6 IMPACT ON DEFENSE-IN-DEPTH AND SAFETY MARGINS

Section 5 discussed the impact of the changes on risk. According to the guidance in Regulatory Guide 1.174, the traditional engineering considerations also need to be addressed. These include defense-in-depth and safety margins. The fundamental safety principles on which the plant design is based cannot be compromised. Design basis accidents are used to develop the plant design. These are a combination of postulated challenges and failure events that are used in the plant design to demonstrate safe plant response. Defense-in-depth, the single failure criterion, and adequate safety margins may be impacted by the proposed change, and consideration needs to be given to these elements.

6.1 IMPACT ON DEFENSE-IN-DEPTH

Events that can occur in reactors can be mitigated by a number of safety systems that provide various levels of defense. Changes in the level of protection afforded by one level of defense, say due to equipment failure, can be compensated for by others. There are three basic levels of defense that ensure the reactor will be protected against RCS overpressurization and possible failure of the RCS pressure boundary with subsequent core damage from ATWS events. These include:

- Prevention: reactor trip with backup operator actions
- Control and Mitigation: the core physics defense barrier (reactor core and moderator feedbacks)
- Control and Mitigation: operation of existing systems to limit the potential pressure/temperature transient and provide reactor coolant inventory addition if necessary

Prevention: Reactor trip with backup operator actions

The first level of protection is provided by the RPS and backup operator actions. The RPS is an automatic system that will shut down the reactor if the RCS or core parameters exceed specified setpoints. The RPS consists of two redundant trains with each train consisting of logic cabinets and reactor trip breakers. The reactor trip breakers can be actuated automatically by two diverse mechanisms: the undervoltage trip and the shunt trip. Analog channels arranged in 2 of 3 or 2 of 4 combinational logic supply signals to each logic cabinet. The channels monitor plant operating parameters and provide signals to both logic cabinets that provide signals to open their respective reactor trip breakers. Reactor trip occurs when the trip combinational logic is met. Signals to trip the plant will be generated from at least two sets of channels for every anticipated transient event that can occur. If the automatic signal fails, then operators can take several actions, which follow, to trip the plant.

- Manually trip the reactor via the trip switch in the control room.
- Manually trip the reactor via interrupting power to the CRDMs from the MG sets (from the control room in many plants; locally at the MG sets near the control room in some plants).
- Manually drive in the control rods via the rod control system.

The first operator action listed provides a signal to open the reactor trip breakers, therefore, it is effective if the automatic trip failed due to failures in the logic cabinets or analog channels. If reactor trip failed due to reactor trip breaker failure or failure of a sufficient number of control rods to drop into the core, this action is ineffective. The second operator action listed interrupts the power to the CRDMs, therefore, it bypasses the RPS completely. This action is effective if the automatic trip failed due to failures in the logic cabinets, analog channels, or reactor trip breakers. If the reactor trip failed due to an insufficient number of control rods dropping into the core, then this operator action is also ineffective. (Note that, as discussed in Section 5.1.1.6, a very large number of control rods must fail to drop into the core in order to present an RCS integrity challenge via overpressure.) The third operator action listed requires the operator to drive the rods into the core by the rod control system. This action is taken if the rod control system is not in the automatic mode of operation. This action is effective if the automatic trip failed due to failures in the logic cabinets, analog channels, or reactor trip breakers. If the reactor trip failed due to an insufficient number of control rods dropping into the core, then this operator action may also be ineffective.

Table 6-1 provides a summary of the operator actions that are available to backup the various failures of the RPS.

One aspect of prevention is the industry trend, since the time that studies such as WCAP-11992 were performed in the late 1980s, to reduce annual plant trip challenges. As plants have matured and efforts to improve plant reliability have been implemented, the number of reactor trips has trended downward from roughly 4-8 per reactor-year to closer to 1 per reactor-year.

Control and Mitigation: Core physics defense barrier (reactor feedbacks)

An additional barrier in defense-in-depth is related to the design of the core with respect to the moderator reactivity feedback. The core is designed to provide negative moderator reactivity feedback to limit the reactor power and the RCS pressure transient if the RCS begins to heat up excessively. This is important for anticipated events, such as, loss of feedwater events that, without a rapid reactor trip, cause the reactor coolant system and core to increase in temperature. The negative reactivity reduces the reactor power and provides the operator time to borate the RCS to bring the reactor to shutdown conditions. Core designs with sufficiently negative reactivity feedback provide a "natural" barrier which limits events that could lead to core damage.

Control and Mitigation: Limit potential pressure transient

In addition to core reactivity feedbacks, in the defense-in-depth scheme, mitigation of the pressure transient by the RCS pressure relief system is also possible. This consists of pressurizer safety valves and PORVs. For a given core, the pressure transient that will need to be accommodated will depend on the time in cycle, the AFW flow rate, and the amount of negative reactivity insertion provided by the control rods. In many ATWS scenarios, partial control rod insertion will occur. In addition, as explained in the preceding paragraph, the operator can take action to manually drive the control rods into the core or the rod control system may be in the automatic mode, which would then automatically move the control rods into the core. Following successful mitigation of the pressure transient, the operator would have a substantial amount of time to borate the RCS to bring the reactor to shutdown conditions.

The AFW system will be started by either AMSAC or the ESFAS signals. AMSAC is a backup to the ESFAS. Signals from the ESFAS will be available to start AFW and trip the turbine under some, but not all, ATWS scenarios. Table 6-2 provides a summary of signals available to actuate the AFW and trip the turbine for the various failures of the RPS.

For ATWS events with peak pressures that do not exceed the safety valve setpoints, the event can be mitigated by emergency boration. Actuation of emergency boration requires an operation action.

Discussion

These barriers work together to provide a total level of plant protection and do not always offer three completely independent safety mechanisms. A partial degradation of one can be compensated for by another. For example, in many ATWS scenarios, partial insertion of the control rods is expected. This will reduce the severity of the pressure transient. For higher reactivity cores, the MTC may not be sufficient early in life to limit the pressure transient to below the pressurizer safety valve setpoints and pressure relief via these valves would be expected. Towards the end of life, pressure relief may not be required since negative reactivity feedback would be sufficient to limit the pressure transient.

If reactor trip fails, that is, a sufficient number of control rods do not drop into the core to shut it down, the pressure relief required to mitigate the potential pressure transient in the RCS will depend on a number of variables. These include core reactivity, time in core life, amount of negative reactivity provided by the controls rods that did drop, and AFW flow. It should also be noted that core design studies show that a large number of the control rod assemblies must fail to insert (i.e., a highly unlikely event) for a severe pressure transient to occur.

For higher reactivity cores, the MTC will be less negative (but always negative) at full power than for lower reactivity cores. The higher reactivity cores will result in higher pressure transients for similar conditions, time in life and AFW flow than low reactivity cores. But actions can be implemented during normal operation with higher reactivity core designs to counter this increased reactivity so that any higher pressure transients can be successfully mitigated.

Tables 4-3, 4-4, 4-7, and 4-8 show the UETs for the low reactivity and high reactivity core designs for 100% power and equilibrium xenon. As previously noted, a comparison of the UET values indicates the following:

- The higher reactivity core has longer UETs.
- Both cores can be operated with 0 UETs, but the lower reactivity core provides more flexibility to achieve this.
- To operate in a plant configuration with a low UET with the high reactivity core, it is important to maintain PORV availability, AFW availability, and control rod insertion from the lead bank (through either manual or automatic control rod insertion).

Tables 6-3 and 6-4 show the probabilities or split fractions for being in certain plant configurations dependent on the state of the rod control system, and PORV and AFW availability. Table 6-3 assumes that the rod control system is in manual, PORVs may be blocked, and AFW may be unavailable due to test or maintenance activities. Table 6-4 assumes that the rod control system is in automatic, a reduced probability that the PORVs are blocked, and the AFW system is available (although it may fail due to random or common cause component failures). A comparison of the information in these tables indicates it is possible to compensate for the degradation of one barrier with another. For example, plant configuration management scheme 2 (Table 6-4) ensures that the plant is operating in a configuration that can compensate for the degradation of the "natural" barrier. The probability of being in a 0 UET configuration is much higher in this scheme than in plant configuration management scheme 1 (Table 6-3).

In addition, and not illustrated in this example, it is also possible to restrict removal of RPS components from service for preventive type activities during unfavorable portions of the cycle. Extending test times to increase the availability of the RPS is also possible, but would require Technical Specification changes. These restrictions will increase the availability of the RPS during the portion of the cycle when the natural reactivity feedback mechanisms are less effective.

Based on the above discussion, it is seen that sufficient defense-in-depth barriers exist such that it is possible to compensate for limited degradation of one barrier with another and, therefore, maintain plant safety afforded by defense-in-depth requirements. This is an effective approach for managing the risk associated with ATWS events when implementing higher reactivity cores or other plant changes.

Elements of Defense-in-Depth

Regulatory Guide 1.174 defines the elements that comprise defense-in-depth that proposed changes need to meet. These elements and the impact of the proposed change on each follow:

- A reasonable balance among prevention of core damage, prevention of containment failure, and consequence mitigation is preserved.

  The proposed change in core design has only a small calculated impact on CDF and LERF as discussed in Section 5. The proposed change impacts both CDF and LERF via higher RCS pressures if an ATWS event occurs. The LERF is impacted primarily from ATWS induced SG tube failures. The change in core design does not degrade core damage prevention and compensate with improved containment integrity nor does it degrade containment integrity and compensate with improved core damage prevention. The balance between prevention of core damage and prevention of containment failure is maintained. Consequence mitigation remains unaffected by the proposed change. Furthermore, no new accidents or transients are introduced with the requested change and the likelihood of an accident or transient is not impacted. The impacts on CDF and LERF are very small as demonstrated in Section 5.

- Over-reliance on programmatic activities to compensate for weaknesses in plant design.

  The core design will change such that higher RCS pressures will occur if an ATWS event occurs. The magnitude of the RCS pressure will depend on the time in life when it occurs and the availability of pressure relief, AFW, and negative reactivity insertion. All safety systems, including the RPS, AFW system, RCS pressure relief capability, and rod control system will continue to function in the same manner with the same reliability, and there will be no additional reliance on additional systems or operator actions. The impact on risk is very small, but depending on the plant configuration, there could be an impact on defense-in-depth. This will be compensated for by plant configuration management programs that improve the preventive aspect or alternate mitigative capabilities as discussed in Section 7.

- System redundancy, independence, and diversity are maintained commensurate with the expected frequency and consequences of challenges to the system.

  No individual system redundancy, independence, or diversity will be impacted by the use of high reactivity cores.

- Defenses against potential common cause failures are maintained and the potential for introduction of new common cause failure mechanisms is assessed.

  Defenses against common cause failures are maintained. The change requested does not impact or introduce any new common cause failure mechanisms. The probability of control rods failing to drop into the core will not be impacted by this change. This change does not impact ATWS preventive or mitigative systems, such as the RPS, AFW system, RCS pressure relief, or the rod control system.

- Independence of barriers is not degraded.

  The barriers protecting the public and the independence of these barriers are maintained. As previously indicated, there will be a small impact on the natural barrier, but it will remain independent of preventive barrier and the RCS pressure mitigation system (PORVs and safety valves). In addition, this change does not provide a mechanism that degrades the independence of the fuel cladding, RCS, and containment barriers.

- Defenses against human errors are maintained.

  No new operator actions related to the change are required to maintain plant safety. No additional operating, maintenance, or test procedures will be introduced or modified due to these changes. During the unfavorable exposure time, a configuration risk management program will be used to control other activities that could impact prevention or mitigation of ATWS events to compensate for an impact on defense-in-depth. This is discussed in Section 7.

6.2 IMPACT ON SAFETY MARGINS

With regard to safety margins, an acceptable guideline to follow, per Regulatory Guide 1.174, for demonstrating compliance with safety margins is as follows. With sufficient safety margins:

- Codes and standards or their alternatives approved for use by the NRC are met.

- Safety analysis acceptance criteria in the licensing basis (FSAR, supporting analyses) are met, or proposed revisions provide sufficient margin to account for analysis and data uncertainty.

Consistent with these guidelines, implementation of the subject risk-informed approach to determine the impact of core design changes on plant safety will not eliminate the requirement to assess the impact of the change on the plant safety analysis licensing basis. All applicable acceptance criteria for the FSAR Chapter 15 design basis events will continue to be met with the implementation of this risk-informed approach. As such, the range of applicability of core design changes included in the risk-informed approach, including moderator temperature coefficient, are limited by the ability to meet applicable acceptance criteria of the FSAR Chapter 15 design basis events and by any existing plant specific Technical Specifications.

Table 6-1 Summary of the Capability of Operator Actions to Trip the Reactor for Various RPS Failures

Failed RPS Element	Backup Operator Action: OA for Reactor Trip from the Control Room	OA to Interrupt Power to MG Sets from the Control Room	OA to Drive in the Control Rods
Analog Channels	Yes	Yes	Yes
Logic Cabinets	Yes	Yes	Yes
Reactor Trip Breakers	No	Yes	Yes
Control Rods	No	No	No

Table 6-2 Summary of the Capability of Automatic Signals to Actuate Auxiliary Feedwater and Trip the Turbine for Various RPS Failures

Failed RPS Element	Actuation Signal: ESFAS	AMSAC	Comments
Analog Channels	No	Yes	ESFAS signal is not available. Reactor trip and ESFAS signals are assumed to be failed due to common cause failure.
Logic Cabinets	No	Yes	ESFAS signal is not available. Reactor trip and ESFAS signals are assumed to be failed due to common cause failure.
Reactor Trip Breakers	Yes (AFW) No (turbine trip)	Yes	ESFAS is still available to start AFW, but the turbine trip signal will not be available since it is developed when a RTB closes. No common cause failure exists between ESFAS and reactor trip signals for reactor trip breaker failures.
Control Rods	Yes	Yes	ESFAS is still available to start AFW and trip the turbine. No common cause failure exists between ESFAS signals and the control rods failing to drop.

Table 6-3 Plant Configuration Probabilities, Plant Configuration Management Scheme 1

Condition	0 PORVs Blocked	1 PORV Blocked	2 PORVs Blocked
Rod Insertion 100% AFW	0.338	0.090	0.023
Rod Insertion 50% AFW	0.034	0.009	0.002
No Rod Insertion 100% AFW	0.338	0.090	0.023
No Rod Insertion 50% AFW	0.034	0.009	0.002

Note: This assumes the following system/component failure probabilities and unavailabilities, and operator action failure probabilities.

- Rod control system in manual - 0.5 operator action failure to drive in control rods
- No PORVs blocked and none fail to open - 0.75
- One PORV blocked or fails to open - 0.20
- Two PORVs blocked or fail to open - 0.05
- 100% AFW = 0.90
- 50% AFW = 0.09
- <50% AFW = 0.01

Table 6-4 Plant Configuration Probabilities, Plant Configuration Management Scheme 2

Condition	0 PORVs Blocked	1 PORV Blocked	2 PORVs Blocked
Rod Insertion 100% AFW	0.848	0.045	0.009
Rod Insertion 50% AFW	0.036	0.002	>0.001
No Rod Insertion 100% AFW	0.045	0.002	>0.001
No Rod Insertion 50% AFW	0.002	>0.001	>0.001

Note: This assumes the following system/component probabilities and unavailabilities.

- Rod control system in automatic - 0.95 reliability of rod control system
- No PORVs blocked and no PORVs fail to open - 0.94
- One PORV blocked or fails to open - 0.05
- Two PORVs blocked or fail to open - 0.01
- 100% AFW = 0.95
- 50% AFW = 0.04
- <50% AFW = 0.01

7 CONFIGURATION MANAGEMENT PROGRAM

The approach for using PRA in risk-informed decisions on plant-specific changes to the licensing basis, specifically Technical Specifications, requires the use of the three-tiered implementation approach. As noted in RG 1.177 (Section 3.1), "Application of the three-tiered approach is in keeping with the fundamental principle that the proposed change is consistent with the defense-in-depth philosophy. Application of the three-tiered approach provides assurance that defense-in-depth will not be significantly impacted by the proposed change." The three-tiered approach includes the following:

Tier 1, PRA Capability and Insights: Assess the impact of the change on CDF, ICCDP, LERF, and ICLERP. This is addressed in detail in Section 5.

Tier 2, Avoidance of Risk-Significant Plant Configurations: Provide reasonable assurance that risk-significant plant equipment outage configurations will not occur when plant specific equipment is out of service consistent with the proposed Technical Specification change.

Tier 3, Risk-Informed Configuration Risk Management: Develop a program that ensures that the risk impact of out of service equipment is appropriately evaluated prior to performing any maintenance activity. This requirement is addressed by the Maintenance Rule.

Although the changes being proposed in this report are not related to Technical Specification requirements and do not impact the licensing basis of the plant, the NRC Staff has indicated on two occasions that they are concerned with how defense-in-depth will be maintained with higher reactivity cores. In the NRC's summary of the WOG/NRC meeting on December 17, 1998 (Reference 18) one major issue is identified as "The staff also noted that there remains a policy question as to what extent MTC would play a role in regulatory space. The staff is not clear as to how the defense-in-depth concept is maintained when MTC is unrestricted." The NRC further stated (Reference 10), "As we understand your proposal, the risk basis will include a configuration risk management program to assure high availability of components to mitigate the severity of ATWS events, such as automatic rod insertion, pressurizer power-operated relief valve (PORV) availability and auxiliary feedwater (AFW) availability. The effectiveness of this program will also be an important element of the staff's review focus. Additionally, in order to provide a sufficient risk informed basis, the staff notes that the WOG submittal should consider the risk impact of an effective configuration risk management program throughout the operating cycle, not solely during the "reference case" UET period (i.e., the UET period assuming all AFW and PORVs are available with rod insertion in manual mode)."

Based on this, the NRC is expecting the issue of potential degradation of defense-in-depth to be addressed by a configuration risk management program. The discussion on defense-in-depth in Section 6.1 states "... it is seen that sufficient defense-in-depth barriers exist such that it is possible to compensate for limited degradation of one barrier with another and, therefore, maintain plant safety afforded by defense-in-depth requirements. This is an effective approach for managing the risk associated with ATWS events when implementing higher reactivity cores or other plant changes." The following discusses two proposed approaches to address this issue. Either can be implemented by utilities via Tier 2 requirements or by a Configuration Risk Management Program.

Approach 1: Assessment of Defense-in-Depth Capability

Tables 4-3, 4-4, 4-7, and 4-8 provide the UETs for the low and high reactivity cores for 100% power and equilibrium xenon. For the low reactivity core there are two configurations, near the start of the cycle, the plant can be operated in which result in a 0 UET. These are for conditions of successful partial rod insertion, both PORVs available, and at least 50% (of total available) AFW flow. For the high reactivity core there is one plant configuration, near the start of the cycle, in which the UET is 0. This is for successful partial rod insertion, both PORVs available, and all AFW available. These are the configurations for which defense-in-depth is not affected early in life. Under other conditions the degree of defense-in-depth, while not necessarily inadequate, may be lessened.

Currently plants can operate with PORVs blocked, with testing and maintenance activities in progress that result in the unavailability of parts of the AFW system (consistent with Tech Spec limitations on AOTs and Maintenance Rule requirements), and with the rod control system in either automatic or manual control. In addition, test and maintenance activities can also take place that result in parts of the reactor protection system being unavailable for short periods of time (again, consistent with the Technical Specifications and Maintenance Rule requirements). These activities can impact defense-in-depth.

By controlling the plant operating configuration plants can maintain defense-in-depth capabilities. Plants can manipulate the plant configuration to ensure they are operating with favorable conditions with regard to UETs, and therefore ATWS events, by limiting the unavailability of systems important to ATWS event mitigation. Possible precautionary actions during UET periods can include the following:

- Operate with the rod control system in the automatic mode
- Limit blocking pressurizer PORVs
- Limit activities on the AFW system, AMSAC, and RPS that result in the unavailability of components within these systems.

These limitations would vary depending on the time in core life and become less restrictive further into the cycle. Certain routine maintenance activities and other non-regulatory activities on these systems could be moved to later in core life when the reactivity feedbacks are favorable.

Based on the PRA results presented and discussed in Section 5, it is seen that configuration restrictions are not required to compensate for large impacts on plant risk. Rather, configuration restrictions are being proposed to address the NRC's concern for possible degradation of defense-in-depth. As previously noted, the time in life when the plant mitigation systems cannot relieve sufficient RCS pressure is dependent on core design, time in core life, and the availability of rod insertion, pressure relief, and AFW. Table 7-1 presents the UET information from Tables 4-7 and 4-8 for the high reactivity core in the form of acceptable plant configurations for different times during the fuel cycle. In this case, defense-in-depth is the basis for acceptable configurations. This table simply shows the plant configuration required to maintain defense-in-depth, with regard to ATWS, at different times in life. It can be used to schedule acceptable times for removal of equipment from service. From this it is seen that later in cycle life offers more configurations that are acceptable from the defense-in-depth perspective. It should be noted that for the situation presented on Table 7-1, no credit for control rod insertion is given if the rod control system is in manual.

The following is an example of the use of this table. From day 14 through day 65, it is necessary to maintain the rod control system in automatic, restrict AFW maintenance activities, and maintain PORVs in the unblocked condition to maintain defense-in-depth. After day 65, AFW maintenance can be performed and defense-in-depth can still be maintained. Therefore, AFW maintenance activities would be scheduled after day 65, providing the other ATWS mitigation features are available.

When components are out of service that are important to ATWS mitigation, acceptable AOTs, or the equivalent of an AOT for systems not included in the Technical Specifications, can be calculated by use of ICCDP and ICLERP assessments. As previously shown in Sections 5.1.7 and 5.2.2, AOTs greater than 3000 hours can be justified for blocked PORVs. Although this is acceptable from a risk perspective, the NRC indicates this is not acceptable from a defense-in-depth perspective. To address the defense-in-depth issue, the following actions are proposed, where appropriate, when operating in an unfavorable time:

- Restrict scheduled maintenance activities on the RPS
- Restrict scheduled maintenance activities on AMSAC
- Restrict scheduled maintenance activities on AFW
- Restrict blocking PORVs
- Place the rod control system in automatic control

The objectives of these actions are to restore defense-in-depth. If defense-in-depth cannot be restored, then several of these actions will reduce the probability of an ATWS event.

As an example, consider a plant operating with the high reactivity core, with the rod control system in manual, the AFW system operable, and no PORVs blocked. At day 120 in the cycle, the plant is in a favorable operating configuration with regard to ATWS. If a PORV is now blocked, this becomes an unfavorable condition. Placing the rod control system in automatic, however, changes the plant back to a favorable condition. If the plant cannot be returned to a favorable condition, then voluntary activities that cause the RPS to be unavailable would be curtailed, reducing the probability of an ATWS event.

None of these restrictions involve changing operation to a plant mode where ATWS events are no longer applicable, such as moving to Mode 3. The risk analysis presented in Section 5 shows that the ATWS risk is small, even when operating in a condition with degraded defense-in-depth. Therefore, a risk argument will not support a plant shutdown. The risk from other potential events during a shutdown and subsequent startup, although small, is not necessarily less than the risk from an ATWS event with degraded defense-in-depth.

In summary, the approach to configuration management is to initially attempt to restore defense-in-depth. If this cannot be accomplished, then activities should be curtailed that cause the RPS and other ATWS mitigative features to be unavailable.

Approach 2: Application of the Plant CRMP

Given the very small contribution of ATWS to plant risk, a second approach is proposed. This relies on using the plant's CRMP to determine the impact of ATWS mitigation equipment unavailability on plant risk. Using this approach, the acceptability of entering and remaining in specific plant configurations is dependent on the plant's Technical Specifications and the risk (CRMP) evaluations. This approach does not specifically address maintaining defense-in-depth, but that is not the purpose of the CRMP evaluations. Given the very small contribution of ATWS events to plant risk, even without RCS pressure mitigation capability, there is very little risk benefit from maintaining high availability of systems providing defense-in-depth for ATWS events. This approach requires that the CRMP include appropriate modeling to address the time dependence of the UET for the various plant configurations (availability of components important to ATWS pressure mitigation).

The CRMP evaluations will assess the impact on risk of removing components for mitigation of ATWS events from service. The compensatory actions taken will be based on the risk impact and may include, as discussed above:

- Restrictions on scheduled maintenance activities on the RPS
- Restrictions on scheduled maintenance activities on AMSAC
- Restrictions on scheduled maintenance activities on AFW
- Restrictions on blocking PORVs
- Placing the rod control system in automatic control

Table 7-1 Configuration Management Approach for the High Reactivity Core

Acceptable Operating Configurations Based on Defense-in-Depth for ATWS

Timeframe (days)	Rod Control System		AFW Maintenance Acceptable[1]	Acceptable Number of Blocked PORVs		
	Automatic	Manual		2	1	0
<14	X		Yes			X
14 to 65	X		No			X
>65	X		Yes			X
>110	X		Yes			X
		X	No			X
>115	X		Yes			X
	X		No		X	
		X	No			X
>134	X		Yes			X
	X		No		X	
		X	Yes			X
>136	X		Yes		X	
		X	Yes			X
>165	X		Yes		X	
	X		No	X		
		X	Yes			X
>170	X		Yes		X	
	X		No	X		
		X	Yes			X
		X	No		X	

Table 7-1 (cont.) Configuration Management Approach for the High Reactivity Core

Acceptable Operating Configurations Based on Defense-in-Depth for ATWS

Timeframe (days)	Rod Control System		AFW Maintenance Acceptable[1]	Acceptable Number of Blocked PORVs		
	Automatic	Manual		2	1	0
>187	X		Yes	X		
		X	Yes			X
		X	No		X	
>192	X		Yes	X		
		X	Yes		X	
>238	X		Yes	X		
		X	Yes		X	
		X	No	X		
>259	X	X	Yes	X		

Note:

1. It is assumed that AFW availability will be controlled by Technical Specification for the AFW system, that is, only one pump is allowed to be out of service at a time. A shutdown is required for two pumps out of service.

