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TECHNICAL EVALUATION REPORT
REVIEW OF TOPICAL REPORT GENE-770-06-1
"BASES FOR CHANGES TO SURVEILLANCE TEST INTERVALS AND ALLOWED OUT-OF-SERVICE TIMES FOR SELECTED INSTRUMENTATION TECHNICAL SPECIFICATIONS"

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ABSTRACT

This EG&G Idaho, Inc. report provides a review of the subject topical report and subsequent submittals from the Boiling Water Reactors Owner's Group which present reliability analyses as the bases for increasing the surveillance test intervals and allowed out-of-service times for testing and repair for certain instrumentation systems included in the BWR4 and BWR6 Technical Specifications.

FOREWORD

This report is supplied as part of the review work performed for the Technical Specification Improvement Program under which the BWROG and G.E. supplied a series of topical reports which provide reliability centered bases for making changes to the Surveillance Test Intervals and Allowed Out-Of-Service Times for certain instrumentation systems included in the BWR4 and BWR6 Technical Specifications. This work is being done for the U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Division of Operational Events Assessment by EG&G Idaho, Inc., NRR&T - Washington Technical Office.

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I. INTRODUCTION

This topical report presents analyses intended to support changes in surveillance test intervals (STIs) and allowed out-of-service times (AOTs) for actuation instrumentation for selected systems for BWR 3,4,5,6 (Relay) and BWR 6 (Solid State) Plants. These systems include:

For the BWR4: Plant Systems Actuation Instrumentation, Main Control Room Environmental Control System (MCRECS), Safety/Relief Valves, and Safety/Relief Valves Low Low Set (LLS) Function.

For the BWR6: Plant Systems Actuation Instrumentation, Control Room Fresh Air Actuation Instrumentation (CRFA), Safety/Relief Valves, and Safety/Relief Valves Low Low Set (LLS) Function.

For the BWR4 and BWR6: EOC-RPT System Actuation Instrumentation, ATWS-RPT System Actuation Instrumentation, RCIC System Actuation Instrumentation and Control Rod Block Instrumentation.

The changes to STIs and AOTs proposed in this report are for actuation instrumentation for systems not covered in previous analyses of RPS, ECCS, and Containment Systems by the BWROG and are intended to be consistent with similar previously reviewed and accepted changes to the RPS, ECCS, and Containment System actuation instrumentation.

The procedure adopted by the BWROG in this report differs from that used in previous Technical Specification Improvement reports in that the systems were not directly modelled, but instead attempts were made to show similarity in components and function with previously reviewed and accepted actuation instrumentation. This procedure was applied to all the identified actuation instrumentation, however, the NRC staff required further, more detailed analysis be performed for the RCIC actuation instrumentation, therefore the RCIC analysis in the current report will not be evaluated or reported on here. This additional analysis is presented in a separate topical report (GENE-770-06-2) and its review reported on in a separate TER.

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II. PROPOSED CHANGES TO TECHNICAL SPECIFICATIONS

The changes to the technical specifications for the identified actuation instrumentation proposed in this topical report include:

(a) Change the Surveillance Test Interval (STI) from once every 31 days to once every 92 days.

(b) Change the Allowed Out-of-Service Time (AOT) for repair from 1 or 2 hours to 12 hours for equipment common to the RPS or 24 hours for all other equipment.

(c) Change the Allowed Out-of-Service Time (AOT) for test from 2 hours to 6 hours.

These proposed changes are the same as those previously approved for the RPS, ECCS, and Containment Isolation actuation instrumentation.

III. ACCEPTANCE CRITERIA

The general criteria for acceptance of the proposed changes to the actuation instrumentation technical specifications differ in this topical report from the criteria in the previous topical reports in this series. Instead of specific percentage limits on increases in system unavailability or failure frequency or specific absolute limits on those increases as before, in this topical report, the new criteria are that if these changes are made to instrumentation of similar type to that used in the RPS, ECCS, or containment isolation instrumentation which has been previously analyzed, then this previous analysis can be used to justify the proposed changes. This can be a reasonable procedure provided that equivalence in components, configuration, redundancy, and action performed is demonstrated in each case.

IV. EVALUATION OF BASES FOR STI AND AOT CHANGES

Topical Report GENE-770-06-1 presented the bases for STI and AOT changes in separate subsections of Section 3 for the various actuation systems. Therefore the evaluations of these bases are presented in this report as subsections which are numbered to correspond to the subsections in the topical report. Enclosure 1 presents a tabulation of the review results.

Section 3.1 - BWR4 Plant Systems Actuation Instrumentation

Changes requested include: Change the Channel Functional Test frequency from monthly to quarterly, and add a note establishing an allowed out-of-service time of 6 hours for surveillance testing. These proposed changes to the STI and the AOT for testing are consistent with similar changes to STIs and AOTs previously made and approved for other similar actuation instrumentation.

Evaluation of Basis For Change

a) The basis for including the Feedwater System/Main Turbine Trip in the technical specifications is that this trip is applicable only to plants which do not have a direct reactor trip on reactor vessel water level 8 signals. The trip is provided mainly as equipment protection against excessive moisture carryover into the main steam system and trips the reactor incidentally on main turbine trip. The trip has no safety significance because three other functions trip the reactor for this initiating event, thus the effect of losing this trip should be much less than the effect of losing the direct reactor vessel water level 8 reactor trip which was shown to be insignificant by the analysis in Ref. 1 (Page 7-6). We therefore agree that it is acceptable to justify the increase in STI and test AOT on the basis of this previous analysis.

b) This topical report shows actuation instrumentation for the Suppression Pool (and Drywell) Spray System consisting of Drywell Pressure--High; Containment Pool Pressure--High; Reactor Vessel Water Level--Low Low Low, Level 1; and Timers, Systems A & B. However, the basis states that the BWR4 design for this system is manually initiated and controlled and that there are no automatic functions or initiation instrumentation for this system. This appeared contradictory and we requested clarification from the BWROG. Their response confirmed that these systems are manually initiated and controlled for most BWR4 plants. However, should a plant have automatic initiation and control, it will be similar to that for the BWR6 design and the bases for STI and AOT changes will be as those for the BWR6 given in Section 3.7. The manual initiation and control statement in the basis and the response agrees with the Hatch 2 BWR4 plant design as an example. We conclude that these systems for most BWR4s are manually initiated

and controlled. However, for those plants which have these systems automatically initiated and controlled, the bases presented in Section 3.7 of the topical report as modified by the response to Question 7 may be used to justify STI and AOT changes to these systems.

Section 3.2 - BWR4 & BWR6 EOC - RPT Actuation Instrumentation

Changes requested include: Change the Channel Functional Test frequency from monthly to quarterly, change the allowed out-of-service time for surveillance testing from 2 to 6 hours, and change the allowed out-of-service time for repair from 1 hour to 12 hours. These proposed changes in STI and AOTs are consistent with similar changes previously made and approved to STIs and AOTs for other similar actuation instrumentation.

Evaluation of Basis For Change

The current topical report states that the End of Cycle-Recirculation Pump Trip (EOC-RPT) is initiated by the Turbine Stop Valve (TSV) Closure and Turbine Control Valve (TCV) Low Hydraulic Pressure signals which are common to the RPS and that the STI and AOT changes for these signals were analyzed in Ref. 1. It further states that although the EOC-RPT trip functions were not explicitly identified in that analysis, the changes to the STI and AOTs can be considered bounded by that analysis. We noted that the logic arrangement is significantly different for these initiating signals (two-out-of-two TSV closure and two-out-of-two for TCV low oil pressure). These two sets of signals are then "anded" with two channels of turbine first stage pressure > 30% RTP which act as a permissive. It was not evident that the analysis of Ref. 1 covered this logic configuration nor was it evident that the Ref. 1 analysis results bounded this case. Additional information was provided by the BWROG which described the instrumentation differences between the RPS and EOC-RPT functions together with calculations which compared the unavailabilities of the RPS and EOC-RPT functions. These comparisons indicated that the unavailabilities for the two functions were nearly the same, thus we now conclude that the proposed STI and AOT changes for the EOC-RPT are acceptable on this basis.

Section 3.3 - BWR4 & BWR6 ATWS-RPT Actuation Instrumentation

Changes requested include: Change the Channel Functional Test frequency and the trip unit calibration frequency from monthly to quarterly; change the allowed out-of-service time for surveillance testing from 2 to 6 hours; change the allowed out-of-service time for repair from 1 hour to 24 hours.

