WCAP-16084-NP
Project 694

November 12, 2003

WOG-03-601

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

Attn: Chief, Information Management Branch
Division of Program Management

Subject: Submittal of WCAP-16084-NP, "Development of Risk-Informed Safety Analysis Approach and Pilot Application"

Transmitted herewith are four (4) non-proprietary copies of the subject topical report. The Westinghouse Owners Group (WOG) is submitting WCAP-16084-NP, Rev 0, under the Nuclear Regulatory Commission (NRC) licensing topical report program.

The Risk-Informed Safety Analysis (RISA) approach presented in WCAP-16084-NP addresses the classification of specific transients and accidents into realistic event categories by considering the frequency of occurrence of the overall event combination (i.e., the initiating event in combination with coincident occurrences and a single failure). Correspondingly, the acceptance criteria proposed for use in assessing conformance to regulatory criteria would be those associated with the realistic overall event frequency rather than the unrealistically higher frequency of the initiating event alone.

Consistent with the Office of Nuclear Reactor Regulation, Office Instruction LIC-500, "Processing Requests for Review of Topical Reports," the WOG requests that the NRC document the acceptance of this report for review, establish a target date for requesting any additional information, and the planned date for issuance of the Safety Evaluation (SE). Also requested is an estimate of the staff hours needed to complete the review of WCAP-16084-NP.

The South Texas Project Units 1 and 2 are used in WCAP-16084-NP to provide an example pilot application of the RISA approach. It is expected that the South Texas Project, as well as other participating utilities, will be referencing WCAP-16084-NP on their respective dockets in support of licensing actions. To this end, the WOG respectfully requests that the NRC complete its review and issue its SE no later than June 2004.

D048

All correspondence and invoices related to the review of WCAP-16084-NP should be addressed to:

Mr. Gordon Bischoff
Manager, Owners Group Program Management Office
Westinghouse Electric Company
Mail Stop ECE 5-16
P.O. Box 355
Pittsburgh, Pennsylvania 15230-0355

If you require further information, please contact Mr. Jim Molkenthin in the Owners Group Program Management Office at (860) 731-6727

Sincerely,

Frederick. P. "Ted" Schiffley, II, Chairman
Westinghouse Owners Group

Enclosures

xc: Management Committee
Analysis Subcommittee
Licensing Subcommittee
Project Management Office
C. B. Brinkman, (W), w/o enclosures
M. C. Jacob, (W), w/o enclosures
J. S. Galembush, (W), w/o enclosures
V. A. Paggen, (W), w/o enclosures
S. Dembek, NRC, w/o enclosures
D. G. Holland, NRC, w/ enclosures (via Federal Express)

Westinghouse Non-Proprietary Class 3

WCAP-16084-NP
Revision 0

September 2003

Development of Risk-Informed Safety Analysis Approach and Pilot Application

WOG Task MUHP1095
CEOG Task 2076

Westinghouse

Westinghouse Non-Proprietary Class 3

WCAP-16084-NP, Rev. 0

Development of Risk Informed Safety Analysis Approach And Pilot Application

September 2003

Author: _____
Mathew C. Jacob
Reliability & Risk Assessment Group

Author: _____
Joseph R. Rezendes
Transients & Setpoints Analysis Group

Reviewer: _____
Robert E. Jaquith
Reliability & Risk Assessment Group

Reviewer: _____
Michael L. Howard
Transients & Setpoints Analysis Group

Approved: _____
Gary A. Brassart
Reliability & Risk Assessment Group

EXECUTIVE SUMMARY

Design basis analyses documented in Safety Analysis Reports (SARs) are performed deterministically using the guidance provided in Regulatory Guide 1.70 (Reference 1) and the Standard Review Plan (NUREG-0800, Reference 2). Regulatory Guide 1.70 suggests the use of risk informed safety analysis (RISA) for plant design basis events. It provides the guidance that the consequences of higher frequency events be evaluated against more stringent acceptance criteria in contrast to the guidance for the consequences of lower frequency events, which can be weighed against less restrictive acceptance criteria. The regulatory guidance also requires the use of coincident occurrences (COs) and single failures (SFs) in the analysis of the initiating event. When the frequency of the initiating event is combined with those of the CO and SF, the combined (or overall) frequency would be orders of magnitude lower than that of the design basis event defined by the initiating event alone. However, current regulatory guidance does not adequately take into consideration this impact when acceptance criteria are specified for an initiating event in combination with a CO and SF. Because the impact of the lower frequency combined event (i.e., IE+CO+SF) is measured against the same acceptance criteria as for the higher frequency event, there exists an inconsistency in terms of the real risk to public health and safety.

Unlike years past, the Nuclear Regulatory Commission (NRC) and the nuclear industry now have sufficient staff members adequately trained in risk assessment methodology and its application to the design, operation and maintenance of nuclear power plants. It is, therefore, appropriate to now assess the RISA approach for use in future safety analyses. Consequently, the RISA Project was conceived and implemented to address the use of risk-informed considerations in conjunction with deterministic safety analyses. This is accomplished by:

(1) developing a risk-informed approach for event frequency re-categorization that considers events by their overall frequency and not just by that of the initiating event,
(2) identifying a new event to replace the event being re-categorized,
(3) analyzes the events, exactly as done today, using traditional NRC-approved deterministic analysis methods,
(4) evaluates the consequences of the events using current regulatory acceptance criteria chosen, however, to be consistent with the re-categorized overall event frequency and not simply the initiating event frequency, and
(5) applying the above approach to a pilot plant and an example design basis event scenario.

With regard to item (5), the viability and feasibility of the RISA approach in appropriately re-classifying design basis events and evaluating the results against applicable acceptance criteria were demonstrated by applying the approach to the Loss of Normal Feedwater (LONF) event analyzed for the South Texas Project plants. The results of the study suggest that (1) the LONF event in combination with a loss of offsite power (LOOP) and a failure of the Engineered Safety Features (ESF) signal has an overall frequency of less than 2.3E-07 per year, and (2) the acceptance criteria applicable for this event are those for the Limiting Fault 2 event category, instead of the current Moderate Frequency event category. Since the LONF in combination with a CO and SF was re-classified to the Limiting Fault 2 event category, a new Moderate

Frequency event would be defined to replace it. For the purpose of the current study, the definition of a replacement event and associated evaluation were not performed since they do not involve any new or different approaches from those traditionally used.