8 WOG ATWS APPROACH AND MODEL

The following presents and discusses: 1) the recommended approach to address ATWS issues rising from higher reactivity cores, and 2) the recommended ATWS model for use in plant specific PRA models. The approach is consistent with RG 1.174 and addresses evaluating the impact on risk, in addition to the impact on defense-in-depth and safety margins.

The ATWS model discussed in the following is based on the model presented in Section 5 and is consistent with the model presented in WCAP-11992. If implemented as presented, it will provide a realistic assessment of ATWS risk, in terms of CDF, and can be used to assess the impact on CDF of PMTC, plant power upgrades, and SG issues. This model can also be used to assess the impact on ATWS risk related to the availability of pressurizer safety valves and PORVs, in addition to the reliability of the RPS and AMSAC.

ATWS events can be divided into five states as discussed in Section 5.1. These states are defined based on power level, which impacts the availability of AMSAC, and xenon concentration, which acts as a poison. Equilibrium xenon concentrations are achieved after approximately 50 hours of operation. During plant startups, that follow shutdowns of sufficient length to allow xenon depletion, the xenon cannot be credited in determining UETs. The five ATWS states are:

- State 1: Startup (no equilibrium xenon), Power level <40% (no AMSAC)
- State 2: Startup (no equilibrium xenon), Power level ≥40% (AMSAC)
- State 3: Power Operation (equilibrium xenon), Power level ~100% (AMSAC)
- State 4: Shutdown (equilibrium xenon), Power level ≥40% (AMSAC)
- State 5: Shutdown (equilibrium xenon), Power level <40% (no AMSAC)

All states have unique characteristics and are evaluated with distinct PRA models except for States 3 and 4. These states are very similar, with State 4 being bounded by State 3. The only difference is that State 4 trips would start from lower power levels which would result in lower RCS pressures.

As concluded and discussed in Section 5.1.5, most of these states do not contribute significantly to ATWS CDF. Power operation, including shutdown and power level ≥40% (State 3/4), is the largest contributor. In Table 5-25 the contribution of State 3/4 contributes at least 88% to CDF for the three core types. The other three states are relatively small contributors to ATWS CDF. Based on this, the WOG model only addresses State 3/4.

8.1 ACCIDENT PROGRESSION

The progression of the ATWS event for State 3/4 as described in the following is taken, in part, from WCAP-11992.

An ATWS event is composed of two different events; the first is an anticipated transient generating a reactor trip signal and the second is the failure to insert control assemblies into the core following the trip requirement. Two categories of ATWS events can be defined based MFW availability; events in which MFW continues to run and events in which MFW is unavailable. For ATWS events with MFW available, a less severe power mismatch between heat source and sink results and the RCS peak pressure is less severe. With the loss of MFW, a large mismatch between the heat source and sink occurs which in turn results in the RCS heatup. This heatup causes rising RCS temperature and pressure.

The water level in the SGs will drop as the remaining water in the secondary system, unreplenished by MFW flow, is boiled off. When the SG water level falls to the point where the SG tubes are exposed, the primary-to-secondary system heat transfer is reduced. The reactor coolant temperature and pressure continue to increase to the point where the PORVs and safety valves open. The peak pressure attained is dependent on the capability of the PORVs and safety valves to release the reactor coolant volumetric insurge into the pressurizer.

Depending on reactivity feedback conditions, these changes in the reactor coolant conditions cause the core power to be reduced. If the reactor control system is in the automatic mode, the control rods would begin to be inserted as the reactor coolant heatup begins, reducing power and mitigating the RCS overpressure.

There are several mechanisms by which a plant may be shut down following failure of the RPS to provide a trip signal. Plant procedures instruct operators to initiate a manual trip. This is done from the control board and requires the reactor trip breakers to open. If this fails, the operator can also trip the reactor by interrupting power from the motor-generator sets to the CRDMs. Due to the short period of time available for operator response, this can only be credited if the action can be done from the control room. The operator will also be instructed to manually insert the control rods. Then the operator is instructed to verify or manually trip the turbine, verify AFW started, and initiate emergency boration. Emergency boration will only be successful when the RCS pressure drops below the pressure limits of the charging pumps.

8.2 ATWS EVENT TREE MODEL

ATWS events can be initiated from a wide range of initiating events. The ATWS analysis for Westinghouse PWRs (Reference 6) established that the limiting events, with regard to RCS peak pressure, are the loss of load with subsequent loss of all MFW and complete loss of normal feedwater. These limiting events are both assumed to be initiated from normal operation at full power. If favorable reactivity conditions exist, the reactor core is expected to shut down prior to core damage following any anticipated transient without reactor trip provided the turbine trips and AFW flow is initiated in a timely manner. If unfavorable reactivity feedback conditions exist, there is the possibility that the allowable RCS component stress limits could be exceeded with possible loss of RCS integrity and core damage. The allowable component stress limits are based on the ASME Service Level C limit of 3200 psi.

As with previous ATWS assessments, core damage is conservatively assumed if any one of the following occur:

- Maximum RCS pressure exceeds the pressure limit corresponding to the ASME Boiler and Pressure Vessel Code Level C service limit stress criterion. This is defined as 3200 psi.
- RCS heat removal function is inadequate (either before or after the core is brought subcritical).
- The operator fails to initiate emergency boration in a timely manner.

As discussed in Section 5, an ATWS event tree was developed based on the event tree in WCAP-11992. The overall approach uses the unfavorable exposure time concept. This concept determines the time during the cycle that the reactor cannot mitigate the ATWS overpressure transient, that is, the time the RCS pressure will exceed the pressure limit corresponding to the ASME Service Level C limit of 3200 psi. This time is referred to as the unfavorable exposure time or UET. The UET is only important if the reactor fails to trip, that is, the rods fail to fall into the core. This failure can be due to failure of automatic RPS signals or manual actions, or mechanical failure of the rods or CRDMs. The UETs for a given core are dependent on the availability of AFW to the steam generators for heat removal, partial insertion of the control rods (if rod insertion for reactor trip fails), availability of RCS pressure relief, and negative reactivity feedback.

Figure 8-1 shows the event tree. The first top event, IEV, is the frequency of a plant event that requires a reactor trip. The next four top events, RT (reactor trip, development of the automatic trip signal), OAMG (operator action to trip the reactor from the motor-generator sets), CRI (operator action or rod control system to drive the control rods into the core), and CR (control rod insertion), are all related to equipment and operator action failures that lead to an ATWS event. The ESFAS and AMSAC signal are modeled as alternate methods to start AFW and trip the turbine given that an ATWS event has occurred. AFW100 and AFW50 model the probability of achieving 100% AFW flow, and less than 100% but at least 50% AFW flow. This, along with the availability of pressurizer safety valves and PORVs, is important in mitigating the overpressure event. PR (pressure relief) accounts for the unavailability or failure of safety valves and PORVs. The UETs are also factored into this top event. The UETs are dependent on the available AFW flow (100% or 50% flow), pressure relief available (number of PORVs available or not blocked), and success of partial control rod insertion. LTS (long-term shutdown) models the ability to shut down the reactor by boration after mitigation of the pressure transient.

Several important clarifications on the event tree follow:

- Control rod insertion (CR) is addressed following success of the reactor trip signal (RT) or failure of reactor trip signal and success of the operator to trip the reactor from the motor-generator (MG) sets (OAMG).

- The ESFAS is credited with starting AFW and tripping the turbine only for failures of reactor trip that cannot be associated with common cause failures between development of the reactor trip signal and ESFAS signals. The ESFAS signal is only credited if reactor trip fails due to failure of the control rods to fall into the core.

- AMSAC is assumed to be a diverse means (diverse from the RPS) of actuating AFW and providing turbine trip.

- It is assumed that if an ATWS event has occurred, core damage will occur if AFW is not initiated or the turbine is not tripped.

- LTS is not addressed if CRI is successful. With successful CRI, it is assumed that the control rods will continue to be inserted and the reactor shut down.

The following describes the ATWS event tree and top events in more detail.

8.2.1 IEV: Initiating Event Frequency

This is the frequency of transient events that can lead to ATWS events. This includes all anticipated transient events with equilibrium xenon and initial power levels greater than 40% except, as previously noted, for LOSP, inadvertent safety injections, and inadvertent and manual reactor trips. The equilibrium xenon requirement eliminates events that occur during plant startups. The first year of operation can be eliminated since this is usually not typical of plant operation in the following years.

The following guidelines can be used to determine an initiating event frequency:

- Since plants operate at 100% power, or close to it, trips in the 95% to 100% power range are at-power trips.
- Trips in the 0% to 95% range occur either during startup or shutdown since plants typically operate at or near 100% power.
- Startup trips occur prior to establishing equilibrium xenon and shutdown trips occur after equilibrium xenon has been established.
- The split between startup and shutdown trips can be determined from the probabilities of a trip during startup (0.088) and during shutdown (0.068). These values are discussed in Section 5.1.1.2 and are from WCAP-14333 (Reference 13).
- WCAP-15210 (Reference 11) is a source for trip events at Westinghouse plants.

The model presented in this section assumes MFW is lost for all anticipated transient events. If MFW continues to operate, then the event does not need to address the pressure relief response, including AFW and AMSAC, but only requires long-term shutdown. A split that accounts for MFW continuing to operate may be added to plant specific ATWS models if desired.

8.2.2 RT: Reactor Trip Signal from the RPS

This top event models the failure of the RPS to provide a reactor trip signal when required. Since the RPS provides the trip signal, the control rods still need to drop into the core. If this event is successful, then the CR event is addressed. If this fails, then alternate means to trip the reactor are addressed.

The RPS fault tree model should include the RTBs, either solid state logic cabinets or relay logic cabinets, and signal processing (analog or digital). The fault tree model for failure of a reactor trip signal should credit signals developed from two sets of analog (instrument) channels. For all transient events reactor trip signals will be generated from at least two sets of analog channels. In addition, an operator action should be credited to trip the reactor from the control room reactor trip switch. This operator action backs up failures in the RPS related to the analog channels and components in the logic cabinets, but not failures involving the RTBs. Consideration also should be given to analog channel testing in the tripped or bypassed conditions.

The RPS fault tree models provided in NUREG/CR-5500, Vol. 2 (Reference 14) are acceptable. The RPS fault tree models provided in WCAP-15376 (Reference 19) or WCAP-14333 (Reference 13) can also be used. References 14 and 19 are also data sources for failures of components in the RPS.

The human error probability for the OA to trip the reactor from the control room is plant specific. It is the first action in a series of several OAs that can be taken to prevent or mitigate the ATWS event. Given that it is the first, there are no dependencies on previous actions that need to be considered.

8.2.3 OAMG: Operator Action to Trip the Reactor via the MG Sets

The operator can take an action to trip the reactor by interrupting power to the CRDMs via the MG sets. Since this trips the reactor by interrupting power to the CRDMs, the control rods still need to drop into the core. To take credit for this action, it is necessary for it to be called out in the plant emergency operating procedures and it must be possible to complete the action from inside the control room. Due to the short timeframe available to respond to an ATWS event, actions outside the control room are not feasible. If this action is successful, then the CR top event is addressed. If this action fails, then the operator can take an action to drive the control rods into the core or if the rod control system is in automatic the rods will begin to move into the core automatically. This last action is addressed in top event CRI.

The failure probability used for OAMG depends on the reason RT failed. If the RT signal failed due to SSPS or channel signal processing (analog channels), then the OA included in the RT top event has also failed and there is a higher probability that this OA will also fail. If the RT signal failed due to RTB failure, then the OA in RT was most likely successful and OAMG can be assumed to be independent of other operator actions already taken.

8.2.4 CRI: Action to Drive the Control Rods into the Core

This event models driving the control rods into the core via the rod control system. The rod control system may be under automatic or manual control. This is a plant specific decision. If the rod control system is in manual, the operator can take the action to manually drive the control rods into the core. If the rod control system is in the automatic mode, the rods will start to insert automatically and the operator will continue to insert the rods, if necessary. This action needs to be taken within a very short time following event initiation (minutes) to limit the pressure transient. Success of this action provides 72 steps of insertion (negative reactivity) from the lead bank. Regardless of whether this action succeeds or fails, the ATWS event can be mitigated depending on the availability of AFW and RCS pressure relief. The UETs are impacted by success or failure of this action.

The value used for this event will depend if the rod control system is in automatic or manual. In manual, the value will depend on success or failure of previous OAs.

- Rod control system in automatic: A conservative failure probability value of 0.1 can be used directly. If a lower value is used, it may be necessary to provide a fault tree evaluation of the rod control system to justify the value.

- Rod control system in manual (no credit taken for OAMG): For plants that do not credit OAMG, this OA will follow the OA to trip the reactor from the control room via the reactor trip switch. If the reactor trip signal failed (RT top event) due to SSPS or channel signal processing (analog channels), then the OA included in the RT top event has also failed and there is a higher probability that this OA will also fail. If the RT signal failed due to RTB failure, then the OA in RT was most likely successful and the OA to drive the rods in can be considered not dependent or conditional on the failure of the previous operator action.

- Rod control system in manual (credit taken for OAMG): For plants that credit OAMG, this OA will follow failure of at least one previous OA (failure to trip from the control room via the reactor trip switch) and possibly failure of two OAs (failure to trip from the control room via the reactor trip switch and OAMG). In the case of one failed OA, some credit can be taken for this OA, but it is conditional on the failure of a previous OA. In the case when two OAs have failed, very little to no credit should be taken for this OA.

Regardless, the value used will need to be plant specific.

8.2.5 CR: Sufficient Number of Control Rods Fall into the Core to Shut Down the Reactor

This top event models insufficient control rods fall into the core to shut down the reactor. If the actions, automatic or manual, to initiate reactor trip are successful, the control rods still need to fall into the core to shut down the reactor. With regard to the rod insertion, three outcomes are possible:

- Sufficient number of rods insert to bring the reactor subcritical

- Sufficient number of rods insert to mitigate or partially mitigate the pressure transient, but not to bring the reactor subcritical. This is equivalent to the rods stepping in automatically by the rod control system or by the operator manually inserting the rods.

- Sufficient number of rods fail to insert so the pressure transient is not mitigated.

NUREG/CR-5500, Vol. 2 (Appendix E, Section E-4.2) calculates a probability of 1.2E-06/d for 10 or more rods failing to fully insert. The NUREG report assumes failure of 10 control rods or more to insert results in a loss of shutdown capability and it does not matter which ten rods fail to insert. The NUREG notes that this is conservative. The number of rods that are required to insert to achieve a subcritical core is dependent on the core design and the location of the failed/successful control rods. In addition, the number of control rods required to insert to mitigate the pressure transient, but not provide shutdown, is also dependent on the core design and control rod failure/success locations.

The number of control rods required to insert to mitigate the pressure transient is less than the number of control rods required to bring the reactor subcritical. The NUREG assumption of 10 or more rods failing to insert may be acceptable for shutting down the core, but significant negative reactivity is provided by those that do insert. That is, the pressure transient will be significantly mitigated. This is a conservative assumption (10 or more control rods fail to insert) with regard to the pressure transient since only D-bank insertion credit of 1 minute (72 out of 230 steps) has a significant impact on the UETs. D-bank insertion of 72 steps is significantly less than the number of control rods required to insert per the assumptions of the NUREG report.

The following guidelines are recommended:

- Failing to insert a sufficient number of control rods (failing CR) to provide an equivalent effect of failing to insert D-bank for one minute is not credible, that is, a sufficient number of control rods will always insert to equal the effect of 72 steps from D-bank. The pressure transient will still need to be mitigated, but the UET will be reduced to those values that assume D-bank insertion success.
- Using this definition for failure of CR (10 or more rods failing to insert), it is assumed that the reactor will be critical, but at a lower power level, and long-term shutdown (boration) will be required.

Therefore, the following approach is recommended:

- A sufficient number of rods will always insert so that the pressure transient will be mitigated or severity reduced.
- Probability of failing to insert sufficient rods to bring the reactor subcritical is 1.2E-06/d.
- If CR fails, it is assumed that sufficient rods have been inserted to be the equivalent to 72 steps of D-bank insertion used in the UET calculations.

Note that it is not necessary to address CR following success of CRI. The probability of rods failing to insert is assumed to be included in the probability of CRI failing (CR is very small compared to CRI).

8.2.6 ESFAS: Turbine Trip and AFW Pump Start by the ESFAS

A key assumption regarding ATWS is that a common cause event occurs that disables the RPS and ESFAS completely inhibiting an ESFAS signal from being generated. But for certain equipment failures that lead to failure of reactor trip, such as control rods failing to drop into the core, the ESFAS signal will still be available for turbine trip and AFW pump start. The conditions when ESFAS signals are not available, assuming a common cause event inhibits all RPS signals, are if reactor trip fails due to RTB, logic cabinet, or analog channel failures.

ESFAS signals to start AFW and trip the turbine should only be credited following failure to trip due to failure of the CR top event (rods fail to insert) following successful RT. A detailed fault tree assessment of the ESFAS can be done to develop a failure probability or a conservative value of 0.01 can be used for failure of the signal. The 0.01 value is considered conservative since it is significantly higher than the unavailability of ESF actuation signals as determined in other studies. A WOG program that analyzed the impact of allowed outage time changes on ESFAS reliability (Reference 13) showed that the unavailability of these signals vary from 3E-03 to 7E-04 depending on the specific signal being considered.

8.2.7 AMSAC: ATWS Mitigation System Actuation Circuitry

AMSAC is a diverse method (diverse from the RPS signals) to trip the turbine and start AFW. No detail fault tree analysis of AMSAC has been done, but WCAP-11992 uses a conservative value of 0.01/demand as a failure probability. This value has also been used in other studies and is an appropriate value. A fault tree analysis would probably be required to justify a lower value.

8.2.8 AF100: AFW System Provides 100% Flow

As previously discussed, the UETs are dependent on available pressure relief and AFW flow. AFW is divided into 100% and 50% levels. The 50% level actually represents AFW flow that is less than 100% and greater than or equal to 50%. AF100 represents 100% AFW flow, which is flow from all the AFW pumps, to all steam generators. For an AFW system design with 1 TD AFW pump and 2 MD AFW pumps, in which one MD pump provides half the flow of the TD pump, 100% flow is flow from the TD pump and both MD pumps.

8.2.9 AF50: AFW System Provides 50% Flow

AF50 represents less than 100% AFW flow but at least 50% AFW flow to all steam generators. For a AFW system design with 1 TD AFW pump and 2 MD AFW pumps, in which one MD pump provides half the flow of the TD pump, 50% flow requires flow from either both MD AFW pumps or the TD AFW pump. A conditional value is used since this event is addressed following failure of AF100. The value required is the probability of at least 50% flow failure given 100% flow has failed.

8.2.10 PR: Availability of Primary Pressure Relief

This event models the availability of primary pressure relief to mitigate the overpressure event. PR is dependent on the AFW flow (100% or 50%) and rod insertion (success or failure), and accounts for the UET, availability of PORVs (PORVs blocked or fail to open), and failure probability of the safety valves. It also accounts for the frequency of initiators that can lead to ATWS events with regard to the time when the events occur during the cycle. UETs occur early in the cycle and transient events are more frequent early in the cycle also.

Four sets of UETs are required that correspond to the various combinations of CRI and AFW. Four fault trees are required for PR, one for each set of UETs. Examples of the PR fault trees are provided in Appendix D. There is one fault tree for each AFW/rod insertion combination:

- Fault tree PRA: control rod insertion success, 100% AFW
- Fault tree PRB: control rod insertion success, 50% AFW
- Fault tree PRC: control rod insertion failure, 100% AFW
- Fault tree PRD: control rod insertion failure, 50% AFW

Successful pressure relief requires opening all three safety valves and the required PORVs when the reactivity feedbacks are favorable. Each PR fault tree consists of four subtrees with each subtree modeling pressure relief requirements for a UET interval. The four UET intervals correspond to:

- pressure relief failure with 2 PORVs and 3 safety valves available
- pressure relief success requiring 2 PORVs and 3 safety valves
- pressure relief success requiring 1 PORV and 3 safety valves
- pressure relief success requiring 0 PORVs and 3 safety valves

Plant specific UETs should be used when possible. A conservative set can also be used if available. Four UET sets will be needed with each set corresponding to a CRI/AFW combination. Within each set, UETs are required for 0, 1, or 2 PORVs available. For plants with 3 PORVs, UETs can be used for 0, 1, 2, or 3 PORVs available, providing the PR fault trees are modified to reflect the availability of 3 PORVs.

The UETs will need to be weighted according to the distribution of transient events over the cycle. This is required since the transient frequency is higher in the beginning of the cycle when unfavorable exposure times occur. The distribution in Table 5-3 can be used for this weighting. Weighting calculations are shown in Section 5.1.1.11. From the weighted UETs, the UET intervals that correspond to basic events PRXI1, PRXI2, PRXI3, and PRXI4 in PR fault trees PRA, PRB, PRC, and PRD (where the X represents A, B, C, or D) are calculated. Sample calculations for this are also provided in Section 5.1.1.11.

Plants operate with PORVs blocked, and blocked PORVs cannot be credited to mitigate an ATWS event since there is insufficient time to open the block valve to unblock the PORV. Plant specific values need to be developed for the probability that PORVs are blocked. Probabilities of blocked PORVs can be assumed to be randomly distributed throughout the fuel cycle unless other information is available that disputes this assumption. For plants with two PORVs, probabilities of blocked PORVs should be developed for each PORV and also for two PORVs. For plants with three PORVs, probabilities for blocked PORVs should be developed for each PORV, combinations of two PORVs, and for three PORVs. This is assuming that for plants with three PORVs, UETs will be used that correspond to the availability of one, two, and three PORVs.

Plant specific failure probabilities, if available, should be used for safety valve and PORV failure to open on demand and for common cause failure of multiple PORVs.

8.2.11 LTS: Long Term Shutdown

This event requires the plant operators to establish long-term shutdown which involves starting emergency boration. This is required on success paths that do not have full control rod insertion. If CRI or CR succeed, then rod insertion has occurred and this is not addressed. Note that CRI requires the lead bank to insert 72 steps, with regard to mitigation of the RCS pressure spike, which is not full control rod insertion. It is assumed that with CRI the operators or automatic rod control system will continue to insert the rods until the core is shut down.

The failure probability for this event is dependent on an operator action for initiation of emergency boration. A plant specific value or fault tree should be used for boration. It can be assumed that this action is independent of the previous OAs since it does not need to be completed in the same short time period as the OAs to trip the reactor, trip the MG sets, or manually drive in the control rods.

8.2.12 Event Tree Sequence Endstates

The core damage endstates can be differentiated from each other according to RCS pressure, if required. Distinctions between high RCS pressure and low RCS pressure endstates are based on whether or not pressure relief was successful. Successful pressure relief maintains the RCS pressure below 3200 psi. The following defines the sequence endstates:

Low RCS pressure: Failure of LTS. LTS is only addressed if pressure relief is successful, so any sequence with LTS failure is a RCS low pressure condition.

High RCS pressure: All other core damage endstates are high pressure RCS conditions since they involve failure of pressure relief PRA, PRB, PRC, or PRD. Failure of AFW50 (less than 50% AFW flow) and failure of AMSAC are also equated to high RCS pressure endstates since insufficient AFW flow is available to mitigate the ATWS event with regard to pressure relief.

8.3 PRA MODEL QUANTIFICATIONS AND APPLICATION OF REGULATORY GUIDE 1.174

To demonstrate the acceptability of higher reactivity core designs, the requirements in Regulatory Guide 1.174 need to be met. This includes addressing the impact on risk, as well as the impact on defense-in-depth and safety margins, of higher reactivity cores. For the plant specific evaluation, only the impact on CDF is required to be evaluated for power operation. Power operation is considered to be the time that the plant is operating above 40% with full power equilibrium xenon. As shown previously, the other ATWS states do not contribute significantly to risk and do not need to be included in the evaluation. LERF also does not need to be addressed as long as the plant and core of interest are bounded by the bounding core evaluated in Section 5.2.

To evaluate the impact on CDF, the PRA model for the plant of interest should be quantified for the current core, that meets current requirements, and for the new higher reactivity core. The plant PRA should use plant specific models, parameters, and values, or values that are set to conservatively represent the plant. The change in CDF should be a small impact as defined in RG 1.174. Sensitivity quantifications can be completed on parameters that the results may be sensitive to, such as, the probability of blocked PORVs, credit for the rod control system being in automatic, and credit for operator actions. It is also recommended that ICCDP values be determined for blocked PORVs. This is to provide an indication of the length of time PORVs can be blocked and meet the ICCDP guideline in RG 1.177 (5E-07).

