. W. Fromm
PF KX-015

*AMG inadvertently
omitted this one letter
in their transmittal*