OG-00-086

Project Number 694

August 11, 2000

Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Attention: Chief, Information Management Branch, Division of Inspection and Support Programs

Subject: Westinghouse Owners Group
Transmittal of August 23, 2000 Risk Informed ATWS Plant Model Meeting Material, "Westinghouse Owners Group Response to NRC Risk Informed ATWS Issues", (Proprietary and Non-Proprietary) (MUHP-1033)

This letter transmits one (1) proprietary and one (1) non-proprietary copy of the "Westinghouse Owners Group Responses to NRC Risk Informed ATWS Issues." The following NRC issues were raised at the December 17, 1998 meeting between representatives from Westinghouse, Westinghouse Owners Group and the NRC:

Issue 1: Defense-in-Depth
Issue 2: Large Early Release Frequency
Issue 3: SG Tube Integrity
Issue 4: Component Aging Considerations
Issue 5: Part Power Considerations
Issue 6: UET/MTC Link
Issue 7: Impact on Safety Margins
Issue 8: ATWS with Loss of Offsite Power
Issue 9: Control Rod Insertion
Issue 10: Regulatory Issues

The enclosed material provides the Westinghouse Owners Group response to each of these issues.

Also attached are:

1. One (1) copy of the Application of Withholding Proprietary Information from Public Disclosure, AW-00-1408 (Non-proprietary).

2. One (1) copy of Affidavit AW-00-1408 (Non-proprietary).

3. One (1) copy of the Copyright Notice.

4. One (1) copy of the Proprietary Information Notice

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The attached information, entitled "Westinghouse Owners Group Response to NRC Risk Informed ATWS Issues", contains information proprietary to Westinghouse Electric Company, it is being transmitted with affidavits signed by Westinghouse, the owner of the information. The affidavits set forth the basis on which the information be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of Section 2.790 of the Commission's regulations. Accordingly, it is respectively requested that the information which is proprietary be withheld from public disclosure in accordance with 10CFR Section 2.790 of the Commission's regulations.

Correspondence with respect to the proprietary aspect of the Applications for Withholding or the supporting Westinghouse affidavits should reference AW-00-1408 as appropriate and should be addressed to Mr. H.A. Sepp, Manager, Regulatory and Licensing Engineering, Westinghouse Electric Company, P.O. Box 355, Pittsburgh, PA 15230-0355.

If you require further information, feel free to contact Mr. Ken Vavrek in the Westinghouse Owners Group Project Office at 412-374-4302.

Very truly yours,

Karl Jacobs, Chairman
Westinghouse Owners Group

enclosures

cc: WOG Steering Committee (1L, 1A)
WOG Primary Representatives (1L, 1A)
WOG Analysis Subcommittee Representatives (1L, 1A)
B. Barron, Duke Energy (1L, 1A)
C. Bakken, AEP (1L, 1A)
A. Passwater, AmerenUE (1L, 1A)
S. Bloom, USNRC (1L, 1A)
H. A. Sepp, W - ECE 4-07a(1L, 1A)
A. P. Drake, W - ECE 5-16 (1L, 1A)

Proprietary Information Notice

Transmitted herewith are proprietary and/or non-proprietary versions of documents furnished to the NRC in connection with requests for generic and/or plant-specific review and approval.

In order to conform to the requirements of 10 CFR 2.790 of the Commission's regulations concerning the protection of proprietary information so submitted to the NRC, the information which is proprietary in the proprietary versions is contained within brackets, and where the proprietary information has been deleted in the non-proprietary versions, only the brackets remain (the information that was contained within the brackets in the proprietary versions having been deleted). The justification for claiming the information so designated as proprietary is indicated in both versions by means of lower case letters (a) through (f) contained within parentheses located as a superscript immediately following the brackets enclosing each item of information being identified as proprietary or in the margin opposite such information. These lower case letters refer to the types of information Westinghouse customarily holds in confidence identified in Sections (4)(ii)(a) through (4)(ii)(f) of the affidavit accompanying this transmittal pursuant to 10 CFR 2.790(b)(1).

Copyright Notice

Westinghouse Electric Company LLC

Box 355
Pittsburgh Pennsylvania 15230-0355

July 21, 2000

AW-00-1408

Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Attention: Mr. Samuel J. Collins

APPLICATION FOR WITHHOLDING PROPRIETARY
INFORMATION FROM PUBLIC DISCLOSURE

Subject: "WOG Risk-Informed ATWS Model: Responses to NRC's Comments"

Dear Mr. Collins:

The application for withholding is submitted by Westinghouse Electric Company LLC ("Westinghouse"), pursuant to the provisions of paragraph (b)(1) of Section 2.790 of the Commission's regulations. It contains commercial strategic information proprietary to Westinghouse and customarily held in confidence.

The proprietary material for which withholding is being requested is identified in the proprietary version of the subject report. In conformance with 10 CFR Section 2.790, Affidavit AW-00-1408 accompanies this application for withholding, setting forth the basis on which the identified proprietary information may be withheld from public disclosure.

Accordingly, it is respectfully requested that the subject information which is proprietary to Westinghouse be withheld from public disclosure in accordance with 10CFR Section 2.790 of the Commission's regulations.

Correspondence with respect to this application for withholding or the accompanying affidavit should reference AW-00-1408 and should be addressed to the undersigned.

Very truly yours,

H. A. Sepp, Manager
Regulatory and Licensing Engineering

Enclosures

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

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COUNTY OF ALLEGHENY:

Before me, the undersigned authority, personally appeared H. A. Sepp, Manager, who, being by me duly sworn according to law, deposes and says that he is authorized to execute this Affidavit on behalf of Westinghouse Electric Company LLC ("Westinghouse"), and that the averments of fact set forth in this Affidavit are true and correct to the best of his knowledge, information, and belief:

H.A. Sepp

H. A. Sepp, Manager
Regulatory and Licensing Engineering

Sworn to and subscribed
before me this 3RD day
of August, 2000

Lorraine M. Piplica

Notary Public

Notarial Seal
Lorraine M. Piplica, Notary Public
Monroeville Boro, Allegheny County
My Commission Expires Dec. 14, 2003
Member, Pennsylvania Association of Notaries

(1) I am Manager, Regulatory and Licensing Engineering, in the Nuclear Services Business Unit, of the Westinghouse Electric Company LLC ("Westinghouse"), and as such, I have been specifically delegated the function of reviewing the proprietary information sought to be withheld from public disclosure in connection with nuclear power plant licensing and rulemaking proceedings, and am authorized to apply for its withholding on behalf of the Westinghouse Electric Company LLC.

(2) I am making this Affidavit in conformance with the provisions of 10CFR Section 2.790 of the Commission's regulations and in conjunction with the Westinghouse application for withholding accompanying this Affidavit.

(3) I have personal knowledge of the criteria and procedures utilized by the Westinghouse Electric Company LLC in designating information as a trade secret, privileged or as confidential commercial or financial information.

(4) Pursuant to the provisions of paragraph (b)(4) of Section 2.790 of the Commission's regulations, the following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld.

(i) The information sought to be withheld from public disclosure is owned and has been held in confidence by Westinghouse.