The final step in the WOG approach is to address the impact on defense-in-depth and safety margins. The impact on safety margins is straight-forward and is addressed in Section 6.2. The impact on defense-in-depth should follow the approach in Section 6.1 and develop a configuration management program, which in this case is equivalent to defining Tier 2 restrictions (since these are predefined restrictions) similar to those proposed in Section 7.

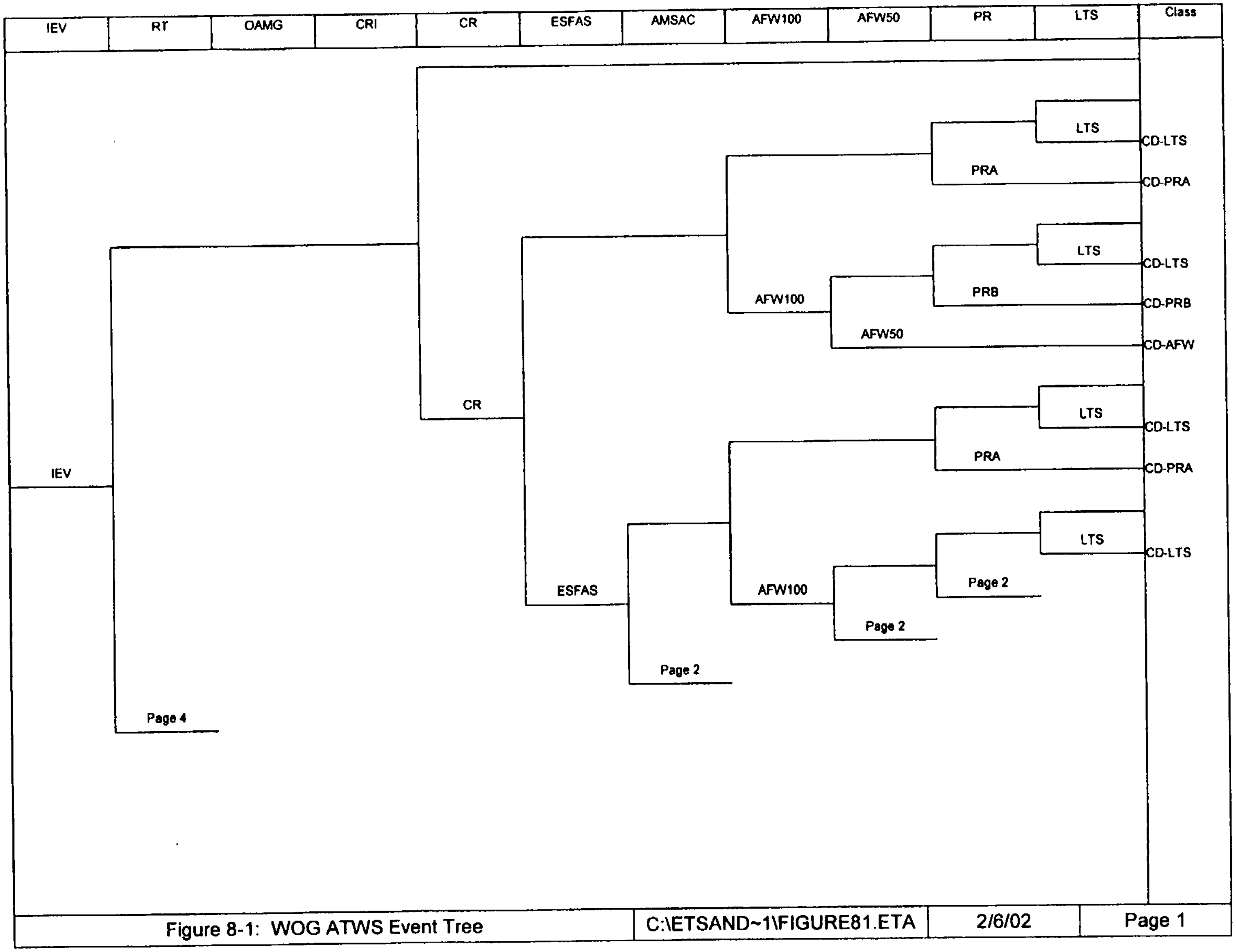

Figure 8-1 WOG ATWS Event Tree, Equilibrium Xenon, Power Level ≥40%

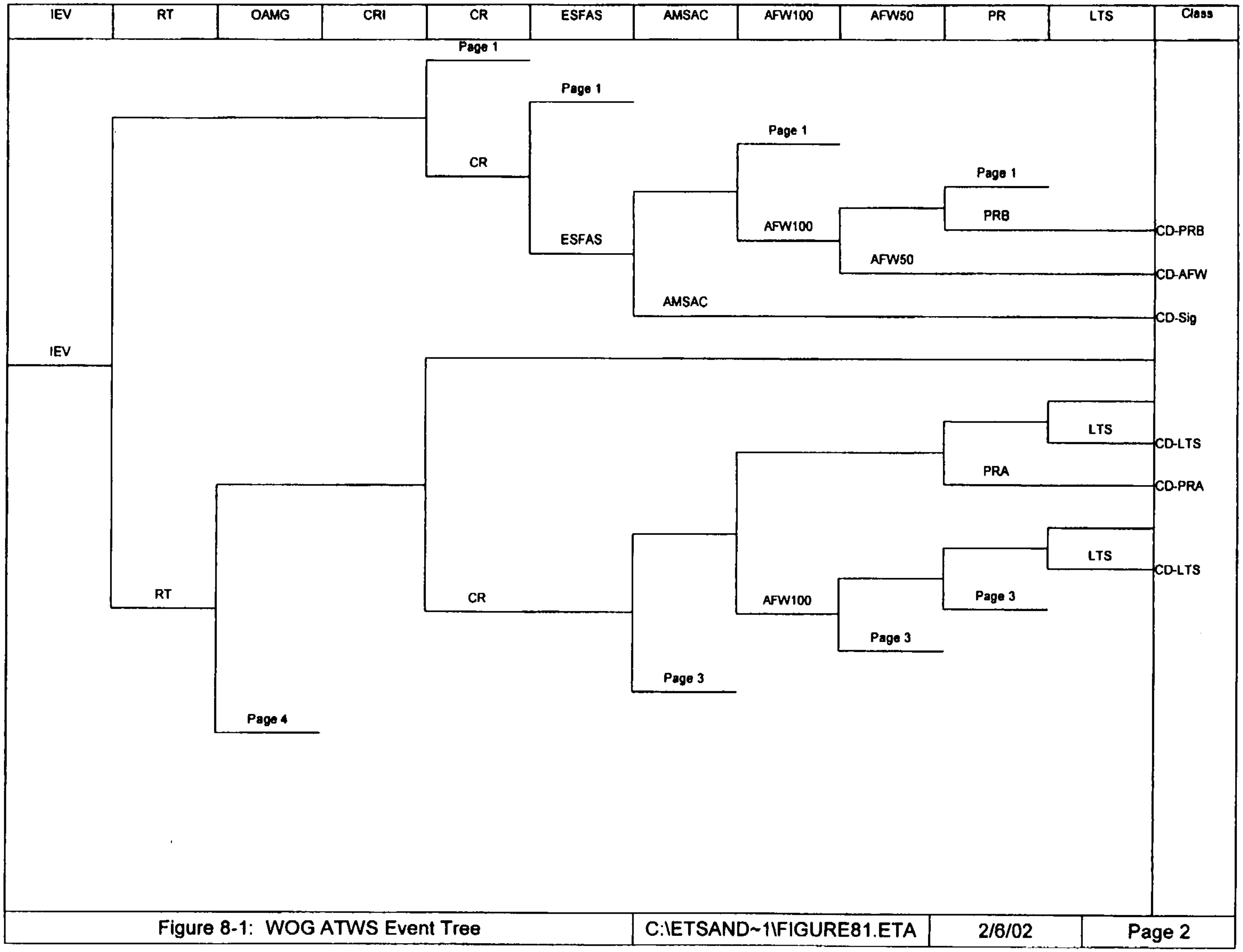

Figure 8-1 WOG ATWS Event Tree, Equilibrium Xenon, Power Level ≥40% (cont.)

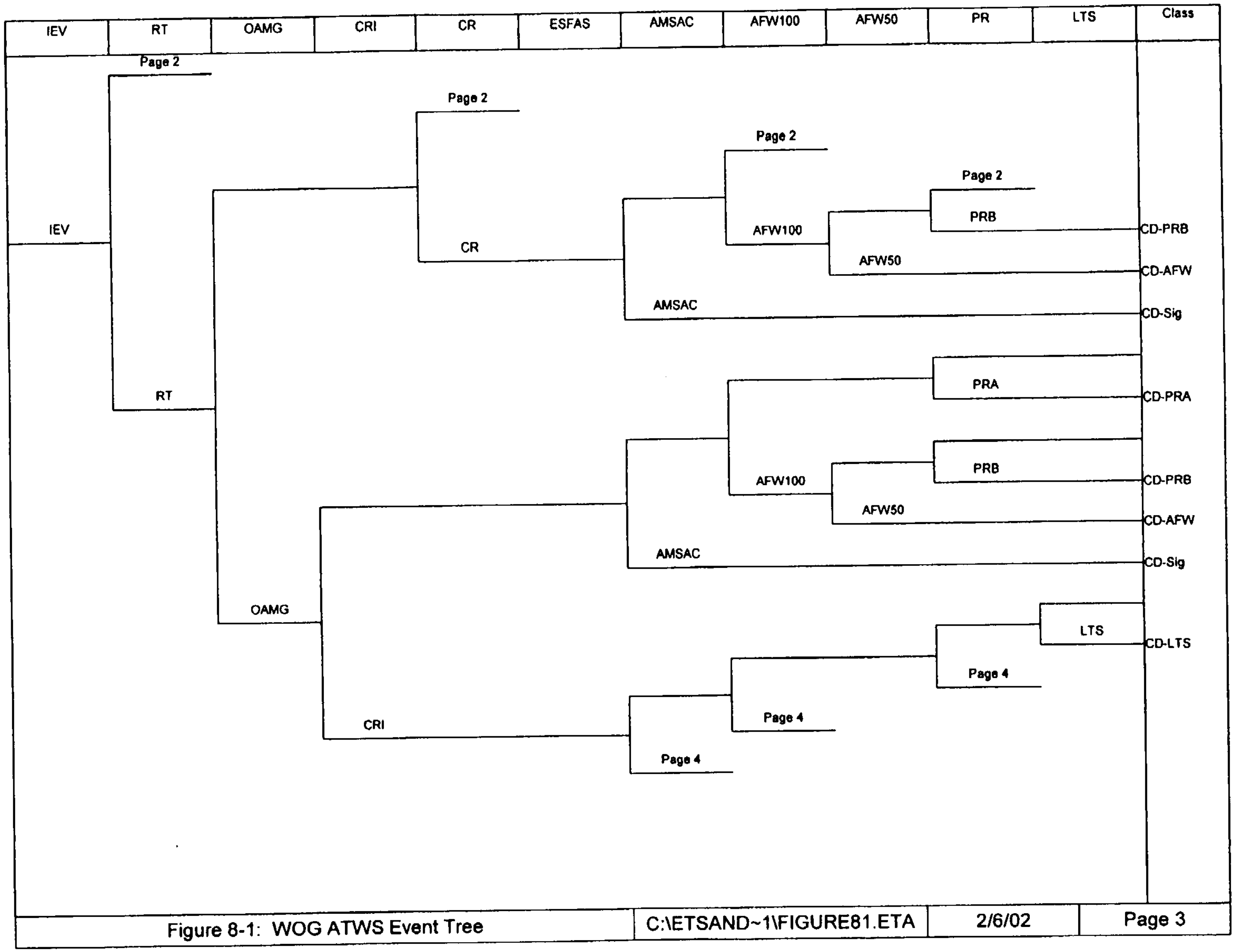

Figure 8-1 WOG ATWS Event Tree, Equilibrium Xenon, Power Level ≥40% (cont.)

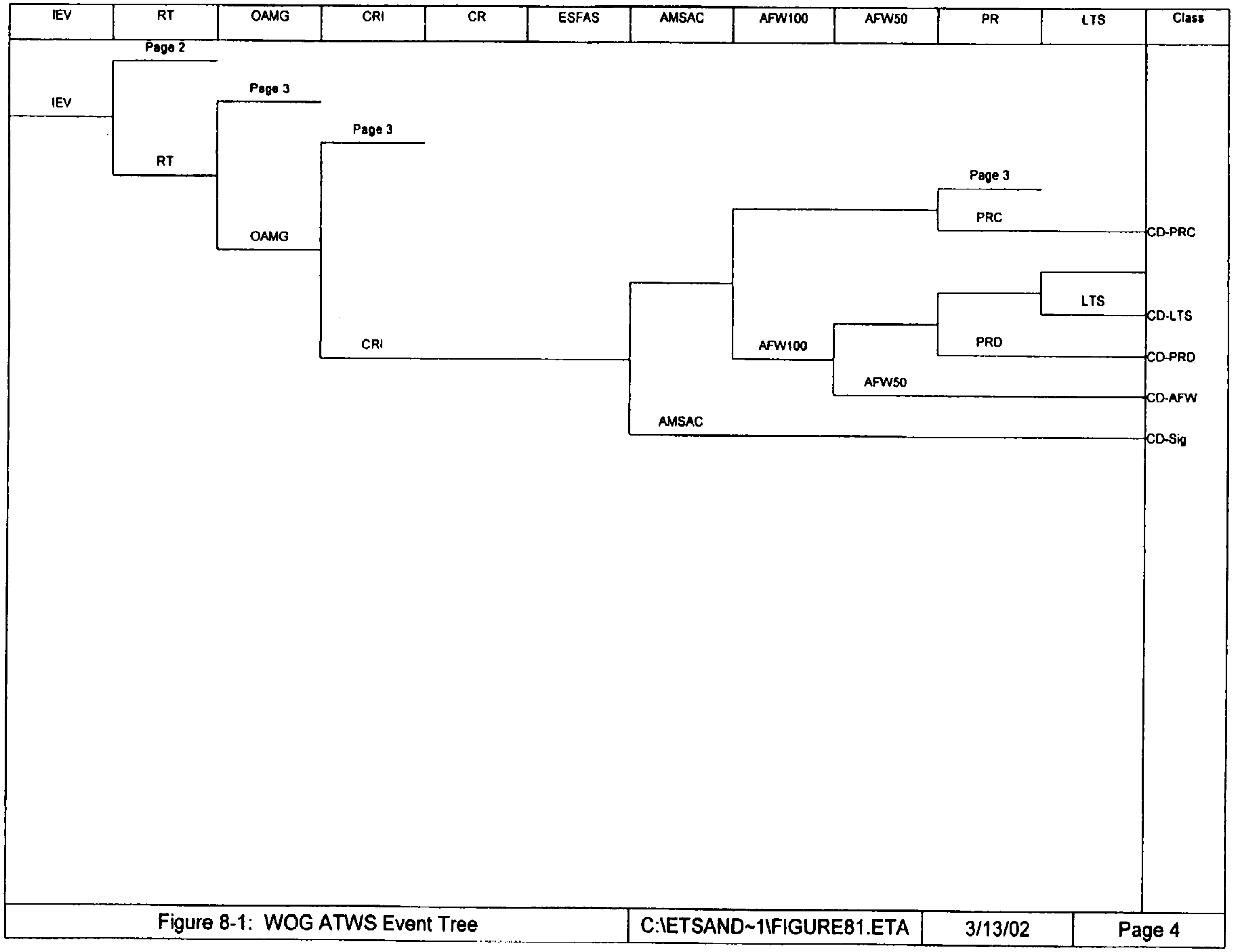

Figure 8-1 WOG ATWS Event Tree, Equilibrium Xenon, Power Level ≥40% (cont.)

9 BRAIDWOOD LEAD PLANT EVALUATION

As discussed in Section 2.3, Byron and Braidwood referenced WCAP-11992 in their PMTC license amendment request. As part of the NRC's review and acceptance of this request, an additional requirement was added to the Byron and Braidwood Technical Specifications that requires core designs to meet a 5% UET for the conditions of no rod insertion, 100% AFW, and no PORVs blocked, referred to as the reference conditions. One objective of the lead plant application is to remove this Technical Specification requirement using the risk-informed approach described in this WCAP.

In this evaluation, the Braidwood PRA model was modified to reflect the WOG model described in Section 8. UETs were developed for the current core design with the 5% UET restriction for the reference conditions and for a core design based on similar requirements, but without the 5% UET restriction. The PRA model was then quantified for both core designs to determine the impact on CDF. ICCDP values were calculated for blocked PORVs. A set of sensitivity evaluations was also completed. A configuration risk management program is provided to address defense-in-depth issues.

9.1 BRAIDWOOD ATWS PRA MODEL

The Braidwood PRA model uses the fault tree linking approach and the CAFTA code system for quantification. The model, as used in this evaluation, includes internal events. The ATWS model in the Braidwood PRA was reviewed for consistency with the WOG model and modified as appropriate. The following discusses each top event in the Braidwood model and how each conforms to the WOG model in Section 8. Table 9-4 provides a summary of the comparison of the WOG and Braidwood models and provides an assessment of the impact of modeling differences on the results. Figure 9-1 shows the Braidwood ATWS event tree.

ATWS Initiators

This top event represents anticipated transient events that have already proceeded to ATWS events. It includes the initiating event frequency for anticipated transients (IEV in the WOG model), failure of the reactor trip signals (RT in the WOG model), and failure to trip the reactor by interrupting power to the MG sets (OAMG in the WOG model).

The Braidwood PRA model uses IE frequencies based on industry and plant specific operating history. These include all events that occur above 40% power. The total IE frequency of anticipated transient events that can result in an ATWS event is approximately 1. This includes LOSP events and events without HFP equilibrium xenon above 40% power. LOSP events are not events that result in increased RCS pressures, but the LOSP IE frequency is small so including this contribution to the total IE frequency has essentially no impact on the results. Including the portion of events that occur above 40% power without equilibrium xenon will increase the IE frequency by a small amount and provide a slightly higher IE frequency.

The reactor trip signal model is based on the model provided in NUREG/CR-5500, Volume 2 (Reference 14) including the component failure probabilities. The Braidwood Station has a solid state protection system as modeled in the NUREG. The trip signal model includes an operator action to trip the reactor from the trip switch in the control room. The HEP for this action is 1.0E-02.

The operator action to trip the reactor by interrupting power to the CRDMs from the MG sets is included in the model. The value used is dependent on previous OA failures. If the previous failures include the OA in the control room from the trip switch, a value of 0.5 is used. If the failures do not include this previous OA, i.e., reactor trip breakers have failed, then a value of 1E-02 is used.

Main Feedwater (ATWS)

Main feedwater is addressed in the Braidwood PRA model, but not the WOG model. If MFW continues to run, then high RCS pressures are not a concern and only long-term shutdown is addressed. The probability that the MFW will not continue to run is 0.23.

Manual Rod Insertion (ATWS)

Manual rod insertion is the operator action to drive the rods into the core or it can represent the probability of the rod control system being in automatic. The value used, if an operator action is assumed, is dependent on the previous failures. If this action follows failure of the OA to trip the reactor via the trip switch in the control room and also failure of the OA to trip the reactor via the MG sets, then the failure probability value used is 1.0. If this follows only the failure of the OA to trip the reactor via MG sets, then a failure probability of 0.5 is used. Several cases use a value of 0.1, which represents unavailability of the automatic rod control system.

Control Rod Failure (mech. binding)

The value used is 1.21E-06/demand and is based on Reference 14.

AMSAC

AMSAC is included in the model to trip the turbine and start AFW. The value used is 1.0E-02.

Auxiliary Feedwater System

The availability of AFW is addressed by a three-way split in the event tree. The bottom path represents less that 50% flow, the middle path represents greater than or equal to 50% flow and less than 100% flow, and the top path represents 100% flow. The Braidwood AFW system consists of two pumps, one motor-driven and one diesel-driven. With this design, 100% flow requires both pumps to operate and 50% requires either pump to operate. Detailed fault trees of the AFW system are used to model this event.

Primary Pressure Relief

The primary pressure relief trees are identical to those used in the WOG model, but a number of inputs are different. These include UETs, probability of blocked PORVs, and failure rates for PORVs and safety values.

The UETs for the current core which meets the 5% restriction and for the higher reactivity core are provided in Tables 4-34 to 4-37. The weighted UETs were developed as discussed in the WOG model using the weighting factors from Table 5-3. The weighted UETs are provided in Tables 9-1 and 9-2. The pressure relief intervals are developed from the weighted UETs following the WOG approach. These are provided in Table 9-3.

The probability of PORVs being blocked are based on plant experience. The values used are:

- Both PORVs blocked = 0.0025
- PORV A blocked = 0.05
- PORV B blocked = 0.05
- No PORVs blocked = 0.8975

PORV failure is modeled by a detailed fault tree and the safety valve failure is modeled as a single basic event. The failure probability for a safety valve failure to open demand is 1.0E-03.

Shutdown of the Reactor

This is equivalent to the LTS (long-term shutdown) in the WOG model. A detailed fault tree for this event is used in the Braidwood model.

Miscellaneous

Note that the only top event not included in the Braidwood model is ESFAS. This models actuation of the AFW and turbine trip by the ESFAS. This is can only be credited if the ATWS event is due to failure of the control rods to insert due to mechanical binding, i.e., the reactor trip signal was present. This represents a small conservatism in the model in comparison to the WOG model.

A summary of the WOG and Braidwood ATWS models is provided on Table 9-4. It is concluded from this review that the Braidwood model follows the WOG model appropriately.

9.2 BRAIDWOOD ATWS CORE DAMAGE FREQUENCY QUANTIFICATIONS

A number of evaluations were performed. The first was for the current core design with the 5% UET restriction (Case B1) and the second was for a future core design without the 5% UET restriction (Case B2). Both assumed that the rod control system is in automatic and the standard probability for blocked PORVs. The only difference between these two cases is the UETs.

The following sensitivity cases were done for the future core design:

- Case B3: Worst Time in Cycle, Standard PORV Blocked Assumptions, Rod Control System in Automatic
- Case B4: End of Cycle, Standard PORV Blocked Assumptions, Rod Control System in Automatic
- Case B5: Yearly CDF, No PORVs Blocked, Rod Control System in Automatic
- Case B6: Yearly CDF, One PORVs Blocked, Rod Control System in Automatic

- Case B7: Yearly CDF, Two PORVs Blocked, Rod Control System in Automatic
- Case B8: Yearly CDF, Standard PORV Blocked Assumptions, Rod Control System in Manual

The results for these cases are provided on Tables 9-5 to 9-7. The following discusses the results:

Table 9-5: By comparison of Cases B1 and B2 it is seen that the impact on CDF of removing the 5% UET core design restriction is very small (ΔCDF = 2.3E-08/yr) which meets the guideline in RG 1.174 that defines a small impact on risk as less than 1E-06/yr. A comparison of Case B8 to Case B2 shows the benefit of operating the plant with the rod control system in automatic. The CDF value decreases a relatively small amount. As discussed in Section 5.1.6, this impact is relatively small when examining the impact across the core life. Placing the rod control system in automatic increases the probability of successful partial rod insertion. Partial rod insertion is not important later in life since it is not necessary to mitigate the RCS pressure transient. Earlier in life the impact is more important since partial rod insertion has more of an impact on the pressure transient.

Table 9-6: This table provides the CDF values for the worst time in the cycle (at the beginning of the cycle, in this case), at the best time in cycle (end of the cycle), and the average CDF for the future core. The end of the cycle value is also applicable to the current core since both cores are favorable in all configurations at the end of the cycle. As seen in this table, the ATWS CDF, which is small to start, decreases significantly through the cycle.

Table 9-7: This table shows the impact of blocking PORVs on CDF. CDF values are provided for 0, 1, and 2 PORVs blocked for the complete cycle with the rod control system in automatic. The safety valves are still available for pressure relief in these cases. Also shown is the CDF for the standard blocked PORV probabilities. This demonstrates that even with both PORVs blocked, the ATWS CDF remains low.

9.3 INCREMENTAL CONDITIONAL CORE DAMAGE PROBABILITY

The ICCDP calculations are discussed in Section 5.1.7 for the generic analysis. As discussed in that section, the ICCDP is used to determine acceptable time periods equipment can be out of service, for example, how long can a PORV be blocked. The ICCDP calculation is generally used to assess changes to the completion times (AOTs) specified in plant Technical Specifications. As shown in Section 5.1.7, an acceptable AOT can be determined based on the acceptance guideline of ICCDP $\leq$ 5E-07 as provided in Regulatory Guide 1.177.

$$\text{AOT(hr)} = (\text{5E-07} \times 8760\ \text{hr/yr})/(\text{CCDF} - \text{CDF}_{\text{baseline}})/\text{yr}$$

where:

CCDF	=	conditional CDF with the subject equipment out of service
$\text{CDF}_{\text{baseline}}$	=	baseline CDF with nominal expected equipment unavailabilities
AOT	=	duration of single AOT under consideration

Given this, the acceptable AOT, based on the yearly average CDF, to have a PORV blocked for the future core follows.

$$AOT = (5E\text{-}07 \times 8760)/(1.67E\text{-}07 - 6.48E\text{-}08) = 42{,}857 \text{ hours} = 4.9 \text{ yr}$$

where:

1.67E-07/yr	=	CDF (yearly average) for future core with one PORV blocked (Case B6)
6.48E-08/yr	=	CDF (yearly average) for future core with standard blocked PORV probabilities (Case B2)

In the generic case presented in Section 5.1.7, the worst time in cycle was used to determine the most limiting time. In this case the yearly average CDF is used which is consistent with the guidelines in RG 1.177.