The results of the current study indicates that by applying the RISA approach, the categorization of plant transients and accidents can be accomplished more rigorously and systematically. In particular, the use of the approach allows the classification of specific transients and accidents into their proper (i.e., more realistic) event categories by considering the overall frequency of occurrence of the event combination, i.e., initiating event in combination with the CO and SF. Correspondingly, the acceptance criteria for the event combination generally becomes less restrictive than those currently used on the basis of the unrealistically higher frequency of the initiating event alone. The use of the RISA approach can lead to meeting the appropriate acceptance criteria for plant transients and accidents more readily, while maintaining the risk to public health and safety at a very low level consistent with the NRC's regulatory mandate. Since the RISA approach is independent of the nuclear steam supply system and/or fuel vendor, Westinghouse requests NRC acceptance of the RISA approach described herein for application in licensing actions for all plant and/or fuel types and for all design basis events.

TABLE OF CONTENTS

TABLES

ACRONYMNS

AFW	Auxiliary Feedwater
AFWS	Auxiliary Feedwater System
ANS	American Nuclear Society
ANSI	American National Standards Institute
AOO	Anticipated Operational Occurrence
AOR	Analysis of Record
CE	Combustion Engineering
CFR	Code of Federal Regulations
CO	Coincident Occurrence
DBE	Design Basis Event
DNB	Departure from Nucleate Boiling
DNBR	Departure from Nucleate Boiling Ratio
ECCS	Emergency Core Cooling System
ESF	Engineered Safety Features
GDC	General Design Criteria
IE	Initiating Event
LF	Limiting Fault
LOCA	Loss-of-Coolant Accident
LOESF	Loss of Engineered Safety Features signal
LONF	Loss of Normal Feedwater
LOOP	Loss of Offsite Power
NRC	Nuclear Regulatory Commission
OG	Owners Group
PC	Plant Condition
PORV	Power Operated Relief Valve
PRA	Probabilistic Risk Assessment
PSE	Public Service Electric
PSV	Primary Safety Valve
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RISA	Risk Informed Safety Analysis
RRA	Reliability and Risk Analysis
SAFDL	Specified Acceptable Fuel Design Limit
SAR	Safety Analysis Report
SBDG	Standby Diesel Generator
SBLOCA	Small Break LOCA
SF	Single Failure
SRP	Standard Review Plan (NUREG-0800)
STP	South Texas Project
TMI	Three Mile Island
UFSAR	Updated Final Safety Analysis Report
USNRC	United States Nuclear Regulatory Commission
WOG	Westinghouse Owners Group

1.0 INTRODUCTION

Design basis analyses documented in Safety Analysis Reports (SARs) are performed deterministically using the guidance provided in Regulatory Guide 1.70 (Reference 1) and the Standard Review Plan (NUREG 0800, Reference 2). Regulatory Guide 1.70 suggests the use of risk informed safety analyses for plant design basis events. It provides the guidance that the consequences of higher frequency events be evaluated against more stringent acceptance criteria in contrast to the guidance for the consequences of lower frequency events, which can be weighed against less restrictive acceptance criteria. The underlying principle of this guidance is that the risk to public health and safety from postulated plant transients and accidents, which is a function of the event frequency and consequences, should be maintained at a very low level.

The regulatory guidance also requires the use of coincident occurrences (COs) and single failures (SFs) in the analysis of the initiating event. When the frequency of the initiating event is combined with those of the CO and SF, the combined (or overall) event frequency would be orders of magnitude lower than that of the design basis event (DBE) defined by the initiating event alone. However, current regulatory guidance does not adequately take into consideration the significance of this impact when acceptance criteria are specified for an initiating event in combination with a CO and SF. Because the impact of the lower frequency combined event (i.e., IE+CO+SF) is measured against the same acceptance criteria as for the higher frequency initiating event alone, there exists an inconsistency in terms of the real risk to public health and safety.

In the late 1970's, Combustion Engineering risk-informed the safety analyses for two plant designs: St. Lucie Unit 2 and the CE Standard Plant design - System 80. Design basis events were categorized into five (5) groups based on the frequency of occurrence and the corresponding acceptance criteria were established consistent with the guidance of Regulatory Guide 1.70 (Reference 1) and the Standard Review Plan (SRP) (Reference 2). Complete safety analyses performed using this categorization scheme and acceptance criteria were documented in the SARs for these designs. The SARs were submitted to the Nuclear Regulatory Commission (NRC) for review. Although supportive of the approach used in performing these safety analyses, the NRC was not fully prepared to actually review such risk-informed applications and would have required significantly longer time periods for completing the safety evaluations of the SARs, thus potentially causing licensing delays for the designs involved. Therefore, the risk-informed safety analyses for these plant designs were withdrawn and replaced with SARs containing conventional deterministic safety analyses.

Unlike years past, the NRC and the nuclear industry now have sufficient staff members adequately trained in risk assessment methodology and its application to the design, operation and maintenance of nuclear power plants. It is, therefore, appropriate to now assess the RISA approach for use in future safety analyses. The RISA Project was conceived and implemented to address the use of risk-informed considerations in conjunction with traditional deterministic safety analyses. This is accomplished by:

(1) developing a risk-informed approach for event frequency re-categorization that considers events by their overall frequency and not just by that of the initiating event,
(2) identifying a new event to replace the event being re-categorized,
(3) analyzing the events exactly as done today, using traditional NRC-approved deterministic analysis methods,
(4) evaluating the consequences of the events using current regulatory acceptance criteria chosen, however, to be consistent with the re-categorized overall event frequency and not simply the initiating event frequency, and
(5) applying the above approach to a pilot plant and an example design basis event scenario.

Since the RISA approach is independent of the nuclear steam supply system and/or fuel vendor, Westinghouse requests NRC acceptance of the RISA approach described herein for application in licensing actions for all plant and/or fuel types and for all design basis events.