Evaluation of Basis For Change

We did not agree on the ATWS-RPT logic for the BWR4 Improved Standard Technical Specifications (ISTS) Lead Plant (Hatch 2) which is given in the

topical report as one-out-of-two per trip system with the tripping of both trip systems needed to trip both recirculation pumps. Our logic diagram as obtained from the Hatch 2 UFSAR showed a two-out-of-two logic per trip system on either high reactor pressure or low reactor vessel water level and trip of either trip system trips both pumps. After discussion with G.E., it appears that the newer BWR4 ATWS-RPT logic arrangement and that which is or will be installed on most BWR4 plants is the logic arrangement from the Hatch 2 UFSAR. The Hatch 1 plant was stated to have been modified to this newer design at the last outage and the utility indicated that Hatch 2 (the lead ISTS plant) will be so modified during the upcoming outage. This newer information should be made part of the topical report.

The current topical report stated that the effect of changes to STI and AOTs for ATWS-RPT instrumentation on the reactivity shutdown failure frequency is negligible because the RPS failure frequency is low (~5.4E-06/yr. from Ref. 1, page 5-29) and the change in overall ATWS-RPT function unavailability due to the STI and AOT changes (<1E-02/demand) is small. This indicated that even though the ATWS-RPT failure frequency may be higher, when it is combined with RTS, the combined effect is small. Additional information was provided by the BWROG which showed the calculation of the changes in ATWS-RPT channel unavailabilities and the small effect of these changes on the reactivity shutdown failure frequency when the test interval is extended from one to three months. Our review of this information confirmed the small effect of the ATWS-RPT STI and AOT changes on the reactivity shutdown failure frequency. Thus we conclude that the basis for these changes is acceptable.

The last sentence on page 6 of the topical report states that the same small change in ATWS-RPT unavailability due to STI and AOT changes can be expected for other logic designs. It was not clear how this could result since it would appear that the logic arrangement should influence the fault trees and their evaluation which means that a change in the logic configuration could result in a change in unavailability and hence failure frequency. Additional information provided by the BWROG indicated that the referenced statement was not intended to indicate that the reliability of a system is nearly independent of its logical arrangement, but that logic designs having similar degrees of redundancy could be expected to have similarly small changes in ATWS-RPT unavailability. This is a reasonable postulate and we conclude the statement as modified is acceptable.

Section 3.4 - BWR4 & BWR6 RCIC System Actuation Instrumentation

The analysis of the RCIC actuation instrumentation and the results of changing STIs and AOTs is more thoroughly presented in Ref. 7, therefore its review and evaluation is presented in a separate technical evaluation report.

Section 3.5 - <u>BWR4 Safety/Relief Valves and Safety/Relief Valves LLS Function Instrumentation</u>

Changes requested include: Change the Channel Functional Test frequency from monthly to quarterly and change the allowed out-of-service time for surveillance testing from 2 to 6 hours.

Evaluation of Basis For Change

Relief Function: For the BWR4, it is stated that there is no automatic initiation instrumentation associated with the relief function, therefore, there is nothing to review or report on.

LLS Function: We agree with the statement in the topical report that failure of the LLS function to initiate is not likely to have an immediate effect on the plant safety and therefore, that any small change in actuation instrumentation unavailability due to changes in the STI and AOTs may be neglected. However, the changes in the SRV LLS function actuation instrumentation unavailability caused by changes in the STI and AOTs had not been shown to be small enough to neglect. The argument was made that the reasoning which supports making the requested changes was bounded by the analyses performed for the ECCS and isolation actuation instrumentation and further that the analyses should apply because the components used in the systems were the same or similar and the number of failures needed to disable the system were comparable. We found this argument, as presented, was not acceptable because it failed to consider the manner in which those components are assembled into the LLS logic. For example, the LLS logic is an armed logic which is not similar to any of the ECCS or isolation actuation logic. Some of the subsystems may be similar but the manner in which they are organized into the logic is different, therefore the fault trees and their evaluation may differ and the results may not be bounded by the previous analyses.

In response to our Question Four, the BWROG submitted additional information consisting of fault trees and evaluations for the BWR4 LLS logic. In our review of this information, we agree that four of the eleven S/RVs can be actuated by the LLS logic and that only one of the four needs to open to fulfill the LLS function. We did, however, question whether the channel logic used for the fault trees followed the channel logic as presented, for example, in the Hatch, Unit 2, BWR4 plant. Our review showed two LLS logic channels in each Division: Channels A&C for Division I and channels B&D for Division II with each channel actuating one of the four LLS valves as follows. A permissive relay receives a signal on reactor vessel high pressure from PT1(x)[N120(x)]. These relays are assigned as follows: Channel A-K340A, Channel C-K370A, Channel B-K340B and Channel D-K370B. The contacts of each such permissive relay are wired in series with the coil of the arming relay and the paralleled contacts of the tail pipe switches associated with that channel. The arming relays are: Channel A-K313A, Channel C-K314A, Channel B-K313B and Channel D-K314B. Actuation of an arming relay seals it in through a set of auxiliary contacts. Each of the 11 S/RV tailpipes is equipped with two pressure switches which senses actuation of the S/RV. One of these switches

from each tailpipe is assigned to one trip system and the other to the other trip system. One channel in each trip system receives input from five of these pressure switches and the other receives input from the remaining six tailpipe pressure switches. Arming of one channel in a trip system provides an arming signal to the other channel in the trip system through auxiliary contacts on its arming relay. The reactor high pressure inputs are derived from two pressure transmitters per channel, PT1(x)[N120(x)] for the permissive and arming function and PT2(x)[N122(x)] to control the S/RV opening and closing. The two-out-of-two logic per channel will open the S/RV assigned to that channel at its LLS setpoint once the channel is armed. Fault trees based on this logic were constructed by us to compare with those submitted by the BWROG. Initially the two sets of fault trees did not agree, principally with respect to the interpretation of the Hatch 2 logic configuration. After further contact and discussion, comparability between the two sets of trees was established.

Evaluation of the fault trees showed the change in unavailability due to the change in surveillance intervals from one to three months was 3.92 E-05 per demand which is relatively small. Because the change in unavailability is small, the changes in surveillance interval and allowed out-of-service times for testing are comparable to those for other similar systems, and failure of the LLS function does not have an immediate direct impact on plant safety, we conclude these changes are acceptable for this system.

Section 3.6 - BWR6 Safety/Relief Valves and Safety/Relief (S/RV) Low Low Set (LLS) Function Instrumentation

Changes requested include: Change the Channel Functional Test and the trip unit calibration frequencies from monthly to quarterly and add a note establishing the allowed out-of-service time for surveillance testing as 6 hours.

Evaluation of Basis For Change

LLS Function: The same arguments were made for the BWR6 LLS actuation instrumentation as were made for the BWR4 LLS actuation instrumentation, therefore the same concerns that were identified in the comments for the BWR4 should also apply for the BWR6 case. In addition, since the same initiating instrumentation may be used to actuate both the Safety/relief and LLS functions, single failures could possibly result in the loss of actuation inputs to both the LLS and relief functions.

The additional information supplied by the BWROG in response to our Question Five addressed our concerns as follows. The information provided a more complete description of the LLS logic and supplied the results of the fault tree evaluation. There are three groups of LLS valves. The high group, consisting of four valves: F047D&G and F051A&F; the mid group, consisting of F051B; and the low group consisting of F051D. Each group is sealed-in or armed by a one-out-of-three-twice logic per division. After arming, the high group of four S/RVs is opened on two-out-of-two reactor vessel high pressure

per division and the mid and low groups of one S/RV each are opened on one-out-of-one reactor vessel high pressure per division. One-out-of-two divisions is required for both seal-in and S/RV actuation. The changes in unavailability caused by the increased surveillance intervals and the AOT were calculated and shown to be 4.48 E-08 for one logic group and negligibly small for the one-out-of-three logic groups. Therefore, since the changes in unavailability are small, the failure of the LLS mode does not have a direct impact on plant safety, and the proposed changes in the surveillance intervals and AOTs are comparable to those for similar safety systems, we conclude they are acceptable.