(ii) The information is of a type customarily held in confidence by Westinghouse and not customarily disclosed to the public. Westinghouse has a rational basis for determining the types of information customarily held in confidence by it and, in that connection, utilizes a system to determine when and whether to hold certain types of information in confidence. The application of that system and the substance of that system constitutes Westinghouse policy and provides the rational basis required.

Under that system, information is held in confidence if it falls in one or more of potential competitive advantage, as follows:

(a) The information reveals the distinguishing aspects of a process (or component, structure, tool, method, etc.) where prevention of its use by any of Westinghouse's competitors without license from Westinghouse constitutes a competitive economic advantage over other companies.

(b) It consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), the application of which data secures a competitive economic advantage, e.g., by optimization or improved marketability.

(c) Its use by a competitor would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing a similar product.

(d) It reveals cost or price information, production capacities, budget levels, or commercial strategies of Westinghouse, its customers or suppliers.

(e) It reveals aspects of past, present, or future Westinghouse or customer funded development plans and programs of potential commercial value to Westinghouse.

(f) It contains patentable ideas, for which patent protection may be desirable.

There are sound policy reasons behind the Westinghouse system which include the following:

(a) The use of such information by Westinghouse gives Westinghouse a competitive advantage over its competitors. It is, therefore, withheld from disclosure to protect the Westinghouse competitive position.

(b) It is information which is marketable in many ways. The extent to which such information is available to competitors diminishes the Westinghouse ability to sell products and services involving the use of the information.

(c) Use by our competitor would put Westinghouse at a competitive disadvantage by reducing his expenditure of resources at our expense.

(d) Each component of proprietary information pertinent to a particular competitive advantage is potentially as valuable as the total competitive advantage. If competitors acquire components of proprietary information, any one component may be the key to the entire puzzle, thereby depriving Westinghouse of a competitive advantage.

(e) Unrestricted disclosure would jeopardize the position of prominence of Westinghouse in the world market, and thereby give a market advantage to the competition of those countries.

(f) The Westinghouse capacity to invest corporate assets in research and development depends upon the success in obtaining and maintaining a competitive advantage.

(iii) The information is being transmitted to the Commission in confidence and, under the provisions of 10CFR Section 2.790, it is to be received in confidence by the Commission.

(iv) The information sought to be protected is not available in public sources or available information has not been previously employed in the same original manner or method to the best of our knowledge and belief.

(v) The proprietary information sought to be withheld in this submittal is that which is appropriately marked in the document "Westinghouse Owners Group, Risk-Informed ATWS Plant Model (MUHP-1033), Response to NRC Comments" for Issues 2 and 4. This information is from documents previously developed by Westinghouse to support Westinghouse NSSS RCS component design and operation.

Public disclosure of this proprietary information is likely to cause substantial harm to the competitive position of Westinghouse because it would enhance the ability of competitors to provide similar RCS component evaluations and assessments and licensing defense services for commercial power reactors without commensurate expenses. Also, public disclosure of the information would enable others to use the information to meet NRC requirements for licensing documentation without purchasing the right to use the information.

The development of the technology described in part by the information is the result of many years of experience and effort and the expenditure of a considerable sum of money.

In order for competitors of Westinghouse to duplicate this information, similar technical programs would have to be performed and a significant manpower effort, having the requisite talent and experience, would have to be expended.

Further the deponent sayeth not.

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PROPRIETARY INFORMATION NOTICE

Transmitted herewith are proprietary and/or non-proprietary versions of documents furnished to the NRC in connection with requests for generic and/or plant-specific review and approval.

In order to conform to the requirements of 10 CFR 2.790 of the Commission's regulations concerning the protection of proprietary information so submitted to the NRC, the information which is proprietary in the proprietary versions is contained within brackets, and where the proprietary information has been deleted in the non-proprietary versions, only the brackets remain (the information that was contained within the brackets in the proprietary versions having been deleted). The justification for claiming the information so designated as proprietary is indicated in both versions by means of lower case letters (a) through (f) contained within parentheses located as a superscript immediately following the brackets enclosing each item of information being identified as proprietary or in the margin opposite such information. These lower case letters refer to the types of information Westinghouse customarily holds in confidence identified in Sections (4)(ii)(a) through (4)(ii)(f) of the affidavit accompanying this transmittal pursuant to 10 CFR 2.790(b)(1).

Westinghouse Electric Company LLC

Nuclear Services Business Unit

Box 355
Pittsburgh, Pennsylvania 15230-0355

July 21, 2000

NSBU-NRC-00-5976

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555

Attention:: Mr. Steven D. Bloom

Subject: "WOG Risk-Informed ATWS Model: Responses to NRC's Comments"

Dear Mr. Bloom:

Enclosed are the proprietary and non-proprietary versions of the WOG responses to NRC comments on the WOG risk-informed ATWS model from the NRC/WOG Meeting on December 17, 1998.

Also enclosed are:

1. One copy of the Application for Withholding, AW-00-1408 with Proprietary Information Notice and Copyright Notice.

2. One copy of Affidavit, AW-00-1408.

This submittal contains Westinghouse proprietary information of trade secrets, commercial, or financial information which we consider privilege or confidential pursuant to 10CFR9.17(a)(4). Therefore, it is requested that the Westinghouse proprietary information attached hereto be handled on a confidential basis and be withheld from public disclosure.

This material is for your internal use only and may be used for the purpose for which it is submitted. It should not be otherwise used, disclosed, duplicated or disseminated, in whole or in part, to any other person or organization outside the Office of Nuclear Reactor Regulation without the express written approval of Westinghouse.

Correspondence with respect to the Application for Withholding should reference AW-00-1408, and should be addressed to H. A. Sepp, Manager of Regulatory and Licensing, Westinghouse Electric Company LLC, P.O. Box 355, Pittsburgh, Pennsylvania 15230-0355.

Very truly yours,

HA Sepp

H. A. Sepp, Manager
Regulatory and Licensing Engineering

Enclosures

cc: W. C. Lyon, NRR/DSSA/SRXB (Mail Stop 10 B3)

bcc:	H. A. Sepp (WECe 4-7A)	1L, 1A
	B. M. Smith (WECe 4-7A)	1L, 1A
	G. R. Andre (WECe 469C)	1L, 1A
	K. J. Vavrek (WECe 528)	1L, 1A

Westinghouse Owners Group Response to NRC Risk Informed ATWS Issues

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 core damage frequency (CDF) is small. With respect to anticipated transients without scram (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

Defense-in-depth is an important concept in nuclear power plant design. 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:

1. Prevention: reactor trip with backup operator actions

2. Control and Mitigation: the core physics defense barrier (reactor core and moderator feedbacks)

3. Control and Mitigation: operation of existing systems to limit the potential pressure/temperature transient and provide reactor coolant inventory addition if necessary

<u>Prevention: Reactor trip with backup operator actions</u>

The first level of protection is provided by the reactor protection system (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 and trip the reactor when the trip combinational logic is met. Signals to trip the plant will be generated from at least two sets of channels for every 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 control rod drive mechanisms (CRDMs) from the motor-generator (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 reactor protection system 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 in this instance, 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 control rod system. This action can be 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 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 1980's, to reduce annual plant trip challenges. As plants have matured and efforts to improve plant reliability have been implemented, the number of reactor trips have 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)

A additional barrier in defense-in-depth is related to the design of the core in conjunction with the moderator. The core is designed with the moderator to provide negative 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 that, without a rapid reactor trip, cause the reactor coolant system and core to increase in temperature, such as, loss of feedwater events. The negative reactivity reduces the reactor power and provides the operator time to borate the RCS to bring the reactor to shutdown conditions. This design with negative reactivity feedback provides a "natural" barrier which limits events that could lead to core damage.

