Post Exam Comment Resolution

Post-Exam Comment Resolution

Summary
Entergy submitted a post-exam comment on Written Exam Q#8, recommending acceptance of two correct answers. The NRC reviewed the proposed exam change and determined that Q#8 is valid as written, that Choice D (the key answer) is correct and Choices A, B and C are all incorrect. The exam key will remain unchanged.

Written Exam Question #8

Facility Recommendation:
Accept both Choice D (the key answer) and Choice C (a distractor) as correct answers.

NRC Comment Resolution:
Question #8 is valid as written. The answer key will remain as-is.

Pass-Fail Statistics:
2 of 11 applicants missed this question. Choice D, the key answer, was selected by 9 applicants. Choice C, a distractor choice, was selected by 2 applicants. No applicant asked for any clarification of this question during exam administration.

Discussion:

NOTE
Documents attached to this comment resolution include:
- the facility justification for proposing two correct answers (which included the full text of the question)
- the background document for event response procedure 2-FR-S.1, Response to Nuclear Power Generation/ATWS

Question #8 asked applicants to determine the required actions on an Anticipated Transient Without Scram (ATWS) event if the main turbine stop valves failed to close <u>and the reason for the actions</u>. Only Choices C and D described the required actions, which are to runback the turbine and, if not successful, to then close MSIVs. Choices A and B did not correctly describe the required actions. Choice D also described the correct reason, which was to prevent an excessive cooldown. Choice C, proposed by the facility as a second correct answer, incorrectly described the reason for the actions as "*to limit RCS pressure transient for all initiating conditions*." This reason is not correct because taking the required actions does not limit RCS pressure for <u>all</u> initiating conditions.

The facility justified Choice C as a second correct answer by explaining that "*tripping the turbine was evaluated for all initiating events to ensure that it* [the action of tripping the turbine following the ATWS] *did not exceed the pressure limit of the RCS*." The background document explains Westinghouse evaluated plant response for a number of Condition II Anticipated Transient initiating events, including loss of normal feedwater, and accidental RCS depressurization, among others. Section 2.5 of the background document describes how the loss of load event and the loss of normal feedwater event have the largest predicted RCS pressure. The analysis for the loss of normal feedwater event showed a higher RCS pressure peak if the turbine trip were delayed by 30 seconds (tripping the turbine at T=60 instead of T=30). It is correct that the turbine is promptly tripped, or

runback, or effectively tripped by closing MSIVs, to limit the RCS pressure transient on a loss of feedwater ATWS.

In contrast, the ATWS analysis for accidental RCS depressurization intentionally assumed no turbine trip to maximize the adverse effects of RCS pressure drop to achieve conservatively low values for DNBR. This implies RCS pressure would be higher (not as limited) if the turbine were tripped during an accidental RCS depressurization event, which is the required procedural response to an ATWS, regardless of the initiating event. Section 2.2 (page 5 of the background document) summarizes the differences in plant response for different initiators, stating:

> *Loss of main feedwater, control bank withdrawal at power, and a spurious opening of a pressurizer PORV are examples of the differing nature of ATWS events. The required operator actions following identification of an ATWS event are the same but the reactor coolant system conditions may be very different; pressurizer pressure can exceed 2785 psig following a loss of main feedwater ATWS but will never exceed the nominal operating pressure following the spurious opening of a pressurizer PORV. Operators must be aware of such system responses and not rely on any signals or indications other than those for reactor trip.*

Additionally, the background "knowledge" for the specific procedure step that tripped the turbine refutes the validity of Distractor C as a 2nd potentially correct answer. Distract C stated the reason for the step action was, "*to limit RCS pressure transient for all initiating conditions.*" The background for Step 2, Verify Turbine Trip, states:

> *A turbine trip is required for an ATWS event where a loss of main feedwater has occurred.* ***For other ATWS events, with the exception of when a turbine trip is the initiating event, manual tripping of the turbine may yield a somewhat higher system pressure, depending on the initiating event and time in core life, than what would otherwise be expected.*** *However, this action has been determined to be necessary due to the analytical results presented and discussed in subsections 2.4, ATWS Analysis and Results, and 2.5, Discussion of Analytical Results. Since there are many initiating ATWS events and some that require immediate mitigating actions, diagnosis of the initiating event would not be feasible and separate guidance for different ATWS events would complicate training and could delay timely performance of necessary operator actions.*

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<u>determined to be necessary due to the analytical results presented and discussed in subsection</u> *2.4……*

CONCLUSION:

The question was developed and validated with D as the correct answer, and D remains a correct answer. Distractor C is also correct because:

- **For the question, the sequence is correct (runback the turbine then close MSIVs).**
- **Tripping the turbine is <u>required</u> for a loss of feedwater ATWS to maintain a secondary heat sink. The mass of water in the SG directly affects the peak RCS pressure. The ATWS analysis assumes the turbine is tripped within 30 seconds.**
- **For other initiating events, tripping the turbine may yield a somewhat higher RCS pressure than would otherwise be expected. Tripping the turbine has been evaluated and the results are acceptable for all other initiating events.**
- **Tripping the turbine is intended to maintain SG inventory which is necessary to limit the RCS pressure. This is true for any initiating event. Thus C is also a correct answer.**

After review, Question 8 has 2 correct answers C and D

NUREG 1122 KA 000029K306 Knowledge of the reasons for the following responses as they apply to the ATWS. Verifying a main turbine trip; methods.

Question# 8

Given:

- *Unit 2 was operating at 100% power when a reactor trip signal failed to trip the reactor, and the reactor cannot be tripped.*
- *The crew has entered 2-FR-S.1 Response to Nuclear Power Generation / ATWS*

Which of the following describes the required actions if the Stop Valves fail to close and the reason for this action?

A. Close MSIVs, IF NOT successful runback the turbine to limit RCS pressure transient for all initiating conditions

B. Close MSIVs, IF NOT successful runback the turbine to prevent an excessive cooldown

C. Runback the turbine, IF NOT successful close MSIVs to limit RCS pressure transient for all initiating conditions

D. Runback the turbine, IF NOT successful close MSIVs to prevent an excessive cooldown

The question asks the correct sequence of actions if the turbine fails to trip (methods) and the reasons the turbine is verified tripped.

Distractors A & B are incorrect because they both state the wrong sequence of turbine trip actions.

Per the Answer Key, D is correct. The EOP Background document for FR-S.1 states:

> *The turbine is tripped to prevent an uncontrolled cooldown of the RCS due to steam flow that the turbine would require. For an ATWS event where a loss of normal feedwater has occurred, analyses have shown that a turbine trip is necessary (within 30 seconds) to maintain SG inventory.*

Distractor C is also correct. Tripping the turbine was evaluated for all initiating events to ensure that it did not exceed the pressure **limit** of the RCS.

> *A turbine trip is required for an ATWS event where a loss of main feedwater has occurred, For other ATWS events, with the exception of when a turbine trip is the initiating event, manual tripping of the turbine may yield a somewhat higher system pressure, depending on the initiating event and time in core life, than would otherwise be expected. <u>However, this action has been</u>*

Reference Document Submitted by the Facility in Support of the Post Exam Comment

Note: Pages 5, 7, 18, 24, 28, and 68 of Reference Document 2-FR-S.1 BG, Revision 2 were flagged by the facility as containing information applicable to the post exam comment.

Entergy

Nuclear Northeast

One Site, One Team, One Fleet

Indian Point Energy Center

Procedure Use Is:

- ☐ Continuous
- ☐ Reference
- ☑ Information

Control Copy: CONTROLLED

Effective Date: 11/12/2015

2-FR-S.1 BG, Revision: 2

RESPONSE TO NUCLEAR POWER GENERATION/ATWS BACKGROUND DOCUMENT

2

Indian Point Operations

Approved By:

Tom Cramer 11/2/15

Procedure Sponsor, RPO/ Designee   Date

Team P

Procedure Owner

EDITORIAL REVISION

BACKGROUND INFORMATION

FOR

INDIAN POINT UNIT TWO

EMERGENCY OPERATING PROCEDURE

2-FR-S.1

RESPONSE TO NUCLEAR POWER GENERATION/ATWS

Rev. 2

10/29/2015

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1. INTRODUCTION

The Function Restoration Procedure FR-S.1, RESPONSE TO NUCLEAR POWER GENERATION/ATWS, provides guidance in the event of an unexpected nuclear flux condition following a reactor trip or safety injection actuation (indicated by the RED or ORANGE priority on the Subcriticality Status Tree), or if an ATWS event has occurred. The operator is provided with instructions to restore the core to a subcritical state; restoration of shutdown margin is desired, but is not a necessity to exit from this Function Restoration Procedure.

This procedure is entered from two separate branches of the Subcriticality Status Tree, one having a RED priority signifying nuclear power generation and the other having an ORANGE priority signifying imminent nuclear power generation. The nuclear instrument indications are either a positive startup rate on the intermediate range startup rate meters (ORANGE), or a power range indication greater than 5 percent power (RED).

If entry into this procedure is due to an anticipated transient without scram (ATWS) event, in which the automatic protection system failed to drop the control rods into the core, then power production and support systems more resemble an at-power configuration than a shutdown configuration. The urgency in this case is based on violation of the design basis protection of the core. Entry into this procedure is then from Step 1 of guideline E-0, REACTOR TRIP OR SAFETY INJECTION, after verifying that the reactor is not tripped (entry into E-0 was based on the need for a reactor trip).

Procedure FR-S.1 is exited when the reactor is subcritical. At this time the operator is instructed to return to the procedure and step that was in effect at the time FR-S.1 was entered.

2. DESCRIPTION

This section was copied without changes from the generic Emergency Response Guidelines and is applicable to Indian Point Unit Two.

Guideline FR-S.1, RESPONSE TO NUCLEAR POWER GENERATION/ATWS, provides the guidance necessary to respond to a loss of subcriticality condition or an ATWS event. The following subsections include descriptions of 1) the potential accident conditions leading to a loss of subcriticality, 2) ATWS background information, and 3) a broad range of ATWS events and the analytical basis for ATWS transients.

2.1 Loss of Subcriticality Condition

Following a reactor trip or safeguards actuation, nuclear flux is expected to drop promptly to the bottom of the power range and then decay off at a fixed rate to a normal shutdown level. Core heat production after several minutes should be limited to that from radioactive decay of fission products, rather than from the fission process itself. The power range indication above 5% is a minimum reading on the power range scale that indicates power production. It is also greater than the decay heat which the shutdown systems were designed to remove and is indicative of a condition requiring immediate response to prevent subsequent core damage. A positive startup rate in the intermediate range represents an early indication of power production since no inherent feedbacks will come into play until the core enters the power range.

Several scenarios can be hypothesized which would provide the positive reactivity insertion producing the symptoms for a loss of subcriticality. First, the initial reactor trip may not be complete - either the control rods do not all insert or do not fully insert, but not to the extent of being considered an ATWS. Second, an injection of water into the RCS could result in a dilution if the water source were not properly borated or if the incorrect source were aligned. Third, any excessive cooldown from either a secondary depressurization or excessive feedwater addition to the steam generators would feed back through a negative moderator temperature reactivity coefficient to produce a positive reactivity insertion.

Though there may be other scenarios which would add positive reactivity to the RCS, they would probably not be as easy to detect as the three conditions mentioned above. However, since the guideline is entered on event independent symptoms, the appropriate actions would still be taken.

2.2 ATWS Background Material

The concept of "anticipated transients without scram" (ATWS) has its origination in concerns raised by the Advisory Committee on Reactor Safeguards (ACRS) more than 10 years ago. At that time, one of the central issues was the need to ensure separation of the instrumentation systems used for reactor control from those used for reactor protection. As originally envisioned, if the control and protection functions were not completely separated, malfunctions or faults in the control system could impair the ability of the protection system to respond to transients requiring reactor trip.

As the issue of control-protection system interaction was resolved both in the regulatory arena and by design changes in the control and protection systems, the concept of ATWS changed. Eventually the definition of ATWS evolved into an unspecified common-cause failure (either electrical or mechanical) which precluded control rods from being inserted into the core in response to an anticipated transient (Condition II event) which requires reactor trip.

As the ATWS issue continued to be a concern during the 1970's, virtually all of the discussion took place in the licensing arena, as opposed to design and operations. The utilities and reactor vendors were confident of the high reliability of the reactor protection system. The NRC (and its predecessor, the AEC) did not share this confidence level and pressed for design changes which would add prevention and mitigation systems to the plants. Throughout this dialog, the potential for changes in plant operating conditions and procedures were discussed, mostly from the standpoint of a pre-event reliability program to provide assurance of the availability of the reactor protection system. After the occurrence of TMI, renewed interest in a post-event mitigation and recovery procedure for ATWS was expressed by the NRC. Events that have occurred at Browns Ferry and Salem, among others, have been regarded as "ATWS precursors" by the NRC and have increased regulatory interest in design and operational changes which represent a prevention or mitigation capability.

The final ATWS Rule, 10CFR50.62, became effective on July 26, 1984. The objective of the ATWS Rule was to reduce the risk from ATWS events to an acceptable level. This rule required the installation of hardware to improve the nuclear plant's capability to prevent an ATWS and mitigate its consequences. The NRC established the following performance criteria:

1)Service level C of the ASME Boiler and Pressure Vessel code is not exceeded for overpressure events. (Ensures no gross loss of structural integrity)
2)Fuel integrity is maintained.
3)No excessive radioactivity release will occur.
4)Containment will not fail.
5)Long term cooling and shutdown is maintained.

The ATWS Rule required that certain plant systems be installed to provide mitigation of the short term effects of ATWS events. For Westinghouse designed plants, the rule added new instrumentation systems to ensure turbine trip and automatic initiation of auxiliary feedwater systems in response to ATWS events. This new instrumentation system is termed the ATWS Mitigating System Actuation Circuitry (AMSAC). In addition to AMSAC, Combustion Engineering and Babcock & Wilcox plants were required to install a diverse scram system independent from the reactor protection system as compensatory measures for higher unfavorable moderator temperature coefficients (MTCs) to prevent potentially excessive reactor coolant system over-pressure. (The period of the fuel cycle time when the MTC is insufficiently negative to maintain RCS pressure below 3200 psig during an ATWS is designated "unfavorable MTC".) Westinghouse plants were excluded from this requirement as they generally maintain an unfavorable MTC of 1 percent. Prior to finalizing the ATWS Rule, the NRC required all plants to develop an ATWS mitigation/recovery procedure consistent with existing plant equipment as part of post-TMI procedures development.

In 1987, the NRC requested the PWR owners groups to quantitatively reassess the initial MTC analyses that support the ATWS position for PWRs and specifically address reload core designs that have less conservative initial MTCs. The responses generally concluded that no significant changes were required in ATWS analyses.

As part of the NRC program to address regulatory effectiveness, the NRC evaluated the effectiveness of the ATWS Rule to determine if the requirements contained in it achieved the desired outcomes. The results of the NRC study are contained in NUREG 1780 which was published in September 2003. The study confirmed that safety has been maintained or otherwise enhanced with plants' implementation of the resolution of the ATWS Rule and associated modifications.

Although the NRC concluded in NUREG 1780 that the risk from ATWS is in the range foreseen when the ATWS Rule was issued, several issues related to less negative MTC effects on ATWS mitigation have the potential to erode past achievements. The ATWS mitigation capability on a PWR is highly dependent on the MTC. Mitigative functions are considered by the ATWS Rule regulatory basis to be non-viable if the ATWS peak pressure exceeds 3200 psig; as a sufficiently negative MTC will limit the ATWS peak pressure. Fuel design to achieve longer cycles and higher power ratings may result in less negative MTCs at full power for a larger fraction of the cycle time, during which time ATWS mitigation may be less effective. Further fuel cycle changes and power upgrades that could affect the ATWS risk may require compensatory measures (e.g., hardware or procedural), consistent with the underlying regulatory basis behind the ATWS Rule.

Guideline FR-S.1, RESPONSE TO NUCLEAR POWER GENERATION/ATWS, specifies the mitigating actions required following ATWS events. The reactor coolant system conditions at the time the operator identifies an ATWS event can be very different depending on the initiating event. Loss of main feedwater, control bank withdrawal at power, and a spurious opening of a pressurizer PORV are examples of the differing nature of ATWS events. The required operator actions following identification of an ATWS event are the same but the reactor coolant system conditions may be very different; pressurizer pressure can exceed 2785 psig following a loss of main feedwater ATWS but will never exceed the nominal operating pressure following the spurious opening of a pressurizer PORV. Operators must be aware of such system responses and not rely on any signals or indications other than those for reactor trip.

Transient response is also highly dependent on the time in fuel cycle at which an ATWS occurs. Response is much more severe very early in cycle lifetime than later in the cycle. For example, primary pressure may exceed 2985 psig for a turbine trip ATWS at beginning of cycle life but not exceed 2560 psig (the pressurizer safety valve setpoint + accumulation) near the end of cycle life.

This guideline only addresses the short-term operator actions. The operator must attempt alternate means of reactor trip and maintain a secondary-side heat sink via turbine trip (for a total loss of main feedwater) and auxiliary feedwater actuation. Turbine trip and actuation of auxiliary feedwater would normally be generated by the reactor protection system but it is assumed that the same fault that prevents a reactor trip also prevents these functions. Therefore, the guideline provides for these functions to be performed without delay by the operator in the control room.

