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Ltr re their 9-15-76 submittal....trans the following:

PLANT NAME: Monticello

ENCLOSURE

Errata Sheets for NEDO-25016 entitled "Evaluation of ATWS for the Monticello Nuclear Generating Plant"........consisting of editorial corrections ....

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NRC FORM 195 (2-76)

NSP

NORTHERN STATES POWER COMPANY

MINNEAPOLIS, MINNESOTA 55401

November 16, 1976

Mr Dennis L Ziemann, Chief
Operating Reactors Branch #2
Division of Operating Reactors
U S Nuclear Regulatory Commission
Washington, DC 20555

Dear Mr Ziemann:

MONTICELLO NUCLEAR GENERATING PLANT
Docket No. 50-263 License No. DPR-22

Errata Sheets for NEDO-25016

On September 15, 1976, we submitted to you a report entitled, "Evaluation of Anticipated Transients Without Scram for the Monticello Nuclear Generating Plant, NEDO-25016". Several editorial corrections were found appropriate after filing the report; the technical conclusions remain unaffected. Please incorporate the attached pages into your copies in accordance with the printed instructions.

Yours very truly,

L. O. Mayer

L O Mayer, PE
Manager of Nuclear Support Services

LOM/MHV/deb

cc: Director, IE, III, USNRC
G Charnoff
MPCA
Attn: J W Ferman

11858

Attachment

APPLICABLE TO
PUBLICATION NO. NEDO-25016
TITLE Evaluation of Anticipated Transients Without Scram for the Monticello Nuclear Generating Plant
ISSUE DATE September 1976

ERRATA And ADDENDA SHEET

NO.	1
DATE	October 1976
NOTE	*Correct all copies of the applicable publication as specified below.*

ITEM	REFERENCES (SECTION, PAGE, PARAGRAPH, LINE)	INSTRUCTIONS (CORRECTIONS AND ADDITIONS)
		Affix the attached new pages over the corresponding pages of your document: 3-1 3-5 3-9 3-11 3-14 3-18 3-21 3-22 3-26 4-1 4-2 4-3 4-5 4-6 4-17 4-18 5-1 A-2 A-5 A-7 A-8 A-11 A-19

3. EVALUATION OF EVENTS

The following discussions summarize performance of the plant during many transients events, including consideration in each case for the postulated failure of normal scram, and the effect of recirculation pump trip, ATWS rod injection (ARI), and operator actions. The previous reports have concentrated upon the most frequent (once in four years) cases with special attention to the closure of all main steam isolation valves as a bounding case. That case still remains the most severe event from most viewpoints, and still receives the most parametric attention in this study. However, the scope of the events has been extended as requested by the NRC to cover all significant events expected at least once within forty years of the plant operation.

3.1 CLOSURE OF ALL MAIN STEAM ISOLATION VALVES

3.1.1 Basic Event Description

Automatic circuitry or operator action can initiate closure of the main steam isolation valves (MSIV). Normally, scram is initiated by position switches on the valves before they have traveled more than 10% from the open position. Subsequent scram signals would be initiated (if needed) from high neutron flux and high vessel pressure. The normal event displays very little, if any, neutron flux increase before shutdown is effective. An abrupt vessel pressure rise occurs when the MSIV's close, lifting the Safety/Relief (S/R) valves for several seconds in their pilot actuated relief mode. Pressure is easily limited below the design pressure of the primary system. Long term heat removal and inventory supply are provided by one of the S/R valves, the HPCI or the RCIC system, and the RHR cooling capability (as long as necessary until the normal heat sink, the main condenser, can be utilized).

3.1.2 Response of Plant in its Present Configuration

Figure 3-1 shows the reactor vessel pressure and neutron flux as a function of time for the MSIV closure when credit for scram is not taken and when the postulated modifications are not assumed. As the MSIV's close the reactor

