WBN2Public Resource

From: Bailey, Stewart
Sent: Wednesday, August 25, 2010 9:49 AM
To: Garg, Hukam; Carte, Norbert; Poole, Justin
Subject: FW: Updated NRC RAI Matrix and reference documents
Attachments: 1553 S00 Pkg.pdf

From: Crouch, William D [mailto:wdcrouch@tva.gov]
Sent: Wednesday, August 25, 2010 6:53 AM
To: Bailey, Stewart
Subject: FW: Updated NRC RAI Matrix and reference documents

Part 2

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From: Clark, Mark Steven
Sent: Tuesday, August 24, 2010 5:40 PM
To: Crouch, William D
Cc: Hilmes, Steven A
Subject: Updated NRC RAI Matrix and reference documents

Bill:

Please forward the attached updated matrix and reference documents to the NRC for the Thursday phone call.

Regards,

Steve

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Bechtel Power Corp.
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Paul L. Pace, ADM 1L, Watts Bar Nuclear Plant

WATTS BAR NUCLEAR PLANT (WBN) - UPDATED FINAL SAFETY ANALYSIS REPORT (UFSAR) CHANGE PACKAGE 1553

The purpose of this memorandum is to provide to you FSAR Change Package 1553.

This package contains approved Nuclear Engineering input for incorporation into the UFSAR. Subsequently, this material should not be changed without approval. These changes have been reviewed by Site Engineering lead and support organizations and approval signatures are included. The changes do not have the potential to affect the conclusions stated in the Safety Evaluation Report and do not modify a commitment to the Nuclear Regulatory Commission.

Frank L. Koontz Jr. for

J. E. Maddox
Engineering and Support Manager
EQB 1A-WBN

Attachments
cc (Attachments):
EDMS, WT 3B-K
W. D. Webb, EQB 2N-WBN

ATTACHMENT 3

UPDATED FSAR

VERIFICATION FORM

UFSAR
SECTION(S): 7.2

THESE UFSAR SECTION(S) ACCURATELY REFLECT PLANT CONSTRUCTION AND OPERATION AND:

NO CHANGES ARE REQUIRED: ___________

CHANGES ARE REQUIRED: Yes

ATTACH FSAR PACKAGE NO. 1553

SA/SE NO: WBPLEE-98-080-0

COMMENTS:

REVIEWED BY: See Attached DATE: ___________

APPROVED BY: See Attached DATE: ___________

SAR CHANGE REQUEST

SAR CHANGE PACKAGE NO. 1553

ORIGINATOR: WD Webb DATE:
ORGANIZATION/ADDRESS WBPLEE / EQB 2N PHONE: 1227

CHANGE REQUIRED DUE TO:

☐ DCN/MODIFICATION*
DATE IMPLEMENTED
JUSTIFICATION:

☐ TECH SPEC CHANGE*:
DATE IMPLEMENTED
JUSTIFICATION:

☒ OTHER* UFSAR Section 7.2
JUSTIFICATION: See attached SA/SE WBPLEE-98-080-0

☐ Nonsignificant*
JUSTIFICATION:

*Attach a marked up copy of applicable SAR page, table, revised figures, table of contents, list of tables, etc.

NOTE A safety assessment/safety evaluation is required to accompany any technical SAR change. Nonsignificant changes do not require a safety assessment/evaluation. Contact Site Licensing organization if confusion or uncertainty exists over whether a change is nonsignificant.

PROPOSED CHANGES HAVE BEEN COORDINATED WITH AND ARE CONCURRED BY THE SUPPORTING ORGANIZATIONS, IF APPLICABLE.

NAME	SUPPORTING ORG	DATE
EWB [signature]	SITE ENGINEERING - MN	12/16/98
[signature]	SYSTEM ENGINEERING	12/16/98
[signature]	MAINTENANCE	12/16/98
[signature]	OPERATIONS	12/17/98

SOS 12-16-98

Prepared By: WD Webb WDW Phone: 1227 Date: 12-16-98

Approved By**: [signature] for MCB 12/17/98 Phone: x1533 Date: 12/17/98
Lead Org. Section Supervisor

References (base on design document, if possible):

**Lead organization approval is not required for typographical changes

Licensing Disposition
Approved ☑ Rejected ☐ Amendment No. 1

Licensing Approval***: Rebecca N Mays Date: 10/16/95

Transit to:
Site Licensing Manager
RIMS
Management Services Living FSAR (Issued by Site Licensing)
***Forward to Originator

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7.2 REACTOR TRIP SYSTEM

7.2.1 Description

7.2.1.1 System Description

The reactor trip system automatically keeps the reactor operating within a safe region by shutting down the reactor whenever the limits of the region are approached. The safe operating region is defined by several considerations such as mechanical/hydraulic limitations on equipment, and heat transfer phenomena. Therefore, the reactor trip system keeps surveillance on process variables which are directly related to equipment mechanical limitations, such as pressure, pressurizer water level (to prevent water discharge through safety valves, and uncovering heaters) and also on
2 variables which directly affect the heat transfer capability of the reactor (e.g., ~~flow and~~ reactor coolant flow and temperatures). Still other parameters utilized in the reactor trip system are calculated from various process variables. In any event, whenever a direct process or calculated variable exceeds a setpoint the reactor will be shutdown in order to protect against exceeding the specified fuel design limit, gross damage to fuel cladding or loss of system integrity which could lead to release of radioactive fission products into the containment.

The following systems make up the reactor trip system:

1. Process Protection and Control System [1] and [11]
2. 1 Nuclear Instrumentation System (NIS) [2] and [15]
3. Solid State Logic Protection System [3]
4. Reactor Trip Switchgear
5. Manual Actuation Circuit

The reactor trip system consists of two to four redundant sensors and associated process protection channels, which monitor various plant variables, and two redundant logic trains, which
3 receive input protection actuation signals from the process protection and NIS channels to complete the logical decisions necessary to automatically open the reactor trip breakers.

Each of the two trains, A and B, is capable of opening a separate and independent reactor trip breaker, RTA and RTB, respectively. The two trip breakers in series connect three phase ac power from the rod drive motor generator sets to the rod drive power cabinets, as shown on Figure 7.2-1, Sheet 1. Normally both the dc undervoltage trip coil and the shunt trip relay for each breaker are kept energized allowing power to be available at the rod control power supply cabinets. For reactor trip, a loss of dc voltage to the undervoltage coil releases the trip plunger
4 and trips open the breaker and the shunt trip relay drops out causing the shunt trip coil to energize and also trip the breaker. When either of the trip breakers opens, power is interrupted to the rod drive power supply, and the control rods fall, by gravity, into the core. The rods cannot be withdrawn until the trip breakers are manually reset. The trip breakers cannot be reset until the abnormal condition which initiated the trip is corrected or no longer requires a reactor trip. Bypass breakers BYA and BYB are provided to permit testing of the trip breakers, as discussed in Section 7.2.2.2.

7.2.1.1.1 Functional Performance Requirements

The reactor trip system automatically initiates reactor trip:

1. Whenever necessary to prevent fuel damage for an anticipated operational transient (Condition II),

2. To limit core damage for infrequent faults (Condition III),

3. So that the energy generated in the core is compatible with the design provisions to protect the reactor coolant pressure boundary for limiting fault conditions (Condition IV).

The reactor trip system initiates a turbine trip signal whenever reactor trip is initiated to prevent the reactivity insertion that would otherwise result from excessive reactor system cooldown and to avoid unnecessary actuation of the engineered safety features actuation system.

The reactor trip system provides for manual initiation of reactor trip by operator action.

7.2.1.1.2 Reactor Trips

The various reactor trip circuits automatically open the reactor trip breakers whenever a condition monitored by the reactor trip system reaches a preset level. To ensure a reliable system, high quality design, components, manufacturing, quality control and testing are used. In addition to redundant channels and trains, the design approach provides a reactor trip system which monitors numerous system variables, therefore providing protection system functional diversity. The extent of this diversity has been evaluated for a wide variety of postulated accidents and is detailed in References [4] and [5].

Table 7.2-1 provides a list of reactor trips which are described below. Protection system interlocks are described in Table 7.2-2. The functional logic for reactor trips is shown on Figure 7.2-1.

1. Nuclear Overpower Trips

   The specific trip functions generated are as follows:

   a. Power range high neutron flux trip

      The power range high neutron flux trip circuit trips the reactor when two of the four power range channels exceed the trip setpoint.

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There are two independent bistables, each with its own trip setting used for a high and a low range trip setting. The high trip setting provides protection during normal power operation and is always active. The low trip setting, which provides protection during startup, can be manually bypassed when two out of the four power range channels read above approximately 10% power (P-10). Three out of the four channels below 10% automatically reinstates the trip function.

b. Intermediate range high neutron flux trip

The intermediate range high neutron flux trip circuit trips the reactor when one out of the two intermediate range channels exceeds the trip setpoint. This trip, which provides protection during reactor startup, can be manually blocked if two out of
5 four power range channels are above ~~approximately 10% power (~~P-10~~)~~. Three out
of the four power range channels below this value automatically reinstates the intermediate range high neutron flux trip. The intermediate range channels (including detectors) are separate from the power range channels. The intermediate range channels can be individually bypassed at the nuclear instrumentation racks to permit channel testing during plant shutdown or prior to startup. This bypass action is annunciated on the control board.

c. Source range high neutron flux trip

The source range high neutron flux trip circuit, trips the reactor when one of the two source range channels exceeds the trip setpoint. This trip, which provides protection during reactor startup and plant shutdown, can be manually bypassed
1 when one of the two intermediate range channels exceeds ~~reads above~~ the P-6
setpoint value and is automatically reinstated when both intermediate range channels decrease below the P-6 setpoint value. This trip is also automatically bypassed by two out of four logic from the power range protection interlock (P-
6 10). This trip function can also be reinstated below P-10 by ~~an administrative~~ a
manual action requiring simultaneous ~~manual~~ actuation of two control board mounted switches, one in each of the two protection logic trains. The source range trip point is set between the P-6 setpoint and the maximum source range power level. The channels can be individually bypassed at the nuclear instrumentation racks to permit channel testing during plant shutdown or prior to startup. This bypass action is annunciated on the control board.

d. Power range high positive neutron flux rate trip

This circuit trips the reactor when a sudden abnormal increase in nuclear power occurs in two out of four power range channels. This trip provides DNB protection against rod ejection accidents of low worth from midpower and is always active.

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e. Power range high negative neutron flux rate trip

This circuit trips the reactor when a sudden abnormal decrease in nuclear power occurs in two out of four power range channels. This trip provides protection against two or more dropped rods and is always active. Protection against one dropped rod is not required to prevent occurrence of DNB per Section 15.2.3.

Figure 7.2-1, Sheet 2, shows the logic for all of the nuclear overpower and rate trips. Detailed functional descriptions of the equipment associated with these functions are given in References [2] and [15].

2. Core Thermal Overpower Trips

The specific trip functions generated are as follows:

a. Overtemperature ΔT trip

This trip protects the core against low DNBR and trips the reactor on two out of four coincidence with one set of temperature measurements per loop. The setpoint for this trip is continuously calculated by the Eagle-21 process protection circuitry for each loop by solving the following equation:

$$OT\Delta T \text{ Setpoint} = \Delta T_0\left[K_1 - K_2\left(\frac{1+\tau_1 s}{1+\tau_2 s}\right)(T - T') + K_3(P - P') - f_1(\Delta I)\right]$$

An overtemperature ΔT reactor trip occurs when

$$\Delta T\left(\frac{1+\tau_4 s}{1+\tau_5 s}\right) > OT\Delta T \text{ Setpoint}$$

where:

ΔT	=	Measured temperature difference between hot and cold leg, °F
ΔT_o	=	Indicated loop ΔT at rated thermal power (RTP), °F
K_1	=	Reference trip setpoint
K_2	=	Penalty or benefit multiplier for deviation from indicated T_{avg}, /°F
K_3	=	Penalty or benefit multiplier for deviation from reference pressure, /psig
τ_1, τ_2	=	Lead/lag time constants for T_{avg} compensation, seconds
τ_4, τ_5	=	Lead/lag time constants for ΔT compensation, seconds
s	=	Laplace transform operator, sec^{-1}
T	=	Measured RCS average temperature (T_{avg}), °F
T′	=	Indicated loop T_{avg} at RTP, °F
P	=	Measured pressurizer pressure, psig
P′	=	Nominal RCS operating pressure, psig
$f_1(\Delta I)$	=	Power shaped penalty - function of the indicated difference between the top and bottom detectors of the power range neutron ion chambers.

7 Values of these parameters are provided in the Technical Specification or are controlled by plant procedures.

Note: Additional information on associated tau values (τ_6 and τ_7) are provided in Section 7.2.1.1.4.

1 A separate long ion chamber unit supplies the flux signals for each overtemperature ΔT trip channel.

Increases in ΔI beyond a predefined deadband result in a decrease in trip setpoint. Refer to Figure 7.2-2.

The required one pressurizer pressure parameter per loop is obtained from separate sensors connected to three pressure taps at the top of the pressurizer. Four pressurizer pressure signals are obtained from the three taps by connecting one of the taps to two pressure transmitters. Refer to Section 7.1.2.2 for a discussion of independence of redundant sense lines.

The logic for this function is shown on Figure 7.2-1, Sheet 3. A detailed functional description of the process equipment associated with this function is contained in Reference [11].

b. Overpower ΔT trip

This trip protects against excessive power (fuel rod rating protection) and trips the reactor on two out of four coincidence with one set of temperature measurements per loop. The setpoint for each channel is continuously calculated by the process protection circuitry using the following equation:

$$OP\Delta T \ \ Setpoint = \Delta T_o \left[K_4 - K_5 \left(\frac{\tau_3 s}{1+\tau_3 s} \right) T - K_6 (T - T") - f_2(\Delta I) \right]$$

An overpower ΔT reactor trip occurs when:

$$\Delta T \left(\frac{1 + \tau_4 s}{1 + \tau_5 s} \right) > OP\Delta T \text{ Setpoint}$$

8 where: ΔT, ΔT_o, T, $\cancel{T'}$, τ_4, τ_5, s are defined in Section 7.2.1.1.2(2)(a) Overtemperature ΔT Trip and

K_4 = Reference Trip setpoint
K_5 = Penalty multiplier for rate of change in T_{avg}/°F
K_6 = Penalty or benefit multiplier for deviation from reference T_{avg}/°F
τ_3 = Lag time constant for T_{avg} compensation, seconds
T" = Indicated loop T_{avg} at RTP, °F
$f_2(\Delta I)$ = Power shape penalty function, typically set to 0 for all ΔI

7 Values of these parameters are provided in the Technical Specifications or are controlled by plant procedures.

Note: Additional information on associated tau values (τ_6 and τ_7) are provided in Section 7.2.1.1.4.

The source of temperature and flux information is identical to that of the overtemperature ΔT trip and the resultant overpower ΔT setpoint is compared to the same ΔT. The trip logic for this function is shown on Figure 7.2-1, Sheet 3. A detailed functional description of the process equipment associated with this function is contained in Reference [11].

3. Reactor Coolant System Pressurizer Pressure and Water Level Trips

The specific trip functions generated are as follows:

a. Pressurizer low pressure trip

The purpose of this trip is to protect against low pressure which could lead to DNB. The parameter being sensed is reactor coolant pressure as measured in the pressurizer. Above P-7 the reactor is tripped when two out of four pressurizer pressure measurements (compensated for rate of change) fall below preset limits. This trip is blocked below P-7 to permit startup. The trip logic and interlocks are given in Table 7.2-1.

The trip logic is shown on Figure 7.2-1, Sheet 2. A detailed functional description of the process equipment associated with the function is contained in References [5] and [11].

b. Pressurizer High Pressure Trip

The purpose of this trip is to protect the reactor coolant system against system overpressure. The same sensors and transmitters used for the pressurizer low pressure trip are used for the high pressure trip except that separate comparators are used for trip. These comparators trip the reactor when two out of four uncompensated pressurizer pressure signals exceed preset limits as listed in Table 7.2-1. There are no interlocks or permissives associated with this trip function.

The logic for this trip is shown on Figure 7.2-1, Sheet 2. The detailed functional description of the process equipment associated with this trip is provided in References [5] and [11].

c. Pressurizer High Water Level Trip

This trip is provided as a backup to the high pressurizer pressure trip and serves to prevent water relief through the pressurizer safety valves. Above P-7, the reactor is tripped when two out of three pressurizer water level measurements exceed preset limits. This trip is blocked below P-7 to permit startup. The coincidence logic and interlocks of pressurizer high water level signals are given in Table 7.2-1.

The trip logic for this function is shown on Figure 7.2-1, Sheet 2. A detailed description of the process equipment associated with this function is contained in References [5] and [11].

4. Reactor Coolant System Low Flow Trips

These trips protect the core from DNB in the event of a loss of coolant flow situation. The means of sensing the loss of coolant flow are as follows:

1 a. Low Reactor Coolant Flow Trip

9 ~~The parameter sensed is r~~Reactor coolant flow~~.~~ ~~Four~~ measurements are derived from elbow taps in each coolant loop ~~are used as flow devices that indicate the status of reactor coolant flow~~. The basic function of ~~this~~ these devices is to provide information as to whether or not a reduction in flow has occurred. An output signal from two out of the three comparators in a loop would indicate a low flow in that loop. Above P-8, low flow in one loop will trip the reactor. Between P-7 and P-8, low flow in two out of four loops will result in a reactor trip. This trip is blocked below P-7 to permit startup.

The coincidence logic and interlocks are given in Table 7.2-1. The logic for this trip is shown on Figure 7.2-1, Sheet 3. A detailed functional description of the process equipment associated with the trip function is contained in References [5] and [11].

b. Reactor Coolant Pump Undervoltage Trip

This trip is required in order to protect against low flow which can result from loss of voltage to more than one reactor coolant pump motor (e.g., from plant loss of voltage or reactor coolant pump breakers opening). This trip is blocked below P-7 to permit startup.

There is one undervoltage sensing relay for each pump motor connected at the load side of each reactor coolant pump breaker. These relays provide an output signal when the pump voltage goes below setpoint ~~approximately 70% of rated voltage~~. Signals from these relays are time delayed to prevent spurious trips caused by short term voltage perturbations. The coincidence logic and interlocks are given in Table 7.2-1. The trip logic is shown on Figure 7.2-1, Sheet 3. 10

c. Reactor Coolant Pump Underfrequency Trip

This trip provides protection against low reactor coolant flow resulting from bus underfrequency (e.g., power grid frequency transients). Above the P-7 interlock setpoint, an underfrequency condition on two out of four reactor coolant pump (RCP) motors will trip the reactor and open all of the RCP circuit breakers.

There is one underfrequency sensing relay connected to the load side of each RCP breaker ~~with a setpoint of approximately 57 Hz~~. The signals from these relays are time delayed to prevent spurious trips caused by short-term frequency perturbations. The coincidence logic and interlocks are given in Table 7.2-1. The trip logic is shown on Figure 7.2-1, Sheet 3. 10

Westinghouse analysis of loss of flow accidents caused by power system frequency transients [Reference 6] has shown that the reactor is adequately protected by the underfrequency reactor trip for frequency ~~of~~ decay rates of less than 6.8 Hz/sec without taking credit for the RCP breaker trip. A grid analysis of the TVA power system determined the maximum system frequency decay rate to be less than 5 Hz/sec. Consequently, the RCP breaker trip on underfrequency is not included in the protection system. 1

5. Low-Low Steam Generator Water Level Trip (including Trip Time Delay)

This trip protects the reactor from loss of heat sink in the event of a loss of feedwater to one or more steam generators or a major feedwater line rupture outside containment. This trip is actuated on two out of three low-low water level signals occurring in any steam generator. If a low-low water level condition is detected in one steam generator, signals are generated to trip the reactor and start the motor-driven auxiliary feedwater pumps. If a low-low water level condition is detected in two or more steam generators, a signal is generated to start the turbine-driven auxiliary feedwater pump as well.

