XN-NF-83-72
REVISION 2
SUPPLEMENT 1

H.B. ROBINSON UNIT 2, CYCLE 10 SAFETY ANALYSIS REPORT, REVISION 2 DISPOSITION OF CHAPTER 15 EVENTS

JULY 1984

RICHLAND, WA 99352

EXXON NUCLEAR COMPANY, INC.

XN-NF-83-72
Revision 2
Supplement 1

Issue Date: 7/20/84

H. B. ROBINSON UNIT 2, CYCLE 10
SAFETY ANALYSIS REPORT, REVISION 2

DISPOSITION OF CHAPTER 15 EVENTS

Written by: C. E. Leach 7/20/84
C. E. Leach, Lead Engineer
PWR Safety Analysis

Reviewed by: C. E. Leach (for) 7/20/84
W. V. Kayser, Manager
PWR Safety Analysis

Concur: J. C. Chandler 7/20/84
J. C. Chandler, Lead Engineer
Reload Fuel Licensing

Prepared by: R. B. Stout 20 JULY 84
R. B. Stout, Manager
Licensing & Safety Engineering

Reviewed by: MRK F. B. Skogen 20 JUL 84
F. B. Skogen, Manager
PWR Neutronics

Prepared by: H. E. Williamson 7/20/84
H. E. Williamson, Manager
Neutronics & Fuel Management

Concur: 7/20/84
J. N. Morgan, Manager
Proposals & Customer Services Engineering

Approved by: H. E. Williamson for 7/20/84
G. A. Sofer, Manager
Fuel Engineering & Technical Services

gf

EXXON NUCLEAR COMPANY, INC.

CUSTOMER DISCLAIMER

IMPORTANT NOTICE REGARDING CONTENTS AND USE OF THIS DOCUMENT

PLEASE READ CAREFULLY

Exxon Nuclear Company's warranties and representations concerning the subject matter of this document are those set forth in the Agreement between Exxon Nuclear Company, Inc. and the Customer pursuant to which this document is issued. Accordingly, except as otherwise expressly provided in such Agreement, neither Exxon Nuclear Company, Inc. nor any person acting on its behalf makes any warranty or representation, expressed or implied, with respect to the accuracy, completeness, or usefulness of the information contained in this document, or that the use of any information, apparatus, method or process disclosed in this document will not infringe privately owned rights; or assumes any liabilities with respect to the use of any information, apparatus, method or process disclosed in this document.

The information contained herein is for the sole use of Customer.

In order to avoid impairment of rights of Exxon Nuclear Company, Inc. in patents or inventions which may be included in the information contained in this document, the recipient, by its acceptance of this document agrees not to publish or make public use (in the patent use of the term) of such information until so authorized in writing by Exxon Nuclear Company, Inc. or until after six (6) months following termination or expiration of the aforesaid Agreement and any extension thereof, unless otherwise expressly provided in the Agreement. No rights or licenses in or to any patents are implied by the furnishing of this document.

XN-NF-F00,766

TABLE OF CONTENTS

LIST OF TABLES

1.0 INTRODUCTION

Exxon Nuclear Company (ENC) methodology requires that all events described in the Standard Review Plan (SRP)[1] be dispositioned into one of four categories:

(1) The event needs to be reanalyzed;

(2) The event is bounded by other events;

(3) The event causes and principal variables which control the results of the event are unchanged from prior accepted analysis; or

(4) The event is not in the licensing basis for the plant.

This report provides the disposition of Chapter 15 events for H.B. Robinson Unit 2 Cycle 10, as well as the justification for that disposition. Results of bounding simplified calculations for selected events are also included.

Section 2.0 provides a summary of the event dispositions. In order to facilitate review, the events are numbered in accordance with the SRP. Section 3.0 presents the results of the analysis and justifications. Section 4.0 presents the references used in this report.

This report is presented as a supplement to Reference 2.

The General Design Criteria in existence at the time H.B. Robinson Unit 2 was licensed for operation in July 1970 are contained in the Proposed Appendix A to 10CFR50, General Design Criteria for Nuclear Plants, published in the Federal Register on July 11, 1967. The proposed 1967 criteria, together with the responses, are presented in Sections 3.1.1 and 3.1.2 of the Updated H.B. Robinson Unit 2 FSAR.