2.0 RISA APPROACH

The governing principle of the RISA approach can be stated in one simple word 'consistency'. That is, the risk associated with a plant transient or accident is a function of the overall frequency of occurrence and the measure of its consequences when compared to consistently selected regulatory acceptance criteria. The conventional deterministic approach in current use is to the contrary 'inconsistent' in its application and, therefore, cannot by definition present an accurate measure of the risk to the public health and safety. Current regulatory guidance stipulates that the risk to public health and safety as a result of transients and accidents be maintained at a very low level. This guidance, therefore, implies that the consequences of a high probability event should be small. Conversely, the consequences of low probability events can have a higher acceptance threshold in comparison to higher probability events.

As indicated in Table 2-1, the current regulatory guidance (References 1 and 2) groups plant transients into three (3) broad classes:

1. Moderate Frequency Events,
2. Infrequent Events,
3. Limiting Faults

References 1 and 2 also provide the guidance that each initiating event should also consider coincident occurrences along with the initiating event (e.g., loss of offsite power, iodine spike, etc.) and the single failure (SF) of an active component or system (e.g., loss of a Diesel Generator, loss of a HPSI pump, etc.). The regulatory guidance does not, however, adequately account for the significant change in the overall frequency that results when the initiating event is combined with a CO or SF, since the only change in the acceptance criteria is the allowance for fuel failure for initiating events "in combination with any single active component failure, or single operator error...". No loss of function of any fission product barrier other than the fuel cladding is allowed. In addition, the accidents are grouped into a single category, namely, Limiting Fault, which is shown to have a wide frequency range (1.0E-6 to 1.0E-2 per year).

In contrast, the RISA approach would re-classify the Design Basis Events (DBEs), including those with COs and/or SFs, and their corresponding (i.e., congruent) acceptance criteria by the overall event frequency. Additionally, when a DBE is re-categorized to a lower frequency class (e.g., Moderate to Infrequent), a new event will be defined to replace it in the category from which the original DBE was removed. Further, the newly defined event will be one that truly belongs in that category based on its overall frequency of occurrence. That is, it would not become a candidate for re-categorization in the future.

2.1 Evolution of Event Categorization and Acceptance Criteria

The evolution of event categorization schemes over a period of time can be seen from Table 2-2, which shows the categorization based on Title 10 of the Code of Federal regulations (10CFR) (References 3 and 6), American National Standards Institute (ANSI) 18.2 (Reference 4), NRC

Regulatory Guide 1.70 (Reference 1), ASME Code and ANSI 51.1-1983 (Reference 5). In particular, it shows the relationship between the various categorization schemes.

Table 2-3 shows the evolution of the acceptance criteria on radiological releases/fuel performance for Title 10 CFR, ANSI 18.2, and the Standard Review Plan (Reference 2). It is seen that for Title 10 CFR, the acceptance criteria are simply for two broad categories of events, namely, Anticipated Operational Occurrences (AOOs) and Accidents, whereas for the SRP, the evolution of the acceptance criteria has resulted in three broad categories of events. The last category, Limiting Faults, is divided into three subcategories with appropriate acceptance criteria attached to each. This approach leads to the breakdown of the Limiting Fault category (which covers a wide range of frequencies, 10^{-6} to 10^{-2}) into smaller frequency groups and results in more meaningful application of the concept of a constant, acceptably low risk due to applicable accident scenarios.

Note that other acceptance criteria (e.g., criteria on peak primary system and secondary system pressures) can also be dealt with in a similar manner as was done for the acceptance criteria for radiation release. Examples are the limits on the peak RCS and secondary side pressures. The peak pressure limit for the Moderate Frequency events would be significantly lower than that for the Limiting Fault 3 category of events.

The systematic application of the RISA approach would start off with a listing of the types of events and their categorization into the various frequency categories using a matrix such as shown in Table 2-4. The limiting events in each type of event class are listed along with the frequency category to which they belong. Based on the frequency categorization, the appropriate acceptance criteria for the limiting event can be identified.

Table 2-5 shows a typical categorization scheme and relevant radiological release acceptance criteria for each category of events.

2.2 RISA Procedure

The procedure for risk-informing the categorization of the DBEs involves the following:

1. Review the existing Analysis of Record (AOR) for the specific DBE for which it is desired to risk-inform the event categorization. Clearly identify the initiating event (IE), and the limiting CO and SF.

2. Using an industry database of event frequencies and probabilities, determine the frequency of occurrence for the IE, and the conditional probabilities for the CO and SF.

3. Calculate the overall frequency of the DBE using the individual frequency of occurrence for the IE, and the conditional probabilities for the CO and SF.

4. Compare the overall frequency of occurrence for the DBE under consideration against the frequency of occurrence for the three (3) broad classifications of events (moderate

frequency, infrequent and limiting faults) provided in the event categorization matrix (e.g., Table 2-1). Assign the DBE to the appropriate frequency category based on the comparison.

5. If the selected DBE can be re-categorized, define a new replacement DBE for the frequency category from which the original DBE is removed (See Section 3.5 for guidelines for defining the replacement event).

6. Identify the congruent acceptance criteria that must be met for the re-categorized DBE and the new replacement event using the criteria defined in the regulatory guidance (e.g., Table 2-3) for the calculated overall frequency of occurrence.

Table 2-1 Event Categorization Matrix

Event Frequency Range (per reactor-year)	Other Categorization Schemes				
	US NRC			ANS/ANSI	
	10CFR (References 3 & 6)	RG 1.48 ASME Code*	RG 1.70 Rev. 3 (Reference 1)	N18.2 (Reference 4)	51.1-1983 (Reference 5)
Planned Operations	Normal	Normal	Normal	Condition I	PC-1
-------10E-1-------	Anticipated Operational Occurrences	Upset	Moderate Frequency	Condition II	PC-2
			Infrequent Incidents	Condition III	PC-3
-------10E-2-------		Emergency			
-------10E-3-------				Condition IV	PC-4
-------10E-4-------	Accidents		Limiting Faults		
-------10E-5-------		Faulted			PC-5
-------10E-6-------					Not Considered

* Information extracted from Reference 5.