Control and Mitigation: Limit potential pressure transient

In addition to core physics, in the defense-in-depth scheme, is mitigation of the pressure transient by the RCS pressure relief system. This consists of pressurizer safety valves and power operated relief valves. For a given core, the pressure transient that will need to be accommodated will depend on the time in cycle, the auxiliary feedwater (AFW) flow rate, and the amount of 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 the ATWS mitigation system actuation circuitry (AMSAC) or the engineered safety feature actuation system (ESFAS) signals. AMSAC is a backup to the ESFAS. Signals from the ESFAS will be available to start AFW and trip the turbine under many, but not all, ATWS scenarios. Table 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 will require 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 moderator temperature coefficient 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 auxiliary feedwater 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) in order to cause a severe pressure transient.

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

Tables 3 and 4 shows unfavorable exposure times (UETs) for low reactivity and high reactivity, 18 month, fuel cycle core designs. The UET indicates the time in life when the pressure transient cannot be mitigated, limited to 3200 psig, for the given conditions. These tables indicate 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 achieve an acceptably low UET with the high reactivity core it is important to maintain PORV availability, AFW availability, and control rod insertion equivalent to 70 steps from the lead bank. The 70 steps from the lead bank corresponds to approximately one minute of bank insertion from the normal operating bank position.

Tables 5 and 6 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 5 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 assumes that the rod control system is in automatic, the PORVs are not blocked, and the AFW system is available (although it may fail due to random or common cause component failures). In this second case, the only contribution to system/component unavailability is from random or common cause 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) 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 5).

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.

Summary

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.

Table 1
Summary of the Capability of Operator Actions to Trip the Reactor for Various Reactor Protection System 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

Nomenclature: MG - Motor generator sets (supply power to the control rod drive mechanisms)
OA - operator action
RPS - reactor protection system

Table 2
Summary of the Capability of Automatic Signals to Actuate Auxiliary Feedwater and Trip the Turbine for Various Reactor Protection System Failures

Failed RPS Element	Actuation Signal		Comments
	ESFAS	AMSAC	
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	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.

Nomenclature: AFW - auxiliary feedwater
AMSAC - ATWS mitigation system actuation circuitry
ESFAS - engineered safety feature actuation system
RPS - reactor protection system

Table 3 Unfavorable Exposure Times for a Low Reactivity Core			
Condition	0 PORVs Blocked	1 PORV Blocked	2 PORVs Blocked
RI, 100% AFW	0 days	0 days	83 days
RI, 50% AFW	0 days	0 days	138 days
No RI, 100% AFW	22 days	236 days	389 days
No RI, 50% AFW	161 days	311 days	443 days

Nomenclature: AFW - auxiliary feedwater
PORV - power operated relief valve
RI - rod insertion (insertion of control rods equivalent to 70 steps from the lead bank

Table 4 Unfavorable Exposure Times for a High Reactivity Core			
Condition	0 PORVs Blocked	1 PORV Blocked	2 PORVs Blocked
RI, 100% AFW	0 days	100 days	154 days
RI, 50% AFW	57 days	121 days	169 days
No RI, 100% AFW	178 days	272 days	411 days
No RI, 50% AFW	214 days	318 days	443 days

Nomenclature: AFW - auxiliary feedwater
PORV - power operated relief valve
RI - rod insertion (insertion of control rods equivalent to 70 steps from the lead bank

Table 5 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 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

The following response addresses two issues raised by the NRC. Both are concerned with the structural integrity of the reactor coolant system (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 systems. 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 psi and a study has been done to show that the RCS will remain intact up to this pressure. It is assumed that a loss-of-coolant accident (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 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 write-up discusses the response of the RCS components to the potential high pressures during an ATWS event. 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 power operated relief valves (PORVs) and safety valves, the auxiliary feedwater (AFW) system, and the rod control system.

A three part approach was taken to address this issue. The first part identifies the most likely ATWS sequences that end with core damage and insufficient pressure relief. The second part calculates the expected RCS pressures corresponding to the most likely sequences identified in Part 1. The third part is a comprehensive examination of the RCS, and interfacing systems and components to determine if they remain intact at the expected RCS pressures, or if missiles are generated that could degrade the containment. Details and results for each part of the assessment are provided in the following sections.

High Pressure ATWS Sequence Endstate Identification
Identification of the most likely sequence endstates is necessary to determine the most likely peak RCS pressure to expect. Calculations were done to determine peak RCS pressures for all conditions and then the RCS systems and components were assessed with regard to failure and missile generation for the maximum pressure, but this maximum pressure will correspond to an extremely low frequency event that is not representative of the most likely RCS pressures.

The core damage sequences from the quantification of the ATWS event tree for the high reactivity core were reviewed and categorized according to the success or failure of systems and actions critical to the RCS peak pressure. This information was then used with the thermal-hydraulic evaluation to determine the most likely expected RCS peak pressures. The systems and actions of interest are the number of PORVs and safety valves available, auxiliary feedwater flow, and rod insertion. For the case being considered, rod insertion (either manually or automatically by the rod control system driving the control rods into the core) success was set at

0.5. Table 1 provides the results of the assessment. It should be noted that all the sequence frequencies are very low. Those that are high, comparatively speaking, usually are associated with successful control rod insertion or have a relatively high level of pressure relief available.

Table 1
Frequency of High RCS Pressure End-states for the High Reactivity Core

Rod Insertion	AFW Capacity (%)	Safety Valves	Number of PORVs	Frequency (per yr)	Comments
Yes/No	<50%	NA	NA	4.7E-09	High pressure end-state, insufficient AFW
Yes/No	≥50%	Adequate Pressure Relief		1.3E-08	Low pressure end-state, failure of long-term shutdown
Yes/No	0%	NA	NA	1.2E-09	High pressure end-state, failure of actuation signals
Yes	100	3	1	7.4E-08	High pressure end-state, insufficient pressure relief
Yes	100	3	0	2.3E-08	High pressure end-state, insufficient pressure relief
Yes	50	3	2	1.5E-08	High pressure end-state, insufficient pressure relief
Yes	50	3	1	7.0E-09	High pressure end-state, insufficient pressure relief
Yes	50	3	0	<1.0E-09	High pressure end-state, insufficient pressure relief
No	100	3	2	1.6E-08	High pressure end-state, insufficient pressure relief
No	100	3	1	5.4E-09	High pressure end-state, insufficient pressure relief
No	100	3	0	1.8E-09	High pressure end-state, insufficient pressure relief
No	50	3	2 or 1 or 0	2.5E-09	High pressure end-state, insufficient pressure relief

ATWS RCS Pressure Assessment
The peak RCS pressure for the loss of load ATWS event was determined for a 4-loop W PWR with Model 51 steam generators at an uprated power level of 3579 MWt. Two higher reactivity cores were considered and are defined as:

- the high reactivity core with a hot full-power moderator temperature coefficient (HFP MTC) of –6.18 pcm/°F
- a bounding core designed to the Technical Specification limit on hot zero-power moderator temperature coefficient (HZP MTC) of +7 pcm/°F (the HFP MTC is equivalent to a –2.9 pcm/°F)

The bounding core is included to show what RCS pressures would be expected for a core designed to the Technical Specification limits. Peak RCS pressures were calculated for ATWS events that initiate from full power with both full and half auxiliary feedwater flow capacity and with varying pressure relief capacities to reflect operation with two, one, or no PORVs available. There is no credit for any control rod insertion as provided by the operators or automatic rod control system driving the rods into the core. The peak RCS pressures are provided on Table 2.