The General Design Criteria applicable to H.B. Robinson Unit 2 as it relates to formulating scenarios for anticipated operational occurrences or postulated accidents are:

(1) Concurrent loss of offsite power is assumed for the specific accidents of main steam line break, feedwater line break, and LOCA.

(2) Each Engineering Safety Feature is designed to perform its intended function while accommodating any single failure of an active component.

(3) The Reactor Protection System is designed with redundancy and independence to assure that no single failure will prevent reactor trip.

2.0 SUMMARY OF DISPOSITION OF EVENTS

Table 2.1 presents a summary of results of the analysis. In accordance with ENC methodology, the events are dispositioned into one of the four categories as identified in Section 1.0. References are presented as is appropriate.

Table 2.1 Summary Disposition of Events

SRP Event Number	Name	Disposition	Bounding Event or Reference
15.1.1	Decrease Feedwater Temperature	Bounded	15.1.3
15.1.2	Increase Feedwater Flow		
	1) Power	Bounded	15.1.3
	2) Startup	Bounded	15.4.1
15.1.3	Increase Load	Analyze	3
15.1.4	Secondary Valve Malfunction		
	1) Power	Bounded	15.1.3
	2) Scram Shutdown Margin	Analyze	3
15.1.5	Steamline Break	Bounded/ Analyze	6
15.2.1	Steam Pressure Regulator	Not applicable; BWR Event	6
15.2.2	Loss of Load	Analyze	3
15.2.3	Turbine Trip	Bounded	15.2.2
15.2.4	MSIV Closure	Bounded	15.2.2
15.2.5	Loss Condenser Vacuum	Bounded	15.2.2
15.2.6	Loss Nonemergency A.C. Power Station Auxiliaries	Bounded	15.2.7, 15.3.1
15.2.7	Loss Normal Feedwater	Analyze	3
15.2.8	Feedwater Pipe Break	Bounded	15.1.5
15.3.1	3-Pump Coastdown	Analyze	3
15.3.2	Locked Rotor	Analyze	3
15.3.3	Broken Shaft	Bounded	15.3.2

Table 2.1 Summary Disposition of Events (Cont.)

SRP Event Number	Name	Disposition	Bounding Event or Reference
15.4.1	Uncontrolled Rod Withdrawal Subcritical or Low Power	Analyze	3
15.4.2	Uncontrolled Rod Withdrawal Power	Analyze	3
15.4.3	RCCA Misalignments	Analyze	3
15.4.4	Inactive Loop Startup	N-1 Operation at Power not Permitted	8
15.4.5	Flow Controller Malfunction	Not applicable; No Flow Controller	6
15.4.6	CVCS Malfunctions which Decrease Boron Concentration		
	1) Refuel	Analyze	3
	2) Startup	Analyze	3
	3) Power	Bounded	15.4.1, 15.4.2
15.4.7	Misloaded Fuel Assembly	Administrative Procedures Preclude this Event	9
15.4.8	RCCA Ejection	Analyze	2, 4, 7
15.4.9	Rod Drop	Not applicable; BWR Event	1
15.5.1	Inadvertent Operation ECCS		
	1) Power	Bounded	See Discussion
	2) Other Modes	Bounded	See Discussion
15.5.2	CVCS Malfunction that Increases Reactor Coolant Inventory		
	1) Power	Bounded	15.4.1, 15.4.2
	2) Other Modes	Bounded	See Discussion

Table 2.1 Summary Disposition of Events (Cont.)

SRP Event Number	Name	Disposition	Bounding Event or Reference
15.6.1	Pressurizer Valve Malfunction	Analysis Results Reported	See Discussion
15.6.2	Small Break	Bounded	6
15.6.3	Steam Generator Tube Rupture Radiological Consequences	Analyze (Simplified Model)	3, 4
15.6.4	Radiological Consequences Main Steamline Failure Outside Containment	Not applicable; BWR Event	1
15.6.5	LOCA		
	1) Fuel Damage Limits	Analyze	5
	2) Radiological Consequences	Analyze	4
15.7.1		Deleted	1
15.7.2		Deleted	1
15.7.3	Radioactive Releases Liquid Containing Tanks	Bounded	6
15.7.4	Radiological Consequences Fuel Handling Accident	Analyze	4
15.7.5	Cask Drop	Bounded	6
15.8	Anticipated Transient Without Scram	Not applicable	

3.0 BASIS AND JUSTIFICATION FOR DISPOSITION OF EVENTS

This section presents the basis and justification for disposition of the events. The section numbers and event names are in accordance with those events described in the SRP.