2.3 ATWS Events

ATWS events are postulated to be initiated from Condition II transients. Commonly termed "Anticipated Transients" to distinguish them from the more severe, lower probability Condition III and IV transients, they include:

1.) Uncontrolled RCCA Bank Withdrawal

2.) Uncontrolled RCCA Bank Misalignment

3.) Partial Loss of Forced Reactor Coolant Flow

4.) Loss of Load and/or Turbine Trip

5.) Loss of Normal Feedwater

6.) Station Blackout (Loss of Offsite Power)

7.) Accidental RCS Depressurization

The basic design criteria for Condition II events require that they be tolerated with, at most, a shutdown of the reactor with the plant capable of returning to operation after corrective action. Fuel damage is not expected for Condition II events, although a small number of fuel rods may experience limited damage. These are within the capability of the plant clean-up systems.

A common characteristic of these events is a power generation-power removal mismatch leading to temperature excursions of the RCS. Some are characterized by increasing RCS pressures and others by RCS depressurization. It is usual to evaluate the core performance in terms of changes in the DNB ratio (or DNBR). The design duty cycle and SAR analyses report the results of these events in terms of DNBR changes. These results are used, in part, to establish the set points for the reactor protection system.

ATWS events are postulated to initiate from the Condition II transients, except that the reactor protection system is assumed to malfunction in a manner to preclude rod drop into the core. Several sets of analyses have been performed and submitted to the NRC. These analyses provide a useful basis for the development of this guideline. The ATWS analyses include five principal transients:

1.) Loss of Load/Turbine Trip

2.) Loss of Normal Feedwater

3.) Loss of Offsite Power

4.) Accidental RCS Depressurization

5.) Uncontrolled RCCA Bank Withdrawal

Reference 1 contains analyses for a more extensive list of transients, although these five are clearly the limiting transients that cover the range of ATWS overpressurization and depressurization. Reference 2 contains the analysis of these five transients with the effects of AMSAC incorporated.

In each of the five transients, there is a characteristic mismatch between power generation in the primary side and heat removal via the secondary side. This results in a primary side heat-up and a corresponding primary pressure increase (except for the "Accidental RCS Depressurization" ATWS), which continue until power production is reduced by a combination of Doppler and moderator temperature effects. The reactor then reestablishes a quasiequilibrium point at some primary pressure and temperature determined by the power generation and heat removal rates. A critical consideration is the evaluation of DNBR minima and/or RCS overpressure maxima which occur during the transient prior to reaching the quasi-equilibrium conditions. If the DNBR decreases too far, the possibility exists that fuel damage will occur. Once released into the coolant, this radioactivity is potentially capable of being released into containment and/or to the environment. If the RCS is overpressurized during the transient (ASME Service Level C is used by the NRC for this purpose (Reference 3)), the possibility that failure of the RCS pressure boundary may occur has been assumed by the NRC. To date much of the analytical work performed for ATWS events has been conducted to show that these limits are not exceeded. Each of the five ATWS transients is discussed in greater detail in the next subsection.

2.4 ATWS Analyses and Results

The ATWS transient analyses were performed using composite plant parameters to bound as many Westinghouse plants as possible, rather than using parameters for any specific plant. Sensitivity studies were performed for the five limiting cases to demonstrate that the conclusions are valid for all plants covered by the generic approach. The analyses considered 2-, 3-, and 4-loop plant configurations with 51 and 44 Series and Model D and F steam generators.

The reference plant was defined to be a 4-loop plant with 51 Series steam generators.

The description and results of the ATWS analyses are included in the following subsections. The reference case analysis, and the results of the sensitivity studies are discussed for each of the five ATWS events analyzed by Westinghouse. Greater detail on ATWS modeling and analyses can be found in References 1 and 2.

2.4.1 Loss Of External Electrical Load And/Or Turbine Generator Trip Without Reactor Trip

A major load loss could result either from a loss of external electrical load or from a turbine/generator trip. In either case, unless a loss of ac power to the station auxiliaries also occurs, offsite power would be available for the combined operation of plant components, such as the reactor coolant pumps. In this section, the loss of load accident is analyzed assuming that the control rods fail to drop into the core following a turbine trip from full power, which would produce the maximum possible load loss.

For turbine trips, the reactor normally trips directly (unless below a specific power level that is related to the amount of steam dump capacity available) from a signal derived from the turbine auto-stop oil pressure (Westinghouse Turbine) or from closure of all turbine stop valves. The automatic steam dump system opens valves to pass the excess generated steam, and, therefore, reactor coolant temperatures and pressure do not significantly increase. If the condenser is not available, the excess steam is dumped into the atmosphere through the steam generator relief and safety valves. In addition, main feedwater flow might be lost if the condenser is not available to run the turbine driven pumps; however, some feedwater flow would be supplied by the auxiliary feedwater system at a rate sufficient to remove the sensible heat of the primary coolant plus residual heat produced in the reactor.

For a complete loss of external electrical load without subsequent turbine trip, no direct reactor trip signal would be generated. Plants designed with full load rejection capability would continue operation without a reactor trip, since the mismatch between core power and turbine load would be accommodated by sufficient steam dump capacity and primary pressure relief. The Reactor Control System would bring the reactor to a turbine/generator electric load of approximately five percent after a complete loss of external electrical load to match the power requirements of the plant auxiliaries. Plants designed with less than full load rejection capability that undergo a full load rejection might possibly have the reactor trip from the first four reactor protection system signals listed in the following paragraph. Plant startup tests, however, have demonstrated that Westinghouse plants with 40 percent steam dump capacity can generally ride through a complete loss of electric load even under the most adverse operating conditions. This transient is normally less severe on the primary system than is the turbine trip transient.

If the steam dump valves fail to open following a large loss of load, or if the plant does not have full load rejection capability, the steam generator safety valves may lift since steam generator shell side pressure increases rapidly. If reactor core or primary system safety limits are approached, a reactor trip signal would be generated by the reactor trip signals which are listed below:

- o Direct reactor trip on turbine trip
- o High pressure reactor trip
- o Over temperature ΔT reactor trip
- o Low feedwater flow coincident with low steam generator water level reactor trip
- o Low-low steam generator water level reactor trip

The most severe plant conditions that could result from a loss of load occur following a turbine trip from full power when the turbine trip is caused by a loss of condenser vacuum. Since the main feedwater pumps may be turbine driven with steam exhaust to the main condenser, loss of feedwater may also result from a loss of condenser vacuum. For this reason, the low feedwater flow coincident with low steam generator water level reactor trip and the low-low steam generator water level trip are included in the above listing.

The pressurizer safety valves and steam generator safety valves are sized to protect the RCS and steam generator against overpressure for all load losses without assuming the operation of the steam dump system, pressurizer spray, pressurizer power-operated relief valves, steam generator power-operated relief valves, automatic rod control, or direct reactor trip on turbine trip. That is, the steam relief capacity of the pressurizer safety valves is selected to match the maximum pressurizer insurge following a turbine trip without credit for the items mentioned above. The steam generator safety valve relief capacity is sized to remove the steam flow at the Engineered Safeguards Design rating (~105 percent of the steam flow at rated power) from the steam generator without exceeding 110 percent of the steam system design pressure. The pressurizer safety valve capacity is sized for a complete loss of heat sink with the plant initially operating at the maximum calculated turbine load and with operation of the steam generator safety valves. The pressurizer safety valves are then able to maintain the RCS pressure to within 110 percent of the RCS design pressure without direct or immediate reactor trip action.

<u>Analysis Of Loss Of Electrical Load and/or Turbine Generator Trip Without Reactor Trip</u>

In this ATWS transient, plant behavior is evaluated for a turbine trip and loss of main feedwater occurring from full power with the assumption that the control rods fail to drop into the core following generation of a reactor trip signal. The evaluation shows the effectiveness of RCS pressure-relief devices and the extent of any approach to the core safety limits.

The following assumptions are made in the analysis:

- o Initial normal full power operation early in core life. Since the negative temperature coefficient of reactivity reduces core power as the coolant temperature rises, and the temperature coefficient becomes more negative with core life, the ATWS loss of load is less severe later in core life.
- o Normal operation of the following control systems:
  1.) Pressurizer pressure control, including heaters, spray, and both the PORVs and the safety relief valves.

  2.) Turbine governor valves in impulse pressure control prior to trip, and valve closure on turbine trip.

- o Loss of condenser vacuum at t = 0.
- o No automatic reactor trip.
- o No automatic control rod insertion as reactor coolant temperature rises.
- o Main feedwater flow falls to zero in the first four seconds of the transient, with no main feedwater flow after that time. This is consistent with a steam driven main feedwater system and a loss of condenser vacuum.
- o Auxiliary feedwater flow begins at 60 seconds at a rate of 1760 gpm.
- o Auxiliary feedwater is injected into the feedwater pipe at a temperature of 130°F, upstream of the steam generator, such that the cooler water enters the steam generator after the feedwater header is purged of main feedwater (440°F).
- o Primary to secondary heat transfer area is reduced as the steam generator shell-side water inventory drops below the level necessary to wet the tubes.

Results for Loss of Electrical Load and/or Turbine Generator Trip Without Reactor Trip

o 51 Series Steam Generator

Figures 1 through 4 show the plant transient response (using 95% MTC*) for a loss of load without reactor trip for a 4-loop plant with a 51 Series steam generator. Sequence of events for this transient is shown in Table 1. The first peak in pressurizer pressure occurs when the steam generator safety valves lift, and the second, higher peak (maximum system pressure** of 2959 psig) occurs after the pressurizer is filled with water due to a coolant volume insurge (RCS fluid swell) resulting from a rapid reduction of steam generator heat transfer. Nuclear power decreases to a value of 68 percent due to negative reactivity feedback caused by moderator (coolant) heating. Further coolant heatup, caused by loss of steam generator heat transfer, decreases nuclear power further, starting at about 110 seconds.

The DNB ratio does not drop below its initial value during the transient. At ten minutes into the transient, conditions are stabilized with auxiliary feedwater providing heat removal capability and with an intact RCS and core. Thus, the operator could begin shutdown operations through rod insertion, actuation of the safety injection system, or through use of the Chemical and Volume Control System to borate or "emergency borate".

*Since the moderator temperature coefficient is a strong function of boron concentration and a somewhat weaker function of power level, conservative assumptions were made concerning the boron concentration and xenon concentration changes in order to make the coefficient less negative. Calculations were then performed using these conservative assumptions to determine what percentage of the time the coefficient is more or less negative than a given value. The results of these calculations show that the coefficient will be more negative than -8 pcm/°F for 95% of the time (95% MTC), and more negative than -7 pcm/°F for 99% of the time (99% MTC) that the core power is greater than 80% of nominal.

**It should be noted that there is a difference between "pressurizer pressure" and "system pressure" as used here. When pressurizer pressure is given, it refers to the pressure in the pressurizer, whereas the system pressure is defined to be the pressure taken at the discharge of the reactor coolant pump, the maximum pressure in the reactor coolant system. The system pressure definition includes pump head and elevation head and will be higher than pressurizer pressure by as much as 100 psi.

TABLE 1

SEQUENCE OF EVENTS FOR LOSS OF LOAD WITHOUT A REACTOR TRIP FOR THE REFERENCE CASE*

Event	Time (seconds)
Turbine trips	0
Reactor trip signal generated on turbine trip	0
Pressurizer relief valves lift	5
High pressurizer pressure reactor trip setpoint reached	6.4
Overtemperature ΔT reactor trip setpoint reached	8.4
Steam generator safety valves lift	11
Auxiliary feed pumps begin delivering flow	60
Pressurizer safety valves lift and pressurizer fills with water	99
Maximum reactor coolant pressure (2959 psig) reached	120

*Reference case: 4-loop plant with a 51 Series steam generator, 95% MTC

Transient results for 3-loop and 2-loop plants with 51 Series steam generators are similar to those presented for the 4-loop case. A peak reactor coolant system pressure of 2847 psig results for a 3-loop plant, and a peak pressure of 2738 psig results for a 2-loop plant configuration.

Since the loss of load transient is a limiting ATWS transient with respect to peak pressure, an analysis was performed using a moderator temperature coefficient that is valid for 99% of core life. The limiting plant configuration, 4-loop, was used for this analysis.

The transient results (using 99% MTC) for a 4-loop plant with a 51 Series steam generator are similar to, but more severe than the results shown earlier. The peak RCS pressure in this case is 3069 psig.

- <u>Model D Steam Generator</u>
  The loss of load ATWS transient results for a Model D steam generator are similar to, but less severe than, the 51 Series steam generator. A 4-loop plant configuration with a Model D steam generator yields a peak reactor coolant system pressure of 2765 psig. The 3-loop Model D configuration results in a peak system pressure of 2770 psig.

- <u>Model F Steam Generator</u>
  The results of a loss of load ATWS transient for the Model F plant configuration follow those of the 51 Series steam generator closely. The base 4-loop plant with a Model F produces a peak system pressure of 2886 psig for the loss of load ATWS. The peak system pressure for a 3-loop plant configuration is 2771 psig.

- <u>44 Series Steam Generator</u>
  The transient results of a loss of load ATWS for a 44 Series steam generator are similar to the 51 Series results. The peak RCS pressure for the 4-loop configuration is 2964 psig. A 3-loop plant produces a peak system pressure of 2824 psig, and a 2-loop, 2738 psig.

<u>Sensitivity Studies for Loss of Electrical Load and/or Turbine Generator Trip Without Reactor Trip</u>

The loss of load transient without reactor trip is also analyzed with changes in certain assumptions and initial conditions to determine the effects of the RCS pressure due to each parameter change. The sensitivity studies were conducted on the base 4-loop plant with a Model 51 steam generator. The sensitivity study effects are presented as variations on this reference case, and are summarized in Table 2.

- o <u>Effect of One Auxiliary Feedwater Pump Failing to Start (Single Failure)</u>
  The single failure effect of one auxiliary feedwater pump failing to start was studied. The pump chosen for the failure is that which delivers the greatest flow, typically the turbine-driven pump. Failure of this pump effectively reduces auxiliary feedwater flow rate by one-half. The peak RCS pressure was increased by 64 psi due to the reduced flow rate.

- o <u>Effect of One PORV Failing to Open (Single Failure)</u>
  The effect of one power-operated relief valve failing to open upon demand was determined. The failure produced a net increase of 166 psi over the 4-loop, 51 steam generator reference case.

- o <u>Effect of Variations in Pressurizer Level</u>
  The initial value of the pressurizer level was varied by +10 and -10 percent to determine the effect on the peak RCS pressure. An increase in the level of 10 percent resulted in an increased peak pressure of 5 psi. Reducing the level by 10 percent resulted in a reduction of 17 psi on the peak pressure.

TABLE 2

SUMMARY OF SENSITIVITY STUDY RESULTS FOR LOSS OF LOAD WITHOUT A REACTOR TRIP

Parameter Change	Resultant Change In Maximum RCS Pressure (psi)*
One Half Auxiliary Feedwater Flow	+64
One PORV Fails to Open	+166
Pressurizer Water Level +10%	+5
Pressurizer Water Level -10%	-17
Steam Generator Water Mass +10%	+0
Steam Generator Water Mass -10%	+2
Main Feedwater Enthalpy +10%	+10
Main Feedwater Enthalpy -10%	-9
RCS Volume +10%	+42
RCS Volume -10%	-44
Auxiliary Feedwater Flow +10%	-11
Auxiliary Feedwater Flow -10%	+12
Fuel heat transfer capability +10%	-6
Fuel heat transfer capability -10%	+8
Pressurizer Spray On	-11
Reactor Power +2%	+44
Reactor Power -2%	-41
60 Second Auxiliary Feedwater Delay	+134
1085 psig Steam Generator Design Pressure	+151

* Change relative to reference case where maximum RCS pressure = 2959 psig.

- Effect of Variations in the Steam Generator Water Inventory
  The steam generator initial water mass was varied to determine peak pressure effects. An increase of 10 percent in initial water mass yielded no difference in the peak pressure. Reducing the initial mass by 10 percent resulted in an increase of only 2 psi.

- Effect of Variations in the Main Feedwater Enthalpy
  Variations in the main feedwater enthalpy resulted in small changes in the peak system pressure. An increase of 10 percent in the enthalpy yielded an increase of 10 psi over the reference case. Decreasing the enthalpy by 10 percent decreased the peak pressure by 9 psi.

- Effect of Variations in the RCS Volume
  The RCS volume was increased by 10 percent, i.e., volume was increased by 10 percent except for the pressurizer, resulting in an increase in the peak system pressure of 42 psi. A decrease of 10 percent in the RCS volume resulted in a decrease in the peak pressure of 44 psi.

- Effect of Variations in the Auxiliary Feedwater Flow Rate
  Increasing the auxiliary feedwater flow rate by 10 percent results in a decrease of 11 psi in the peak RCS pressure when compared to the reference case. Decreasing the flow rate by 10 percent increases the peak system pressure by 12 psi.

- Effect of Variations in the Fuel Heat Transfer Capability (UA)
  Variations in the fuel heat transfer capability (UA) produce small effects on the peak RCS pressure. Increasing the fuel UA by 10 percent results in a decrease of 6 psi in the peak pressure. A 10 percent decrease in the fuel UA increased the peak pressure by 8 psi.

- Effect of the Pressurizer Spray
  Since the loss of load analysis assumes that the pressurizer spray is inoperable, the effect of proper operation was studied. Assuming the pressurizer spray system operates, the peak RCS pressure is decreased by 11 psi.

- Effect of Variations in Reactor Power
  The initial reactor power was increased by 2 percent, resulting in a peak pressure increase of 44 psi. Decreasing the initial reactor power by 2 percent resulted in a decrease of 41 psi in the peak pressure.