Relief Function: Some concerns with the statements in the last paragraph on page 14 of the topical report arose. First, in the level of redundancy argument, it was not clear just what was being taken as redundant. The three actuation logic sets did not appear to be entirely redundant because in two out of three cases, two of the logic sets plus part of the third were needed to open 13 out of the 20 relief valves required to preclude reactor overpressurization. Second, it was not clear how it is known that the relief actuation function is a small contributor to the overall S/RV function unavailability. Third, it was not clear what the term "overall S/RV function unavailability" referred to, nor how it was determined. Fourth, it was not clear how it has been determined that changes in STI and AOTs for the S/RV actuation logic result in small contributions to the S/RV actuation logic unavailability. A previously reviewed actuation instrumentation system or subsystems had not been identified which this actuation instrumentation system or its subsystems could be compared against. Also, an analysis or calculation of the actuation instrumentation system unavailability did not appear to have been performed. Therefore, there originally appeared to be no basis for accepting the statement that the contributions to unavailability caused by increases to STIs and AOTs are insignificant and therefore acceptable.

Additional information supplied by the BWROG in response to our Question Six addressed the definition of what was redundant by showing that the ASME Code allows the relief (logic actuated) and safety (spring) modes of actuation to be taken as redundant. This information also included fault trees and their evaluation to show the changes in unavailabilities which occurred when the surveillance intervals were changed from one to three months. The changes in unavailability were found to be 2.37 E-03 for the relief mode and 7.14 E-06 for the relief and safety mode which are relatively small, and the changes in surveillance intervals and allowed out-of-service times for testing are comparable to those granted for other similar systems. We therefore conclude the change in surveillance intervals from one to three months and the inclusion of a six hour AOT for testing are acceptable.

Section 3.7 - BWR6 Plant Systems Actuation Instrumentation

This section contains three subordinate systems: The RHR Containment Spray System instrumentation, the Feedwater and Main Turbine Trip System instrumentation, and the Suppression Pool Makeup System instrumentation. Each of these systems will be addressed independently.

a) RHR Containment Spray Instrumentation

Changes requested include: Change the Channel Functional Test and the trip unit calibration frequencies from monthly to quarterly, the allowed out-of-service time for repair from 1 hour to 24 hours, and the allowed out-of-service time for surveillance testing from 2 hours to 6 hours. These proposed changes are consistent with those proposed and accepted for other BWR6 actuation instrumentation.

Evaluation of Basis For Change

The actuation instrumentation performs functions similar to those performed by the isolation actuation instrumentation, therefore since the configuration is similar to that for the isolation actuation instrumentation, we agree that the analyses performed for that instrumentation can be extended to this instrumentation also. Our review of the referenced analyses showed only a statement that valve unavailability is the dominant contributor to system unavailability. We were unable to confirm the dominance of the valve unavailability because information on this unavailability analysis and its results was not presented. Additional information was provided by the BWROG in response to our Question Seven which included a more complete system description, a fault tree for the initiation system, and calculation of the increase in unavailability caused by the increases in surveillance interval and AOT. The increase in unavailability for one-out-of-two trip systems is a relatively small 1.45 E-04 for the most conservative case and the valve opening and closing is shown to be the largest contributor to the individual trip system unavailability. On the basis of this information and the actuation system's similarity to those previously reviewed and accepted for the Emergency Core Cooling and Containment Isolation Systems, we conclude the proposed STI and AOT changes are acceptable.

b) Feedwater and Main Turbine Level 8 Trip Instrumentation

Changes requested include: Change the Channel Functional Test frequency from monthly to quarterly and the allowed out-of-service time for surveillance testing from 2 hours to 6 hours. These proposed changes are consistent with those proposed and accepted for other BWR6 actuation instrumentation.

Evaluation of Basis For Change

The actuation instrumentation for the Feedwater/Main Turbine Level 8 Trip uses output from level transmitters that are similar to those for the RPS Level 8 trip units. These compose a two-out-of-three logic for a single trip system. The single trip system does not meet the single failure criterion but because BWR6 plants have a direct reactor trip on Reactor Vessel Water Level--High, Level 8

that is part of the RPS, the feedwater and main turbine trip on high reactor vessel water level 8 does not serve a safety function, but is for equipment protection only. It was not clear however, that the RPS Level 8 trip analysis bounds the feedwater and main turbine trip because the feedwater and main turbine trip instrumentation is a single trip system using a two-out-of-three logic where the RPS Level 8 trip is a one-out-of-two per trip system with both trip systems required for trip. Additional information describing the system and calculating the increase in system unavailability caused by increasing the surveillance interval to three months and the AOT from two to six hours was provided by the BWROG. The most conservative calculations using design analysis failure rates show an increase in single trip system unavailability of 1.8E-03. Using more realistic current experience failure rates shows an increase of 1.2E-04. These values for single system unavailability are small enough to conclude the requested increases in surveillance interval and AOT are acceptable.

c) Suppression Pool Makeup System Instrumentation

Changes requested include: Change the Channel Functional Test and the trip unit calibration frequencies from monthly to quarterly, the allowed out-of-service time for repair from 1 hour to 24 hours, and the allowed out-of-service time for surveillance testing from 2 hours to 6 hours. These proposed changes are consistent with those proposed and accepted for other BWR6 actuation instrumentation.

Evaluation of Basis For Change

The actuation instrumentation for the Suppression Pool Makeup System (SPMS) does perform functions similar to those performed by the isolation system actuation instrumentation. However, the initiation and actuation logic configuration for this system appears to be quite different from that in the isolation actuation logic referred to. Our review of the referenced analyses shows only a statement that valve unavailability is the dominant contributor to system unavailability. Additional information was provided by the BWROG in response to our Question Seven which included a more detailed system description which showed similarity to the LPCS ECCS initiation logic and certain primary/secondary isolation valve logic and calculation of the increase in SPMS unavailability caused by the increases in surveillance interval and AOT. Using the more conservative design failure rates gives a change in system unavailability of 2.0E-05 in going from one month to three month surveillance intervals which is small enough to conclude the requested increases in surveillance interval and AOT are acceptable.

Section 3.8 - BWR4 Main Control Room Environmental Control System (MCRECS) Actuation Instrumentation

Changes requested include: Change the Channel Functional Test frequency from monthly to quarterly, change the allowed out-of-service time for repair from 2 hours to 24 hours, and change the allowed out-of-service time for surveillance testing from 2 hours to 6 hours. These changes are consistent with those proposed and accepted for other BWR4 actuation instrumentation.

Evaluation of Basis For Change

The reference given for the reactor vessel water level 1, high drywell pressure, and main steam line high flow inputs to the MCRECS was not acceptable because it did not appear to provide analyses against which to compare the reactor vessel water level 1 and main steam line high flow inputs. Further, in the basis, such a comparison was not made and initiation of MCRECS was not addressed.

The current basis should also provide additional information on how the high control room and the refueling floor area radiation functions are similar to the reactor building exhaust high radiation function to enable judgement to be made as to whether the fault trees and their evaluation are applicable to the high control room and the refueling floor area radiation functions. Additional information provided by the BWROG in response to our Question Eight addressed our concerns by providing, for each of six initiating events, the applicable logic configuration per subsystem, references to previously reviewed actuation instrumentation, and logic diagrams. Each subsystem was shown to be similar to a previously analyzed system, thus the change in unavailabilities previously calculated for that system should also apply to this system. References were also provided to similar previously reviewed systems for the Refueling Floor Area High Radiation, Control Room Inlet High Radiation, and Control Room Inlet High Chlorine Level Function actuation instrumentation. The small increases in unavailability previously calculated for the referenced systems were found acceptable for justifying similar changes in surveillance intervals and AOTs for those systems. Therefore, by similarity, they will also be acceptable justification for the requested similar changes in surveillance intervals and AOTs for the MCRECS actuation instrumentation.

Section 3.9 - BWR6 Control Room Fresh Air (CRFA) Actuation Instrumentation

Changes requested include: Change the Channel Functional Test and the trip unit calibration frequencies from monthly to quarterly and the allowed out-of-service time for surveillance testing from 2 hours to 6 hours. These changes are consistent with those proposed and accepted

for other BWR6 actuation instrumentation.