Each event described in the SRP is considered in accordance with the plant licensing basis and dispositioned. Events which are not bounded by other events or by existing accepted analysis, and are in the plant licensing basis are dispositioned to be analyzed. In dispositioning, the cause for each event is identified as an event initiator. The magnitude of the initiator for each event is calculated and compared to the magnitude of the initiator for other events. The comparison basis includes the plant operating modes and state. This allows, in several cases, a ranking of the event initiators as to severity allowing the lesser events to be dispositioned as bounded by the greater event. Similar logic is applied in determination of the applicability and bounding nature for existing accepted analysis.

15.1 INCREASE IN HEAT REMOVAL BY SECONDARY SYSTEM

15.1.1 Feedwater Malfunctions that Result in a Decrease in Feedwater Temperature

The event is caused by malfunction of a feedwater bypass valve which diverts flow around the low pressure feedwater heaters at full power. A conservative calculation of the increase in feedwater flow rate was performed, since it was assumed that all flow in that train bypassed the preheaters. The incremental heat removal capacity was calculated to be $3.8*10^8$ Btu/hr. The added heat removal capacity of the 10% load increase event (15.1.3) is $7.8*10^8$ Btu/hr. The magnitude of the initiator of this event is less than that of the 10% load increase event.

The thermal inertia of the steam generator will slow the reactivity insertion of the feedwater temperature decrease event in comparison to the load increase event; the cooler feedwater must mix with the generator inventory. The load increase event, however, results in a pressure decrease in the steam generator which results in a cooldown of inventory. Therefore, the event is bounded both in rate and magnitude by the 10% load increase event. The acceptance criteria for these events are the same.

15.1.2 Feedwater System Malfunctions that Result in an Increase in Feedwater Flow

This event is caused by malfunction (opening) of the main feedwater valve during startup or low power operation when feedwater is controlled by the bypass valves. A step increase in feedwater to one steam generator to rated flow at a feedwater temperature of 70°F could occur.

Two events are relevant:

(1) During full power operation, one of the steam generator feedwater regulating valves opens to full capacity; and

(2) During startup, the reactor is operated on the feedwater bypass system. While operating in bypass, a steam generator feedwater flow regulating valve opens to full capacity.

To evaluate the increase in cooling capacity of the subevent at power, the feedwater regulating valve is assumed to deliver full flow with the increased coolant delivered at 70°F. The increased heat removal capacity was calculated to be $4*10^8$ Btu/hr. The 10% load increase event results in an increased cooling demand of $7.8*10^8$ Btu/hr. This subevent is therefore bounded by the 10% load increase event (15.1.3).

The calculation of the increased cooling capacity subevent during startup conservatively assumed the feedwater regulating valve permitted full flow, even though only one feedwater train is operated in this mode and some of its capacity is used by the bypass line. Since this is a cooldown event, it is most limiting at end of cycle when the moderator temperature coefficient is most negative. Using the most negative moderator coefficient, the maximum reactivity insertion due to this malfunction was calculated to be $4.4*10^{-4}$ $\Delta\rho$/sec. This is compared to the insertion rate used in the rod withdrawal event at subcritical or low power (15.4.1). The latter is greater than the calculated insertion rate for this event and is therefore bounded by the results of 15.4.1.

This subevent may be conservatively compared to the rod withdrawal event, even though this is a cooldown in comparison to a heatup event.

In 15.4.1 the coolant temperature will increase. The net effect is a reduction in thermal margin. Conversely, in 15.1.2 the coolant temperature decreases, which increases thermal margin relative to a heatup event. Therefore, the events being otherwise equal in terms of reactivity insertion rate, the rod withdrawal event produces a lower MDNBR because of the increase in coolant temperature. Similar to the rod withdrawal event, the reactor would trip on the power range (low setting) flux trip set at approximately 25% of rated power.

It is concluded that this event is bounded by the results of 15.1.3 and 15.4.1.

15.1.3 Excessive Increase in Secondary Steam Flow

This event is being reanalyzed for a 10% load increase.