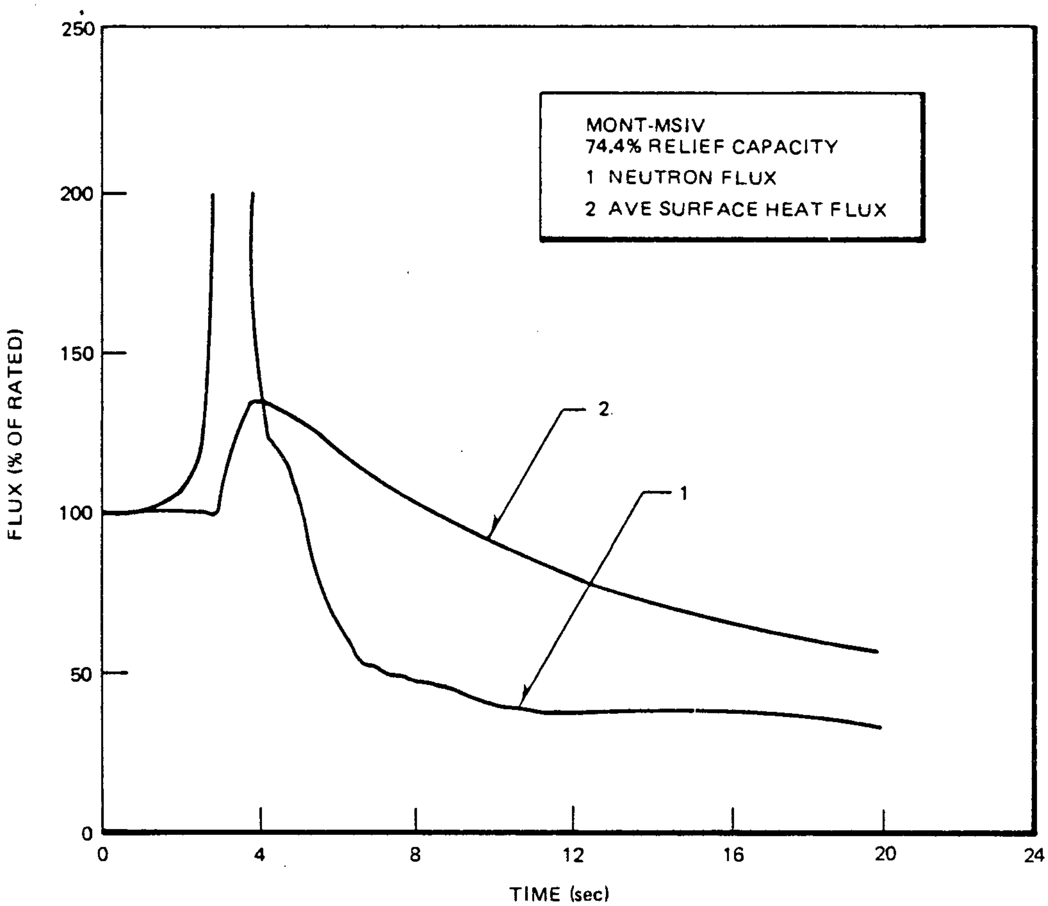

Figure 3-3. MSIV Closure Transient With 2-RPT Modification

effects neutronic shutdown, the reactor power is only due to decay heat. Reactor core cooling is adequately provided by the HPCI and RCIC systems, the heat being removed by relief valve flow into the suppression pool. Cold shutdown is achieved by normal operator actions. The peak bulk suppression pool temperature reached is 153.2°F with 6 valves (153.4°F with 8 valves) and is below the guideline value of Section 1.1. The containment pressure peaks at 7.6 psig (with 6 valves) and 7.4 psig (with 8 valves). Both of these peak values are well below the guideline value.

3.2 LOSS OF NORMAL AC POWER

3.2.1 Basic Event Description

Loss of normal AC auxiliary power de-energizes all busses that supply power to the unit's auxiliary equipment such as the recirculation pumps, condensate pumps and circulating water pumps. Coastdown of all pumps occurs and condenser vacuum is gradually lost. Turbine-generator trip occurs at the start if this is a general grid disturbance. Scram is normally activated by any of a number of signals (T-G trip, low vacuum, low water level) and MSIV closure occurs (low water level, low vacuum). Momentary opening of the S/R valves occurs to easily limit the pressure rise in the vessel. Long term heat removal and inventory supplies are provided by one of the S/R valves, the HPCI or RCIC system and RHR cooling capabilities.

3.2.2 Response of Plant in its Present Configuration

When auxiliary power is lost, all pumps (circulating water pumps, feedpumps and recirculation pumps) coast down immediately. The reduction in core flow begins to reduce the reactor power. The MSIV's close due to loss of condenser vacuum and contribute to the pressurization of the reactor. The short term response of the reactor is shown in Figure 3-7. Short term response is much less severe than the MSIV ATWS with RPT modification for the following reasons: (1) The recirulation pumps trip at time zero rather than wait for the reactor pressure to reach the ATWS setpoint. (2) The feedwater pumps are tripped at time zero which result in a lower core flow and lower core inlet subcooling, and thereby lower power without normal scram. The peak reactor pressure reached is 1194 psig (with 6 relief valves) which remains under the guideline number. The peak fuel enthalpy reached is < 150 cal/gm, also below its guide.

In the long term, with the feed pumps lost, the reactor water level begins to drop. At the low low level, the HPCI and RCIC systems are initiated. However, as the reactor core power is still higher than the combined flow capacity of HPCI and RCIC, the reactor water level continues to drop. The combination of reactor power, core flow and water level eventually reached before the standby liquid control system can be assumed to be effective is such that the analytical model is not capable of continuing to predict the results of the event. Therefore, the results do not show reactor behavior during the time before manual neutronic shutdown becomes effective and normal reactor water inventory is restored. If initiation of SLC takes place at 5 minutes after the reactor high pressure is reached, hot shut down would be achieved at about 20 minutes (with the normal configuration of only one SLC pump working).

