Reactivity Accidents *
A Reassessment of the Design-Basis Events

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ABSTRACT

This paper summarizes a study of light water reactor event sequences which have been investigated for their potential to result in reactivity accidents with severe consequences. The study is an outgrowth of the concern which arose after the accident at Chernobyl and was recommended by the report of the U.S. Nuclear Regulatory Commission (NRC) on the implications of that accident (NUREG-1251). The work was done for the NRC to reconfirm or bring into question previous judgements on reactivity events which must be analyzed for licensing. Event sequences were defined and then a probabilistic assessment was completed to estimate the frequency of the reactivity event and/or a deterministic calculation was completed to estimate the consequences to the fuel. Using the results of this analysis, analysis done by others, and a set of screening criteria developed within this study, judgements were made for each sequence as to its importance, and recommendations were made as to whether the NRC ought to be considering the important sequences as part of the design basis or for further, more detailed, investigation.

Summary

The required safety analysis for light water reactors includes various reactivity events which must result in acceptable consequences in order for plants to be licensed. Although it is possible to hypothesize reactivity events that might lead to unacceptable consequences, in the past these unanalyzed events were either precluded by specific design features or required combinations of causative events which were judged too improbable to be of concern. The objective of this study was to analyze these latter events, and events newly identified in this study, to reconfirm or bring into question previous judgements on the adequacy of the events selected as the design-basis, and to determine if any events needed more analysis in the context of severe accident research. This was done by performing probabilistic and/or deterministic calculations and, where available, using the research of others.

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The incentive for the study was in part the accident at the Chernobyl plant in the Soviet Union. That accident called attention to the issue of potential events with relatively large reactivity insertions in a short enough period of time to cause rapid fuel damage. In a subsequent Nuclear Regulatory Commission (NRC) report [1] one of the recommendations was to make such a study. The complete study is to be published as an NRC report [2].

In order to focus the study, the pressurized water reactor (PWR) analysis concentrated on a Westinghouse plant of four-loop design and the boiling water reactor (BWR) analysis on a BWR/4 design. However, reference is made to other designs throughout the report, as the goal was to reach general conclusions that would be applicable to all PWRs or BWRs.

The primary results from the study are:

- ° Many sequences were studied in 17 broad categories. All but two of the sequences studied that have the potential to cause rapid fuel damage were estimated to have frequencies too low to warrant further consideration at this time.
- ° Of the two sequences that have estimated frequencies in the range of interest for severe accidents, the most important is a refueling accident in a BWR. This is caused by the loading of fuel surrounding two or more positions where control blades have been removed and are inoperative. Understanding of this event and related sequences should be improved so that corrective actions (such as changes in Technical Specifications), which may be deemed necessary, may be taken.
- ° The other sequence with an estimated frequency that is significant is the result of the flushing of boron during a BWR anticipated transient without scram (ATWS) because of an uncontrolled depressurization and injection of unborated water from low pressure cooling systems. This sequence has been studied by the NRC but no detailed deterministic calculations exist to fully understand core behavior. These calculations will be done in a follow-up study in order to provide a basis for a fully informed decision on the disposition of this sequence.
- ° Several sequences under shutdown conditions were found not to lead to rapid fuel damage but to lead to core melt; however they have low estimated frequencies of occurrence.

Methodology

The general approach was to first identify sequences of interest. This was done by studying reactor design, procedures in use at plants and plant experience. The experience of other analysts was also used in the search for what could go wrong. Plant experience was obtained both by searching data bases [3, 4, 5] as well as by using studies of accident precursors [6].

Once a sequence was identified, a judgement was made as to whether it was possible to estimate the consequences in terms of fuel damage using deterministic methods or the frequency of occurrence using probabilistic methods. If one type of analysis was sufficient to determine that the sequence was unimportant, then the other type of analysis was not done.

An important adjunct to reaching conclusions in this study was the defining of screening criteria to help judge whether a particular sequence may need further, more detailed study. BNL defined such a set of criteria in terms of expected damage to the fuel and the estimated frequency of the sequence.

The consequences of a sequence were either acceptable in terms of current NRC acceptance criteria or unacceptable leading to either a relatively slow core melt or rapid fuel damage. The latter implies the possibility of fuel fragmentation and the rapid changing of core geometry. In this study rapid fuel damage is judged to take place if the fuel enthalpy at any location in the core exceeds 280 cal/g during a time interval of the order of five seconds. Although this amount of energy deposition does have the potential to cause a serious accident, note that it is considerably less than what has been estimated for the Chernobyl reactor.

There were three frequency ranges considered. The highest range extends down to a frequency of $1.0\text{E-}6/\text{RY}$ (1.0×10^{-6} per reactor year), which is the value that the NRC is currently using for functional sequences that might lead to core damage which should be reported as part of the Individual Plant Examination Program. Below that was a range which might still be of interest if the event leads to rapid fuel damage. This was taken as one order of magnitude lower, extending down to $1.0\text{E-}7/\text{RY}$. Any event with frequency less than this is considered of sufficiently low probability not to warrant further investigation at this time.

PWR Events

PWR reactivity events were categorized according to the 10 types of events listed in Table 1. The table also summarizes the results of analysis, performed at BNL and elsewhere, of the expected consequences to the fuel for the worst sequence in the category and the estimated frequency of occurrence. A frequency is considered significant or insignificant according to whether it is above or below the criteria discussed above. The criterion used depends on whether the sequence can lead to a core melt or rapid fuel damage.

The scenarios in the first category involve adding diluted water from the accumulator to the core when the vessel has been depressurized below the pressure of the accumulators (600-650 psig) during normal operation. The potential for this occurs during the cooldown of the reactor from hot shutdown (Mode 4 operation) to cold shutdown (Mode 5 operation) and then persists if the reactor is put in the refueling mode (Mode 6) where it is at atmospheric pressure. A motor operated valve (MOV) is closed prior to depressurization to isolate the accumulator and prevent any flow from the accumulator into the vessel.