Evaluation of Basis For Change

The reference given for the reactor vessel water level 2 and the high drywell pressure inputs to the CRFA system in Ref. 5, page 5-11 was not satisfactory because it did not present any information on these two trip inputs and it presented no information which would indicate that the CRFA system initiation instrumentation was included in the analysis. The secondary containment statement mentions only the fuel handling area ventilation exhaust high radiation and the pool sweep exhaust radiation inputs to the secondary containment isolation actuation system. No fault trees or other analysis is presented in that reference for initiation of secondary containment isolation or isolation of the MCR and initiation of the CRFA system. Additional information provided by the BWROG in response to our Question Nine addressed the lack of information in the topical report for the Control Room Fresh Air (CRFA) actuation instrumentation by confirming that the same isolation logic which isolates the secondary containment isolation valves also actuates the CRFA system. Also provided were logic diagrams that identify the equipment and logic configurations for the subsystems which actuate the CRFA system. These logic configurations have been previously analyzed in similar systems for the secondary containment isolation system and found acceptable for justifying similar requested changes in surveillance intervals and AOTs for that actuation instrumentation. Therefore, by similarity, the results will also be acceptable justification for the requested similar changes in surveillance intervals and AOTs for the CRFA system actuation instrumentation.

Section 3.10 - <u>BWR4 and BWR6 Control Rod Block Instrumentation</u>

Changes requested include: Change the allowed out-of-service time for repair in Table 3.3.6-1, Items 5 and 6 from 1 hour to 12 hours and establish the allowed out-of-service time for surveillance testing at 6 hours by adding a note to Surveillance Requirement 4.3.6. These requested changes are consistent with similar changes proposed and accepted for other BWR4 and BWR6 actuation instrumentation that is common to the RPS and ECCS actuation instrumentation.

Evaluation of Basis For Change

The basis for justifying these requested changes is stated to be included as part of the basis for changing the STI given in Ref. 2. We reviewed this reference and found that it does not explicitly address the extension of the AOTs. However, in the case of the SDV level rod block, the same type of instrumentation is used to provide the rod block as is used to provide the RPS scram signal. From Ref. 1, the effect on core damage frequency of changing the repair AOT to 12 hours and the surveillance testing AOT to 6 hours was found to be negligible for the RPS. Since the rod block instrumentation is similar in type and

configuration to that used for the RPS trip, it is reasonable to assume that the increase in rod block unavailability caused by the increases in the AOTs for the SDV level input is also negligible. On this basis we conclude the requested changes are acceptable.

For the RCS Recirculation Flow sensors, the increase in AOTs for test and repair was not explicitly addressed. However, in Ref. 1, increasing the STI from 31 to 92 days does not result in a significant increase in the APRM Flow-Biased Neutron Flux scram unavailability and it is further stated that analyses indicated that increasing the test AOT to 6 hours and the repair AOT to 12 hours would produce an even smaller effect on the system unavailability. Therefore, if these increases are acceptable for the signal input to the flow-biased APRM trip, similar test and repair AOT increases should also be acceptable for the same type of input to the rod block instrumentation since it should produce a correspondingly insignificant effect on the rod block instrumentation unavailability. On this basis we conclude the requested changes are acceptable.

V. REFERENCES

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2. S. Visweswaran, et al., "Technical Specification Improvement Analysis for BWR Control Rod Block Instrumentation," General Electric Company, October 1988 (NEDC-30851P-A, Supp. 1).

3. D. B. Atcheson, C. Ha, C. L. Larson, R. P. Raftery, W. P. Sullivan, "BWR Owners' Group Technical Specification Improvement Methodology (With Demonstration for BWR ECCS Actuation Instrumentation) Part 2," General Electric Company, December 1988 (NEDC-30936P-A).

4. L. G. Frederick, et al., "Technical Specification Improvement Analysis for BWR Isolation Instrumentation Common to RPS and ECCS Instrumentation," General Electric Company, March 1989 (NEDC-30851P-A, Supp. 2).

5. W. P. Sullivan, et al., "Technical Specification Improvement Analysis for BWR Isolation Actuation Instrumentation," General Electric Company, July 1990 (NEDC-31677P-A).

6. D. B. Atcheson, L. G. Frederick, W. P. Sullivan, P. T. Tran, "BWR Owners' Group Technical Specification Improvement Methodology (With Demonstration for BWR ECCS Actuation Instrumentation) Part 1," General Electric Company, December 1988 (NEDC-30936P-A).

7. W. P. Sullivan, "Addendum to Bases For Changes to Surveillance Test Intervals And Allowed Out-Of-Service Times For Selected Instrumentation Technical Specifications," General Electric Company, GENE 770-06-2, February 1991.

8. W. P. Sullivan, "BWROG Response to NRC Questions On Topical Report GENE-770-06-1," General Electric Company, December 1991 (BWROG-91157).

Enclosure 1
REVIEW OF TOPICAL REPORT GENE 770-06-1
TABLE OF RECOMMENDATIONS
FOR PROPOSED CHANGES

System	Appl	CFT Surv. Int.			Surv. Test AOT			Repair AOT		
		Curr	Prop	Rec. Act.	Curr	Prop	Rec. Act.	Curr	Prop	Rec. Act.
FW/MTT	BWR4	31d	92d	Accept.	2 hr	6 hr	Accept.	NR	NR*	NR*
SP&DW SPR	BWR4	N/A**								
EOC-RPT	Both	31d	92d	Accept.	2 hr	6 hr	Accept.	1 hr	12 hr	Accept.
ATWS-RPT	Both	31d	92d	Accept.	2 hr	6 hr	Accept.	1 hr	24 hr	Accept.
RCIC Inst	Both	N/A))								
S/R Valves	BWR4	N/A**								
S/R Valves	BWR6	31d	92d	Accept.	2 hr	6 hr	Accept.	NR	NR*	NR*
LLS	BWR4	31d	92d	Accept.	2 hr	6 hr	Accept.	NR	NR*	NR*
LLS	BWR6	31d	92d	Accept.	2 hr	6 hr	Accept.	NR	NR*	NR*
RHR Con Sp	BWR6	31d	92d	Accept.	2 hr	6 hr	Accept.	1 hr	24hr)	Accept.
FW/MTT	BWR6	31d	92d	Accept.	2 hr	6 hr	Accept.	NR	NR*	NR*
SPMU Sys	PWR6	31d	92d	Accept.	2 hr	6 hr	Accept.	1 hr	24hr)	Accept.
MCREC	BWR4	31d	92d	Accept.	2 hr	6 hr	Accept.	2 hr	24 hr	Accept.
CRFA	BWR6	31d	92d	Accept.	2 hr	6 hr	Accept.	NR	NR*	NR*
CR Block	Both	92d	NR	Accept.	2 hr	6 hr	Accept.	1 hr	12 hr	Accept.

* NR = Not Requested in topical report
** N/A = No Auto initiation involved
) = If taking channel for Surveillance does not cause loss of functions, otherwise no change.
)) = Refer to Review of T.R. GENE 770-06-2

BWROG'S RESPONSE TO NRC QUESTIONS

ON TOPICAL REPORT GENE-770-06-1

"BASES FOR CHANGES TO SURVEILLANCE TEST INTERVALS AND ALLOWED OUT-OF-SERVICE TIMES FOR SELECTED INSTRUMENTATION TECHNICAL SPECIFICATIONS", FEBRUARY 1991

BWROG'S RESPONSE TO NRC QUESTIONS ON TOPICAL REPORT GENE-770-06-1

Reference:

1) Nine (9) questions from Don Lasher, EG&G, Faxed May 13, 1991.

2) GENE-770-06-1, "Bases for Changes to Surveillance Test Intervals and Allowed Out-Of-Service Times for Selected Instrumentation Technical Specifications", February 1991.

3) NEDC-30851P-A, "Technical Specification Improvement Analysis for BWR Reactor Protection System", March 1988.

4) NEDC-30936P-A, "Technical Specification Improvement Methodology (With Demonstration for BWR ECCS Actuation Instrumentation, Part 2", December 1988.

5) NEDC-30936P-A, "Technical Specification Improvement Methodology (With Demonstration for BWR ECCS Actuation Instrumentation, Part 1", December 1988.

6) NEDC-31677P-A, Technical Specification Improvement Analysis for BWR Isolation Actuation Instrumentation", July 1990.

QUESTION 1

For the Suppression Pool (and Drywell) Spray System (BWR4), the basis presented appears contradictory in that actuation instrumentation consisting of Drywell Pressure-High; Containment Pool Pressure-High (I don't know where this comes from); Reactor Vessel Water Level-Low Low-Low, Level 1; and Timers, Systems A&B is listed but in the Basis it is stated that the BWR design for this system is manually initiated and controlled and that there are no automatic functions or initiation instrumentation. Can you clarify and correct this apparent contradiction?