The calculated AOT value is high since the PORVs, with regard to being blocked, are not important to total plant CDF. With regard to ATWS risk, the CDF increases by a factor of approximately 3 when one PORV is blocked as opposed to no PORVs blocked (see Table 9-7). Even though there is a factor of 3 increase, the magnitude of the increase (1.1E-07/yr) is small since ATWS CDF is small.

9.4 CONFIGURATION MANAGEMENT PROGRAM

This section presents the Configuration Management Program that will be used to address the identified Tier 2 restrictions at Braidwood with higher reactivity cores in the future. The approach to these restrictions is discussed in detail in Section 7.

Tables 4-34 to 4-37 provide UETs for the current and future cores for Braidwood. For the current core there are a number of plant configurations, near the start of the cycle, the plant can be operated in which result in a 0 UET. For the high reactivity core there is one plant configuration, near the start of the cycle, for which the UET is 0. This is for successful partial rod insertion, both PORVs available, and all AFW available. These are the conditions, component and system unavailability, for which defense-in-depth is not affected early in life. For other conditions, the degree of defense-in-depth, while not necessarily inadequate, may be lessened.

As noted in Section 7, currently plants can operate with PORVs blocked, with testing and maintenance activities in progress that result in the unavailability of parts of the AFW system (consistent with Tech Spec limitations on AOTs and Maintenance Rule requirements), and with the rod control system in either automatic or manual control. In addition, test and maintenance activities can also take place that result in parts of the reactor protection system being unavailable for short periods of time (again, consistent with the Technical Specifications and Maintenance Rule requirements). These activities can impact defense-in-depth.

By controlling the plant operating configuration, defense-in-depth capabilities can be maintained. The plant configuration can be controlled to enhance the probability of operating with favorable conditions with regard to UETs, and therefore, ATWS events. The following were noted in Section 7 as possible precautionary actions to take during UET periods:

- Operate with the rod control system in the automatic mode

- Limit blocking pressurizer PORVs

- Limit activities on the AFW system, AMSAC, and RPS that results in the unavailability of components within these systems.

Based on the PRA results presented and discussed in Sections 9.2 and 9.3, it is seen that configuration restrictions are not required to compensate for large impacts on plant risk. Rather, configuration restrictions are being proposed to address the NRC's concern for possible degradation of defense-in-depth. Table 9-8 presents the UET information from Tables 4-36 and 4-37 for the future core in the form of acceptable plant configurations for different times during the fuel cycle. In this case, defense-in-depth is the basis for acceptable configurations. This table shows the plant configuration required to maintain defense-in-depth, with regard to ATWS, at different times in life.

When components are out of service that are important to ATWS mitigation, acceptable AOTs, or the equivalent of an AOT for systems not included in the Technical Specifications, can be calculated by use of ICCDP and ICLERP assessments. As previously shown in Section 9.3, very large AOTs can be justified for blocked PORVs. Although this is acceptable from a risk perspective, the NRC indicates this is not acceptable from a defense-in-depth perspective. To address the defense-in-depth issue, the following actions are proposed, where appropriate, when operating in an unfavorable exposure condition according to Table 9-8:

- Restrict scheduled maintenance activities on the RPS
- Restrict scheduled maintenance activities on AMSAC
- Restrict scheduled maintenance activities on AFW
- Restrict blocking PORVs
- Place the rod control system in automatic control

The objectives of these actions are to restore defense-in-depth. If defense-in-depth cannot be restored, then some of these actions will reduce the probability of an ATWS event.

9.5 CONCLUSIONS FROM THE LEAD PLANT EVALUATION

The following provides the key conclusions from the Braidwood analysis.

- The Braidwood ATWS PRA model follows the WOG model appropriately and can be used to evaluate the impact of plant issues and design changes on ATWS contributions to CDF.

- The impact on CDF of removing the 5% UET core design restriction is very small (ΔCDF = 2.3E-08/yr) and meets the guideline in RG 1.174 that defines a small impact on risk as less than 1E-06/yr.

- Operating with the rod control system in automatic and with the PORVs not blocked reduces the plant CDF. The impact of these actions on the total plant CDF is small, but is significant on the ATWS contribution to the total CDF.

- A PORV can be out of service, or blocked, for a significant length of time based on the ICCDP calculation and the guidelines provided RG 1.177.

- Tier 2 restrictions have been developed that can be implemented into the Braidwood Configuration Risk Management Program to enhance maintaining defense-in-depth during unfavorable exposure times in the cycle. This is not required to compensate for large impacts on plant risk, but rather to address the NRC's concern of possible degradation of defense-in-depth.

Table 9-1 Braidwood Weighted UET Values, Current Core Design with the 5% UET Restriction			
100% Power, Equilibrium Xenon			
Condition	**0 PORVs Blocked**	**1 PORV Blocked**	**2 PORVs Blocked**
RI, 100% AFW	0.00	0.00	0.22
RI, 50% AFW	0.00	0.00	0.42
No RI, 100% AFW	0.00	0.25	0.71
No RI, 50% AFW	0.00	0.48	0.78

Table 9-2 Braidwood Weighted UET Values, New Core Design without the 5% UET Restriction			
100% Power, Equilibrium Xenon			
Condition	**0 PORVs Blocked**	**1 PORV Blocked**	**2 PORVs Blocked**
RI, 100% AFW	0.00	0.35	0.47
RI, 50% AFW	0.23	0.38	0.50
No RI, 100% AFW	0.34	0.51	0.69
No RI, 50% AFW	0.39	0.55	0.75

Table 9-3 Braidwood: Summary of Pressure Relief Intervals		
100% Power, Equilibrium Xenon		
PR Interval Basic Event	**Current Core Design With 5% UET Restriction**	**New Core Design w/o 5% UET Restriction**
PRAI1	0.00	0.00
PRAI2	0.00	0.35
PRAI3	0.22	0.12
PRAI4	0.78	0.53
PRBI1	0.00	0.23
PRBI2	0.00	0.15
PRBI3	0.42	0.12
PRBI4	0.58	0.50
PRCI1	0.00	0.34
PRCI2	0.25	0.17
PRCI3	0.46	0.18
PRCI4	0.29	0.31
PRDI1	0.00	0.39
PRDI2	0.48	0.16
PRDI3	0.30	0.20
PRDI4	0.22	0.25

Table 9-4 Summary of Comparison of WOG and Braidwood ATWS PRA Models

Parameter	WOG Model	Braidwood Model	Comments
IE Frequency	1.0/yr	~1.0/yr	No impact
Reactor Trip Model	NUREG/CR-5500	NUREG/CR-5500	No impact
OA to Trip via Trip Switch in the Control Room	1.0E-02	1.0E-02	No impact
OA to Trip via MG Sets	0.5 HEP after failure of one OA 0.01 HEP after failure of no OA	0.5 HEP after failure of one OA 0.01 HEP after failure of no OA	No impact
OA to Drive Control Rods In	Automatic operation assumed 0.5 base value, 0.1 sensitivity	for automatic operation If in manual: 1.0 HEP after failure of two OAs 0.5 HEP after failure of one OA	Braidwood assumed automatic operation in the base model and later did sensitivities for manual operation of the rod control system
Main Feedwater Availability	Conservatively not addressed	Included in model	Including this results in fewer ATWS events and lower ATWS CDF
Control Rods Fail to Drop	1.2E-06/demand	1.21E-06/demand	No impact
ESFAS	Included in model	Conservatively not addressed	Including this results in slightly lower ATWS CDF
AMSAC	0.01	0.01	No impact
Auxiliary Feedwater	Scalars based on typical AFW system design	Detailed fault tree developed for Braidwood AFW system	Appropriate to use plant specific model
Primary Pressure Relief Model	Detailed fault trees	Detailed fault trees, same as WOG model	No impact

Table 9-4 (cont.) Summary of Comparison of WOG and Braidwood ATWS PRA Models

Parameter	WOG Model	Braidwood Model	Comments
UETs	Low reactivity (5%) core High reactivity core Bounding reactivity core	Current core (with 5% limitation) Future core (without 5% limitation)	WOG cores based on 4-loop plant with model 51 SGs Braidwood cores are plant specific and based on the replacement SGs
Blocked PORV Probabilities	0.10 for either PORV 0.05 for both PORVs Assumed conservative values	0.05 for either PORV 0.0025 for both PORVs Based on plant experience	The less the PORVs are blocked the lower probability of being in a UET. Provides lower CDF results
PORV Failure Probability	7.0E-03/demand	Detailed fault tree developed	Appropriate to use plant specific model
Safety Valve Failure Probability	1.0E-03/demand	1.0E-03/demand	No impact
Long-term Shutdown	1.0E-02	Detailed fault tree developed for Braidwood emergency boration	Appropriate to use plant specific model

Table 9-5 Core Damage Frequency Summary, Current and Future Cores			
Standard Probabilities for Blocked PORVs			
Case	**Core**	**Rod Control System**	**CDF (per yr)**
B1	Current Core	Automatic[1]	4.15E-08
B2	Future Core	Automatic[1]	6.48E-08
B8	Future Core	Manual[2]	7.11E-08

Notes:

1. Failure probability of rod control system = 0.1
2. OA HEP = 1.0 following failure to trip the reactor via the control room trip switch and via the MG sets; OA HEP = 0.5 following to trip the reactor via the MG sets only.

Table 9-6 Core Damage Frequency Summary, Sensitivity Studies, Future Core, Time in Cycle			
Standard Probabilities for Blocked PORVs			
Case	**Time in Cycle**	**Rod Control System**	**CDF (per yr)**
B3	Worst time in cycle	Automatic[1]	1.18E-07
B2	Yearly average	Automatic[1]	6.48E-08
B4	End of cycle	Automatic[1]	4.10E-08

Note:

1. Failure probability of rod control system = 0.1

Table 9-7 Core Damage Frequency Summary, Sensitivity Studies, Future Core, Blocked PORVs			
Rod Control System in Automatic			
Case	**Time in Cycle**	**Number of PORVs Blocked**	**CDF (per yr)**
B2	Yearly average	Standard Probabilities	6.48E-08
B5	Yearly average	0	5.30E-08
B6	Yearly average	1	1.67E-07
B7	Yearly average	2	2.09E-07

Table 9-8 Configuration Management Approach for the Future Braidwood Core						
Acceptable Operating Configurations Based on Defense-in-Depth for ATWS						
Timeframe (days)	**Rod Control System**		**AFW Maintenance Acceptable[1]**	**Acceptable Number of Blocked PORVs**		
	Automatic	**Manual[2]**		**2**	**1**	**0**
≤81	X		No			X
>81	X		Yes			X
>141	X		Yes			X
		X	No			X
>143	X		Yes			X
	X		No		X	
		X	No			X
>163	X		Yes		X	
		X	No			X
>166	X		Yes		X	
		X	Yes			X
>208	X		Yes		X	
	X		No	X		
		X	Yes			X
>225	X		Yes	X		
		X	Yes			X

Table 9-8 (cont.) Configuration Management Approach for the Future Braidwood Core

Acceptable Operating Configurations Based on Defense-in-Depth for ATWS

Timeframe (days)	Rod Control System		AFW Maintenance Acceptable[1]	Acceptable Number of Blocked PORVs		
	Automatic	Manual[2]		2	1	0
>231	X		Yes	X		
		X	No		X	
		X	Yes			X
>256	X		Yes	X		
		X	Yes		X	
>333	X		Yes	X		
		X	Yes		X	
		X	No	X		
>362	X		Yes	X		
		X	Yes	X		

Notes:

1. It is assumed that AFW availability will be controlled by Technical Specification for the AFW system, that is, only one pump is allowed to be out of service at a time.
2. If the rod control system is in manual, no credit is taken for partial rod insertion.

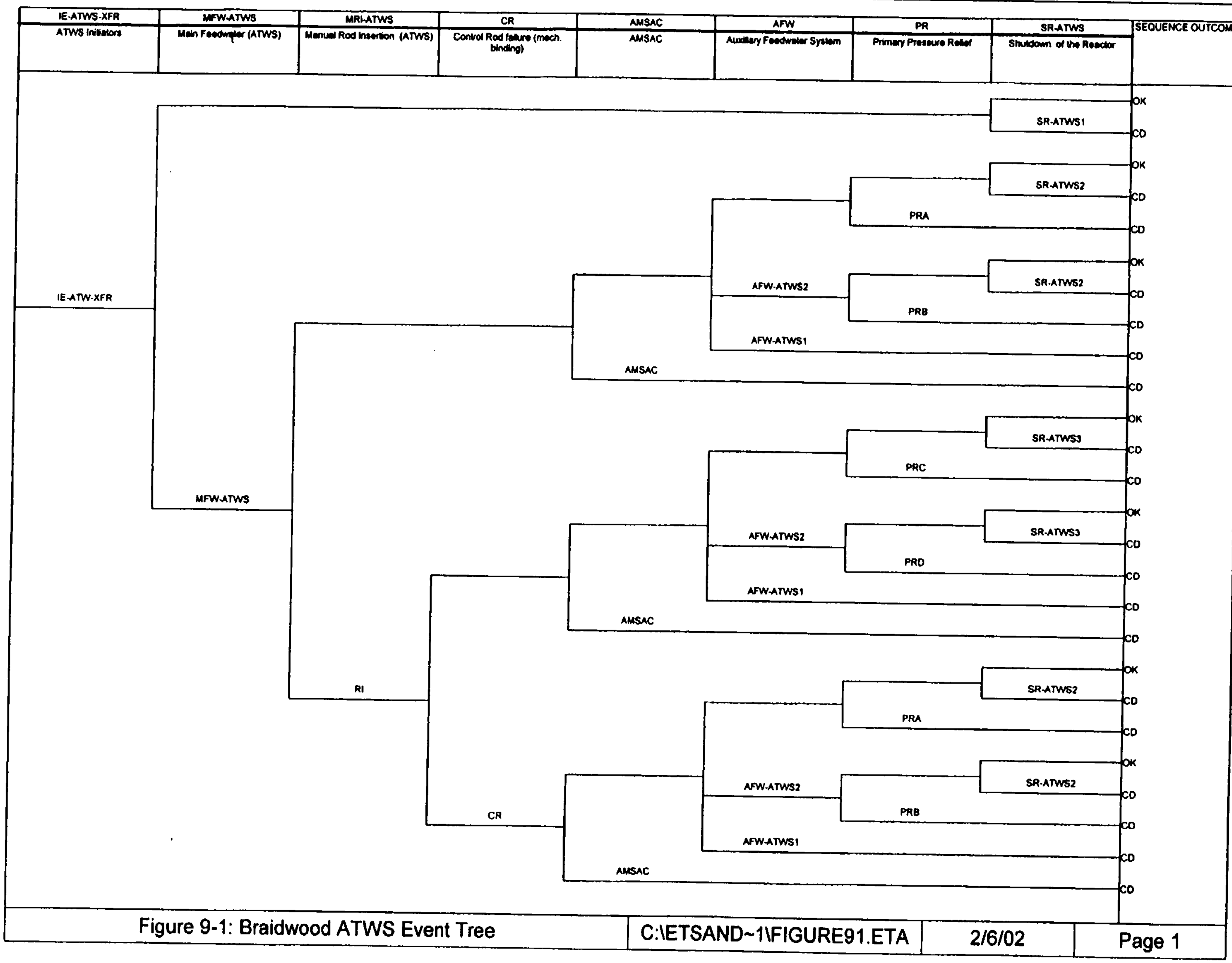

Figure 9-1 Braidwood ATWS Event Tree

10 RELOAD IMPLEMENTATION PROCESS

Plants, as currently licensed, only need to install AMSAC to meet the ATWS Rule (see Section 2.1). A key assumption in the NRC PRA for the ATWS Rule (Reference 4) was that an unfavorable MTC would exist for 10% of the cycle for non-turbine trip events and 1% of the cycle for turbine trip events. The Westinghouse generic analysis for ATWS used a full power MTC of –8 pcm/°F with a sensitivity analysis using an MTC of –7 pcm/°F. In 1979, these values represented MTCs that Westinghouse PWRs would be more negative than for 95% and 99% of the cycle, respectively. The base case of 95% represents an unfavorable MTC for 5% of the cycle. In more recent activities, the NRC imposed a Technical Specification on the Byron and Braidwood Stations, in response to a license amendment request for PMTC, that requires the UET to be no greater than 5% for the reference conditions (no control rod insertion, 100% AFW, and no PORVs blocked). Although not explicitly stated as a licensing requirement in the ATWS Rule, this limit on UET is a restriction on core design for Byron and Braidwood.

Regulatory Guide 1.174 provides an approach for using PRA in risk-informed decisions on plant-specific changes to the licensing basis. In implementing RI decision-making, several key principles are expected to be met. These are (from RG 1.174, Section 2):

1. The proposed change meets the current regulations unless it is explicitly related to a requested exemption or rule change, i.e., a "specific exemption" under 10 CFR 50.12 or a "petition for rulemaking" under 10 CFR 2.802.

2. The proposed change is consistent with the defense-in-depth philosophy.

3. The proposed change maintains sufficient safety margins.

4. When proposed changes result in an increase in core damage frequency or risk, the increases should be small and consistent with the intent of the Commission's Safety Goal Policy Statement.

5. The impact of the proposed change should be monitored using performance measurement strategies.

This report uses the RI approach to demonstrate the acceptability of eliminating the 5% UET restriction in the Byron and Braidwood Technical Specifications. Also, this report demonstrates that ATWS-specific MTC restrictions are unnecessary for all Westinghouse plants.

With the elimination of this requirement, the NRC is concerned about the design and use of reload cores with higher reactivity levels. This issue, higher reactivity reload cores, is applicable to Byron and Braidwood and other plants that have requested PMTC or have already implemented PMTC Technical Specifications. Currently the acceptability of reload cores is addressed via the 50.59 process and does not require prior approval of the NRC provided the provisions of 10 CFR 50.59 are met. The following is the proposed approach to address the NRC's concern.

The reload core implementation will continue under 10 CFR 50.59, but with the additional requirement of following the RI approach to evaluate changes to the plant's licensing basis. That is, the key principles specified above will be met. Applying this RI approach will only be done by utilities with plants that are not consistent with the bases for the ATWS Rule (Reference 6). This report demonstrates that these principles will be met on a generic basis for realistic or typical core designs that utilities would like to use in the future, as well as for a core design that is bounding for most Westinghouse plants. For reload cores, licensees will need to demonstrate that either the 5% UET restriction is met on a best estimate basis for the reference ATWS scenario, or if not, that the generic RI analysis presented in this report is applicable. Licensees can demonstrate that the generic analysis is applicable by evaluating the impact of the new core design on CDF relative to a core design that meets a 5% UET for the reference conditions. This should be done consistent with the WOG model presented in Section 8. The CDF impact should meet the acceptance criteria in RG 1.174. The CDF impact only needs to consider ATWS State 3/4 (power level ≥40% and HFP equilibrium xenon). There is no need to evaluate the other ATWS states since they are small contributors to ATWS risk. In addition, there is no need to evaluate the impact on LERF since this was assessed and found to be acceptable for the bounding core. This plant specific assessment only needs to be done when initially transitioning to a high reactivity reload core (a core that does not meet the 5% UET for the reference conditions). Similar assessments for reload cores for following cycles are not required providing the analysis for the initial change to the high reactivity core remains applicable. In addition, configuration assessments based on either (ATWS) defense-in-depth or risk (as discussed in Section 7) will be required to assess the acceptability of plant operating configuration changes and identification of compensatory actions.

In summary, to demonstrate that a reload core is acceptable, given that the plant is not consistent with the bases for the ATWS Rule, licensees should either:

1. Demonstrate that the best estimate UET assuming no control rod insertion, 100% AFW, and no PORVs blocked is 5% or less.

   OR

2. Implement the WOG ATWS model to demonstrate that the impact on CDF meets the RG 1.174 acceptance guideline shown on Figure 3 of RG 1.174 **AND** implement a Configuration (Risk) Management Program similar to either Approach 1 or 2 described in Section 7 of this report. Note that by meeting the CDF acceptance guideline using the WOG ATWS model, the licensee will demonstrate that the generic ATWS probabilistic risk analysis presented in this report remains applicable.

11 CONCLUSIONS

A WOG ATWS model was developed and presented that can be used to evaluate the impact of ATWS related issues on plant risk. The RI approach presented in RG 1.174, along with the WOG ATWS model, has been used in this program to demonstrate the acceptability of removing the stated or implied 5% restriction on UET for core designs. In particular, this model was used to evaluate the acceptability of high reactivity reload core designs. The following are the key conclusions from this study:

Key Conclusions from the Generic Analysis

- The CDF increase from the low reactivity core to the high and bounding reactivity cores meets the ΔCDF acceptance guideline (<1.0E-06/yr) defined in Regulatory Guide 1.174, and the CDF contribution from ATWS events to plant total CDF is small for all core designs.

- ATWS State 3/4, operation with the power level ≥40% and HFP equilibrium xenon, is the largest contributor to CDF. This state contributes 88% or more to the total ATWS CDF depending on the core reactivity. The other ATWS states (startup without equilibrium xenon for all power levels, and shutdown with xenon equilibrium for power level <40%) are small contributors to plant risk and will not be important to the plant risk profile or to the risk-informed decision process involving changes to a plant.

- Since the CDF and the impact on CDF are dominated by ATWS State 3/4, LERF assessments only need to consider this operating regime. The other ATWS states will be small contributors to LERF and ΔLERF.

- The LERF increase from the low reactivity core to the bounding reactivity core slightly exceeds the acceptance guideline (<1.0E-07/yr) defined in Regulatory Guide 1.174. This is based on the conservative approach that applies the peak configuration specific RCS pressures across the whole cycle. The LERF increase from the low reactivity core to the bounding reactivity core meets the acceptance guideline (<1.0E-07/yr) defined in Regulatory Guide 1.177 for the sensitivity case that assumes that the peak RCS pressures are applicable to 50% of the cycle. That is, the fraction of cycle time for each plant configuration that yields RCS pressures that exceed 3584 psi is 0.5. An RCS pressure of 3584 psi is noted as the pressure where SG tubes will fail resulting in a large release. SG tubes were identified as the first component of the RCS pressure boundary to fail as the RCS pressure increases during an ATWS event.

- ICCDP and ICLERP analysis show that PORV availability is not important to plant risk. Based on the RG 1.177 guideline, one PORV may be blocked for more than 3000 hours per year. This is not because PORVs are not required for ATWS mitigation, but as a result of the low importance of ATWS events to plant risk.

- The impacts on CDF and RCS integrity from LOSP/ATWS events are very small, therefore, this event is not important to the plant risk profile or to risk-informed decision process involving changes to a plant.

- Plant specific ATWS models and risk evaluations only need to consider CDF analyses for ATWS State 3/4 (power level ≥40% with HFP equilibrium xenon) since this state accounts for the largest contribution from ATWS events to plant risk. The plant specific model and evaluation can be used to assess the impact of plant changes on ATWS risk and also to demonstrate that the generic analysis and results are applicable to the individual plant.

- All applicable acceptance criteria for the FSAR Chapter 15 design basis events will continue to be met with the implementation of this risk-informed approach. Therefore, all applicable safety margins will continue to be maintained.

- Tier 2 restrictions can be developed and implemented via a Configuration Management Program that addresses defense-in-depth issue during unfavorable exposure times. This is not required to compensate for large impacts on plant risk, but rather to address the NRC's concern of possible degradation of defense-in-depth.

Key Conclusions from the Braidwood Lead Plant Evaluation

- The Braidwood ATWS PRA model follows the WOG model appropriately and can be used to evaluate the impact of plant issues and design changes on ATWS contributions to CDF.

- The impact on CDF of removing the 5% UET core design restriction is very small (ΔCDF = 2.3E-08/yr) and meets the guideline in RG 1.174 that defines a small impact on risk as less than 1E-06/yr.

- A PORV can be out of service, or blocked, for a significant length of time (greater than a year) based on the ICCDP calculation and the guidelines provided RG 1.177.

- Tier 2 restrictions have been developed that can be implemented into the Braidwood Configuration Risk Management Program to enhance maintaining defense-in-depth during unfavorable exposure times in the cycle.

Reload Implementation Process

To demonstrate that a core reload is acceptable, with regard to ATWS considerations, given that the plant is not consistent with the bases for the ATWS Rule, licensees should either:

- Calculate the UET for the condition of no control rod insertion, 100% AFW, and no PORVs blocked to demonstrate it is 5% or less.

  OR

- Implement the WOG ATWS model and demonstrate that the impact on CDF meets RG 1.174 acceptance guideline shown on Figure 3 of RG 1.174 **AND** implement a Configuration (Risk) Management Program similar to either Approach 1 or 2 described in Section 7 of this WCAP. Note that meeting the CDF acceptance guideline using the WOG ATWS model demonstrates that the generic analysis is applicable.

Based on the analysis presented in this WCAP, it is concluded that restrictions on UETs for higher reactivity core designs should be eliminated. This is based on the RI approach, which demonstrates that the impact on risk is small, safety margins are not impacted, and defense-in-depth can be addressed via a Configuration Management Program.