If it is assumed that the water in one of the accumulators has been diluted to a sufficiently low boron concentration (as defined in [2]), then there are two fundamentally different sequences that could result in a potentially significant reactivity event. Sequence 1A is defined as the abrupt opening of the MOV leading to a surge of diluted accumulator water into the vessel. The hydraulic conditions in this situation are difficult to predict. The flow rate will depend on the pressure differential which can be as much as 650 psi during Mode 6. This water will enter the cold leg and then reach the core after going through the downcomer and lower plenum. The water will mix with the water in the vessel which has been borated to maintain the shutdown margin. The degree of mixing and the flow rate into the core will determine whether a slug of diluted water can add reactivity rapidly enough to cause a serious power excursion.

Table 1

SUMMARY OF REACTIVITY ACCIDENTS - PWR EVENTS

<u>Event</u>	<u>Expected Consequences for Worst Sequence</u>	<u>Estimated Frequency</u>
1. Addition of diluted accumulator water during shutdown due to slow leakage or blowdown thru single valve	Core melt or rapid fuel damage but calculations needed to verify possibility of latter	$<1\text{E}-8/\text{RY}$; insignificant
2. Addition of diluted RWST water during shutdown due to inadvertent safety injection, leakage because of improper valve opening/closing, or filling of cavity	Same as above	$<1\text{E}-7/\text{RY}$; insignificant
3. LOCA with diluted ECCS water from more than one accumulator or RWST	Same as above	$<1\text{E}-7/\text{RY}$; insignificant
4. LOCA with sump water diluted	Core melt	Insignificant
5. LOCA/SGTR with secondary diluting primary	Core melt	Analysis by NRC to be published
6. Uncontrolled boron dilution from CVCS during shutdown	Core melt	$<1\text{E}-6/\text{RY}$; insignificant
7. Startup of RCP after improper boron dilution	Acceptable	No analysis
8. Rod ejection accident	Rapid fuel damage	Insignificant; requires (unknown) mechanism for multiple ejections
9. Transients with positive MTC	Acceptable	No analysis
10. Other beyond-design-basis events	No rapid fuel damage	No analysis

In Sequence 1B there is leakage past the MOV and accumulator water slowly makes its way into the vessel. This water can be at a much lower temperature than the reactor coolant system (RCS) water which, in Mode 5 or Mode 6, must be less than 200°F. Hence, if there is not much circulation in the vessel, stratification may occur and the diluted water may accumulate in the lower plenum. When sufficient water has entered the vessel the bottom of the core could become critical. The resulting power surge could be rapid or slow depending on the rate at which reactivity was added to the bottom of the core. However, the power rise will cause a surge in flow when the heat is transferred to the coolant and this in turn will draw more diluted water into the core, increasing the rate of dilution, and causing the entire core to return to power. Hence, the initial power surge could cause an autocatalytic reaction which may have the potential to be destructive.

An event tree depicting the inadvertent blowdown of an accumulator during refueling, i.e., Sequence 1A, is given in Figure 1. The frequency of the initial condition (0.75/RY), is that obtained for the Zion plant for which such data were readily available [7]. The next top event on the tree is the deenergizing of the accumulator MOVs after closing. When RCS pressure falls below 1000 psig during the shutdown sequence, the accumulator isolation valves are required to be closed to prevent the accumulators from discharging into the reactor pressure vessel (RPV). As RCS pressure is further reduced, the accumulator valves are required to be locked shut to prevent their accidental opening on a spurious signal. The human-error probability for failure to follow this procedure is estimated to be 0.01 [8].

An estimate of the probability that a MOV would be opened suddenly due to the failure of the valve controls was obtained by assuming that the mean rate for MOV's to "transfer open" is identical to that for "transfer closed" (1.2E-7/hr) [9]. The mean refueling outage duration time (1996 hr) was taken from the Zion plant [7]. Since there are four accumulators at a plant, the probability that any MOV opens is four times the probability for a single valve. The probability (9.7E-5) that the corresponding accumulator has been diluted to at least the critically low concentration is obtained from a fault tree analysis explained in [2]. The resulting probability for a potentially damaging reactivity event is 6.9E-10/RY.

An event tree depicting Sequence 1B, which involves the leaking of the accumulator valve, is given in Figure 2. In this sequence it is the slow buildup of diluted water in the lower plenum followed by criticality of the bottom of the core and then an increase in core flow rate which is of concern. This event tree is quantified using the same outage data as for Sequence 1A. The leakage frequency (1.0E-7/hr) was derived from Licensee Event Report (LER) data compiled for all plants [10]. When this frequency is multiplied by the refueling outage duration time (1996 hrs) and then by four, the result is the probability that any one of the four MOVs at the plant leak.

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BLOWDOWN OF ACCUMULATOR DURING REFUELING

Figure 1 Event tree for accumulator injection.

EVENT CAUSED BY LEAKING ACCUMULATOR VALVE

Figure 2 Event tree for leaking accumulator.

The top event "sufficient core flow" on the event tree refers to the fact that if there is sufficient circulation in the vessel, then the water which has leaked from the accumulator will not remain stratified in the lower plenum but rather will be mixed with the water which was originally in the vessel. If this is the case then any return to criticality would be very slow and would require an even larger accumulator dilution than is being considered herein. In order to estimate the probability that there is sufficient core flow to mix the accumulator water, it is assumed that the residual heat removal (RHR) system supplies this function. During the refueling outage the RHR system is normally realigned to the RHR mode for maintaining balanced charging and letdown. The probability of RHR failure during shutdown ($7.0E-2$) is based on experience [11]. The probability for the last top event on the event tree is again from the fault tree explained in [2]. Although this sequence takes place over a long period of time, no credit is given for operations people to discover the leak and take corrective action. The resulting probability for a potentially significant reactivity event using this tree is $4.1E-9/RY$.

Since the frequencies for these two sequences are less than the $1.0E-7/RY$ criterion, they are considered insignificant, independent of what the actual consequences might be. In order to be more certain as to consequences, detailed thermal-hydraulic calculations would be necessary. These calculations would not be easy due to the fact that multi-dimensional flow patterns, low flows, stratification, and boron transport are involved and these can each be difficult to model.