Table 2
ATWS Loss of Load Peak RCS Pressures: 100% Power

Core	HFP MTC (pcm/°F)	Number of PORVs	AFW Capacity (%)	Peak RCS Pressure (psia)
High Reactivity	-6.18	2	100	3222
High Reactivity	-6.18	1	100	3440
High Reactivity	-6.18	0	100	3748
High Reactivity	-6.18	2	50	3301
High Reactivity	-6.18	1	50	3530
High Reactivity	-6.18	0	50	3862
Bounding	-2.90	2	100	3558
Bounding	-2.90	1	100	3846
Bounding	-2.90	0	100	4097
Bounding	-2.90	2	50	3652
Bounding	-2.90	1	50	3998
Bounding	-2.90	0	50	4113

In addition to these full power cases, two part-power cases based on the high reactivity core were analyzed to demonstrate that the peak full-power RCS pressures bound the peak part-power RCS pressures. Initial power conditions corresponding to 85% and 70% full power were considered. For the 85% power case, the MTC was changed to –4.59 pcm/°F and for the 70% power case, the MTC was reduced to –3.08 pcm/°F. AFW flow was assumed to be at 100% and 2 PORVs were assumed to be available. The results are shown in Table 3. This confirms that the full power initial conditions are bounding.

Table 3
ATWS Loss of Load Peak RCS Pressures: Reduced Power Levels

Core	Power Level (%)	MTC (pcm/°F)	Number of PORVs	AFW Capacity (%)	Peak RCS Pressure
High Reactivity	100	-6.18	2	100	3222
High Reactivity	85	-4.59	2	100	2999
High Reactivity	70	-3.08	2	100	2570

Based on the results from Tables 1 and 2, and not crediting the effect of successful control rod insertion which would lower the RCS peak pressures, the peak RCS pressures for the more likely endstates are less than 3750 psia for the high reactivity core. The more likely endstates are defined as those contributing greater than 1.0E-08/yr to the core damage frequency. For the bounding core, the peak RCS pressures for the more likely endstates are less than 4100 psia. Again, this does not credit successful control rod insertion.

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 defined above for the bounding core (4113 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 strophic 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**

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- **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 4 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 4, would remain below recognized ASME Code limits for faulted one-time events. Clearly, the piping will not fail.

Table 4 Stress Intensity in psi for an Applied Pressure Stress of 4100 psi					
Nominal Pipe Size	Schedule				
(inches)	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 5 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 5
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	

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- **Pressurizer**

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- **Steam Generator**

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- **<u>Reactor Vessel</u>**

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- **Reactor Coolant Pumps**

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References

1. CE Report: CENC-1273, "Analytical Report for Georgia Power Company Alvin W. Vogtle Unit No. 1 Reactor Vessel".
2. WCAP-12866, "Bottom-Mounted Instrumentation Flux Thimble Wear", January 1991.
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-431, 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 3: SG Tube Integrity
Current studies have indicated that the steam generator (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 would need to keep in mind and re-visit as necessary.

Response
The NRC identified this as an issue that will need to be addressed in the future if SG tube structural requirements are relaxed such that the 5% tube failure probability is no longer applicable. Thus, no formal response to this specific issue is necessary at this time. It should be noted that degradation of steam generator tubes is discussed within the response to Issue 2, Large Early Release Frequency, and Issue 4 Component Aging Considerations.

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.

It typically takes approximately 50 hours to achieve an equilibrium xenon level, regardless of power level, following a shutdown that was of sufficient length for complete xenon decay. At 24 hours the xenon level has built up to 70% to 90%, depending on the power level, of the full concentration level for that power level.

Xenon buildup is important during the initial period of reactor operation following shutdowns of sufficient length that allow the xenon concentration to deplete to a low enough level so that it does not provide negative feedback. A shutdown of approximately 3 days is sufficient in length to achieve complete xenon decay. Therefore, xenon concentration is an important consideration with regard to ATWS events for reactor startups after any outage that is long enough to allow significant xenon decay. For relatively short shutdowns (hours) the xenon concentration remains sufficiently high so as to eliminate this issue as an ATWS concern during reactor startups.

UETs were calculated for the low and high reactivity cores with no xenon. These are provided on Tables 1 and 2. As expected, there are times during the core life when the exposure time is unfavorable (RCS pressure will exceed 3200 psi). The length of the UET is dependent on rod insertion success, auxiliary feedwater flow, and the number of PORVs available.

Startup ATWS Risk Assessment
An analysis was completed to determine the probability of core damage from an ATWS event on plant restart. The probability of core damage was evaluated for the low reactivity core and the high reactivity core. The ATWS model used is the same that was used for the at-power ATWS risk assessment with modifications as discussed below.

The probability of an ATWS event is dependent on the reliability of the reactor trip system; development of trip signals and insertion of the control rods. A significant number of control rods failing to insert due to either 1) failure to develop a trip signal either automatically or manually or 2) failure of the control rods to drop due to mechanical problems results in an ATWS event. Studies done on the reliability of the reactor trip system assume that the plant is operating at power, and that specified test and maintenance activities demonstrate the operability of the reactor trip system on a periodic basis. The reliability of the reactor trip system for plant startups closely following a reactor trip is significantly higher. That is, a successful reactor trip demonstrates that the reactor trip system is fully operable and its reliability in the following startup is greater than its reliability during typical plant at-power operation when its operability is demonstrated only periodically. A plant startup also exercises the shutdown and control rods; both need to be pulled out of the core via the control rod drive mechanisms (CRDMs). This operation demonstrates their operability. In addition, during plant startup, test and maintenance activities that render parts of the reactor trip system unavailable will not be in progress.

The ATWS risk associated with plant startup needs to consider three types of startups:

1. Startup following refueling

2. Startup following a controlled plant shutdown

3. Startup following a reactor trip

For each of these types of startups, the following conditions will exist, or will be conservatively assumed to exist, in order to simplify the evaluation:

For a type 1 startup, there will be a zero xenon concentration level; control rods and CRDMs are exercised for startup; no test or maintenance activities on the RPS are in progress (an expected condition during startup); and there will have been no recent activities that demonstrated RPS operability other than typical periodic tests.

For a type 2 startup, there will be zero xenon concentration level (it will be conservatively assumed that the shutdown time was long enough for complete xenon decay); control rods and CRDMs are exercised for startup; no test or maintenance activities on the RPS are in progress (an expected condition during startup); and there will have been no recent activities that demonstrated RPS operability other than typical periodic tests.