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Appendix A
Issues Identified by the NRC at the NRC/WOG December 17, 1998 Meeting

WOG Responses are Provided for Each Issue

Issue 1: Defense-in-Depth

The NRC has noted that maintaining existing defense-in-depth features of licensed nuclear power plants is important even when the impact of a desired plant change on CDF is small. With respect to ATWS in particular, a concern has been expressed that the use of core designs with more positive moderator temperature coefficients might be undesirable because it reduces the inherent core reactivity feedbacks (one of the defense-in-depth features of existing PWRs), which serve to shut down the core in the event of a plant transient. NRC views defense-in-depth in three layers for ATWS concerns: the first is core feedback, the second is the reactor trip system with backup operator actions, and the third is the set of plant features that serve to limit the pressure transient (or core heatup) that results from an ATWS event. The NRC requested information regarding how the loss of the "prevention" barrier is compensated for by the other barriers.

Response: The defense-in-depth issue is discussed in detail in Section 6.1. A process for compensating for the loss of the "prevention" barrier with other barriers is discussed in Section 7 (Configuration Management Program). Since the impact on risk has been shown to be small, this process is directed at maintaining defense-in-depth, and not to compensate for large impacts on plant risk.

Issues 2, 3, and 4: Large Early Release Frequency and Component Aging Considerations

The following response addresses three issues raised by the NRC. These are concerned with the structural integrity of the RCS pressure boundary during potential ATWS events. The basic issue concerns failure of the RCS and subsequent releases from containment either through containment failure, containment isolation failure, or containment bypass. Containment bypass could be via either the steam generator tubes or systems that interface with the RCS, such as the residual heat removal or letdown system. A statement of the issues follows.

Issue 2: Large Early Release Frequency

The NRC is concerned with how the containment and safety systems inside containment will respond to the potentially large RCS pressure increase and ensuing high energy break that could occur during an ATWS event. The WOG approach assumes core damage occurs if the pressure exceeds 3200 psig and a study has been done to show that the RCS will remain intact up to this pressure. It is assumed that a LOCA, that cannot be mitigated, will eventually occur that will relieve the RCS pressure in a relatively controlled manner; containment systems and the containment will not be degraded. The specific NRC concern is directed at the level of confidence that the assumed LOCA will occur, as the RCS pressure exceeds 3200 psi, and relieves the pressure increase, as opposed to a catastrophic failure of the RCS that results in missile generation, degradation of containment safety systems, and possible containment failure.

Issue 3: Steam Generator Tube Integrity

Current studies have indicated that the SG tubes will withstand an ATWS pressure peak that results in RCS failure. A 5% probability of SG tube failure is generally used if the RCS pressure increases to a point that the RCS fails (RCS pressure > 3200 psi). The NRC is concerned that with relaxation of SG tube structural requirements that ATWS induced SG tube ruptures could become an issue in the future. This was seen as an issue that the NRC and industry will need to keep in mind and re-visit as necessary, but no specific response is expected. (Note that even though no specific response was expected when the issue was stated by the NRC, SG tube integrity has been addressed in the response to Issues 2 and 4.

Issue 4: Component Aging Considerations

The NRC agrees that previous analyses done indicate that the RCS components will maintain their integrity up to 3200 psi, but these analyses assumed new or like-new component conditions. The concern is that with aged components this conclusion may not remain valid. This question arose with regard to valves that function to provide part of the RCS pressure boundary, and potentially interfacing system LOCAs and containment bypass issues.

Response: The following discusses the response of the RCS components to the potential high pressures during an ATWS event and the impact on large early release frequency. The RCS pressure during an ATWS event is dependent on the core design and time in core life, in addition to the availability of pressure mitigating systems and negative reactivity insertion. The systems and components that are important in mitigating the RCS pressure are the pressurizer PORVs and safety valves, the AFW system, and the rod control system.

As discussed in Section 5.2, a three part approach was taken to address LERF. Part 1 involves a comprehensive examination of the RCS, and interfacing systems and components, to determine if these components remain intact at the expected RCS pressures, or if missiles may be generated or interfacing systems fail such that containment integrity is degraded. Part 2 calculates the expected RCS pressures that correspond to the various combinations of control rod insertion, AFW, and PORV availability. These are used, in conjunction with the results from Part 1, to identify sequences that lead to containment degradation. Part 3 determines the frequencies of these sequences and calculates the LERF for the low, high, and bounding cores.

The RCS integrity assessment (Part 1) is provided in the following. The ATWS RCS pressure analysis and results (Part 2) are provided in Section 4. The ATWS LERF analysis and results (Part 3) are provided in Section 5.2.

RCS Integrity Assessment

A comprehensive examination of the RCS components, and systems and components that interface with the RCS was completed to identify any components that would fail at or below the RCS peak pressure for the bounding core (4110 psia). These components were divided in the following groups:

- Valves
- RCS Piping and Interfacing System Piping
- Pressurizer
- Steam generators
- Reactor vessel
- Reactor Coolant Pumps

A review of the design requirements of the components was completed as well as an assessment of the potential impact of aging on the component's structural integrity. It is important to note that the boundaries of this investigation are consistent with the traditional system boundaries of normally closed valves, isolation valves, check valves, and closed loop configurations. It was recognized that for closed loop configurations (i.e., steam generator tubes) a catastrophic failure of the wall would result in extended boundaries. The review considered external deadweight loads as the only additional source of stress beyond the pressure transient generated stress. Thermal expansion stresses will exist in many of these systems and may be reasonably large, but because of their nature, they tend to be self-limiting and redistribute with system deflections (unlike the pressure and deadweight stresses). The following sections discuss the findings for each group.

- Valves

a,b,c

a,b,c

a,b,c

- RCS Piping and Interfacing System Piping

The RCS and interfacing system piping in plants is designed in accordance with the requirements of ASME Section III or the equivalent B31.1 requirements. Under original design conditions, the design pressure of these systems is typically 2485 psi for design temperatures up to 680°F, and there is a nominal margin of safety for the pressure design of a factor of three. This piping is expected to retain structural integrity for the projected ATWS pressure of 4100 psi. Class 1 or piping with design pressure of 2500 psi would typically be schedule 160. Piping with design pressure of 1000 psi to 2000 psi would typically be schedule 80 or schedule 120 depending upon pipe size. The piping under discussion is typically at least schedule 80 or higher. Table A-1 provides a summary of stress intensity based on principal stress calculations for hoop, radial, and axial stress resulting from an applied pressure of 4100 psi to the straight stainless steel pipe typically attached to the RCS and interfacing systems. The only additional contributor to stress under the ATWS scenario is applied loads due to deadweight. The resulting stress for deadweight loads is typically less than 5000 psi for nuclear applications and, if added directly to the stress tabulated in Table A-1, would remain below recognized ASME Code limits for faulted one-time events. Clearly, the piping will not fail.

Table A-1 Stress Intensity in psi for an Applied Pressure Stress of 4100 psi					
Nominal Pipe Size (inches)	Schedule				
	80	120	140	160	XXS
1/8	6512				
1/4	6825.6				
3/8	7783.3				
1/2	8208			6745.5	5044.5
3/4	9537.6			7122.6	5580.2
1	10180			7668.4	5859.1
1 1/4	11826			9321.6	6605.6
1 1/2	12822			9468.9	7070.8
2	14544			9641.8	7890.1
2 1/2	13954			10571	7607.4
3	15501			10967	8350.3
3 1/2	16631				
4	17593	13777		11560	9365.9
5	19434	14831		12084	10266
6	20057	15651		12470	10572
8		15908	14207	12847	13263
10		16828	14366	12891	
12		16844	15088	13091	
14		16902	14923	13386	
16		17310	14832	13485	
18		17267	15323	13571	
20		17568	15206	13633	
22		17823	15583	13875	
24		17458	15466	13734	

Table A-2 summarizes the resulting stress intensity in straight pipe for a principal stress based calculation for thickness reduced to 2/3 of nominal and applied deadweight loads equal to 5000 psi. This 33% allowance for potential wall thinning is impossible for stainless steel class 1 or class 2 piping and is included only to illustrate the margin available in piping. Again the calculated values remain within Code limits for stainless steel pipe, and so failure will not occur.

Table A-2 Stress Intensity in psi for an Applied Pressure Stress of 4100 psi and Deadweight Load Stresses of 5000 psi on Pipe with 2/3 of Original Thickness

Nominal Pipe Size (inches)	Schedule				
	80	120	140	160	XXS
1/8	13987				
1/4	14483				
3/8	15973				
1/2	16627			14356	11513
3/4	18657			14948	12466
1	19631			15796	12932
1 1/4	22119			18328	14135
1 1/2	23619			18552	14867
2	26207			18815	16138
2 1/2	25320			20223	15702
3	27642			20823	16845
3 1/2	29338				
4	30779	25055		21718	18396
5	33536	26638		22508	19762
6	34468	27868		23089	20225
8		28254	25701	23656	24281
10		29632	25939	23722	
12		29657	27023	24023	
14		29744	26775	24467	
16		30355	26639	24616	
18		30291	27377	24744	
20		30742	27200	24839	
22		31123	27766	25202	
24		30577	27591	24990	

a,b,c

a,b,c

a,b,c

a,b,c

a,b,c

a,b,c

a,b,c

a,b,c

a,b,c

a,b,c

a,b,c

a,b,c

a,b,c

a,b,c

a,b,c

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3. WCAP-13587, Revision 1, "Reactor Vessel Upper Shelf Energy Bounding Evaluation For Westinghouse Pressurized Water Reactors," September 1993.

4. Manufacturer's Literature on Swage Lock Fittings.

5. "Criteria for Design of Elevated Temperature Class 1 Components," Section III, Division 1 of the ASME Boiler and Pressure Vessel Code, ASME, New York, 1976.

6. EM-4531, Rev 2.

7. EM-4860, Rev 1.

8. WCAP-13673, "Background and technical Basis: Handbook on Flaw Evaluation for the Sequoyah Units 1 and 2 Main Coolant System and Components," November 1993.

9. WCAP-14576, "Aging Management Evaluation for Class 1 Piping and Associated Pressure Boundary Components," August 1999.

10. ASME Boiler and Pressure Vessel Code, Section XI, ASME, New York, 1995.

Issue 5: Part Power Considerations

The NRC is interested in the risk associated with part power operation. Particularly of concern is the risk when the reactor is initially started or re-started following a shutdown earlier in the cycle, when unfavorable exposure times exist. The concern is the risk related to the plant startup and the increased potential for a trip during this plant transient operation. The current ATWS models only include reactor trips at power levels greater than 40%. It was not clear if this risk is adequately addressed in ATWS models.

Response: The issue concerns plant risk due to ATWS events that occur during plant start-up following refueling and start-up following a reactor trip or required plant shutdown during the fuel cycle. The specific concern is the unfavorable exposure time during and immediately following the restart as related to the time it takes to build up equilibrium xenon concentration. The analysis used to determine the at-power UETs assumes that full power equilibrium xenon concentration exists. Without full power equilibrium xenon concentration, those UETs are not applicable.

The ATWS probabilistic risk analysis presented in Section 5 addresses ATWS risk due all phases of plant operation. This includes power operation before and after establishing equilibrium xenon concentration, and below 40% power without AMSAC and above 40% with AMSAC. Table 5-1 defines five different ATWS states analyzed. UETs were determined for each ATWS state and are provided in Section 4. Table 5-25 provides a summary of the ATWS CDF contributions for the five ATWS states. It was concluded from this study that the most important ATWS state, from the risk perspective, is with xenon equilibrium and the power level greater than or equal to 40%. The other states are relatively minor contributors to risk from ATWS events. Based on this it was concluded that plant PRA models that include ATWS events initiated from above 40% power with xenon equilibrium do adequately address ATWS risk and that the operating time prior to establishing xenon equilibrium does not need to be included in plant PRA models.

Issue 6: UET/MTC Link

The NRC is interested in the link between MTC (moderator temperature coefficient) and UET (unfavorable exposure time). They are concerned that all the inter-dependencies are not known and that some simplifications may lead to a secure feeling, but that a cliff may loom nearby. The NRC is interested in the range of the various coefficients that are used in the UET calculations. Sensitivity studies will need to be done to address this concern.

Response:

Unfavorable Exposure Time and Critical Power Trajectories

For a given plant and core design, the UET represents the period of time during the operating cycle when an ATWS event could lead to primary system pressures of greater than 3200 psi. The methodology used to determine the UET involves comparing two critical power trajectory (CPT) curves. The ATWS analysis is performed using LOFTRAN. The first CPT curve is calculated based on the reactivity feedback model used in the LOFTRAN analysis that results in a peak RCS pressure of 3200 psi. This CPT represents the change in power as a function of inlet temperature for this reactivity feedback model. To generate these curves, the transient analyst simulates an ATWS event and adjusts the moderator feedback (moderator density coefficient) in the point kinetics core model until the peak pressure limit is reached.

The second CPT curve is the set of inlet temperature and power level combinations that lead to criticality at the ATWS peak pressure <u>in the actual core and using realistic feedback mechanisms</u>. This second curve is generated by the core designer using a three-dimensional core model (ANC). This is the same core model that is used to assess key safety parameters for design basis events for the Reload Safety Evaluation. Realistic moderator, Doppler, and power feedbacks are employed. Using the core model, the core designer calculates a series of critical power levels as a function of inlet temperature and cycle burnup. The core designer then compares these critical power levels with the CPT curve from the system code. If, at a given cycle burnup step, the core critical power (CPT curve 2) is less than the peak pressure power (CPT curve 1), then that burnup is favorable with respect to meeting the 3200 psig limit. If, on the other hand, the critical power from the core model is greater than the peak pressure power from the system code, then that burnup is unfavorable. By calculating the fraction of the cycle that is unfavorable, the core designer determines the UET, usually in terms of number of effective full power days (EFPD) or percent of the cycle.

The limiting ATWS event for peak pressure is the Loss of Load event. Here, the increase in core inlet temperature drives the transient and the core response. As the core inlet temperature and system pressure increase, the natural core reactivity feedback mechanisms will respond and cause the power to drop. These feedback mechanisms effectively balance one another so that the core remains critical, albeit at a new statepoint condition. Briefly, a typical ATWS scenario is as follows: The core begins at steady state conditions operating at full power with nominal temperatures and pressures. When the ATWS event occurs, the inlet temperature rises causing a corresponding increase in system pressure. Since the full power moderator temperature coefficient is always negative, the core responds by dropping power. The positive reactivity increase caused by the drop in power effectively balances the negative reactivity effect of the increase in inlet temperature, resulting in a new critical condition. The two primary reactivity effects, then, are the moderator density feedback and the power coefficient feedback. The power feedback includes both moderator and Doppler components. These primary feedback mechanisms and their relationship to ATWS events are discussed below.

Moderator Density Feedback

Increases in coolant inlet temperatures will add negative reactivity to the core because of the negative moderator temperature coefficient. In response, the core power decreases, and equivalent positive reactivity is added to the core due to the combined effects of Doppler and moderator feedback (see power feedback discussion below).

ATWS events, however, involve not only an increase in core inlet temperature, but also an increase in system pressure. This complicates the moderator feedback since it becomes more than simply a temperature feedback at constant pressure; it involves a change in the moderator density associated with changes in inlet temperature, pressure, and power. For this reason, one cannot simply multiply the moderator temperature coefficient by the inlet temperature increase to determine the amount of negative reactivity added to the core during the event. This method will tend to overestimate the negative reactivity addition core since it doesn't account for the positive reactivity component associated with the pressure increase. Another reason that this simple approach will not work is that the moderator temperature coefficient is not a static value; it is a function of the dynamic reactor conditions, becoming more negative with decreasing moderator density and increasing moderator temperature.

Another factor that complicates the moderator feedback is axial flux redistribution. Whenever the inlet temperature or system pressure increases, the core axial power shape will change even if the reactor power is held constant. This change in axial power shape affects core reactivity since the axial burnup distribution of the core is not uniform. Generally, the net effect of an increase in both system pressure and core inlet temperature is a shift in the axial power distribution toward the bottom of the core, making the core less reactive due the higher fuel burnup there and higher axial power peaking. Reactivity changes due to redistribution are subtle reactivity effects that are usually implicitly included in the moderator, Doppler, and power coefficients.

As the above suggests, the moderator feedback during this kind of event has several components and complicating factors. Consequently, in any assessment of the core reactivity balance for an ATWS event, moderator feedback must be accounted for as part of an integrated reactivity effect between reactor states.

Doppler Feedback

Doppler feedback comes into play in association with the inlet temperature increase and power feedback (see power feedback discussion below). Generally, Doppler temperature feedback is a function of fuel type and power density. It is not a strong function of the core loading pattern or fuel burnup. Consequently, for a given plant, Doppler temperature feedback will not vary much from cycle to cycle or within a cycle.

The negative Doppler feedback that occurs due solely to the moderator temperature increase does not play a dominant role in an ATWS event, but it is important and must be accounted for in the overall reactivity balance. As the coolant temperature increases, the fuel temperature will also increase, adding negative reactivity to the core. Like the moderator feedback, Doppler feedback also has a redistribution component associated with changes in axial power shape and peaking factors. Higher power peaking and more highly skewed power shapes yield increased Doppler feedback.

More important than the Doppler feedback due to moderator temperature increase is the positive Doppler feedback in conjunction with the drop in core power. This is discussed in the following section.

Power Feedback

In the ATWS reactivity balance, a drop in reactor power effectively balances the negative reactivity effects associated with the inlet temperature increase. The overall power feedback is the sum of the moderator, Doppler, and redistribution reactivity components associated with this drop in reactor power, with the moderator component being the most dominant in the latter half of the cycle.

As reactor power drops, moderator density increases, fuel temperatures decrease, and power shifts toward the top of the core. Each of these effects adds positive reactivity to the core. The critical power level is that reactor power that just balances the net negative reactivity due to the inlet temperature increase. Because the moderator temperature coefficient generally becomes more negative with cycle burnup, power feedback becomes stronger with cycle burnup. Early in the cycle, the critical boron concentration is at its highest value. During this time, the moderator temperature feedback is at its weakest. As the core burns and the critical boron concentration decreases, the MTC becomes increasingly negative with cycle burnup. For a given inlet temperature increase, then, a larger drop in reactor power will occur at end-of-life than at beginning-of-life. For this reason, the unfavorable portion of the cycle is always nearest the beginning of the cycle.

ATWS Reactivity Balance

To characterize the interplay of these various reactivity components, reactivity balances were quantified for the low, high, and bounding reactivity cores designs for selected core inlet temperatures and cycle burnups. The reactivity balance is associated with five successive reactor states defined so as to separate the reactivity components:

1. Nominal HFP Steady State Condition (2250 psi, 556.6°F T_{in}, 3565 MWt)

2. Increased Pressure Condition at Nominal Inlet Temperature (3200 psi, 556.6°F T_{in}, 3565 MWt)

3. Increased Pressure with Higher Inlet Temperature, Moderator Feedback Held Constant (3200 psi, T_{in} of 600° to 660°F, 3565 MWt, moderator feedback same as State 2)

4. Increased Pressure and With Higher Inlet Temperature, Moderator Feedback Included (3200 psi, T_{in} of 600° to 660°F, 3565 MWt)

5. Critical Power Condition (3200 psi, T_{in} of 600° to 660°F, critical power level)

States 1 and 5 are critical states representing the initial and final reactor states. State 2 is a supercritical state resulting from the pressure increase. States 3 and 4 add in the negative Doppler and moderator density feedback, respectively. State 4 is always subcritical because of the negative reactivity associated with the inlet temperature increase (decreased moderator density). State 5 is the final critical power condition where the power is decreased to balance the negative reactivity resulting from States 1-4.

Tables A-7, A-8, and A-9 show these reactivity balances as well as moderator density coefficients, Doppler temperature coefficients, pressure coefficients, and power coefficients for the low, high, and bounding reactivity cores, respectively. In these tables, the pressure coefficient was calculated using the core k_{eff} values from States 1 and 2 above. The Doppler coefficient was calculated using States 2 and 3. The moderator density feedback was calculated using States 3 and 4. Finally, the power coefficient was calculated using States 4 and 5. Note that these coefficients represent average values between the reactor states. Furthermore, slightly different values would have been obtained if the order of the reactor states were changed. For example, if the inlet temperature were increased in State 2 and the pressure increased in State 3, the coefficient values would change somewhat. The above order was chosen primarily to avoid coolant voiding in the model, which would occur in the high inlet temperature cases if the pressure were not increased first.

Tables A-7, A-8, and A-9 also provide the HFP MTC at nominal conditions, the calculated critical powers, and the critical power limits for 3200 psig system pressure. These critical power limits correspond to the reference ATWS scenario, which assumes all PORVs available and full auxiliary feedwater.

Tables A-7, A-8, and A-9 illustrate the differences between cores with different excess reactivities with respect to ATWS performance. The low reactivity core achieves more negative MTC values early in the cycle through the use of a much larger loading of burnable absorbers. As a result, this core exhibits lower critical NSSS powers early in the cycle and a much smaller UET overall. With increasing cycle burnup, the critical powers of the high and bounding reactivity cores approach those of the low reactivity core. This occurs since, as burnup progresses and the burnable absorbers deplete, the cores have similar reactivity coefficients and reactivity balance values, i.e., their reactivity feedbacks become comparable.

Figure A-1 illustrates how the critical power varies with HFP MTC. Figure A-1 plots the calculated critical powers for the low, high, and bounding reactivity cores as a function of HFP MTC for both the 600° and 640°F inlet temperature cases. The plotted values come from Tables A-7, A-8, and A-9. Note that all three cores follow the same critical power versus MTC trendlines. This means that for a given inlet temperature and HFP MTC, these cores would be expected to have similar critical powers.

Note also that the MTC that yields a "favorable" critical power is slightly different for the two inlet temperatures. For this particular core and for inlet temperatures of 600°F, the MTC must be more negative than approximately -7.5 pcm/°F to achieve a favorable critical power. For the 640°F inlet temperature, the "favorable" MTC value is about -7 pcm/°F. Thus, the MTC requirement for a favorable critical power will vary somewhat depending on the inlet temperature assumption. For these scenarios, then, the UET is determined by the fraction of the cycle for which the HFP nominal MTC is less negative than these values.

The MTC requirement will also vary depending on the ATWS scenario being considered (number of PORVs available, auxiliary feedwater assumption, control rod insertion assumption) and the plant specific operating conditions (nominal power level, nominal inlet temperature, etc.) since these assumptions affect the peak pressure critical power limits calculated by LOFTRAN. If, for example, one were to assume a different ATWS scenario where only one PORV was available instead of two, the critical power limits corresponding to 3200 psig would be lower, and a more negative MTC value would be required to achieve a favorable critical power. Conversely, for a given core design and MTC versus burnup behavior, these lower critical power limits would lead to a higher UET for this particular ATWS scenario relative to the reference case.

In the risk-informed approach being proposed, all of the reactivity effects discussed above (Doppler, moderator, and power feedbacks) are implicitly included in the evaluation of each ATWS scenario through the reactivity balance that is inherent in the critical power and UET calculations. In this way, the particular feedback characteristics of a specific core design and the critical power limits appropriate for a particular plant are accounted for in the overall risk evaluation.

Table A-7 Low Reactivity Core; Critical Powers, Reactivity Balance, and Reactivity Coefficients; Reference ATWS Scenario

Initial Conditions and Final T_{in}	T_{in} = 600°F				T_{in} = 620°F				T_{in} = 640°F				T_{in} = 660°F			
Cycle Burnup (MWD/MTU)	150	3000	10000	21700	150	3000	10000	21700	150	3000	10000	21700	150	3000	10000	21700
NSSS Power (MWt)	3579	3579	3579	3579	3579	3579	3579	3579	3579	3579	3579	3579	3579	3579	3579	3579
Initial T_{in} (°F)	556.4	556.4	556.4	556.4	556.4	556.4	556.4	556.4	556.4	556.4	556.4	556.4	556.4	556.4	556.4	556.4
Final T_{in} (°F)	600	600	600	600	620	620	620	620	640	640	640	640	660	660	660	660
Critical Power																
Critical NSSS Power (MWt)	2624	2588	2331	2021	1971	1911	1533	1059	1255	1173	702	146	431	313	<0*	<0*
Critical Power Limit (MWt)	2627	2627	2627	2627	2008	2008	2008	2008	1288	1288	1288	1288	429	429	429	429
Unfavorable Power (MWt)	-3	-39	-296	-606	-37	-97	-475	-949	-33	-115	-586	-1142	2	-116	<-415	<-415
Reactivity Balance (values in pcm)																
Pressure Reactivity	80	77	163	403	80	77	162	403	80	77	162	403	79	77	162	404
Doppler Reactivity	-51	-52	-55	-57	-74	-75	-79	-83	-95	-96	-101	-106	-114	-116	-122	-128
Moderator Reactivity	-444	-447	-813	-1785	-851	-871	-1476	-3044	-1481	-1533	-2427	-4692	-2579	-2688	-3919	-7025
Power Reactivity	416	422	705	1439	845	869	1393	2723	1496	1552	2365	4394	2613	2727	3706	5666
Net Reactivity Change	0	0	0	0	0	0	0	0	0	0	0	0	0	0	-173*	-1083*
Reactivity Coefficients																
HFP Nominal MTC (pcm/°F)	-7.79	-7.59	-14.53	-33.11	-7.79	-7.59	-14.53	-33.11	-7.79	-7.59	-14.53	-33.11	-7.79	-7.59	-14.53	-33.11
Pressure Coefficient (pcm/psi)	0.084	0.081	0.171	0.425	0.084	0.081	0.171	0.425	0.084	0.081	0.171	0.424	0.084	0.081	0.171	0.425
Doppler Temp. Coefficient (pcm/°F)	-1.26	-1.29	-1.35	-1.41	-1.26	-1.28	-1.35	-1.40	-1.25	-1.27	-1.34	-1.40	-1.25	-1.27	-1.34	-1.39
Moderator Density Coef. ($\Delta\rho$/gm/cm^3)	0.078	0.079	0.143	0.316	0.096	0.098	0.167	0.345	0.118	0.122	0.194	0.377	0.150	0.156	0.227	0.410
Power Coefficient (pcm/%)	-15.5	-15.2	-20.1	-32.9	-18.7	-18.6	-24.3	-38.5	-23.0	-23.0	-29.3	-45.6	-29.6	-29.8	-37.1	-56.7

* A power level of 0 was calculated. Statepoint is subcritical. A negative power would be required for criticality.