The scenarios in the second category involve adding diluted water from the refueling water storage tank (RWST) during one of the shutdown modes of operation. The discussion of these scenarios is broken into three types of sequences. In Sequence 2A there is inadvertent actuation of safety injection. If the reactor is depressurized, as it would be during shutdown conditions, then the safety injection flow rate can be appreciable. The source of this water is the RWST and if the boron concentration in the tank has been diluted to at least the critical value discussed in [2], then the situation is analogous to Sequence 1A where the accumulator water rapidly entered the reactor pressure vessel (RPV). As in Sequence 1A, the degree of mixing and the flow rate into the core will determine whether a slug of diluted water can add reactivity rapidly enough to cause a serious power excursion with rapid fuel damage.

The RWST water can also leak into the vessel under gravity flow during shutdown and this is referred to as Sequence 2B. This involves the incorrect opening/closing of valves which has occurred three times based on a survey of Licensee Event Reports (LER) and Nuclear Power Experience reports (NPE). None of these events led to a serious dilution but this sequence needs to be considered as was Sequence 1B, the inadvertent leakage of accumulator water. Since the water would be entering the vessel slowly relative to Sequence 2A, the concern is that the water could accumulate in the lower plenum and then when sufficient water had entered, the bottom of the core could become critical. If the resulting power rise causes a surge in flow then the rate of dilution would increase and the entire core would become part of the transient. The concern is

that the initial power rise is autocatalytic and, therefore, has the potential to become destructive. If the RWST water mixes with the water in the vessel then the scenario might lead to a slow return to power.

Sequence 2C occurs when the reactor cavity is being filled from the RWST in order for refueling to take place. This follows when the vessel head has been removed and the residual heat removal system has been aligned to take suction from the RWST and to discharge into the cavity through the hot legs. The cavity is filled at the rate of approximately 3 in/min, equivalent to 7500 gpm, for approximately an hour. If the water is again assumed to be diluted, then there is the possibility that the diluted water will reach the core. However, typically the water enters the hot leg and it is expected that the water will flow directly into the cavity. This is in contrast to the situation where the water flows into the cold leg, e.g., as during an inadvertent safety injection. In that case the water is channelled to the downcomer from which it can flow directly into the lower plenum and core. With the cavity and upper vessel initially empty before beginning the filling of the cavity, it is expected that the RWST water will flow from the hot leg directly into the empty region and the only concern might be a very slow dilution event which could be mitigated by the operator. Only if there are atypical plants which fill through the cold leg could the sequence become more severe.

The event tree shown in Figure 3 depicts Sequence 2A. The probabilities of the top events except for having RWST critically diluted, are from the probabilistic risk assessment done for the residual heat removal system (RHR) for the Zion plant [7]. The frequency of being in the cold startup condition after refueling or maintenance is 3.8/RY. The probability that safeguards are not in service ($5.0\text{E-}4$) and that there is an inadvertent safety injection ($3.5\text{E-}4$) are based on plant experience. The probability that the boron concentration of the RWST is critically low is taken from the fault tree discussed in [2]. The resulting probability for the potential for a significant reactivity event is $3.7\text{E-}8/\text{RY}$.

Sequence 2B, the inadvertent admission of RWST water into the vessel due to operator errors in opening/closing valves, can be analyzed in a fashion analogous to the way in which Sequence 1B, for accumulator water leaking into the vessel, was analyzed. The event tree is given in Figure 4. For this type of sequence three initiating conditions are considered: a refueling, a drained maintenance outage and an undrained maintenance outage. Again, the analysis done for the Zion plant is used for the frequency of each initial condition and the duration of each outage. In order to obtain the frequency of this type of event the LER and NPE data bases were searched for the historical experience. There were three events that led to RWST water draining into the vessel in the 487 reactor years of operation covered by the data bases. Taking into account that historically 0.27 of these years represent outages [12] the frequency of this event in any outage is $[3/(487)(.27)(8760)]$ $2.5\text{E-}6/\text{hr}$. For Zion the average duration times for refueling and for drained and undrained maintenance are 1996, 982, and 358 hours, respectively [7], and hence, the probability of having an inadvertent leakage of RWST water into the vessel is $5.0\text{E-}3$, $2.5\text{E-}3$, and $9.0\text{E-}4$ for each of these periods, respectively.

EVENT CAUSED BY INADVERTENT SAFETY INJECTION

Figure 3 Event tree for sequence with RWST diluted.

EVENT CAUSED BY IMPROPER VALVE ALIGNMENT

Figure 4 Event tree for "leaking" RWST.

The next top event on the event tree of Figure 4, "sufficient core flow," is the same as used on the event tree of Figure 2 and relates to having RHR operational. The last top event is again the probability that the RWST is diluted to a critically low concentration. By summing up the contributions from all three types of outages, the resulting frequency for a potential reactivity event is $1.5\text{E-}8/\text{RY}$.

The result of the analysis of both of these sequences places the frequencies of occurrence in the insignificant range. The possibility of rapid fuel damage is predicated on understanding the thermal-hydraulics of these situations and such calculations are currently not available. With regard to Sequence 2B, note that if the event proceeded with mixing of the RWST water, it would take a long time to achieve criticality, operator action would be likely and the possibility of a core melt would be even more remote.

The scenarios in the third category involve adding diluted water to the core during the early phase of a loss-of-coolant accident (LOCA) as a result of having the water in the emergency core cooling system (ECCS) diluted. If the LOCA is caused by a large break then several systems are involved. Consider as Sequence 3A the situation when the vessel has depressurized below the 625 psia pressure of the accumulators and the accumulators blow down and flood the core. If the boron concentration in the accumulators had been diluted, then the flow of this water into the vessel has the potential to cause a significant reactivity event.