In order to calculate the containment response, the initial relief valve flow predicted by the model is extrapolated through hot shut down. Assuming this steam dump into the containment, the containment pressure and suppression pool bulk temperature peaks were calculated. The peaks reached are 21.8 psig and 214°F respectively.

The results of this transient, if the unlikely ATWS event is assumed to occur, are uncertain because of analytical limitations and, since the suppression pool temperature is above the guideline, the results are considered to be unsatisfactory without the postulated modifications.

3.2.3 Plant Response with the Postulated Modifications

The short term response of this case is the same as the As-Built Case. The reactor pressure peak reached is 1194 psig (with 6 relief valves). The peak fuel enthalpy reached is < 150 cal/gm. When the dome pressure reaches 1150 psig and the ATWS logic initiates the ATWS Rod Injection (ARI).

The reactor will be in hot shutdown near 20 seconds by ARI, and combined flow of the RCIC and HPCI systems easily restore water level to the normal range. The cold shutdown condition can be reached by performing the normal manual actions. Containment response with ARI function is shown in Figures 3-8 and 3-9. The containment pressure peak reached is 3.6 psig at 12 hours and the suppression pool bulk temperature peak reached is 153.4°F at 11 hours. Both of these are below their respective guideline values.

3.3 LOSS OF NORMAL FEEDWATER FLOW

3.3.1 Basic Event Description

Inadvertant trip of all the feedwater pumps or water level controller failure (zero demand) have been postulated as potential causes of loss of all normal feedwater flow to the vessel. Loss of auxiliary power also causes this event as described above. Reactor core flow is reduced when the feedwater flow reduction occurs, dropping power gradually until it is totally shut down in the normal case when scram is initiated from low water level. Continued gradual inventory loss occurs until isolation is initiated and the RCIC/HPCI systems are brought on automatically to maintain proper water level to the conclusion of the event.

3.3.2 Response of Plant in its Present Configuration

When the feedwater pumps are lost, the reactor water level drops to the low low level in a few seconds, causing the MSIV's to close, the recirculation pumps to trip and the HPCI, RCIC and RHR systems to initiate. From this point the short term response of the plant will be similar to that of the MSIV closure but milder. The peak vessel pressure and fuel enthalpy reached are less than the respective guideline values of Section 1.1. The long term response of the reactor is not analyzed as it would be very similar to the case of loss of normal AC power transient.

3.3.3 Plant Response With the Postulated Modifications

When the reactor water level reaches the low low level, the ATWS logic initiates ARI. Hot shutdown is achieved near 20 seconds. The minimum water level reached in this case would be about 1.5 to 2.0 ft below the low low level. The peak bulk temperature reached in the pool would be slightly less than the value for the MSIV closure (Viz. 153.4°F). This is because of the smaller power burst due to feedwater trip at the beginning and the recirculation pump trip being simultaneous with the MSIV closure. Cold shutdown condition can be achieved in the same manner as MSIV closure transient; by normal manual actions.

3.4 TURBINE-GENERATOR TRIP

3.4.1 Basic Event Description

Loss of generator electrical load initiates fast closure of the turbine control valves to provide overspeed protection for the unit. A variety of equipment protection signals can lead to trip of the turbine stop valves directly. Both types of events are very similar from the reactor point of view. Normally, scram is initiated almost simultaneously with the start of fast valve closure. Inherent control logic opens the steam bypass valves directing some steam

3.5.3 Plant Response With Postulated Modifications

The closure of stop valves causes the vessel pressure to increase (Figure 3-11) resulting in an ATWS signal. When the dome pressure reaches 1150 psig, the recirculation pumps are tripped and the ARI is initiated. The tripping of the recirculation pumps causes an immediate reduction in power level. The peak vessel pressure reached is 1248 psig. The rod injection continues to reduce power to hot shut down. Further normal manual actions will bring the reactor to cold shutdown conditions. Heat is removed via the relief valves to the suppression pool. The RHR's are manually initiated at 10 minutes to remove excess heat from the pool. The pool reaches a peak bulk temperature of 153.3°F and the containment reaches a peak pressure of 7.3 psig in about 11 hours, which are within the guides of Section 1.1.

3.6 LOSS OF A FEEDWATER HEATER

3.6.1 Basic Event Description

Historically, this event has been addressed in most SAR analyses even though on many plants no direct means of eliminating the action of a feedwater heater is provided. It is included here at NRC request, covering whatever chance of extraction steam loss or other heater interruption that could occur. Individual heaters (or linked groups) have traditionally been bounded by analyses with 100°F changes in feedwater temperature. Conventional designs for most plants including Monticello really limit that change to about 70°F or less. The expected plant behavior, should this event happen at full power conditions, is therefore a gradual power increase toward a new value consistent with the colder core inlet conditions. The change is gradual because of the thermal capacity of the heater and the mixing characteristics of the reactor downcomer and lower plenum. If the plant happened to be in automatic load control, the core flow would be reduced in such a way that steam flow to the turbine would be held essentially constant (although neutron flux would rise slightly above the initial value). In base-loaded, manual operation of the plant, power increases somewhat more (without compensation by the flow control) and the bypass valves would be opened slightly, if needed to pass the excess steam.