Note: A positive unfavorable power indicates that the primary system pressure limit will be exceeded, while a negative unfavorable power indicates that the primary system pressure limit will not be exceeded.

Table A-8 High Reactivity Core; Critical Powers, Reactivity Balance, and Reactivity Coefficients; Reference ATWS Scenario

Initial Conditions and Final T_{in}	**T_{in} = 600°F**				**T_{in} = 620°F**				**T_{in} = 640°F**				**T_{in} = 660°F**			
Cycle Burnup (MWD/MTU)	150	3000	10000	22006	150	3000	10000	22006	150	3000	10000	22006	150	3000	10000	22006
NSSS Power (MWt)	3579	3579	3579	3579	3579	3579	3579	3579	3579	3579	3579	3579	3579	3579	3579	3579
Initial T_{in} (°F)	556.4	556.4	556.4	556.4	556.4	556.4	556.4	556.4	556.4	556.4	556.4	556.4	556.4	556.4	556.4	556.4
Final T_{in} (°F)	600	600	600	600	620	620	620	620	640	640	640	640	660	660	660	660
Critical Power																
Critical NSSS Power (MWt)	2656	2688	2374	2014	2014	2053	1600	1041	1312	1351	788	117	495	527	<0*	<0*
Critical Power Limit (MWt)	2627	2627	2627	2627	2008	2008	2008	2008	1288	1288	1288	1288	429	429	429	429
Unfavorable Power (MWt)	29	61	-253	-613	6	45	-408	-967	24	63	-500	-1171	66	98	<-415	<-415
Reactivity Balance (values in pcm)																
Pressure Reactivity	71	55	147	408	71	55	148	408	71	55	147	408	71	55	147	407
Doppler Reactivity	-51	-52	-55	-57	-74	-75	-79	-82	-95	-96	-101	-105	-114	-116	-122	-127
Moderator Reactivity	-412	-361	-751	-1800	-800	-734	-1377	-3066	-1401	-1328	-2280	-4720	-2426	-2360	-3701	-7055
Power Reactivity	393	359	659	1450	803	754	1308	2740	1424	1370	2234	4418	2469	2421	3588	5641
Net Reactivity Change	0	0	0	0	0	0	0	0	0	0	0	0	0	0	-87*	-1133*
Reactivity Coefficients																
HFP Nominal MTC (pcm/°F)	-7.05	-5.75	-13.36	-33.5	-7.05	-5.75	-13.36	-33.5	-7.05	-5.75	-13.36	-33.5	-7.05	-5.75	-13.36	-33.5
Pressure Coefficient (pcm/psi)	0.075	0.058	0.155	0.429	0.075	0.058	0.155	0.429	0.075	0.058	0.155	0.429	0.075	0.058	0.155	0.429
Doppler Temp. Coefficient (pcm/°F)	-1.26	-1.29	-1.34	-1.40	-1.26	-1.28	-1.34	-1.39	-1.26	-1.28	-1.34	-1.39	-1.25	-1.27	-1.33	-1.38
Moderator Density Coef. ($\Delta\rho$/gm/cm^3)	0.073	0.063	0.132	0.319	0.090	0.083	0.155	0.348	0.112	0.106	0.182	0.379	0.141	0.137	0.215	0.412
Power Coefficient (pcm/%)	-15.2	-14.3	-19.5	-33.0	-18.3	-17.6	-23.6	-38.5	-22.4	-21.9	-28.5	-45.5	-28.5	-28.3	-35.9	-56.4

* A power level of 0 was calculated. Statepoint is subcritical. A negative power would be required for criticality.

Note: A positive unfavorable power indicates that the primary system pressure limit will be exceeded, while a negative unfavorable power indicates that the primary system pressure limit will not be exceeded.

Table A-9 Bounding Reactivity Core; Critical Powers, Reactivity Balance, and Reactivity Coefficients; Reference ATWS Scenario

Initial Conditions and Final T_{in}	**T_{in} = 600°F**				**T_{in} = 620°F**				**T_{in} = 640°F**				**T_{in} = 660°F**			
Cycle Burnup (MWD/MTU)	150	3000	10000	22470	150	3000	10000	22470	150	3000	10000	22470	150	3000	10000	22470
NSSS Power (MWt)	3579	3579	3579	3579	3579	3579	3579	3579	3579	3579	3579	3579	3579	3579	3579	3579
Initial T_{in} (°F)	556.4	556.4	556.4	556.4	556.4	556.4	556.4	556.4	556.4	556.4	556.4	556.4	556.4	556.4	556.4	556.4
Final T_{in} (°F)	600	600	600	600	620	620	620	620	640	640	640	640	660	660	660	660
Critical Power																
Critical NSSS Power (MWt)	2923	2862	2420	2014	2399	2303	1661	1041	1775	1647	862	117	1012	855	<0*	<0*
Critical Power Limit (MWt)	2627	2627	2627	2627	2008	2008	2008	2008	1288	1288	1288	1288	429	429	429	429
Unfavorable Power (MWt)	296	235	-207	-613	391	295	-347	-967	487	359	-426	-1171	583	426	<-415	<-415
Reactivity Balance (values in pcm)																
Pressure Reactivity	22	23	131	407	22	23	131	407	22	23	131	407	22	23	131	406
Doppler Reactivity	-52	-52	-55	-57	-74	-75	-79	-82	-95	-96	-101	-106	-114	-116	-122	-127
Moderator Reactivity	-211	-231	-684	-1799	-478	-527	-1268	-3066	-938	-1032	-2123	-4720	-1794	-1951	-3479	-7061
Power Reactivity	240	260	608	1449	530	578	1217	2741	1011	1105	2093	4419	1886	2044	3449	5648
Net Reactivity Change	0	0	0	0	0	0	0	0	0	0	0	0	0	0	-21*	-1133*
Reactivity Coefficients																
HFP Nominal MTC (pcm/°F)	-2.84	-2.99	-11.92	-33.46	-2.84	-2.99	-11.92	-33.46	-2.84	-2.99	-11.92	-33.46	-2.84	-2.99	-11.92	-33.46
Pressure Coefficient (pcm/psi)	0.023	0.025	0.137	0.429	0.023	0.024	0.138	0.429	0.023	0.025	0.137	0.428	0.023	0.025	0.137	0.428
Doppler Temp. Coefficient (pcm/°F)	-1.26	-1.28	-1.35	-1.40	-1.26	-1.28	-1.35	-1.40	-1.25	-1.28	-1.34	-1.39	-1.25	-1.27	-1.34	-1.38
Moderator Density Coef. ($\Delta\rho/gm/cm^3$)	0.037	0.041	0.120	0.318	0.054	0.059	0.143	0.348	0.075	0.082	0.169	0.379	0.104	0.113	0.202	0.412
Power Coefficient (pcm/%)	-13.0	-12.9	-18.7	-33.0	-16.0	-16.2	-22.6	-38.5	-20.0	-20.4	-27.5	-45.5	-26.2	-26.8	-34.5	-56.5

* A power level of 0 was calculated. Statepoint is subcritical. A negative power would be required for criticality.

Note: A positive unfavorable power indicates that the primary system pressure limit will be exceeded, while a negative unfavorable power indicates that the primary system pressure limit will not be exceeded.

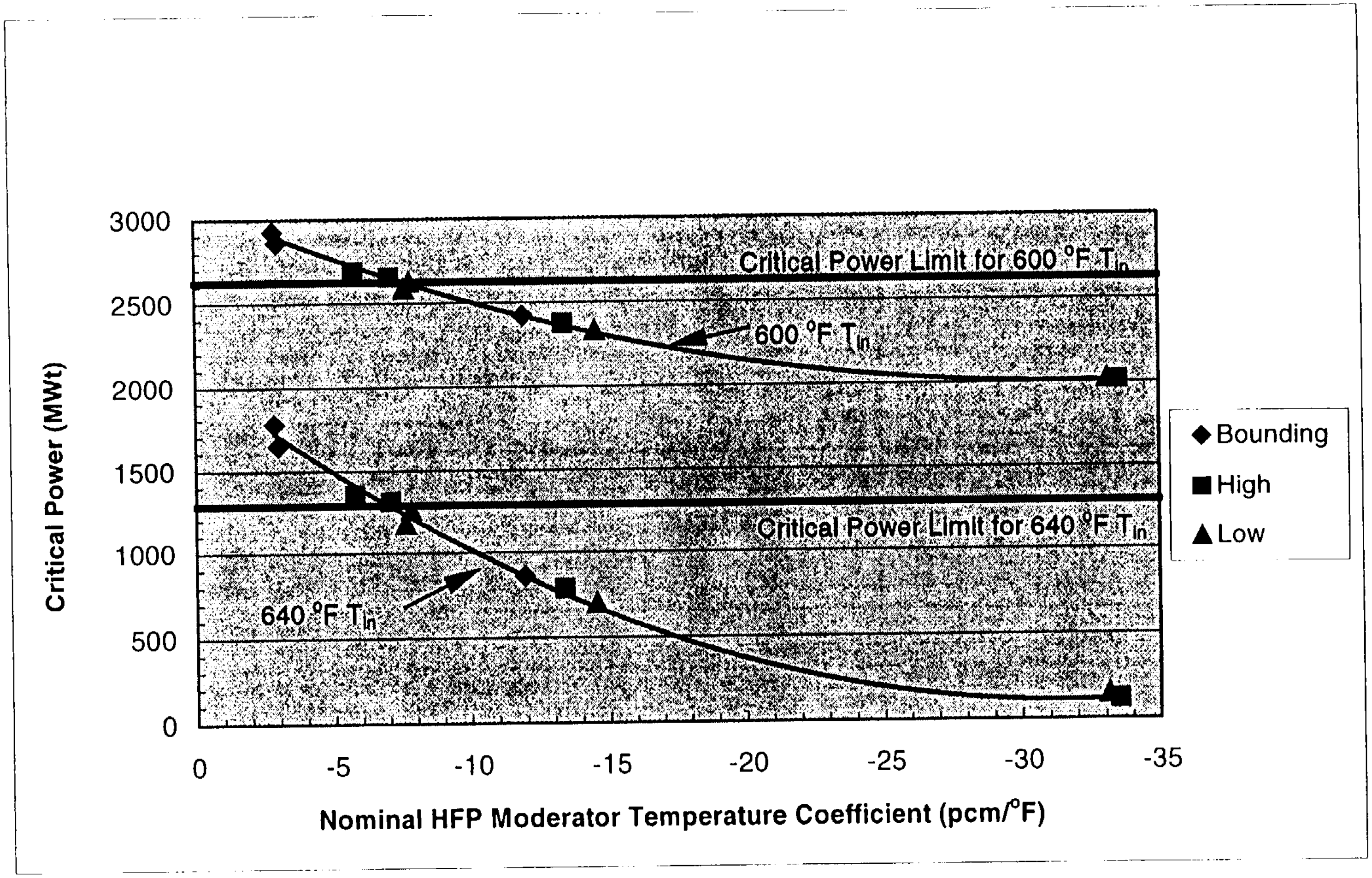

Figure A-1 Critical Powers for the Bounding, High, and Low Reactivity Core Designs for the Reference ATWS Scenario

Issue 7: Impact on Safety Margins

Requirements from other Chapter 15 events need to be maintained. It will be necessary to show that there is no impact on design basis event margins. The NRC noted that this issue is not directly a PRA issue.

Response: The current WOG program is developing a risk-informed approach consistent with Regulatory Guide 1.174 that can be used on a plant specific basis to demonstrate that the impact of core design changes, specifically those related to the moderator temperature coefficient, on plant safety is acceptable. Regulatory Guide 1.174 requires that the impact on plant risk, as measured by core damage frequency and large early release frequency, in addition to the impact of the change on defense-in-depth and plant safety margins be assessed. The impact on safety margins is discussed in Section 6.2. But in summary, all applicable acceptance criteria for the FSAR Chapter 15 design basis events will continue to be met with the implementation of this risk-informed approach.

Issue 8: Loss of Offsite Power with ATWS Events

Failure of the control rods to insert following a loss of offsite power (LOSP) event is not specifically addressed in the generic PRA ATWS model. The NRC would like to see this addressed on a generic and/or plant specific basis; whichever is necessary.

Response: Section 5.3 addresses the LOSP/ATWS event on a generic basis and contains a discussion of the analysis. The following is concluded in Section 5.3:

- LOSP/ATWS events are not significant contributors to plant CDF or plant ATWS CDF.
- LOSP/ATWS events to not produce high RCS pressures and do not impact RCS integrity.
- The increase in CDF from LOSP/ATWS events in moving from the low reactivity core to the bounding reactivity core is very small.
- Since the impacts on CDF and RCS integrity from LOSP/ATWS events are very small, this event will not be important to the plant risk profile or to risk-informed decision process involving changes to a plant.

Therefore, the LOSP/ATWS event does not need to be included in plant specific PRA models.

Issue 9: Control Rod Insertion

The model currently assumes there is no link between burnup and control rod insertion requirements. This will need to be addressed to either: 1) show it is not important, 2) use a conservative value on the control rod insertion requirements, or 3) use different requirements for different times in the fuel cycle. Another comment on control rod insertion requirements is related to the event used to determine the number of rods required to insert. It was asked if this assumption covers all events.

Response:

During an ATWS event, the reactor coolant inlet temperature increases. The natural reactivity feedback mechanisms (moderator and Doppler) respond by reducing the core power level, effectively limiting the primary system pressure transient. Near beginning-of-life (BOL), the natural reactivity feedback mechanisms are weaker relative to middle-of-life (MOL) and end-of-life (EOL) due to a less negative moderator temperature coefficient. This results in higher peak pressures near BOL for a given inlet temperature increase.

Along with the natural reactivity feedback mechanisms, automatic or manual control rod insertion can mitigate the system pressure transient by introducing negative reactivity, further reducing the core power level. The effectiveness of the control rods in reducing the power level is assessed at all cycle burnups through the calculation of burnup dependent critical powers. Calculations are performed assuming a pressure of 3200 psi, a range of inlet temperatures, and D-Bank insertion of 72 steps. The resulting power levels are compared to the critical power trajectory (CPT) curves, generated by Transient Analysis, that yield a peak pressure of 3200 psi. If the calculated critical power for a given burnup is less than the CPT curve value, then that burnup is "favorable." If the calculated critical power is greater than the CPT curve value, then that burnup is "unfavorable." By quantifying the fraction of the cycle that is unfavorable, we obtain the unfavorable exposure time (UET) for the given scenario.

For the most probable plant configurations (e.g., full auxiliary feed, 0 PORVs blocked), cycle burnups beyond the first half of the cycle are generally not limiting since the natural feedback mechanisms are strong enough to limit the peak system pressure to less than 3200 psi. Thus, control rod insertion is primarily a benefit near BOL for these scenarios since the negative reactivity of the control rods augments the natural reactivity feedback, significantly reducing the UET for the cycle. For the reference ATWS scenario (loss of normal feedwater with 0 PORVs blocked and full auxiliary feed), 72 steps of control rod insertion reduces the UET to 0% for the low and high reactivity core models; i.e., the 3200 psig limit is never reached at any time during the cycle.

In the risk-informed approach being proposed, credit for rod insertion is taken based upon the probability of operator action to drive in the control rods or the probability of the automatic rod control system to function properly. When credit for rod insertion is taken in this fashion, only insertion of the lead control bank (Control Bank D) is credited and only one minute of rod insertion is assumed (~72 steps of insertion). The amount of control rod insertion assumed is not event specific; i.e., the same assumptions are made for all ATWS events in evaluating whether the peak pressure limit is met. (With regard to ATWS events caused by mechanical binding of the control rods, it is expected that a sufficient number of rods will insert to provide the equivalent of 72 steps insertion of the lead bank.) Furthermore, the probability of control rod insertion is not a function of the cycle burnup or the burnup of the fuel assemblies in control rod positions. There are currently no specific burnup restrictions or limits on fuel assemblies placed in control rod locations. Control rods are expected to insert properly into all fuel assemblies that meet the generic licensed fuel burnup limit.

Control rod insertion, even the modest amount of rod insertion assumed here, is very effective in reducing the core power level. Credit for rod insertion, within the framework of a risk-informed approach, is justified based upon the high probability that a sufficient number of control rods will insert or that manual rod insertion or automatic rod insertion through the rod control system will be successful.

Issue 10: Regulatory Issue

The NRC is concerned with how plant operation would be regulated with regard to ATWS. The NRC asked how would a plant that tripped early in its cycle and wanted to restart with one power operated relief valve blocked and a main feedwater pump unavailable, as permitted by Tech Specs, be treated from the regulatory perspective.

Response: Limitations on the unavailability of systems important to mitigation of an ATWS event during UETs may be implemented if required to maintain plant safety for higher reactivity core designs. As discussed in Section 7, and based on the PRA results in Section 5, configuration restrictions are not necessary to compensate for large impacts on plant risk related to higher reactivity core designs. It was shown in Section 5 that the impact on plant risk of higher reactivity cores is small. But a concern does exist relative to the impact of higher reactivity core designs on defense-in-depth. To address this, configuration restrictions via plant specific configuration management programs are being proposed.

Section 7 presents and discusses the proposed configuration management program. The objective of the program is to maintain defense-in-depth though out the cycle by managing the plant configuration and, if that cannot be accomplished, to take actions to reduce the probability of an ATWS event. This is done by the following when operating in an unfavorable exposure time:

- Restrict scheduled maintenance activities on the RPS
- Restrict scheduled maintenance activities on AMSAC
- Restrict scheduled maintenance activities on AFW
- Restrict blocking PORVs
- Place the rod control system in automatic control

Details of the approach are provided in Section 7.

Appendix B
Issues and Additional Information Needs Identified by the NRC in the Summary for the NRC/WOG August 23, 2000 Meeting

WOG Responses are Provided for Each Issue and Information Need

Issue 1: The WOG did not address the potential for no auxiliary feedwater (AFW) to be available (e.g., resulting from maintenance and/or equipment failures) as a parameter of the possible plant configurations (i.e., auto or manual rod insertion, power-operated relief valve (PORV) availability, and AFW availability). The inclusion and consideration of no AFW being available would aid the staff in future reviews.

Response: The availability of AFW is considered in the analysis and in the configuration management approach as discussed in the following.

Unfavorable Exposure Time Analysis: UETs are developed for success or failure of rod insertion (72 steps from the lead bank), number of blocked PORVs (0, 1, or 2), and amount of AFW flow to the steam generators. AFW flow level of 100% and 50% are considered. 100% AFW flow is flow from all AFW pumps and 50% AFW flow is half the 100% flow value. UETs from these different conditions, including AFW, are provided in Section 4.

Probabilistic Risk Analysis: The PRA model provided in Section 5 includes AFW flow as an event in the ATWS event tree (see Figure 5-1). Splits between 100%, 50% (<100% but ≥50%), and <50% are provided. These correspond to the UET AFW dependencies. The AFW unavailability values used for these split include component failure probabilities, common cause failure of components, and component unavailability due to test and maintenance.

Configuration Management Approach: The configuration management approach to enhance defense-in-depth is discussed in Sections 6.1 and 7. AFW availability is a key part of the configuration management approach for maintaining a plant configuration corresponding to a favorable exposure. This is shown on Figure 7-1 in the column "AFW Maintenance Acceptable" and also as discussed in Section 7 where it's noted that possible precautionary actions during UET periods can include, among others, limiting activities on the AFW system that result in its unavailability. Eliminating scheduled AFW maintenance and testing activities that result in partial AFW system degradation, as in the unavailability of a single pump, increases the probability that 100% AFW flow will be supplied if an ATWS event occurs.

Conclusion: The AFW plays an important role in the plant configuration management approach to maintain defense-in-depth and is included in the approach to configuration management described in Section 7.

Issue 2: The WOG did not provide any information on the incremental conditional core damage probability (ICCDP) when operating in the various plant configurations. For each core design, calculating the ICCDP for operating in each plant configuration during: 1) the estimated operating time within the configuration's unfavorable exposure time (UET) period (e.g., based on a plant configuration management scheme and considering random failures), 2) operating throughout the UET period (i.e., assuming the plant operates throughout the entire UET period in that configuration), and 3) operating throughout the entire cycle (i.e., assuming the plant remains in that configuration throughout the cycle) would aid the staff in future reviews.

Response: ICCDP calculations have not been completed for the three different core designs for the different plant configurations for the three conditions as requested above. First, it's not possible to estimate the amount of time plants will operate within each configuration. This is a plant specific decision and will be impacted by the requirements to maintain defense-in-depth via a configuration management program. Second, plants cannot operate in a number of these configurations for a significant length of time due to other limitations, primarily the Technical Specification requirements on AFW. The AFW Technical Specification would not allow the AFW system to be degraded for a significant length of time. Typically, 72 hours is the maximum for one AFW train inoperable.

Two plant configuration cases were evaluated to determine acceptable configuration specific operating times based on the ICCDP and the Regulatory Guide 1.177 acceptance guideline of 5E-07. The ICCDP is defined in Reg. Guide 1.177 as:

$$ICCDP = (CCDF - CDF_{baseline}) \times AOT$$

where:

CCDF	=	conditional CDF with the subject equipment out of service
$CDF_{baseline}$	=	baseline CDF with nominal expected equipment unavailabilities
AOT	=	duration of single AOT under consideration (in this case the acceptable configuration specific operating time)

An acceptable configuration specific operating time can be determined based on the acceptance guideline of ICCDP $\leq$ 5E-07.

$$AOT(hr) = (5E\text{-}07 \times 8760\ hr/yr)/(CCDF - CDF_{baseline})/yr$$

Two cases are considered. The first is a bounding case that represents any plant configuration that cannot mitigate the pressure transient. This will be based on the high reactivity core. The second case is for blocked PORVs, which will be provided for both the high and bounding reactivity cores. This second case is identical to those presented in Section 5.1.7.

Case 1: Configurations in which the pressure transient cannot be mitigated (any configuration with a UET)

CCDF = 1.51E-06/hr (CDF given the pressure transient cannot be mitigated. This is taken from Table 5-31/Case 11 and is for the worst time in the cycle for the bounding core. For the bounding core, at the worst time in the cycle, the pressure transient cannot be mitigated even with all equipment available. This value also represents the CCDF for any core in a plant configuration in which the pressure transient cannot be mitigated.)

$CDF_{baseline}$ = 1.70E-07/yr (high reactivity core baseline CDF)

AOT = (5E-07 x 8760 hr/yr)/(1.51E-06/yr – 1.70E-07/yr) = 3269 hr = 0.37 yr

Case 2A: Configurations in which one PORV is blocked – high reactivity core

This turns out to be the same as Case 1 if we consider the time in the cycle a UET exists with a blocked PORV.

CCDF = 1.51E-06/yr (from Table 5-30/Case 5)

$CDF_{baseline}$ = 1.70E-07/yr (high reactivity core baseline CDF)

AOT = (5E-07 x 8760 hr/yr)/(1.51E-06/yr – 1.70E-07/yr) = 3269 hr = 0.37 yr

Case 2B: Configurations in which one PORV is blocked – bounding reactivity core

The CCDF used is the same as Case 1 since the pressure transient cannot be mitigated during the worst time in the cycle regardless of the number of PORVs available.

CCDF = 1.51E-06/yr (from Table 5-31/Case 15)

$CDF_{baseline}$ = 4.69-07/yr (bounding reactivity core baseline CDF)

AOT = (5E-07 x 8760 hr/yr)/(1.51E-06/yr – 4.69E-07/yr) = 4207 hr = 0.48 yr

Conclusion: Although the ICCDPs requested have not been provided, conservative acceptable configuration specific operating times were determined based on the ICCDP and Regulatory Guide 1.177 acceptance guideline of 5E-07. Based on this it is seen that ATWS pressure mitigating components can be unavailable for significant lengths of time.

Issue 3: The number of days within a UET condition for the various plant configurations for the bounding core design was not provided. A chart that identifies the number of days within the UET condition for the various plant parameters for the bounding core design would aid the staff in future reviews.

Response: The UETs for the bounding core are provided in Tables 4-11 to 4-15, 4-28, and 4-29.

Issue 4: Since there are different UETs calculated for plant configurations in which the only parameter that changed is the rod insertion mode (i.e., Auto or Manual), it appears that at least some partial rod insertion is credited in the analyses when in the Auto rod insertion mode. An explanation of how the WOG addresses this parameter in the UET calculations would aid the staff in future reviews.