- <u>Effect of Auxiliary Feedwater Initiation Delay</u>
  A delay of 60 seconds over the reference case for the start of auxiliary feedwater was studied. This case assumes the auxiliary feedwater pumps start 120 seconds into the transient. The peak RCS pressure is increased by 134 psi due to the delay.

- <u>Effect of Variation in Steam Generator Design Pressure</u>
  The 4-loop, 51 Series steam generator reference plant is analyzed utilizing a steam generator design pressure of 1185 psig. A sensitivity study on the reference case with a design pressure of 1085 psig resulted in an increase in peak pressure of 151 psi.

<u>Conclusions for Loss of Electrical Load and/or Turbine Generator Trip Without Reactor Trip</u>

During a loss of load with failure of rod insertion after a reactor trip signal generation, core safety limits are not exceeded since the DNB ratio does not go below its initial value. Peak RCS pressure is limited to 2959 psig for the 4-loop 51 Series steam generator reference case. Thus, no core damage or impairment of RCS integrity would occur for the "Loss of Load/Turbine Trip" ATWS.

2.4.2 Complete Loss of Normal Feedwater Without Reactor Trip

Loss of normal feedwater could result from a malfunction in the feedwater or condensate system or its control system from such causes as a simultaneous trip of all condensate pumps, a simultaneous trip of both main feedwater pumps (or closure of their discharge valves), or a simultaneous closure of all feedwater control valves. The most likely cause of a complete loss of feedwater would be loss of offsite power which is presented in Section 2.4.3.

The loss of main feedwater produces a large imbalance in the heat source/sink relationship. When feedwater flow to the steam generators is terminated, the secondary system can no longer remove all of the heat that is generated in the reactor core. This heat buildup in the primary system is indicated by rising RCS temperature and pressure, and by increasing pressurizer level, which is due to the insurge of expanding reactor coolant. Water level in the steam generators drops as the remaining water in the secondary system, unreplenished by main feedwater flow, is boiled off. When the steam generator level falls to the point where the steam generator tubes are effectively exposed and primary-to-secondary system heat transfer is reduced, the reactor coolant temperature and pressure begin to increase at a greater rate. This greater rate of primary system temperature and pressure increase is maintained as the pressurizer fills and discharges water through the PORVs and safety valves. Reactivity feedback, due to the high primary system temperature, reduces core power. Eventually, the system pressure begins to decrease, and a steam space is again formed in the pressurizer.

For protection for loss of feedwater, under normal design transient conditions, the reactor should be tripped automatically when any of the following conditions are reached:

- Steam/feedwater flow mismatch (low feedwater flow) and low steam generator water level
- Overtemperature ΔT reactor trip
- High pressurizer pressure
- High pressurizer level
- Steam generator low-low water level
- Low reactor coolant flow

None of these trips is assumed during the loss of feedwater ATWS analysis.

Analysis of Complete Loss of Normal Feedwater Without Reactor Trip

The analysis includes a moderator temperature coefficient that is valid for at least 95 percent of core life. The effect of the 99 percent value is studied for the reference case, a 4-loop, 51 Series steam generator plant configuration.

The following assumptions were made in the analysis:

- o Initial normal full power operation early in core life. Since the negative temperature coefficient of reactivity reduces core power as the coolant temperature rises, and the temperature coefficient becomes more negative with core life, the loss of feed ATWS is less severe later in core life.
- o Normal operation of the following control systems:
  1.) Both the PORVs and the safety relief valves are operable.

  2.) Turbine governor valves in impulse control prior to trip, and valve closure on turbine trip.

  3.) Steam dump to condenser at 40 percent of rated turbine flow following turbine trip.

- o Turbine trip 30 seconds after loss of feed.
- o No automatic reactor trip.
- o No automatic control rod insertion as reactor coolant temperature rises.
- o Main feedwater flow falls to zero in the first four seconds of the transient, with no main feed after that time.
- o Auxiliary feedwater flow begins at 60 seconds, at a rate of 1760 gpm.
- o Auxiliary feedwater is injected into the feedwater pipe at a temperature of 130°F, upstream of the steam generators, such that the cooler water enters the steam generator only after the feedwater header volume is purged.
- o Primary-to-secondary heat transfer area is reduced as the steam generator shell-side water inventory drops below the value necessary to cover the tubes.

Results for Complete Loss of Normal Feedwater Without Reactor Trip

- 51 Series Steam Generator

  The 4-loop, 51 Series steam generator plant configuration was analyzed for the loss of normal feedwater transient as described above. This plant configuration, assuming a 95% MTC, is considered the reference case.

  The peak pressure in the RCS for the reference case was 2833 psig and occurred approximately 106 seconds after the termination of feedwater supply to the steam generators. The pressurizer reached a peak pressure of 2731 psig at the same time.

  The chronology of events for this case is shown in Table 3 and plots are presented in Figures 5 through 8. The gradual drop in flow rate, before RCP cavitation occurs, is due to coolant expansion (density decreases). The volumetric flow rate, however, is relatively constant before the pump is assumed to cavitate.

  The loss of normal feedwater ATWS reference case was also analyzed utilizing a moderator temperature coefficient that is valid for 99 percent of core life (99% MTC). The transient results for this case are similar to the results shown in Figures 5 through 8 but are slightly more severe. The RCS pressure reached a peak of 2899 psig.

  Three-loop and 2-loop plant configurations with 51 Series steam generators were analyzed for a loss of normal feedwater ATWS to determine the effect on the peak system pressure when compared to the reference 95% case. In both plant configurations, the transient trends are similar to the reference case. The 3-loop case results in a peak system pressure of 2768 psig, while the resulting peak pressure for the 2-loop case is 2738 psig.

- Model D Steam Generator

  The loss of normal feedwater ATWS was analyzed for 4-loop and 3-loop plants with a Model D steam generator with the same assumptions described above. The resulting plant parameter transients are similar in nature to the loss of normal feedwater ATWS reference case. The 4-loop Model D plant configuration results in a peak RCS pressure of 2710 psig. The 3-loop Model D plant attained a peak pressure of 2720 psig.

- Model F Steam Generator

  The loss of normal feedwater ATWS transient results for a Model F steam generator compare closely with the results of the reference case presented in this subsection. A 4-loop Model F plant configuration yields a peak RCS pressure of 2815 psig, while a 3-loop configuration attains a peak pressure of 2735 psig.

TABLE 3

SEQUENCE OF EVENTS FOR LOSS OF FEEDWATER WITHOUT A REACTOR TRIP FOR THE REFERENCE CASE*

Event	Time (seconds)
Main feedwater supply to all steam generators is terminated	0-4
Power-operated relief valves on the pressurizer open and release steam	17
Turbine is assumed to trip	30
Reactor/turbine trip signal: overtemperature ΔT	36
Reactor/turbine trip signal: high pressurizer pressure	40
Steam generator safety valves open and hold steam pressure constant	44
All auxiliary feedwater pumps are assumed to start	60
Pressurizer fills with water	90
Peak reactor coolant system pressure reached (2833 psig)	106

*Reference case: 4-loop plant with a 51 Series steam generator, 95% MTC

- 44 Series Steam Generator
  A 44 Series steam generator plant configuration was analyzed for the loss of normal feedwater ATWS, with the assumptions described earlier in this subsection. The transients resulting from this model of steam generator are similar to the reference case. The peak RCS pressure for a 4-loop, 44 Series plant loss of normal feedwater ATWS is 2842 psig. The 3-loop plant configuration yields a 2702 psig peak pressure.

Sensitivity Studies for Complete Loss of Normal Feedwater Without Reactor Trip

The loss of normal feedwater ATWS transient reference case is evaluated with changes in certain assumptions and initial conditions to determine their effect on the results of the transient. The results of these sensitivity studies are discussed below and listed in Table 4 as variations in pressure with respect to the reference case provided in this subsection.

- <u>Effect of One Auxiliary Feedwater Pump Failing (Single Failure)</u>
  The single failure criteria of assuming that the largest auxiliary feedwater pump fails to start during a loss of normal feedwater ATWS is considered. This assumption effectively reduces the auxiliary feedwater available during the transient by one half. The effect of reduced auxiliary feedwater is a reduction in primary to secondary heat transfer, with a corresponding increase in the primary system temperature and pressure. In this case, the peak RCS pressure that is attained during the transient is increased by 31 psi over the reference case.

- <u>Effect of One PORV Failing to Open (Single Failure)</u>
  If one power-operated relief valve fails to open upon demand during a loss of normal feedwater ATWS transient, the mass and energy release is reduced, resulting in an increase in the primary system pressure. A sensitivity study on the reference case shows that under these conditions, the peak pressure reached during the transient is increased by 108 psi in the RCS.

TABLE 4

SUMMARY OF SENSITIVITY STUDY RESULTS FOR LOSS OF NORMAL FEEDWATER WITHOUT A REACTOR TRIP

Parameter Change	Resultant Change In Maximum RCS Pressure (psi)*
One Half Auxiliary Feedwater Flow	+31
One PORV Fails to Open	+108
Pressurizer Water Level +10%	+4
Pressurizer Water Level -10%	-5
Steam Generator Water Mass +10%	+0
Steam Generator Water Mass -10%	+2
Main Feedwater Enthalpy +10%	+3
Main Feedwater Enthalpy -10%	-3
RCS Volume +10%	+18
RCS Volume -10%	-12
Auxiliary Feedwater Flow +10%	-3
Auxiliary Feedwater Flow -10%	+3
Fuel Heat Transfer Capability +10%	-2
Fuel Heat Transfer Capability -10%	+3
Pressurizer Spray On	-6
Reactor Power +2%	+23
Reactor Power -2%	-15
60 Second Auxiliary Feedwater Delay	+108
1085 psig Steam Generator Design Pressure	+136
Turbine Trip at 60 Seconds	+57

* Change relative to reference case where maximum RCS pressure = 2833 psig.

- Effect of Variation in Initial Pressurizer Water Level
  Variation of ±10 percent in initial pressurizer water level is considered. The maximum pressurizer pressure attained during a loss of feedwater ATWS, with the pressurizer water level at 10 percent above the nominal level, is increased by 4 psi over the base case. Another transient which is based on a lower pressurizer water level (10 percent below nominal level) produces a maximum pressurizer pressure that is lower by 5 psi. The higher initial water level means that the pressurizer fills to capacity earlier in the transient when the core power is still relatively high. A lower than normal water level delays the filling of the pressurizer, and provides more steam for volumetric relief through the relief valves, resulting in a lower pressurizer pressure.

- Effect of Variation In Steam Generator Water Inventory
  The initial steam generator water mass was varied to determine the effect on the loss of normal feedwater transient. An increase in the initial water mass of 10 percent does not affect the peak RCS pressure. The peak pressure increases by 2 psi when the initial mass is decreased by 10 percent.

- Effect of Variation in Main Feedwater Enthalpy
  A variation in the initial main feedwater enthalpy does not appreciably affect the peak RCS pressure because the assumed transient is a loss of main feedwater. The only effect is due to the difference in purge volume enthalpy when auxiliary feedwater is initiated. An increase of 10 percent in the main feedwater enthalpy produces a 3 psi peak pressure increase. Likewise, a 10 percent decrease in enthalpy results in a 3 psi pressure decrease.

- Effect of Variation in the RCS Volume
  An increase in the RCS volume of 10 percent corresponds to an increase of 18 psi in the peak pressure over the reference case. A decrease in the volume of 10 percent yields a decrease in the peak pressure of 12 psi.

- Effect of Variation in the Auxiliary Feedwater Flow Rate
  As described in Section 2.2.1, a decrease in auxiliary feedwater flow will cause an increase in peak RCS pressures. A sensitivity study reducing the flow rate by 10 percent shows an increase of 3 psi in the peak pressure. Likewise, an increase of 10 percent in the available flow yields a decrease in the peak system pressure of 3 psi.

- Effect of Variation in Fuel Heat Transfer Capability (UA)
  Sensitivity studies with changes in the fuel UA show that there is little effect on the peak RCS pressure. An increase in fuel UA of 10 percent results in a decrease of 2 psi in the peak pressure, while a corresponding 10 percent decrease yields a 3 psi increase in the peak pressure.

- Effect of the Pressurizer Spray
  Assuming the pressurizer spray system is operable during a loss of normal feedwater transient tends to reduce the pressures attained during the transient. An analysis was done assuming proper operation of the pressurizer spray, resulting in a decrease in the peak system pressure of 6 psi.

- Effect of Variation in Reactor Power
  If the initial reactor power level is increased by 2 percent during a loss of normal feedwater ATWS, the resulting peak system pressure attained during the transient is increased by 23 psi when compared to the reference case. A decrease of 2 percent in initial reactor power results in a decrease of 15 psi in the peak pressure attained.

- Effect of Delay in Auxiliary Feedwater Initiation
  A delay in the initiation of the auxiliary feedwater system during a loss of normal feedwater ATWS affects the ability of the steam generators to remove excess heat. The reduced heat transfer ability causes the primary system pressures to be increased when compared to the reference case with no initiation delay. (Normal initiation time is 60 seconds). A 60 second delay, or auxiliary feedwater initiation within 120 seconds, results in increasing the peak primary system pressure attained by 90 psi.

- Effect of Variation in Steam Generator Design Pressure
  The steam generator design pressure assumed in the reference 51 Series case is 1185 psig. An analysis was made to determine the effect of an 1085 psig design pressure. The net result is an increase in peak pressure of 136 psi.

- Effect of Turbine Trip Delay
  In the reference loss of normal feedwater ATWS, the turbine is assumed to be tripped within 30 seconds. If the turbine is assumed to be tripped with a 30 second delay over the base case, this results in turbine trip within 60 seconds. The peak RCS pressure is increased by 57 psi for this case.

Conclusions for Complete Loss of Normal Feedwater Without Reactor Trip

Table 4 summarizes the results for the loss of feedwater ATWS reference case and sensitivity studies. The DNB ratio increases above its initial value during the transient as pressure increases. The peak RCS pressure is about 2833 psig for the reference 95% MTC case. Thus, no core damage or impairment of RCS integrity would occur for the loss of feedwater ATWS.

2.4.3 Loss of Offsite Power Without Reactor Trip

A complete loss of normal ac power to the station auxiliaries would result from a loss of offsite power combined with a trip of the station turbine/ generator.

If site and offsite power are lost, plant components requiring ac power would lose their normal power source. These components include reactor coolant pumps, condensate pumps, circulating water pumps, and main feedwater pumps (if main feedwater pumps are motor-driven). The emergency diesel generators are started on an undervoltage signal on the plant emergency busses and begin to supply vital plant loads. Emergency power is also provided by the station batteries.

Loss of power to the control rod motor/generator sets results in a loss of power to the rod drive mechanism gripper coils. This releases the rods to fall into the core independently of any protection system action to open the reactor trip circuit breakers. This method of rod release into the core is not part of the plant protection system, but is nevertheless a consequence of a loss of offsite power.

As a result of the power loss to the reactor coolant pumps, forced reactor coolant flow is lost as the pumps coast down. If the reactor is at power at the time of the transient, the immediate effect of loss of coolant flow is an increase in the coolant temperature. The decrease in flow and increase in coolant temperature causes reduced margin to DNB resulting in prompt protection system action to generate a reactor trip. There are about 25 trip inputs to the Reactor Protection System (varies slightly among plants), all of which operate on the deenergize-to-trip principle. In addition to rod mechanisms being deenergized by loss of power to the motor/generator sets, the following trip demands would occur:

- o Undervoltage or underfrequency on the reactor coolant pump power supply busses
- o Low reactor coolant loop flow
- o Open reactor coolant pump circuit breakers
- o Overtemperature ΔT
- o Overpower ΔT
- o High pressurizer pressure reactor trip
- o High pressurizer water level reactor trip

For the loss of offsite power ATWS event, none of these factors are assumed to be effective in causing rod insertion into the core.

The auxiliary feedwater system will be actuated on trip of the main feedwater pumps and/or a blackout signal during a loss of offsite power. The turbine driven auxiliary feed pump uses steam from the secondary system and exhausts to the atmosphere. The motor-driven auxiliary feed pump is supplied with power from the emergency diesel-generators. The pumps take suction directly from a condensate storage tank for delivery to the steam generators.

Analysis of Loss of Offsite Power without Reactor Trip

During a loss of offsite power when the normally expected protection system action occurs, the reactor is promptly tripped by reactor coolant pump bus undervoltage and no DNB or fuel damage occurs, even with extremely conservative initial conditions being assumed. ATWS analyses assume that loss of power to the rod power supply motor/generator sets will be disregarded (i.e., the control rods do not fall in). In addition, all of the reactor trip signals are postulated not to result in a reactor trip.

In the analysis, the following assumptions are made:

- o Initial normal full power operation early in core life. Since the negative temperature coefficient of reactivity reduces core power as the coolant temperature rises, and the temperature coefficient becomes more negative with core life, the loss of offsite power ATWS is less severe later in core life.
- o Loss of offsite ac power occurs, causing:
  1.) Reactor coolant pump coastdown in the coolant loops.

  2.) Loss of all main feedwater pumps.

  3.) Turbine trip.

  4.) Actuation of auxiliary feedwater pumps 60 seconds after start of emergency diesel generators.

- o Pressurizer relief valves are operable.
- o Remaining plant control systems are not operable as a consequence of the loss of ac power.
- o No automatic reactor trip.

Results for Loss of Offsite Power Without Reactor Trip

- 51 Series Steam Generator

  Analysis of the transient results due to a loss of offsite power ATWS was done for 51 Series steam generator plant configurations. The 4-loop plant configuration (51 Series) is considered the reference case.