For a type 3 startup, there will be a zero xenon concentration level (it will be conservatively assumed that the shutdown time was long enough for complete xenon decay); control rods and CRDMs are exercised for startup; no test or maintenance activities on the RPS are in progress (an expected condition during startup); and the reactor trip that caused the shutdown demonstrated RPS operability.

The most conservative startup to evaluate, with regard to the RPS reliability, is one following a refueling outage or one following a controlled shutdown. For these startups the RPS has not been recently actuated, it was not required for the shutdown prior to the startup, and its operability has been demonstrated only by periodic testing. The most conservative startup to evaluate, with regard to unfavorable exposure time, is one following an outage of sufficient duration to allow complete xenon decay. The time of the startup during the cycle is also important since the UETs change as the fuel burns and the reactor trip rate is typically higher during the beginning of the cycle.

Another factor that needs to be considered is the time to return to power and the xenon buildup during this time period. The return to power time for a new core is longer than for a core previously in operation due to restraints imposed by required startup tests, calibrations, and data collection. During this time period, xenon concentration is increasing. Theoretically it is possible to determine unfavorable exposure times for various levels of xenon concentrations that could be used to construct a probabilistic model to determine ATWS risk during startup at any point in life, but the level of effort would be high and the model complex, and such detail is not necessary to respond to this issue.

The approach used in this assessment will be conservative and envelope all startup scenarios. The following assumptions apply:

1. The startup will be assumed to be a rapid startup that will be considered a step change to full power. Therefore, the UETs provided in Tables 1 and 2, for no xenon buildup at full power, will be used. At lower power levels the UETs are expected to be of shorter duration.

2. The time the reactor is down following a reactor trip is assumed to be long enough for complete xenon decay.

3. Equilibrium xenon concentration will be achieved within 50 hours. The full power UETs with equilibrium xenon concentration will be applicable after 50 hours and the ATWS risk is no longer related to startup.

4. The startup will be assumed to follow a shutdown that did not require generation of a reactor trip signal, therefore, the probability of failure of the reactor trip signal is assumed to be the same as the normal at-power probability of failure value since there is no comprehensive testing of the RPS prior to startup.

5. No test or maintenance activities are in progress that cause any part of the RPS to be unavailable.

6. The startup, with the movement of the control and shutdown rods, demonstrates the operability of the control and shutdown rods.

Three cases will be analyzed; one for the high reactivity core and two for the low reactivity core. The worst case for each core, with regard to UETs (without xenon buildup), will be analyzed.

Case 1, Low reactivity core: This case models the low reactivity core at the beginning of the cycle when the reactor trip probability is the highest, such as a startup following refueling. For this case the exposure time is favorable for the conditions of successful rod insertion with 100% or 50% AFW flow and at least one PORV available.

Case 2, Low reactivity core: This case models the low reactivity core during the cycle time period from approximately 71 days to 101 days. During this time period the exposure time is unfavorable except for the conditions of successful rod insertion with 100% AFW and two PORVs available. For this time period the reactor trip probability is lower than during the beginning of the cycle.

Case 3, High reactivity core: This case models the high reactivity core at the beginning of the cycle when the reactor trip probability is the highest, such as a startup following refueling. For this case the exposure time is unfavorable for all rod insertion, PORV, and AFW conditions.

Startup ATWS Event Tree Assessment

The ATWS model used is the same that was used for the at-power ATWS risk assessment with modifications as discussed below.

IE: Probability of reactor trip

The probability of a reactor trip occurring during a reactor startup or during the first 50 hours following startup early in the cycle is determined by considering the probability of a trip during the actual plant startup and then during the following 50 hours.

Probability of reactor trip during plant startup: WCAP-14333 (Reference 1, Section 8.4) collected information from utilities on the probability of a reactor trip during a startup event. This was determined to be 0.088.

Probability of a reactor trip during the 50 hour time period following startup: The previous ATWS work has shown the yearly reactor trip frequency to be 0.85/yr. This work indicated that a reactor trip in the first 30 days of operation following startup is more likely than any following 30 day period by a ratio of 0.134/0.051 = 2.6. Therefore, the probability of a reactor trip in the first 50 hours following startup is:

= 0.85/yr x 2.6 x 50hr/8760 hr/yr = 0.013

Total probability of reactor trip = 0.088 + 0.013 = 0.10

The reactor trip probability during a startup following the initial 30 day period is calculated to be 0.093 using the same approach as above. Since these trip probabilities are essentially the same, the value of 0.1 is used in all three cases.

RT: Reactor trip signal by the reactor protection system
The reactor trip signal unavailability model from the NRC study "Reliability Study: Westinghouse Reactor Protection System, 1984-1995" (Reference 2) is used. This includes credit for tripping the reactor manually in the control room via the reactor trip switch. The test and maintenance activities as unavailability contributors during startup were eliminated since these types of activities will not be scheduled for that time.

OAMG: Operator action to trip the reactor by cutting power to the CRDMs from the motor-generator sets
The following human error probabilities are used:

- 0.5 is used when RT fails due to reasons related to the operator action to trip the reactor in RT in conjunction with logic cabinet or analog channel processing failures – this is a conservative conditional failure probability (conditional on a previous operator action already failing).
- 0.01 is used when RT fails due to reasons not related to failure of the operator action to trip the reactor in RT, that is, due to reactor trip breaker failures.

RI: Action to drive the control rods into the core
It is assumed that there is a 0.5 probability automatic rod insertion will fail (i.e., 0.5 probability the rods are in automatic).

CR: Sufficient number of control rods fall into core to shut down the reactor
The value presented in Reference 2 (1.2E-06/d) assumes normal reactor operation which means the reactor has been at power for some relatively long period of time and the control rods have not been fully exercised since the last startup. In the situation being considered in this analysis, the reactor trip is required within 50 hours of startup when the rods were withdrawn from the core. Therefore, the probability for a sufficient number for control rods failing to insert will be reduced by a factor of 10 to credit the recent movement of control and shutdown rods.

Probability of failing to insert sufficient rods to bring the reactor subcritical is 1.2E-07/d.

Other top events: ESFAS, AMSAC, AFW100, AFW50, LTS
These top events remain the same as used in the base ATWS at-power model.

PR: Availability of primary pressure relief
Three cases are considered; two for the low reactivity core and one for the high reactivity core. The high reactivity core case and one of the low reactivity core cases correspond to the beginning of the fuel cycle. The second low reactivity core case corresponds to a time in the fuel cycle with the worst set of unfavorable exposure times (71 days to 101 days). During this time period the only configuration with a favorable exposure is with rod insertion, 100% AFW, and both PORVs available.

Model Quantification Results
The probability of core damage as quantified in this model represents the core damage probability for a reactor startup following a refueling outage or controlled outage. As noted, three cases are quantified; two for the low reactivity core and one for the high reactivity core.