Sequence 3B is the similar low pressure injection of water from the refueling water storage tank when the depressurization due to the break brings the pressure below the approximately 1620 psia setpoint for safety injection. Assuming a large break, this injection becomes significant after the accumulators have reflooded the core. Although some plants have boron injection tanks with a higher boron concentration than the RWST, this is not the case in all Westinghouse plants. If the RWST has been diluted to a critically low boron concentration (see [2]), then the flow of this water into the vessel may cause a significant reactivity event.

To estimate the probability of the accident sequences described above, the functional event tree shown in Figure 5 was constructed depicting combinations of failures of safety functions. Since LOCA as an initiating event is used normally in probabilistic risk assessments (PRAs), an existing event tree [13] for the Byron plant was used as the starting point for Figure 5. Sequences which could lead to a reactivity event were not considered in the original tree and hence modifications were made to include the sequences being considered herein. Top events ACCM (boron concentration in 2 out of 3 accumulators critically low and not detected or ignored) and RWSTLOW (boron concentration in RWST water critically low and not detected) were added. The top events SUMP (boron concentration in sump water diluted) and TR1 (multiple steam generator tube ruptures) are also relevant to reactivity events that may occur in a LOCA, but will be discussed with categories 4 and 5 of Table 1.

Initiating Event	Availability of RWST as a source of ECCS injection	Multiple steam generator tube ruptures	Safeguards actuation signal	Operation of low pressure ECCS injection	Boron in 2/3 accumulators critically low and not detected or ignored	Boron conc. in RWST critically low and not detected	Containment spray system in injection phase	Operation of containment fan coolers	Operation of low pressure ECCS in long term recirc. mode	Boron conc. in sump water diluted
ETI	TK	TRI	SAI	LP	ACCM	RWSTLOW	CS	CF	R-1	SUMP

a: Insignificant
 b: Potential Reactivity Event; 2.7E-8/RY
 c: Potential Reactivity Event; 5.4E-8/RY
 d: Potential Reactivity Event; 9.4E-9/RY

Figure 5 Event tree for LOCA sequences.

The event tree shown in Figure 5 has end states that lead to core damage due to insufficient cooling. These are marked by initials which represent the core damage state in the original PRA. There are also end states where there is the potential for a reactivity event. Based solely on this tree, the probability that there is a reactivity event due to the accumulators being diluted is $5.4\text{E-}8/\text{RY}$ and due to the RWST being diluted is $2.7\text{E-}8/\text{RY}$. Both of these are considered to be insignificant.

These estimates are based on the initiating event frequency and the probabilities for dilution. The frequency of a LOCA event is estimated to be $9.4\text{E-}4/\text{RY}$ for Byron [13]. Note that this value varies from plant to plant ranging over at least two orders of magnitude. The probability of a large LOCA (used in the PRA) for Indian Point 2 is $2.2\text{E-}3/\text{RY}$ [14], for Oconee, $9.3\text{E-}4/\text{RY}$ [9], and for Sequoyah, $4.7\text{E-}5/\text{RY}$ (point estimate) [15].

The probabilities for dilution of the RWST and accumulators are explained in [2]. Note that in the LOCA sequences, for the necessary core dilution to occur due to the accumulators, at least two of the four must have been diluted sufficiently.

Since human errors play an important role in the accident sequences in the three categories discussed above, and because the human error probabilities (HEPs) usually have a large uncertainty, the sensitivity to the HEPs of each of the calculated frequencies for a reactivity accident was assessed. It was found that when the HEPs were individually increased by a factor of 10, the frequency of a potentially significant event was increased above $1.0\text{E-}7/\text{RY}$ for the case with inadvertent safety injection during shutdown and for a LOCA where accumulator water was the source of dilution.

The fourth category in Table 1 is the dilution of the core during the long-term cooling phase of a LOCA because the sump water became diluted. This event was studied probabilistically by one utility and reactor vendor taking into account several sources of unborated water [16]. Their study was reviewed and a judgement was made that the event falls into the lowest frequency range. Moreover, the most severe consequences of this event are expected to be core damage due to melting rather than rapid fuel damage. This sequence is discussed more fully in [2].

The next event category is the dilution of the core during a LOCA as a result of steam generator tube ruptures and the ingress of unborated secondary water. This event is currently being probabilistically analyzed by the NRC and also is not expected to lead to rapid fuel damage.

The sixth category in Table 1 refers primarily to dilution of the core boron concentration due to the chemical and volume control system (CVCS). It also includes sequences where diluted demineralized water from the purification system enters the vessel inadvertently. Hence, this category considers the systems that normally are used with the reactor coolant system (RCS) as opposed to the potential sources of dilution considered above which are not normally active during operation. Of concern are the shutdown modes when all control rods are

inserted into the reactor and cannot be used to counter the effect of a recriticality due to the dilution. Shutdown is also of interest because many PWRs have no positive means of detecting the dilution in this mode of operation.

The significance of inadvertent dilution from the CVCS had been of concern to the NRC for some time and had been made into a generic issue. The resolution of this issue is explained in an NRC report [17]. The estimated frequency of criticality from an unplanned boron dilution was determined to be $2\text{E-}4/\text{RY}$. This number is the product of two factors. Using information from the events that did occur and extrapolating to the point where shutdown margin is reduced to zero results in a frequency of $2\text{E-}3/\text{RY}$. This is then multiplied by a probability of 0.1 for the failure of the operator to take action. The latter takes into account that neutron flux alarms will at some point come into play and that at this point the operator can still prevent criticality by stopping the dilution or reborating the RCS. The human error failure probability for this step (0.1) was based on judgement.

The consequences expected [17] once the core becomes critical are a heating up "with a positive period governed by the rate of dilution and by moderator temperature and Doppler feedback. Eventually the coolant may boil and the peak power level will be limited by void generation in the moderator. Preliminary calculations indicate that, assuming BOC [beginning-of-cycle] parameters (worst case), a power level of about 3% of rated would be reached [18]." Hence, core damage is not expected.

There are also sequences in which it is demineralized water from the purification system which enters the core via the residual heat removal system. These have been studied by the Belgians [19]. In that work the initial conditions are cold shutdown with the primary partially drained in order to enhance the effect of the dilution. They give probabilities for events that include very low flow rates and do the core analysis in order to determine consequences. Only for the case where all control rods are removed from the core do they calculate any fuel damage.