  The transient results of the loss of offsite power transient for the reference 4-loop, 51 Series steam generator case are shown in Figures 9 through 12 and the sequence of events is listed in Table 5. The figures show that the rapid decrease in core flow, due to loss of the RCPs, causes a loss of secondary heat transfer with an associated rise in core inlet temperature, core average temperature, and pressurizer pressure. The minimum DNB ratio for the reference case is 1.37 at 21 seconds. Analysis utilizing a 3-loop configuration show a resulting minimum DNBR of 1.39.

  The lifting of the secondary safety valves limits the reactor coolant temperature and pressure increase. The peak pressurizer pressure is 2517 psig, and the peak pressure in the RCS is 2596 psig. An increase in pressure with a lower peak occurs later when the pressurizer fills with water. Core nuclear power decreases, due to the effect of negative reactivity feedback from a reduction in moderator density as the core average temperature increases. Core flow due to natural circulation equilibrates at about 9 percent of its nominal value. Primary coolant temperature increases during the transient, causing a slight nuclear power decrease due to the moderator heating. The steam space is recovered in the pressurizer at 500 seconds into the transient and the primary pressure begins to drop below the PORV setpoint shortly thereafter.

- Model D Steam Generator

  The loss of offsite power ATWS was analyzed for the minimum DNB ratio for plant configurations of a Model D steam generator. The general plant transients that resulted from this analysis are similar to those presented in this subsection for the reference loss of offsite power case. The 4-loop, Model D plant configuration results in a minimum DNB ratio of 1.43. The minimum DNB ratio is 1.41 for the 3-loop plants.

- Model F Steam Generator

  The transient results for the Model F loss of offsite power ATWS are similar to the reference case. The minimum DNB ratio that resulted from a 4-loop Model F loss of offsite power analysis is 1.30, with a corresponding DNBR of 1.68 for 3-loop Model F plants.

- 44 Series Steam Generator

  The loss of offsite power ATWS was analyzed for plants with a 44 Series steam generator to determine the effect of the minimum DNB ratio when compared to the reference case. The 4-loop, 44 Series plant configuration results in a minimum DNBR of 1.32.

TABLE 5
SEQUENCE OF EVENTS FOR LOSS OF OFFSITE POWER WITHOUT A REACTOR TRIP FOR THE REFERENCE CASE*

Event	Time (seconds)
Loss of site ac power and offsite ac power	0
Undervoltage reactor trip setpoint reached and underfrequency reactor trip setpoint reached	0
Low reactor coolant flow reactor trip setpoint reached	2.9
Pressurizer power-operated relief valves open	4.0
Overtemperature ΔT reactor trip setpoint reached	4.8
Pressurizer safety valves open	8
Steam generator safety valves open	13
High pressurizer water level reactor trip setpoint reached	18
Minimum DNBR occurs	21
Pressurizer fills with water	30
Auxiliary feedwater pumps start delivering flow	60
Steam space regained in pressurizer	500

*Reference Case: 4-loop plant with a 51 Series steam generator, 95% MTC

Sensitivity Studies for Loss of Offsite Power Without Reactor Trip

Sensitivity studies on parameters that could affect the minimum DNB ratio occurring during the transient have been done on the loss of offsite power ATWS. The effect of various assumptions and initial conditions on the results of the transient are described below. A summary of the studies follow and can be found in Table 6. Sensitivity studies on RCS pressure have not been included since this ATWS transient clearly is not limiting when compared to the "Loss of Load" and "Loss of Normal Feedwater" transients.

- <u>Effect of Variation in Reactor Power</u>
  The initial power level was varied for the loss of offsite power ATWS reference case to determine the effect on the minimum DNB ratios attained during the transient. An increase in initial power level of 2 percent corresponds to a decrease of 12.5 percent in the minimum DNB ratio. A decrease of 2 percent in the initial power level corresponds to an increase in the minimum DNB ratio of 10 percent.

- <u>Effect of the Pressurizer Spray</u>
  The reference loss of offsite power ATWS assumes that the pressurizer spray does not operate during the transient. An analysis assuming proper operation of the pressurizer spray shows that there is no effect on the minimum DNB ratio.

- <u>Effect of Variation in Pressurizer Level</u>
  The reference case was analyzed with changes in the initial pressurizer water level of ±10 percent. Neither case studied resulted in any significant change in the minimum DNB ratio attained during the transient.

- <u>Effect of Variation in Steam Generator Water Inventory</u>
  The initial steam generator water mass was varied by ±10 percent. The resulting analyses show that there is essentially no effect on the minimum DNB ratio.

TABLE 6
SUMMARY OF SENSITIVITY STUDY RESULTS FOR LOSS OF OFFSITE POWER WITHOUT A REACTOR TRIP

Parameter Change	Resultant Change In Mimimum DNB Ratio*
Initial Power +2%	-12.5%
Initial Power -2%	+10.0%
Pressurizer Spray On	+0%
Pressurizer Level +10%	+0%
Pressurizer Level -10%	+0%
Initial SG Mass +10%	+0%
Initial SG Mass -10%	+0%
One PORV Fails to Open	+1%

* Change relative to reference case where minimum DNB ratio = 1.37.

o Effect of One PORV Failing to Open

The reference case was analyzed with one power-operated relief valve failing to open upon demand during a loss of offsite power ATWS. The effect on the minimum DNB ratio was an increase of slightly less than 1 percent.

Conclusions for Loss of Offsite Power Without Reactor Trip

For the loss of offsite power without reactor trip the transient results show that, based upon the calculated DNB ratio, no significant clad damage is expected, and the peak RCS pressure will not cause impairment of RCS mechanical integrity.

The transient equilibrates to a condition from which the operator can begin shutdown procedures by boration, with decay heat removal and cooldown accomplished with the auxiliary feedwater system.

2.4.4 Accidental Depressurization of the Reactor Coolant System Without Reactor Trip

Depressurization of the RCS as an anticipated transient could result from accidental opening of a pressurizer relief valve, or from a leak in a sample line or instrument line connected to the RCS (Condition II events).

Two types of pressure relieving devices are provided: power-operated relief valves and spring-loaded safety valves. Continuous blowdown from either type is not likely for the following reasons:

- o The power-operated relief valves are pneumatic, air-to-open, with air pressure to the valve operator controlled by a deenergize-to-vent electric solenoid. The solenoids for the two relief valves are actuated by independent pressure control channels. A single failure in the actuation system could cause a single relief valve to open when not needed. However, an electric interlock is provided to independently close the valve on low pressure. This interlock is actuated by an independent pressure signal, and deenergizes the solenoid when pressurizer pressure drops significantly below normal operating pressure. Therefore, two independent simultaneous failures must be assumed to cause continuous relief from a power-operated pressurizer relief valve. In addition, a remotely operated block valve is located downstream of the relief valve. This block valve could be used to terminate the depressurization if the relief valve cannot be closed.

- o The spring-loaded safety valves are self-actuated by system pressure such that no external failure could cause an undesired opening. Only a major mechanical failure, such as failure of the spring, could cause the valve to remain open when the system pressure is below the set pressure. This type of mechanical failure is generally considered as an ANSI-18.2 Condition III or IV event, i.e., the probability of occurrence is too low to be considered as a Condition II event or "anticipated" transient.

Nevertheless, a 3.6 in.^{2} vent area at the top of the pressurizer is selected for evaluation purposes to bound all credible (Condition II) depressurization incidents. This size is equal to the throat area of a spring-loaded safety valve, and twice the throat area of a power-operated relief valve. The area is larger than that which could result from any credible leak in a sample line or instrument sensor line.

Of the five ATWS events analyzed, this ATWS event releases the most mass and energy into the containment. The discharge from both the relief and safety valves is routed to the pressurizer relief tank. When this tank is overpressurized, rupture discs will open and a discharge of coolant to the containment can occur. This event is presented here to show that ATWS events will not cause the containment design pressure to be exceeded.

Results for Accidental Depressurization of the Reactor Coolant System

- 51 Series Steam Generator

The accidental depressurization of the reactor coolant system ATWS was analyzed utilizing a 51 Series steam generator. The 4-loop, 51 Series plant configuration is the reference case for the depressurization ATWS. The resulting DNB ratio for this reference case is 1.60. The 3-loop plant transients are similar to the reference case with a minimum DNB ratio of 1.71.

The system transients for the reference plant are shown in Figures 13 through 16. Table 7 shows the sequence of events for the base case. Initially, the pressure decreases rapidly at a rate of about 11 psi/sec until the system pressure reaches a value corresponding to the hot leg saturation pressure. At that time the pressure decrease slows considerably. Nuclear power decreases slowly as the density decreases with reduced pressure. Following saturation in the hot leg and the upper part of the core, the lower moderator density in the core causes the nuclear power to decrease at a faster rate. This continues until the pressurizer fills with water; the lower energy relief rate then retards the rate of pressure decrease. The pressurizer fills with water because swelling of the RCS volume occurs at a rate faster than steam relief capacity through the valve. The rapidly decreasing nuclear power, together with the relatively small change in the rate of energy removal across the steam generators following hot leg saturation, causes the average temperature to decrease rapidly. The pressurizer fills with water at about 128 seconds.

TABLE 7
SEQUENCE OF EVENTS FOR ACCIDENTAL DEPRESSURIZATION OF
THE RCS WITHOUT A REACTOR TRIP
FOR THE REFERENCE CASE*

Event	Time (seconds)
Safety valve opens	0
Overtemperature ΔT reactor trip setpoint reached	16.3
Low pressurizer pressure reactor trip setpoint reached	37.9
High pressurizer water volume reactor trip setpoint reached	93.3
Pressurizer fills	128

*Reference Case: 4-loop plant with a 51 Series steam generator, 95% MTC.

Safety injection on low pressurizer pressure signal will be actuated at 50 seconds. Safety injection will further decrease nuclear power and provide makeup for coolant lost by blowdown. However, automatic safety injection is not assumed in the analysis.

Steam flow remains constant at its initial value until approximately 85 seconds. The flow then starts to decrease, following the decrease in core power and temperature. The steam generator level remains constant throughout the transient. It is conservative to assume that the normal feedwater system remains in operation during this ATWS event. The enhanced heat removal via the steam generators maximizes the RCS depressurization and leads to conservatively low values for DNBR.

The containment pressure increase, due to the mass and energy discharge from the pressurizer relief tank during the depressurization ATWS, is insignificant compared to the containment pressure produced by a loss of coolant accident (LOCA).

- <u>Model D Steam Generator</u>
  The depressurization ATWS analyzed for Model D plant configurations shows transient results that are similar to the reference case. The minimum DNB ratio for the 4-loop Model D is 1.61. A 3-loop plant configuration results in a 1.59 DNBR.

- <u>Model F Steam Generator</u>
  A minimum DNB ratio of 1.60 results for an accidental depressurization of the RCS ATWS, when a 4-loop Model F plant configuration is analyzed. The 3-loop plant also results in a 1.60 minimum DNBR.

- <u>44 Series Steam Generator</u>

  The transient results for a depressurization ATWS analyzed for a 4-loop, 44 Series steam generator follow those of the reference case closely. The DNB ratio for this case reaches a minimum of 1.45.

2.4.5 Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power Without Reactor Trip

A rod withdrawal accident could result from a Reactor Coolant System malfunction which would cause the rod speed programmer to request control rod withdrawal in the absence of either a temperature deviation or a power mismatch signal. In the event of such an occurrence, a reactor trip signal from any one of the several protection systems would terminate the rod withdrawal.

The result of an uncontrolled rod withdrawal would be the addition of positive reactivity to the reactor core resulting in an increase in core nuclear power and thermal flux. Because the heat extraction from the steam generator lags the increasing core power generation, the reactor coolant temperature rises, and if no action terminates the process, DNB may occur in the core resulting in possible fuel and cladding damage. Because of the nature of the transient, the magnitude of the nuclear and thermal excursions and the margin to DNB in the core are primarily a function of the total excess reactivity inserted by the rods and is only slightly affected by the rates of reactivity insertion.

There are several features of the automatic Reactor Protection System which normally would act to prevent core damage in the event of this accident. These include the following coincident trips:

- o Two power range nuclear flux instrumentation channels in excess of the nuclear overpower setpoint actuate a reactor trip.
- o Two ΔT channels exceeding the overtemperature ΔT setpoint actuate a reactor trip. The setpoint is automatically varied with axial power distribution, reactor coolant temperature, and reactor coolant pressure to protect against DNB.
- o Two ΔT channels exceeding the overpower ΔT setpoint actuate a reactor trip. This setpoint is also automatically varied with axial power distribution to ensure that the allowable transient heat generation rate is not exceeded.
- o Two pressurizer level channels exceeding a fixed high pressurizer water level setpoint actuate a reactor trip.
- o Two pressurizer pressure channels exceeding a fixed high pressure setpoint actuate a reactor trip.

In addition to the above reactor trip functions, the following Rod Cluster Control Assembly withdrawal blocking setpoints would be reached in the event of an uncontrolled rod withdrawal transient:

- o One power range nuclear flux channel exceeding a high nuclear flux setpoint.
- o Two ΔT channels exceeding a overtemperature ΔT setpoint.
- o Two ΔT channels exceeding a overpower ΔT setpoint.

<u>Analysis of Uncontrolled Rod Cluster Control Assembly Bank Withdrawal At Power Without Reactor Trip</u>

The following assumptions were made in the analysis:

- o Initial normal full power operation early in core life. Since the negative moderator temperature coefficient of reactivity limits the overtemperature-overpower transient, and the moderator coefficient becomes more negative during core life, rod withdrawal later in core life would be less severe than the case studied. The moderator temperature coefficient assumed is valid for 95% of core life.
- o Normal operation of the following control systems:
  1.) Automatic regulation of feedwater flow to maintain steam generator water level.

  2.) Both the PORVS and the safety valves on the pressurizer.

  3.) Turbine governor valves in impulse pressure control.
- o No automatic reactor trip.
- o No automatic rod stops.
- o Continuous rod withdrawal at maximum rod speed of 45 in./minute (72 steps/minute) until control rods are fully withdrawn.

Results for Uncontrolled Rod Cluster Control Assembly Bank Withdrawal At Power Without Reactor Trip

- 51 Series Steam Generator

  The "Rod Cluster Control Assembly Bank Withdrawal at Power" ATWS was analyzed for a 4-loop, 51 Series steam generator plant configuration. This is the reference rod withdrawal at power ATWS case.

  As the rods are withdrawn core power increases. This, in turn, increases core temperature and RCS pressure because of the mismatch between core power and secondary plant power. The high nuclear flux trip was reached approximately 12.8 seconds after the rods began to be withdrawn. Core power increased to about 113 percent and core average temperature to about 609°F. The nuclear power increase was stopped by Doppler and moderator feedback. The rapid insurge into the pressurizer resulted in opening of the PORVS and a peak RCS pressure of 2413 psig.

  After the initial surge in power, the core power and secondary power extraction stabilized at about 99 percent of the nominal power level. The inserted reactivity was balanced by moderator feedback due to increasing core average temperature until the rod withdrawal ceased at 62 seconds.

  Figures 17 through 20 show the transient response of this type plant to a 0.3% $\Delta k/k$ rod withdrawal from 100-percent power. An inserted rod worth of 0.3% $\Delta k/k$ is typical of the available control rod reactivity at 100-percent power. Table 8 lists the sequence of events and the time of their occurrence during the transient. Included in the table are the times at which various Reactor Protection System trip points are reached. The minimum DNB ratio for the reference case during the transient was 1.58. A 3-loop plant configuration results in a minimum DNBR of 1.58 also.

- Model D Steam Generator

  The rod withdrawal at power ATWS was studied for plants with a Model D steam generator. The transient results are similar to those described above. A 4-loop, Model D analysis resulted in a minimum DNB ratio of 1.59, slightly better than the 51 Series steam generator case.

- Model F Steam Generator

  Plants with Model F steam generator were also studied for a rod withdrawal at power ATWS, with transient results similar to the reference case. A 4-loop, Model F configuration minimum DNB ratio was 1.54, while a 3-loop Model F resulted in a minimum DNB of 1.55.

- 44 Series Steam Generator

  The rod withdrawal at power ATWS analyzed with a 44 Series steam generator resulted in a similar transient to the reference case. The minimum DNB ratio of 1.51 was slightly lower than the reference case.

TABLE 8

SEQUENCE OF EVENTS FOR ROD WITHDRAWAL AT POWER WITHOUT A REACTOR TRIP FOR THE REFERENCE CASE*

Event	Time (seconds)
Rod withdrawal begins	0.0
High nuclear flux trip setpoint reached	12.8
Overtemperature ΔT reactor trip setpoint reached	15.7
Overpower ΔT reactor trip setpoint reached	18.5
Pressurizer power-operated relief valves open	20.0
Pressurizer high level reactor trip setpoint reached	96.2

*Reference Case: 4-loop plant with 51 Series steam generator, 95% MTC

<u>Conclusions for Uncontrolled Rod Cluster Control Assembly Bank Withdrawal At Power Without Reactor Trip</u>

Based upon the calculated DNB ratios, no significant clad damage is expected. In addition, since RCS pressures are below limiting values, no damage to the RCS is expected from an uncontrolled rod withdrawal at power without trip.

2.5 Discussion of Analytical Results

The ATWS event analyses discussed in the previous subsections, and the more comprehensive reports listed as References 1 and 2 have provided a strong basis for addressing the licensing issues associated with ATWS. They provide assurance that the licensing performance criteria can be met.