10 psi and the relief valve steadily blows 164 lbs/sec of steam into the pool. The guides for suppression pool temperature and containment pressure shown in Section 1.1 would eventually be exceeded if scram could not be initiated and there were no modifications or manual corrective actions taken.

3.7.3 Plant Response With Postulated Modifications

Should the normal scram not take place, the manual initiation of recirculation pump trip and ARI in 5 minutes after receiving the high torus temperature alarms would reduce the power and shut down the reactor. The high temperature alarms are initiated when the pool temperature reaches 110°F. The bulk pool temperature reached in this case is 120°F at hot shutdown, well below the guideline value. The long term pool temperature response is discussed in the appendix in answer to Questions B.4, 5 and 6.

3.8 LOSS OF CONDENSER VACUUM

3.8.1 Basic Event Description

The reduction or loss of vacuum in the main turbine condenser can be caused by loss of cooling water pumps or ineffectual operation of the vacuum support equipment. It sequentially trips the turbine stop valves closed (which normally scrams the reactor) and, if the event is severe enough and the reduction of flow from the turbine still is not enough to help condenser performance, the steam bypass valves are closed. These actions would occur normally over a period of several minutes or at worst, 20-30 seconds. The initial part of the event is the same as a turbine-generator trip since all systems function in the same way as they do in that event. The long term behavior of the event is similar to any isolation unless enough vacuum can be maintained to preserve bypass flow thereby permitting decay heat removal through the condenser instead of relying upon the pool and the shutdown cooling systems.

3.8.2 Response of Plant in its Present Configuration

This case begins identically to the T-G trip and then the bypass valves would close. The response without the postulated modifications is less severe than MSIV closure. The peak vessel pressure would exceed the guide of Section 1.1.

3.8.3 Plant Response With the Postulated Modifications

The event results in short term peak values less severe than the MSIV case. All the ATWS logic is activated quickly by the high pressure transient. The longer term nature of this case (assuming vacuum continues to deteriorate) is converted to a nearly normal isolation by action of the ATWS rod injection which completes the nuclear shutdown less than 20 seconds into the event.

3.9 FEEDWATER FAILURE - MAXIMUM DEMAND

3.9.1 Basic Event Description

A postulated failure of the feedwater/water level controls in the direction of maximum demand results in a moderator temperature and void fraction decrease causing a reactor power increase at the same time water level increases toward high level protection. The feedwater pumps are tripped as well as the main turbine when level reaches the high trip setpoint. Scram normally occurs with the turbine stop valve closure, limiting any further power increase in such a way that satisfactory thermal margins are maintained. The resulting pressure rise is controlled by the turbine bypass (throughout) and S/R valves (momentarily). Final aspects of the event are similar to the loss of normal feedwater since RCIC/HPCI system initiation eventually are expected to occur.

3.9.2 Response of Plant in its Present Configuration

When the failure of the feedwater in the direction of maximum demand occurs, the high level turbine trip (with bypass) and feedwater pump trip will occur near 143 seconds. The feedwater pump trip results in the water level dropping to the low low level causing the MSIV closure and recirculation pump trip. So, without the postulated modifications the short term response is similar to the MSIV closure but milder. The long term response is similar to the loss of AC power event.

3.10.2 Plant Response With the Postulated Modifications - Not Applicable

3.11 SUMMARY OF TRANSIENT ANALYSES

ATWS impact of all the key transients expected in the life time of the plant were considered in this section. The results show the impact of modification, in terms of reactor pressure, fuel transient and containment conditions by comparison to the guides of Section 1.1. The specific postualted modifications considered are recirculation pump trip and ARI. The results of the analyses can be summarized as follows:

1. For most pressurization transients the reactor pressure exceeds the comparison guide for the plant in its present configuration.

2. For all transients, the recirculation pump trip modification effectively mitigates the short term ATWS response. The reactor peak pressure satisfies the 1500 psig guide value and the peak enthalpy of the hottest fuel satisfies the guide value of 280 cals/gm.

3. For all transients, ARI successfully mitigates the long term response. The containment pressure and suppression pool temperature satisfy their respective comparison guide values.

4. In the loss of feedwater heater and rod withdrawal error transients all the key variables are within the comparison guides even without the benefit of the postulated modifications.