Response: In the risk-informed approach being proposed, credit can be taken for insertion of the lead bank either through operator action (manual rod insertion) or through actuation of the rod control system when the reactor is in automatic rod control (auto rod insertion). When credit is taken for manual or auto rod insertion, only the lead control bank (D-Bank) is assumed to insert and the credit for insertion is limited to 72 steps (~1 minute of control rod insertion). In the UET calculation, there is no distinction between manual or auto rod insertion. For a given plant configuration, the UET is calculated with and without rod insertion assumed. When control rod insertion is assumed, D-Bank is inserted 72 steps from the all rods out position for the critical power calculation. This has the effect of reducing core reactivity and dropping reactor power to a lower value than would be achieved with no control rod insertion. The net result is a smaller UET.

Issue 5: Based on the meeting discussions, apparently all maximum ATWS pressure calculations were performed with equilibrium xenon levels at 100 percent power, except for the part-power calculations, which actually were 100 percent power with no xenon. The information provided gave UETs and probabilities for these conditions. A table of the peak pressures for the part-power conditions would aid the staff in future reviews.

Response: The probabilistic analysis discussed in Section 5 evaluated the ATWS CDF for all ATWS states which are:

- ATWS State 1: power level <40%, without equilibrium xenon
- ATWS State 5: power level <40%, with equilibrium xenon
- ATWS State 2: power level ≥40%, without equilibrium xenon
- ATWS State 3/4: power level ≥40%, with equilibrium xenon

From this assessment it was determined that ATWS State 3/4 is the largest contributor to core damage frequency; 88% for the low reactivity core, 89% for the high reactivity core, and 93% for the bounding reactivity core (see Table 5-25). Since this state is the dominant contributor to ATWS CDF, peak RCS pressures have been calculated only for this state. The remaining states are small contributors to CDF so they were not considered in the LERF assessment, therefore, peak RCS pressures have not been calculated. Peak RCS pressures are provided in Tables 4-20 and 4-21 only for ATWS events that initiate at 100% power with equilibrium xenon.

Table B-1 provides a summary of the CDF contributors for the low and bounding reactivity cores for each ATWS state. Also shown are the increases in CDF from each state. This again shows that ATWS State 3/4 is the dominant contributor to the increase in CDF, accounting for ~ 95% of the increase.

Table B-1 Core Damage Frequency Results Summary for Low and Bounding Reactivity Cores, Standard Blocked PORV Probabilities, Control Rod Insertion Failure Probability = 0.5

ATWS State	Low Reactivity Core CDF (per yr)	Bounding Reactivity Core CDF (per yr)	ΔCDF (per yr)	Percent of Total ΔCDF (per yr)
1	1.32E-09	1.34E-08	1.21E-08	3.2%
2	1.17E-08	1.36E-08	1.90E-09	0.5%
3/4	1.09E-07	4.69E-07	3.60E-07	94.7%
5	1.57E-09	8.15E-09	6.58E-09	1.7%
Total	1.24E-07	5.04E-07	3.80E-07	100.1%

As stated in Section 5.2.1, LERF values were only calculated for ATWS State 3/4. If it is very conservatively assumed that all the core damage sequences for the other ATWS states proceed to large early release sequences, the increase in LERF, from the low reactivity core to the bounding reactivity core, for the other ATWS states is:

$$\Delta\text{LERF (ATWS States 1, 2, \& 5)} = 1.21\text{E-08} + 1.90\text{E-09} + 6.58\text{E-09} = 2.1\text{E-08/yr}$$

Adding this to the ΔLERF for ATWS State 3/4 from Section 5.2.1, for the sensitivity case that assumes the peak pressures are applicable to 50% of the cycle, the total LERF impact is then:

$$\Delta LERF = 6.05E\text{-}08/yr + 2.1E\text{-}08/yr = 8.2E\text{-}08/yr$$

Even with this conservative approach, the impact on LERF from the low to bounding reactivity core meets the LERF guideline in Regulatory Guide 1.177 (<1E-07/yr defines a small impact), and the conservative estimate for ATWS States 1, 2, and 5 account for approximately 25% of the total ΔLERF. This also shows that the LERF contributions from these states are not significant. Based on this, peak RCS pressures were not calculated for part power conditions, and it was concluded that ATWS risk from part power conditions plays only a small role in ATWS risk and can be neglected in the decision-making process.

Issue 6: It is not clear that the peak ATWS pressures at 100% power with no xenon bound the peak ATWS pressures at lower powers with no xenon, especially since the moderator temperature coefficient (MTC) can be more positive at lower powers (i.e., at 100 percent power MTC is limited by technical specifications to 0 pcm/°F, but at 70% percent power MTC can be as high as +5 pcm/°F). The identification of the part-power levels that produce the bounding peak ATWS pressures for the low, high, and bounding core cases and what bounding pressures are reached would aid the staff in future reviews.

Response: As discussed in the response to Issue 5, the ATWS states without equilibrium xenon are very small contributors to CDF. Also as discussed, if it is assumed that all core damage sequences in ATWS states without equilibrium xenon are assumed to proceed to large early release sequences, the impact on LERF, from the low to bounding reactivity core, meets the LERF guideline in Regulatory Guide 1.177.

Based on this conservative analysis, it was concluded that the RCS pressures for conditions without equilibrium xenon are not necessary and would not provide any benefit in the decision-making process.

Issue 7: For some cores, the peak ATWS pressures would occur at some point in time substantially after the beginning of the fuel cycle, when the different rates of burnable poison depletion, uranium depletion, fission product accumulation and plutonium breeding create the maximum net surplus reactivity. The full power UETs described in the information provide account for these factors. However, it is not clear if the peak pressures from an ATWS during a restart from a forced outage (>3 days) in the period of maximum core reactivity is bounded by the pressures calculated for other conditions. The bounding pressure for the above conditions would aid the staff in future reviews.

Response: As noted in the response to Issue 5, peak pressures have only be calculated and provided for 100% power operation with equilibrium xenon. By the probabilistic risk analysis provided in Section 5, it was shown that the condition of power level ≥40% with equilibrium xenon accounts for the majority of the ATWS CDF and it was concluded from this that LERF analysis and the supporting RCS pressure analysis only needs to be done for these conditions.

Also as discussed in the response to Issue 5, if it is assumed that all core damage sequences in ATWS states without equilibrium xenon are assumed to proceed to large early release sequences, the impact on LERF, from the low to bounding reactivity core, meets the LERF guideline in Regulatory Guide 1.177.

Based on this conservative analysis, it was concluded that the RCS pressures for conditions without equilibrium xenon are not necessary and would not provide any benefit in the decision-making process.

Issue 8: Identifying the initiating event conditions that result in the highest pressures and what pressures are reached would aid the staff in future reviews. Specifically, a table of the pressures reached for the different cores if no AFW, no PORVs, and no rod insertion are available for the initiating event that results in the highest pressure would aid the staff in future reviews.

Response: Tables 4-20 and 4-21 provide the RCS peak pressures for the twelve UET conditions for the low and bounding reactivity cores. These are based on the pressure limiting loss of load ATWS event for a 4-loop W PWR with model 51 steam generators at an uprated power level of 3579 MWt.

Issue 9: The information provided includes distinct values for moderator temperature coefficient (MTC) at 150, 4000, 9000, and 21,512 MWD/MTU. Though helpful, this does not provide insight into the MTC behavior at low power and in-between these four points. A plot of MTC and peak pressure as a function of time for the limiting power level and limiting initiating event while in the UET domain for each case would aid the staff in future reviews.

Response: Figure B-1 gives the HFP moderator temperature coefficients as a function of cycle burnup for the low, high, and bounding reactivity cores. Similarly, Figure B-2 gives the HZP moderator temperature coefficients as a function of burnup. These figures clearly show that the bounding core has the weakest moderator feedback early in the operating cycle. The low reactivity core, on the other hand, has the strongest moderator feedback. As the cycle burnup proceeds and the burnable absorbers deplete, all three cores tend to approach roughly the same moderator coefficient values. Note in Figure B-2 that the bounding reactivity core reaches the +7 pcm/°F limit at a burnup of about 2000 MWD/MTU.

Figure B-1 can be used to approximate the UET for a given ATWS scenario and inlet temperature assumption. For the reference ATWS scenario (all PORVs available, full auxiliary feed, no rod insertion) and an assumed ATWS inlet temperature of 600°F, the "favorable" MTC is approximately -7.5 pcm/°F (see response to the UET/MTC link issue, Appendix A, Issue 6). Thus, the portion of the cycle for which the MTC is less negative than -7.5 pcm/°F is the unfavorable portion of the cycle. Figure B-1 shows that the low reactivity core is always more negative than -7.5 pcm/°F; therefore, for this ATWS scenario and inlet temperature assumption, the UET would be ~0% of the cycle. For the bounding reactivity core, the MTC does not become more negative than -7.5 pcm/°F until a burnup of about 6800 MWD/MTU. Since the EOL burnup is ~22,000 MWD/MTU, this corresponds to a UET of approximately 30% of the cycle. For the high reactivity core, the favorable portion of the cycle occurs for burnups greater than about 4900 MWD/MTU, corresponding to a UET of about 22%. The "favorable" MTC value will depend upon the ATWS scenario being considered (number of PORVs available, etc.) and the inlet temperature assumption.

Figures B-3 and B-4 show the MTC behavior as a function of power level. In Figure B-3, equilibrium xenon was assumed at each power level, while in Figure B-4 no xenon was assumed. (At HZP, no xenon and equilibrium xenon are the same condition.) As these figures show, the MTC is a monotonically decreasing function of power level. Figure B-3 demonstrates that even for the bounding core, the expected MTC at full power with equilibrium xenon is negative (-2.7 pcm/°F was calculated). In Figure B-4, the MTC values are more positive than in Figure B-3 due to the higher critical boron concentrations that result from the no xenon assumption. While Figure B-4 gives MTC values at HFP with no xenon, this is not a realistic condition. Typical power ramp rates are slow enough such that significant xenon build-up prior to reaching full power is expected. If necessary, control rod withdrawal limits can be specified as a function of boron concentration to ensure that the MTC Technical Specification is met. This is consistent with current MTC Technical Specifications.

The plots of peak pressure as a function of time have not been developed. As discussed in Section 5.2, a conservative approach was used in the LERF assessment that assumed the peak pressures are applicable to the full cycle length for each rod insertion, AFW flow, and blocked PORV configuration. The peak pressures are provided in Section 4.3.

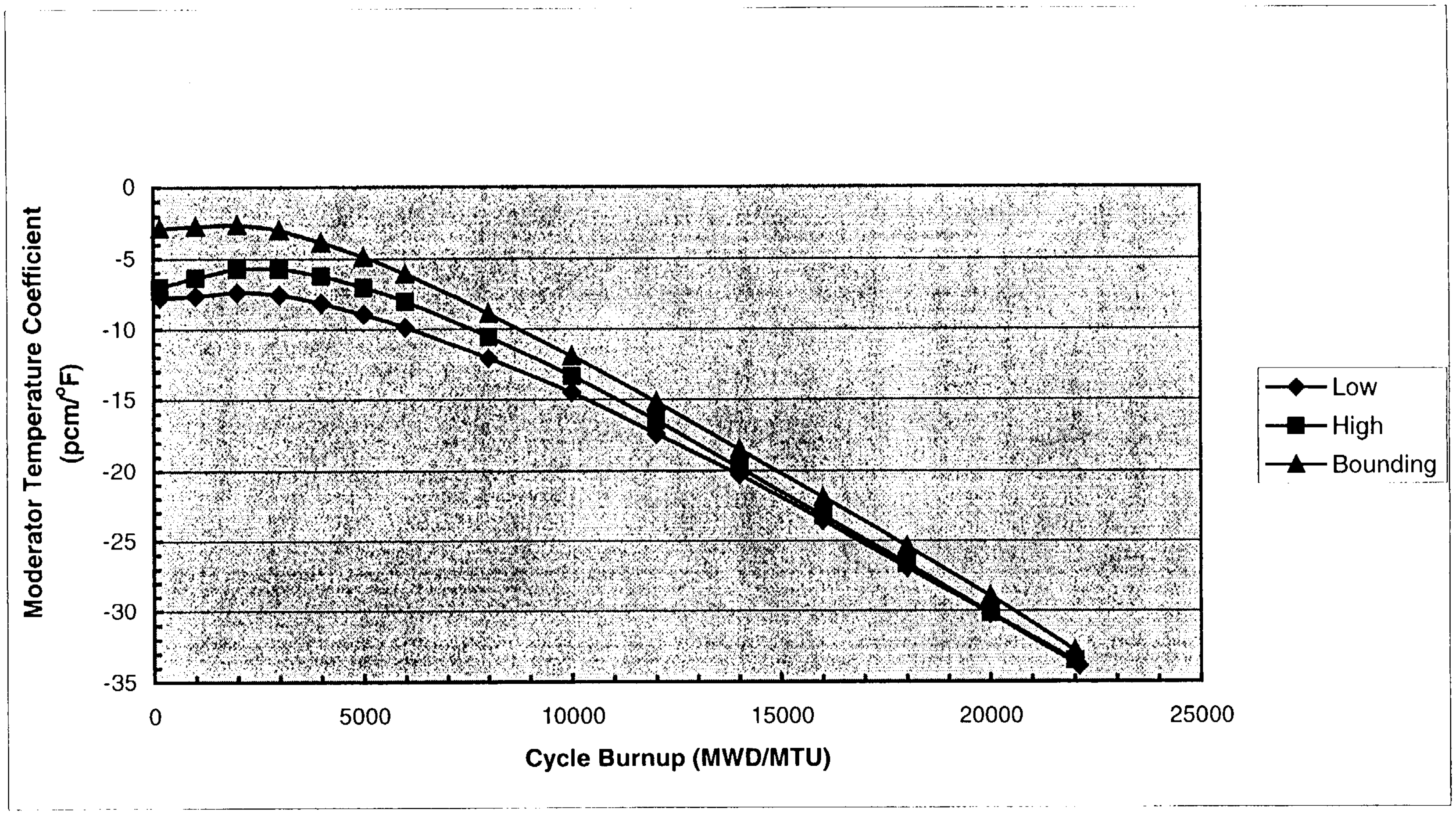

Figure B-1 HFP Moderator Temperature Coefficient versus Cycle Burnup

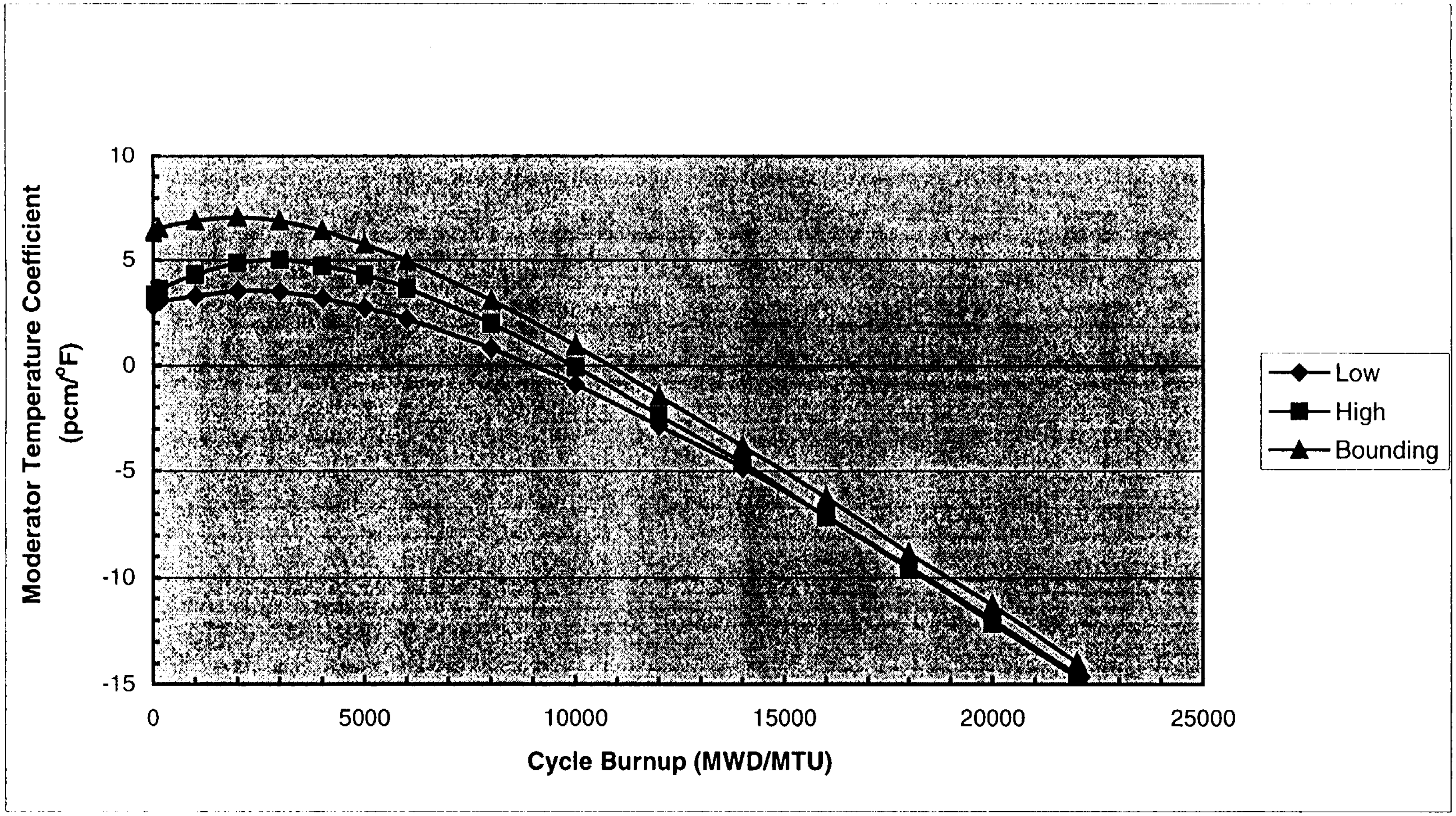

Figure B-2 HZP Moderator Temperature Coefficient versus Cycle Burnup

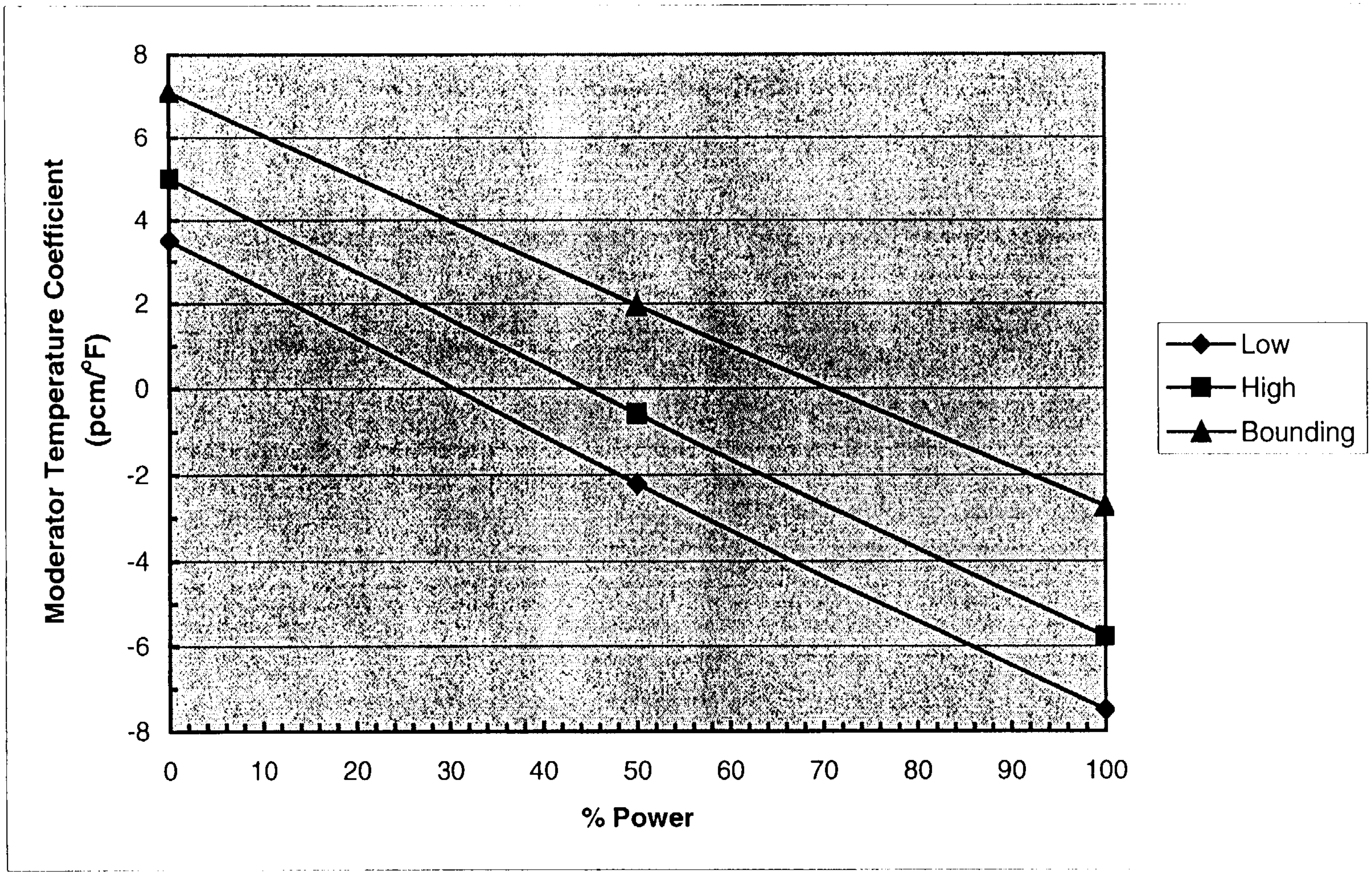

Figure B-3 Moderator Temperature Coefficient versus Power Level Assuming Equilibrium Xenon

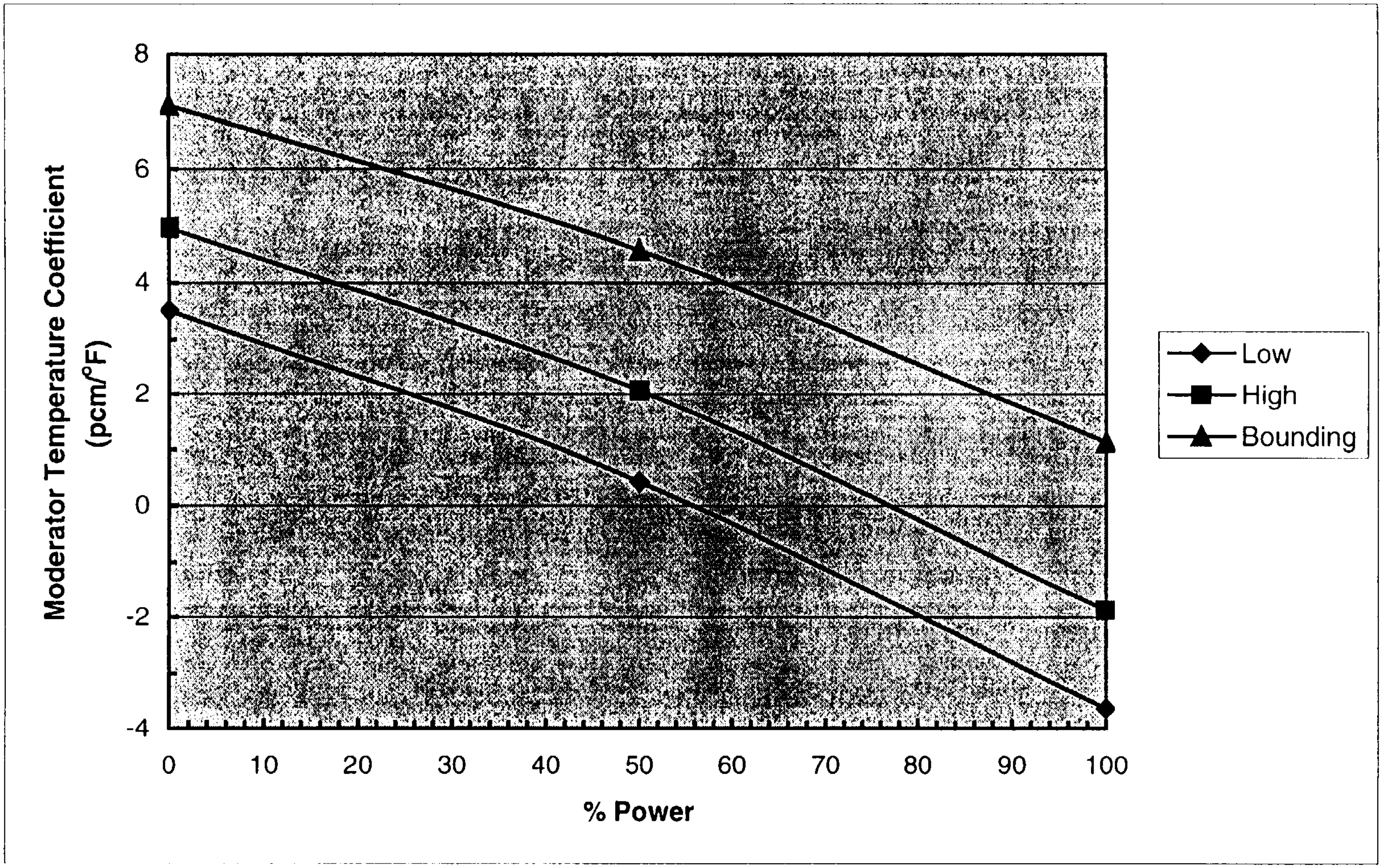

Figure B-4 Moderator Temperature Coefficient versus Power Level Assuming No Xenon

Issue 10: Regulatory Guide (RG) 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," addresses risk-informed approaches. As part of the engineering analysis, the RG indicates that consideration should be given to defense-in-depth and safety margins. One of the conditions for maintaining consistency with defense-in-depth philosophy is "Over-reliance on programmatic activities to compensate for weaknesses in plant design is avoided." The WOG-proposed approach includes compensating for increased UETs (i.e., periods of inadequate plant capability to withstand the peak pressures resulting from an ATWS) that results from using higher reactivity core designs by some form of plant configuration management. The current information provided by the WOG may not result in any limitation on unfavorable MTC, and concomitantly on UET. The WOG needs to justify how their approach satisfies the above condition for maintaining consistency with the defense-in-depth philosophy. The WOG also needs to address how the plant configuration management schemes should be controlled by the utility.