The signals to actuate the reactor trip and start auxiliary feedwater pumps are delayed through the use of a Trip Time Delay (TTD) system for reactor power levels below 50% of RTP. Low-Low water level in any steam generator will generate a signal which starts an elapsed time trip delay timer. The allowable trip time delay is based upon the prevailing power level at the time the low-low level trip setpoint is reached and the number of steam generators that are affected. If power level rises after the trip time delay setpoints have been determined, the trip time delay is re-determined (i.e., decreased) according to the increase in power level.

At this point the timer will continue timing from the original timer initiation. However, the trip time delay setpoints are not increased if the power level decreases after the TTD timer has started. The use of this delay allows added time for natural steam generator level stabilization or operator intervention to avoid an undesirable inadvertent protection system actuation.

There are no interlocks or permissives associated with this trip function. The logic for this protective function is shown on Figure 7.2-1, Sheet 4. A detailed functional description of the process equipment associated with this function is contained in References [11] and [14].

6. <u>Reactor Trip on a Turbine Trip</u>

The reactor trip on a turbine trip is actuated by two out of three logic from low autostop oil pressure signals or by closed signals from all four turbine steam stop valves. A turbine trip causes a direct reactor trip above P-9.

The reactor trip on turbine trip provides additional protection and conservatism beyond that required for the health and safety of the public. This trip is included as part of good engineering practice and prudent design. No credit is taken in any of the accident analyses (Chapter 15) for this trip.

11 ~~Separate routing is maintained for the four turbine trip channel sets. Each of three sets includes the signal from low autostop oil pressure and the signal from closing of the steam stop valve. The fourth set consists of the signal from closing of the steam stop valve. The~~ Channel separation ~~of these four channel sets~~ is maintained from the sensors to the reactor protection system logic input cabinets for both the low autostop oil pressure signals and the steam stop valves closed signals. ~~These channel routings~~ This design meets the redundance and separation requirements identical to those for Class 1E circuits. Mounting and location is in non-seismic Category I structures.

The turbine provides anticipatory trips to the reactor protection system from contacts which change position when the turbine stop valves close or when the turbine autostop oil pressure goes below its setpoint.

One of the design bases considered in the protection system is the possibility of an earthquake. With respect to these contacts, their functioning is unrelated to a seismic event in that they are anticipatory to other diverse parameters which cause reactor trip. The contacts are closed during plant operation and open to cause reactor trip when the turbine is tripped. No power is provided to the protection system from the contacts; they merely serve to interrupt power to cause reactor trip.

This design functions in a de-energize-to-trip fashion to cause a reactor trip if power is interrupted in the trip circuitry. This ensures that the protection system will in no way be degraded by this anticipatory trip because seismic design considerations do not form part of the design bases for anticipatory trip sensors. (The reactor protection system cabinets which receive the inputs from the anticipatory trip sensors are seismically qualified as discussed in Section 3.10.). The anticipatory trips thus meet the intent of IEEE-279-1971, including redundancy, separation, single failure, etc. Seismic qualification of the contacts sensors is not required.

The logic for this trip is shown on Figure 7.2-1, Sheet 3.

7. Safety Injection Signal Actuation Trip

A reactor trip occurs when the Safety Injection System is actuated. The means of actuating the Safety Injection System are described in Section 7.3. This trip protects the core against a loss of reactor coolant or heat sink.

1 Figure ~~7.2-1~~7.3-3, Sheet ~~1~~3, shows the logic for this trip. A detailed functional description of the process equipment associated with this trip function is provided in References [5] and [11].

8. Manual Trip

The manual trip consists of two switches with two outputs on each switch. One output is used to actuate the train A reactor trip breaker, the other output actuates the train B reactor trip breaker. Operating a manual trip switch removes the voltage from the undervoltage trip coil and energizes the shunt trip coil.

There are no interlocks which can block this trip. Figure 7.2-1, Sheet 2, shows the manual trip logic.

7.2.1.1.3 Reactor Trip System Interlocks

1. Power Escalation Permissives

1 The overpower protection provided by the ~~out-of-core~~ excore nuclear instrumentation consists of three overlapping~~,~~ ranges. Continuation of startup operation or power increase requires a permissive signal from the higher range instrumentation channels before the lower range level trips can be manually blocked by the operator.

A one of two intermediate range permissive signal (P-6) is required prior to source range trip blocking. Source range level trips are automatically reactivated when both intermediate range channels are below the permissive (P-6) level. There are two manual reset switches for administratively reactivating the source range trip when between permissive P-6 and P-10 if required. Source range trip block is always maintained when above permissive P-10.

The intermediate range trip and power range (low setpoint) trip can only be blocked after satisfactory operation and permissive information are obtained from two of four power range channels. Four individual blocking switches are provided so that the low range power range trip and intermediate range trip can be independently blocked (one switch for each train). These trips are automatically reactivated when any three of the four power range channels are below permissive P-10, thus ensuring automatic activation to more restrictive trip protection.

12 The development of permissives P-6 and P-10 is shown on Figure 7.2-1, Sheet 2. These ~~All of the~~ permissives ~~are digital; they and~~ are derived from analog signals in the nuclear power range and intermediate range channels.

See Table 7.2-2 for the list of protection system interlocks.

2. Blocks of Reactor Trips at Low Power

Interlock P-7 blocks a reactor trip below approximately 10% of full power on a low reactor coolant flow in more than one loop, reactor coolant pump undervoltage, reactor coolant pump underfrequency, pressurizer low pressure, or pressurizer high water level. ~~See Figure 7.2-1, Sheets 2 and 3, for permissive applications.~~ The low power block signal
1 is derived from three out of four power range neutron flux signals below the setpoint in coincidence with two out of two turbine impulse chamber pressure signals below the setpoint (low plant load). See Figure 7.2-1, Sheets 2 and 3, for the derivation and applications of P-7.

The P-8 interlock blocks a reactor trip when the plant is below approximately 48% of full power, on a low reactor coolant flow in any one loop. The block action (absence of the P-8 interlock signal) occurs when three out of four neutron flux power range signals are
13 below the setpoint. Thus, below the P-8 setpoint, the reactor ~~will be allowed to operate with one inactive loop and~~ trip will not occur until two loops are indicating low flow. See Figure 7.2-1, Sheet 3, for derivation of P-8 and applicable logic.

The P-9 interlock blocks a reactor trip on a turbine trip when the plant is below approximately 50% of full power. The block action (absence of the P-9 interlock signal) occurs when three out of four neutron flux power range signals are below the setpoint. Thus, below the P-9 setpoint, the reactor will not trip directly from a turbine-tripped signal but will allow the reactor control system, utilizing steam dump to the condenser as an artificial load, to bring the reactor to zero power. See Figure 7.2-1, Sheet 2, for derivation of P-9, and Sheet 3 for logic applications.

See Table 7.2-2 for the list of protection system blocks.

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7.2.1.1.4 Reactor Coolant Temperature Sensor Arrangement and Calculational Methodology

The individual narrow range cold and hot leg temperature signals required for input to the reactor trip circuits and interlocks are obtained using RTDs installed in each reactor coolant loop.

The cold leg temperature measurement on each loop is accomplished with two narrow range RTDs mounted in thermowells. The cold leg sensors are inherently redundant in that either sensor can adequately represent the cold leg temperature measurement.

The hot leg temperature measurement on each loop is accomplished with three narrow range RTDs mounted in thermowells spaced 120 degrees apart around the circumference of the reactor coolant pipe for spatial variations.

These cold and hot leg narrow range RTD signals are input to the process protection system digital electronics and are processed as follows:

The two cold leg temperature signals are subjected to range and consistency checks and then averaged to provide a group value for T cold.

A consistency check is performed on the T_{cold} input signals. If these signals agree within an acceptance interval (DELTAC), the group quality is set to GOOD. If the signals do not agree within the acceptance tolerance DELTAC, the group quality is set to BAD and the individual signal qualities are set to POOR. The average of the two signals is used to represent the group in either case. If an input signal is manually disabled or subject to a diagnosed hardware failure, the group is represented by the active signal. DELTAC is a fixed input parameter based on operating experience. One DELTAC value is required for each loop/protection set.

The following parameters are used in conjunction with the Overtemperature ΔT and Overpower ΔT reactor trips:

$T_{c_{ji}}$ = ~~jth~~ narrow range T_{cold} input signal ~~from loop i~~

14 $T^f_{c_{ji}}$ = Filtered T_{cold} signal ~~for the jth RTD~~; = $T_{c_{ji}}(1/(1 + \tau_7 s))$

~~where: j = 1, 2 and i = 1 to 4~~

τ_7 = Time constant utilized in the lag compensator for T_{cold}. Typically set to 0.0 sec.

$T^f_{c\,ave_i}$ = Group average of the valid T_{cold} input signals

14 ~~= $(T^f_{c(j-1)i} + T^f_{cji})/2$ for two valid input signals (j=2)~~

~~= T^f_{cji} for one valid input signal (j=1)~~

~~where: i = 1 to 4~~

S is defined in Section 7.2.1.1.2

Each of the three hot leg temperature signals is subjected to a range check~~;~~ and utilized to calculate an estimated average hot leg temperature which is consistency checked against the other two estimates for average hot leg temperature.

Then an average of the three estimated hot leg temperatures is computed and the individual signals are checked to determine if they agree within $\pm$DELTAH of the average value. If all of the signals do agree within $\pm$DELTAH of the average value, the group quality is set to GOOD. The
14 group value ~~($T^f_{h\,ave-i}$)~~ is set to the average of the three estimated average hot leg temperatures.

If the signal values do not all agree within $\pm$DELTAH of the average, the algorithm will delete the signal value which is furthest from the average. The quality of this signal will be set to POOR and a consistency check will then be performed on the remaining GOOD signals. If these signals pass the consistency check, the group value will be taken as the average of these GOOD signals and the group quality will be set to POOR. However, if these signals again fail the consistency check (within $\pm$DELTAH), then the group value will be set to the average of these two signals; but the group quality will be set to BAD. All of the individual signals will have their quality set to POOR. If one or two input signals is manually disabled or subject to a diagnosed hardware failure, the group value is based on the unaffected signal(s). DELTAH is a fixed input parameter based on temperature distribution tests with the hot leg and operating experience. One DELTAH value is required for each loop/protection set.

The following parameters are used in conjunction with the Overtemperature ΔT and Overpower ΔT reactor trips:

T_{hji} = ~~jth~~ narrow range T_{hot} input signal ~~from loop i~~

T^f_{hji} = Filtered T_{hot} signal ~~for the jth RTD~~; = $T_{hji}(1/(1 + \tau_6 s))$

14 ~~where: j = 1 to 3 and i = 1 to 4~~

τ_6 = Time constant utilized in the lag compensator for T_{hot}. Typically set to 0.0 sec

$T^f_{h\,ave-i}$ = Group average of the valid T_{hot} input signals

~~= $(T^f_{h(j-2)i} + T^f_{h(j-1)i} + T^f_{hji})/3$ for three valid input signals (j=3)~~

~~= $(T^f_{h(j-1)i} + T^f_{hji})/2$ for two valid input signals (j=2)~~

~~= T^f_{hji} for one valid input signal (j=1)~~

~~where: i = 1 to 4~~

14 The estimated average hot leg temperature is derived from each T_{hot} ~~hot~~ input signal as follows:

~~$\overline{T}_{H_{ji}} = T^f_{h_{ji}} - P_{Bi} S^o_{ji}$ = estimated T_{hot} average~~

$$\overline{T}_h = T^f_h - P_B S^o$$

Where:

P_B ~~P_{Bi}~~ = power fraction being used to correct the bias value being used for any power level

$$\cancel{P_{B_i} = \left(\frac{(T^f_{h_{ave_i}} - T^f_{c_{ave_i}})}{\Delta T_{oi}}\right)} \quad P_B = \left(\frac{(T^f_{h_{ave}} - T^f_{c_{ave}})}{\Delta T_o}\right)$$

ΔT_o ~~ΔT_{oi}~~ = ~~is~~ the indicated loop ΔT at rated thermal power.

S^o ~~S^o_{ji}~~ = manually input bias which corrects the individual T_{hot} ~~hot~~ RTD value to the loop average.

ΔT and T_{avg} are calculated as follows:

$$\cancel{\Delta T_i = T^f_{h_{ave_i}} - T^f_{c_{ave_i}}}$$

$$\Delta T = T^f_{h_{ave}} - T^f_{c_{ave}}$$

14

$$\cancel{T_{avg_i} = \frac{(T^f_{h_{ave_i}} + T^f_{c_{ave_i}})}{2.0}}$$

$$T_{avg} = \frac{(T^f_{h_{ave}} + T^f_{c_{ave}})}{2.0}$$

The calculated values for ΔT and T_{avg} ~~T_{ave}~~ are then utilized for both the remainder of the Overtemperature and Overpower ΔT protection channels and channel outputs used for control purposes.

15 ← **INSERT (7.2.1.1.4)**

7.2.1.1.5 Pressurizer Water Level Reference Leg Arrangement

16 ~~The design of the pressurizer water level instrumentation includes a slight modification of the usual tank level arrangement using differential pressure between an upper and a lower tap. The modification consists of the use of a sealed reference leg instead of the conventional open column of water. Refer to Section 7.2.2.3.4 for an analysis of this arrangement.~~

INSERT

15

INSERT for 7.2.1.1.4 (revised/moved from 7.2.2.3.2)

The accuracy of the narrow range ~~resistance temperature detector~~ RTD loop temperature measurements is demonstrated during plant startup tests and periodically with surveillance tests. Testing compares temperature measurements from the narrow range ~~resistance temperature detectors~~ RTDs with one another as well as with the temperature measurements obtained from the wide range ~~resistance temperature detectors~~ RTDs located in the hot leg and cold leg piping of each loop. The comparisons are done with the reactor coolant system in an isothermal condition. ~~As part of the plant startup tests, t~~The narrow range ~~resistance temperature detector~~ RTD signals ~~will be~~ are also compared with the core exit thermocouple signals during plant startup tests.

During plant startup tests, ΔT measurements obtained from the hot leg and cold leg narrow range loop ~~resistance temperature detectors~~ RTDs are compared to plant power, and, if required, normalized to plant power. The absolute value of ΔT versus plant power is not important, per se, as far as reactor protection is concerned. Reactor trip system setpoints are based upon percentages of the indicated ΔT at nominal full power rather than on absolute values of ΔT. This is done to account for loop differences which are inherent. Therefore the percent ΔT scheme is relative, not absolute, and thus provides better protective action without ~~the expense of~~ sacrificing accuracy. ~~As part of the plant startup tests, the narrow range resistance temperature detector signals will be compared with the core exit thermocouple signals.~~

16 **INSERT for 7.2.1.1.5 (revised text from 7.2.2.3.4)**

The pressurizer water level instrumentation consists of three independent, redundant instrument channels which provide reactor trip and control functions. The associated high and low pressure sense lines for each level channel connect to the upper (vapor-filled) and lower (liquid-filled) regions of the pressurizer, respectively, and satisfy the independence requirements specified in Section 7.1.2.2. The high pressure sense line is called a reference leg because the line must be liquid filled and the fill elevation must be maintained at a known point by use of a condensing chamber. The main portion of the reference leg consists of a remote-seal/capillary system (integral to the level transmitter) which provides a mechanical seal (bellows) between the process fluid and the capillary line fill-fluid. The location of the remote seal is required to be 12-inches or less (measured vertically) from the associated condensing chamber. The condensing chamber and downstream piping is uninsulated and is thus cooled by the ambient environment. This remote seal location requirement minimizes the potential adverse effects of a loss of condensate between the condensing chamber and the remote seal due to a sudden RCS depressurization event. During reactor operation, the condensate could contain high concentration of dissolved hydrogen gas. Upon a rapid RCS depressurization event, the resulting dissolution of the hydrogen gas would force the condensate from the line segment between the remote seal and the condensing chamber. This remote seal location requirement limits the maximum head pressure loss error for this event to approximately 12-inches.

Pressurizer level channel maintenance features include transmitter/remote seal isolation and equalization capability without affecting other redundant channels. Also, the condensing chamber and remote seal housing can be remotely vented by use of permanently installed vent lines with manual isolation valves.

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7.2.1.1.6 Process Protection System

The process protection instrumentation system is described in References [1] and [11]. The nuclear instrument system is described in References [2] and [15]. Reference [2] is applicable to the power range only.

7.2.1.1.7 Solid State Logic Protection System

The solid state logic protection system takes binary inputs from the process protection and nuclear instrument channels and other plant equipment corresponding to conditions (normal/abnormal) of plant parameters. The system combines these signals in the required logic combination and generates a trip signal (no voltage) to the undervoltage coils and the shunt trip relays (which energize the shunt trip coils) of the reactor trip circuit breakers when the necessary combination of signals occurs. The system also provides annunciator, status light and computer
17 input signals which indicate the condition of ~~comparator input signals,~~ partial trip and full trip functions and the status of the various blocking, permissive and actuation functions. In addition, the system includes means for semi-automatic testing of the logic circuits. A detailed description of this system is given in Reference [3].

7.2.1.1.8 Isolation Devices

In certain applications, control signals and other non-protective functions are derived from individual protection channels through isolation devices contained in the protection channel, as permitted by IEEE Standard 279-1971. The isolation devices are part of the protection system and are located in the process protection racks. By definition, non-protective functions include those signals used for control, remote process indication, and computer monitoring.

Isolation device qualification type tests are described in References [7], [8], and [11].

7.2.1.1.9 Energy Supply and Environmental Variations

The energy supply for the reactor trip system is described in Chapter 8. The environmental variations, throughout which the system will perform, are given in Section 3.11 and Chapter 8.

As documented in Reference [7], testing was performed on the Eagle 21 Process Protection System to demonstrate that the Eagle 21 system remained operational before, during and after applied noise, fault, surge withstand, electro-magnetic interference (EMI) and Radio Frequency Interference (RFI) operating conditions. Objectives accomplished by the test demonstrated that the physical independence of the non-class 1E and Class 1E circuitry was maintained and that the system was designed to withstand worst-case noise environment conditions.

7.2.1.1.10 Setpoints

The setpoints that require trip action are given in the Technical Specifications.

7.2.1.1.11 Seismic Design

The seismic design considerations for the reactor trip system are given in Section 3.10. This design meets the requirements of Criterion 2 of the 1971 General Design Criteria (GDC).

7.2.1.2 Design Bases Information

The information given below presents the design bases information requested by Section 3 of IEEE Standard 279-1971, Reference [9]. The reactor trip logic is presented in Figure 7.2-1, Sheets 1 through 4.

7.2.1.2.1 Generating Station Conditions

The following are the generating station conditions requiring reactor trip.

1. DNBR approaching the limiting value.

2. Power density (kilowatts per foot) approaching rated value for Condition II events (See Chapter 4 for fuel design limits).