15.1.4 Inadvertent Opening of a Steam Generator Relief or Power Operated Relief Valve

The event is caused by malfunction of a single power operated (PORV) or safety relief valve (SRV). There are two subevents: (1) at power; and (2) no return to criticality after reactor trip. The safety relief valves are independent mechanical type valves allowing no common mode failure. Similarly, the PORVs have no common mode malfunction. Therefore, the consequences of this event are limited to malfunction of a single valve. The maximum capacity of one PORV is 580,000 lb/hr and one SRV is 1,000,000 lb/hr. The total steam demand at rated load for H.B. Robinson is 10,000,000 lb/hr. Therefore, the 10% load increase of the 15.1.3 analysis will conservatively treat the consequences of the secondary valve malfunction event at power. This event is therefore bounded by the results of 15.1.3.

The event acceptance criteria for this event in the H.B. Robinson 2 FSAR(6) is for a postulated accident. Since the event is shown to be bounded by an anticipated operational occurrence (AOO), protection of the specified acceptable fuel design limits (SAFDLs) is assured.

The second subevent is postulated to occur after reactor trip. The acceptance criteria is that there be adequate shutdown margin to preclude return to criticality after trip due to spurious release of steam. The analysis results are reported in Reference 3.

15.1.5 Main Steam Line Break

The event is most limiting at end of cycle and at hot zero power conditions. The event was previously analyzed by ENC(10) and reviewed and accepted by the NRC. The controlling parameters of this event have been reviewed and remain bounded by that analysis. The analysis demonstrated that the acceptance criteria was met.

ENC has reviewed its steam line break model and is developing a new model based on use of RELAP5. The new model is planned for submittal in 1984, with plant specific applica n planned for 1985. This work will provide additional confirmation of the Reference 10 analysis.

15.2 DECREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM

15.2.1 Steam Pressure Regulator Malfunction that Results in Decreasing Steam Flow

The H.B. Robinson Unit 2 plant has no main steam line pressure regulators. This event is not applicable to H.B. Robinson Unit 2.

15.2.2 Loss of External Electric Load

This event is analyzed(3) for two cases:

(1) Primary pressure limiting case; and

(2) SAFDL limiting case.

The following set of assumptions relevant to other events in this category are imposed on the analysis of this event:

- Direct trip on turbine trip is disabled.
- Turbine stop valve rather than steam flow regulating valve is closed.
- The trip delay (overtemperature ΔT, DNB case; high pressure, overpressurization) is equal to or greater than the turbine trip delay time.
- Steam bypass is disabled.

15.2.3 Turbine Trip

The event is caused by a turbine trip signal. Turbine trip causes a direct reactor trip which results in earlier trip than analyzed in Section 15.2.2. Since the turbine stop valve closure time is used in the loss of load analysis, the consequences of this event are bounded by the conditions assumed for analysis of the loss of load event.

15.2.4 Loss of Condenser Vacuum and Other Events Resulting in Turbine Trip

This event may be caused by loss of the condenser cooling pumps. Loss of condenser vacuum disables steam bypass. Since steam bypass and direct trip on turbine trip is defeated in the loss of load event, the results of this event will be less severe than loss of load and are therefore bounded by the loss of load event.

15.2.5 Inadvertent Closure of Main Steam Isolation Valves (MSIVs)

The event is assumed to be caused by malfunction of the valve controllers. The MSIVs close more slowly than the turbine stop valves, as is imposed in the loss of load event. Therefore, the consequences of this event will be bounded by the results of the loss of load analysis.

15.2.6 Loss of Non-Emergency AC Power to the Station Auxiliaries

The event may be caused by a break in connection to the main grid. The H.B. Robinson plant is designed to accept a substantial load loss without trip, maintaining all auxiliaries as turbine load. The plant is designed to bring the station to safe shutdown even with turbine trip and consequent loss of auxiliaries.

The most likely result of break in connection to the grid is a sudden loss in load but maintaining station auxiliaries such as primary coolant pumps and main feedwater pumps. This branch of the event is similar to the loss of load event with respect to safety except the steam bypass and dump are expected to operate. The station sheds extra load in an orderly fashion and DNB margin increases. The challenging aspects are bounded by the loss of load results.