Table 2-2 Evolution of Event Categorization

10 CFR	ANSI 18.2	Reg. Guide 1.70	ANSI 51.1
Normal Operation	Condition I Normal Operation	-----	PC-1
Anticipated Operational Occurrences (AOOs) "...are expected to occur one or more times during the life..."	Condition II Moderate Frequency "...may occur during a calendar year..."	Moderate Frequency	PC-2
	Condition III Infrequent Incidents "...may occur during a lifetime..."	Infrequent	PC-3
Accidents "...exceedingly low probability of occurrence..."	Condition IV Limiting Faults	Limiting Faults	PC-4
			PC-5

PC: Plant Condition

Table 2-3 Evolution of the Acceptance Criteria on Radiological Release Based on Categories

10 CFR	ANSI 18.2	Section 15 Standard Review Plans
Anticipated Operational Occurrences (AOOs) • Specified Acceptable Fuel Design Limits (SAFDLs) are not exceeded	Condition II Moderate Frequency • 10 CFR 20, p. 20.1 • No loss of function of any barrier to radioactive product escape	Moderate Frequency • SAFDLs are not exceeded
	Condition III Infrequent Incidents • > 10 CFR 20 • Damage to small fraction of fuel elements	Infrequent • Event dependent
Accidents • 10 CFR 100	Condition IV Limiting Faults • 10 CFR 100	Limiting Faults • 1) Small fraction of 10 CFR 100 • 2) Well within 10 CFR 100 • 3) 10 CFR 100

Table 2-4 Event Categorization Matrix Based on RISA

Type of Events	Moderate Frequency Events	Infrequent Events	Limiting Faults		
			LF-1	LF-2	LF-3
			Incidents Not likely to Occur (Small Fraction of 10 CFR 100)	Incidents of Low Probability (Well within 10 CFR 100)	Incidents of Exceedingly Low Probability (10 CFR 100)
1					
2					
3					
4					
5					
6					
7					
8					

Note: This table is provided as an example of an event categorization matrix that displays potential event categories. Entries under column headings are not provided, since the purpose of the table is to identify only the event categories.

Table 2-5 Categorization Probabilities/Frequencies and Radiological Release Acceptance Criteria

10 CFR	AOOs		Accidents		
Reg. Guide 1.70 (Ref. 1) and SRP (Ref. 2)	**Moderate Frequency Event**	**Infrequent Events**	**Limiting Faults**		
			Incidents Not Likely to Occur	**Incidents of Low Probability**	**Incidents of Exceedingly Low Probability**
Probability (per reactor year)	1 – 0.5	0.5 – 1.7E-2	1.7E-2 - 1.0E-3	1.0E-3 – 1.0E-4	1.0E-4 – 1.0E-6
Frequency (per reactor year)	≥0.693	0.693 – 1.7E-2	1.7E-2 - 1.0E-3	1.0E-3 – 1.0E-4	1.0E-4 – 1.0E-6
Acceptance Criteria	Appendix I DNBR≥SAFDL	1% 10CFR100 DNBR≥1.0	10% of 10CFR100	25% of 10CFR100	10CFR100

3.0 APPLICATION OF THE RISA APPRAOCH

The viability and feasibility of the RISA approach in classifying DBEs and evaluating the results against congruent acceptance criteria will be best understood by applying the approach to a specific DBE and plant. The objective is to use the RISA approach to:

(1) identify the current classification of the specific DBE,

(2) identify the CO and SF used in the analysis,

(3) determine the frequencies for the IE, CO and SF,

(4) calculate the overall event frequency by combining the individual frequencies,

(5) reclassify the selected DBE into a new frequency category and define a replacement event for the selected DBE and the original frequency classification, and

(6) specify the new congruent acceptance criteria that the selected DBE would be subjected to based on the revised frequency classification.

3.1 Selection of Pilot Plant and Initiating Event

A survey of Westinghouse Owners Group (WOG) plants was conducted to identify the pilot plant for the study and to narrow the list of IEs to be considered. Responses received from WOG members identified a suggested list of IEs that included the loss of normal feedwater, steam line break, boron dilution, and inadvertent ECCS actuation as example events. These events were suggested based on the potential difficulty faced in satisfying the acceptance criteria based solely on the IE frequency classification. The goal was to gain relief from the potential difficulty in satisfying the incongruent acceptance criteria by reclassification of the overall event frequency and use of the resulting congruent acceptance criteria.

Based on a review of the IEs recommended and discussions with the WOG members, the Loss of Normal Feedwater (LONF) event was chosen as the example IE and the South Texas Project units as the pilot plant. This IE benefits substantially from the application of the RISA approach.

3.2 Description of Initiating Event at STP

The following is a brief description of the example initiating event for the South Texas Project Units 1 and 2. The IE under consideration is a total loss of normal feedwater that can be caused by valve malfunctions or pump failures that result in a reduction in the capability of the secondary system to remove heat generated in the reactor core. A loss of offsite power can also cause a total LONF. However, for the analysis considered, a LOOP subsequent to reactor/turbine trip caused by low-low steam generator level resulting from the LONF is assumed. If an alternate supply of feedwater were not provided to the plant, core residual heat

following the trip would heat the primary system to the point where water relief from the pressurizer would occur, resulting in a substantial loss of water from the RCS. Since the plant is tripped well before the steam generator heat transfer capability is reduced, the primary system variables never approach a DNB condition.

In the short term, the event is either less limiting than other Condition II (moderate frequency) events or of little concern with respect to the SRP criteria on primary and secondary pressure and DNBR. The event is thus analyzed to ensure the pressurizer does not go solid (i.e., fill with water) and subsequently discharge water from the power operated relief valves (PORVs) and/or the Primary Safety Valves (PSVs). This ensures long term satisfaction of the above mentioned acceptance criteria. The PORVs are conservatively assumed to function in order to maximize the swell in pressurizer water level, thus maximizing the potential for pressurizer fill. Pressurizer fill can result in primary coolant being discharged to the containment due to a rupture of the pressurizer relief tank rupture disk. The rupture disk would perform its pressure relief function if sufficient water is discharged from the pressurizer via the PSVs and/or the PORVs. Further, an uncontrolled release of primary fluid to the containment may occur in the event the PSVs or PORVs fail to reseat due to the discharge of subcooled water. Failure of these valves to reseat is likely in this case since neither of these valves is qualified to discharge water. Note, however, that in some plants the PORVs have been qualified to discharge water.