3. Reactor coolant system overpressure creating stresses approaching the limits specified in Chapter 5.

7.2.1.2.2 Generating Station Variables

The following are the variables required to be monitored in order to provide reactor trips (see Table 7.2-1).

1. Neutron flux
2. Reactor coolant temperature
3. Reactor coolant system pressure (pressurizer pressure)
4. Pressurizer water level
5. Reactor coolant flow
6. Reactor coolant pump bus voltage and frequency
7. Steam generator water level
8. Turbine-generator operational status (autostop oil pressure and stop valve position).

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7.2.1.2.3 Spatially Dependent Variables

Reactor coolant temperature is a spatially dependent variable. See Section 7.3.1.2.3 for a discussion.

7.2.1.2.4 Limits, Margins and Levels

18 The parameter values that will require reactor trip are given in the Technical Specifications ~~and in Chapter 15, Accident Analyses~~. Chapter 15 demonstrates that the setpoints used in the Technical Specifications are conservative.

The setpoints for the various functions in the reactor trip system have been analytically determined such that the operational limits so prescribed will prevent fuel rod clad damage and loss of integrity of the reactor coolant system as a result of any ANS Condition II incident. As such, during any ANS Condition II incident, the reactor trip system limits the following parameters to:

1. Minimum DNBR - limiting value.

2. Maximum system pressure = 2750 psia

3. Fuel rod maximum linear power - maximum rated power

The accident analyses described in Section 15.2 demonstrate that the functional requirements as specified for the reactor trip system are adequate to meet the above considerations, even assuming, for conservatism, adverse combinations of instrument errors (Refer to Table 15.1-3). A discussion of the safety limits associated with the reactor core and reactor coolant system, plus the limiting safety system setpoints, are presented in the Technical Specifications. The Technical Specifications incorporate both nominal and limiting setpoints. Nominal settings of the setpoints are more conservative than the limiting settings. This allows for calibration uncertainty and instrument channel drift without violating the limiting setpoint. Automatic initiation of protective functions occurs at the nominal setpoints (plus or minus the allowed tolerances). The
1 methodology used to derive the setpoints is documented in References [13] and [16]. A further discussion on trip setpoints is given in Section 7.2.2.1.1.

7.2.1.2.5 Abnormal Events

The malfunctions, accidents or other unusual events which could physically damage reactor trip system components or could cause environmental changes are as follows:

1. Earthquakes (see Sections 2.5 and 3.7).

2. Fire (see Section 9.5)

3. Explosion (hydrogen buildup inside containment) (see Section 6.2).

4. Missiles (see Section 3.5).

5. Flood (see Sections 2.4 and 3.4).

6. Wind and Tornadoes (see Section 3.3).

The reactor trip system fulfills the requirements of IEEE Standard 279-1971 to provide automatic protection and to provide initiating signals to mitigate the consequences of faulted conditions. The reactor trip system provides protection against destruction of the system from fires, explosions, floods, wind, and tornadoes (see each item above). The discussions in Section 7.1.2.1.7 and this section adequately address or reference the Safety Analysis Report coverage of the effects of abnormal events on the reactor trip system in conformance with applicable General Design Criteria.

7.2.1.2.6 Minimum Performance Requirements

1. Reactor Trip System Response Time

   Reactor trip system response time is defined in Section 7.1. The maximum allowable time delays in generating the reactor trip signal are provided in the Technical Requirements Manual. These values are verified in accordance with the Technical Specifications and are consistent with the safety analyses. See Table 7.1-1 Note 1 for a discussion of periodic response time verification capabilities.

2. Reactor Trip Accuracies

   Accuracy is defined in Section 7.1. Reactor trip accuracies are given in Reference [13].

3. Protection System Ranges

   Typical Protection System ranges are tabulated in Table 7.2-3.

7.2.1.3 Final Systems Drawings

19 Functional logic ~~block~~ diagrams, electrical schematics ~~elementaries~~ and other drawings documenting the protection system design ~~required to assure electrical separation and perform a safety review~~ are listed ~~provided~~ in Table 1.7-1.

7.2.2 Analyses

A reliability study for the reactor trip and engineered safety features function of the Eagle 21
1 process protection system hardware was ~~has been~~ performed~~. The basis for this study was~~ to compare the availability of the Eagle 21 digital system with the previous ~~existing~~ implementation of the same function using analog hardware. Availability is defined as the probability that a system will perform its intended function (e.g., actuate a partial trip) at a randomly selected instant in time. Results of the availability study determined that the Eagle 21 digital system is commensurate with an equivalent analog process protection system availability although no credit was given to the Eagle 21 process protection features of automatic surveillance testing, self calibration and self-diagnostics when the study was performed. It is expected that if credit were
20 given to the Eagle 21 self diagnostic features ~~(EPSOM checksums, RAM checks, Math Co-Processor checks and Loop Cycle Time checks)~~, automatic surveillance testing and self calibration capabilities, system availability would be improved. Therefore, the impact on the system operation due to channel drift being corrected by the Eagle 21 self-calibration feature and the impact on system downtime because of the automatic surveillance/self-diagnostic features~~,~~ is ~~will be~~ minimized. Additionally, with the MMI test unit provided with the Eagle 21 system, the
1 amount of technician and engineering time required for maintenance and troubleshooting is ~~will be~~ minimized. Thus, large quantities of engineering time required for the review of the quarterly functional tests, prior to restoring the channel to an operable condition, are ~~is~~ eliminated because of the user-friendly printout provided from the MMI. In total, interface with the Eagle 21 process
1 protection system is ~~will be~~ reduced, resulting in a decreased potential for technician induced error which results in improved system reliability and availability.

In the Eagle 21 process protection system design, there are failure modes which could result in the failure of an entire protection rack. During these conditions, the rack will fail to the preferred failure mode (tripped/not tripped condition) providing maximum protection for the plant. The failure of a single rack is considered to be bounded by the loss of an entire protection set, which is the existing licensing basis. This failure has been shown not to adversely impact plant safety due to the existence of redundancy, functional diversity and defense-in-depth design measures employed in the design of the process protection system. Use of these design measures ensures that in the event of a single failure, the remaining protection system channels would be available for plant protection if required. Additional discussion of the defense-in-depth, redundancy and functional diversity design measures used in the design of the Eagle 21 process protection system can be found in References [5] and [14].

1 A failure mode and effects analysis (FMEA) of the logic portion of the reactor trip system was ~~has been~~ performed. The basis of the FMEA is that the reactor protection system is designed to sense abnormal plant conditions and to initiate action necessary to assure that acceptable fuel design limits are not exceeded for anticipated operational occurrences. Results of this study and a fault tree analysis are presented in Reference [4]. The results of the study show that the probability of protection system failure in anticipated transients is sufficiently low that no provision need be made in plant design to accommodate such hypothetical failure.

7.2.2.1 Evaluation of Design Limits

While most setpoints used in the reactor protection system are fixed, there are variable setpoints, most notably the overtemperature ΔT and overpower ΔT. All setpoints in the reactor trip system have been selected on the basis of engineering design or safety studies. The capability of the reactor trip system to prevent loss of integrity of the fuel cladding and/or reactor coolant system pressure boundary during Condition II and III transients is demonstrated in Chapter 15. These accident analyses are carried out using those setpoints determined from results of the engineering design studies. Setpoint limits are presented in the Technical Specifications. A discussion of the intent for each of the various reactor trips and the accident analyses (where appropriate) which
1 utilize this trip is presented below, in Section 7.2.1.1.2, and in Table 7.2-4. The selection of trip setpoints provides for margin before protection action is actually required to allow for uncertainties and instrument errors (Reference 13). The design meets the requirements of Criteria 10, 15, 20, and 29 of the 1971 GDC.

7.2.2.1.1 Trip Setpoint Discussion

It has been pointed out previously that below the limiting value of DNBR there is likely to be significant local fuel cladding failure. The DNBR existing at any point in the core for a given core design can be determined as a function of the core inlet temperature, power output, reactor coolant operating pressure and flow. Consequently, core safety limits in terms of the limiting value of DNBR for the hot channel can be developed as a function of core ΔT, T_{avg}, and pressure for a specified flow as illustrated by the dashed lines in Figure 15.1-1. Shown as a dashed line in Figure 15.1-1 are the loci of conditions designed to prevent exceeding 121% of power as a function of ΔT and T_{avg}, thus, representing the overpower (KW/ft) limit on the fuel. The solid lines indicate the maximum permissible setpoints (ΔT) as a function of T_{avg} and pressure for the overtemperature and overpower reactor trips. Actual setpoint constants in the equation representing the solid lines are as given in the Technical Specifications. These values are conservative to allow for instrument errors. The design meets the requirements of Criteria 10, 15, 20 and 29 of the 1971 GDC.

DNBR is not a directly measurable quantity; however, the process variables that determine DNBR are sensed and evaluated. Small isolated changes in various process variables may not individually result in violation of a core safety limit, whereas the combined variations, over sufficient time, may cause the overpower or overtemperature safety limit to be exceeded. The design concept of the reactor trip system takes cognizance of this situation by providing reactor trips associated with individual process variables in addition to the overpower/overtemperature safety limit trips. Process variable trips prevent reactor operation whenever a change in the monitored value is such that a core or system safety limit is in danger of being exceeded should operation continue. Basically, the high pressure, low pressure and overpower/overtemperature ΔT trips provide sufficient protection for slow transients as opposed to such trips as low flow or high flux which will trip the reactor for rapid changes in flow or flux, respectively, that would result in fuel damage before actuation of the slower responding ΔT trips could be effected.

Therefore, the reactor trip system has been designed to provide protection for fuel cladding and reactor coolant system pressure boundary integrity where: 1) a rapid change in a single variable or factor which will quickly result in exceeding a core or a system safety limit, and 2) a slow change in one or more variables will have an integrated effect which will cause safety limits to be exceeded. Overall, the reactor trip system offers diverse and comprehensive protection against fuel cladding failure and/or loss of reactor coolant system integrity for Condition II and III accidents. This is demonstrated by Table 7.2-4 which lists the various trips of the reactor trip system, the corresponding technical specification on safety limits and safety system settings and the appropriate accident discussed in the safety analyses in which the trip could be utilized.

The plant is prohibited by Technical Specifications from operating with an inactive loop for extended periods of time, and administrative procedures require that the unit be brought to a load of less than 25% of full power prior to starting the pump in the inactive loop in order to bring the inactive loop hot leg temperature closer to the core inlet temperature.

21 ~~However, it should be noted that the reactor trip system automatically provides core protection during this non-standard operating configuration, i.e., no protection system setpoints need to be reset. This is because the nominal value of the power (P-8) interlock setpoint restricts the power levels such that DNB ratios below the design basis limit will not be realized during any Condition II transients occurring during this mode of operation. This restricted power level is considerably below the boundary of permissible values for core safety limits for operation with a loop out of service. Thus t~~The P-8 interlock acts essentially as a high nuclear power reactor trip when operating in this condition ~~with one loop not in service~~.

The reactor trip system design was evaluated in detail with respect to common mode failure and is
1 presented in References [4] and [5]. The design meets the requirements of Criterion 23 ~~21~~ of the 1971 GDC.

1 Preoperational testing was ~~is~~ performed on reactor trip system components and systems to determine equipment readiness for startup and served ~~. This testing serves~~ as confirmation of the system design.

Analyses of the results of Condition II, III and IV events, including considerations of instrumentation installed to mitigate their consequences, are presented in Chapter 15. The instrumentation installed to mitigate the consequences of load rejection and turbine trip is given in
1 Section 7.7 ~~7.4~~.

7.2.2.1.2 Reactor Coolant Flow Measurement

22 ~~The elbow taps used on each loop in the primary coolant system are instrument devices that are used to indicate reactor coolant flow. The basic function of this measurement is to ensure that thermal design flow is achieved. The correlation between flow and elbow tap signal is given by the following equation:~~

INSERT

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$$\cancel{\frac{\Delta P}{\Delta P_0} = \left(\frac{w}{w_0}\right)^2,}$$

~~where ΔP_0 is the pressure differential at the reference flow w_0, and ΔP is the pressure differential at the corresponding flow, w. The full flow reference point is established during initial plant startup. The low flow trip point is then established by extrapolating along the correlation curve. The expected absolute accuracy of the channel is within ±10% of full flow and field results have shown the repeatability of the trip point to be within ±1%.~~

INSERT

7.2.2.2 Evaluation of Compliance to Applicable Codes and Standards

1 The reactor trip system meets the requirements of the General Design Criteria and ~~as indicated. The reactor trip system meets the requirements of~~ Section 4 of IEEE Standard 279-1971[9] as indicated below.

1. General Functional Requirement

   The protection system automatically initiates appropriate protective action whenever a condition monitored by the system reaches a preset value. Functional performance requirements are given in Section 7.2.1.1.1. Section 7.2.1.2.4 presents a discussion of limits, margins and setpoints; Section 7.2.1.2.5 discusses unusual (abnormal) events; and Section 7.2.1.2.6 presents minimum performance requirements.

2. Single Failure Criterion

   The protection system is designed to provide two, three, or four redundant process protection channels for each protective function and two logic train circuits. These redundant channels and trains are electrically isolated and physically separated. Thus, any single failure within a channel or train will not prevent protective system action when required. Loss of input power, the most likely mode of failure, to a channel or logic train will result in a signal calling for a trip. This design meets the requirements of Criteria 21, 22 and 23 of the 1971 GDC.

   To prevent the occurrence of common mode failures, such additional measures as functional diversity, physical separation, and testing as well as administrative control during design, production, installation and operation are employed, as discussed in
   1 references [4] and [5]. The design meets the requirements of Criteria 21 and 22 of the 1971 GDC.

3. Quality of Components and Modules

   For a discussion on the quality of the components and modules used in the reactor trip system, refer to Chapter 17. The quality assurance applied conforms to Criterion 1 of the 1971 GDC.

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INSERT for 7.2.2.1.2

Elbow taps installed in each loop of the primary coolant system are used to measure reactor coolant flow. The correlation between flow and elbow tap differential pressure signal is given by the following equation:

$$\frac{\Delta P}{\Delta P_0} = \left(\frac{w}{w_0}\right)^2,$$

where ΔP_0 is the pressure differential at the reference flow w_0, and ΔP is the pressure differential at the corresponding flow, w. Nominal full power flow is established at the beginning of each fuel cycle by performance of the RCS calorimetric flow measurement, the results of which are used to normalize the RCS flow indicators. This provides a reference point for the low flow reactor trip setpoint, and also provides a relatively simple method for periodic verification of the thermal design flow assumed in the safety analysis, as required by the Technical Specifications. Accuracy and repeatability of the flow measurement instrumentation are considered in establishment of the low flow setpoint and the minimum required flow and are adequate for these functions.

4. Equipment Qualification

For a discussion of the type tests made to verify the performance requirements, refer to Section 3.11. The test results demonstrate that the design meets the requirements of Criterion 4 of the 1971 GDC.

5. Channel Integrity

Protection system channels required to operate in accident conditions maintain necessary functional capability under extremes of conditions relating to environment, energy supply, malfunctions, and accidents. The energy supply for the reactor trip system is described in Chapter 8. The environmental variations, throughout which the system will perform is given in Section 3.11. The design meets the requirements of Criteria 21 and 22 of the 1971 GDC.

6. Independence

Channel independence is carried throughout the system, extending from the sensor through the devices actuating the protective function. Physical separation is used to achieve separation of redundant transmitters.

Separation of wiring is achieved using separate wireways, cable trays, conduit runs and containment penetrations for each redundant channel. Redundant protection channels are separated by locating the processing electronics of the redundant channels in different protection rack sets. Each redundant protection channel set is energized from a separate AC power feed. This design meets the requirements of Criteria 21 and 22 of the 1971 GDC.

Independence of the logic trains is discussed in Reference[3]. Two reactor trip breakers are actuated by two separate logic matrices which interrupt power to the control rod drive mechanisms. The breaker main contacts are connected in series with the power supply so that opening either breaker interrupts power to all control rod drive mechanisms, permitting the rods to free fall into the core. See Figure 7.1-1.

The design philosophy is to make maximum use of a wide variety of measurements. The protection system continuously monitors numerous diverse system variables. The extent of this diversity has been evaluated for a wide variety of postulated accidents and is discussed in Reference [5]. Generally, two or more diverse protection functions would terminate an accident before intolerable consequences could occur. This design meets the requirements of Criterion 22 of the 1971 GDC.

7. Control and Protection System Interaction

The protection system is designed to be independent of the control system. In certain applications the control signals and other non-protective functions are derived from individual protective channels through isolation devices. The isolation devices are classified as part of the protection system and are located in the process protection racks. Non-protective functions include those signals used for control, remote process indication, and computer monitoring. The isolation devices are designed such that a short circuit, open circuit, or the application of credible fault voltages on the isolated output portion of the circuit (i.e., the non-protective side of the circuit) will not affect the input (protective) side of the circuit. The signals obtained through the isolation devices are never returned to the process protection racks. ~~This design meets the requirements of Criterion 24 of the 1971 GDC and paragraph 4.7 of IEEE Standard 279-1971[9].~~

23

~~Specific control and protection system interactions are discussed in Section 7.2.2.3.~~

A detailed discussion of the design and testing of the protection system isolation devices is given in References [7], [8], and [11]. These reports include the results of applying various malfunction conditions on the output portion of the isolation devices. The results show that no significant disturbance to the isolation devices' input signal occurred.
← **INSERT**

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~~Where failure of a protection system component can cause a process excursion which requires protective action, the protection system can withstand another, independent failure without loss of protective action. This is normally achieved by means of two-out-of-four (2/4) trip logic for each of the protective functions except Steam Generator Protection. The Steam Generator Low Water Level protective function relies upon two-out-of-three (2/3) trip logic and a control system Median Signal Selector (MSS). The use of a control system MSS prevents any protection system failure from causing a control system reaction resulting in a need for subsequent protective action. This meets the requirements of Criterion 24 of the 1971 GDC. A detailed discussion of the function of the MSS relative to control and protection system interaction is contained in References [12] and [14].~~

8. Derivation of System Inputs

To the extent feasible and practical, protection system inputs are derived from signals which are direct measures of the desired variables. Variables monitored for the various reactor trips are listed in Section 7.2.1.2.2.

9. Capability for Sensor Checks

The operational availability of each system input sensor during reactor operation is accomplished by cross checking between channels that bear a known relationship to each other and that have read-outs available. Channel checks are discussed in the Technical Specifications.

23 **INSERT for 7.2.2.2(7)**

Where failure of a protection system component can cause a process excursion which requires protective action and can also prevent the channel from performing its protective action, the protection system can withstand a second independent failure without loss of the protection function. The means of achieving this are provided in the discussions of specific control and protection system interactions in Section 7.2.2.3. Typically this requirement is satisfied by utilizing 2/4 logic for the trip function or by providing a diverse trip. This design meets the requirements of Criterion 24 of the 1971 GDC and paragraph 4.7 of IEEE Standard 279-1971[9].

10. Capability for Testing

The reactor trip system is capable of being tested during power operation. Where only parts of the system are tested at any one time, the testing sequence provides the necessary overlap between the parts to assure complete system operation. The testing capabilities are in conformance with Regulatory Guide 1.22 as discussed in Table 7.1-1.