The second branch of this event is the possibility that the turbine trips with consequent loss of primary coolant and main feedwater pumps. The early stages of this subevent will be bounded by the results of the loss of forced reactor coolant flow event because direct trip will occur prior to low flow trip (for 3-pump coastdown), and MDNBR occurs prior to significant reduction in steam generator inventory. In the longer term, the results are bounded by the loss of normal feedwater event with concurrent loss of primary coolant pumps as the reactor trips directly on turbine trip and steam generator level at or above the low low trip. Therefore, this event is bounded by the loss of normal feedwater and 3-pump coastdown events.

15.2.7 Loss of Normal Feedwater Flow

This event may be caused by malfunction of main feedwater valves or trip of the main feedwater pumps. Loss of forced primary coolant flow may also occur and the event is analyzed both with and without forced primary coolant flow.[3]

15.2.8 Feedwater System Pipe Break

This event is postulated to be caused by the instantaneous severence of a feedwater line.

The H.B. Robinson steam generators are fed by a single 16 inch line. Auxiliary feedwater enters the same nozzle. The sparger is approximately 3 feet above the top of the tube bends and approximately 7 feet below the top of the downcomer. The sparger is approximately 2 feet above the low range liquid level tap. The feedwater sparger is of the J tube type. Upon rupture, some liquid may initially blow down; however, substantial liquid will remain. In many PWRs, feedwater is introduced at the bottom of the steam generator and a feedwater pipe break potentially results in a major or total loss of steam generator liquid inventory and subsequent primary system heatup. In the case of H.B. Robinson, however, this event will be a cooldown event and will be bounded by the steam line break results as the feedwater pipe is much smaller in area than the minimum area for flow in the new steam generator integral flow restrictors.

15.3 DECREASE IN REACTOR COOLANT FLOW

15.3.1 Loss of Forced Primary Coolant Recirculating Flow

This event is caused by trip of the three primary coolant recirculating pumps. The purpose of the analysis is to verify protection of the SAFDLS by the low flow trip setpoint. Analysis of this event is reported in Reference 3.

15.3.2 Flow Controller Malfunction

The H.B. Robinson Unit 2 plant has no primary coolant flow controllers. Therefore, this event is not applicable to H.B. Robinson Unit 2.

15.3.3 Reactor Coolant Pump Shaft Seizure (Locked Rotor)

This event is postulated to be caused by the instantaneous seizure of one primary coolant pump. The reactor trips on low flow in the affected loop. Flow reverses direction in this loop but is limited by the substantial reverse flow pump loss coefficient. The MDNBR of the event occurs shortly after trip. Analysis of this event is reported in Reference 3.

15.3.4 Reactor Coolant Pump Broken Shaft

This event is postulated to be caused by the instant severence of the pump impeller shaft. The reactor trips on low flow slightly later than for the pump seizure event because of the higher flow associated with a free-wheeling impeller in comparison to a locked impeller. The reverse flow associated with a free-wheeling impeller in reverse direction is larger, but MDNBR occurs prior to significant flow reversal due to momentum effects. The results of this event are bounded by those of the pump seizure event.

15.4 REACTIVITY AND POWER DISTRIBUTION ANOMALIES

15.4.1 Uncontrolled RCCA Withdrawal From a Subcritical or Low Power Startup Condition

The cause of this event could be a malfunction of the reactor control system or malfunction of the control rod drive system. This can occur with the reactor subcritical or at power. The high power rod withdrawal events are treated in 15.4.2. The analysis is performed at a conservatively high reactivity insertion rate. Two-pump operation is analyzed since this event occurs during the startup mode. The results are reported in Reference 3.

15.4.2 Uncontrolled RCCA Withdrawal From Power

The event is assumed to be caused by direct reactivity insertions due to RCCA withdrawal. A spectrum of insertion rates are analyzed at selected powers in the range of power operation from the low neutron flux power trip reset point to full power. The insertion rates cover the range that are characteristic of chemical shim dilution to bounding reactivity insertion rates. The purpose of the analysis is verification of proper setting of the high power neutron flux and overtemperature-delta T trips with bounding power distribution to prevent penetration of SAFDLs.

The results are reported in Reference 3.

15.4.3 Control Rod Misoperation

There are four subevents in this event:

(1) Dropped full length RCCA with and without turbine run-back;

(2) Dropped RCCA bank or group;

(3) Statically misaligned assembly; and

(4) Single rod withdrawal.