The ATWS event analyses contain several conservative assumptions that are not entirely addressed by the sensitivity studies performed. In addition to assuming a non-specified, non-mechanistic failure of the Reactor Protection System in order to preclude rod insertion, several other potentially beneficial systems have also been assumed to be non-functional. In all ATWS event analyses, the Engineered Safeguards Features actuation system was assumed to be non-responsive to signals which typically would have resulted in generation of the SI actuation signal. For example, in the case of RCS depressurization, the primary pressure would have fallen below the SI setpoint, but no credit was taken for the boron which would have been added as a result. In another example, no credit was taken in any of the ATWS analyses for operation of the rod control system. Sensitivity studies performed in WCAP-8330 (Reference 1) indicate that operation of this system would have lowered peak RCS pressures by approximately 100 psi for the "Loss of Normal Feedwater" ATWS.

The "Loss of Load" ATWS and "Loss of Normal Feedwater" ATWS events have the largest predicted RCS overpressure. Fuel integrity, as measured by DNBR, is not an issue for these ATWS events. However, for both transients, there are less severe types which would not necessarily show such large departures from the design/SAR transient. For example, the NRC has indicated (Reference 3) that approximately 70% of the events which they regard as ATWS precursors involve turbine trip for which steam dump to the condenser is available. (The remaining 30% of the transients are conservatively lumped as loss of normal feedwater transients by the NRC.) The loss of load ATWS event analyzed is the most severe overpressurization transient. However, if the condenser were available for steam dump and continued operation of the main feedwater system is assumed, the overpressurization transient would be less severe and may result in the "Loss of Normal Feedwater" ATWS being the limiting case with respect to overpressurization. If the design transients for loss of load and loss of normal feedwater are examined, it will be seen that even with reactor trip the RCS pressure rises to values between the pressurizer PORV setpoint and the pressurizer safety valve setpoint. For these cases, there would be steam relief, not water relief as in the ATWS event transients. Thus, for the vast range of ATWS events where secondary side heat removal is a major consideration, it may be reasonable to expect RCS pressures only slightly elevated compared to the non-ATWS design transients. RCS pressure transients cannot be used as a sole determinant of ATWS events; the usual indicators of reactor trip (rod bottom lights, nuclear flux rates, etc.) must be evaluated to confirm presence of an ATWS event.

The "Loss of Off-Site Power" ATWS presents only moderate RCS pressure increases, but is more of a concern with respect to DNBR than either the "Loss of Load" ATWS or "Loss of Normal Feedwater" ATWS events. The predominant reason for this is that it typically causes a trip of the reactor coolant pumps resulting in reduced primary coolant flow. For the "Loss of Offsite Power" ATWS, it can be seen that nuclear power generation declines more rapidly and RCS temperature and pressure increase less than in either of the two previously discussed ATWS events. DNBRs stay at acceptably high values.

The "Depressurization" ATWS transient does not result in an RCS overpressure condition. DNBR stays at acceptable levels although lower than in the design transient.

The "Rod Withdrawal" ATWS event results in a mild overpressure transient. The peak pressure will lie between the pressurizer PORV and safety valve setpoints; however, if more realistic assumptions about the moderator temperature coefficient were made, the peak RCS pressure may not even reach the PORV setpoint. DNBR values are reduced relative to the design transient, but remain acceptable.

As seen from these results, the RCS conditions at the time the operator identifies an ATWS event can be very different depending on the initiating event. In addition, the analyses has shown that having a turbine trip within 30 seconds and actuation of AFW within 60 seconds provides acceptable consequences for the limiting RCS overpressurization ATWS event - loss of load plus loss of normal feedwater. Since time is a critical concern in mitigating certain ATWS events, the required operator actions must be independent of the initiating ATWS event. Therefore, guideline FR-S.1 contains immediate actions for tripping the reactor and tripping the turbine (necessary for a loss of normal feedwater ATWS). Though some of these actions may degrade RCS conditions for particular ATWS events, they are necessary to cover the wide range of initiating events.

Figure 1. NUCLEAR POWER VERSUS TIME FOR A LOSS OF LOAD ATWS

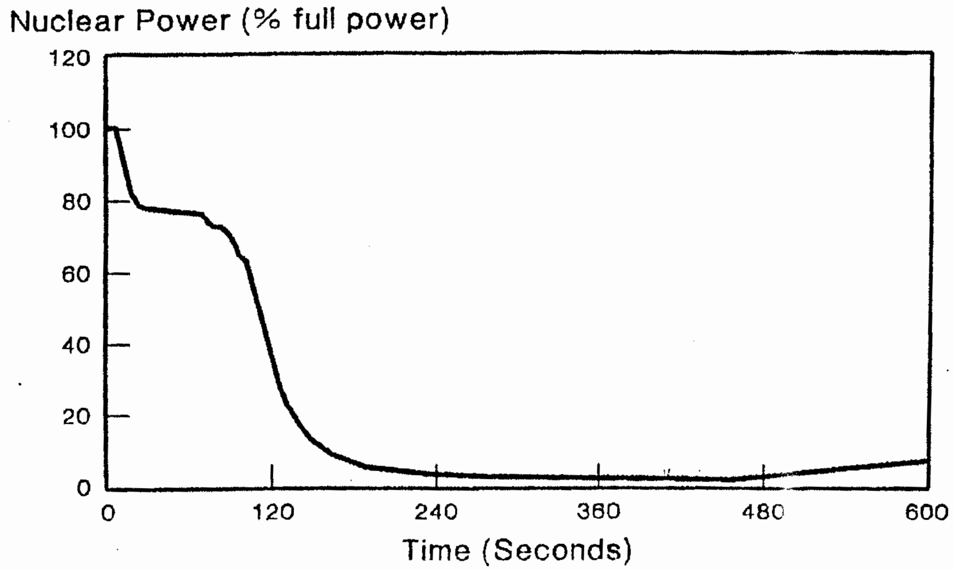

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Figure 2. RCS AVERAGE COOLANT TEMPERATURE VERSUS TIME FOR A LOSS OF LOAD ATWS

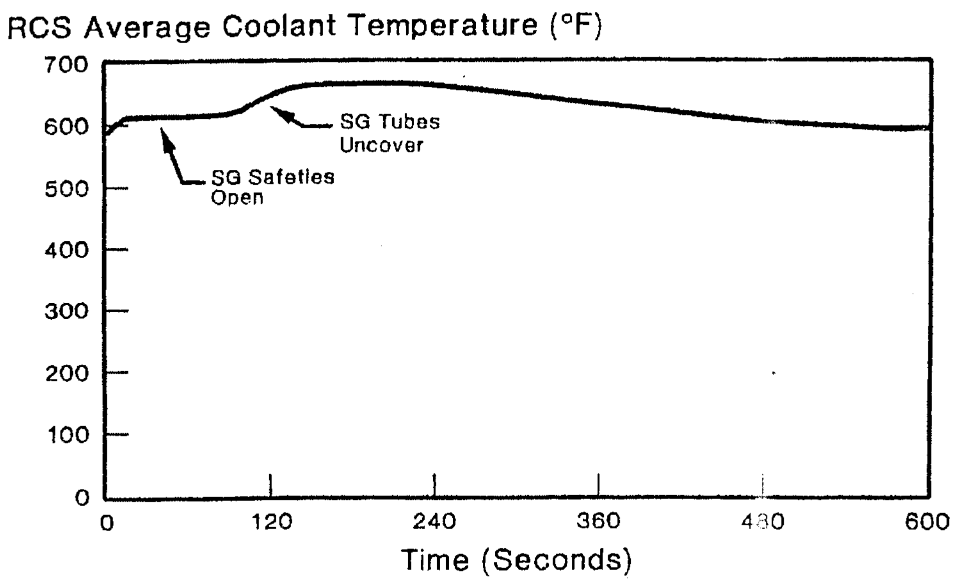

Figure 3. PRESSURIZER PRESSURE VERSUS TIME FOR LOSS OF LOAD ATWS

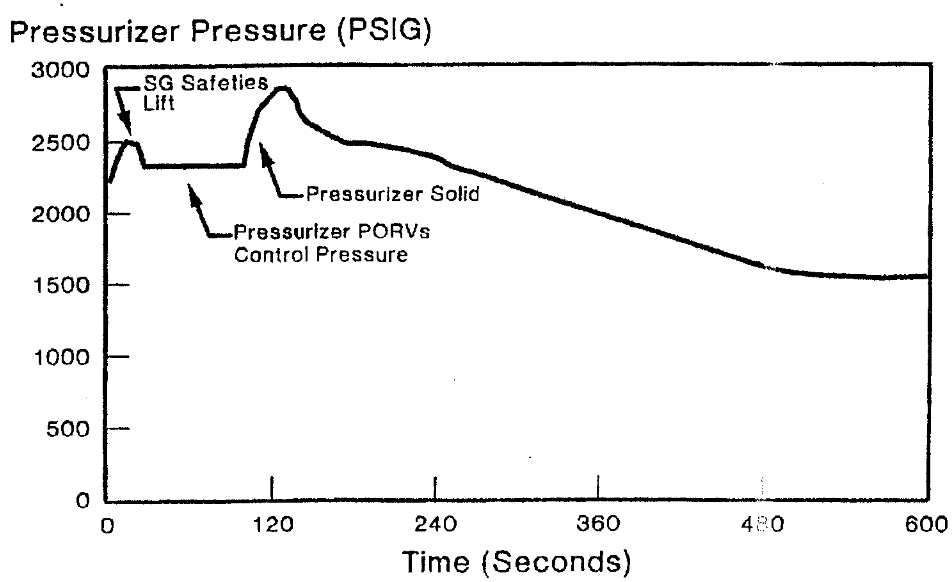

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Figure 4. PRESSURIZER LEVEL VERSUS TIME FOR A LOSS OF LOAD ATWS

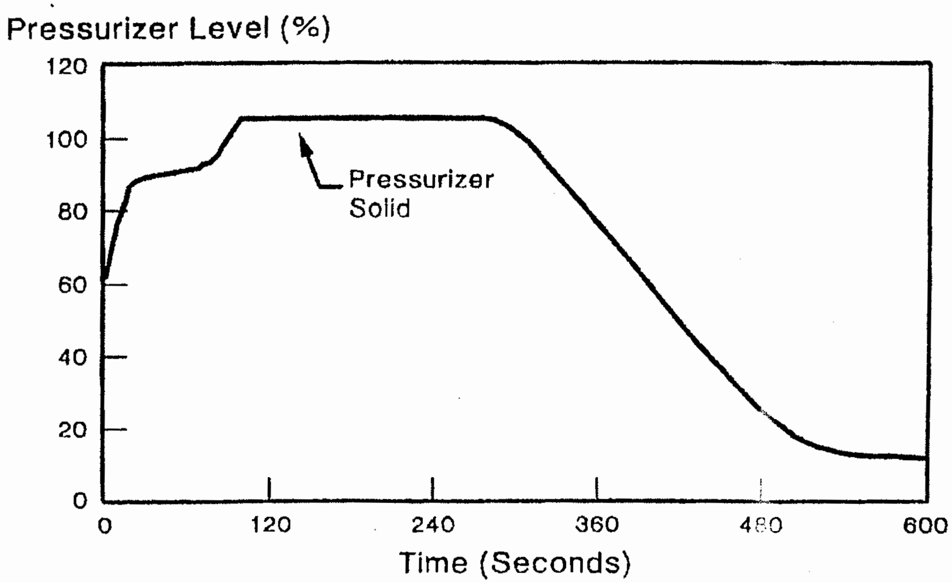

Figure 5. NUCLEAR POWER VERSUS TIME FOR A LOSS OF NORMAL FEEDWATER ATWS

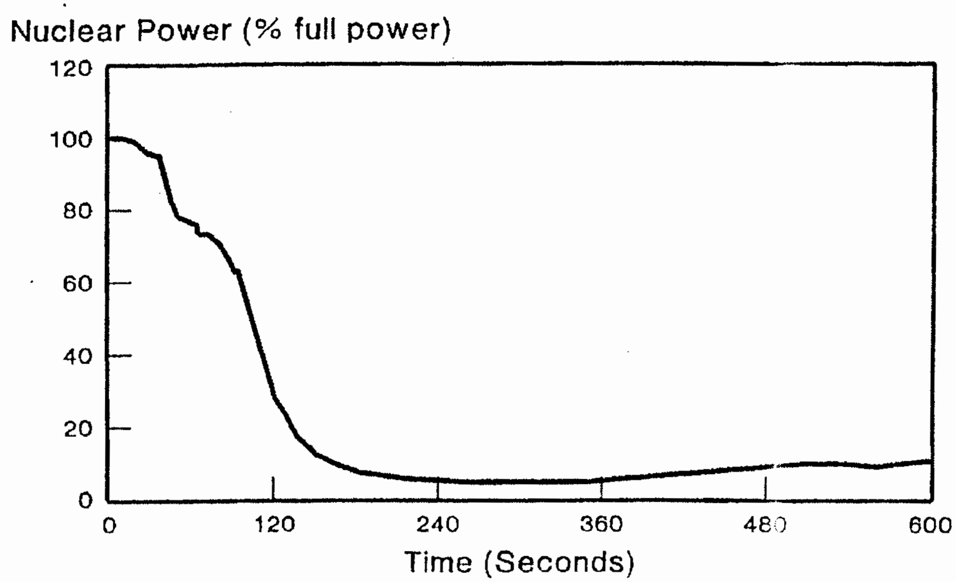

Figure 6. RCS AVERAGE COOLANT TEMPERATURE VERSUS TIME FOR A LOSS OF NORMAL FEEDWATER ATWS

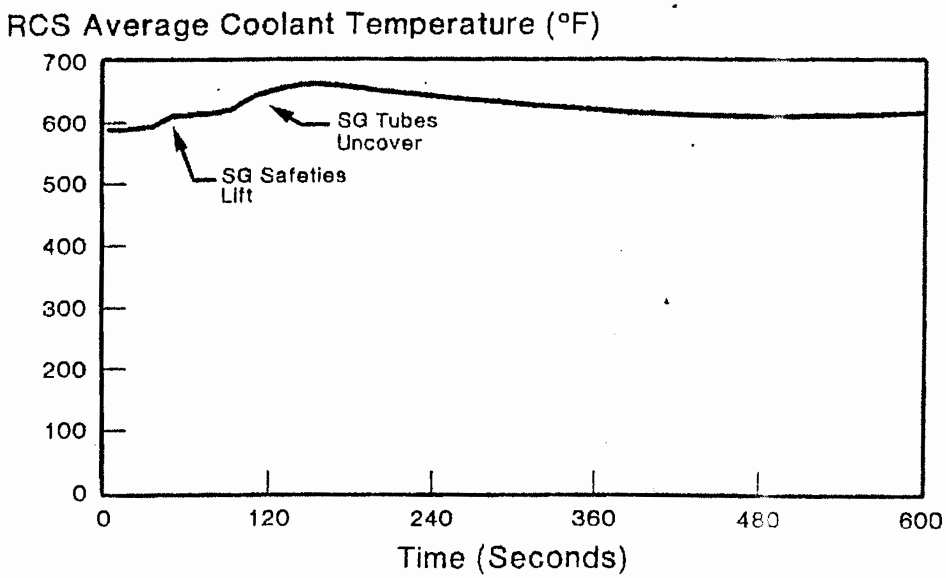

Figure 7. PRESSURIZER PRESSURE VERSUS TIME FOR A LOSS OF NORMAL FEEDWATER ATWS

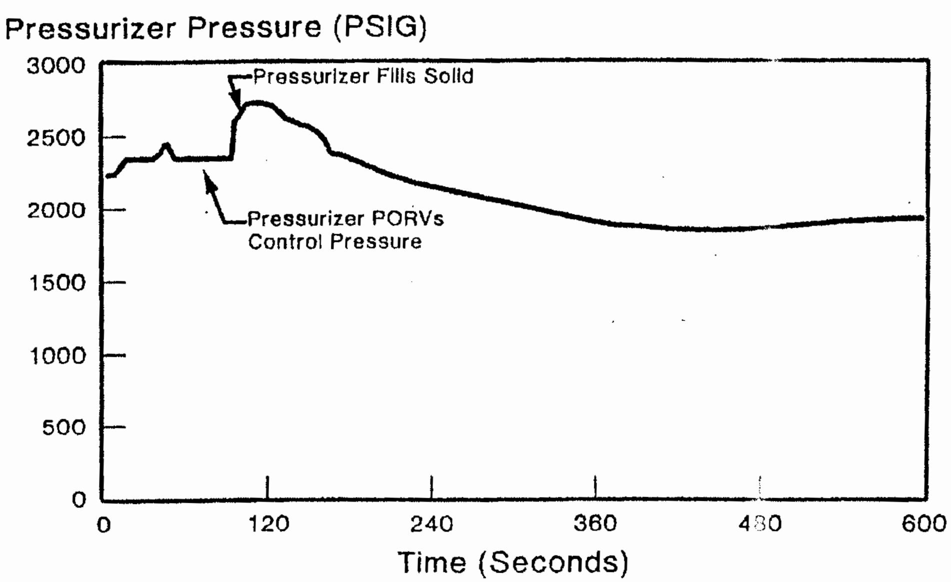

Figure 8. PRESSURIZER LEVEL VERSUS TIME FOR A LOSS OF NORMAL FEEDWATER ATWS

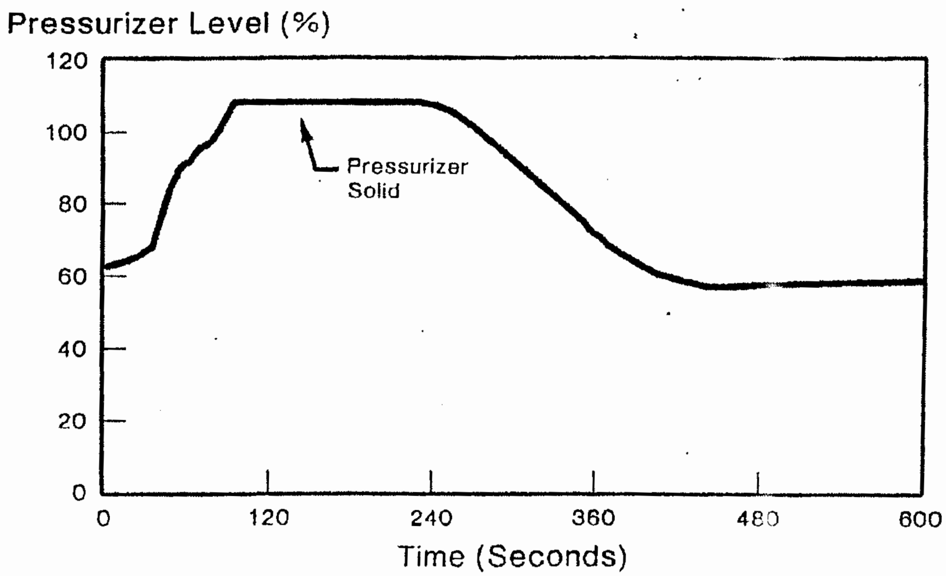

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Figure 9. NUCLEAR POWER VERSUS TIME FOR A LOSS OF OFFSITE POWER ATWS