The protection system is designed to permit periodic testing of the signal processing portion of the reactor trip system during reactor power operation without initiating a protective action unless a trip condition actually exists. This is because of the coincidence logic required for reactor trip. Source and intermediate range high neutron flux trips must be bypassed during testing. These tests may be performed at any plant power from cold shutdown to full power. Before starting any of these tests with the plant at power, all redundant reactor trip channels associated with the function to be tested must be in the normal (untripped) mode or bypass mode, according to the Technical Specifications, in
24 order to avoid spurious trips. ~~Setpoints are documented in Reference [13] and incorporated into the Technical Specifications.~~

The Protection System is also designed to permit periodic response time testing of the reactor trip system, excluding neutron detectors.

Process Protection Channel Tests

The Eagle 21 process protection system accommodates automatic or manual surveillance testing of the digital process protection racks via a portable Man Machine Interface (MMI) test cart. The
1 MMI test cart is connected to the process rack test panel with ~~by inserting~~ a cable/connector assembly ~~into the process rack test panel~~. The rack installed test processor permits performance of operations such as channel calibration, channel response time tests, partial trip actuation tests, and maintenance activities. Administrative controls and multiple levels of security are provided to limit access to setpoint and tuning constant adjustments. The system is designed to permit testing of any protection channel during power operations without initiating a protective action at the systems level.

Individual channels can be tested in either the "Channel Trip" or "Bypass" mode:

The Channel Trip mode interrupts the individual channel comparator output. Interruption of a comparator output in this mode for any reason (test, maintenance purposes or removed from service) causes that portion of the logic to be actuated and initiates a channel trip alarm and status light in the control room. Status lights on the process rack test panel indicate when the associated comparators have tripped.

The Bypass mode disables the individual channel comparator trip circuitry. Interruption of a comparator output in this mode effectively "bypasses" the channel in test causing the associated logic relays to remain in the non-tripped state until the "bypass" is removed. This feature of the protection system eliminates the potential for an unwarranted actuation in the event of a failure. This condition is also accompanied by an alarm in the control room.

Nuclear Instrumentation Channel Tests

The power range channels of the nuclear instrumentation system are tested by using the actual detector input to the channel and injecting test currents obtained from the detector response curves at various power levels. The output of the bistable is not placed in a tripped condition prior to testing. Also, since the power range channel logic is two out of four, bypass of this reactor trip function is not required.

Testing of a power range channel requires deliberate operator action and is annunciated in the control room. Bistable operation is tested by increasing the test signal up to its trip setpoint and verifying bistable relay operation by control board annunciator and trip status lights.

It should be noted that a valid trip signal would cause the channel under test to trip at a lower actual reactor power. A reactor trip would occur when a second bistable trips. No provision has been made in the channel test circuit for reducing the channel signal below that signal being received from the nuclear instrumentation system detector.

A nuclear instrumentation system channel which can cause a reactor trip through one of two protection logic (source or intermediate range) is provided with a bypass function which prevents the initiation of a reactor trip from that particular channel during the short period that it is undergoing test. These bypasses are annunciated in the control room.

~~The following p~~Periodic tests of the source, intermediate, and power range channels of the nuclear instrumentation system are performed~~:~~ in the applicable modes/power levels in accordance with the Technical Specifications.

25 ~~a. Testing at plant shutdown~~
~~1) Source range testing~~
~~2) Intermediate range testing~~
~~3) Power range testing~~

~~b. Testing between P-6 and P-10 permissive power levels~~
~~1) Intermediate range testing~~
~~2) Power range testing~~

~~c. Testing above P-10 permissive power level~~
~~1) Power range testing~~

For a detailed description of the nuclear instrumentation system see References [2] and [15]. Reference [2] is applicable to the power range only.

Solid State Logic Testing

The logic trains of the reactor trip system are designed to be capable of complete testing at power. Logic matrices are tested from the Train A and Train B logic rack test panels. During this test, the logic inputs are actuated automatically in all combinations of trip and non-trip logic. Trip logic is not maintained sufficiently long enough to permit opening of the reactor trip breakers. The reactor trip undervoltage coils are 'pulsed' in order to check continuity. During logic testing of one train, the other train can initiate any required protective functions. Annunciation is provided in the control room to indicate when a train is in test (train output bypassed) and when a reactor trip breaker is bypassed. Details of the logic system testing are given in Reference [3].

A direct reactor trip resulting from undervoltage or underfrequency on the pump side of the
1 reactor coolant pump breakers is provided as discussed in Section 7.2.1.1.2 and shown on Figure 7.2-1, Sheet 3. The logic for these trips is capable of being tested during power operation. When parts of the trip are being tested, the sequence is such that an overlap is provided between parts so that a complete logic test is provided.

This design complies with the testing requirements of IEEE Standard 279-1971 and IEEE Standard 338-1971[10] as discussed in Table 7.1-1. Details of the method of testing and compliance with these standards are provided in References [l], [3], and [11].

The permissive and block interlocks associated with the reactor trip system and engineered safety features actuation system are given on Tables 7.2-2 and 7.3-3 and designated protection or 'P' interlocks. As a part of the protection system, these interlocks are designed to meet the testing requirements of IEEE Standards 279-1971 and 338-1971 as discussed in Table 7.1-1.

Testability of the interlocks associated with reactor trips for which credit is taken in the accident analyses is provided by the logic testing and semi-automatic testing capabilities of the solid state protection system. In the solid state protection system the undervoltage coils (reactor trip) and master relays (engineered safeguards actuation) are pulsed for all combinations of trip or actuation logic with and without the interlock signals. Interlock testing may be performed at power.

Testing of the logic trains of the reactor trip system includes a check of the input relays and a logic matrix check. The following sequence is used to test the system:

1) Check of input relays

During testing of the process protection system and nuclear instrumentation system channels, each channel comparator/bistable is placed in a trip mode causing one SSPS input relay in train A and one in train B to de-energize except when individual channels are tested in bypass with the reactor at power. A contact of each relay is connected to a universal logic printed circuit card. This card performs both the reactor trip and monitoring functions. Each reactor trip input relay contact causes a status lamp and an annunciator on the control board to operate. Either the Train A or Train B input relay operation will light the status lamp and annunciator.

Each train contains a multiplexing test switch, one of which (either train) normally remains in the A + B position. The A + B position allows information to be transmitted alternately from each train to the control board. During testing a steady status lamp indicates that both trains are receiving a trip mode logic input for the channel being tested. A flashing lamp indicates a failure in one train. Contact inputs to the logic protection system such as reactor coolant pump bus underfrequency relays operate input relays which are tested by operating the remote contacts as described above and using the same type of indications as those provided for comparator/bistable input relays.

Actuation of the SSPS input relays provides the overlap between the testing of the logic protection system and the testing of those systems supplying the inputs to the logic
1 protection system. These tests are performed periodically in accordance with the Technical Specifications. Test indications are status lamps and annunciators on the control board. Inputs to the logic protection system are checked one channel at a time, leaving the other channels in service. For example, a function that trips the reactor when two out of four channels trip becomes a one out of three trip when one channel is placed in the trip mode. Both trains of the logic protection system remain in service during this portion of the test.

2) Check of logic matrices

Logic matrices are checked one train at a time. Input relays are not operated during this test. Partial reactor trips to the train being tested are inhibited with the use of the input error inhibit switch on the semi-automatic test panel in the train. Details of semi-automatic tester operation are given in Reference [3]. At the completion of the logic matrix tests, closure of the input error inhibit switch contacts is checked using an
26 appropriate test method ~~(~~such as verification of existing trip status lamps/computer points~~, or trip of one comparator/bistable for the appropriate protection system channel)~~.

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The logic test scheme uses pulse techniques to check the coincidence logic. All possible trip and non-trip combinations are checked. Pulses from the tester are applied to the inputs of the universal logic card at the same terminals that connect to the input relay contacts. Thus there is an overlap between the input relay check and the logic matrix check. Pulses are fed back from the reactor trip breaker undervoltage coil to the tester. The pulses are of such short duration that the reactor trip breaker undervoltage coil armature cannot respond mechanically.

1 Test indications ~~that are~~ provided are an annunciator in the control room indicating that reactor trips from the train have been blocked and that the train is being tested, and green and red lamps on the semi-automatic tester to indicate a good or bad logic matrix test. Protection capability provided during this portion of the test is from the train not being tested.

The general design features and details of the testability of the logic system are described in Reference [3]. The testing capability meets the requirements of Criterion 21 of the 1971 GDC.

<u>Testing of Reactor Trip Breakers</u>

27 Normally, reactor trip breakers ~~52/~~RTA and ~~52/~~RTB are in service, and bypass breakers ~~52/~~BYA and ~~52/~~BYB are withdrawn (out of service). In testing the protection logic, pulse techniques are used to avoid tripping the reactor trip breakers thereby eliminating the need to bypass them during this testing. Each of the reactor trip breakers is tested with the corresponding bypass breaker in service.

Auxiliary contacts of the bypass breakers are connected into the SSPS General Warning Alarm System of their respective trains such that if either train is placed in test while the bypass breaker of the other train is closed, both reactor trip breakers and both bypass breakers will automatically trip.

Auxiliary contacts of the bypass breakers are also connected in such a way that if an attempt is made to close the bypass breaker in one train while the bypass breaker of the other train is already closed, both bypass breakers and both reactor trip breakers will automatically trip.

The Train A and Train B alarm systems operate separate annunciators in the control room. The two bypass breakers also operate an annunciator in the control room. Bypassing of a protection train with either the bypass breaker or with the test switches will result in audible and visual indications.

The complete reactor trip system is normally required to be in service. However, to permit online testing of the various protection channels or to permit continued operation in the event of a subsystem instrumentation channel failure, the Technical Specifications define the minimum number of operable channels. The Technical Specifications also define the required restriction to operation in the event that the channel operability requirements cannot be met.

11. Channel Bypass or Removal from Operation

28

The Eagle 21 Process Protection System is designed to permit any channel to be maintained in a bypassed condition and, when required, tested during power operation without initiating a protective action at the systems level. This is accomplished without lifting electrical leads or installing temporary jumpers. ~~If a~~ Bypass of any channel in an Eagle 21 protection system rack ~~has been bypassed~~ for any purpose~~, a signal (1 per protection set) is provided to allow this condition to~~ will be continuously indicated in the control room via the plant annunciator at the protection set level. In addition, the Eagle 21 design has provided for administrative controls and multiple levels of security for bypassing a protection channel.

The channel bypass feature of the Eagle 21 system will be used for the following purposes:

1. To allow for an inoperable Reactor Trip (RT) or Engineered Safety Features Actuation System (ESFAS) channel to be maintained in a bypassed condition up to six hours for the purpose of troubleshooting.

2. To allow for a failed RT or ESFAS channel to be bypassed up to four hours for the purpose of surveillance testing a redundant channel of the same function.

3. To routinely allow testing of a RT or ESFAS channel in the bypassed condition instead of the tripped condition for the purpose of surveillance testing.

The Nuclear Instrumentation System (NIS) is designed to permit routine periodic testing of the Source Range and Intermediate Range portion of the reactor trip system during reactor power operation. To enable testing of the one-out-of-two channel logic for the NIS Source Range and Intermediate Range during reactor power operation, a channel bypass feature has been provided. Use of this feature will permit routine required surveillance testing to be completed without initiating a protective action unless a trip condition exists.

12. Operating Bypasses

Where operating requirements necessitate automatic or manual bypass of a protective function, the design of the protection system is such that the bypass is removed automatically whenever permissive conditions are not met. Devices used to achieve automatic removal of the bypass of a protective function are considered part of the protection system and are designed in accordance with the criteria of this section. Indication is provided in the control room if some part of the system has been administratively bypassed or taken out of service. Bypasses associated with the reactor trip system are identified in Table 7.2-2.

13. Indication of Bypasses

Bypass of a process protection channel during testing is indicated by an alarm in the control room. This is discussed further in Section 7.2.2.2, subsections 10 and 11.
1 Operating bypasses are discussed in Section 7.2.2.2, subsection 12.

14. Access to Means for Bypassing

The design provides for administrative control of access to the means for manually bypassing channels or protective functions. For details, refer to References [1] and [11].

15. Multiple Setpoints

For monitoring neutron flux, multiple setpoints are used. When a more restrictive trip setting becomes necessary to provide adequate protection for a particular mode of operation or set of operating conditions, the protective system circuits are designed to provide positive means or administrative control to assure that the more restrictive trip setpoint is used. The devices used to prevent improper use of less restrictive trip settings are considered part of the protective system and are designed in accordance with the criteria of this section.

16. Completion of Protective Action

The protection system is so designed that, once initiated, a protective action goes to completion. Normal operation is restored in accordance with established procedures.

17. Manual Initiation

29 Switches are provided on the control board for manual initiation of protective action. A single ~~F~~failure in the automatic system will ~~does~~ not prevent the manual actuation of the protective functions. Manual actuation relies on the operation of a minimum of
1 equipment. Additional discussion of manual actuation of protective functions is provided in Section 7.3.2.2.6.

18. Access to Setpoint Adjustments, Calibration and Test Points

The design provides for administrative control of access to all setpoint adjustments, processing electronics calibration adjustments, and test points. For details refer to References [1], [2], [11] and [15].

19. Identification of Protective Actions

Indication and identification of protective actions is discussed in Item 20 below.

20. Information Read Out

The protective system provides the operator with complete information pertinent to system status and safety. All transmitted signals (flow, pressure, temperature, etc.) which
1 can cause a reactor trip are ~~will be~~ either indicated or recorded for every channel, including all neutron flux power range currents (top detector,~~;~~ bottom detector, algebraic difference and average of bottom and top detector currents).

Any reactor trip will actuate an alarm and an annunciator. Such protective actions are indicated and identified by the parameter being measured.

Alarms and annunciators are also used to alert the operator of deviations from normal operating conditions so that he may take appropriate corrective action to avoid a reactor trip. Actuation of any rod stop or trip of any reactor trip channel will actuate an alarm,
30 except for the source and intermediate range channels which have one out of two reactor trip logic. For these two functions, a channel trip alarm is not provided since a channel trip will also initiate a reactor trip and reactor trip alarm as described above.

21. System Repair

The system is designed to facilitate the recognition, location, replacement, and repair of
1 malfunctioning components or modules. Refer to the discussion in Section 7.2.2.2, subsection ~~Item~~ 10 above.

22. Identification

Identification of protection system equipment is discussed in Section 7.1.2.3.

7.2.2.3 Specific Control and Protection Interactions

1 ← **INSERT**

7.2.2.3.1 Neutron Flux

Four power range neutron flux channels are provided for overpower (high flux) protection. An isolated auctioneered high signal is derived by auctioneering of the four channels for automatic rod control. If any channel fails in such a way as to produce a low output, that channel is incapable of proper overpower protection but will not cause control rod movement because of the auctioneer. Two out of four overpower trip logic will ensure an overpower trip if needed even with an independent failure in another channel.

1 **INSERT for 7.2.2.3**

A general discussion of control and protection system interaction criteria and compliance is provided in Section 7.2.2.2, subsection 7.

In addition, channel deviation signals in the control system will give an alarm if any neutron flux channel deviates significantly from the average of the flux signals. Also, the control system will respond only to rapid changes in indicated neutron flux; slow changes or drifts are compensated by the temperature control signals. Finally, an overpower signal from any nuclear power range channel will block manual and automatic rod withdrawal. The setpoint for this rod stop is below the reactor trip setpoint.

7.2.2.3.2 Reactor Coolant Temperature

15 ~~The accuracy of the narrow range resistance temperature detector loop temperature measurements is demonstrated during plant startup tests and periodically with surveillance tests. Testing compares temperature measurements from the narrow range resistance temperature detectors with one another as well as with the temperature measurements obtained from the wide range resistance temperature detectors located in the hot leg and cold leg piping of each loop. The comparisons are done with the reactor coolant system in an isothermal condition. During plant startup tests, ΔT measurements obtained from the hot leg and cold leg narrow range loop resistance temperature detectors are compared to plant power, and if required normalized to plant power. The absolute value of ΔT versus plant power is not important, per se, as far as reactor protection is concerned. Reactor trip system setpoints are based upon percentages of the indicated ΔT at nominal full power rather than on absolute values of ΔT. This is done to account for loop differences which are inherent. Therefore the percent ΔT scheme is relative, not absolute, and thus provides better protective action without the expense of accuracy. As part of the plant startup tests, the narrow range resistance temperature detector signals will be compared with the core exit thermocouple signals.~~

Revised/Moved to 7.2.1.1.4

Reactor control is based upon signals derived from protection system channels after isolation by isolation devices such that no feedback effect from the control system can perturb the protection channels.

Since control is based on the highest of the loop average temperatures, the control rods are always moved based upon the most pessimistic temperature measurement with respect to margins to DNB. A spurious low average temperature measurement from any loop temperature control channel will cause no control action. A spurious high average temperature measurement will cause rod insertion (safe direction). The 2/4 trip logic ensures that the overpower and
31 overtemperature ΔT trip functions can provide the required protection even if degraded by a second random failure.

Channel deviation signals in the control system will give an alarm if any temperature channel deviates significantly from the auctioneered (highest) value. Automatic rod withdrawal blocks
32 and turbine runback (power demand reduction) will also occur prior to reaching the reactor trip setpoint if any two of the ΔT channels indicate an overtemperature or overpower condition.

1 A discussion of reactor coolant temperature measurement is provided in Section 7.2.1.1.4.

31 ~~Section 4.7 of IEEE 279-1971 and GDC 24 requirements concerning control and protection systems interaction are satisfied, even though control signals are derived from protection sets, because the 2/4 voting coincidence logic of the protection system is maintained. Where a single random failure can cause a control system action that results in a condition requiring protective action and can also prevent proper action of a protection system channel designed to protect against the condition, the remaining three redundant protection channels are capable of providing the required protective action even if degraded by a second random failure.~~

7.2.2.3.3 Pressurizer Pressure

The pressurizer pressure protection channel signals are used for high and low pressure protection and as inputs to the overtemperature ΔT trip protection function. Isolated output signals from these channels are used for pressure control. These are used to control pressurizer spray and heaters and power operated relief valves. A coincident high pressure signal from two independent channels is needed for the actuation of each pressurizer PORV.

33 A spurious high pressure signal from one channel can cause decreasing pressure by turning off the heaters and actuation of spray. Additional redundancy is provided by ~~in~~ the 2/4 low pressurizer pressure reactor trip logic ~~to~~ ensures low pressure protection in the event of a second independent failure.

Overpressure protection is based upon the positive surge of the reactor coolant produced as a result of turbine trip under full load, assuming the core continues to produce full power. The self-actuated safety valves are sized on the basis of steam flow from the pressurizer to accommodate this surge at a setpoint of 2500 psia and an accumulation of 3%. Note that no credit is taken for the relief capability provided by the power-operated relief valves during this surge.

In addition, operation of any one of the power-operated relief valves can maintain pressure below the high pressure trip point for most transients. The rate of pressure rise achievable with heaters is slow, and ample time and pressure alarms are available to alert the operator of the need for appropriate action.