The above situation leads to a Condition II event becoming a more severe, potentially a Condition III or Condition IV event, since in this case a Small Break Loss of Coolant Accident (SBLOCA) would be in progress. For the STP units, the worst postulated LONF event with respect to potentially overfilling the pressurizer with eventual discharge of water is one in which a LOOP occurs coincident with the reactor trip. This is due to rapid depletion of the steam generator secondary inventory prior to the reactor trip on steam generator low-low level followed by reactor coolant pump (RCP) coastdown which further degrades the capability of the reactor coolant system to remove residual core heat.

The following events occur upon a LOOP coincident with a reactor trip:

1. Plant vital instruments are supplied from emergency DC power sources,
2. Increasing secondary pressure following a reactor trip results in the steam generator PORVs opening automatically to the atmosphere. Turbine bypass to the condenser is not available due to the loss of power. If steam flow through the PORVs is not available, the steam generator safety valves may lift to dissipate the sensible heat of the fuel and coolant in addition to the residual decay heat produced in the reactor.
3. As the no-load temperature is approached, the steam generator PORVs (or the safety valves if the PORVs are not available) are used to dissipate the residual decay heat and to maintain the plant at the hot standby condition.
4. The Standby Diesel Generators (SBDGs), started on loss of voltage to the plant emergency busses, begin to supply plant vital loads.

The auxiliary feedwater system (AFWS) for the STP units consists of four AFW pumps, three motor-driven and one turbine-driven, along with the necessary piping and valves to deliver

feedwater flow to the four steam generators. The AFW pumps, which take suction from the auxiliary feedwater storage tank, are started automatically by an actuation logic that is controlled by the steam generator water level. The turbine-driven AFW pump utilizes steam from the secondary system and exhausts it to the atmosphere. The motor driven AFW pumps receive power from the SBDGs. The pumps take suction directly from the auxiliary feedwater storage tank for delivery to the steam generators. The subject event analyzed for the STP units assumes a SF in the AFW system. The failure assumed is that of a single train of AFW actuation logic causing the failure of two out of the four AFW pumps to automatically start. Hence, two motor driven pumps are started automatically delivering flow to two of the four steam generators. Operator action is relied upon to manually start a third AFW pump resulting in flow being delivered to a third steam generator. Hence, three out of the four steam generators eventually receive AFW. Note that one AFW pump out of service is not allowed by Technical Specifications for the STP plants.

Upon a loss of power to the RCPs, coolant flow necessary for core cooling and removal of residual heat is maintained by natural circulation in the reactor coolant loops. In the long term and subsequent to AFW actuation, the addition of feedwater is manually controlled to maintain proper steam generator water level.

3.3 Regulatory Guidance for the Initiating Event

The Regulatory guidance are best summarized in SRP Section 15.2.7 "Loss of Normal Feedwater Flow" (Revision 1, 1981).

The General Design Criteria (GDC) which are imposed on the acceptability of this event are:

GDC 10 – Reactor design: Requires that SAFDLs are not exceeded during AOOs,

GDC 15 – Reactor coolant system design: Requires that the RCS be designed with appropriate margin to assure that the pressure boundary will not be breached during AOOs,

GDC 26 – Reactivity control system redundancy and capability: Requires reliable control of reactivity changes to assure that SAFDLs are not exceeded during AOOs,

TMI Action Plan (NUREG-0737) Items II.E.1.1 and II.E.1.2 require that the design and performance of the auxiliary feedwater system should be such that it (1) can automatically start-up, (2) is capable of adequately removing the decay heat, and (3) has protection from a SF.

Based on the above guidance, the NRC has developed the following specific acceptance criteria:

a. Primary and main steam pressure must be less than 110% of design values,
b. Fuel cladding integrity must be maintained by ensuring DNBR remains above the 95/95 DNBR limit,

c. An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently,
d. An incident of moderate frequency in combination with any single active component failure, or single operator error, shall be considered and is an event for which the amount of fuel failure must be calculated for radiological dose calculations. There shall be no loss of function of any fission product barrier other than the fuel cladding.
e. Regulatory Guide 1.105, "Instrument Spans and Setpoints" positions are used with regard to their impact on plant response.
f. The most limiting plant systems SF, as defined in the "Definitions and Explanations" of Appendix A to 10 CFR Part 50, shall be identified and assumed in the analysis and shall satisfy the positions of Regulatory Guide 1.53, "Application of the Single-Failure Criterion to Nuclear Plant Protection Systems".
g. The analysis of the LONF transient should be performed using an acceptable analytical model.

The above criteria are thus applicable to the analysis of two specific event combinations, those being the LONF and the LONF with a SF.

One of the objectives of this topical report is to recommend a set of congruent acceptance criteria consistent with the above requirements for the specific event combination under consideration based on the frequency of the specific event combination. Hence, it may be determined that the IE in combination with certain COs and SFs can deviate from certain currently established acceptance criteria if the frequency of the event combination is sufficiently low. Table 3-1 provides the recommended revised congruent acceptance criteria that are based on the SRP criteria presented above. The frequency definitions are comparable to the Regulatory Guide 1.70 (Reference 1) definitions for Moderate Frequency events, Infrequent Events and Limiting Faults with the exception that the Limiting Fault category has been further divided into three subcategories: Limiting Fault 1 (LF-1), Limiting Fault 2 (LF-2) and Limiting Fault 3 (LF-3). The table also provides the associated frequencies in terms of events per reactor year.

Table 3-1 Recommended Revised Event Acceptance Criteria

Parameter	Moderate Frequency+ ≥ 0.693	Infrequent Events+ $0.693 - 1.7 \times 10^{-2}$	Limiting Faults		
			LF-1+ $1.7 \times 10^{-2} - 10^{-3}$	LF-2+ $10^{-3} - 10^{-4}$	LF-3+ $10^{-4} - 10^{-6}$
RCS Pressure	< 110% of Design	< 110% of Design	< 110% of Design	< 110% of Design	< 120% of Design
Secondary Pressure	< 110% of Design	< 110% of Design	< 110% of Design	< 110% of Design	< 120% of Design
Fuel Performance	DNBR > 95/95 Limit, No Fuel Melting	DNBR > 95/95 Limit, No Fuel Melting	Maintain Coolable Geometry	Maintain Coolable Geometry Radially Averaged Enthalpy < Licensed Limit*	Maintain Coolable Geometry
Radiological Release	Appendix I	Very Small Fraction of 10CFR100 (1%)	Small fraction of 10CFR100 (10%)	Well Within 10CFR100 (25%)	10 CFR 100

* Since changes have been considered for this parameter based on industry research.
+ Frequencies are in units of events per reactor year.