After reviewing the sequences cited above and (proprietary) work by Combustion Engineering, it was concluded [2] that the worst consequences expected are a core melt rather than rapid fuel damage. The frequency of occurrence is not expected to be significant.

The seventh category includes those sequences in which there is the restart of reactor coolant pumps after a period when a dilution has been carried out. The consequences of this due to misoperation of loop stop valves have been shown by Westinghouse in each plant Safety Analysis Report to be acceptable. Swedish investigators have found no rapid fuel damage in sequences they investigated [20] and French work on similar sequences is expected to be reported on in 1989 [21]. Although all PWR sequences are affected by design, the sequences in this category (and in the first category) may especially be sensitive to the number of loops in the plant. Note too that although the Swedish analysis did not find rapid fuel damage, they did recommend changes to operating procedures. This category is discussed more fully in [2].

The next category is the rod ejection accident which includes extensions of the design-basis accident. The extensions which must be considered are sequences in which more than one rod is ejected and sequences in which the ejected rod has an excessive reactivity worth. There are two ways in which more than one rod could be ejected which are discussed in [2]. One way is consequential damage to a second housing, the other way is a common mode failure.

To obtain a control rod with a reactivity worth beyond the design-basis during power operation would necessitate the violation of Technical Specifications which define rod insertion limits. Control rod positions are displayed in the control room and monitored by the operator. There are alarms which sound when the rod insertion limits are reached and in some plants there are two alarms indicating both an approach to the limits as well as reaching the limits. If the limits are exceeded by a large amount it is likely that there would be a reactor trip due to a flux imbalance or some other indication of the anomaly. Even if a bank were completely inserted the maximum worth rod would still not be excessive as it is not expected to be larger than the worth of a rod at zero power. At zero power all control banks may be in the core and the worth of an ejected rod is at its maximum for normal operation. This maximum is less than 1.0% for all PWR designs and results in acceptable enthalpy increases.

In order to obtain a rod worth that is larger at zero power one has to consider the situation during the startup physics tests and question whether there is a rod pattern which would result in a higher worth. The only possibility that has been considered occurs for Combustion Engineering (C-E) plants in which there are double-yoked control element assemblies. These assemblies have a worth that typically is 1.9%. Although they are not in the core during normal operation, there may be times during the physics testing when one of the shutdown banks containing such a double assembly is inserted. This pattern, or any other potential high-worth pattern, would only be possible perhaps two days out of 400 when the testing is done.

Also discussed in [2] are analyses by BNL and U. S. PWR vendors that show that the rod worth needed to obtain rapid fuel damage is significantly greater than typical rod worths. It is anomalous to note that the French find [21] that it requires only a worth of 1.2% to obtain a stored energy of 300 cal/g.

The scenarios in the ninth category are initiated by a disturbance which causes the reactor coolant system temperature to increase. This occurs when, on the secondary side, there is a pressure increase or there is a reduction in fluid inventory; i.e., degradation of the ability of the secondary to remove heat. If the reactor has a positive moderator temperature reactivity coefficient (MTC), then an increase in temperature leads to an increase in reactivity and this could lead to a significant increase in power. Since this is an autocatalytic process analogous to the situation at Chernobyl, where it was a positive void reactivity coefficient that exacerbated the transient, it is obviously of concern.

The MTC becomes positive only when the boron concentration in the core is relatively high. This is the case only at beginning-of-cycle (BOC) and at low powers or shutdown. The effect of a positive MTC has always been a licensing

concern and Technical Specifications limit the time during the cycle and the operating mode when a positive coefficient is allowed.

The amount of reactivity that can be added by the positive MTC is limited because the coefficient becomes negative if the temperature becomes high enough. In order to determine the limits of reactivity addition both operating and shutdown conditions must be considered. This has been done [2] and the conclusion was that there is no expectation that a potentially disruptive accident could occur.

The last category are those sequences which represent extensions of events normally analyzed as part of a plant's safety analysis. This includes an uncontrolled rod/bank withdrawal, a steamline break, and a rapid increase in primary coolant temperature. No rapid fuel damage is expected due to the extensions considered [2].

BWR Events

BWR reactivity events were categorized according to the 7 types of events listed in Table 2. The table also summarizes the results of analysis, performed at BNL and elsewhere, in the same way as was done in Table 1.

Extensions of the design-basis control rod drop accident are in the first category. The sequence of events which could lead to excessive fuel enthalpy has been analyzed probabilistically by the NRC [22]. Based on the extremely low frequency of occurrence, it was concluded that this type of sequence did not need further study at this time.

The rod ejection accident in a BWR (Category 2) has always been precluded from consideration because of the design of the control blade housing support structure. An analysis of the consequences of this event [2] indicates that it would be no more severe than a design-basis rod drop accident. Hence, this event is not expected to have either a significant frequency or unacceptable fuel damage.

The scenarios in the third category are initiated by an overpressurization event caused by the closure of a steamline valve. Either the main steam isolation valves (MSIV) or the turbine stop or control valves could close initiating a pressure pulse that travels up the steamline into the vessel. The overpressurization causes the collapse of steam voids and, therefore, the addition of reactivity. The resulting power surge is normally mitigated by thermal-hydraulic feedback and the reactor protection system which causes both reactor trip and trip of the recirculation pumps (RPT). The latter reduces flow in the core which increases voids. The opening of safety and relief valves (SRVs) is also important to control these events. The standard safety analysis submitted for licensing must demonstrate that the plant response is acceptable.