7.2.2.3.4 Pressurizer Water Level

~~Three pressurizer water level channels are used for reactor trip. Isolated signals from these channels are used for pressurizer water level control. A failure in the level control system could fill or empty the pressurizer at a slow rate.~~

16 ~~Experience has shown that hydrogen gas can accumulate in the upper part of the condensate pot on conventional open reference leg systems in pressurizer water level service. At RCS operating pressures, high concentrations of dissolved hydrogen in the reference leg water are possible. On sudden depressurization accidents, it has been hypothesized that rapid effervescence of the dissolved hydrogen could blow water out of the reference leg and cause a large level error, measuring higher than actual level. To eliminate the possibility of such effects, a bellows is used in a pot at the top of the reference leg to provide an interface seal and prevent dissolving of hydrogen gas into the reference leg water.~~

INSERT (Portions Revised/Moved to 7.2.1.1.5)

~~The reference leg is uninsulated and will remain at local ambient temperature. This temperature will vary somewhat over the length of the reference leg piping under normal operating conditions but will not exceed 140°F. During a blowdown accident, any reference leg water flashing to steam will be confined to the condensate steam interface in the condensate pot at the top of the temperature barrier leg and will have only a small (about one inch) effect on measured level. Some additional error may be expected due to effervescence of hydrogen in the temperature barrier water. However, even if complete loss of this water is assumed, the error will be small and can be tolerated.~~

~~The sealed reference leg design has been installed in various plants since early 1970 and operational accuracy was verified by use of the sealed reference leg system in parallel with an open reference leg channel. No effects of operating pressure variations on either the accuracy or integrity of the channel have been observed.~~

16 ~~Calibration of the sealed reference leg system is done in place after installation by application of known pressure to the low pressure side of the transmitter and measurement of the transmitter output. The effects of static pressure variations are predictable. The largest effect is due to the density change in the saturated fluid in the pressurizer itself. The effect is typical of level measurements in all tanks with two phase fluid and is not peculiar to the sealed reference leg technique. In the sealed reference leg, there is a slight compression of the fill water with increasing pressure, but this is taken up by the flexible bellows. A leak of the fill water in the sealed reference leg can be detected by comparison of redundant channel readings on line and by physical inspection of the reference leg off line. Leaks of the reference leg to atmosphere will be immediately detectable by off scale indications and alarms on the control board. A closed pressurizer level instrument shut off valve would be detected by comparing the level indications from the redundant level channels (three channels). In addition, there are alarms on one of the three channels to indicate an error between the measured pressurizer water level and the programmed pressurizer water level. There is no single instrument valve which could affect more than one of the three level channels.~~

~~The high water level trip setpoint provides sufficient margin such that the undesirable condition of discharging liquid coolant through the safety valves is avoided. Even at full power conditions, which would produce the worst thermal expansion rates, a failure of water level control would not lead to any liquid discharge through the safety valves. This is due to the automatic high pressurizer pressure reactor trip actuating at a pressure sufficiently below the safety valve setpoint.~~

INSERT

Portions Revised/Moved to 7.2.1.1.5

16 **INSERT for 7.2.2.3.4**

Three independent, redundant instrument channels are provided for pressurizer high water level protection. This reactor trip condition is generated based on a 2-out-of-3 logic and serves to prevent water discharge through the pressurizer safety relief valves. The pressurizer level channels also provide isolated output signals which are used for pressurizer water level control (reference Section 7.7). A failure in a level control channel could increase or decrease pressurizer level at a slow rate.

The high water level trip setpoint provides sufficient margin such that the undesirable condition of discharging liquid coolant through the safety valves is avoided. Even at full power conditions, which would produce the worst thermal expansion rates, a failure of water level control would not lead to any liquid discharge through the safety valves. This is due to the automatic high pressurizer pressure reactor trip actuating at a pressure sufficiently below the safety valve setpoint.

In addition, alarms are actuated on high or low water level and on significant deviations from programmed level. Channel failure can also be detected by comparison to the other two redundant level channel indicators located in the main control room. A discussion of the pressurizer water level reference leg arrangement is provided in Section 7.2.1.1.5.

7.2.2.3.5 Steam Generator Water Level

The basic function of the reactor protection circuits associated with low steam generator water level is to preserve the steam generator heat sink for removal of long term residual heat. Should a complete loss of feedwater occur, the reactor would be tripped on low-low steam generator water level. In addition, redundant auxiliary feedwater pumps are provided to supply feedwater in order to maintain residual heat removal after trip. This reactor trip acts before the steam generators are dry to reduce the required capacity and increase the starting time requirements of the auxiliary feedwater pumps and to minimize the thermal transient on the reactor coolant system and steam generators.

INSERT

34 ~~Therefore, a low-low steam generator water level reactor trip is provided for each steam generator to ensure that sufficient initial thermal capacity is available in the steam generator at the start of the transient. It is desirable to minimize thermal transients on a steam generator for a credible loss of feedwater accident. To minimize perturbations on the feedwater control system, a control grade Median Signal Selector (MSS) is installed in the control system. Implementation of the MSS will prevent failure of a single steam generator water level channel from causing a feedwater control system disturbance requiring subsequent protective action. The application of the MSS in the feedwater control system will improve system reliability by providing a "median" signal for use by the control system to initiate control system actions based on this signal and eliminate the need for the low feedwater flow trip previously required to meet the intent of IEEE Std. 279 Section 4.7, Control and Protection System Interaction. Thus, because of the design of the MSS (accepting three isolated level channel inputs and providing a "median" signal to the control system), the potential for a control and protection system interaction is eliminated. A more detailed discussion of the MSS relative to compliance with control and protection system interaction criteria is contained in References [12] and [14].~~

7.2.2.4 Additional Postulated Accidents

Loss of plant instrument air or loss of component cooling water is discussed in Section 7.3.2. Load rejection and turbine trip are discussed in further detail in Section 7.7.

35 ~~The control interlocks, called rod stops, that are provided to prevent abnormal power conditions which could result from excessive control rod withdrawal are discussed in Section 7.7.1.4.1 and listed on Table 7.7-1. Excessively high power operation (which is prevented by blocking of automatic rod withdrawal), if allowed to continue, might lead to a safety limit (as given in the Technical Specifications) being reached. Before such a limit is reached, protection will be available from the reactor trip system. At the power levels of the rod block setpoints, safety limits have not been reached; therefore these rod withdrawal stops do not come under the scope of safety-related systems and are considered as control systems.~~

INSERT

34 **INSERT for 7.2.2.3.5 (2nd paragraph)**

Therefore, a low-low steam generator water level reactor trip is provided for each steam generator to ensure that sufficient initial thermal capacity is available in the steam generator at the start of the transient. It is desirable to minimize thermal transients on a steam generator for a credible loss of feedwater accident. Implementation of the control grade Median Signal Selector (MSS) feature in the feedwater control system prevents failure of a single steam generator water level channel from causing a feedwater control system disturbance requiring subsequent protective action. Isolated outputs from all three narrow range level channels are input to the MSS. The MSS selects the median signal for use by the control system and control system actions are then based on this signal. Since the high and low signals are rejected, the control system is prevented from acting on a single, failed protection system instrument channel. Since no adverse control system action can then result from a failed protection channel, the potential for a control and protection system interaction is eliminated and it is not necessary to consider a second random protection system failure as would otherwise be required by IEEE 279-1971. A more detailed discussion of the MSS relative to compliance with control and protection system interaction criteria is contained in References [12] and [14].

35 **INSERT for 7.2.2.4 (2nd paragraph)**

The control interlocks and permissives, called rod stops, are provided to inhibit automatic and/or manual rod withdrawal and initiate turbine runback. The rod stops indicate certain abnormal reactor operating conditions exist. The rod stop control action is used to stop positive reactivity additions due to rod withdrawal and to prevent reactor system parameters from reaching a condition requiring protective action (i.e., reactor trip actuation). The rod stops are not considered a protective feature. A listing of the initiating input signal and control function of each rod stop is provided in Section 7.7.1.4.1 and Table 7.7-1.

7.2.3 Tests and Inspections

The reactor trip system meets the testing requirements of IEEE Standard 338-1971, Reference [10], as discussed in Section 7.1.2. The testability of the system is discussed in Section 7.2.2.2. The test intervals are specified in the Technical Specifications. Written test procedures and documentation, conforming to the requirements of Reference [10], are utilized in the performance of periodic tests. Periodic testing complies with Regulatory Guide 1.22 as discussed in Sections
1 7.1.2 ~~and 7.2.2.1.3~~.

To ensure the Median Signal Selector (MSS) functions as described in Section 7.2.2.3.5, operability of the MSS is verified commensurate with the Technical Specification surveillance interval for the associated narrow range steam generator level channels.

Signal selector testing consists of monitoring the three input signals and the one output signal. Comparison of the output signal to the input signals permits determination of whether or not the median signal is being passed and, consequently, whether the signal selector is functioning properly. Any output signal at a value other than that corresponding to the median signal is indicative of a unit failure.

The signal selector is tested concurrently with the process protection channels which provide inputs to the unit. Test signals are received from the protection system, as would normal process signals, when the individual protection channels are placed in the test mode. As the test signal magnitude is varied, that protection channel which represents the median signal will also be altered allowing the technician to determine the presence of proper signal selector action.

REFERENCES

1. J. A. Nay, "Process Instrumentation for Westinghouse Nuclear Steam Supply Systems,"
1 WCAP 7671, ~~May~~ April 1971.

2. Lipchak, J. B., "Nuclear Instrumentation System," WCAP-8255, January 1974. Applicable to Power Range NIS only.

3. Katz, D. N., "Solid State Logic Protection System Description," WCAP-7488-L, January 1971 (Proprietary) and WCAP-7672, June 1971 (Non-Proprietary).

4. Gangloff, W. C. and Loftus, W. D., "An Evaluation of Solid State Logic Reactor Protection In Anticipated Transients," WCAP-7706-L, July 1971 (Proprietary) and WCAP-7706, July 1971 (Non-Proprietary).

5. Burnett, T. W. T., "Reactor Protection System Diversity in Westinghouse Pressurized Water Reactors," WCAP-7306, April 1969.

6. Baldwin, M. S. et al., "An Evaluation of Loss of Flow Accidents Caused by Power System Frequency Transients in Westinghouse PWR's," WCAP-8424, Revision 1, May 1975.

7. Doyle, J. P., "Noise, Fault, Surge and Radio Frequency Interference Test Report for Westinghouse Eagle 21 Process Protection Upgrade System," WCAP-11733 June 1988 (Westinghouse Proprietary Class 2); WCAP-11896 ~~June~~ July 1988 (Westinghouse Proprietary Class 3).

1

8. Lipchak, J. B. and Bartholomew, R. R., "Test Report Nuclear Instrumentation System Isolation Amplifier," WCAP-7506-P-A, April 1975 (Proprietary) and WCAP-7819-Revision 1-A, April 1975 (Non-Proprietary).

9. The Institute of Electrical and Electronic Engineers, Inc., "IEEE Standard: Criteria for Protection Systems for Nuclear Power Generating Stations," IEEE Standard 279-1971.

10. The Institute of Electrical and Electronic Engineers, Inc., "IEEE Trial Use Criteria for the Periodic Testing of Nuclear Power Generating Station Protection Systems," IEEE Standard 338-1971.

11. Erin, L. E., "Topical Report, Eagle 21 Microprocessor-Based Process Protection System," WCAP-12374 Rev. 1 December 1991 (Westinghouse Proprietary Class 2); WCAP-12375 Rev. 1 December 1991 (Westinghouse Proprietary Class 3).

12. Mermigos, J. F., "Median Signal Selector for Foxboro Series Process Instrumentation Application to Deletion of Low Feedwater Flow Reactor Trip," WCAP-12417 October 1989 (Westinghouse Proprietary Class 2); WCAP-12418 October 1989 (Westinghouse Proprietary Class 3).

13. Reagan, J. R., "Westinghouse Setpoint Methodology for Protection Systems, Watts Bar Units 1 and 2, Eagle 21 Version," WCAP-12096 Rev. 7 (Westinghouse Proprietary Class 2).

14. "Summary Report Process Protection System Eagle 21 Upgrade, NSLB, MSS, and TTD Implementation, Watts Bar Unit 1 and 2", WCAP-13462, ~~June 1993~~ Revision 2, September 1994.

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15. System Description, "Neutron Monitoring System" N3-92-4003.

16. ISA-DS-67.04, 1982, "Setpoints for Nuclear Safety-Related Instrumentation Used in Nuclear Power Plants."

TABLE 7.2-1
(Sheet 1 of 2)

LIST OF REACTOR TRIPS

		Reactor Trip	Coincidence Logic	Interlocks	Comments
	1.	High neutron flux (Power Range)	2/4	Manual block of low setting permitted by P-10	High and low settings; manual block and automatic reset of low setting by P-10
	2.	Intermediate range neutron flux	1/2	Manual block permitted by P-10	Manual block and automatic reset
	3.	Source range neutron flux	1/2	Manual block permitted by P-6, interlocked with P-10	Manual block and automatic reset. Automatic block above P-10
	4.	Power range high positive neutron flux rate	2/4	No interlocks	
	5.	Power range high negative neutron flux rate	2/4	No interlocks	
1	6.	Overtemperature ΔT	2/4	No interlocks	
	7.	Overpower ΔT	2/4	No interlocks	
	8.	Pressurizer low pressure	2/4	Interlocked with P-7	Blocked below P-7
	9.	Pressurizer high pressure	2/4	No interlocks	
	10.	Pressurizer high water level	2/3	Interlocked with P-7	Blocked below P-7
	11.	Low reactor coolant flow	2/3 in any loop	Interlocked with P-7 and P-8	Low flow in one loop will cause a reactor trip when above P-8 and a low flow in two loops will cause a reactor trip when above P-7. Blocked below P-7
36	12.	Reactor coolant pump bus undervoltage	2/4	Interlocked with P-7	~~Low voltage on all pumps permitted~~ Blocked below P-7.

TABLE 7.2-1
(Sheet 2 of 2)

LIST OF REACTOR TRIPS (Cont'd)

	Reactor Trip	Coincidence Logic	Interlocks	Comments
13.	Reactor coolant pump bus underfrequency	2/4	Interlocked with P-7	Underfrequency on 2 pumps will trip all reactor coolant pump breakers and cause reactor trip; reactor trip and pump trip blocked below P-7
14.	Low-low steam generator water level	2/3 in any loop	No interlocks	Features Trip Time Delay (TTD) upgrade
15.	Turbine-generator trip*			
	a) Low auto stop oil pressure	2/3	Interlocked with P-9	Blocked below P-9
	b) Turbine stop valve close	4/4	Interlocked with P-9	Blocked below P-9
16.	Safety injection signal	Coincident with actuation of safety injection	No interlocks	(See Section 7.3 for Engineered Safety Features actuation conditions)
17.	Manual	1/2	No interlocks	

* Reactor trip on turbine trip is anticipatory in that no credit is taken for it in accident analyses.

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TABLE 7.2-2
(Sheet 1 of 2)

PROTECTION SYSTEM INTERLOCKS

Designation	Derivation	Function
	I POWER ESCALATION PERMISSIVES	
P-6	Presence of P-6: 1/2 neutron flux (intermediate range) above setpoint	Allows manual block of source range reactor trip
	Absence of P-6: 2/2 neutron flux (intermediate range) below setpoint	Defeats the block of source range reactor trip
P-10	Presence of P-10: 2/4 neutron flux (power range) above set-point	Allows manual block of power range (low setpoint) reactor trip
		Allows manual block of intermediate range reactor trip and intermediate range rod stops (C-1)
		Blocks source range reactor trip (back-up for P-6)
37		Input to P-7
	Absence of P-10: 3/4 neutron flux (power range) below setpoint	Defeats the block of power range (low setpoint) reactor trip
		Defeats the block of intermediate range reactor trip and intermediate range rod stops (C-1)
37		Input to P-7

TABLE 7.2-2
(Sheet 2 of 2)

PROTECTION SYSTEM INTERLOCKS

Designation	Derivation	Function
	II BLOCKS OF REACTOR TRIPS	
P-7	Absence of P-7: 3/4 neutron flux (power range) below set-point (from P-10) and 2/2 turbine impulse chamber pressure below setpoint (from P-13)	Blocks reactor trip on: Low reactor coolant flow in more than one loop, undervoltage, under-frequency, pressurizer low pressure, and pressurizer high level
P-8	Absence of P-8: 3/4 neutron flux (power range) below set-point	Blocks reactor trip on low reactor coolant flow from one loop only
P-9	Absence of P-9: 3/4 neutron flux (power range) below setpoint	Blocks reactor trip on turbine trip
	Presence of P-9	Defeats block of reactor trip on turbine trip
P-13	Absence of P-13: 2/2 turbine impulse chamber pressure below setpoint	Input to P-7

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TABLE 7.2-3
(Sheet 1 of 2)

REACTOR TRIP SYSTEM INSTRUMENTATION

	Reactor Trip Signal	Typical Range
1.	Power range high neutron flux	1 to 120% power
2.	Intermediate range high neutron flux	10 decades of neutron flux overlapping source range by 2 decades and including 100% power
3.	Source range high neutron flux	6 decades of neutron flux (1 to 10^6 counts/sec)
4.	Power range high positive neutron flux rate	+2 to +30% of full power
5.	Power range high negative neutron flux rate	-2 to -30% of full power
6.	Overtemperature ΔT:	T_H 530 to 650°F T_C 510 to 630°F T_{avg} 530 to 630°F 1 P_{PRZR} 1700 TO 2500 ~~PSI~~psig $F(\Delta I)$ -60 to + 60% ΔT Setpoint 0 to 150% power
7.	Overpower ΔT	T_H 530 to 650°F T_C 510 to 630°F T_{avg} 530 to 630°F ΔT Setpoint 0 to 150% power
8.	Pressurizer low pressure	1700 to 2500 psig
9.	Pressurizer high pressure	1700 to 2500 psig

TABLE 7.2-3
(Sheet 2 of 2)

REACTOR TRIP SYSTEM INSTRUMENTATION (Cont'd)

	Reactor Trip Signal	Typical Range
10.	Pressurizer high water level	Entire cylindrical portion of pressurizer
11.	Low reactor coolant flow	0 to 110% of rated flow
12.	Reactor coolant pump bus undervoltage	0 to 100% rated voltage
13.	Reactor coolant pump bus underfrequency	50 to 65 Hz
14.	Low-low steam generator water level	+ 6 ft., - 12 ft. from nominal full load water level
15.	Turbine Trip (1)	

NOTES:

(1) The reactor trip on turbine trip is anticipatory in that no credit is taken for it in the accident analyses.