4. SYSTEM DESCRIPTION WITH POSTULATED MODIFICATIONS

All of the anticipated transients, which would require mitigation with the plant in its present configuration in the unlikely event of an ATWS, quickly reach at least one of two conditions which are readily sensed and from which the actions of the postulated modifications may be initiated. These conditions are: 1. High vessel pressure, and 2. Low water level. The vessel pressure was chosen to be slightly above the relief valve setpoint. The value used in the analysis was 1150 psig. The low low level point chosen is that level at which the recirculation pumps already trip and HPCI and RCIC are initiated. A simplified block diagram of the postulated modification is shown in Figure 4-1. The overall requirements for these modifications are as follows:

A. The system should be diverse from current RPS.

B. No single component failure in the instrument channels or logic shall cause inadvertent injection of all control rods.

C. The system should be testable in service.

D. The system should be designed so that as much as possible no single component failure can prevent 2-RPT and ARI.

E. All hardware should be high quality and be environmentally qualified.

Certain manual actions are required of the operator. Paragraphs 4.1.3 and 4.2.3 show that capability to manually initiate recirculation pump trip and ATWS Rod Injection is available as a backup to automatic initiation. Suppression pool cooling must be initiated manually within 10 minutes of the ATWS event (see paragraph 4.5.2).

Certain alarms and indications are given to the operator to allow him to perform the required manual actions within the time limits. The response to request number 2 of reference 5 discusses the information available to the operator. Additionally, annunciator windows have been added that alarm when the reactor water level or reactor pressure reach the ATWS setpoints. Therefore, at the beginning of the ATWS event, when the recirculation pumps are signalled to trip and the ATWS Rod Injection is automatically initiated, the operator is alarmed that an ATWS has occurred. He then has sufficient time to perform the required manual actions.

4.1 TRIP OF FIELD CIRCUIT BREAKERS OF BOTH RECIRCULATION PUMPS

Since normal scram is assumed to be unavailable for reducing the reactor power and since the transient event is one in which power reduction is necessary, another method of reducing the power is needed for the first 15 seconds of the event. The trip of both recirc pumps causes a quick reduction in core flow which increases the core void generation, thus introducing a negative reactivity thus decreasing the power. In short term considerations, the quick power reduction brings the reactor pressure, neutron flux and fuel surface heat flux down in time to acceptably limit the peak pressure, clad oxidation and peak fuel enthalpy. The analysis is done tripping generator field breakers and the results are applicable to alternate breakers installed between the m-g set and pump motor, since the field breakers result in a slower trip of the pumps and yield more limiting results.

4.1.1 Performance Characteristics

Logic Delay for Trip (Sec)*	≤ 0.53
Pump Inertial Constant (Sec)	≤ 3.0

4.1.2 Automatic ATWS Actuation

Higher Reactor Dome Pressure Setpoint (psig)	<1150
Reactor Low Water Level Setpoint	Low-Low

*Including dynamic response of the sensors, logic, action of the breakers and collapse of generator field.

4.1.3 Manual Actuation

High Torus Water Average Temperature Alarm Setpoint (°F)	≤110
High Reactor Dome Pressure Alarm Setpoint (psig)	≤1150
Reactor Low Water Level Alarm Setpoint	Low-Low

4.2 ATWS ROD INJECTION

ATWS rod injection (ARI) is a means of predominantly diverse blade injection which is motivated mechanically by the normal hydraulic control units and control rod drives, but which utilizes totally separate and diverse logic. The advantage of this method is that the initial signals of high vessel pressure or low water level are used to dump separate valve(s) which cause the pilot air header to bleed down. This bleed down takes approximately 15 seconds after which the reactor is shut down by rod injection. Although this type of rod injection does not eliminate the short term consequences of the assumed failure of normal scram action, it does reduce the long term consequences to nearly those of normal scram situations. The short term consequences are controlled by the early trip of the recirc pumps.

4.2.1 ARI Performance Characteristics

Delay After Air Trip (Sec)	≤15
Logic Delay for Rod Injection (sec)*	≤0.53
Rod Injection Rate After Delay	Same as normal scram

4.2.2 Automatic Actuation

High Reactor Dome Pressure Setpoint (psig)	≤1150
Reactor Low Water Level Setpoint	Low-Low

*Including dynamic response of the sensors and logic.