3.4 Probabilistic Consideration of the Initiating Event

The LONF event is identified as a Moderate Frequency event in the SRP (Reference 2). From plant specific PRA information for the STP plants, the frequency of occurrence for this event (F_{LONF}) is 5.08E-02 per year (mean value) and 1.13E-01 per year (95th percentile value). The CO is the LOOP on assumed turbine trip. This CO has a conditional probability (P_{LOOP}) of 6.21E-04 (mean value) and 1.50E-03 (95th percentile value) for the STP plants. The SF is assumed to be the failure of an ESF signal that results in two out of the four AFW pumps not starting up based on a low steam generator water level signal. For the STP plants, this event has a conditional probability (P_{LOESF}) of 6.05E-04 (mean value) and 1.33E-03 (95th percentile value). Thus, the overall frequency of this DBE is:

$$
\begin{aligned}
F_{overall} &= F_{LONF} * P_{LOOP} * P_{LOESF} \\
&= 5.08E\text{-}02 * 6.21E\text{-}04 * 6.05E\text{-}04 = 1.91E\text{-}08 \text{ (mean)} \\
&= 1.13E\text{-}01 * 1.50E\text{-}03 * 1.33E\text{-}03 = 2.254E\text{-}07 \text{ (95}^{th}\text{ percentile)}
\end{aligned}
$$

This suggests that the LONF analyzed here has a significantly lower frequency than that for the "Moderate Frequency" classification due principally to the addition of the CO and the SF. Consequently, the acceptance criteria for the event are justifiably less restrictive than those for the Moderate Frequency event, based on maintaining a very low risk to public health and safety. Currently, the acceptance criteria to be met for the LONF event are the ones for a Moderate Frequency event.

Consistent with the approach described in Section 3.2.3 of Reference 5, a comparison of the overall frequency value calculated above with the frequencies shown in Table 2-5 was made. It suggests that the LONF IE in combination with the CO of a LOOP and the SF of the ESF signal would fall into the outer fringes of the Limiting Fault-3 category. This is a shift of four categories (shift from moderate frequency event to Limiting Fault-3 category). To avoid excessive change in categorization that is simply based on the change in the frequency of occurrence, the RISA approach also considers the characteristics of the event under consideration in the categorization process. For the LONF event in combination with the given CO and SF, the pressurizer can fill up and potentially discharge two phase fluid or liquid through the PORV and the relief tank, if no mitigating operator action is assumed. This event would thus represent a subset of the SBLOCA event which has been categorized as a LF-2 event.

This leads to correspondingly higher acceptance criteria on the radiological releases as shown in Table 2-5. The analyses documented in the STP UFSAR show that the acceptance criteria on relevant parameters for a SBLOCA are met with adequate margins.

3.5 Replacement Event for the Re-categorized Event

The shifting of the LONF IE in combination with a CO and SF into the Accident category, based on the rationale described above, would require the identification of another event to replace it in the Moderate Frequency event category. This would be accomplished using the same approach previously employed to group the "decrease in heat removal by the secondary system" (SAR

section 15.2) events into the various frequency categories. In general, the procedure for identifying the replacement event involves the following steps.

(1) Identify/compile initiating events in the "decrease in heat removal from the secondary system" classification whose frequencies fall within the Moderate Frequency category.

(2) Evaluate/order these events to identify the events that would be limiting within the Moderate Frequency event category. The objective is to reduce the number of events that need to be quantitatively analyzed. Qualitative evaluations/comparisons may be sufficient to identify the Moderate Frequency event that leads to the most limiting consequences.

(3) Perform limited amount of quantitative analyses to determine the event which leads to the most limiting consequences if qualitative evaluations do not identify the initiating event that results in the most limiting consequences.

(4) Choose the initiating event that gives the most limiting consequences as the replacement Moderate Frequency event. The event chosen, including the CO and SF, will have a frequency of occurrence that truly falls in the Moderate Frequency classification. That is, the replacement event will not become a candidate for re-classification at a future date.

As a consequence of applying the above guidelines, an appropriate replacement Moderate Frequency event for the LONF IE in combination with a CO and SF may be the LONF event by itself. This is because the SRP specifically requires the analysis of the LONF event whose frequency of occurrence falls within the Moderate Frequency event category.

For the purpose of the study documented in this topical report, the definition of a replacement event and associated evaluation were not performed since they do not involve any new or different approach from those traditionally used.

4.0 THERMAL HYDRAULIC EVALUATION

4.1 Description of Safety Analysis of the Initiating Event

As mentioned in Section 3.2, the LONF event for the STP plants is such that one criterion is imposed (pressurizer does not go solid) to ensure long term satisfaction of Criteria "a" and "b" of Section 3.3, i.e., fission product barrier integrity. The short term satisfaction of these criteria is ensured either (1) by being bounded by other events or (2) by being of minor concern for this event. The pressurizer fill criterion also ensures satisfaction of Criterion "c": i.e. the LONF event does not progress to a more severe event, that being a SBLOCA.

The safety analyses supporting the UFSAR thus focuses on selecting parameters and utilizing assumptions that exacerbate the potential for pressurizer fill. The analysis was performed utilizing the RETRAN computer code. The event combination analyzed was as follows:

(1) Initiating Event	Total Loss of Normal Feedwater
(2) Reactor Trip Credited	Low-Low Steam Generator Water Level
(3) Coincident Occurrence (CO)	Loss of Offsite Power, Two (2) Seconds Following Reactor Trip (Rods Begin to Drop)
(4) Single Failure	Failure of an Engineered Safety Features (ESF) Signal: Two (2) Out of Four (4) Auxiliary Feedwater (AFW) Pumps Do Not Start
(5) Automatic AFW	Two Motor Driven AFW Pumps Start 60 Seconds Following the AFW Actuation Signal
(6) Third AFW Pump	A Third AFW Pump is Assumed to be Started Manually 15 Minutes Following AFW Actuation

The inclusion of a SF addresses Criteria "d" and "f" of Section 3.3. The remaining Criteria "e" and "g" are not relevant to the current study, since this study focuses on the change in the overall frequency of the LONF event in combination with a CO and SF.