TABLE 7.2-4
(Sheet 1 of 5)

REACTOR TRIP CORRELATION

	TRIP[a]		ACCIDENT[b]	TECH SPEC
1.	Power Range High Neutron Flux Trip (Low Setpoint)	1.	Uncontrolled Rod Cluster Control Assembly Bank Withdrawal From a Subcritical Condition (15.2.1)	3.3.1 Table 3.3.1-1 #2
		~~2.~~	~~Uncontrolled Boron Dilution (15.2.4) (Modes 1 and 2)~~	
		~~3.~~ 2.	Excessive Heat Removal Due to Feedwater System Malfunctions (15.2.10)	
		~~4.~~ 3.	Rupture of a Control Rod Drive Mechanism Housing (Rod Cluster Control Assembly Ejection) (15.4.6)	
2.	Power Range High Neutron Flux Trip (High Setpoint)	1.	Uncontrolled Rod Cluster Control Assembly Bank Withdrawal From a Subcritical Condition (15.2.1)	3.3.1 Table 3.3.1-1 #2
		2.	Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power (15.2.2)	
		~~3.~~	~~Uncontrolled Boron Dilution (15.2.4) (Modes 1 and 2)~~	
		~~4.~~ 3.	Excessive Heat Removal Due to Feedwater System Malfunctions (15.2.10)	
		~~5.~~ 4.	Excessive Load Increase Incident (15.2.11)	
		~~6.~~ 5.	Rupture of a Control Rod Drive Mechanism Housing (Rod Cluster Control Assembly Ejection) (15.4.6)	
3.	Intermediate Range High Neutron Flux Trip	1.	Uncontrolled Rod Cluster Control Assembly Bank Withdrawal From a Subcritical Condition (15.2.1)	3.3.1 Table 3.3.1-1 #4
4.	Source Range High Neutron Flux Trip	1.	Uncontrolled Rod Cluster Control Assembly Bank Withdrawal From a Subcritical Condition (15.2.1)	3.3.1 Table 3.3.1-1 #5
		2.	Uncontrolled Boron Dilution (15.2.4) (Modes 2, 3, 4, and 5)	
		3.	Excessive Heat Removal Due to Feedwater System Malfunctions (15.2.10)	

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TABLE 7.2-4
(Sheet 2 of 5)

REACTOR TRIP CORRELATION

	TRIP[a]		ACCIDENT[b]	TECH SPEC
5.	Power Range High Positive Neutron Flux Rate Trip	1.	Uncontrolled Rod Cluster Control Assembly Bank Withdrawal From a Subcritical Condition (15.2.1)	3.3.1 Table 3.3.1-1 #3
		2.	Rupture of a Control Rod Drive Mechanism Housing (Rod Cluster Control Assembly Ejection) (15.4.6)	
6.	Power Range High Negative Flux Rate Trip	1.	Rod Cluster Control Assembly Misalignment (15.2.3)	3.3.1 Table 3.3.1-1 #3
7.	Overtemperature ΔT Trip	1.	Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power (15.2.2)	3.3.1 Table 3.3.1-1 #6
		2.	Uncontrolled Boron Dilution (15.2.4)	
		3.	Loss of External Electrical Load and/or Turbine Trip (15.2.7)	
		4.	Excessive Load Increase Incident (15.2.11)	
		5.	Accidental Depressurization of the Reactor Coolant System (15.2.12)	
		~~6.~~	~~Accidental Depressurization of the Main Steam System (15.2.13)~~	
		~~7.~~	~~Loss of Reactor Coolant From Small Ruptured Pipes or From Cracks in large Pipes Which Actuates ECCS (15.3.1)~~	
38		~~8.~~ 6.	Single Rod Cluster Control Assembly Withdrawal at Full Power (15.3.6)	
		~~9.~~	~~Major Rupture of a Main Steam Line (15.4.2.1)~~	
		~~10.~~ 7.	Major Rupture of a Main Feedwater Pipe (15.4.2.2)	
		8.	Excessive Heat Removal Due to Feedwater System Malfunctions (15.2.10)	

TABLE 7.2-4
(Sheet 3 of 5)

REACTOR TRIP CORRELATION

	TRIP[a]		ACCIDENT[b]	TECH SPEC
8.	Overpower ΔT Trip	1.	Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power (15.2.2)	3.3.1 Table 3.3.1-1 #7
		~~2.~~	~~Loss of External Electrical Load and/or Turbine Trip (15.2.7)~~	
		~~3.~~ 2.	Excessive Heat Removal Due to Feedwater System Malfunctions (15.2.10)	
38		~~4.~~ 3.	Accidental Depressurization of the Main Steam System (15.2.13)	
		~~5.~~ 4.	Major Rupture of a Main Steam Line (15.4.2.1)	
		5.	Excessive Load Increase Incident (15.2.11)	
9.	Pressurizer Low Pressure Trip	1.	Excessive Load Increase Incident (15.2.11)	3.3.1 Table 3.3.1-1 #8
		2.	Accidental Depressurization of the Reactor Coolant System (15.2.12)	
		3.	Accidental Depressurization of the Main Steam System (15.2.13)	
		4.	Inadvertent Operation of Emergency Core Cooling System (15.2.14)	
		5.	Loss of Reactor Coolant From Small Ruptured Pipes or From Cracks in Large Pipes Which Actuates ECCS (15.3.1)	
		6.	Major Reactor Coolant System Pipe Ruptures (LOCA) (15.4.1)	
		7.	Major Rupture of a Main Steam Line (15.4.2.1)	
		8.	Major Rupture of a Main Feedwater Pipe (15.4.2.2)	
		9.	Steam Generator Tube Rupture (15.4.3)	

TABLE 7.2-4
(Sheet 4 of 5)

REACTOR TRIP CORRELATION

	TRIP[a]		ACCIDENT[b]	TECH SPEC
10.	Pressurizer High Pressure Trip	1.	Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power (15.2.2)	3.3.1 Table 3.3.1-1 #8
		2.	Loss of External Electrical Load and/or Turbine Trip (15.2.7)	
		~~3.~~	~~Loss of Normal Feedwater (15.2.8)~~	
		~~4.~~	~~Loss of Offsite Power to Station Auxiliaries (Station Blackout) (15.2.9)~~	
		~~5.~~ 3.	Major Rupture of a Main Feedwater Pipe (15.4.2.2)	
11.	Pressurizer High Water Level	1.	Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power (15.2.2)	3.3.1 Table 3.3.1-1 #9
		2.	Loss of External Electrical Load and/or Turbine Trip (15.2.7)	
38		~~3.~~	~~Loss of Normal Feedwater (15.2.8)~~	
		~~4.~~	~~Loss of Offsite Power to Station Auxiliaries (Station Blackout) (15.2.9)~~	
		~~5.~~ 3.	Major Rupture of a Main Feedwater Pipe (15.4.2.2)	
12.	Low Reactor Coolant Flow	1.	Partial Loss of Forced Reactor Coolant Flow (15.2.5)	3.3.1 Table 3.3.1-1 #10
		2.	Complete Loss of Forced Reactor Coolant Flow (15.3.4)	
		3.	Single Reactor Coolant Pump Locked Rotor (15.4.4)	
13.	Reactor Coolant Pump Bus Undervoltage Trip	1.	Complete Loss of Forced Reactor Coolant Flow (15.3.4)	3.3.1 Table 3.3.1-1 #11
		2.	Partial Loss of Forced Reactor Coolant Flow (15.2.5)	
14.	Reactor Coolant Pump Bus Underfrequency Trip	1.	Complete Loss of Forced Reactor Coolant Flow (15.3.4)	3.3.1 Table 3.3.1-1 #12
		2.	Partial Loss of Forced Reactor Coolant Flow (15.2.5)	

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TABLE 7.2-4
(Sheet 5 of 5)

REACTOR TRIP CORRELATION

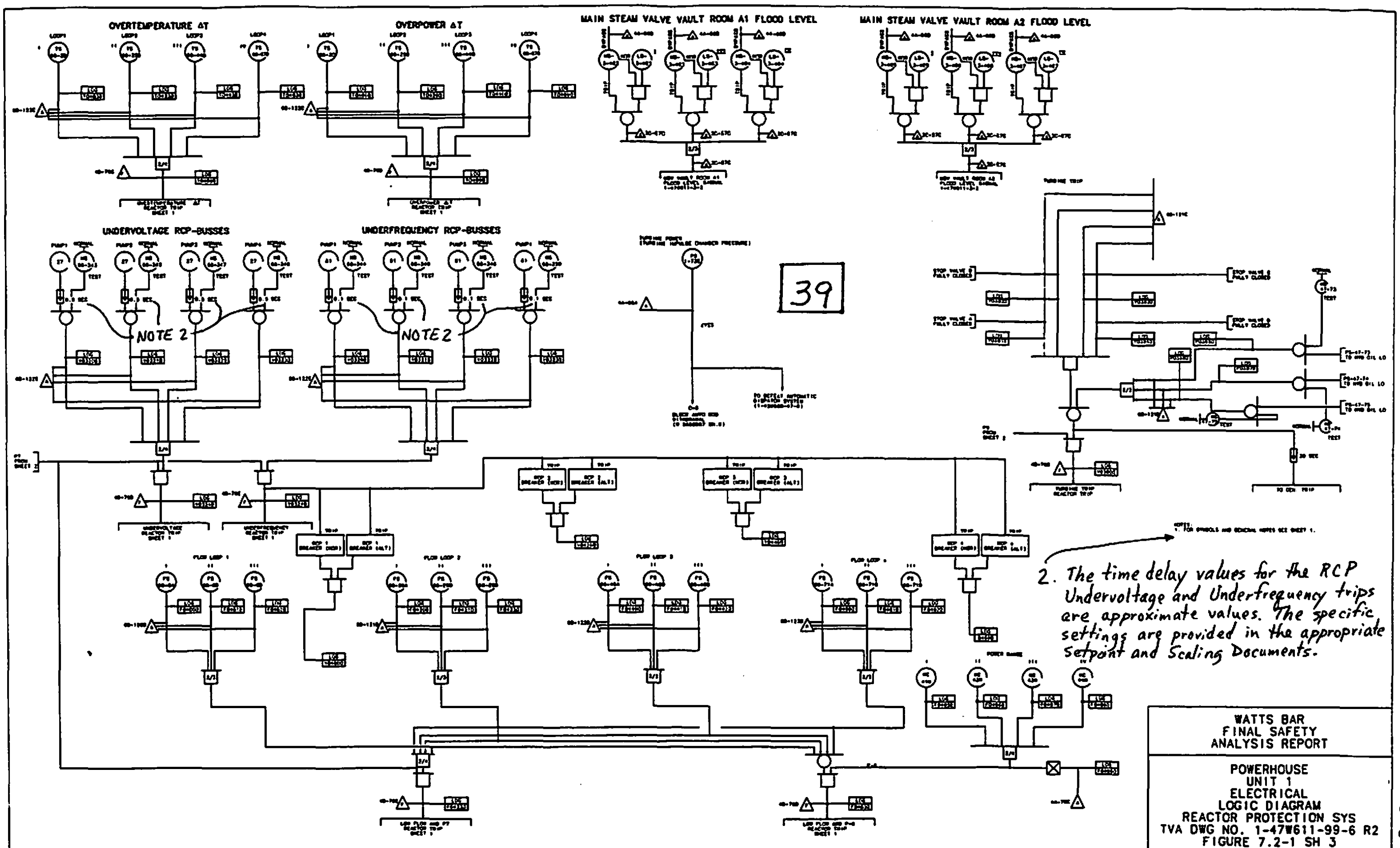
		TRIP[a]		ACCIDENT[b]	TECH SPEC
	15.	Low-low Steam Generator Water Level Trip	1.	Loss of Normal Feedwater (15.2.8)	3.3.1 Table 3.3.1-1 #13
			2.	Loss of Offsite Power to the Station Auxiliaries (Station Blackout)(15.2.9)	
			3.	Major Rupture of a Main Feedwater Pipe (15.4.2.2)	
38			4.	Loss of External Electrical Load and/or Turbine Trip (15.2.7)	
	16.	Turbine Trip-Reactor Trip	~~1.~~	Loss of External Electrical Load and/or Turbine Trip (15.2.7)[c]	3.3.1 Table 3.3.1-1 #14
	17.	Safety Injection Signal Actuation Trip	1.	Accidental Depressurization of the Main Steam System (15.2.13)	3.3.1 Table 3.3.1-1 #15
			~~2.~~	~~Inadvertent Operation of Emergency Core Cooling System (15.2.14)~~	
			~~3.~~ 2.	Major Rupture of a Main Steam Line (15.4.2.1)	
			~~4.~~ 3.	Major Rupture of a Main Feedwater Pipe (15.4.2.2)	
	18.	Manual Trip		Available for all Accidents (Chapter 15)	3.3.1 Table 3.3.1-1 #1

NOTES:

a. Trips are listed in order of discussion in Section 7.2

38 b. References refer to Chapter 15 accident analyses ~~presented in Chapter 15~~ in which the trip may be utilized, either as a primary or backup trip.

c. Reactor trip on turbine trip is an anticipatory trip and is not credited in the accident analyses.



2. The time delay values for the RCP Undervoltage and Underfrequency trips are approximate values. The specific settings are provided in the appropriate Setpoint and Scaling Documents.

WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

POWERHOUSE
UNIT 1
ELECTRICAL
LOGIC DIAGRAM
REACTOR PROTECTION SYS
TVA DWG NO. 1-47W611-99-6 R2
FIGURE 7.2-1 SH 3

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SAFETY ASSESSMENT/SCREENING REVIEW/SAFETY EVALUATION COVERSHEET

Document No: UFSAR Change 1553, EDC E-50045 Rev. 0

805

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Screening Review Only ☐

Safety Assessment/Screening Review ☐

Safety Assessment/Screening Review/Safety Evaluation ☒

Procedure Exemption ☐

Procedure Change Evaluation ☐

RT 12-16-98

Plant WBN

Affected Unit(s) 1

Preparing Group WBP-LEE-DFF

Preparer W. D. Webb

Reviewer

Activity	Number (Include Revision No.)	
☐ Design Change	DCN No.	
☒ Engineering Document Change	EDC No.	E-50045
☐ Temporary Alteration	TACF No.	
☐ Special Test/Experiment	Special Test No.	
☐ Temporary Shielding Request	TSRF No.	
☐ Procedure Change	Procedure No. and	
☐ New Procedure	PCF No. (if applicable)	
	Procedure No.	
☐ Maintenance	WR/WO No.	
☒ Other (Identify)	SAR change package number 1553	

Comments:
Tracking no. WBPLEE-098-080-0.
SA/SR/SE supports both SAR change package 1553 resulting from review of Section 7.2 and EDC E-50045 which makes documentation changes identified during the SAR review.

Revision: (Provide a brief summary of the reason for the revision to the SR, SA, or SE)

Distribution:
cc: EDMS - WT 3B-K
W. D. Webb - Return original to originating document

SAFETY ASSESSMENT FORM

Page 1 of ____

Document No: UFSAR Change 1553, EDC E-50045 Rev. 0

I. SAFETY ASSESSMENT

A. Description

1. Brief synopsis of the change, special test, experiment or condition including the systems, structures, and components affected. Include the number of the activity proposed (e.g., ECN/DCN No., procedure No.).

FSAR change package 1553 documents changes resulting from a review of Section 7.2. A copy of the section showing the changes is attached. The attachment contains reference numbers near each change which correspond to the item numbers used in this SA/SE. Some items appear several times. Items 1-15 and 17-38 are corrections of non-numerical typographical errors, administrative changes, or minor editorial changes that do not change the intent. As such, they are considered to be non-significant changes as defined in NADP-7, Section 5.0, and do not require an SA, SR or SE. Therefore, these non-significant changes will not be further addressed. Item 1 includes those non-significant changes for which no explanation is needed. The remaining items (16 and 39) will be addressed in the Screening Review and Safety Evaluation. Design basis document changes are being made by EDC E-50045 to ensure consistency with existing design and licensing bases, including changes associated with the UFSAR changes described below.

1 (pages 7.2-1, 3, 5, 7, 8, 10, 11, 17, 19-22, 25, 27-29, 31-33, 37, 38; and Tables 7.2-1 sheet 1 and 7.2-3 sheet 1): Minor changes which do not need explanation and, therefore, will not be addressed individually. Examples of these are addition of acronyms, correction of reference or figure numbers, addition of references to other sections which address related topics, and editorial changes such as verb tense, word choice, grammatical corrections.

2 (page 7.2-1): Clarify that flow refers to reactor coolant flow.

3 (page 7.2-1): Clarify that the NIS, as well as the process protection system, provides input signals to the protection system logic trains as described elsewhere in the UFSAR (e.g., Section 7.2.1.1.7).

4 (page 7.2-1): Clarify that the referenced shunt relay is the shunt trip relay, consistent with a previous description of the relay.

5 (page 7.2-3): Delete redundant information - the power level for the P-10 interlock is given in the preceding paragraph.

6 (page 7.2-3): Editorial change to clarify reinstatement of the source range high flux trip function below P-10.

7 (page 7.2-5, 6): Clarify that the values of some of the parameters of the OTΔT and OPΔT trip functions are not given in the Technical Specifications (TS) but are controlled by plant procedures. The values of such parameters (e.g., ΔT_0) are based on plant conditions and may be periodically adjusted as described in the TS.

8 (page 7.2-5): Delete the reference to the T′ parameter since it is not used in the OPΔT trip function which is the subject of the section.

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9 (page 7.2-7): Clarify that flow measurements used in the low reactor coolant flow trip are derived from elbow taps. As noted later in the paragraph, there are three flow signals per loop. The present wording concerning the number of elbow taps is ambiguous and could be interpreted to mean that there are four signals per loop instead of four coolant loops. Reactor Protection System Description N3-99-4003 Section 3.1.1.1(d)(1) is similarly revised.

10 (page 7.2-8): Delete reference to approximate setpoint values for the Reactor Coolant Pump undervoltage (70%) and underfrequency (57 Hz) reactor trips. This is consistent with the discussions of other reactor trip functions, which do not give the trip setpoint, and with Section 7.2.1.1.10 which states that reactor trip setpoints are given in the Technical Specifications. In addition the underfrequency setpoint statement is redundant to information provided in Section 15.3.4.1. The values given in the TS are 4830 v (70%) and 57.5 Hz, respectively.

11 (page 7.2-9): Simplify the discussion of compliance with protection system separation criteria for the reactor trip - turbine trip channels and eliminate reference to the number of channels since this is described in a previous paragraph.

12 (page 7.2-11): Clarify that the section does not concern all permissives, only P-6 and P-10. Also the statement that permissives are digital is not useful since they are inherently on/off (enable/block) functions.

13 (page 7.2-11): The statement that **the reactor will be allowed to operate with one inactive loop** is misleading. In the context of the description of the P-8 interlock, it means that low flow in only one loop will not cause a reactor trip below P-8. Deleting the statement will eliminate potential misinterpretation and is consistent with SAR Section 15.2.6 and Technical Specification 3.4.4, which require all reactor coolant pumps to be operating in modes 1 and 2.

14 (pages 7.2-12, 13, 14): The deleted subscript notations in Section 7.2.1.1.4 identify the number of reactor coolant temperature measurement channels per loop and the number of loops. This information is repetitive and is considered excessive detail which complicates the discussion of temperature measurements used in the OTΔT and OPΔT trip functions and reactor control system. The changes will simplify the description of the methodology used for calculating reactor coolant temperature.

15 (page 7.2-14, 33): The insert for Section 7.2.1.1.4 contains additional information pertaining to RCS temperature measurement which is more appropriately included in this section than Section 7.2.2.3.2 from which it was moved. The changes to the relocated text are editorial.

16 (page 7.2-14, 34, 35): The update to Section 7.2.1.1.5 is taken from text in Section 7.2.2.3.4 with clarifications and editorial changes. The relocated discussion of the pressurizer water level instrumentation is more appropriately included in this section than Section 7.2.2.3.4, which deals with control and protection system interaction. The changes to 7.2.1.1.5 are based on a general description of the Westinghouse pressurizer level design, channel independence, and actual installation attributes found on TVA physical drawings. Also, the hydrogen gas entrainment issue documented in NRC Information Bulletin No. 92-54, Level Instrumentation Inaccuracies Caused by Rapid Depressurization, is retained and clarified. Similar clarification is made to Reactor Protection System Description

N3-99-4003 Section 3.1.1.2(d). The original text in 7.2.2.3.4 provides some information that is too detailed and is not pertinent to the subject of discussion. It also includes a statement that the error effect on the level measurement during a blowdown accident would be about one inch. The basis for this value is not known; however, the worst case reference leg loss of fill error due to a rapid RCS depressurization event is no more than 12 inches elevation head. This value is based on the relative elevation difference between the condensing chamber and the reference leg sensor bellows. The Westinghouse Owners Group response to this issue is found in RIMS # L44930216800. The channel error value discrepancy is documented in WBPER980417. The remaining text in 7.2.2.3.4 is revised to clarify the control and protection system interaction discussion.