Table 2

SUMMARY OF REACTIVITY ACCIDENTS - BWR EVENTS

<u>Event</u>	<u>Expected Consequences for Worst Sequence</u>	<u>Estimated Frequency</u>
1. Rod drop accident	Rapid fuel damage	$<1E-8/RY$; insignificant
2. Rod ejection accident	Acceptable	Insignificant
3. Overpressurization events	Rapid fuel damage not expected	Insignificant; (only short-term behavior con- sidered)
4. Flushing of boron during an ATWS	Rapid fuel damage ^a	$>1E-6/RY$; significant
5. Reactivity event initiated while in unstable operation	Acceptable (with respect to reactivity event)	Significant
6. Refueling accidents	Rapid fuel damage ^b	$>1E-6/RY$; significant
7. Other beyond-design- basis events	Core melt	Insignificant

^a BNL will do a follow-on study to verify whether rapid fuel damage is possible.

^b GE calculations show fuel enthalpy increases to more than 280 cal/g which may lead to rapid disintegration of fuel.
In general the results show that the initial power surge due to the

These events can become more serious if additional failures are postulated. Failure of recirculation pump or reactor trip will exacerbate the power surge. If some, or all, of the SRVs do not open then a power surge can theoretically become autocatalytic as an increase in power causes a continuing increase in pressure which in turn continues to collapse voids and add reactivity.

In order to determine the consequences of these types of sequences, calculations were done with the RELAP5/MOD2 code [23] with a model of the Browns Ferry plant based on previous studies [24]. The reactor was assumed to be initially operating at nominal power.

In general the results show that the initial power surge due to the overpressurization is not limiting, but if there is failure of RPT and reactor trip, the power following this surge can remain high and oscillate. This is seen in Figure 6 which shows the power when the initiating event is a turbine trip and there is failure of the bypass, RPT, and reactor trip. The four curves represent cases with 0, 1, 2, or 3 banks of SRVs assumed to fail to open. The power remains at approximately 50-150% above rated power during the 10 seconds after the initial peak.

The traces of system pressure corresponding to the four cases shown in Figure 6 are given in Figure 7. Note that the pressure increase is calculated to be excessive with any SRV bank failures. Depending on whether it is 1, 2, or 3 banks failing to open, the pressure exceeds 100 bar (1470 psia) in approximately 8, 5, or 4 seconds, respectively. As the pressure increases beyond 100 bar, it is problematic as to whether there will be pressure relief due to a break in the primary system pressure boundary.

The most important result of these calculations is the fuel enthalpy during the power surge since this is to be compared with the 280 cal/g criterion to determine if rapid fuel damage could occur. These results are shown in Figure 3 for the four cases being considered. The curves are the increase in enthalpy averaged over the core. The enthalpy, and the fuel temperature, increases due to the power surge and then decreases when the effect of heat transfer from the fuel to the coolant becomes important. The time constant for heat transfer is approximately 3-5 seconds and the peak enthalpy occurs at times between 9 and 14 seconds. A power peaking factor must be applied to the maximum average enthalpy of Figure 8 to obtain the maximum enthalpy increase within the core. Based on the various space-dependent calculations of overpressurization events that exist, the shape of the axial power distribution is expected to change dramatically but not the peaking factor. Hence, we use a peaking factor of two as a representative number and in the worst case, with the failure of all SRVs, the maximum fuel enthalpy rise would be approximately 140 cal/g. This provides sufficient margin to the criterion of 280 cal/g that we do not expect to see rapid fuel damage even if the peaking factor is considerably higher. Note too that this case represents an extremely large pressure rise which would likely lead to a pipe break and pressure relief. Since the calculations do not take this into account, the actual pressure with failure of all valves might be less than is calculated.

Figure 6 Turbine trip with multiple failures - core power.

Figure 7 Turbine trip with multiple failures - system pressure.

In order to see the effect of a different initiating event, calculations were done with MSIV closure. Graphs of average fuel enthalpy are given in Figure 9, for a case with two SRV banks failing to open, with no reactor trip and with and without RPT. Focusing on the case without RPT, it can be seen that the effect of having MSIV closure rather than a turbine trip is not to change the fuel enthalpy significantly.

The effect of RPT is seen to be important in determining power and fuel enthalpy. Indeed, with RPT the enthalpy rise even with two banks failing to open is trivial. Note too that these calculations are affected by a decrease in feedwater which occurs after MSIV closure (and as a result of an increase in pressure).

In order to have the remote possibility of core damage during an overpressurization event, failure of reactor trip must be assumed. ATWS events have been studied extensively elsewhere and some of those efforts can be applied to the present problem. The estimated frequency of an ATWS event for the Shoreham plant is $1.7\text{E-}4/\text{RY}$ [25]. The conditional probability of recirculation pump trip failure is $1.0\text{E-}4$ which would make the frequency of the combined sequence $1.7\text{E-}8/\text{RY}$. For the Shoreham plant the probability of failure to open on demand for all (11 in this plant) SRVs during an ATWS is $1.0\text{E-}4$ [25]. The estimated frequency for an overpressurization event with failure of reactor trip and all SRVs is then $1.7\text{E-}8/\text{RY}$ and if we include failure of RPT then the frequency of occurrence is $1.7\text{E-}12/\text{RY}$.

On the basis of both deterministic and probabilistic analysis it was concluded that beyond design-basis overpressurization events do not need further study at this time.

In the tenth category are ATWS sequences in which the problem is during the latter phase of the event after borated water has been added to the vessel through the standby liquid control system. At this point in time there is sufficient boron in the core to shut down the reactor. If it is assumed that at this time reactor pressure is abruptly lowered beyond the setpoint for low pressure injection, then large amounts of unborated water could enter into the vessel if the operator does not control the flow. The concern is that there might be a flushing of the boron in the core which could occur so rapidly that the resulting power excursion leads to catastrophic fuel damage. This reactivity addition would be aggravated by the fact that the injectant is at a much lower temperature than the water in the core and adds reactivity due to a negative moderator temperature coefficient (analyzed in [26]).

In some plants the operator is required to inhibit the Automatic Depressurization System (ADS) during the latter phase of an ATWS. The actuation of the ADS may lead to severe consequences although a deliberate depressurization by an operator (e.g., to comply with the pressure suppression pool heat capacity temperature limits) may be acceptable. Failure to inhibit ADS actuation is a top event in ATWS event trees done for the probabilistic risk assessment (PRA) of these plants. For the Shoreham PRA [25], if there is failure, then this leads to core damage, although in that study the core damage may be from a relatively slow core melt rather than from a rapid reactivity excursion.