The results of the analyses are reported in Reference 3.

15.4.4 Startup of an Inactive Reactor Coolant Loop at an Incorrect Temperature

The H. B. Robinson plant technical specifications(8) do not permit operation with less than three primary coolant pumps during power operation. Therefore, analysis of this event is unnecessary.

15.4.5 Recirculation Loop at Incorrect Temperature or Flow Controller Malfunction

The H.B. Robinson Unit 2 plant has no primary loop isolation valves nor means to control primary flow. Therefore, this event is not applicable to H.B. Robinson Unit 2.

15.4.6 Chemical and Volume Control System Malfunctions That Result in a Decrease in the Boron Concentration in the Reactor Coolant

The event is caused by malfunction of the primary makeup water system. The maximum rate of dilution is due to the inadvertent operation of all three charging pumps with makeup water pumps in operation taking suction from the unborated makeup water tanks.

There are three operational modes of concern in analysis of the consequences of this malfunction:

(1) Dilution during refueling operation.

(2) Unplanned dilution during startup.

(3) Unplanned dilution at power.

The operational modes of refueling and startup are analyzed. The dilution rate at power is bounded by the rod withdrawal event at power (15.4.2).

The results of the analysis are reported in Reference 3.

15.4.7 Inadvertent Loading of a Fuel Assembly into the Improper Location

This event is a result of placement of a fuel assembly in a position other than designated. Differences between fuel can be due to burnable poison, burnup or part length shielding assemblies. During core reload, fuel assembly serial numbers are double checked against the core load map to preclude this event.[9]

The H.B. Robinson core has 48 in-core thimble tubes to accommodate miniature in-core neutron flux probes. There are ten in-core calibration locations. There are also 51 thermocouples to measure coolant temperature as it leaves the fuel. H.B. Robinson operating procedures[9] require monitoring of the core during startup at 30%, 70%, 90% and 100% of rated power, and thereafter at intervals of not greater than one effective full power month. These measurement results are then reviewed and compared to the Technical Specification limits. Administrative procedures therefore preclude occurrence of this event.

15.4.8 Spectrum of Rod Cluster Control Assembly (RCCA) Ejection Accidents

The accident is postulated to be caused by a mechanical failure of a drive housing. Because of the high pressure of the primary coolant system, the RCCA will be driven out rapidly. The results of the analysis are reported in References 2 and 4. The methodology is in accordance with Reference 7.

15.4.9 Spectrum of Rod Drop Accidents

This event is not applicable to pressurized water reactors.

15.5 INCREASES IN REACTOR COOLANT SYSTEM INVENTORY

Increase in reactor coolant system inventory can be caused by inadvertent operation of the ECCS or primary coolant system charging pumps.

15.5.1 Inadvertent Operation of Emergency Core Cooling System

The shutoff head of the H.B. Robinson high pressure safety injection system pumps is approximately 1500 psia, which is much less than the trip setpoint pressure of 1850 psia and therefore cannot increase the primary inventory during power operation.

Pressurized thermal shock (PTS) is being addressed in the Unreviewed Safety Issue program A-49. Typical Combustion Engineering, Babcock & Wilcox and Westinghouse early design operating plants were modeled in this effort. The plants modeled were Calvert Cliffs, Oconee, and H.B. Robinson Unit 2. Approximately 200 cases have been analyzed in the thermal hydraulics portion of the H.B. Robinson program. Representative events examined were steam line break, loss of coolant accidents, and arbitrarily large step changes in coolant temperature.

Break spectrums were examined with the specific objective of achieving stagnation conditions in the primary system. In each event when primary pressure dropped below 1300 psia, the reactor coolant pumps were shut off. As required by the reactor protection logic, the safety systems were enabled injecting cold ECC water. All events were initiated at hot zero power or at power conditions in order to bound lower temperature operations. Thus, the effect of inadvertent operation of the ECCS in stagnant conditions in addition to a much broader spectrum of more limiting events has been addressed.

Probabilistic fracture mechanics analysis using these thermal hydraulic results is in progress. While not yet completed, extremely low probability of reactor vessel failure is indicated from preliminary results.