Case 1: Low Reactivity Core, Startup following refueling
The conservatively estimated core damage probability per startup = 7.4E-09
As compared to a yearly ATWS CDF = 6.5E-08/yr (from the generic ATWS CDF calculations for a low reactivity core)

Case 2: Low Reactivity Core, Startup during most unfavorable time in the cycle
The conservatively estimated core damage probability per startup = 1.4E-08
As compared to a yearly ATWS CDF = 6.5E-08/yr (from the generic ATWS CDF calculations for a low reactivity core)

Case 3: High Reactivity Core, Startup following refueling
The conservatively estimated core damage probability per startup = 3.0E-08
As compared to a yearly ATWS CDF = 1.7E-07/yr (from the generic ATWS CDF calculations for a high reactivity core)
This case represents the most unfavorable time in the cycle.

For the situation of a startup following a reactor trip event, the core damage probability for both cores would be less due to the previous actuation that would have demonstrated that the reactor trip system functioned properly. No estimates of this improvement are provided.

For the situation of a startup later in life, when the UETs for some conditions turn favorable, the core damage probability per startup would be less. The reduction is dependent on the time in life.

These calculations are based on several conservative assumptions of which one of the most limiting is assuming a step change to full power. This leads to the assumption that there is no xenon concentration up to 50 hours when full equilibrium xenon concentration is reached.

Summary
From this assessment the following is concluded:

- The ATWS worst case startup core damage probability for both the low and high reactivity cores is very low.
- The worst case increase in risk from ATWS startup events between the low and high reactivity cores is small.

References
1. "Probabilistic Risk Analysis of the RPS and ESFAS Test Times and Completion Times", WCAP-14333-P-A, Rev. 1, October 1998.

2. "Reliability Study: Westinghouse Reactor Protection System, 1984-1995", NUREG/CR-5500, Vol. 2, December 1998.

Table 1 Unfavorable Exposure Times for a Low Reactivity Core Without Xenon Buildup, 18 Month Fuel Cycle			
Condition	0 PORVs Blocked	1 PORV Blocked	2 PORVs Blocked
RI, 100% AFW	none	24 – 151	0 - 212
RI, 50% AFW	71 - 101	10 – 178	0 - 234
No RI, 100% AFW	0 - 241	0 – 349	0 - 479
No RI, 50% AFW	0 - 276	0 – 400	0 - 490

Notes:
RI – Rod insertion
PORV – Power operated relief valve
AFW – Auxiliary feedwater

Table 2 Unfavorable Exposure Times for a High Reactivity Core Without Xenon Buildup, 18 Month Fuel Cycle			
Condition	0 PORVs Blocked	1 PORV Blocked	2 PORVs Blocked
RI, 100% AFW	0 - 129	0 - 174	0 - 218
RI, 50% AFW	0 - 145	0 - 193	0 - 236
No RI, 100% AFW	0 - 261	0 - 343	0 - 494
No RI, 50% AFW	0 - 301	0 - 377	0 - 494

Notes:
RI – Rod insertion
PORV – Power operated relief valve
AFW – Auxiliary feedwater

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 in the actual core and using realistic feedback mechanisms. 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 psi 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 slightly, 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. 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 which just balances the 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 and high 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 Tin, 3565 MW_{th})
2. Increased Pressure Condition at Nominal Inlet Temperature (3200 psi, 556.6 °F Tin, 3565 MW_{th})
3. Increased Pressure with Higher Inlet Temperature, Moderator Feedback Held Constant (3200 psi, Tin of 580-660 °F, 3565 MW_{th}, moderator feedback same as State 2)
4. Increased Pressure and With Higher Inlet Temperature, Moderator Feedback Included (3200 psi, Tin of 580-660 °F, 3565 MW_{th})
5. Critical Power Condition (3200 psi, Tin of 580-660 °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).

Tables 1 and 2 shows these reactivity balances as well as moderator density coefficients, Doppler temperature coefficients, pressure coefficients, and power coefficients for the high and low reactivity cores, respectively. In this table, the pressure coefficient was calculated using the core keff 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 1 and 2 also provide the HFP MTC at nominal conditions, the calculated critical powers, and the critical power limits for 3200 psi system pressure. These critical power limits correspond to the reference ATWS scenario, which assumes all PORVs available and full auxiliary feedwater.

Tables 1 and 2 illustrate the differences between a high reactivity core and a low reactivity core 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 reactivity core 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 1 illustrates how the critical power varies with HFP MTC. Figure 1 plots the calculated critical powers for the low and high reactivity cores as a function of HFP MTC for both the 580 °F and 620 °F inlet temperature cases. The plotted values come from Tables 1 and 2. Note that both cores follow the same critical power versus MTC trendlines. This means that for a given inlet temperature and HFP MTC, both cores would be expected to have similar critical powers.

Note also, however, that the MTC that yields a "favorable" critical power is different for the two inlet temperatures. For this particular core and for inlet temperatures of 580 °F, the MTC must be more negative than approximately –10.5 pcm/°F to achieve a favorable critical power. For the 620 °F inlet temperature, the "favorable" MTC value is about –7 pcm/°F. Thus, the MTC requirement for a favorable critical power will vary depending on the inlet temperature. Similarly, the MTC requirement will vary depending on the ATWS scenario being considered (number of PORVs available, auxiliary feedwater 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 for 3200 psi 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 1
High Reactivity Core Design
Critical Powers, Reactivity Balance, and Reactivity Coefficients
Reference ATWS Scenario

Initial Conditions and Final Tin	Tin = 580°F				Tin = 620°F				Tin = 660°F	
Cycle Burnup (MWD/MTU)	150	4000	9000	21512	150	4000	9000	21512	150	4000
Initial NSSS Power (MW_{th})	3579	3579	3579	3579	3579	3579	3579	3579	3579	3579
Initial Tin (°F)	556.6	556.6	556.6	556.6	556.6	556.6	556.6	556.6	556.6	556.6
Final Tin (°F)	580	580	580	580	620	620	620	620	660	660
Critical Power										
Critical NSSS Power (MW_{th})	3237	3226	3141	3016	2042	1993	1650	1084	559	428
Critical Power Limit (MW_{th})	3164	3164	3164	3164	2008	2008	2008	2008	429	429
Unfavorable Power (MW_{th})	73	62	-23	-148	34	-15	-358	-925	130	-1
Reactivity Balance (values in pcm)										
Pressure Reactivity	72	69	143	408	72	69	143	408	72	69
Doppler Reactivity	-28	-29	-29	-31	-74	-75	-78	-81	-114	-116
Moderator Reactivity	-167	-163	-311	-832	-778	-813	-1336	-3096	-2428	-2717
Power Reactivity	123	122	197	456	780	820	1270	2770	2470	2764
Net Reactivity Change	0	0	0	0	0	0	0	0	0	0
Reactivity Coefficients										
HFP Nominal MTC (pcm/°F)	-7.1	-6.9	-13.1	-33.7	-7.1	-6.9	-13.1	-33.7	-7.1	-6.9
Pressure Coefficient (pcm/psi)	0.076	0.073	0.151	0.429	0.076	0.073	0.151	0.429	0.076	0.073
Doppler Temp. Coefficient (pcm/°F)	-1.28	-1.30	-1.33	-1.40	-1.26	-1.29	-1.33	-1.39	-1.26	-1.28
Moderator Density Coef. ($\Delta\rho/gm/cm^3$)	0.056	0.055	0.103	0.264	0.084	0.087	0.140	0.307	0.128	0.143
Power Coefficient (pcm/%)	-12.8	-12.3	-16.0	-28.9	-18.1	-18.4	-23.5	-39.6	-29.2	-31.3