Figure 8 Turbine trip with multiple failures
core average fuel enthalpy.

Figure 9 MSIV closure with multiple failures
core average fuel enthalpy.

The importance of this contribution to ATWS core damage frequency (CDF) can be determined from the ATWS event trees that are usually constructed for different initiating events. The result [2] is that the total CDF with this failure is $4.1\text{E-}6/\text{RY}$ or 23% of the total CDF due to ATWS. As a result of this relatively high frequency, deterministic calculations will be done in a follow-on study to assess whether rapid fuel damage is really possible.

The sequences in the fifth category originate when the reactor is brought into a region of instability such as is possible when a plant is operated with a relatively high power to flow ratio. In this condition the reactor may be undergoing global and/or spatial oscillations. One of several safety concerns is that if a reactivity event takes place during this time, it might lead to an unacceptable power excursion. The reactivity event might be any of those that are normally considered in safety analyses. Ordinarily these events lead to acceptable results, however, a priori, it is not clear if, in this unstable situation, the event could lead to unacceptable fuel damage.

In order to assess consequences, calculations were carried out using the BNL Plant Analyzer (BPA) [27]. The initial conditions were based on operation of a BWR/4, similar to the LaSalle plant, initially at 85% of rated power and 75% of rated flow. When this model was subjected to a recirculation pump trip, the flow rate and power decreased. When this was combined with the modeling of a feedwater heater failure, the power to flow ratio increased and the power began to oscillate. During this oscillatory phase reactivity events were initiated in order to determine the resulting power excursion.

The plant model for these calculations had been set up for an extensive study of BWR instabilities [28] which included calculations related to the present study. The calculations done to date that are germane encompass the following reactivity events:

- Restoration of recirculation pump flow
- Restoration of flow and MSIV closure
- Turbine trip without scram
- Turbine trip without bypass and without scram
- Turbine trip with feedwater pump trip without scram
- MSIV closure without scram.

There are several conclusions that can be drawn from these calculations. What is pertinent in the present study is that the power excursion due to the reactivity event (turbine trip etc.) is not excessive relative to the power pulse which is expected had the event taken place at nominal operating conditions. This is based on varying the type of reactivity events and the timing for one of them; namely, for MSIV closure with no scram. The latter parametric [2] study was to determine if the time of the reactivity addition relative to the time during the oscillating cycle had any effect. Although it did change the magnitude of the power peak and the overall plant response, the effect was not significant with regard to fuel damage.

Sequences that occur during refueling (Category 6) are of particular concern because the vessel head is removed and containment may not be sealed. This means that in case of an accident there may be a direct path to the environment. These sequences are also unique because control rods are usually inserted or taken out of service, thereby negating the possibility of inserting negative reactivity. An exhaustive study of potential sequences has not been done as part of the present work. However, it is known that there is at least one sequence that is important by virtue of work which has been done by the reactor vendor, General Electric (GE) [29, 30, 31].

The sequence of interest requires that two or more control blades have been removed from the core and made inoperable. The surrounding fuel bundles would also have been removed. As part of the refueling, fuel is then erroneously loaded into the locations adjacent to the control blade position. These bundles would be uncontrolled and when the last bundle is placed in the core the reactivity insertion would be sufficient to cause a power excursion with the potential for rapid fuel damage.

The potential for this event and the concern of GE has evolved over the years. The design of a BWR includes refueling interlocks which prohibit fuel loading with any control rod withdrawn. This is to prevent the possibility of criticality during refueling. However, some utilities wanted more flexibility during shutdown and Technical Specifications were changed to allow bypass of the refueling interlock signals to facilitate multiple control rod drive maintenance. The bypass of the refueling interlock signals was acceptable as long as several criteria were met, including the removal of the fuel assemblies surrounding the control rod to be withdrawn.

The probability of occurrence calculated by GE is somewhat greater than $1.0E-6/R\bar{Y}$. The deterministic analysis done by GE takes into account that the reactivity worth of the last assembly is approximately 4% and this would be inserted at the maximum grapple speed of 40 fpm (18 seconds for full insertion). The maximum fuel enthalpy under these conditions would be greater than 280 cal/g and significant localized fuel damage would be expected.

For BWR's with Mark I or Mark II containment designs, both the vessel and drywell heads are removed during refueling; therefore, the only barrier to potential release of fission products from this excursion is the secondary containment. Because of this containment, and a standby gas treatment system, the offsite radiological releases for this event are calculated not to exceed the limits set by 10CFR100. However, the effect on plant personnel, particularly those on the refueling floor, would be severe.

The sequence of events described for a reactivity accident during refueling has the potential to lead to rapid fuel damage with the release of significant radioactivity and has an estimated frequency that is in the range of interest for severe accidents. It is, therefore, the most significant of the sequences that have been considered in this study and requires further investigation.

Any specific corrective actions which may be ordered by the NRC would depend on the results of this future analysis and analysis currently underway at GE. These actions could be along the lines of the current GE recommendations [31] which would reduce the probability of occurrence of this type of event to less than $1.0E-8/R\bar{Y}$ [29].

The last category consists of sequences which are the result of multiple failures after an initiator which adds reactivity to the core. The initiator could be a rod withdrawal or a thermal-hydraulic event. All of the events in this category are relatively slow and are not expected to lead to rapid fuel damage. The expected frequency of occurrence is expected to be insignificant.