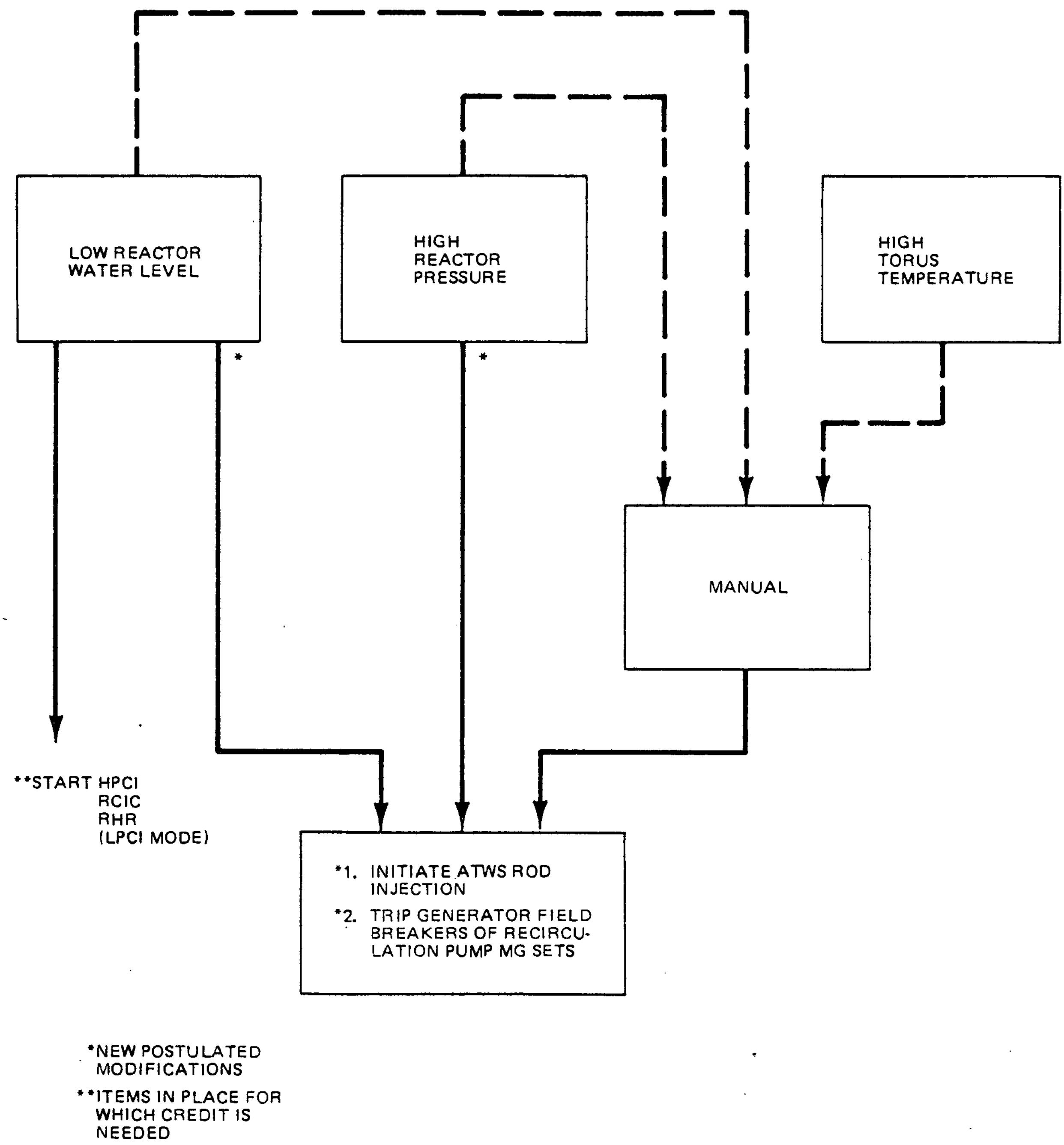

Figure 4-1. Monticello Postulated ATWS Modification Block Diagram

4.4.2 Manual Actuation

High Reactor Dome Pressure Alarm Setpoint (psig)	$\leq$1150
Reactor Low Water Level Alarm Setpoint	Low-Low
High Torus Water Average Temperature Alarm Setpoint (°F)	$\leq$110
Time Required for Manual Initiation After Alarm (Min)	$\leq$10

4.5 DIVERSITY, TESTABILITY, SEPARATION, REDUNDANCY

In order to properly explain the diversity between the normal RPS and ATWS logic schemes, it is necessary to first describe the two individual logic schemes. Within each discussion the process sensor variables which initiate RPS and ATWS are discussed as well as the logic configuration and testing methods. A summary discussion on the diversity of sensor inputs to RPS and the diversity between the RPS logic and the ATWS logic is provided.

Each transient that is expected to be more frequent than one in 40 years is discussed in the attached appendix (answer to question B.13). The discussion of each transient consists of:

1. A description of the transient and how it can occur.

2. The order of normal scram parameter trips that will be generated within the first three minutes after the transient given a normal scram does not occur.

3. The order that the Anticipated Transient Without Scram (ATWS) trips of high reactor pressure or low reactor water that will occur if the transient is severe enough, given that normal scram does not occur.

4.5.1 Description of the Reactor Protection System (RPS) Logic

The RPS logic is described in this section. The description is brief but covers the major design concepts. More detailed information is found in the Monticello FSAR.

The main steam line radiation detectors are located along the main steam lines so they can monitor the radiation levels inside the main steam lines. The detector output is processed through an electronic system to produce an output that is proportional to the radiation activity inside the main steam line. When the radiation trip point is reached, the bistable in the system provides an output to the normal scram system. Calibration of the system is performed by making amplification adjustments to generate a trip output while inputting a known standard signal.

4.5.3.2 RPS Sensor Diversity

Diversity for the RPS sensor inputs is achieved because of the diverse input device trips that operate in diverse environments and have diverse calibration procedures and calibration standards.