To further address the concerns of this issue, Carolina Power and Light is implementing a low radial leakage fuel management program and is installing part length shielding fuel assemblies. These actions assure that H.B. Robinson 2 will not reach the NRC screening criteria for RT_{NDT}.

The Westinghouse Owners Group (WOG) has previously addressed this issue.(11) This effort addressed all transients which may subject the reactor pressure vessel to overcooling thermal effects from loss of loop flow. The results of the report support the NRC screening criteria, i.e., plant operation is acceptable if the screening criteria for RT_{NDT} is not reached.

Therefore, the causes and consequences of this event and all other events which could lead to PTS have been addressed by the NRC and WOG programs and need not be further addressed in this license action.

15.5.2 CVCS Malfunction that Increases Reactor Coolant Inventory

The consequences of unplanned additions to inventory and effect of reactivity additions due to dilution during refueling and startup are treated in Section 15.4.6. The consequences of dilutions at power are bounded by the analysis of Section 15.4.2, Uncontrolled RCCA Bank Withdrawal at Power.

The consequences of volumetric addition and effect on pressure boundary during all operational modes have been earlier addressed in the H.B. Robinson 2 FSAR, Updated, Section 15.5.

These effects are mitigated by resetting the pressurizer PORV set pressure to 400 psig prior to going below 350°F. There are two PORVs on the pressurizer, each independently actuated. Any one valve has adequate relief capacity and response time to prevent overpressurization due to malfunction of the CVCS. Therefore, this branch of the event is bounded by the previous analysis.

15.6 DECREASES IN REACTOR COOLANT SYSTEM INVENTORY

15.6.1 Inadvertent Opening of a Pressurizer Safety or Power Operated Relief Valve

This event is caused by malfunction of either the pressurizer PORV or safety relief valve. The safety relief valve capacity is greater than that of the PORV, and analysis with malfunction of the larger will bound the two possible events.

The H.B. Robinson 2 licensing basis acceptance criteria for this event is as for postulated accidents. However, the analysis performed shows that the SAFDLs are not penetrated.

A calculation was performed to evaluate the consequences of this event. It was assumed that a single code valve failed open at full power. The maximum relief capacity of the valve is 288,000 lb/hr at set pressure. The conditions are:

Power	2346 MWt (102% of rated)
T_{ave}	575.4 + 4°F
Pressure	2250-30 psia
Active Core Flow	$95.8*10^6$ lb/hr (-3% uncertainty included)
$F^N_{\Delta H}$	1.65
F_e	1.03
F^T_Q	2.32
Low Pressure Trip	1850-30 psia
Trip Delay	1.1 sec
Scram Delay	.9 sec

A quasi-steady state calculation was performed to determine the DNBR for the event.

The event was assumed to be initiated at the above conditions. Reactor power was assumed to remain at 102% of full power to the trip overshoot pressure less 20 psi. The time from event initiation to trip was conservatively calculated to be 9.2 seconds based on rated valve flow at 2500 psia. A trip power-reduction delay of two seconds was used. The pressure at scram was calculated to be 1733 psia with the 20 psi conservative subtractor. The 20 psi conservatism would also allow ~.5 sec larger delay to power reduction. The reduction in coolant temperature due to the primary coolant performing thermodynamic work on the pressurizer was calculated to be 2.4°F. Conservatively, credit was not taken in the calculation of DNBR for the reduction in coolant temperature. The MDNBR so calculated was greater than the XNB DNB correlation SAFDL limit of 1.17, which assures with 95% probability and confidence limits that DNB does not occur.

15.6.2 Loss of Reactor Coolant from Rupture of Small Pipes or from Cracks in Large Pipes which Actuate the Emergency Core Cooling System

An analysis was performed in 1975 with ENC's approved small break model. Substantial margin to acceptance criteria was demonstrated in that analysis.(6) The analysis was performed for breaks ranging in size from a 6 inch diameter break to a 1.0 double ended guillotine cold leg break. That analysis demonstrated that the large break bounded the small breaks with substantial margin. There have been no changes which would change the relative aspects of small and large breaks. Therefore, the event as analyzed in 15.6.5 (large break LOCA) bound the results of this event.

15.6.3 Steam Generator Tube Rupture

This event is analyzed with a conservative break flow calculation, including a stuck open atmospheric dump valve. The results are reported in Reference 3. The radiological consequences are treated in Reference 4.