References

1. "Implications of the Accident at Chernobyl for Safety Regulation of Commercial Nuclear Power Plants in the United States," NUREG-1251, U.S. Nuclear Regulatory Commission, April 1989.
2. D.J. Diamond, C.J. Hsu, and R. Fitzpatrick, "Reactivity Accidents--A Reassessment of the Design-Basis Events," NUREG/CR-5368, Brookhaven National Laboratory, to be published.
3. Public Document Room Bibliographic Retrieval System (U.S. Nuclear Regulatory Commission).
4. Sequence Coding and Search System for Licensee Event Reports (Nuclear Safety Information Center).
5. Nuclear Power Experience Automated Retrieval System (S. M. Stoller Corp.).
6.
 - a) J.W. Minarick and C.A. Kukiela, "Precursors to Potential Severe Core Damage Accidents: 1969-1979 A Status Report," NUREG/CR-2497, Science Applications, Inc. and Oak Ridge National Laboratory, June 1982.
 - b) W.B. Cottrell et al., "Precursors to Potential Severe Core Damage Accidents: 1980-81, A Status Report," NUREG/CR-3591, Science Applications, Inc. and Oak Ridge National Laboratory, June 1984.
 - c) J.W. Minarick et al., "Precursors to Potential Severe Core Damage Accidents: 1984, A Status Report," NUREG/CR-4674, Vols. 3 & 4, Science Applications, Inc. and Oak Ridge National Laboratory, April 1987.
 - d) J.W. Minarick et al., "Precursors to Potential Severe Core Damage Accidents: 1985, A Status Report," NUREG/CR-4674, Vols. 1 & 2, Science Applications, Inc. and Oak Ridge National Laboratory, November 1986.
 - e) J.W. Minarick et al., "Precursors to Potential Severe Core Damage Accidents: 1986, A Status Report," NUREG/CR-4674, Vols. 5 & 6, Science Applications, Inc. and Oak Ridge National Laboratory, February 1988.
7. D.C. Bley et al., "Zion Nuclear Plant Residual Heat Removal PRA," NSAC-84, Electric Power Research Institute, July 1985.
8. A.D. Swain and H.E. Guttmann, "Handbook of Human Reliability Analysis with Emphasis on Nuclear Power Plant Applications," NUREG/CR-1278, Sandia National Laboratories, 1983.
9. "Oconee PRA: A Probabilistic Risk Assessment of Oconee Unit 3," NSAC-60, Vol. 1, Electric Power Research Institute, June 1984.
10. C.F. Miller et al., "Data Summaries of Licensee Event Reports of Valves at U.S. Commercial Nuclear Power Plants," NUREG/CR-1363, Idaho National Engineering Laboratory, October 1982.

11. T.L. Chu et al., "Improved Reliability of Residual Heat Removal Capability in PWRs as Related to Resolution of Generic Issue 99," NUREG/CR-5015, Brookhaven National Laboratory, May 1988.
12. "Licensed Operating Reactors Status Summary Report," NUREG-0020, U.S. Nuclear Regulatory Commission, through September 1988.
13. J.C. Butler et al., "Byron Generating Station Limiting Conditions for Operation Relaxation Program," WCAP-10526, Vols. 1 & 2, Westinghouse Electric Corporation, April 1984.
14. "Indian Point Probabilistic Safety Study," Power Authority of the State of New York, Consolidated Edison Company of New York, December 1983.
15. D.D. Carlson et al., "Reactor Safety Study Methodology Application Program: Sequoyah #1 PWR Plant," NUREG/CR-1659, Sandia National Laboratories, April 1981.
16. "The Potential Exists for Unacceptable Dilution of the ECCS During a Large Break LOCA Due to Postulated Non-Borated Water Addition to Containment Sump," Licensee Event Report 046, Sequoyah Unit 1, July 1987 and Revision 1, March 1988.
17. R. Emrit et al., "A Prioritization of Generic Safety Issues," NUREG-0933, U.S. Nuclear Regulatory Commission, June 1988.
18. N.S. DeMuth, "Analysis of Unmitigated Boron Dilution Events," letter to R.T. Curtis (NRC), Los Alamos National Laboratory, November 18, 1981.
19. M.I. Colard et al., "PWR Reactivity Accident in Cold Conditions," Proceedings of the International ENS/ANS Conference on Thermal Reactor Safety, Avignon, France, October 1988.
20. S. Jacobson, "Some Local Dilution Transients in a Pressurized Water Reactor," LIU-TEK-LIC-1989:11, Linkoping University Institute of Technology, Sweden, 1989.
21. L. Bregeon et al., "Considerations on Nuclear Safety in France Two Years After Chernobyl," Proceedings of the International ENS/ANS Conference on Thermal Reactor Safety, Avignon, France, October 1988.
22. H. Richings, "A Statistical Examination of the RDA in Some BWRs," U.S. Nuclear Regulatory Commission, June 1975.
23. V.H. Ransom and R.J. Wagner, "RELAP5/MOD2 Code Manual," EGG-SAAM-6377, Idaho National Engineering Laboratory, April 1984.
24. R. Jack Dallman et al., "Severe Accident Sequence Analysis Program - Anticipated Transient Without Scram Simulations for Browns Ferry Nuclear Plant Unit 1," NUREG/CR-4165, Idaho National Engineering Laboratory, May 1987.
25. "Probabilistic Risk Assessment Shoreham Nuclear Power Station Long Island Lighting Company, Final Report," Science Applications, Inc., June 1983.

26. H.S. Cheng and D.J. Diamond, "Reactivity Transients During a Blowdown in a MSIV Closure ATWS," Trans. Amer. Nucl. Soc., 57, 147, November 1988.
27. W. Wulff et al., "The BWR Plant Analyzer," NUREG/CR-3943, Brookhaven National Laboratory, August 1984.
28. H.S. Cheng, "Simulations of the LaSalle 2 Incident," in "Safety Research Programs Sponsored by Office of Nuclear Regulatory Research," NUREG/CR-2331, Vol. 8, Nos. 1 and 2, January-June 1988, and No. 4, October-December 1988.
29. "Telecon - Condition Germane to Safety," Letter from C.A. Cameron, General Electric to U.S. Nuclear Regulatory Commission, June 25, 1982.
30. "Recommended Technical Specifications for Fuel Loading," SIL No. 372, General Electric, June 1982.
31. "Germane to Safety - Core Monitoring During Fuel Loading," Letter from G.B. Stramback, General Electric to U.S. Nuclear Regulatory Commission, March 2, 1989.