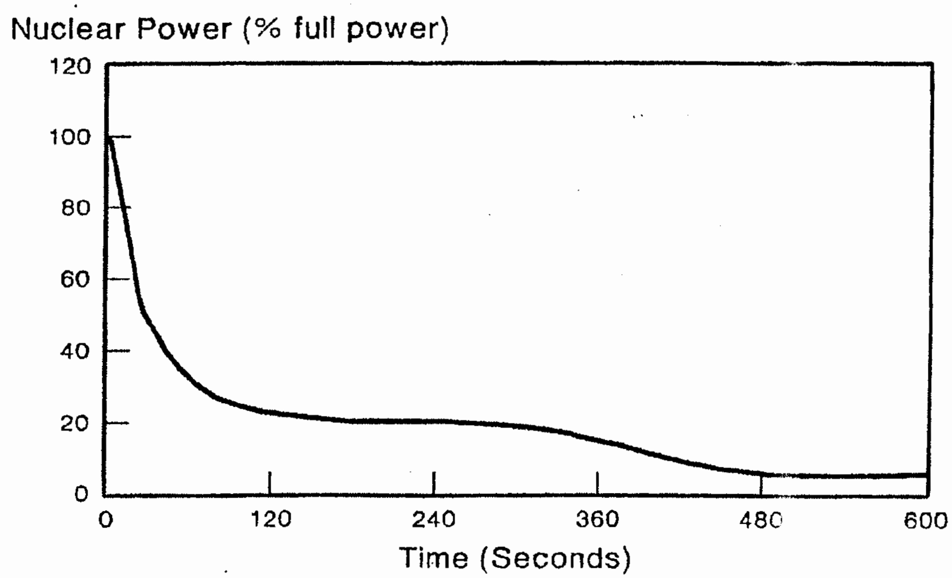

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Figure 10. RCS AVERAGE COOLANT TEMPERATURE VERSUS TIME FOR A LOSS OF OFFSITE POWER ATWS

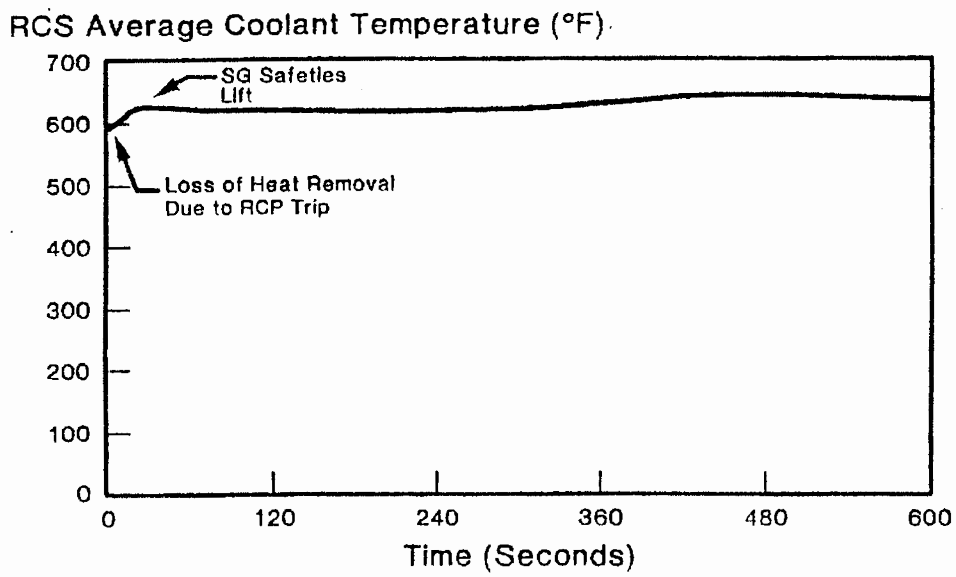

Figure 11. PRESSURIZER PRESSURE VERSUS TIME FOR A LOSS OF OFFSITE POWER ATWS

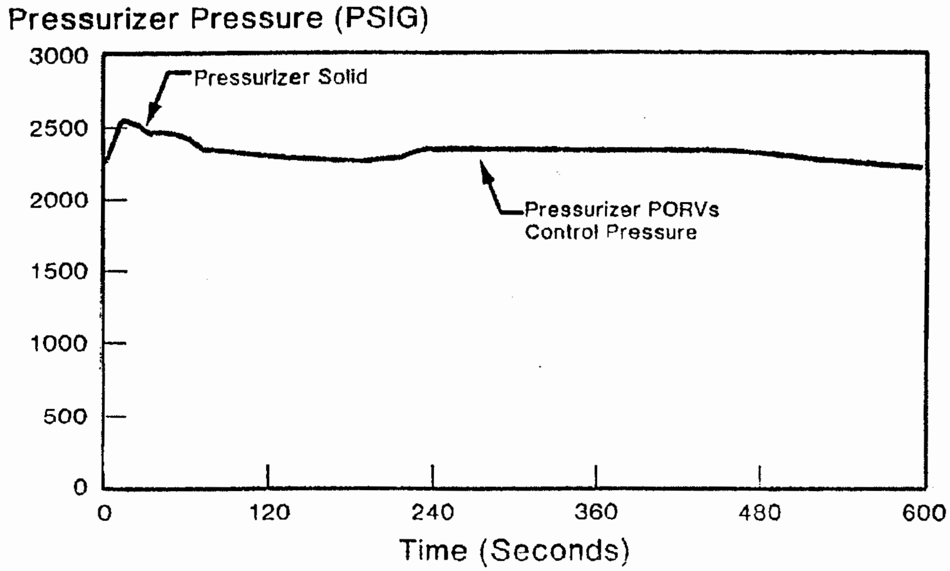

Figure 12. PRESSURIZER LEVEL VERSUS TIME FOR A LOSS OF OFFSITE POWER ATWS

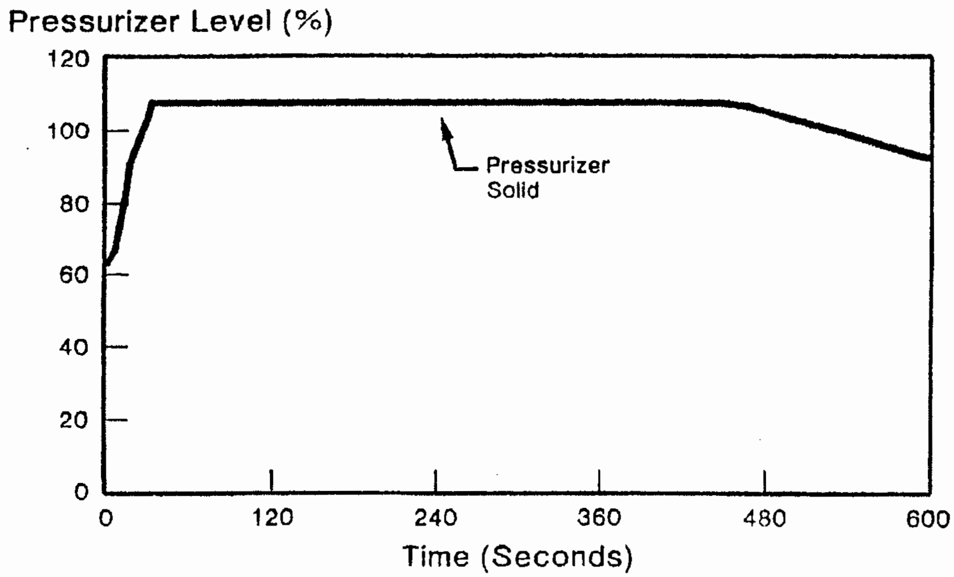

Figure 13. NUCLEAR POWER VERSUS TIME FOR AN ACCIDENTAL RCS DEPRESSURIZATION ATWS

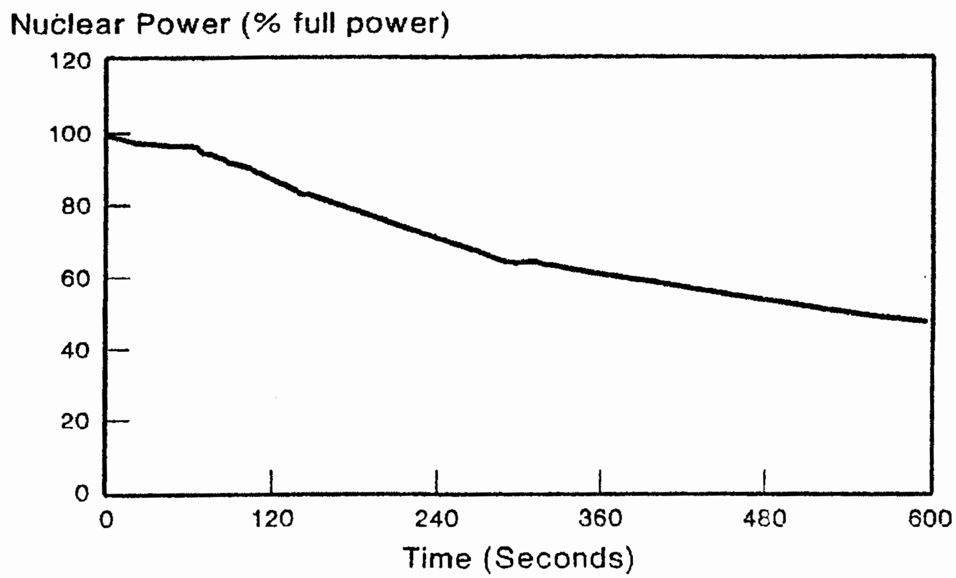

Figure 14. RCS AVERAGE COOLANT TEMPERATURE VERSUS TIME FOR AN ACCIDENTAL RCS DEPRESSURIZATION ATWS

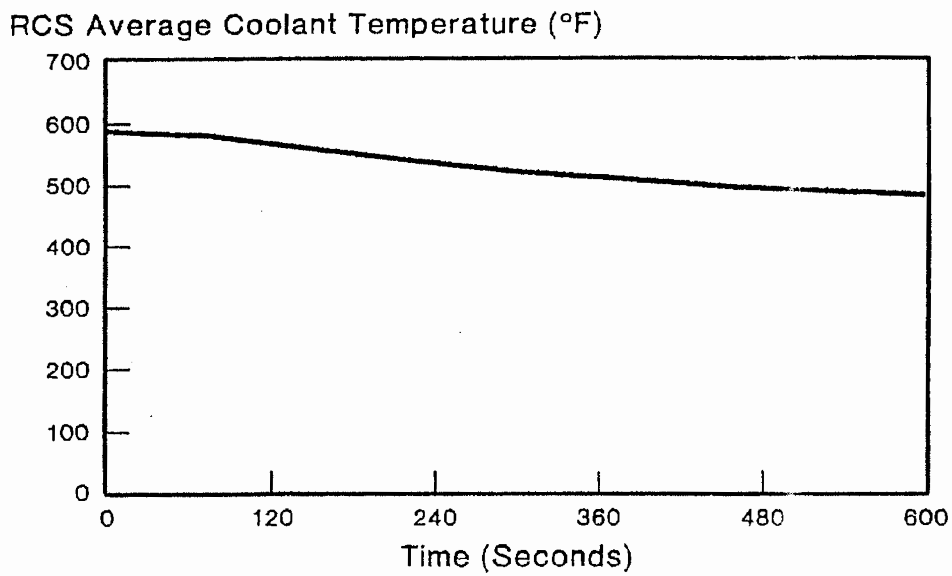

Figure 15. PRESSURIZER PRESSURE VERSUS TIME FOR AN ACCIDENTAL RCS DEPRESSURIZATION ATWS

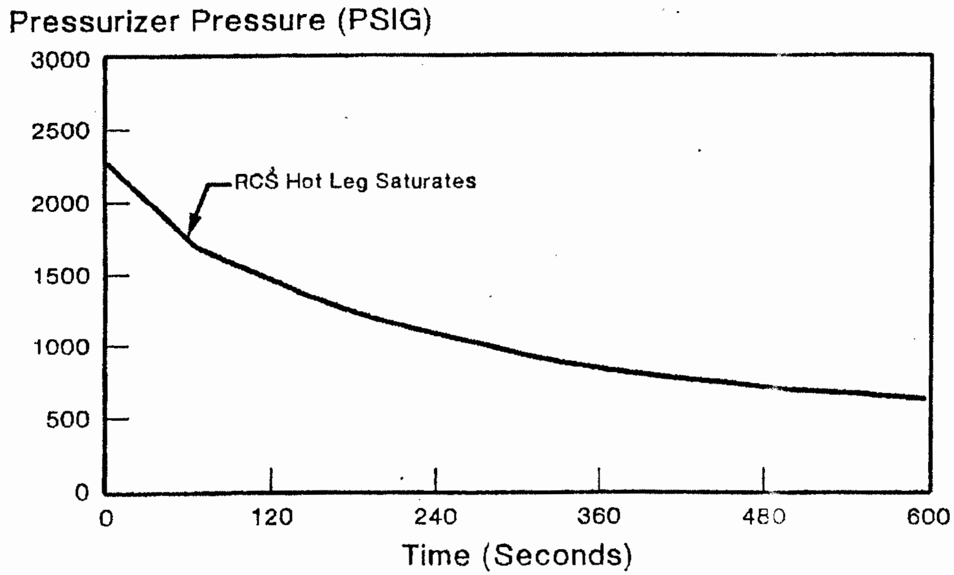

Figure 16. PRESSURIZER LEVEL VERSUS TIME FOR AN ACCIDENTAL RCS DEPRESSURIZATION ATWS

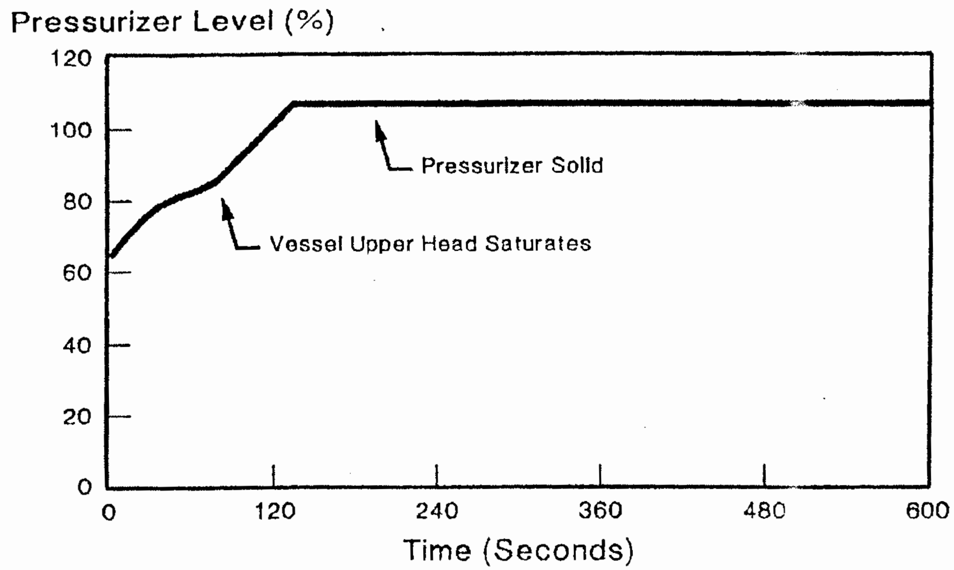

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Figure 17. NUCLEAR POWER VERSUS TIME FOR AN UNCONTROLLED ROD WITHDRAWAL ATWS