4.5.3.3 Diversity Between RPS Logic and ATWS Logic

The diversity between the RPS logic and ATWS logic is achieved by functional application of the logic elements, and location of the logic elements. Some logic elements used in synthesizing the design may be the same, such as trip units, pressure transmitter, pressure switches, relays, power supplies, wires, terminal boards, etc. However, if common elements are used, the application in relation to the trip status will be diverse. Diversity between the RPS and ATWS logic is shown in the table below.

System	Location of logic	Power Source	Logic Contacts During Operation	End Diverse Status	Logic Equation
RPS	RPS cabinets	115 volt AC	closed	energized	One-out-of-two twice
ATWS	ECCS cabinets	125 volt DC	open	de-energized	Two-out-of two or two-out-of-two

4.6 RELIABILITY IMPROVEMENT DUE TO ARI

The proposed addition of a scram air header trip initiated by the ATWS logic provides a diverse means of tripping the reactor protection logic system.

This modification reduces the top major contributor to scram unreliability, i.e., common mode failure of the eight scram contactors. It is estimated that this modification would result in approximately a one (1) order of magnitude reduction in scram system unreliability.

The proposed change does not significantly reduce the potential for miscalibration of all scram sensors as given in WASH-1400. However, it is felt that the probability of miscalibration of several sets of diverse sensors was overly conservative in the WASH-1400 analysis. Current preliminary analyses indicate that this failure mode may drop from the list of major contributors.

5. SUMMARY

The ATWS events were analyzed, showing the need for plant modifications for events if the failure to scram is postulated. The events analyzed covered those transients expected to occur within 40 years, coupled with a failure to scram. The ATWS Prevention system identified includes automatic recirculation pump trip, ATWS Rod Injection, manual suppression pool cooling and reactor vessel water level maintenance by the core cooling systems. The ATWS Prevention system provides satisfactory recovery of the plant for all transients analyzed. The ATWS Rod Injection feature decreases scram unreliability by approximately one order of magnitude. Responses to all identified unanswered NRC questions are provided. In many cases, the responses are generic and reference is given.

RESPONSE TO QUESTION 1

QUESTION

Provide the peak torus water temperature reached during the MSIV closure ATWS. Provide and justify a torus water temperature limit. If the calculated temperature exceeds the limit, discuss the plant modifications needed to keep torus water temperature below the proposed limit. If the peak torus water temperature exceeds 170°F discuss plant modifications needed to keep this temperature below 170°F.

RESPONSE

The peak bulk pool temperature reached following the postualted MSIV closure ATWS event with the postulated modifications is 153.4°F.

The torus water temperature limit during S/RV discharge is dependent on the configuration of the discharge pipe where the steam enters the suppression pool. A local limit of 170°F for an ATWS event has been established for pipes with a single discharge point, such as a uniform cross section pipe. Data presented in NEDO-21078, Test Results Employed by General Electric for BWR Containment and Vertical Vent Loads, October 1975, indicates that additional discharge points (greater energy dispersion) generally improves the thermal performances of the discharge device. Therefore the rams head discharge thermal performance would be expected to be better than a single pipe discharge. Even so, the same thermal limit of 170°F (ATWS) is currently recommended for the rams head discharge.

If the calculated pool temperature exceeds the thermal limit established for the ATWS event, then modifications could be made to the NSS to reduce the energy released to the pool during the event (for the existing discharge device), or the discharge configuration could be modified to raise the thermal limit to a higher temperature.

RESPONSE TO QUESTION B.1

QUESTION

Section 7.1 of NEDO-20626 identifies the systems relied upon to mitigate the consequences of ATWS. Demonstrate the diversity of these systems and their initiating signals from the Reactor Scram System. Further discuss the reliability of these systems to perform their functions during an ATWS event.

RESPONSE

The systems identified in NEDO-20626 for B category plants are not appropriate for Monticello. The appropriate systems for Monticello are specified in Section 4 of this report. The response to the diversity question is contained in Sections 4.5 and 4.6.

e. RHR flow and temperature when RHR is used for suppression pool cooling and decay heat removal

f. Storage capacity of each source of water used to maintain level and remove energy from vessel

g. Operator actions including the time action taken

QUESTION B.5

In an October 7, 1974 letter from I. F. Stuart to V. Stello, GE stated (responses to Question 4) that the condensate storage tank would provide water for HPCI and RCIC for 24 minutes and that the suppression pool is not needed as a source of water for an ATWS event. If this is the case, explain how the plant can be brought to a cold shutdown condition.

RESPONSE

These questions will be addressed below first for all the transients analyzed in Section 3 except the inadvertent opening of a relief valve. Discussion of the latter transient follows. The case of a S/R valve failing to reclose during an ATWS event constitutes a single failure in addition to the common mode failure and is not analyzed for Monticello, an ATWS "C" plant.