RESPONSE TO QUESTION 1

In the proposed BWR 4 Improved Standard Technical Specification (ISTS), the suppression pool and drywell spray auto initiation signals have been removed (See Enclosure 1 for reason for removal). The only remaining auto initiation signal from Table 3.3.9-1 that will appear in the BWR 4 ISTS is for the Feedwater System/Main Turbine Trip System. If a plant should have suppression pool and drywell spray auto initiation similar to that in a BWR 6 plant, the bases for STI and AOT changes are as provided in Section 3.7 for BWR 6 plants.

QUESTION 2

For the EOC-RPT (BWR4 and BWR6), we need more specific reference to the system or subsystems in NEDC 30851P because it appears that the logic arrangement is significantly different for these initiating signals (two-out-of-two TSV closure and two-out-of-two TCV Low Oil Pressure). These signals are "anded" with two channels of Turbine 1st Stage Pressure >30% RTP. It is not evident that the analysis of NEDC 30851P

covered this particular logic configuration or that the results of that analysis bound this case.

RESPONSE TO QUESTION 2:

The logic for the turbine control valve (and turbine stop valve) trip for RPS-Scram and End of Cycle (EOC)-Recirculation Pump Trip (RPT) is presented in Enclosure 2. The same sensors and logic relays perform both the RPS-Scram and EOC-RPT functions. The dual trip functions are achieved by different contacts in the logic relays. The RPS-Scram logic is 1 out of 2 channels twice required for trip. For the turbine stop trip function, two signals from individual stop valve position switches are required to trip an individual channel. For the turbine control valve closure, one sensor input is required to trip an individual channel. A trip circuit bypass is provided in each channel for reactor operation below 30% power. For reactor operation above 30% power, a relay contact in each channel opens. If a bypass relay fails to transfer out of the bypass position, the failure is annunciated in the control room. Therefore, a failure to remove the bypass would be detected within a short time period.

The EOC-RPT logic is 2 out of 2 channels required to trip an individual logic Division. Each of the two logic Divisions trip both recirculation pumps. Failure of the EOC RPT function therefore requires a failure of both logic Divisions. Each logic Division has contacts from two of the bypass relays. These contacts are in the closed position when reactor power is above 30% (trip function not bypassed). This is different than the bypass relay contacts in the RPS trip circuit which are open during operation above 30% power. As with the RPS trip circuit, a failure to remove the bypass would be detected within a short time period.

A calculation of the unavailability of the two trip functions of each trip logic is provided in Enclosure 2. The change in EOC-RPT trip function unavailability when the surveillance interval is extended from 1 to 3 months is lower for the turbine stop valve trip function and slightly higher for the turbine control valve trip function than the same trip functions for RPS-Scram. However, the small increase in EOC-RPT unavailability (represented by small increase risk of an MCPR violation) is judged to be offset by the benefits associated with the similar approved STI and AOT changes for the RPS-Scram function. Therefore, it can be concluded that the STI and AOT changes for EOC-RPT trip function is bounded by the approved RPS analysis.

QUESTION 3

For the ATWS-RPT (BWR4) as we discussed over the phone, the newer BWR4 (Hatch) logic configuration and the one you indicated is implemented for most BWR4s, is the two-out-of-two per trip system on either reactor high pressure or reactor vessel water level low with trip of either trip system tripping both Recirculation Pumps.

The topical report states that the effect of changes to STI and AOTs for ATWS-RPT instrumentation on the reactivity shutdown failure frequency is negligible because the RPS failure frequency is low (5.4E-06/yr from NEDC 30851P, page 5-29) and the change in overall ATWS-RPT function unavailability due to the STI and AOT changes (<1E-02/demand) is small. This appears to say that even though the ATWS-RPT failure frequency may be higher, when it is combined with that for the RPS, the combined effect is small. It is not clear how this combination is made or how the numbers were calculated. Can you supply some additional information explaining how these numbers were arrived at and how they were combined with the RPS?

The last sentence on page 6 of the report seems to state that the same small change in ATWS-RPT unavailability due to STI and AOT changes can be expected for other logic designs. It is not clear how this can result since it would appear that the logic arrangement should have a large influence on the fault trees and their evaluation which means a change in the logic configuration could result in a large change in unavailability and hence failure frequency. This would indicate that the reliability of a system is nearly independent of its logical arrangement. Please clarify this concern.

RESPONSE TO QUESTION 3:

It was stated in Reference 2 that the trip logic for ATWS-RPT for the Hatch2 lead ITS plant is one out of two channels per trip system for each trip function. Both trip systems are required to trip the two recirculation pumps. This was the configuration at the time of the analysis. The ATWS-RPT configuration that is being installed during the current Hatch-2 refueling outage has the same type logic as the BWR 6 lead plant.

The ATWS-RPT logic for the BWR 6 lead plant (currently being installed for the BWR 4 lead plant) is two out of two channels required to trip each trip system for each trip function (low water level and high reactor pressure). It was stated in Reference 2 that the change in unavailability when the surveillance interval is extended from 1 to 3 months is small (< 1E-02/demand). Calculation of this unavailability change is presented in Enclosure 3. Unavailabilities are calculated using trip unit failure rates from original design analyses and failure rates that reflect more current operating experience. The calculated change in unavailabilities for each trip function is approximately 2E-03/demand based on failures rates from the original design analysis and 2E-04/demand based on failure rates reflecting current operating experience.

The relative effect of the ATWS-RPT unavailability change on the reactivity shutdown failure frequency is also shown in Enclosure 3. Assuming only one ATWS-RPT trip function (low water level or high reactor pressure) and the RPS failure frequencies calculated in Reference 3, the relative change in reactivity shutdown failure frequency when the ATWS-RPT channel functional test is extended from 1 to 3 months is negligible (<2.0E-08/year). This calculation does not include other ATWS-RPT components which are not part of the channel

functional tests (e.g., circuit breakers). However, the model provides adequate indication of the small relative effect of the surveillance interval extension. This negligible change in reactivity shutdown failure frequency is offset by the benefits from reduced inadvertent scrams which is discussed in Reference 3.

It is also stated in Reference 3 that a same small change in ATWS-RPT unavailability due to STI and AOT extensions can be expected at other BWR plants. This does not imply that the reliability of the ATWS-RPT is independent of logic arrangement. However, the logic designs of the different BWR plants can be expected to have redundancy comparable to the above analyzed design based on the specified design requirements for ATWS-RPT. A small change in ATWS-RPT unavailability can be expected if comparable redundancy exists.

QUESTION 4

For the BWR4 LLS function, it is stated that failure of the LLS function to initiate will not have an immediate adverse effect on the plant safety and therefore, that any small change in actuation instrumentation unavailability due to changes in the STI and AOTs may be neglected. However, the changes in the SRV LLS function actuation instrumentation unavailability caused by changes in the STI and AOTs have not been shown to be small enough to neglect. The argument is made that the reasoning which supports making the requested changes is bounded by the analyses performed for the ECCS and isolation actuation instrumentation and further that the analyses apply because the components used in the systems are the same or similar and the number of failures needed to disable the system are comparable. This argument as presented is unacceptable because it appears to fail to consider the manner in which those components are assembled into the LLS logic. For example, the LLS logic is an armed logic which is not similar to any of the ECCS or isolation actuation logic. Some of the subsystems may be similar but the manner in which they are organized into the logic is different, therefore the fault trees and their evaluation may differ and the results may not be bounded by the previous analyses. This has not been shown and neither has it been shown that the number and type of failures needed to disable the LLS actuation instrumentation is comparable to that for an identified previously reviewed ECCS or isolation actuation system. The topical report should be revised to demonstrate the similarity of this logic system to a specific reference logic system or systems.

RESPONSE TO QUESTION 4:

Enclosure 4 provides a fault tree of the low-low set (LLS) for the Hatch 2 plant. There are 11 safety/relief valves (S/RVs) at the Hatch 2 plant. Four of these are actuated by the LLS. Failure of the LLS function requires failure of all 4 LLS valves. Each LLS valve is armed by a relay in a logic channel initiated by a high reactor pressure trip unit and one of the S/RV tail pipe pressure switches. Two of the valve arming logic channels receive trip signals from 5 tail pipe pressure switches. The remaining two arming logic channels receive trip signals from 6 tail pipe pressure switches. Each of the 11 S/RV tail pipes

have two pressure switches. After arming, each LLS valve is actuated by two out of two relays each initiated by a high reactor pressure trip unit. There are a total of 22 individual pressure switches (2 per S/RV) and and 12 individual reactor pressure trip units.