17 (page 7.2-15): Delete **comparator input signals**, which is redundant to the next term, **partial trip**.

18 (page 7.2-17): The statement that the parameter values which will require reactor trip are given in Chapter 15 is misleading and is being deleted. It implies that the trip setpoints are identified in Chapter 15, which is not the case. Chapter 15 analyses do identify the protection system functions required for mitigation of the analyzed events and may give the limiting values of certain parameters. The adequacy of the setpoints is also demonstrated. As noted, the trip setpoints are given in the Technical Specifications.

19 (page 7.2-18): Editorial change to more accurately describe the drawings which reflect the design of the protection systems.

20 (page 7.2-19): Delete the examples of Eagle 21 self-diagnostic features; they are considered unnecessary detail. Eagle 21 hardware and software are not affected.

21 (page 7.2-21): Delete part of the discussion of operation with one reactor coolant loop out of service. The language is no longer applicable since a previous SAR change deleted the discussion of N-1 operation, for which WBN has not been analyzed. This change will make the text consistent with Section 15.2.6 and with Technical Specification 3.4.4 and is therefore considered non-significant.

22 (page 7.2-21, 22): The discussion on reactor coolant flow measurement is revised to clarify how the nominal full power flow reference point is established and its relation to the low flow trip and thermal design flow verification. Values for accuracy and repeatability are not given in the SAR for other protection system instrumentation and, therefore, the statement is replaced with a more general statement. Uncertainty of the flow measurement instrumentation is documented in the setpoint calculations as noted in Section 7.2.1.2.6.

23 (page 7.2-24): Portions of the discussion of control and protection system interaction are revised to clarify the requirement. The discussion of how the SG low-low water level protective function and the control system Median Signal Selector satisfy this requirement is deleted since it is redundant to the information provided in Section 7.2.2.3.5. Reactor Protection System Description N3-99-4003 is also revised to move and clarify the discussion of the requirements for control and protection system interaction from Section 3.1.1.2 to Section 2.2.11, where the issue is also discussed.

24 (page 7.2-25): Clarify that protection system channels may be tested in the bypass mode consistent with the Process Protection Channel Tests discussion which follows in the SAR and as described in the Technical Specifications. Delete the reference to the setpoint document since it repeats information in Section 7.2.1.2.6. Reactor Protection System Description N3-99-4003 Section 3.1.1.4 is also revised to clarify that channels may be tested in the bypass mode and to add that the source or intermediate range channels must be tested in bypass because of their 1/2 trip logic.

25 (page 7.2-26): The change summarizes the deleted information concerning testing of the NIS and adds reference to the Technical Specifications, which govern periodic testing.

26 (page 7.2-28): The change deletes one of the two listed methods which may be used to verify that the SSPS input error inhibit switch contacts are in their normal state after completion of a logic test. This is done to confirm that the SSPS logic is operational. Either of the listed methods is adequate, but only one is actually used. This information is not addressed in the SER.

27 (page 7.2-29): Editorial change for identification of the reactor trip and bypass breakers to be consistent with Section 7.2.1.1.

28 (page 7.2-30): Clarify that the plant annunciator provides the indication of an Eagle 21 protection system channel in bypass.

29 (page 7.2-31): Clarify that the failure being discussed falls within the single failure criterion. This is implicit from the discussion of compliance with the single failure criterion in Section 7.2.2.2 subsection 2.

30 (page 7.2-32): The change clarifies that channel trip alarms are not provided for actuation of source or intermediate range channels. The present wording implies that actuation of any protection system channel (partial trip) will actuate an alarm to alert the operator of an abnormal operating condition. This is correct except for trip functions which require only one channel (i.e., 1/2 logic) to initiate reactor trip. An alarm is provided for each trip function. A partial trip alarm for these channels would be redundant to the reactor trip alarm for each function and, therefore, would serve no purpose. Reactor Protection System Description N3-99-4003 Section 2.2.10.2 is similarly revised.

31 (page 7.2-33, 34): The deleted paragraph contains criteria information which is redundant to the discussion in Section 7.2.2.2 subsection 7 on control and protection system interaction. The statement concerning the 2/4 logic is retained as part of the basis for demonstrating compliance with the criteria for the reactor coolant temperature channels. Reactor Protection System Description N3-99-4003 Section 3.1.1.2(b) is also revised to clarify compliance of the RCS temperature channels with the requirement.

32 (page 7.2-33): Clarify that the overtemperature and overpower rod block and turbine runback occurs before the OTΔT/OPΔT reactor trip setpoints are reached. Reactor Protection System Description N3-99-4003 Section 3.1.1.2(b) is similarly revised.

33 (page 7.2-34) Clarify compliance of the pressurizer pressure channels with the control and protection system interaction criteria. Reactor Protection System Description N3-99-4003 Section 3.1.1.2(c) is similarly revised.

34 (page 7.2-36): Clarify the role of the Median Signal Selector in eliminating the potential for SG water level control and protection system interaction.

35 (page 7.2-36): Clarify the function of the rod stop interlocks with respect to reactor control and protection. The revised text is more consistent with the rod stop discussion found in Section 7.7.1.4.

36 (Table 7.2-1 Sheet 1): Clarify the purpose of the P-7 permissive with respect to the RCP undervoltage trip for consistency with comments on similar features for other trip functions.

37 (Table 7.2-2 Sheet 1): Clarify the function of the P-10 permissive. It is also an input to P-7 as shown on Figure 7.2-1 Sheet 2. Reactor Protection System Description N3-99-4003 Table 1 is similarly revised.

38 (Table 7.2-4): This table lists the reactor trips and the various accident analyses for which each trip could provide protection. The intent of the table is to demonstrate the diversity of and comprehensive protection provided by the reactor trip system against various postulated events and to correlate the trip functions with the analyses in which they may be utilized, either as a primary or secondary protective function. Chapter 15, along with the Accident Analysis Parameters Checklist, WB-DC-40-70, provides the accident analysis discussion and identifies the protection system functions which provide accident mitigation. The additions and deletions to the table are made for consistency with the safety analyses of record as reflected in the design and licensing basis and do not represent analysis changes or protection system changes. Therefore, they are considered to be non-significant as discussed at the beginning of this section. Neutron Monitoring System Description N3-85-4003 Table 2 is also revised for consistency with WB-DC-40-70.

39 (Figure 7.2–1 Sheet 3): This drawing, 1-47W611-99-6, shows time delays of 0.5 and 0.1 seconds, respectively, for Reactor Coolant Pump undervoltage (UV) and underfrequency (UF) reactor trip signals. NE SSDs specify settings of 23 cycles (0.383 sec) for the UV and 5 cycles (0.087 sec) for the UF as determined by calculations WBPE0689009007 and WBPE0689009008. This discrepancy is documented in WBPER980417. The drawing will be revised by DCN E-50045.

In addition to the UFSAR changes described above, Reactor Protection System Description N3-99-4003 is revised by EDC E-50045 to clarify the design basis and functions, correct minor errors, and make editorial changes to maintain consistency between the design basis and the UFSAR. Also reference 7.5.26 of N3-99-4003 is changed from WCAP-14419 to WCAP-14738, "Revised Thermal Design Procedure Instrument Uncertainty Methodology," to reflect the current design and licensing basis for the RCS flow and reactor power calorimetrics instrument uncertainty, which became effective at cycle 2 startup (reference DCN M-39293-A, FSAR change 1473, and TS Amendment 7). WCAP-14419 was superseded by WCAP-14738. These documents are not listed in the UFSAR or TS.

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2. References (SAR/Technical Specifications/etc.).

1. UFSAR
2. FSAR change package 1473
3. Safety Evaluation Report and Supplements 1-20
4. Technical Specifications and Bases, including Amendment 7
5. Technical Requirements and Bases Manual
6. NADP-7, FSAR Management
7. EDC E-50045
8. DCN M-39293-A
9. SPP-9.4, 10CFR50.59 Evaluations of Changes, Tests, and Experiments
10. System Description Documents
    - N3-03-4003, Feedwater
    - N3-68-4001, Reactor Coolant
    - N3-85-4003, Rod Control
    - N3-92-4003, Nuclear Instrumentation
    - N3-99-4003, Reactor Protection
11. Design Criteria
    - WB-DC-30-4, Separation/Isolation
    - WB-DC-40-64, Design Basis Events
    - WB-DC-40-70, Accident Analysis Parameters Checklist
12. Plant Procedures
    - AOI-5, Unscheduled Removal of One RCP
    - TI-68.011, Acquisition of T', T", ΔT0, THOT, TCOLD, and RCS Streaming Factors for Periodic Update
    - 1-SI-68-31, Reactor Coolant System Total Flow Measurement
    - Annunciator Response Instructions (ARI)
13. WBPER980417
14. Vendor Manual WBN-VTM-W120-2991, Eagle 21
15. WBPE0689009007 and WBPE0689009008, RCP Undervoltage and Underfrequency uncertainty calculations
16. SSDs 1-27-68-8A, 1-81-68-8, 1-T-68-2, 1-F-68-6A
17. NRC Information Bulletin No. 92-54, Level Instrumentation Inaccuracies Caused by Rapid Depressurization, and Westinghouse Owners Group response (L44930216800).
18. Vendor Documents
    - WCAP-12096, Westinghouse Setpoint Methodology for Protection Systems
    - WCAP-14738, Westinghouse Revised Thermal Design Procedure Instrument Uncertainty Methodology
    - Westinghouse Precautions, Limitations and Setpoints Document (T33950913801)
19. IEEE 279-1971, Criteria for Protection Systems for Nuclear Power Generating Stations.
20. Drawings
    - 1-47W610-3, -68 series
    - 1-47W611-3-8; 99-2, 6, 7
    - 47W600-331, 332
    - 5655D87-series (54114-1)
    - 1682C30-23 (54114-1)

B. **Safety Assessment Checklist (Form SPP-9.4-4) - (required for all changes)**

See attached.

C. **Acceptability from a Nuclear Safety Standpoint**

A determination if the proposed activity is acceptable from a nuclear safety perspective. This includes a written justification for the acceptability.

The following address applicable Safety Assessment Checklist items.

6. **Design Basis Document**

Item 16: The UFSAR statement concerning the pressurizer water level instrumentation measurement error due to a rapid RCS depressurization event is not supported by the design basis. Instrument uncertainty calculations were based on the attributes found on the installation drawings and, therefore, the change to the UFSAR has no impact on design basis documents and no protection system parameters are affected by the discrepancy.

Item 39: The relay settings for the time delays for the Reactor Coolant Pump undervoltage and underfrequency reactor trip signals are based on NE SSDs which reflect the results of uncertainty calculations WBPE0689009007 and WBPE0689009008. The response time values given in System Description N3-99-4003 for these functions are based on these calculations. Therefore, the drawing change has no impact on other design basis documents and no protection system parameters are affected by the discrepancy.

System Description N3-99-4003 is revised by DCN E-50045 as required to support changes described in Part A of this SA, correct minor errors, and make editorial changes to maintain consistency between the DBD and the FSAR. These changes do not require any physical modifications to the plant and do not adversely impact nuclear safety.

23. **Instrument Setpoints**

Item 16: The high pressurizer water level trip setpoint calculation (WCAP-12096) is based on the installation and is not affected by the UFSAR change concerning the level measurement error.

Item 39: The actual relay settings for the time delays for the Reactor Coolant Pump undervoltage and underfrequency reactor trip signals are based on uncertainty calculations WBPE0689009007 and WBPE0689009008 and NE SSDs. The drawing change is documentation only to reflect the current design basis and installation.

50. **System Design Parameters**

Item 16: Setpoints and scaling of the pressurizer water level instrumentation were based in part on installation attributes. The level measurement error described in the UFSAR was not a factor in the design and, therefore, no system parameters are affected by the UFSAR change.

Item 39: The actual relay settings for the time delays for the Reactor Coolant Pump undervoltage and underfrequency reactor trip signals are based on NE SSDs which reflect the results of uncertainty calculations WBPE0689009007 and WBPE0689009008. The time delay values shown on drawing 1-47W611-99-6 were not used in setting the relays or in the uncertainty analyses. Therefore, no protection system parameters are affected by this discrepancy.

Conclusion

Based on the above, the SAR and design basis document changes described in Part A do not impact any component or system safety functions, do not require plant modifications, and are acceptable under the existing safety analyses. The changes do not affect any Technical Specifications. Therefore, these changes have no impact on nuclear safety.

D. Review and Approvals

	Name	Signature	Date
Preparer:	W. D. Webb	W D Webb	12-14-98
Reviewer:	DAN F. FAULKNER	Dan F Faulkner	12/16/98
Other: Reviewers (as appropriate)			

SAFETY ASSESSMENT CHECKLIST

Document No. UFSAR Change 1553, EDC E-50045 Page ____ of ____

	Potential Impact On Nuclear Safety	N/A	
1.	☐	☒	ASME Section XI
2.	☐	☒	Chemistry Changes or Chemical Release Pathways
3.	☐	☒	Compensatory Measure
4.	☐	☒	Control Room Habitability
5.	☐	☒	Decay Heat Removal Capability
6.	☒	☐	Design Basis Document
7.	☐	☒	Digital Upgrade (NRC Generic Letter 95-02)
8.	☐	☒	Electrical Breaker Alignment Changes
9.	☐	☒	Electrical Loads
10.	☐	☒	Electrical Separation/Isolation
11.	☐	☒	EMI/RFI Potential
12.	☐	☒	Environmental Impact Statement (See SPP-5.5)
13.	☐	☒	Environmental Qualification Category
14.	☐	☒	Equipment Diversity
15.	☐	☒	Equipment Failure Modes
16.	☐	☒	Equipment Redundancy
17.	☐	☒	Equipment Reliability
18.	☐	☒	Erosion/Corrosion/MIC
19.	☐	☒	Fire Protection (Appendix R)
20.	☐	☒	Hazardous Material
21.	☐	☒	Heavy Load Lifts or Safe Load Paths (NUREG-0612)
22.	☐	☒	Human Factors
23.	☒	☐	Instrument Setpoints
24.	☐	☒	Instrument/Relay Settings
25.	☐	☒	Internal Flooding Protection (MELB)
26.	☐	☒	Internal/External Missiles
27.	☐	☒	Jet Impingement Effects
28.	☐	☒	Materials Compatibility
29.	☐	☒	Modification to Non-Seismic Areas in CB/AB

SAFETY ASSESSMENT CHECKLIST			
Document No. UFSAR Change 1553, EDC E-50045			Page ______ of ______
	Potential Impact On Nuclear Safety	N/A	
30.	☐	☒	Physical Separation
31.	☐	☒	Pipe Breaks
32.	☐	☒	Pipe Vibration
33.	☐	☒	Pipe Whip
34.	☐	☒	Primary Containment Integrity/Isolation
35.	☐	☒	Protective Coatings Inside Containment
36	☐	☒	Radioactive Effluent (Liquid or Gaseous) Release Pathways
37.	☐	☒	Radwaste System Changes
38.	☐	☒	Reactor Coolant Pressure boundary
39.	☐	☒	Reactor Core Parameters
40.	☐	☒	Requires an increase in operator staffing to complete newly required actions
41.	☐	☒	Response Time of Emergency Safeguards Equipment
42.	☐	☒	Safety Injection/Core Cooling Capability
43.	☐	☒	Scaffolding
44.	☐	☒	Secondary Containment Integrity/Isolation
45.	☐	☒	Security System
46.	☐	☒	Seismic/Dead Weight
47.	☐	☒	Shield Building Integrity (SQN/WBN)
48.	☐	☒	Shutdown Reactivity Control
49.	☐	☒	Single Failure Criteria
50.	☒	☐	System Design Parameters
51.	☐	☒	Temporary Shielding
52.	☐	☒	Test and Retest Scoping Document (Post Modification Test)
53.	☐	☒	Tornado or External Flood Protection
54.	☐	☒	Toxic Gases
55.	☐	☒	Valve Alignment Changes
56.	☐	☒	Ventilation Cooling for Electronic Equipment
57.	☐	☒	Water Spray/Condensation

SCREENING REVIEW FORM

Document No: UFSAR Change 1553, EDC E-50045 Rev. 0

A. Potential Technical Specification (T/S) Impact (List TS sections reviewed)

Yes ☐ No ☒ Is a change to the T/S required for conducting or implementing the change (design or procedure), test, or experiment?

Justification:

There are no plant modifications associated with the SAR changes or the design basis document changes processed by DCN E-50045. The changes are primarily editorial changes and clarifications. Based on review of TS Sections 3.3.1, 3.3.2, 3.3.3, 3.3.4 and 3.4.9, no TS changes are required.

If the answer is "Yes," a T/S change is required prior to implementation or the activity needs to be revised or canceled.

B. Potential Safety Analysis Impact (List FSAR sections reviewed)

Yes ☐ No ☒ Is this a special test, or experiment not described in the SAR?

Does the proposed activity affect (directly or indirectly) any information presented in the SAR or deviate from the description given in the SAR?

Yes ☒ No ☐ By changing: The system design or functional requirements; the technical content of text, tables, graphs, or figures? (For radwaste changes see Note in Appendix B for guidance.) If the answer is "Yes," process an FSAR change.

Justification:

This SA/SE supports changes to Section 7.2 documented in SAR change package 1553 and design documentation changes made in DCN E-50045. A Safety Evaluation will be performed for those changes which were determined to be significant.

Does the proposed change involve new procedures or instructions or revisions thereof that:

Yes ☐ No ☒ N/A ☐ Differ with system operation characteristics from that described in the SAR?

Yes ☐ No ☒ N/A ☐ Conflict with or affect a process or procedure outlined, summarized, or described in the SAR?

Justification:

SE is required as noted above.

If the questions are answered "No" or "N/A," the activity may be implemented without a safety evaluation. If any question is answered "Yes," an SE is required.

SCREENING REVIEW FORM

Page 2 of ____

Document No: UFSAR Change 1553, EDC E-50045 Rev. 0

C. **Review and Approvals**

Preparer:	W. D. Webb	W D Webb	12-14-98
	Name	*Signature*	*Date*
Reviewer:	DAN F. FAULKNER	D F Faulkner	12/16/98
	Name	*Signature*	*Date*
Other: Reviewers (as appropriate)	*Name/Organization*	*Signature*	*Date*

SAFETY EVALUATION FORM

Document No.: UFSAR Change 1553, EDC E-50045 Rev.: 0

A. Description and Accident Evaluation:

Detailed description of the change, test, or experiment, including the design basis accident, and credible failure modes of activity.