Table 2
Low Reactivity Core Design
Critical Powers, Reactivity Balance, and Reactivity Coefficients
Reference ATWS Scenario

Iniitial Conditions and Final Tin	**Tin = 580°F**				**Tin = 620°F**				**Tin = 660°F**	
	150	4000	9000	21512	150	4000	9000	21512	150	4000
Cycle Burnup (MWD/MTU)	3579	3579	3579	3579	3579	3579	3579	3579	3579	3579
Initial NSSS Power (MW_{th})	556.6	556.6	556.6	556.6	556.6	556.6	556.6	556.6	556.6	556.6
Initial Tin (°F)	580	580	580	580	620	620	620	620	660	660
Final Tin (°F)										
Critical Power										
Critical NSSS Power (MW_{th})	3119	3162	3123	3012	1586	1757	1575	1076	<0*	121
Critical Power Limit (MW_{th})	3164	3164	3164	3164	2008	2008	2008	2008	429	429
Unfavorable Power (MW_{th})	-45	-2	-41	-152	-422	-251	-433	-932	<-429	-308
Reactivity Balance (values in pcm)										
Pressure Reactivity	157	108	160	411	157	108	159	412	157	108
Doppler Reactivity	-28	-29	-29	-31	-75	-76	-78	-81	-115	-117
Moderator Reactivity	-335	-242	-345	-840	-1304	-1073	-1448	-3116	-3483	-3257
Power Reactivity	206	163	215	460	1221	1040	1367	2785	3368	3265
Net Reactivity Change	0	0	0	0	0	0	0	0	-74*	0
Reactivity Coefficients										
HFP Nominal MTC (pcm/°F)	-14.1	-10.2	-14.4	-33.9	-14.1	-10.2	-14.4	-33.9	-14.1	-10.2
Pressure Coefficient (pcm/psi)	0.165	0.114	0.168	0.433	0.165	0.114	0.168	0.434	0.165	0.114
Doppler Temp. Coefficient (pcm/°F)	-1.28	-1.29	-1.34	-1.40	-1.28	-1.29	-1.33	-1.39	-1.27	-1.29
Moderator Density Coef. ($\Delta\rho/gm/cm^3$)	0.112	0.081	0.114	0.267	0.139	0.115	0.152	0.309	0.182	0.171
Power Coefficient (pcm/%)	-16.0	-13.9	-16.8	-28.9	-21.8	-20.3	-24.3	-39.7	-33.7	-33.7

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

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 plant risk and defense-in-depth are being addressed in other parts of this program and responses to NRC questions.

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 of assessing 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.

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
During a LOSP event, the motor-generator sets, which provide power to the control rod drive mechanisms (CRDMs), lose their power supply and coast down which interrupts power to the CRDMs. The CRDMs, in turn, release the control rod assemblies which drop into the core. During a LOSP event it is not necessary to generate a reactor trip signal in the reactor protection system to trip the plant. Therefore, the only way for an ATWS event to occur with a LOSP is for the control rods to fail to drop into the core due to mechanical binding of the control rod assemblies or failure of the CRDMs to release when they lose power.

The failure of a sufficient number of control rods to drop due to mechanical binding of the control rod assemblies or failure of the CRDMs to release when they lose power is a highly unlikely event. These are the only possible failure mechanisms that can lead to an ATWS event following a loss of offsite power. The probability of failure of a sufficient number of control rods to drop due to these causes is extremely low and this event (LOSP/ATWS) does not need to be further evaluated to determine its contribution to plant risk, that is, its contribution to risk is very low. But assuming such a highly unlikely event can occur, a conservative PRA assessment was completed to demonstrate the very small potential contribution to plant risk.

Due to the loss of forced reactor coolant flow and subsequent core heatup that occurs following the LOSP event, power in the core decreases as the result of negative reactivity feedback effects. This, combined with the coolant conditions that occur during the transient, preclude this event from reaching the high RCS pressure conditions experienced in the other more limiting loss of feedwater ATWS events. For this event, continued core cooling capability is the concern following the loss of forced reactor coolant flow.

For the lower reactivity cores, previous analyses (Reference 1) have demonstrated that there is sufficient DNB margin, such that no core damage would occur. In the short term, the reactor power would be limited by a combination of negative reactivity additions (Doppler, MTC, and voiding). In the long term, the reactor would be shut down by boration. In addition, decay heat removal, via the auxiliary feedwater (AFW) system, would be required. Since offsite power is unavailable, onsite power (diesel generators) would be required to start and run until offsite power is restored. Core damage would occur from this event if all diesel generators (DGs) failed, all AFW failed, or boration failed. In the absence of new LOSP/ATWS analyses for higher reactivity cores, it is simpler and conservative just to assume that an LOSP/ATWS event goes to core damage for the purpose of this illustration. Based on this, the contribution of LOSP/ATWS events to core damage frequency for a low and high reactivity core can then be conservatively determined as follows.

Low Reactivity Core

The contribution of the LOSP/ATWS event to core damage can be calculated by:

(LOSP IE frequency) x (probability of failure of control rods to insert) x (probability of failure of DGs or AFW or boration)

<u>LOSP IE frequency:</u> The initiating event frequency for a LOSP event was obtained from the WOG PSA Database, Rev. 2. LOSP IE frequencies from PRA models for WOG plants were reviewed. The LOSP value used in this calculation is the midpoint of the range of values.

- LOSP IE Frequency = 4.4E-02/year

Failure of control rods to insert (CR): The value provided in Reference 2 for failure of 10 or more control rods failing to insert is used in this assessment. This is a conservative approach since this represents successful insertion of up to 40 rods (assuming a typical plant has 50 control rods) even though CR has failed. Even with failure of control rod insertion as defined here, a significant negative reactivity insertion has been achieved for most cases which is not credited.

- CR = 1.2E-06

Failure of DGs or AFW or boration: Success will require only one DG, one AFW pump, and one boration path. Consistent with the failure probability values used in other parts of the generic PRA analysis in the RI ATWS model, it will be assumed that boration failure probability is 1E-02. Failure of all DGs and all AFW will contribute little to this value.

Core damage frequency = 4.4E-02/yr x 1.2 E-06 x 1E-02 = 5.3E-10/yr

High Reactivity Core

Since it is conservatively assumed that all LOSP/ATWS events will go to core damage for the high reactivity core, the contribution of the LOSP/ATWS events to core damage frequency is calculated by:

(LOSP IE frequency) x (probability of failure of control rods to insert)
= 4.4E-02/yr x 1.2E-06 = 5.28E-08/yr

Impact on Core Damage Frequency

The impact on core damage frequency from a low reactivity core to a high reactivity core from the LOSP/ATWS event is conservatively estimated to be:

CDF(high reactivity core) – CDF (low reactivity core) = 5.28E-08/yr – 5.3E-10/yr = 5.2E-08/yr

This is a very small impact on core damage frequency and well below the guideline of 1E-06/yr defined as a small impact per Regulatory Guide 1.174 (Reference 3).