A calculation of the LLS unavailability as a function of surveillance interval is given in Enclosure 4. Two LLS valves were assumed to open. The change in LLS unavailability when the surveillance interval is extended from 1 to 3 months is 3.92E-05/demand. This change is acceptably low based on the function and failure consequences of the LLS (i.e., reduce the number of load cycles on the containment and S/RV discharge lines and reduce the number of S/RV actuations during the plant lifetime).

QUESTION 5

For the BWR6 LLS function, the same arguments are made for the actuation instrumentation as were made for the BWR4 LLS actuation instrumentation, therefore the comments made for the BWR4 case apply for the BWR6 case also. In addition, since the same initiating instrumentation may be used to actuate both the relief mode and the LLS mode, it appears that single failures could result in the loss of actuation inputs to more than one relief function. These concerns should be addressed to first demonstrate the similarity of the LLS actuation instrumentation logic to an identified previously reviewed ECCS or isolation actuation system logic and second to analyze the apparent condition in which a single failure could result in the loss of actuation inputs to more than one relief function.

RESPONSE TO QUESTION 5:

Enclosure 5 provides information on the LLSL for BWR 6 Grand Gulf plant. There are a total of 20 S/RVs of which 6 perform the LLS function. The seal in logic is the same for all 6 LLS valves (i.e., 1 out of 3 twice/Division). The LLS logic is 2 out of 2 per Division for one logic group controlling 4 LLS valves and 1 out of 1 per Division for the remaining two logic groups which each control one LLS valve. One out of 2 Divisions are required to initiate the LLS function for both the seal in logic and LLS logic. There are a total of 12 individual reactor pressure trip units for the seal in logic (6/Division) and 8 individual reactor pressure trip units for the LLS logic (4/Division).

The unavailability as a function of the test interval was calculated at the LLS logic channel level. The change in LLSL unavailability for a single logic group when the surveillance interval is extended from 1 to 3 months is 4.5E-08/demand. The change in LLSL unavailability for 1 out of 6 LLS valves (or 1 out of 3 logic groups) is negligible. This change is acceptably low based on the function and failure consequences of the LLSL (i.e., reduce the number of load cycles on the containment and S/RV discharge lines and reduce the number of S/RV actuations during the plant lifetime).

QUESTION 6

For the BWR6 Relief function, several concerns with statements in the last paragraph on page 14 need to be addressed in the topical report. First, in the level of redundancy argument, it is not clear just what is being taken as redundant. The three actuation logic sets are not entirely redundant because in two out of three cases, two of the logic sets plus part of the third are needed to open 13 out of the 20 relief valves required to preclude reactor overpressurization. [The safety mode may not be truly redundant because of the different (higher) opening pressures.] Second, it is not clear how it is known that the relief actuation function is a small contributor to the overall S/RV function unavailability. Third, it is not clear what "overall S/RV function unavailability" refers to, nor how it is determined. Fourth, it is not clear how it has been determined that changes in STI and AOTs for the S/RV actuation logic result in small contributions to the S/RV actuation logic unavailability. A previously reviewed actuation instrumentation system or subsystems have not been identified which this actuation instrumentation or its subsystems can be compared. An analysis or calculation of the actuation instrumentation system unavailability does not appear to have been performed. We do not appear to have a firm basis for accepting the statement that the contributions to unavailability caused by STI and AOT increases are insignificant and therefore, are acceptable.

RESPONSE TO QUESTION 6:

For the BWR 6 lead plant 13 of the 20 S/RVs are required to prevent reactor overpressure. The S/RVs are designed to limit the primary system pressure, including transients, to the requirements of the ASME Section III. The ASME code allows one-half of the S/RVs to open in the safety (spring) mode and one-half to open in the pressure relief mode.

Enclosure 6 provides a description of the pressure relief logic. One group of 9 S/RVs are actuated by 1 out of 2 logic Divisions. Each Division is actuated by tripping 2 out of 2 reactor pressure trip unit channels. A second group of 10 S/RVs are tripped by another set of reactor pressure trip units using a similar logic configuration. One S/RV is tripped by a third set of reactor pressure trip units. As discussed in response (5), 6 of the 20 S/RVs also have a LLS function in addition to the pressure relief function. The LLS seal in logic uses the same trips units as the pressure relief logic. Because of this commonality, no credit was taken for the LLS relief mode when calculating the pressure relief function unavailability.

A fault tree model of failure of the S/RV relief and safety mode at the logic channel level is provided in Enclosure 6. The unavailability was calculated using trip unit failure rates from original design analyses and failure rates that reflect more current operating experience. The calculated change in relief and safety mode unavailabilities when the surveillance test interval is extended from 1 to 3 months is approximately 7E-06/demand based on failures rates from the original design analysis and 4E-08/demand based on failure rates reflecting

current operating experience. This increase in calculated unavailability is insignificant and is offset by the similar benefits of reduced testing discussed in Section 4.2 of the ECCS Actuation Instrumentation Analysis, Reference 4.

QUESTION 7

For the BWR6 Plant Systems Actuation Instrumentation (RHR Containment Spray System, Feedwater & Main Turbine Trip- System, and Suppression Pool Makeup System Instrumentation) provide a more specific reference system to which each of these three systems can be compared. Also provide additional information describing the valve unavailability analyses and results.

RESPONSE TO QUESTION 7:

a) RHR Containment Spray System Instrumentation - The containment spray is initiated by 1 out of 2 trip systems. Each system has a 10.85 minute timer which is initiated by either high drywell pressure or low reactor water level in a 1 out of 2 twice logic. One RHR valve is opened and closed in each system after the time delay when both high containment pressure and high drywell pressure are present in a 1 out of two logic. The reactor level and drywell pressure trip units are common to the ECCS initiation logic which were considered in the Reference 4 analysis.

A fault tree of the containment spray initiation including the valves that have to open and close is presented in Enclosure 7. The unavailability was calculated as a function of the level/pressure trip channel surveillance interval for a single trip system and 1 out of 2 trip systems. Using a level/pressure trip channel failure rate based on current experience, the change in calculated unavailability when the surveillance interval is extended from 1 to 3 months is 8.75E-05/demand. No credit is taken here for manual actuation if the automatic initiation logic should fail. This small increase in calculated unavailability is insignificant and is offset by the similar benefits of reduced testing discussed in Section 4.2 of the ECCS Actuation Instrumentation Analysis, Reference 4.

b) Feedwater and Main Turbine Level 8 Trip - This trip system is not included in the the BWR 6 Improved Standard Technical Specification. The reason for this is because a direct scram is provided for the level 8 trip (uses separate set of trip units and logic that was considered in the Reference 3 analysis). For those plants that do not have a direct scram on level 8 trip (such as BWR 4 plants), the effect of changes to the surveillance intervals of the feedwater and main turbine level 8 trip system is discussed in Section 3.1 of Reference 2. Calculations of the unavailability for this specific trip function for different surveillance intervals are provided in Enclosure 7. The calculated change in unavailabilities when the surveillance interval is extended from 1 to 3 months is acceptably low (1.24E-04/demand).

c) Suppression Pool Makeup System Instrumentation - The same sensors that initiate the low pressure ECCS and certain primary/secondary containment isolation valves are also used in the suppression pool

makeup system initiation logic. The logic is discussed in Enclosure 7. Also included in Enclosure 7 is calculated change in unavailability when the surveillance intervals are changed from 1 to 3 months. The small increase in calculated unavailability is insignificant and is offset by the similar benefits of reduced testing discussed in Section 4.2 of the ECCS Actuation Instrumentation Analysis, Reference 4.

QUESTION 8

For the BWR4 Main Control Room Environmental Control (MCREC) Actuation Instrumentation, please provide specific references for the reactor vessel water level 1, high drywell pressure, and main steam line high flow inputs to the MCREC systems. The quoted reference did not address the water level 1 and main steam line high flow inputs. More specific references should be provided for the MCREC inputs and the similarities in configuration, components and qualification addressed to allow comparison to be made and conclusions drawn. Also, please provide additional information on how the high control room and the refueling floor area radiation functions are similar to the reactor building exhaust high radiation function to enable judgement to be made as to whether the fault trees and their evaluation are applicable to the high control room and the refueling floor area radiation functions.