Section 4.2 details the assumptions utilized to establish the limiting case. The most limiting combination was used in the final analysis. The pressurizer volume of the STP plants is 2100 ft^3. The limiting case as presented in the UFSAR yielded a maximum pressurizer water volume of 2040 ft^3 for the long term analysis of the LONF event with a LOOP. In order to achieve this acceptable maximum water volume it was necessary to require that two (2) motor driven pumps be available to supply auxiliary feedwater to the steam generators automatically and a third auxiliary feedwater pump be available to be manually started 15 minutes following auxiliary feedwater actuation. This requirement prevents the Technical Specifications from being relaxed to allow a single motor driven auxiliary feedwater pump to be out of service indefinitely. This

condition along with an assumed SF of the auxiliary feedwater turbine driven pump would result in only two (2) motor driven pumps being available to supply auxiliary feedwater to the steam generators, an unacceptable condition that will result in the pressurizer going solid.

The RISA approach, however, has the potential to allow relaxation of applicable Technical Specifications in this case depending on the frequency of the event resulting in the pressurizer going solid.

4.2 Discussion of Assumptions and Single Failures

The assumptions utilized in the performance of the STP plant LONF analysis with a LOOP are detailed in Table 4-1. As mentioned in Section 3.2, the SF assumed was the failure of an ESF train to actuate auxiliary feedwater which results in only two out of the four auxiliary feedwater pumps starting automatically.

These assumptions are specific to the STP Units 1 and 2 plants. Future analyses for other plants, that utilizes the RISA approach discussed within this document, will need to define the assumptions that are specific to the plant being analyzed as was done here for the STP plants.

Table 4-1 Assumptions Used in the Pilot Plant Total Loss of Normal Feedwater Analysis

ITEM	ASSUMPTION	COMMENTS
1	Both Minimum Tube Plugging (0%) and Maximum Tube Plugging (10%) considered.	Minimum tube plugging is limiting.
2	A-AFW pump (loop 1) no longer "out of service" indefinitely (T.S. 3.7.1.2 ACTION a.)	
3	Limiting Single Failure is failure of ESFAS actuation "Train A" which precludes automatic delivery of AFW flow to steam generators A and D.	
4	AFW delay time for auto start: 60 seconds	Assumes diesel start time as well; hence very conservative for cases without LOOP.
5	Operator action time to start third AFW pump: 15 minutes following reactor trip on Low-Low Steam Generator Water Level.	
6	Δ94 Steam Generators.	
7	Credit taken for a LOOP resulting in 50% of the Backup Heater Capacity (as opposed to 100%) being available due to Emergency Diesel Generator loading.	Applicable to LOOP cases only.
8	Backup Heaters actuate on both pressurizer level deviation and pressure effects.	
9	Low Steam Generator Water Level Trip Setpoint of 10.1% Narrow Range Level Span (NRS). This was reduced from previous analyses to account for the pressure drop across the steam generator mid-deck plate.	
10	Credit for thick metal masses associated with the reactor vessel and steam generator inlet and outlet plenums was not taken in this analysis. This differs from the previous analysis.	
11	LOOP subsequent to reactor trip.	LONF event analyzed with and without LOOP.
12	LOOP is assumed with a 2 second delay following reactor trip.	
13	Initial NSSS Power 3821 MWt + 2% uncertainty	
14	Pump Heat- maximum assumed 24 MWt	Value assumes 4 RCPs operating. Upon RCP trip pump heat will coast down proportional to rotational speed of the pumps.
15	Both low and high nominal Tavg considered with ± 5.1°F uncertainty.	High nominal minus uncertainty limiting.
16	Both low and high initial Pressurizer Pressure considered, nominal 2250 psia ± 46 psi uncertainty	Nominal plus uncertainty limiting.
17	Initial RCS Flow = Thermal Design Flow =98,000 gpm/loop	
18	Initial Pressurizer Level = 64.1% NRS (high Tavg, full power nominal level (57% NRS) plus 7.1% NRS uncertainty) and 47.1% NRS (low Tavg, full power nominal level (40% NRS) plus 7.1% NRS uncertainty)	High Tavg Program plus uncertainty limiting.
19	Initial Steam Generator Level = 76.3% NRS (full power nominal level (70.7% NRS) plus 5.6% NRS uncertainty)	
20	Main Feedwater Temperature: 390 °F and 440 °F	440 °F limiting
21	Low-Low Steam Generator Water Level Trip delay time = 2.0 seconds	
22	Physics Parameters: Minimum Reactivity Feedback	Note: for the acceptance criterion of interest, a stuck rod is of little consequence since pressurizer level is governed by long term heat removal. Decay heat rate would have a first order effect on meeting this criterion.

Table 4-1 Assumptions Used in the Pilot Plant Loss of Main Feedwater (Continued)