15.6.4 Radiological Consequences of Main Steam Line Failure Outside Containment

BWR event. Not applicable to PWRs.

15.6.5 Loss of Coolant Accidents Resulting from Spectrum of Postulated Piping Breaks within the Reactor Coolant System Pressure Boundary

A spectrum of double-ended guillotine pipe breaks with varying Moody discharge coefficients and split breaks with varying flow areas has been evaluated with an approved and accepted model for H.B. Robinson 2. This work identified the limiting break size to a double-ended cold leg guillotine break with Moody discharge coefficient of 0.8. This event is reanalyzed for the previously identified limiting break with EXEM/PWR LOCA/ECCS models. The results of the analysis are presented in Reference 5.

15.6.6 Radiological Consequences of a Design Basis Loss-of-Coolant Accident

The analysis is based on the maximum core damage and release which could occur as a consequence of the design basis LOCA. As such, the fission product inventory within the gap region for all fuel rods in the reactor is released. The maximum average core burnup is assumed. The results of the analysis are reported in Reference 4.

15.7 CONSEQUENCES OF RADIOACTIVE RELEASE FROM A SUBSYSTEM OR COMPONENT

15.7.3 Postulated Radioactivity Releases due to Liquid Tank Failure

This event has been reviewed and results documented in the H.B. Robinson Unit 2 FSAR. The results of this evaluation are unchanged by the planned licensing actions and therefore bound present operations.

15.7.4 Radiological Consequences of Fuel Handling Accidents

The accident is assumed to be caused by damage and release of fission products from a single fuel assembly during handling in the spent fuel pool. The maximum average burnup of the H.B. Robinson fuel is being increased. The results of the analysis for increased burnup are presented in Reference 4.

15.7.5 Spent Fuel Cask Drop Accidents

The results of this accident are unchanged from those documented in the H.B. Robinson Unit 2 FSAR. The results of the analysis therefore bound present operations.

15.8 ANTICIPATED TRANSIENTS WITHOUT SCRAM

Not applicable.

4.0 REFERENCES

(1) "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," NUREG-0800, U.S. Nuclear Regulatory Commission, July 1981.

(2) "H.B. Robinson Unit 2, Cycle 10 Safety Analysis Report," XN-NF-83-72, Rev. 2, July 1984.

(3) "Plant Transient Analysis for H.B. Robinson Unit 2 at 2300 MWt with Increased $F^N_{\Delta H}$." (To be issued)

(4) "H.B. Robinson Unit 2 Radiological Assessment of Postulated Accidents," XN-NF-84-68(P), June 1984.

(5) "H.B. Robinson Unit 2 Limiting Break LOCA-ECCS Analysis with Increased Enthalpy Rise Factor," XN-NF-84-72, July 1984.

(6) Final Safety Analysis Report (Updated), H.B. Robinson Steam Electric Plant Unit No. 2.

(7) "A Generic Analysis of the Control Rod Ejection Transient for Pressurized Water Reactors," XN-NF-78-44(A), October 1983.

(8) H.B. Robinson Unit 2 Technical Specifications.

(9) Plant Operating Manual for H.B. Robinson Unit 2, Procedure FHP-031, Core Mapping Following Fuel Loading; EST-050, Refueling Startup Procedures; EST-054, Power Distribution Maps.

(10) "Plant Transient Analysis of the H.B. Robinson Unit 2 PWR for 2300 MWt," XN-75-14, July 1975.

(11) "A Generic Assessment of Significant Flow Extension, Including Stagnant Loop Conditions, from Pressurized Thermal Shock of Reactor Vessels on Westinghouse Nuclear Plants," WCAP-10319, December 1983.

XN-NF-83-72
Revision 2
Supplement 1

Issue Date: 7/20/84

H. B. ROBINSON UNIT 2, CYCLE 10

SAFETY ANALYSIS REPORT, REVISION 2

DISPOSITION OF CHAPTER 15 EVENTS

Distribution

FT Adams
GJ Busselman
JC Chandler
LJ Federico
NF Fausz
TJ Helbling
JS Holm
WV Kayser
MR Killgore
CE Leach
JN Morgan
FB Skogen
GA Sofer
IZ Stone
RB Stout
T Tahvili
HE Williamson

Document Control (5)

CP&L/TJ Helbling (82)