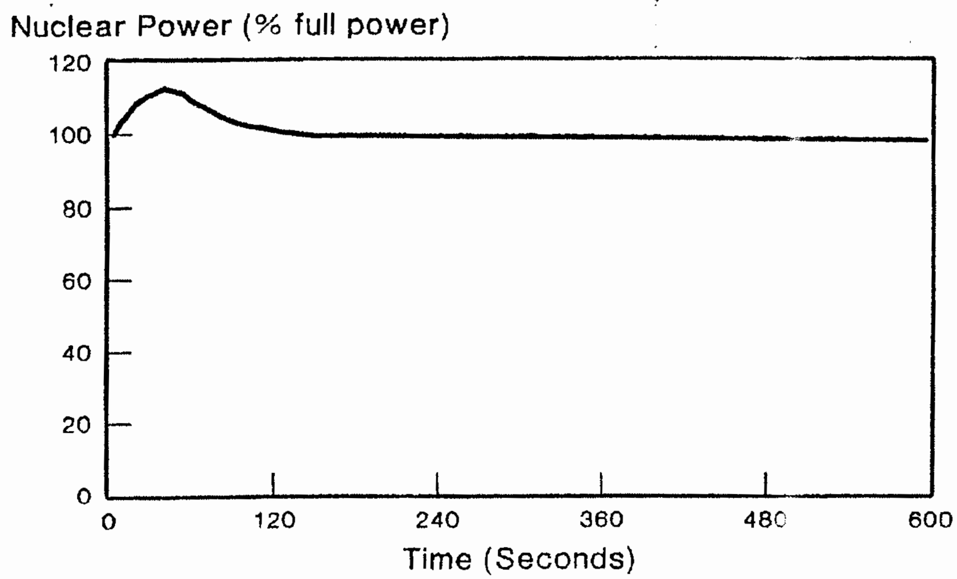

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Figure 18. RCS AVERAGE COOLANT TEMPERATURE VERSUS TIME FOR AN UNCONTROLLED ROD WITHDRAWAL ATWS

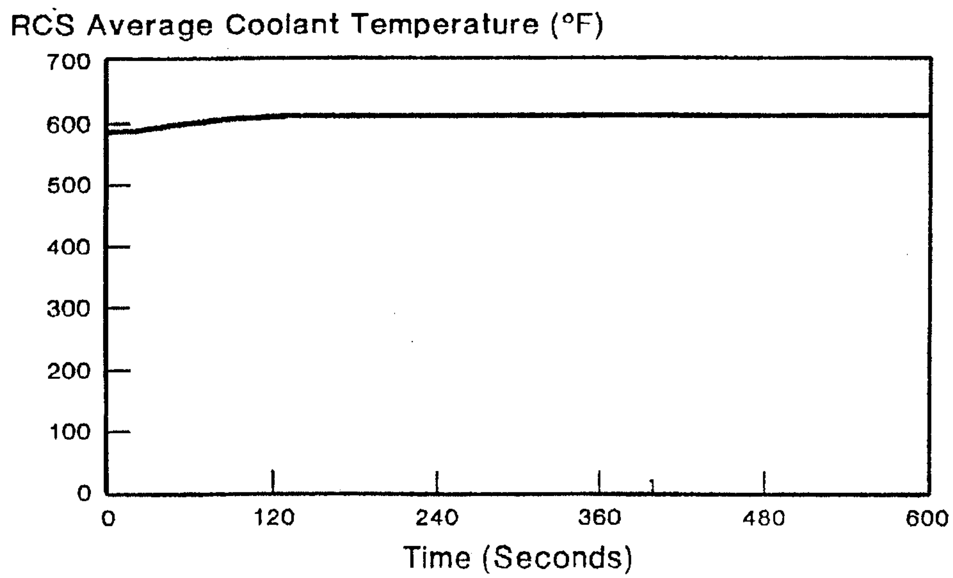

Figure 19. PRESSURIZER PRESSURE VERSUS TIME FOR AN UNCONTROLLED ROD WITHDRAWAL ATWS

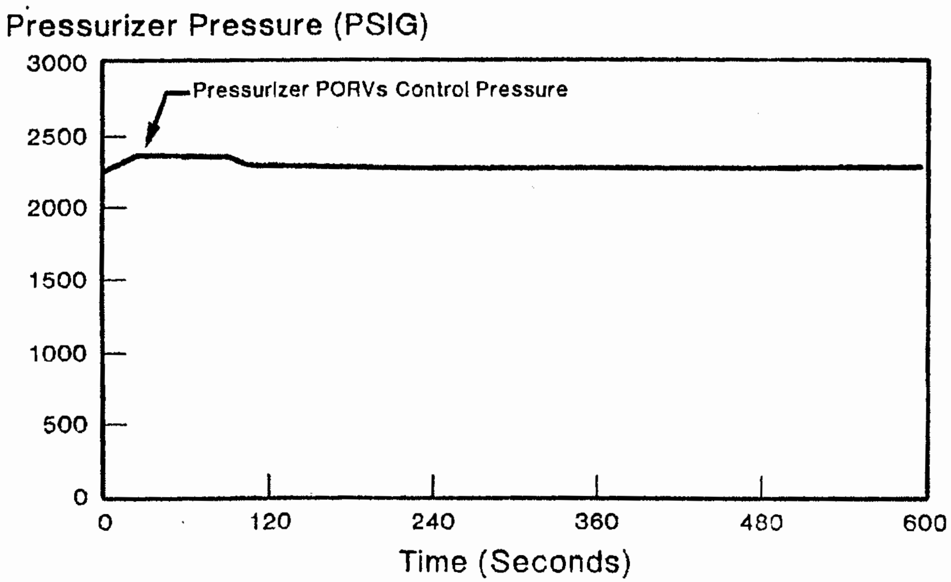

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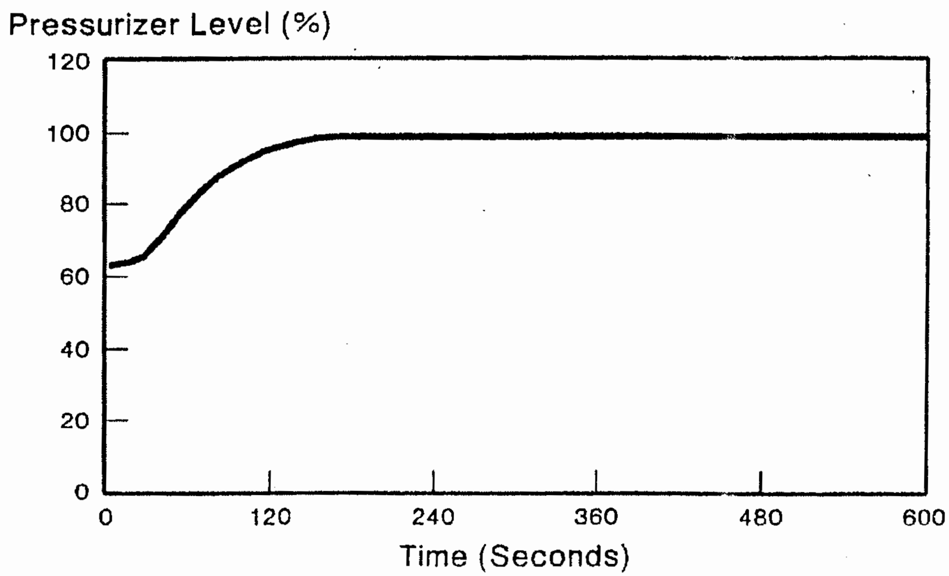
Figure 20. PRESSURIZER LEVEL VERSUS TIME FOR AN UNCONTROLLED ROD WITHDRAWAL ATWS



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3. RECOVERY/RESTORATION/TECHNIQUE

The objective of the recovery/restoration technique incorporated into procedure FR-S.1 is to add negative reactivity to the core after it has been determined to be critical when it should be shut down.

The following subsections provide a summary of the major categories of operator actions and the key utility decision points for procedure FR-S.1, RESPONSE TO NUCLEAR POWER GENERATION/ATWS.

3.1 High Level Action Summary

A high level summary of the actions performed in FR-S.1 is given in the form of major action categories. These are discussed below in more detail.

MAJOR ACTION CATEGORIES IN FR-S.1

- o Verify Automatic Actions or Perform Manual Actions to Reduce Core Power
- o Emergency Borate
- o Check for Possible Sources of Positive Reactivity and Eliminate Them
- o Verify Subcriticality

- Verify Automatic Actions or Perform Manual Actions to Reduce Core Power

  A full reactor trip is the preferred way to shut down the reactor. If automatic functions have still not been effective, any manual trips from the control room are to be actuated. If these are still not effective, the rods should be inserted using the Rod Control System. Subcriticality is checked to determine the effectiveness of these manual actions.

  Supplemental turbine trip and AFW actuation checks provide consistency with the supporting ATWS analysis.

- Emergency Borate

  Several methods of emergency borating are available in the control room. This action is taken prior to initiating more time-consuming local actions to trip the reactor and/or turbine.

- Check for Possible Sources of Positive Reactivity and Eliminate Them

  Possible sources of positive reactivity are checked and eliminated at this time. Actions include isolation of all dilution paths and identification/isolation of faulted SG(s) causing an uncontrolled RCS cooldown.

  These actions address the return-to-power condition and will probably not be required for an ATWS.

- Verify Subcriticality

  This final action checks on the effectiveness of previous steps in mitigating the transient prior to departing the guideline. Departure is not allowed until subcriticality is verified.

3.2 Key Utility Decision Points

In the development of plant specific EOPs, the utility should generate lists of alternate means of boration and various means of performing control room and local trips (both reactor and turbine). At this time thought should be given to prioritization of the methods listed, considering both the time required and personnel available. Since time is a critical factor during a full power ATWS event, such advance prioritization will expedite plant recovery.

4. DETAILED DESCRIPTION OF PROCEDURE

This section provides a very detailed discussion of the generic procedure FR-S.1 to facilitate utility EOP writing and training efforts. By presenting guideline background information in greater detail through the use of a structured format (i.e., step description tables, step sequence tables, and logic diagrams), plant specific applicability can be more easily determined. The separate and unique subsections containing this information follow.

4.1 Detailed Description of Steps, Notes, and Cautions

This section contains a one-page (or more) step description table for each separate procedure step, note, and caution. Notes and cautions are always presented relative to the step they precede. Refer to the Users Guide in the Executive Volume for a discussion on the use of the step description tables.

The Step Description Tables for the 16 steps and associated notes of procedure 2-FR-S.1 are presented on the following pages.

CAUTION: RCPs should NOT be tripped with Reactor Power GREATER THAN 5%.

PURPOSE: To inform the operator that the RCPs should not be tripped even if all normal running conditions are not satisfied

BASIS:

During an ATWS, RCP operation could be beneficial by temporarily cooling the core under voided RCS conditions. If reactor power is greater than 5%, the RCPs should not be tripped even if all normal running conditions are not satisfied. Manually tripping the RCPs during some ATWS events could result in reduced heat removal and a challenge to fuel integrity. An ATWS is not a design basis event; therefore the licensing requirement to trip the RCPs within a timely manner to remain within the small-break LOCA design basis is not applicable.

KNOWLEDGE:

This caution is applicable during the performance of the Immediate Action Steps and should be known by the operator without availability of the written guideline.

NOTE: Steps 1 and 2 are IMMEDIATE ACTION steps.

PURPOSE: To remind the operator that the first 2 steps are immediate actions

BASIS:

Immediate actions are those actions which the operator should be able to perform before opening and reading the emergency procedures. In general, immediate actions are limited to time critical actions that verify automatic protection features of the plant but are not so complex or extensive that reliance on procedure is preferred to reliance on memory.

Although the immediate actions should be memorized by the operator, they need not be memorized verbatim. The operator should know them well enough to complete the intent of each step. The order in which they should be performed should also be consistent with the procedure.

KNOWLEDGE:

The intent of the immediate action steps should be committed to memory

STEP: Verify Reactor Trip:

PURPOSE: To ensure that the reactor has tripped

BASIS:

Reactor trip must be verified to ensure that the only heat being added to the RCS is from decay heat and reactor coolant pump heat. The safeguards systems that protect the plant during accidents are designed assuming that only decay heat and pump heat are being added to the RCS. If the reactor cannot be tripped, then the control rods should be manually inserted into the core in order to decrease reactor power.

KNOWLEDGE:

If RCS temperature has increased above the current reference temperature, then the rods should automatically be driven in by the Rod Control System. This action satisfies the intent of the contingency requirement.

For non-adverse containment conditions the NIS power range instruments may be used to determine if the reactor has tripped. If adverse containment conditions exists, environmentally qualified instruments must be used (i.e., core exit TCs, T_{HOT} and T_{COLD}) since IP2 NIs are NOT qualified and may be inaccurate or inoperable under adverse containment conditions. (See Document ID RA-95-175, TAC No. M81727)

STEP: Verify Turbine Trip:

PURPOSE: To ensure that the turbine is tripped

BASIS:

The turbine is tripped to prevent an uncontrolled cooldown of the RCS due to steam flow that the turbine would require. For an ATWS event where a loss of normal feedwater has occurred, analyses have shown that a turbine trip is necessary (within 30 seconds) to maintain SG inventory.

If the turbine will not trip, a turbine runback (manual decrease in load) at maximum rate will also reduce steam flow in a delayed manner. If the turbine stop valves cannot be closed by either trip or runback, the MSIVs should be closed.

KNOWLEDGE:

A turbine trip is required for an ATWS event where a loss of main feedwater has occurred. For other ATWS events, with the exception of when a turbine trip is the initiating event, manual tripping of the turbine may yield a somewhat higher system pressure, depending on the initiating event and time in core life, than what would otherwise be expected. However, this action has been determined to be necessary due to the analytical results presented and discussed in subsections 2.4, ATWS Analysis and Results, and 2.5, Discussion of Analytical Results. Since there are many initiating ATWS events and some that require immediate mitigating actions, diagnosis of the initiating event would not be feasible and separate guidance for different ATWS events would complicate training and could delay timely performance of necessary operator actions.

STEP: Check AFW Pumps Running:

PURPOSE: To ensure AFW pumps are running

BASIS:

The MD AFW pumps start automatically on an SI signal or SG low level to provide feed to the SGs for decay heat removal. If SG levels drop below the appropriate setpoint, the turbine-driven AFW pump will also automatically start to supplement the MD pumps. The ATWS analyses have shown that actuation of AFW within 60 seconds after the failure to scram/trip provides acceptable results.

KNOWLEDGE:

N/A

STEP: Initiate Emergency Boration of RCS:

PURPOSE: To add negative reactivity to bring the reactor core subcritical

BASIS:

After control rod trip and rod insertion functions, boration is the next most direct manner of adding negative reactivity to the core. The intended boration path here is the most direct one available using the normal charging pump(s).

Several methods are available for rapid boration and are specified in order of preference. Methods of rapid boration include emergency boration via MOV-333, switching charging pump suction to the RWST with maximum charging flow or normal boration with maximum boric acid and charging flow. To ensure flow is NOT impeded from the RWST via LCV-112B, the RCS Makeup Control switch is placed in STOP. To allow for maximum flow during a boration using positive displacement charging pumps, it will be necessary to transfer to manual and increase the speed controller of these pumps. Once the charging pumps(s) are started, charging pump suction is aligned to the RWST. This ensures that a continuous borated suction supply is maintained.

The check on RCS pressure is intended to alert the operator to a condition which would reduce charging pump injection into the RCS and, therefore, boration. The PRZR PORV pressure setpoint is chosen as that pressure at which flow into the RCS is insufficient. The contingent action is a rapid depressurization to a pressure which would ensure maximum charging flow. When primary pressure drops 200 psi below the PORV pressure setpoint, the PORVs should be closed. The operator must verify successful closure of the PORVs, closing the isolation valves, if necessary.

KNOWLEDGE:

The RCS makeup control switch is placed in stop to ensure flow is delivered to the charging pumps suction from the RWST. RCS makeup would impede flow due to primary water pressure maintaining check valve 290 in the closed position.

CAUTION: Radiation levels and harsh environment conditions should be evaluated prior to performing local actions.

PURPOSE: To remind the operator to perform an evaluation of the local environment prior to sending personnel out to perform local actions.

BASIS:

Depending on the local area in question and the event which is in progress, the local environmental conditions may have a high temperature, high pressure, or high radiation. An evaluation of the pressure, temperature, and radiation level in the local area should be performed prior to dispatching personnel to that local area. ALARA procedures must be considered.

KNOWLEDGE:

High radiation, high pressure, or high temperatures may exist in the local areas where actions need to be taken. ALARA procedures must be followed.

STEP: Verify Containment Ventilation Isolation:

PURPOSE: To ensure non-essential containment ventilation penetrations are isolated

BASIS:

Non-essential ventilation penetrations are isolated to prevent potential release of radioactive materials from containment.

This step is addressed in 2-FR-S.1 in accordance with the ATWS analytical case of the "Accidental Depressurization of the RCS Without Reactor Trip" (See WOG background Section 2.0, page 48) which results in the most releases of mass and energy into the containment. As a result, verification of containment ventilation should conservatively always be performed independent of the RCS pressure.

KNOWLEDGE:

The above valves are located in the fan room, and have local position control. If they cannot be closed either from the control room or fan room, send an operator with an HP technician to locally close outside valves by isolating instrument air.

CAUTION: If an SI signal exists or occurs, Steps 1 through 9 of 2-E-0, REACTOR TRIP OR SAFETY INJECTION, should be performed while continuing with this procedure.

PURPOSE: To alert the operator that he should verify proper actuation of all SIS actuated equipment.

BASIS:

It is possible to make a transition to this procedure without having performed the verification of automatic SI actions in 2-E-0. This caution specifically instructs the operator to perform the verification. This verification is started after Steps 1 through 5 of 2-FR-S.1 since the first five steps deal directly with ATWS mitigation while the 2-E-0 actions deal with system alignment for design basis events.

KNOWLEDGE:

Verification of automatic SI actions should be initiated and performed in parallel with the subsequent steps of this procedure as manpower and time permit.