Response: Section 6.1 addresses the impact of the higher reactivity cores on defense-in-depth. Section 6.1 also addresses the elements that comprise defense-in-depth as defined in Regulatory Guide 1.174. Section 7 presents the configuration management approach that will be used to maintain defense-in-depth. It is important to note that the objective of the configuration management program is to operate the plant in a configuration that preserves defense-in-depth. As noted in Section 7, this requirement is not the result of a need to address a large impact on risk. The analysis in Section 5 demonstrates that the risk impact is small and the Section 5 analysis did not credit a configuration management program to show this.

With regard to the issue concerning over-reliance on programmatic activities to compensate for weaknesses in plant design, the following is provided in Section 6.1.

> *The core design will change such that higher RCS pressures will occur if an ATWS event occurs. The magnitude of the RCS pressure will depend on the time in life when it occurs and the availability of pressure relief and AFW, and available reactivity insertion. All safety systems, including the RPS, AFW system, RCS pressure relief capability, and rod control system will continue to function in the same manner with the same reliability, and there will be no additional reliance on additional systems or operator actions. The impact on risk is very small, but depending on the plant configuration, there could be an impact on defense-in-depth. This will be compensated for by plant configuration management programs that improve the preventive aspect or alternate mitigative capabilities as discussed in Section 7.*

Control of the configuration management program will depend on the licensee. It is expected that the program will be incorporated into the plant's program to address (a)(4) of the Maintenance Rule. This requirement states:

> *Before performing maintenance activities (including but not limited to surveillances, post-maintenance testing, and corrective and preventive maintenance), the licensee shall assess and manage the increase in risk that may result from the proposed maintenance activities. The scope of the assessment may be limited to structures, systems, and components that a risk-informed evaluation process has shown to be significant to public health and safety.*

Plants with a UET > 5% for the reference case (no rod insertion, 100% AFW, and no PORVs blocked) will develop a set of acceptable operating configurations that address maintaining defense-in-depth for ATWS events. An example is provided in Figure 7-1. This is also discussed in Section 10 (Reload Implementation Process).

Issue 11: The updated event tree and fault tree models used in support of the analyses were not provided. To aid the staff in future reviews, the updated models, the identification of the dominant cutsets and/or sequences, and the bases and/or references for the failure rates or unavailability values used for the basic events in the models should be provided.

Response: The updated event trees and fault tree models are provided on the enclosed disk. The output files for the base quantifications for the low, high, and bounding reactivity cores are also provided on the disk. The basic event failure rates and unavailability values are provided in Section 5 of the WCAP.

Appendix C
Issues Identified in NRC letter "Westinghouse Owners Group Risk-Informed Anticipated Transient without Scram Approach" (Reference 10)

WOG Responses are Provided for Each Issue

Issue 1: Peak Pressure

Meet ASME Service Level C (3200 psig)

In a PWR, the ATWS transient results in a primary system pressure rise, the magnitude of which is dependent upon the MTC, the primary relief capacity, and how much energy the steam generators can remove. If the pressure cannot be reduced, reactor coolant will be lost through the relief valves and the core will eventually be uncovered. If an ATWS occurs when the MTC is either positive or insufficiently negative to limit reactor power, the ATWS pressure increase will exceed the ASME Service Level C pressure and all subsequent mitigative functions are likely to be ineffective. The proposed WOG approach should address this situation.

Response: In the WOG approach, if the RCS pressure exceeds 3200 psig, core damage is assumed to occur. If the pressure remains below 3200 psig, the RCS remains intact and mitigative functions remain operable. The function of particular interest after the pressure transient has been mitigated is boration. Check valves connecting the boration system to the RCS, and providing part of the RCS pressure boundary, will be exposed to the 3200 psig RCS pressure. The following provides the assessment regarding the operability for these valves following the pressure transient.

An evaluation of the charging line check valves described below has been performed to determine the effect of an ATWS transient on their integrity and operability. The ATWS transient would result in a 3200 psig differential pressure across the disc at a maximum fluid temperature of 550°F with the valves closed. The evaluation included the effect of the transient on the pressure boundary components which consist of the main flange joint, bonnet or cover, and disc to determine if the valve would function following the transient.

Base on the evaluation of the four design configurations that makeup a majority of the installed valves, the valves should operate following the ATWS transient condition. Following the transient some seat leakage may be present which may require the valve be disassembled and inspected.

Following is a description of the four design configurations and with a summary of each evaluation.

1. Westinghouse 3" Model 03000CS88 and 4" Model 04000CS88 (Style A) Swing Check Valves
2. Westinghouse 3" Model 03001CS88 and 4" Model 04001CS88 (Style B) Swing Check Valves
3. Velan 3" Swing Check Valves built to drawing 78409 (B10-3114B-13MS)
4. Velan 3" Swing Check Valve built to drawing 78431

1. Westinghouse 3" Model 03000CS88 and 4" Model 04000CS88 (Style A) Swing Check Valves

These swing check valves were manufactured to the requirements of the ASME Code Section III Class 1 requirements and supplied over a range of applicable Codes of construction. Therefore, the effect of the applicable Codes of construction was considered in the evaluation. Also, the design changes, i.e., main flange bolting hole depth, etc., to the valve model has been taken into account in the evaluation. In addition to the Code of construction and the design changes, the valve main flange bolting remains SA-453 Gr 660 material, the bonnet (cover) is SA-240 Type 316 or SA-182 F316 material and the disc is SA-182 F316 material.

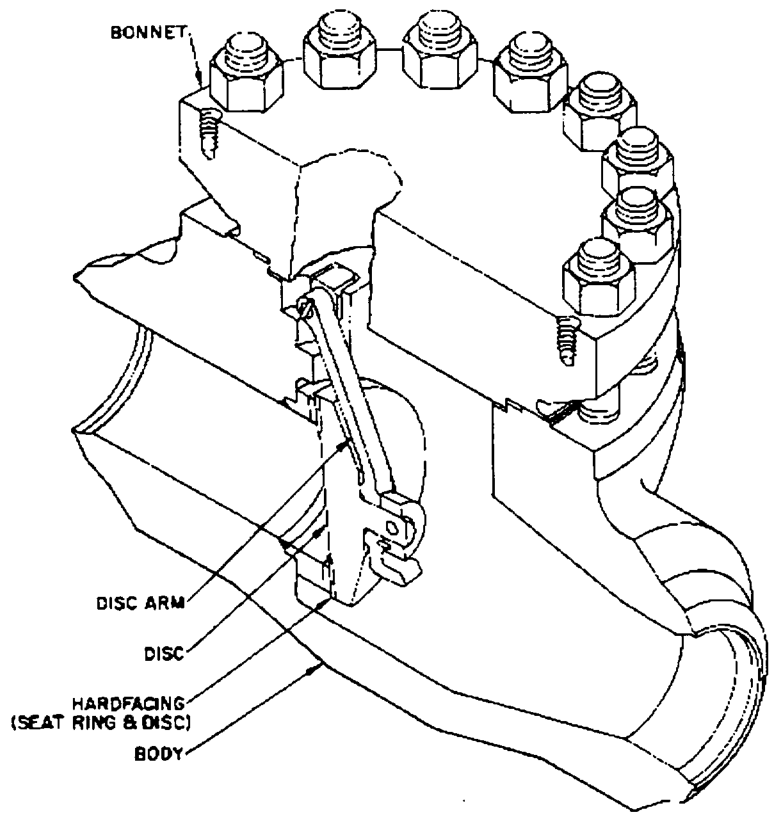

Figure C-1 Westinghouse Style A Swing Check Valve Models 03000CS88 and 04000CS88

Main Flange Joint

The main flange joint was evaluated using the methodology from the original design analysis of the valve that was derived from the ASME Code Section III. This methodology was applied to the analysis of the main flange joint at the ATWS transient conditions to determine if the main flange and main flange bolting stresses were acceptable. The evaluation concluded that the main flange and bolts stresses were less than normal ASME Code Class 1 allowable stresses for the range of Codes of construction.

Bonnet (Cover)

The cover was evaluated using classical methods used in the original design analysis at the ATWS transient conditions. The evaluation concluded that the cover stresses are less than normal ASME Code Class 1 allowable stresses for the range of Codes of construction.

Disc

The disc was evaluated in the closed position using classical methods based on a flat plate theory used in the original design analysis at the ATWS transient conditions. The evaluation concluded that the disc stresses were less than normal ASME Code Class 1 allowable stresses for the range of Codes of construction.

Summary

The results of the evaluation show that the Westinghouse 3" Model 03000CS88 and 4" Model 04000CS88 (Style A) Swing Check Valves should operate following the ATWS transient condition described.

2. Westinghouse 3" Model 03001CS88 and 4" Model 04001CS88 (Style B) Swing Check Valves

These swing check valves were manufactured to the requirements of the ASME Code Section III Class 1 requirements and supplied over a range of applicable Codes of construction. Therefore, the effect of the applicable Codes of construction was considered in the evaluation. Also, the design changes, i.e., main flange bolting hole depth, etc., to the valve model has been taken into account in the evaluation. In addition to the Code of construction and the design changes, the valve main flange bolting remains SA-453 Gr 660 material, the bonnet (cover) is SA-240 316 or SA-182 F316 material and the disc is SA-564 Gr 630 H1150 material.

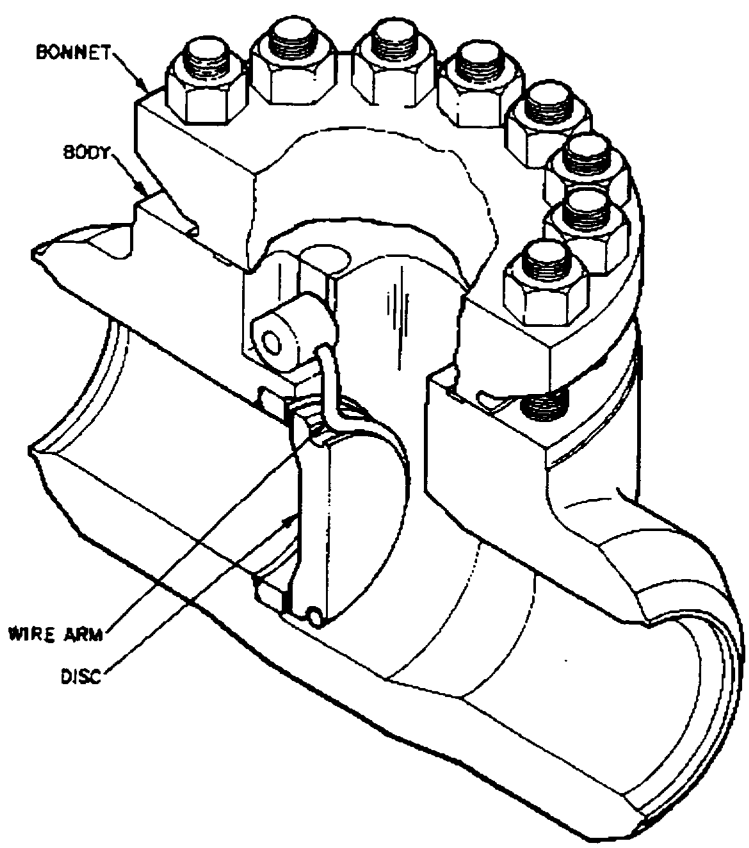

Figure C-2 Westinghouse Style B Swing Check Valve Models 03001CS88 and 04001CS88

Main Flange Joint

The main flange joint was evaluated using the methodology from the original design analysis of the valve that was derived from the ASME Code Section III. This methodology was applied to the analysis of the main flange joint at the ATWS transient conditions to determine if the main flange and main flange bolting stresses were acceptable. The evaluation concluded that the main flange and bolts stresses were less than normal ASME Code Class 1 allowable stresses for the range of Codes of construction.

Bonnet (Cover)

The cover was evaluated using classical methods used in the original design analysis at the ATWS transient conditions. The evaluation concluded that the cover stresses are less than normal ASME Code Class 1 allowable stresses for the range of Codes of construction.

Disc

The disc was evaluated in the closed position using classical methods based on a flat plate theory used in the original design analysis at the ATWS transient conditions. The evaluation concluded that the disc stresses were less than normal ASME Code Class 1 allowable stresses for the range of Codes of construction.

Summary

The results of the evaluation show that the Westinghouse 3" Model 03001CS88 and 4" Model 04001CS88 (Style B) Swing Check Valves should operate following the described ATWS transient condition.

Velan 3" Swing Check Valves (B10-3114B-13MS) built to drawings 78409 and 78431

A review of records show that the swing check valves have had some design and material changes to the pressure retaining parts since the time of initial construction. Major design changes have been made to the disc that could affect the results of this evaluation. These changes consist of design changes to the configuration of the disc and the materials used for its construction. Therefore, the evaluation considered the various disc design evolutions that could be installed in these valves.

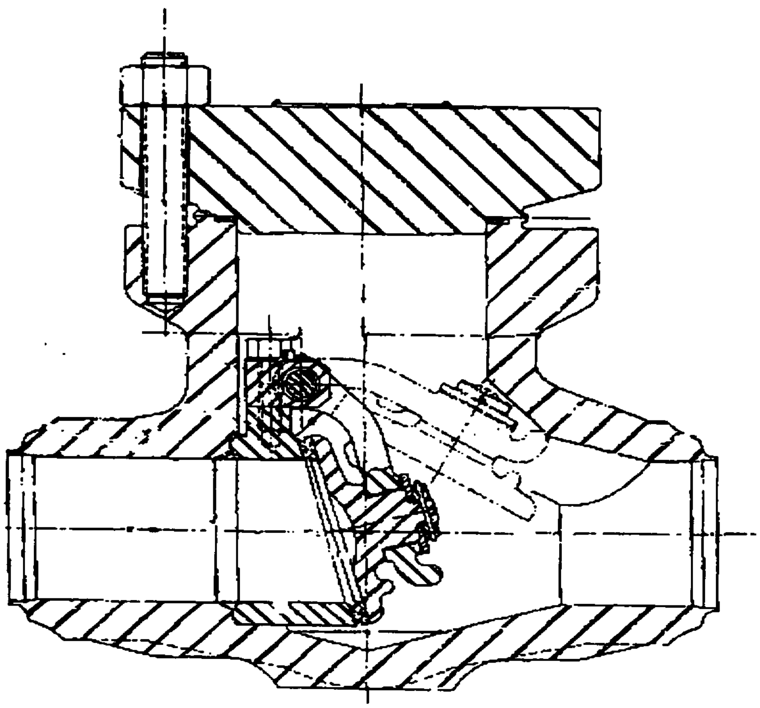

Figure C-3 Velan Swing Check Valve

3. Velan Swing Check Valves built to drawing 78409

These valves were built prior to the use of the ASME Code Section III. They were manufactured to the requirements of ANSI B16.5.

Following are the major components used in this valve design.

Component	Part Number(Drawing)	Material
Body	7904-2-13 (Dwg 79042)	A-182 F316
Cover	8164-5-13 (Dwg 8164-004)	A-182 F316
Disc- Forged*	8204-5-35 (Dwg 8204-15)	A-182 F316
Disc – Cast	8204-5-35 (Dwg 82045)	A-351 CF8 casting pattern 82044E
Main Flange Bolts	9144-2-54 (3/4"-10UNC x 4.25" lg)	A-193 Gr B7

* Forged discs were supplied as replacement parts.

Main Flange Joint

The main flange joint was evaluated using the methodology in the ASME Code Section III for flanged joints at the ATWS transient conditions, and the flange and bolts stresses were less than normal ASME Code allowable stresses.

Cover (Bonnet)

The cover was evaluated using the design rules in the ASME Code Section III for blind flanges at the ATWS transient conditions. The evaluation concluded that the cover stresses were less than normal ASME Code allowable stresses.

Disc

There are various design disc configurations that could be installed in the valves. They include both cast and forged discs.

- Cast Disc

The cast disc identified above was evaluated by both elastic and plastic analysis. The conclusion of the elastic analysis is that, while the stresses in the disc exceed normal allowable stresses at 3200 psig and 550°F, the stresses do not exceed faulted condition allowable stresses at that temperature. Since the stresses exceed normal allowable stresses localized yielding may occur that could result is seat leakage, but the valve should still function as required. The stresses at the plant design conditions do not exceed normal allowable stresses.

The plastic analysis provided further evidence that the cast disc identified above will survive the ATWS transient.

Older revisions of the drawing may be present in some of the valves, but sufficient detail of the actual configuration of the pattern is not available to determine the effect of the pattern changes have on the analysis.

- Forged Disc

The replacement part forged disc was evaluated in the closed position using classical methods based on a flat plate theory at the ATWS transient conditions. The evaluation concluded that the disc stresses were less than normal ASME Code allowable stresses.

Summary

The results of the evaluation show that the Velan swing check valves built to drawing 78409 (B10-3114B-13MS) should operate following the ATWS transient condition described.

4. Velan Swing Check Valves built to drawing 78431

These valves were built to the requirements of the ASME Code Section III.

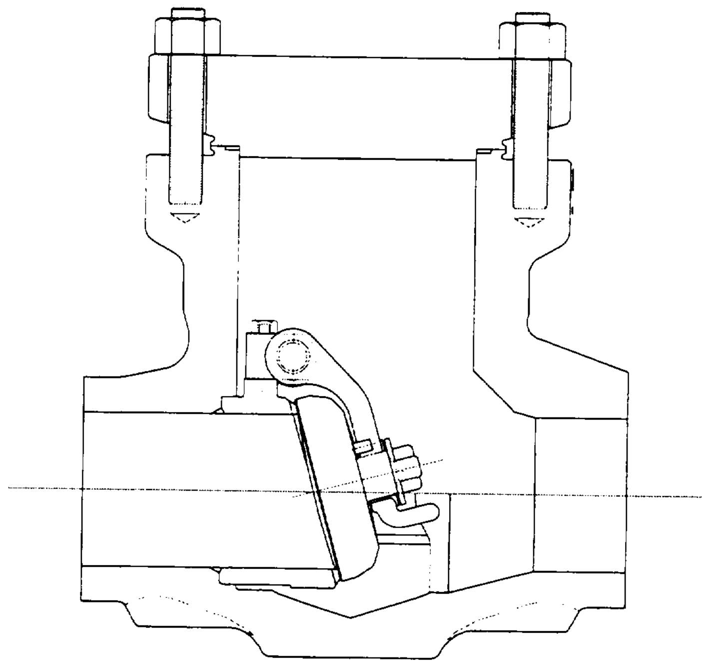

Figure C-4 Velan Swing Check Valve

Following are the major components used in this valve design.

Components	Part Number (Drawing)	Material
Body	8155-19-13 (Dwg 81551)	SA/A 182 F316
Cover	8164-7-13 (Dwg 81647)	SA/A-182 F316
Disc- Casting	8204-7-139 (Dwg 82047 Pattern 82044E)	SA/A-351 CF3M
Disc – Forged*	8204-5-35 (Dwg 8204-15)	A-182 F316

* Forged discs were supplied as replacement parts.

Main Flange Bolts	8244-3-163 (3/4"-10UNC x 4.25" Lg)	SA/A-453 Gr 660

Main Flange Joint

The main flange joint was evaluated using the methodology in the ASME Code Section III for flanged joints at the transient conditions, and the flange and bolts stresses were less than normal ASME Class 1 Code allowable stresses.

Cover (Bonnet)

The cover was evaluated using the design rules in the ASME Code Section III for blind flanges at the transient conditions. The evaluation concluded that the cover stresses were less than normal ASME Code allowable stresses.

Disc

There are various design disc configurations that could be installed in the valves. They include both cast and forged discs.

- Cast Disc

The cast disc identified above was evaluated by both elastic and plastic analysis. The conclusion of the elastic analysis is that, while the stresses in the disc exceed normal allowable stresses at 3200 psig and 550°F, the stresses do not exceed faulted condition allowable stresses at that temperature. Since the stresses exceed normal allowable stresses localized yielding may occur that could result is seat leakage, but the valve should still function as required.

The plastic analysis provided further evidence that the cast disc identified above will survive the ATWS transient.

Older revisions of the drawing may be present in some of the valves, but sufficient detail of the actual configuration of the pattern is not available to determine the effect of the pattern changes have on the analysis.

- Forged Disc

The forged disc was evaluated in the closed position using classical methods based on a flat plate theory at the ATWS transient conditions. The evaluation concluded that the disc stresses were less than normal ASME Code allowable stresses.

Summary

The results of the evaluation show that the Velan swing check valves built to drawing 78431 should operate following the described ATWS transient condition.

Issue 2: MTC/UET

Technical Specification MTC=0 at Beginning of Cycle, Hot Standby, Zero Power

The MTC is a natural process that reduces the core reactivity as the water temperature increases. For a PWR with a negative MTC, an increase in the primary coolant temperature provides negative reactivity feedback to limit the power increase. During the first part of the fuel cycle below 100 percent power, the MTC can possibly be positive for a very short period of time. The MTC is more negative (less positive) at 100 percent power than at lower power. The MTC also becomes more negative (less positive) later in the fuel cycle. When the MTC is insufficient to maintain the primary system pressure below 3200 psig during an ATWS, it is designated in the basis of the ATWS rule as "unfavorable MTC" and in the WOG topical reports the equivalent condition is referred to as an UET. A Westinghouse analysis in December 1979 indicated that the MTC will be more negative than -8 pcm/°F for 95 percent of the cycle time, and more negative than -7 pcm/°F for 99 percent of the cycle time that the core is greater than 80 percent of nominal power. The -7 pcm/°F was determined to be the point at which the core conditions become unfavorable. Under the approach proposed by the WOG, the values of MTC and the doppler coefficient (DC) will have to be carefully examined to ensure that an accident does not result in a situation where the contribution from the MTC and DC effects results in an unacceptable reduction in the margin associated with the total temperature coefficient or results in a net positive reactivity feedback condition.

Response: As part of the Reload Safety Evaluation process, the reactivity coefficients for each reload core are evaluated to ensure that they are within the bounding values assumed in the reference safety analyses. This process is described in WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology." The reload core evaluations include moderator coefficients, Doppler temperature coefficients, and Doppler-only power coefficients. Most negative and least negative limits are checked since, depending on the transient, strong or weak reactivity feedback may be limiting. For plants with positive moderator temperature coefficient technical specifications, the reference safety analyses assume the technical specification limit values for those transients where least negative moderator feedback is limiting (e.g., heatup transients). The MTC values are evaluated for each reload core design to ensure that the MTC technical specification limit bounds the expected MTC values.

Even for cores with positive moderator temperature coefficient technical specifications, the total power coefficient will be significantly negative over the power operating range. For example, total power coefficients were examined for the bounding reactivity core at 2000 MWD/MTU, the cycle burnup where the MTC reached the tech spec limit of +7 pcm/°F. At HFP, equilibrium xenon, nominal conditions, the total power coefficient (TPC) was -9.4 pcm/% power. At HZP, the TPC was -10.5 pcm/% power. Consequently, despite the positive moderator temperature coefficient, the overall reactivity feedback due to an increase in power was negative over the power operating range.

Appendix D
Fault Trees for Primary Pressure Relief with Power Level ≥40% used for CDF Analysis

PRA: Control rod insertion success, 100% AFW
PRB: Control rod insertion success, 50% AFW
PRC: Control rod insertion failure, 100% AFW
PRD: Control rod insertion failure, 50% AFW

The information provided in this appendix is proprietary to Westinghouse Electric Company. Due to the volume of information, it has not been bracketed. The coding associated with this information is "a,c".

Appendix E
Fault Trees for Primary Pressure Relief with Power Level <40% used for CDF Analysis

PR: No control rod insertion, No AMSAC (no AFW)

The information provided in this appendix is proprietary to Westinghouse Electric Company. Due to the volume of information, it has not been bracketed. The coding associated with this information is "a,c".

Appendix F
Fault Trees for Primary Pressure Relief Used for LERF Analysis

PRA: Control rod insertion success, 100% AFW
PRB: Control rod insertion success, 50% AFW
PRC: Control rod insertion failure, 100% AFW
PRD: Control rod insertion failure, 50% AFW

The information provided in this appendix is proprietary to Westinghouse Electric Company. Due to the volume of information, it has not been bracketed. The coding associated with this information is "a,c".