ITEM	ASSUMPTION	COMMENTS
23	Decay Heat: ANS 1979 + 2σ	2σ uncertainty included on decay heat
24	Pressurizer spray assumed to actuate via the pressurizer pressure control system	Spray actuation more adverse for maximizing pressurizer level. Hence, the relevant control systems are assumed to operate.
25	Pressurizer proportional and backup heaters actuate as part of the pressurizer pressure control system. Backup heaters also actuate on high pressurizer level deviation 5% NRS.	Heater actuation more adverse for maximizing pressurizer level. Hence, the relevant control systems are assumed to operate.
26	Pressurizer PORV actuation is assumed to occur on an uncompensated signal setpoint of 2350 psia (for an initial pressurizer pressure of 2296 psia).	PORV actuation swells the pressurizer level; hence is an adverse assumption for this acceptance criterion. For low initial pressurizer pressure a compensated signal setpoint of 100 psid was assumed (i.e. for an initial pressurizer pressure of 2204 psia, 2304 psia was assumed). However, high initial pressurizer pressure is limiting.
27	Vessel Mixing; Perfect mixing is assumed for LONF with LOOP.	Design mixing is assumed for LONF cases.
28	Fuel Heat Transfer Data: Minimum UAs.	As opposed to maximum.
29	Auxiliary Feedwater Enthalpy a maximum; corresponds to 120 °F.	Limits RCS heat removal.
30	Maximum AFW purge volume: 40 ft^3/loop.	
31	Minimum AFW flow: 500 gpm/pump	Limits RCS heat removal.
32	Rod control system : Off	No credit for this control system
33	Pressurizer Level Control System: Off	No credit for charging and letdown is assumed.
34	Initial Steam Generator Conditions Consistent with Initial NSSS Power, Thermal Design Flow, Initial T_{avg}, Initial Pressurizer Pressure, Nominal Steam Generator Level + Uncertainty, and Steam Generator tube Plugging.	GENF code generates IGOR input for eventual use in RETRAN
35	Tech Spec MSSV setpoints increased by 3% (tolerence + drift) and a 21.8 psi ΔP between the steam generator and the MSSV sensing point.	Minimize heat removal
36	Secondary PORV actuation credited. Opening pressure is 1282.0 psia which includes a 3% tolerance. The assumed full open pressure is 1345.35 psia which includes 5% accumulation.	Credited with customer consent. One secondary PORV per steam generator was modeled.

4.3 Description of Results in Relation to Meeting Regulatory Requirements

The criterion of interest for the analysis of the LONF event is the maintenance of the pressurizer water volume to less than 2100 ft^3, which is the volume of a filled pressurizer. Satisfaction of this criterion ensures no water relief via the PSVs or PORVs. The discharge of water from either of these valves can result in a stuck open valve with subsequent rupture of the pressurizer relief tank rupture disk, causing primary coolant to be discharged to the containment. This would result in violation of two (2) NRC acceptance criteria as defined by the SRP: 1) the event would progress to a more serious event (i.e. a LONF would propagate to a SBLOCA); and 2) for the event with a SF, the criterion which states that there shall be no loss of function of any fission product barrier other than the fuel cladding would be violated since the primary coolant barrier would be bypassed. The event as analyzed for the pilot plant meets the 2100 ft^3 requirement, however, this requires overburdening restrictions on auxiliary feedwater system availability via the plant Technical Specifications. The Utility's preference would be to allow a single auxiliary feedwater pump to be out of service such that a SF in conjunction with a LONF event would result in only two (2) auxiliary feedwater pumps delivering flow to two (2) steam generators. In this scenario, the SF would be assumed to be in the turbine driven pump, and since the remaining motor driven pump is out of service, AFW delivery would then be limited to two (2) pumps.

Exploring the frequency of the event being analyzed helps to alleviate the burdensome restriction on the AFW system availability since the event with all its conservatism (LONF, LOOP, SF in a safety system etc.) is shown to be a Limiting Fault rather than an AOO. The pressurizer becoming water solid with water being discharged to the containment is an acceptable consequence for a Limiting Fault event, provided all regulatory acceptance criteria for the consequences of the event are met.

Section 3.4 addresses these possibilities from a risk perspective. It indicates that the overall frequency of the LONF event in combination with a LOOP and a failure of the ESF signal is 2.254E-07 based on 95th percentile values of IE frequency and conditional probabilities and 1.91E-08 based on mean values. These exceedingly low frequency values place this event in the Limiting Fault 2 category as discussed in Section 3.4. This categorization would allow the specific LONF event being considered to meet less restrictive acceptance criteria (e.g., those applicable to a SBLOCA).

5.0 RESULTS OF RISA APPLICATION

The application of the RISA approach resulted in the reclassification of the IE (i.e., LONF) in combination with a CO and SF into a frequency category having frequencies that are several orders of magnitude smaller. This category is Limiting Fault 2. Consequently, the acceptance criteria for this event are shifted to those for a SBLOCA from the current acceptance criteria that are applicable to the Moderate Frequency event. The SBLOCA event was chosen, since the increasing RCS pressure and pressurizer level could potentially lead to a SBLOCA via the PORVs/PSVs and the pressurizer relief tank if no operator actions are assumed.

6.0 CONCLUSIONS AND RECOMMENDATIONS

By applying the RISA approach, the categorization of plant transients and accidents can be accomplished more rigorously, systematically, and more reasonably. In particular, the use of the RISA approach allows the classification of specific transients and accidents into more meaningful event categories by considering the frequency of occurrence of the IE and conditional probabilities of COs and SFs. Correspondingly, the acceptance criteria for an IE in combination with a CO and SF would become less restrictive than those resulting from the use of the current deterministic approach. Current regulatory guidance suggests that the IE in combination with a CO and SF should use acceptance criteria that are essentially the same as those applicable to the initiating event by itself, even though the frequencies for both scenarios differ by orders of magnitude.

The use of the RISA approach can more easily lead to acceptable results for plant transients and accidents, since the re-categorization based on this approach would support the use of less restrictive acceptance criteria. The risk to public health and safety would be maintained at a very low level consistent with the regulations due to the fact that the event consequences would still be judged commensurate with the frequency of occurrence.

The viability and feasibility of the RISA approach to bring event categorization and acceptance criteria into congruence have been demonstrated by specific application to a pilot plant and an example event. The application of the RISA approach to other PWR designs would be analogous, although the plant system/component configuration, alignment, COs and SFs are very plant dependent.

7.0 REFERENCES

1. US NRC Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants (LWR Edition)", USNRC, Revision 3, November, 1978.
2. NUREG-0800, "Standard Review Plan", Revision 2, USNRC, July 1981.
3. Title 10, Code of Federal Regulations, Part 100, "Reactor Site Criteria".
4. ANSI N18.2, "American National Standard, Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants", August 6, 1973.
5. ANSI/ANS-51.1-1983, "American National Standard Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants", April 29, 1983.
6. Title 10, Code of Federal Regulations, Part 50, "Domestic Licensing of Production and Utilization Facilities".