STEP: Check If The Following Trips Have Occurred:

PURPOSE: To determine if earlier control room actions were successful in producing reactor and turbine trips and, if not, to initiate local actions

BASIS:

Reactor trip is the fastest mechanism for adding negative reactivity to the reactor core. Turbine trip removes a large source of positive reactivity addition (heat removal by steaming), and will conserve SG inventory for the limiting ATWS event. If any of these actions have not been successfully achieved when attempted from the control room, an operator should be dispatched to perform the actions locally. Local actions were delayed until now because they will be more time consuming to initiate and complete, but may still be effective. Local reactor trip actions are performed first since the sooner a trip is obtained the less severe the ATWS transient will be.

KNOWLEDGE:

N/A

STEP: Check If Reactor Is Subcritical:

PURPOSE: To check if previous actions were successful in returning the reactor to a subcritical condition

BASIS:

Previous actions to trip the reactor and insert control rods may have been successful in adding sufficient negative reactivity to the reactor core to return the core to a subcritical condition. This step specifies two conditions which must both be satisfied to verify that the reactor is indeed subcritical. Power range channels below 5%% ensure that the heat load to available heat sinks is just the decay heat level normally accommodated with AFW flow. The negative intermediate range startup rate ensures that the reactor is subcritical. Notice that no degree of subcriticality is specified and, therefore, any negative startup rate is acceptable.

If the reactor is verified to be subcritical (the RED or ORANGE priority no longer exists on the Subcriticality Status Tree), the operator should continue plant recovery operations by returning to the guideline and step that was in effect at the time 2-FR-S.1 was entered. The transition to the last step in the guideline ensures that reactivity effects are addressed prior to exiting the guideline. If either of the specified conditions for subcriticality is not satisfied, the operator is directed to the next step in the guideline to continue to address reactivity concerns.

KNOWLEDGE:

This step is a continuous action step

CAUTION: City water for AFW pumps will be necessary if CST level decreases to less than 2 ft.

PURPOSE: To alert the operator that CST level should be monitored, and that switching to city water as an alternate source may be necessary.

BASIS:

If CST level decreases below 2 ft, inadequate suction pressure may result in AFW pump damage and require manual pump trip. To ensure continued AFW pump operation, the suction of the AFW pumps must be connected to the city water supply if the CST is depleted.

KNOWLEDGE:

There are check valves in the AFW pump suction lines from the CST which may not seat properly when city water is valved in. An NPO may have to be dispatched to close the manually operated valves in those AFW pump suction lines.

STEP: Check SG Levels:

PURPOSE: To ensure that sufficient AFW flow is present to remove heat generated from power operation during an ATWS event or a return to criticality

BASIS:

ATWS analyses have shown that AFW flow of 800 gpm is acceptable to adequately remove the heat generated from power operation prior to reactor shutdown. If AFW flow is not greater than 800 gpm, it is important to increase AFW flow in order to maintain a secondary heat sink. For the loss of normal feedwater ATWS, the SG tubes are uncovered in about two minutes.

For other transients, such as a return to criticality, this feed flow requirement would be excessive. Narrow range SG level can be maintained with lower AFW flow rates. As long as level can be maintained with the lower flow rate, the higher flow rate is not necessary.

KNOWLEDGE:

This step is a continuous action step. (DW-93-025)

AFW flow is preset for 200 gpm per SG, or a total flow of 800 gpm.

STEP: Verify All Dilution Paths - ISOLATED:

PURPOSE: To insure that any possible dilution path is isolated

BASIS:

A possible cause of power generation would be an inadvertent dilution of the RCS. Removal of this source of positive reactivity will make the boration performed earlier more effective.

Since the control room operator is not able to completely verify the isolation of some potential dilution paths, it may be necessary to dispatch an operator locally to verify the proper alignment of the manual valves in these dilution paths.

KNOWLEDGE:

N/A

STEP: Check For Reactivity Insertion From Uncontrolled RCS Cooldown:

PURPOSE: To see if an uncontrolled or controlled RCS cooldown is in progress

BASIS:

An uncontrolled cooldown of the RCS is indicated by either an uncontrolled RCS temperature decrease or an uncontrolled SG pressure decrease. Such an RCS cooldown could add a significant amount of positive reactivity to the core, depending on the current value of the moderator temperature coefficient.

If an uncontrolled cooldown is not in progress, the operator is instructed to stop any controlled cooldown and proceed to Step 14. The actions required for stopping the controlled cooldown could include closing the atmospheric or condenser steam dump valves if steaming was in progress, or reducing AFW flow to that of one MD AFW pump if the maximum flow was established in Step 8 in response to low SG level. These actions, within the control of the operator, could reduce the RCS cooldown to minimize the amount of positive reactivity that is being added to the core. Once the controlled RCS cooldown has been addressed, the identification and isolation of the faulted steam generator(s) is bypassed and the next action (Step 14) is to determine if core exit TCs are less than 1200°F.

KNOWLEDGE:

- o "Uncontrolled" means not under the control of the operator, and incapable of being controlled by the operator using available equipment. The intent of this step is not to identify a Faulted Steam Generator based on a decreasing pressure due to an RCS cooldown (or other known cause) even though it may not be under the control of the operator. If the rate at which pressure is decreasing is small or the cause is known, it should not be considered DECREASING IN AN UNCONTROLLED MANNER. (DW-98-045)
- o Since the severity of a reactivity addition due to uncontrolled cooldown is directly a function of the current moderator temperature reactivity coefficient, training should emphasize the increased magnitude of this parameter at high-burnup, low-ppm core condition
- o "Controlled" means capable of being controlled through operator action using available equipment. With the identification of a controlled cooldown, the operator can proceed to minimize the effect (stop steaming, reduce AFW flow) of the cooldown on adding positive reactivity to the core.

STEP: Check MSIVs - CLOSED

PURPOSE: To determine if MSIVs are closed

BASIS:

This step is only performed if an uncontrolled RCS cooldown is observed, and is the initial step in isolating any faulted SG(s).

KNOWLEDGE:

The instructions in Steps 11 through 13 are intended to limit any uncontrolled cooldown caused by a faulted steam generator (secondary side break). Allowance is made for multiple (but less than all) steam generators to be faulted.

The main steam bypass valves can allow uncontrolled steam release if open. These valves are closed during power operation and can only be closed locally.

STEP: Identify Faulted SG(s):

PURPOSE: To identify any faulted (failure in secondary pressure boundary) SG(s)

BASIS:

An uncontrolled SG pressure decrease (following MSIV closure) or a completely depressurized (near containment or atmospheric pressure) SG indicates an unisolable failure of the secondary pressure boundary. These symptoms are sufficient to identify the affected SG(s).

KNOWLEDGE:

"Uncontrolled" means not under the control of the operator, and incapable of being controlled by the operator using available equipment. The intent of this step is not to identify a Faulted Steam Generator based on a decreasing pressure due to an RCS cooldown (or other known cause) even though it may not be under the control of the operator. If the rate at which pressure is decreasing is small or the cause is known, it should not be considered DECREASING IN AN UNCONTROLLED MANNER. (DW-98-045)

CAUTION: At least one SG must be maintained available for RCS cooldown.

PURPOSE: To alert the operator that at least one SG must be available as a heat sink for decay heat removal and RCS cooldown

BASIS:

During the attempt to determine the faulted loop(s), the operator must maintain at least one loop available for cooldown capability. Otherwise, RCS pressure and temperature will increase if all SGs are isolated.

KNOWLEDGE:

System transient characteristics and symptoms for different size breaks

CAUTION: If all SGs are faulted, at least 85 gpm feed flow should be maintained to each SG.

PURPOSE: To alert the operator to maintain a minimum feed flow to minimize any subsequent thermal shock to SG components

BASIS:

IF feed flow to a SG is isolated and the SG is allowed to dry out, subsequent reinitiation of feed flow to the SG could create significant thermal stress conditions on SG components. Maintaining a minimum verifiable feed flow to the SG allows the components to remain in a "wet" condition, thereby minimizing any thermal shock effects if feed flow is increased.

KNOWLEDGE:

N/A

CAUTION: If the turbine-driven AFW pump is the only available source of feed flow, steam supply to the turbine-driven AFW pump must be maintained from one SG.

PURPOSE: To alert the operator that the steamline to the turbine-driven AFW pump must not be isolated if it is the only source of feed flow to the steam generators

BASIS:

If the turbine-driven AFW pump is the only operable source of feed flow to the steam generators (i.e., no other MD AFW pumps or other operable pumps are capable of providing feed flow to the SGs), then isolation of its steam supply line may degrade system conditions and result in a transition to 2-FR-H.1. Therefore, this isolation must not be performed.

KNOWLEDGE:

N/A

STEP: Isolate Faulted SG(s):

PURPOSE: To isolate all feedwater to and steam flow from the faulted SG(s)

BASIS:

Isolation of the feedwater to the faulted SG maximizes the cooldown capability of the nonfaulted loops following a feedline break and minimizes the cooldown and mass and energy release following a steamline break. Isolation of steam paths from the faulted SG also minimizes the RCS cooldown and mass and energy release to the containment. In addition, isolation of these steam paths could isolate the break.

KNOWLEDGE:

Recovery actions affected by isolation of the faulted SG or secondary break

STEP: Check Core Exit TCs - LESS THAN 1200°F

PURPOSE: To ensure severe conditions do not exist that require a transition to the SAMGs

BASIS:

The Severe Accident Management Guidelines (SAMGs) are entered from the EOPs by the control room operators when core damage occurs. The EOP to SAMG transition uses, as part of the transition criteria, a core exit thermocouple temperature indication of greater than 1200°F to indicate the need to transition from the EOPs to the SAMGs. The 1200°F criteria for transition from the EOPs to the SAMGs is identical to the 1200°F criteria on the Core Cooling Critical Safety Function Status Tree.

If the operator enters this step and core exit TC temperatures are greater than 1200°F and increasing, the operator should transition to SACRG-1, Step 1. This condition indicates that all attempts to restore core cooling have failed and core damage can not be prevented and the operator should go to the SAMGs.

If the operator enters this step and core exit TC temperatures are less than 1200°F or core exit TC temperatures are greater than 1200°F and decreasing, the operator will stay in the loop between steps 4 and 15 in procedure 2-FR-S.1 to continue efforts to emergency borate the RCS and check for sources of positive reactivity.

KNOWLEDGE:

N/A

STEP: Verify Reactor Subcritical:

PURPOSE: To see if previous actions were successful in returning the reactor to a subcritical condition

BASIS:

By this time all attempts to identify and isolate the most obvious sources of positive reactivity addition to the RCS have been performed. Furthermore, the boration initiated in Step 4 may already have had some effect in returning the core to a subcritical condition. Hence, a check on subcriticality is in order. This step specifies conditions which must be satisfied to verify that the reactor is indeed subcritical. Power range channels below 5% ensure that the heat load to available heat sinks is just the decay heat level normally accommodated with AFW flow. The negative intermediate range startup rate ensures that the reactor is subcritical. Notice that no degree of subcriticality is specified and, therefore, any negative startup rate is acceptable.

For non-adverse containment conditions the NIS power range instruments may be used to determine if the reactor is subcritical. If containment conditions are adverse, the neutron flux indications may be incorrrect due to non-qualified instruments so boration should be continued. (See Document ID RA-95-175, TAC No. M81727)

If any of the above conditions for verification of subcriticality are not satisfied, the operator is directed to continue the boration. If boration is not available, then the RCS should be allowed to heat up in order for the negative reactivity feedback mechanisms (moderator temperature coefficient and Doppler effect) to take effect in reducing nuclear power.

In addition, actions of other Function Restoration Procedures in effect can be performed at this time (even though the Subcriticality Status Tree may still indicate a RED or ORANGE priority) as long as they do not cool down or otherwise add positive reactivity to the core. The operator is then returned to Step 4 of 2-FR-S.1 to continue efforts to emergency borate the RCS and check for sources of positive reactivity.

KNOWLEDGE:

Other FRPs in effect can mean previous FRPs in effect or lower priority FRPs that may be identified, e.g., high containment pressure.

CAUTION: Boration should continue during subsequent actions until adequate shutdown margin is obtained.

PURPOSE: To inform the operator of the required extent of boration after the reactor is subcritical

BASIS:

Subcriticality (negative startup rate) is a sufficient condition to satisfy the Subcriticality Critical Safety Function RED or ORANGE priority. However, establishing adequate shutdown margin reflects a conservative operational approach.

KNOWLEDGE:

Confirmation of subcriticality is a minimum condition for terminating the boration currently in progress. Prior to termination, the operator should consider future actions in the recovery process and their effect on core reactivity. The "should continue" wording of the caution suggests that the current level of subcriticality may not be adequate throughout these future actions. Boration may be terminated at this time, based on the judgement that near-term actions will not add positive reactivity to the core. However, the operator should be aware that additional boration may be necessary in the future, for example, during cooldown operations.

STEP: Return To Procedure And Step In Effect

PURPOSE: To direct the operator to the proper procedure following successful completion of the steps in this procedure

BASIS:

Since the reactor has been verified to be subcritical (the RED or ORANGE priority no longer exists on the Subcriticality Status Tree), the operator should continue plant recovery operations by returning to the procedure and step that was in effect at the time 2-FR-S.1 was entered.

KNOWLEDGE:

N/A

4.2 Step Sequence Requirements

This section consists of a table which presents the existing WOG sequence and identifies the allowed interchangeability of procedure steps for the benefit of the utility EOP writer. The WOG High Level step wording is typically used for future referencing.

The Step Sequence Table for FR-S.1 is provided on the following page. The interchangeability of procedure steps is identified by the numbers in the column to the right of each procedure step. Refer to the Users Guide in the Executive Volume for information on use of the step sequence tables.

STEP SEQUENCE FOR FR-S.1

STEP		SEQUENCE
1	Verify Reactor Trip	1
2	Verify Turbine Trip	2
3	Check AFW Pumps Running	3
4	Initiate Emergency Boration Of RCS	4
5	Verify Containment Ventilation Isolation	5
6	Check If The Following Trips Have Occurred	6
7	Check If Reactor Is Subcritical	7
8	Check SG Levels	8
9	Verify All Dilution Paths - ISOLATED	9
10	Check For Reactivity Insertion From Uncontrolled RCS Cooldown	10
11	Check Main Steamline Isolation And Bypass Valves - CLOSED	11
12	Identify Faulted SG(s)	12
13	Isolate Faulted SG(s)	13
14	Check Core Exit TCs - LESS THAN 1200°F	14
15	Verify Reactor Subcritical	15
16	Return To Guideline And Step In Effect	16

5. FREQUENT QUESTIONS

The following are questions which have been frequently asked about FR-S.1, RESPONSE TO NUCLEAR POWER GENERATION/ATWS:

Q.

If the ATWS event results in rising RCS temperatures, why are the control rods inserted manually? Auto rod control would be faster.

A.

For those events which involve rising temperatures, it is true that leaving the rod control system in AUTOMATIC will cause rods to drive in at 72 steps per minute (assuming the turbine is tripped and reference temperature is no-load) compared to the 48 steps per minute speed in MANUAL. In this combination of circumstances, the AUTOMATIC function satisfies the intent of this step. However, for other ATWS events, rod insertion in MANUAL will be required.

Q.

What is the expected ERG usage if SI is actuated to implement the "Emergency Boration?"

A.

Assuming that the reactor is promptly shut down due to the boration, the operator is returned to the guideline and step in effect. For an ATWS, this is E-0, REACTOR TRIP OR SAFETY INJECTION, Step 1, where the operator must recognize that the equivalent of a trip has been implemented, and the reactor is shut down. He would then proceed in E-0, terminate the SI as spurious using ES-1.1, and then transition to some normal plant procedure.

If entry to FR-S.1 is due to a return-to-power following reactor trip without SI actuation, then the transition would be back to ES-0.1, REACTOR TRIP RESPONSE, at the appropriate step. However, a CAUTION at the front of ES-0.1 states "If SI actuation occurs during this guideline, E-0, REACTOR TRIP OR SAFETY INJECTION, should be performed". So the operator would go to E-0, and terminate the SI as spurious, just as above.

If entry to FR-S.1 is based on Status Tree diagnosis during a normal plant shutdown/cooldown, the SI actuation implies entry to E-0 once FR-S.1 is completed.

Q.

Why is the turbine tripped for all ATWS events regardless of whether this action may degrade RCS conditions for particular ATWS events?

A.

Analyses have shown that a turbine trip is required for an ATWS event where a loss of main feedwater has occurred. For other ATWS events, with the exception of when a turbine trip is the initiating event, manual tripping of the turbine may yield a somewhat higher system pressure, depending on the initiating event and time in core life, than what would otherwise be expected. However, since there are many initiating ATWS events and some that require immediate mitigating actions, diagnosis of the initiating event would not be feasible and separate guidance for different ATWS events would complicate training and could delay timely performance of necessary operator actions. Therefore, guideline FR-S.1 contains immediate actions for tripping the reactor and tripping the turbine.

6. REFERENCES

1)

Westinghouse Electric Corporation, Westinghouse Anticipated Transients Without Trip Analysis, WCAP-8330, August 1974.

2)

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3)

Memorandum from W. J. Dircks to NRC Commissioners, Amendments to 10CFR50 Related to Anticipated Transients Without Scram (ATWS) Events, SECY-83-293, July 19, 1983.

4)

T.W.T. Burnett, J. C. McIntyre, J. C. Baker, LOFTRAN Code Description, WCAP-7907 (NES Class 3), October 1972.

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5)

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WOG ERG Revision 3, 1/30/2015