RESPONSE TO QUESTION 8:

The trip unit channels and logic that actuate the BWR 4 Main Control Room Environmental Control (MCREC) are provided in Enclosure 8. MCREC is made up of two redundant independent subsystems which are initiated by Trip System Logic A & B (one subsystem by logic A and one subsystem by logic B). The following is a summary of the different trips, type of logic, and where covered by previous analysis.

EVENT	LOGIC PER SUBSYSTEM	ANALYSIS REFERENCE	
LOCA	1 out of 2 twice	Reference 5 & Reference 6	Page B-5 Page 5-21
Main Steam Line High Rad	2 out of 2	Reference 6	Page 5-21
Main Steam Line Break	2 out of 2	Reference 6	Page 5-21
Refueling Floor Area High Rad	1 out of 1	Reference 6	Page 5-21
Control Room Inlet High Rad	1 out of 1	Reference 6	Page 5-21
Control Room Inlet High Chlorine	1 out of 1	Reference 6	Page 5-21

The logic that initiates MCREC is similar to the logic given in the case studies for BWR plants. Instead of closing an inboard and outboard valve, the logic initiates redundant MCREC subsystems. The logic therefore is analogous to the logic given under the column headed "Logic Type Per Valve Per Variable". The only exception is the logic for a LOCA signal. The trip units and logic for the LOCA signal is the same signal that actuates the ECCS low pressure systems analyzed in Reference 5. The LOCA signal is also similar to the 1 out of 2 twice logic in Reference 6 except there are 2 variables instead of 4.

QUESTION 9

In the BWR6 Control Room Fresh Air (CRFA) Instrumentation, the reference given for the reactor water level 2 and the high drywell pressure trip in NEDC 31677P-A, page 5-11 does not present any information on these two trip inputs and it presents no information that would indicate that the CRFA system initiation was included in the analysis. The secondary containment statement mentions only the fuel handling area ventilation exhaust high radiation and the pool sweep exhaust radiation inputs. No fault trees or other analysis is presented in that reference for initiation of secondary containment isolation or isolation of the MCR and initiation of the CRFA system. Please provide more complete references which describe the system and show the similarities between the referenced instrumentation and this actuation instrumentation so we can complete our evaluation and reach a conclusion regarding the acceptability of this instrumentation.

RESPONSE TO QUESTION 9:

The BWR 6 CRFA is actuated by the same isolation logic that isolates the secondary containment isolation valves. The trip unit channels and logic for the BWR 6 CRFA are presented in Enclosure 9. The BWR 6 logic is the same type of logic that initiates the BWR 4 MCREC. The same case studies given in Reference 6, page 5-21, and Reference 5 for the LOCA initiated signal (reactor Level 1 or high drywell pressure) apply for the BWR 6 CRFA.

ENCLOSURES

Enclosure 1
Page 1 of 2

TABLE 3.3.9-1

PLANT SYSTEMS ACTUATION INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM	APPLICABLE OPERATIONAL CONDITIONS	
1. SUPPRESSION POOL (AND DRYWELL) SPRAY SYSTEM			
a. Drywell Pressure-High	1	1, 2, 3	85
b. Containment Pool Pressure-High	1	1, 2, 3	85
c. Reactor Vessel Water Level - Low Low Low, Level 1	1	1, 2, 3	85
d. Timers			
1) System A	1	1, 2, 3	85
2) System B	1	1, 2, 3	85
			(4)
2. FEEDWATER SYSTEM/MAIN TURBINE TRIP SYSTEM	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (5)		
a. Reactor Vessel Water Level-High, Level (8)	3	1 (2)	85

7/31/89

DISCUSSION OF CHANGES TO STS REV. 4
(Addendum)

LCO 3.3.2.2: FEEDWATER AND MAIN TURBINE TRIP INSTRUMENTATION (continued)

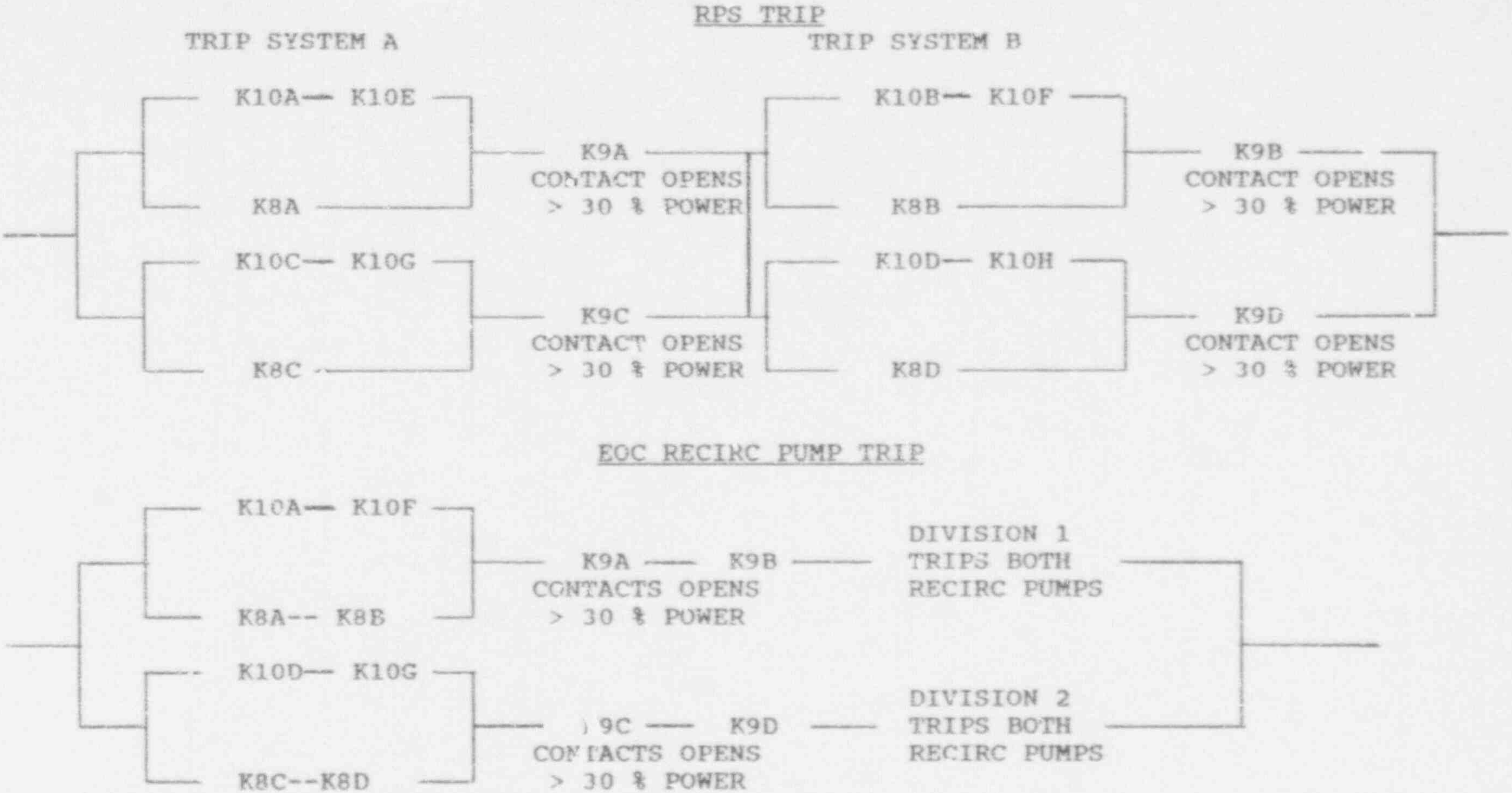
2. The Level 8 Function is assumed to function in plant specific MCPR analyses (if it is not, this specification would not be applicable to that plant). MCPR limits are only required to be met when ≥ 25% RTP, therefore this function, which only serves to support MCPR, has its applicability consistent with Specification 3.2.2.

4. The BWR design for suppression pool and drywell spray (where included in the design) are manually initiated and controlled systems. There are no automatic functions and therefore no corresponding instrumentation.

5. The requirement for system operability in the LCO requires the same three channels to be operable. The specific number of channels in the FW/turbine Level 8 trip system is a detail of the system design which is located in the Bases.