FSAR change package 1553 documents changes resulting from a review of Section 7.2. A copy of the section showing the changes is attached. The attachment contains reference numbers near each change which correspond to the item numbers used in this SA/SE. Some items appear several times. Items 1-15 and 17-38 are corrections of non-numerical typographical errors, administrative changes, or minor editorial changes that do not change the intent. As such, they are considered to be non-significant changes as defined in NADP-7, Section 5.0, and do not require an SA, SR or SE. Therefore, these non-significant changes will not be further addressed. Item 1 includes those non-significant changes for which no explanation is needed. The remaining items (16 and 39) will be addressed in the Screening Review and Safety Evaluation. Design basis document changes are being made by EDC E-50045 to ensure consistency with existing design and licensing bases, including changes associated with the UFSAR changes described below.

1 (pages 7.2-1, 3, 5, 7, 8, 10, 11, 17, 19-22, 25, 27-29, 31-33, 37, 38; and Tables 7.2-1 sheet 1 and 7.2-3 sheet 1): Minor changes which do not need explanation and, therefore, will not be addressed individually. Examples of these are addition of acronyms, correction of reference or figure numbers, addition of references to other sections which address related topics, and editorial changes such as verb tense, word choice, grammatical corrections.

2 (page 7.2-1): Clarify that flow refers to reactor coolant flow.

3 (page 7.2-1): Clarify that the NIS, as well as the process protection system, provides input signals to the protection system logic trains as described elsewhere in the UFSAR (e.g., Section 7.2.1.1.7).

4 (page 7.2-1): Clarify that the referenced shunt relay is the shunt trip relay, consistent with a previous description of the relay.

5 (page 7.2-3): Delete redundant information - the power level for the P-10 interlock is given in the preceding paragraph.

6 (page 7.2-3): Editorial change to clarify reinstatement of the source range high flux trip function below P-10.

7 (page 7.2-5, 6): Clarify that the values of some of the parameters of the OTΔT and OPΔT trip functions are not given in the Technical Specifications (TS) but are controlled by plant procedures. The values of such parameters (e.g., ΔT_0) are based on plant conditions and may be periodically adjusted as described in the TS.

8 (page 7.2-5): Delete the reference to the T″ parameter since it is not used in the OPΔT trip function which is the subject of the section.

9 (page 7.2-7): Clarify that flow measurements used in the low reactor coolant flow trip are derived from elbow taps. As noted later in the paragraph, there are three flow signals per loop. The present wording concerning the number of elbow taps is ambiguous and could be interpreted to mean that there are four signals per loop instead of four coolant loops. Reactor Protection System Description N3-99-4003 Section 3.1.1.1(d)(1) is similarly revised.

10 (page 7.2-8): Delete reference to approximate setpoint values for the Reactor Coolant Pump undervoltage (70%) and underfrequency (57 Hz) reactor trips. This is consistent with the discussions of other reactor trip functions, which do not give the trip setpoint, and with Section 7.2.1.1.10 which states that reactor trip setpoints are given in the Technical Specifications. In addition the

underfrequency setpoint statement is redundant to information provided in Section 15.3.4.1. The values given in the TS are 4830 v (70%) and 57.5 Hz, respectively.

11 (page 7.2-9): Simplify the discussion of compliance with protection system separation criteria for the reactor trip - turbine trip channels and eliminate reference to the number of channels since this is described in a previous paragraph.

12 (page 7.2-11): Clarify that the section does not concern **all** permissives, only P-6 and P-10. Also the statement that permissives are digital is not useful since they are inherently on/off (enable/block) functions.

13 (page 7.2-11): The statement that **the reactor will be allowed to operate with one inactive loop** is misleading. In the context of the description of the P-8 interlock, it means that low flow in only one loop will not cause a reactor trip below P-8. Deleting the statement will eliminate potential misinterpretation and is consistent with SAR Section 15.2.6 and Technical Specification 3.4.4, which require all reactor coolant pumps to be operating in modes 1 and 2.

14 (pages 7.2-12, 13, 14): The deleted subscript notations in Section 7.2.1.1.4 identify the number of reactor coolant temperature measurement channels per loop and the number of loops. This information is repetitive and is considered excessive detail which complicates the discussion of temperature measurements used in the OTΔT and OPΔT trip functions and reactor control system. The changes will simplify the description of the methodology used for calculating reactor coolant temperature.

15 (page 7.2-14, 33): The insert for Section 7.2.1.1.4 contains additional information pertaining to RCS temperature measurement which is more appropriately included in this section than Section 7.2.2.3.2 from which it was moved. The changes to the relocated text are editorial.

16 (page 7.2-14, 34, 35): The update to Section 7.2.1.1.5 is taken from text in Section 7.2.2.3.4 with clarifications and editorial changes. The relocated discussion of the pressurizer water level instrumentation is more appropriately included in this section than Section 7.2.2.3.4, which deals with control and protection system interaction. The changes to 7.2.1.1.5 are based on a general description of the Westinghouse pressurizer level design, channel independence, and actual installation attributes found on TVA physical drawings. Also, the hydrogen gas entrainment issue documented in NRC Information Bulletin No. 92-54, Level Instrumentation Inaccuracies Caused by Rapid Depressurization, is retained and clarified. Similar clarification is made to Reactor Protection System Description N3-99-4003 Section 3.1.1.2(d). The original text in 7.2.2.3.4 provides some information that is too detailed and is not pertinent to the subject of discussion. It also includes a statement that the error effect on the level measurement during a blowdown accident would be about one inch. The basis for this value is not known; however, the worst case reference leg loss of fill error due to a rapid RCS depressurization event is no more than 12 inches elevation head. This value is based on the relative elevation difference between the condensing chamber and the reference leg sensor bellows. The Westinghouse Owners Group response to this issue is found in RIMS # L44930216800. The channel error value discrepancy is documented in WBPER980417. The remaining text in 7.2.2.3.4 is revised to clarify the control and protection system interaction discussion.

17 (page 7.2-15): Delete **comparator input signals**, which is redundant to the next term, **partial trip.**

18 (page 7.2-17): The statement that the parameter values which will require reactor trip are given in Chapter 15 is misleading and is being deleted. It implies that the trip setpoints are identified in Chapter 15, which is not the case. Chapter 15 analyses do identify the protection system functions required for mitigation of the analyzed events and may give the limiting values of certain parameters. The adequacy of the setpoints is also demonstrated. As noted, the trip setpoints are

given in the Technical Specifications.

19 (page 7.2-18): Editorial change to more accurately describe the drawings which reflect the design of the protection systems.

20 (page 7.2-19): Delete the examples of Eagle 21 self-diagnostic features; they are considered unnecessary detail. Eagle 21 hardware and software are not affected.

21 (page 7.2-21): Delete part of the discussion of operation with one reactor coolant loop out of service. The language is no longer applicable since a previous SAR change deleted the discussion of N-1 operation, for which WBN has not been analyzed. This change will make the text consistent with Section 15.2.6 and with Technical Specification 3.4.4 and is therefore considered non-significant.

22 (page 7.2-21, 22): The discussion on reactor coolant flow measurement is revised to clarify how the nominal full power flow reference point is established and its relation to the low flow trip and thermal design flow verification. Values for accuracy and repeatability are not given in the SAR for other protection system instrumentation and, therefore, the statement is replaced with a more general statement. Uncertainty of the flow measurement instrumentation is documented in the setpoint calculations as noted in Section 7.2.1.2.6.

23 (page 7.2-24): Portions of the discussion of control and protection system interaction are revised to clarify the requirement. The discussion of how the SG low-low water level protective function and the control system Median Signal Selector satisfy this requirement is deleted since it is redundant to the information provided in Section 7.2.2.3.5. Reactor Protection System Description N3-99-4003 is also revised to move and clarify the discussion of the requirements for control and protection system interaction from Section 3.1.1.2 to Section 2.2.11, where the issue is also discussed.

24 (page 7.2-25): Clarify that protection system channels may be tested in the bypass mode consistent with the Process Protection Channel Tests discussion which follows in the SAR and as described in the Technical Specifications. Delete the reference to the setpoint document since it repeats information in Section 7.2.1.2.6. Reactor Protection System Description N3-99-4003 Section 3.1.1.4 is also revised to clarify that channels may be tested in the bypass mode and to add that the source or intermediate range channels must be tested in bypass because of their 1/2 trip logic.

25 (page 7.2-26): The change summarizes the deleted information concerning testing of the NIS and adds reference to the Technical Specifications, which govern periodic testing.

26 (page 7.2-28): The change deletes one of the two listed methods which may be used to verify that the SSPS input error inhibit switch contacts are in their normal state after completion of a logic test. This is done to confirm that the SSPS logic is operational. Either of the listed methods is adequate, but only one is actually used. This information is not addressed in the SER.

27 (page 7.2-29): Editorial change for identification of the reactor trip and bypass breakers to be consistent with Section 7.2.1.1.

28 (page 7.2-30): Clarify that the plant annunciator provides the indication of an Eagle 21 protection system channel in bypass.

29 (page 7.2-31): Clarify that the failure being discussed falls within the single failure criterion. This is implicit from the discussion of compliance with the single failure criterion in Section 7.2.2.2 subsection 2.

30 (page 7.2-32): The change clarifies that channel trip alarms are not provided for actuation of source or intermediate range channels. The present wording implies that actuation of any protection system channel (partial trip) will actuate an alarm to alert the operator of an abnormal operating condition. This is correct except for trip functions which require only one channel (i.e., 1/2 logic) to initiate reactor trip. An alarm is provided for each trip function. A partial trip alarm for these channels would be redundant to the reactor trip alarm for each function and, therefore, would serve no purpose. Reactor Protection System Description N3-99-4003 Section 2.2.10.2 is similarly revised.

31 (page 7.2-33, 34): The deleted paragraph contains criteria information which is redundant to the discussion in Section 7.2.2.2 subsection 7 on control and protection system interaction. The statement concerning the 2/4 logic is retained as part of the basis for demonstrating compliance with the criteria for the reactor coolant temperature channels. Reactor Protection System Description N3-99-4003 Section 3.1.1.2(b) is also revised to clarify compliance of the RCS temperature channels with the requirement.

32 (page 7.2-33): Clarify that the overtemperature and overpower rod block and turbine runback occurs before the OTΔT/OPΔT reactor trip setpoints are reached. Reactor Protection System Description N3-99-4003 Section 3.1.1.2(b) is similarly revised.

33 (page 7.2-34) Clarify compliance of the pressurizer pressure channels with the control and protection system interaction criteria. Reactor Protection System Description N3-99-4003 Section 3.1.1.2(c) is similarly revised.

34 (page 7.2-36): Clarify the role of the Median Signal Selector in eliminating the potential for SG water level control and protection system interaction.

35 (page 7.2-36): Clarify the function of the rod stop interlocks with respect to reactor control and protection. The revised text is more consistent with the rod stop discussion found in Section 7.7.1.4.

36 (Table 7.2-1 Sheet 1): Clarify the purpose of the P-7 permissive with respect to the RCP undervoltage trip for consistency with comments on similar features for other trip functions.

37 (Table 7.2-2 Sheet 1): Clarify the function of the P-10 permissive. It is also an input to P-7 as shown on Figure 7.2-1 Sheet 2. Reactor Protection System Description N3-99-4003 Table 1 is similarly revised.

38 (Table 7.2-2): This table lists the reactor trips and the various accident analyses for which each trip could provide protection. The intent of the table is to demonstrate the diversity of and comprehensive protection provided by the reactor trip system against various postulated events and to correlate the trip functions with the analyses in which they may be utilized, either as a primary or secondary protective function. Chapter 15, along with the Accident Analysis Parameters Checklist, WB-DC-40-70, provides the accident analysis discussion and identifies the protection system functions which provide accident mitigation. The additions and deletions to the table are made for consistency with the safety analyses of record as reflected in the design and licensing basis and do not represent analysis changes or protection system changes. Therefore, they are considered to be non-significant as discussed at the beginning of this section. Neutron Monitoring System Description N3-85-4003 Table 2 is also revised for consistency with WB-DC-40-70.

39 (Figure 7.2–1 Sheet 3): This drawing, 1-47W611-99-6, shows time delays of 0.5 and 0.1 seconds, respectively, for Reactor Coolant Pump undervoltage (UV) and underfrequency (UF) reactor trip signals. Setpoint and Scaling Documents specify settings of 23 cycles (0.383 sec) for the UV and 5 cycles (0.087 sec) for the UF as determined by calculations WBPE0689009007 and WBPE0689009008. This discrepancy is documented in WBPER980417. The drawing will be revised by DCN E-50045.

In addition to the UFSAR changes described above, Reactor Protection System Description N3-99-4003 is revised by EDC E-50045 to clarify the design basis and functions, correct minor errors, and make editorial changes to maintain consistency between the design basis and the UFSAR. Also reference 7.5.26 of N3-99-4003 is changed from WCAP-14419 to WCAP-14738, "Revised Thermal Design Procedure Instrument Uncertainty Methodology," to reflect the current design and licensing basis for the RCS flow and reactor power calorimetrics instrument uncertainty, which became effective at cycle 2 startup (reference DCN M-39293-A, FSAR change 1473, and TS Amendment 7). WCAP-14419 was superseded by WCAP-14738. These documents are not listed in the UFSAR or TS.

These changes do not involve any physical modifications to the plant or modify the safety function of any equipment. The changes do not alter any design basis accident or operational transient analyses previously performed, and no new accidents or equipment failure modes are created. The changes do not affect setpoints or safety limits and, thus, do not reduce any margins of safety as defined in the Technical Specifications. Therefore, it is concluded that the proposed change is acceptable from a nuclear safety standpoint and no unreviewed safety question exists.

B. Evaluation of Effects

B.1 May the proposed activity increase the probability of an accident previously evaluated in the SAR? Yes ☐ No ☒
Justification:

The proposed changes do not result in any changes in the design, material and construction standards of equipment important to safety. There are no hardware modifications or changes in system operating characteristics. The changes do not alter any design or safety functions. Therefore, the change does not affect any accident previously evaluated in the SAR and will not increase the probability of occurrence of any accidents.

B.2 May the proposed activity increase the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR? Yes ☐ No ☒
Justification:

The proposed changes do not involve any new or different type of equipment or modifications to existing equipment and will not create any additional or different failure modes. Therefore, the change will not affect the performance of any safety systems or increase the probability of a malfunction of equipment important to safety.

B.3 May the proposed activity increase the consequences of an accident previously evaluated in the SAR? Yes ☐ No ☒
Justification:

There are no hardware modifications associated with the proposed changes. They will not create any new limiting single failures or affect the accident mitigation capability of any systems or components. No fission product barriers will be affected. Thus, it is concluded that the consequences of an accident will not be increased.

B.4 May the proposed activity increase the consequences of a malfunction of equipment important to safety previously evaluated in the SAR? Yes ☐ No ☒
Justification:

The proposed changes do not require any physical modifications. No new failure modes or equipment malfunctions for safety related equipment are created. The changes will not degrade the performance of accident mitigation equipment or the integrity of fission product barriers. Therefore, no increased radiological consequences due to equipment malfunctions will result from the changes.

B.5 May the proposed activity create a possibility for an accident of a different type than any evaluated previously in the SAR? Yes ☐ No ☒
Justification:

The proposed changes do not affect the design or safety function of any plant equipment. There are no modifications associated with the proposed changes which could create any new limiting single failures of safety related equipment and there is no change in the response of accident mitigation equipment as assumed in the safety analyses. Therefore, the change cannot cause the initiation of any accident and the possibility of an accident of a different type will not be created.

B.6 May the proposed activity create a possibility for a malfunction of a different type than any evaluated previously in the SAR? Yes ☐ No ☒
Justification:

The proposed changes do not involve any new or different type of equipment, make any hardware modifications, or create any different failure modes or equipment malfunctions for accident mitigation equipment. No design requirements and performance specifications will be altered. Thus, there is no possibility for the creation of a different type of equipment malfunction than previously evaluated.

B.7 May the proposed activity reduce the margin of safety as defined in the basis for any Technical Specification?
Yes ☐ No ☒
Justification:

The changes do not alter any setpoints or other component design specifications, equipment or system operating limits, or safety analysis limits as given in the Technical Specifications. The change will not affect the performance of any systems or components required for accident mitigation and, therefore, will not reduce any margins of safety.

C. **Unreviewed Safety Question Determination Conclusion**

The change, test, or experiment:
Does not involve an unreviewed safety question. ☒
Involves an unreviewed safety question and must be revised, canceled, or reviewed by the NRC prior to implementation. ☐
Summarize why the activity does or does not constitute a USQ.

FSAR change 1553 documents changes resulting from a review of Section 7.2; design change EDC E-50045 makes related documentation changes identified during the SAR review. Corrections of non-numerical typographical errors, administrative changes, or other minor editorial changes that do not change the intent are considered to be non-significant changes as defined in procedure NADP-7, *FSAR Management*, and as such do not require a safety evaluation. The remaining items are as follows:

- Part of the discussion of the pressurizer water level instrumentation in Section 7.2.2.3.4 is moved to Section 7.2.1.1.5 since it is not specifically related to control and protection system interaction. The changes to this relocated discussion are based on a general description of the Westinghouse pressurizer level design, channel independence, and actual installation attributes found on TVA physical drawings. Also, the hydrogen gas entrainment issue documented in NRC Information Bulletin No. 92-54, Level Instrumentation Inaccuracies Caused by Rapid Depressurization, is retained and clarified. The original text in 7.2.2.3.4 provides some information that is too detailed and is not pertinent to the subject of discussion. It also includes a statement that the error effect on the level measurement during a blowdown accident would be about one inch. The basis for this value is not known; however, the worst case reference leg loss of fill error due to a rapid RCS depressurization event is no more than 12 inches elevation head. The value of 12 inches is based on the relative elevation difference between the condensing chamber and the reference leg sensor bellows. Instrument uncertainty calculations were based on the attributes found on the installation drawings and, therefore, the change to the UFSAR has no impact on the design basis or Technical Specifications and no protection system parameters such as setpoints or scaling are affected by the discrepancy. The remaining text in 7.2.2.3.4 is revised to clarify the control and protection system interaction discussion. The change does not require any plant modifications and is consistent with the existing design and licensing bases.

- Drawing 1-47W611-99-6 (Figure 7.2–1 Sheet 3) shows time delays of 0.5 and 0.1 seconds, respectively, for Reactor Coolant Pump undervoltage (UV) and underfrequency (UF) reactor trip signals. The Setpoint and Scaling Documents (SSDs) specify settings of 23 cycles (0.383 sec) for the UV and 5 cycles (0.087 sec) for the UF as determined by calculations WBPE0689009007 and WBPE0689009008. The values on the drawings were not used in establishing the settings and the actual settings are based on the SSDs. No plant modifications are required. The drawing will be revised by DCN E-50045.

These changes are documentation only and do not involve any physical modifications to the plant, modify the safety function of any equipment, or affect fission product barriers. The changes do not alter any design basis accident or operational transient analyses previously performed, and no new accidents or equipment malfunction failures are created. The changes do not affect setpoints or safety limits and, therefore, do not reduce any margins of safety as defined in the Technical Specifications. Therefore, it is concluded that the proposed change is acceptable from a nuclear safety standpoint and no unreviewed safety question exists.

D. **Reviews and Approvals**

	Name	Signature	Date
Preparer:	W. D. Webb	W D Webb	12-16-98
Reviewer:	DAN F. FAULKNER	Dan F Faulkner	12/16/98
Reviewer: (PORC)*	N/A	Not Significant No PORC Req'd	12-16-98
Other: Reviewers (as appropriate)			

*As required by Technical Specification