Discussion of Key Assumptions

The conclusions reached above were reviewed in light of several of the key assumptions in the evaluation and found to be robust. The key assumptions assessed are:

LOSP frequency: The base evaluation used the mid-point value for the LOSP initiating event frequency. If the upper end value was used, the impact on the core damage frequency between the high and low reactivity cores would increase, but would still be significantly less than the 1E-06/yr guideline for defining a small impact on risk.

Control rods required for shutdown: This evaluation conservatively assumed that 10 or more rods failing to insert into the core would result in an ATWS condition. Detailed analyses were not performed to determine the minimum number of rods that are required to assure reactor shutdown. However, it is recognized that the available control rod worth is a function of both the number of rods that insert and the depth of the insertion, such that rods that fail to fully insert due to mechanical binding contribute to the core power reduction. Detailed analyses were not performed to assess the amount of control rod worth required to assure the reactor power level is sufficiently low to preclude core damage, however, if it is assumed that 20 or more rods failing

to insert is required to result in an ATWS condition, the resultant impact on core damage frequency would decrease by a factor of approximately two.

Short term core cooling capability: This evaluation conservatively assumed that short term heat transfer from the fuel rods would not be sufficient to prevent fuel rod damage (e.g., DNB occurs and results in core damage) for high reactivity cores. Analyses with tools different from those used in Reference 1 show that short term core damage would not occur even if no control rods were inserted into the core. The impact of this is that the core damage frequency for low and high reactivity cores would be the same.

References

1. "ATWS Submittal", NS-TMA-2182, December 30, 1979.

2. "Reliability Study: Westinghouse Reactor Protection System, 1984-1995", NUREG/CR-5500, Vol. 2, December 1998.

3. "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis", Regulatory Guide 1.174, July 1998.

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 (MRI) 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 (for the MRI cases). 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 PORV's blocked and full auxiliary feed), manual rod insertion reduces the UET to 0%, i.e., the 3200 psi 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.

Summary

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 (PORV) 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 unfavorable exposure times (UET) may be implemented if the risk assessment for a new core design shows that such limitations are warranted. As discussed in the following paragraphs, a risk assessment will probably not show the need to impose such restrictions. But restrictions may be necessary in order to demonstrate that defense-in-depth is adequately maintained when a plant is operating during a UET. This is also discussed in the following paragraphs.

Due to the high reliability of the reactor protection system (RPS) and backup systems available to mitigate an ATWS event, the ATWS contribution to plant risk is small across all Westinghouse plants. As plants move to higher reactivity cores, the risk analysis shows that the contribution of ATWS events to plant risk increases, but remains small. ATWS core damage frequency (CDF) contributions for a low and high reactivity core in a typical Westinghouse plant are calculated in the current WOG ATWS program, using the most current reliability estimates for the RPS, to be:

- Low reactivity core ATWS CDF contribution – 6.5E-08/yr
- High reactivity core ATWS CDF contribution – 1.7E-07/yr

The low reactivity core has been defined such that it meets the assumption that the overpressure transient can be mitigated 95% of the time (for the conditions of no rod insertion, full auxiliary feedwater flow, and all pressurizer PORVs available). This represents a UET of 5%, that is, for 5% of the cycle the pressure transient will exceed 3200 psi. The high reactivity core represents a core design with a UET of approximately 35% for the previously noted conditions. Note that full (or 100%) auxiliary feedwater flow is the total available from all AFW pumps.

Note: This analysis assumes the following with regard to availability of mitigating systems.
Rod insertion failure probability (by either the automatic rod control system or operator action, following failure of reactor trip) = 0.5
Probability of one PORV blocked = 0.2
Probability of both PORVs blocked = 0.05
Typical reactor protection system (RPS) and auxiliary feedwater system (AFW) test and maintenance unavailabilities

Even though the annual impact of core reactivity on CDF is small, the actual impact on CDF will vary during plant life, dependent on a number of variables including the availability of the RPS, AFW pumps, PORVs, and the time in core life. The time in life is important since the UETs occur early in life and the critical powers become more favorable later in core life. The following provides the ATWS CDF contributions, calculated under the same assumptions as listed above, but assuming core conditions representative of early in life and late in life for the high reactivity core on a yearly basis:

- High reactivity core ATWS CDF contribution early in life – 5.4E-07/yr
- High reactivity core ATWS CDF contribution late in life – 3.6E-08/yr

Neither of these CDF values represent a high contribution to total plant core damage frequency.

Even though the impact on risk as measured by CDF is small, the issue of maintaining defense-in-depth features of licensed nuclear power plants is important. This issue is discussed in detail in the WOG's response to Issue 1, Defense-in-Depth. Tables 1 and 2 provided the UETs for the low and high reactivity cores. For the low reactivity core there are a number of configurations the plant can be operated in which result in a 0 UET. These are for successful partial rod insertion, one or both PORVs available, and at least 50% (of total available) AFW flow. For the high reactivity core there is one plant configuration in which the UET is 0. This is for successful partial rod insertion, both PORVs available, and all AFW available. These are the conditions under 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 allowed outage time and Maintenance Rule requirements), and with the automatic 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 Tech Spec and Maintenance Rule requirements).

By controlling the plant operating configuration plants can maintain defense-in-depth capabilities. Plants with high reactivity cores 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 might include the following:

- Operate with the rod control system in the automatic mode
- Limit blocking pressurizer PORVs
- Limit activities on the AFW system and RPS that results 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 UETs are favorable.

The response to Issue 1, Defense-in-Depth, discusses this issue further. Tables 5 and 6 of the Issue 1 response shows the plant configuration probabilities for two different plant configuration management schemes. The first corresponds to the conditions listed above and the second incorporates restrictions that increase the availability of the PORVs and the AFW system and, increases the probability of operating with the rod control system in automatic. This indicates that through plant configuration management, the probability of operating a plant under non UET conditions can be increased.

This response does not suggest that these restrictions be added to the plant Technical Specifications, but that they be contained in guidance documents outside of plant regulatory documents. Since there may be times and good reasons for operating the plant in a manner that is inconsistent with this guidance, this guidance should not become part of the plant's licensing basis, but represent good practices.

Table 1
Unfavorable Exposure Times for a Low Reactivity Core

Condition	0 PORVs Blocked	1 PORV Blocked	2 PORVs Blocked
RI, 100% AFW	0 days	0 days	83 days
RI, 50% AFW	0 days	0 days	138 days
No RI, 100% AFW	22 days	236 days	389 days
No RI, 50% AFW	161 days	311 days	443 days

Nomenclature: AFW - auxiliary feedwater
PORV - power operated relief valve
RI - rod insertion (insertion of control rods equivalent to 70 steps from the lead bank)

Table 2
Unfavorable Exposure Times for a High Reactivity Core

Condition	0 PORVs Blocked	1 PORV Blocked	2 PORVs Blocked
RI, 100% AFW	0 days	100 days	154 days
RI, 50% AFW	57 days	121 days	169 days
No RI, 100% AFW	178 days	272 days	411 days
No RI, 50% AFW	214 days	318 days	443 days

Nomenclature: AFW - auxiliary feedwater
PORV - power operated relief valve
RI - rod insertion (insertion of control rods equivalent to 70 steps from the lead bank)