Discussion for Transients Other Than Inadvertent Opening of a Relief Valve

In section 3, the time evolution of plant variables in ATWS transients until after achieving hot shutdown was discussed. In this state, the reactor power is due only to decay heat which is removed by steam flow through relief valves into the suppression pool (or, in the case of transients like turbine trip with bypass available, by steam flow into the main condenser). The reactor inventory in the hot shutdown state is made up by HPCI and/or feedwater system flows. The source of the HPCI system flow is the condensate storage tanks which is (75,000 gallons each) sufficient for at least fifty minutes worth of full HPCI flow (3000 gpm). Similarly, the Monticello main condenser wet well is large enough to supply the feedwater system for an approximation of 3 minutes at its full flow even when there is no steam flow into the condenser. Therefore, the HPCI system alone can maintain makeup water to compensate for reactor inventory loss due to decay heat for more than 40 hours after hot shutdown.

When hot standby is achieved by ATWS Rod Injection (ARI), the situation of reactor depressurization can be achieved by manipulating the ADS system and relieving the stored energy of the reactor into the suppression pool. This would result in a suppression pool temperature rise. Whenever the main condenser is available for duty steam flow into the condenser can be established through the turbine bypass line, thus avoiding further steam flow into the suppression pool. The reactor depressurization rate can be controlled by manipulating the ADS valves or the pressure regulator setpoint.

Energy release during reactor depressurization down to 150 psig is expected to result in a suppression pool temperature rise of about 40°F (with only one RHR heat exchanger in the pool cooling mode). The action of one RHR heat exchanger can accomplish a suppression pool cooling rate of ∿6°F/hr when a bulk pool temperature of 160°F is assumed). Therefore if the reactor depressurization (to 150 psig) is performed over a period longer than ∿6.8 hours, the energy release into the suppression pool would not exceed 160°F when both RHR heat exchangers are in the pool cooling mode.

In the Monticello plant the RHR system heat exchangers can assume the function of directly cooling the reactor water only after the reactor is depressurized. Thus, the HPCI system will assume the long term cooling function through reactor depressurization. The HPCI system is automatically initiated by the reactor low low water level and does not require specific operator action.

Figures A.1 and A.2 show the long term traces of reactor power and reactor pressure prior to the start of depressurization for the MSIV closure transient. Reactor water level and containment conditions are shown in Figures 3-4, 3-5 and 3-6. These plots are typical of pressurization transients which are not accompanied by feedwater loss.

Figures A.3, A.4 and A.5 show the plots of reactor power, pressure and water level for the case of loss of AC power ATWS. Containment temperature and pressure plots are shown in Figures 3-8 and 3-9, respectively. These plots are typical of pressurization transients accompanied by loss of feedwater flow early in the transient.

Discussion for the Case of Inadvertent Opening of a Relief Valve

In this case hot shutdown in achieved at approximately 850 seconds with ARI. The reactor will continue to blow down through the inadvertently opened relief valve. The feedwater system can continue to make up the reactor inventory loss for some time after hot shutdown is achieved. Thereafter the HPCI (initiated by Low-Low water level) will assume this function. Since the HPCI pumping capacity is more than twice the rated relief valve flow, reactor inventory can be adequately maintained. Moreover, the condensate storage tank can supply the HPCI system in this reactor water makeup function for more than one hour. Beyond this time, the suppression pool forms a second source of water for the HPCI system which can be utilized by opening appropriate valves. Since in this situation a closed flow path is established, reactor water inventory can be maintained indefinitely.

When ARI function is assumed, the suppression pool temperature at hot shutdown is 120°F. As mentioned before, there would be another estimated rise of 40°F as the reactor blows down to a pressure of 150 psig. (This rise would be somewhat less than 40°F when effect of both RHR heat exchangers in the pool cooling mode is considered.) Thus, the suppression pool temperature would reach 160°F before the reactor pressure is below 150 psig. At reactor pressures equal to or less than this pressure, the steam mass flux through the relief valve would be low enough that the 170°F pool temperature limit would not be critical. Higher temperature limits are justifiable for these conditions. Figures A.6 and A.7 show the long term plots of reactor power and water level.

RESPONSE TO QUESTION B.16

QUESTION

The sensitivity of peak pressure to relief valve capacity is presented in Table 6-3 of NEDO-20626. For each product line, what is the minimum relief capacity? For the minimum relief capacity plants, what is the relief capacity of each valve? What is the probability that a relief valve will not open upon reaching the pressure setpoint? Identify B class plants with lower relief capacity than that used in NEDO-20626. Provide ATWS analyses using the plant with the least relief capacity as basis.

RESPONSE

Monticello has eight relief valves of combined capacity equal to 74.4% of the rated reactor steam flow. The analysis reported in Section 3 takes credit only for six of the eight valves in the reactor peak pressure calculations. The containment conditions are not very sinsitive to whether six or eight valves are used in the analysis. The failure of a relief valve to open in addition to an ATWS is not considered applicable to the Monticello plant.