QUESTION 2
TURBINE CONTROL VALVE/STOP VALVE TRIP

ENCLOSURE 2
PAGE 1 OF 3



K8A,B,C,D - CHANNEL LOGIC RELAYS - CONTROL VALVE TRIP
K10A,B,C,D,E,F,G,H - CHANNEL LOGIC RELAYS - STOP VALVE TRIP
K9A,B,C,D - CHANNEL LOGIC RELAYS - REACTOR PRESSURE BYPASS FOR
REACTOR POWER > 30 % - RPS LOGIC CONTACTS OPEN
- EOC-RPT LOGIC CONTACTS CLOSED

QUESTION 2
TURBINE CONTROL VALVE/STOP VALVE TRIP

REFERENCE IEEE - 352

TURBINE CONTROL VALVE TRIP FUNCTION UNAVAILABILITY

$$RPS = 2\ [\ (XY)^2/3 + (XY)(P_{BP}) + (P_{BP})^2\ ]\ + CCF$$

$$EOC\text{-}RPT = (2XY)^2/3 + (2XY)(2P_{BP}) + (2P_{BP})^2\ + CCF$$

WHERE: X = CHANNEL FAILURE RATE (FAILURES/HR.)

Y = CHANNEL SURVEILLANCE INTERVAL (HRS.)

P_{BP} = BYPASS FAILURE PROBABILITY

TURBINE STOP VALVE TRIP FUNCTION UNAVAILABILITY

$$RPS = 2\ [\ (2XY)^2/3 + (2XY)(P_{BP}) + (P_{BP})^2\ ] + CCF$$

$$EOC\text{-}RPT = (2XY)^2/3 + (2XY)(2P_{BP}) + (2P_{BP})^2 + CCF$$

WHERE: X = CHANNEL FAILURE RATE (FAILURES/HR.)

Y = CHANNEL SURVEILLANCE INTERVAL (HRS.)

P_{BP} = BYPASS FAILURE PROBABILITY

CCF = COMMON CAUSE MISCAL. OF TRIP UNITS PROB.

QUESTION 2 - TURBINE CONTROL VALVE/STOP VALVE TRIP

	FAILURE RATE FAILURES/HR.
TB STOP VALVE	6.00E-06
TB CONTROL VALVE	2.00E-07
1ST STAGE PRESSURE (BYPASS)	2.00E-05
RELAY	4.00E-07

	Failure Rate (FR) (SENSOR + RELAY)
TB STOP VALVE	6.40E-06
TB CONTROL VALVE	6.00E-07
1ST STAGE PRESSURE (BYPASS)	2.04E-05

UNAVAILABILITY (RPS)

TRIP FUNCTION	1 MONTH	3 MONTHS	RPS CHANGE
TB STOP VALVE	4.41E-05	2.08E-04	1.64E-04
TB CONTROL VALVE	2.04E-05	2.11E-05	7.70E-07

UNAVAILABILITY (EOC-RPT)

TRIP FUNCTION	1 MONTH	3 MONTHS	EOC-RPT CHANGE
TB STOP VALVE	3.45E-05	1.21E-04	8.68E-05
TB CONTROL VALVE	2.08E-05	2.23E-05	1.54E-06

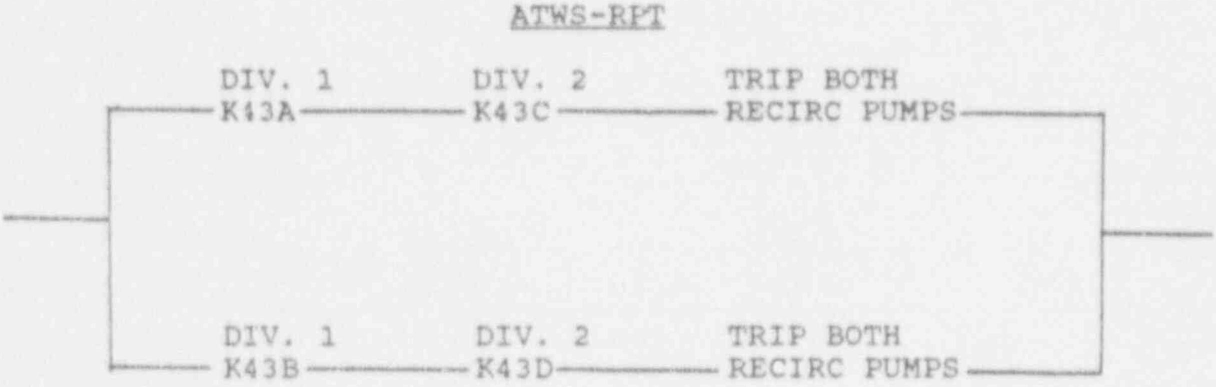
WHERE RPS UNAVAILABILITY =
2*[((FR*T)*2)/3+(FR*T)*PF+(PF)*2] + CCF (STOP VALVE)
2*[((2FR*T)*2)/3+(2FR*T)*PF+(PF)*2] + CCF (CONTROL VALVE)

FR = CHANNEL FAILURE RATE
T = CHANNEL FUNCTIONAL TEST INTERVAL
PF = PROBABILITY OF BYPASS FAILURE
CCF =COMMON CAUSE MISCAL. OF TRIP UNITS
= 2.00E-05

EOC-RPT UNAVAILABILITY =
((2*FR*T)*2)/3+(2*FR*T)*(2*PF)+(2*PF)*2 + CCF

1ST STAGE PRESSURE (BYPASS) - IF ANY OUTPUT RELAY ENERGIZES (BYPASS POSITION) FAILURE WILL BE ANNUNCIATED

QUESTION 3 - ATWS-RPT LOGIC



TRIPS ON LOW REACTOR WATER LEVEL

OR

HIGH REACTOR PRESSURE

QUESTION 3 - ATWS-RPT LOGIC

	FAILURE RATES (FAILURES/HR.)	
	DESIGN ANALYSIS	CURRENT EXPERIENCE
LEVEL	2.00E-05	5.00E-06
PRESSURE	2.00E-05	5.00E-06
RELAY	4.00E-07	4.00E-07

	Failure Rate (FR) (SENSOR + RELAY)	
LEVEL	2.04E-05	5.40E-06
PRESSURE	2.04E-05	5.40E-06

UNAVAILABILITY (ATWS-RPT) PER FUNCTION

TRIP FUNCTION	1 MONTH	3 MONTHS	ATWS-RPT FUNCTION CHANGE
(1) DESIGN ANALYSIS FRs			
LOW LEVEL	3.16E-04	2.68E-03	2.37E-03
HIGH Rx PRESSURE	3.16E-04	2.68E-03	2.37E-03
(2) OPERATING EXPR. FRs			
LOW LEVEL	4.07E-05	2.06E-04	1.66E-04
HIGH Rx PRESSURE	4.07E-05	2.06E-04	1.66E-04

WHERE: ATWS-RPT FUNCTION UNAVAILABILITY = (1/3)*(2*FR*T)*2 + CCF

CCF =COMMON CAUSE MISCAL. OF TRIP UNITS= 2.00E-05

REACTIVITY SHUTDOWN FAILURE FREQUENCY

		1 MONTH	3 MONTHS	CHANGE
RPS FAILURE FREQ. (EVENTS/YEAR)		4.60E-06	5.40E-06	8.00E-07
ATWS-RPT UNAVAIL.	(1)	3.16E-04	2.68E-03	2.37E-03
(/DEMAND)	(2)	4.07E-05	2.06E-04	1.66E-04
REACTIVITY SHUTDOWN	(1)	1.45E-09	1.45E-08	1.30E-08
FREQ. (EVENTS/YEAR)	(2)	1.87E-10	1.11E-09	9.28E-10

QUESTION 4

FAILURE RATES USED IN LLS FAULT TREE FOR HATCH 2

COMPONENT	FAILURE RATE (1)	TEST INTERVAL (2)	[(1)*(2)]/2 UNAVAIL.
RESET SWITCH	1.30E-08	730	4.75E-06
	1.30E-08	2190	1.42E-05
RELAY	4.00E-07	730	1.46E-04
	4.00E-07	2190	4.38E-04
TRIP UNIT	2.00E-05	730	7.30E-03
	2.00E-05	2190	2.19E-02
PRESSURE SWITCH	2.00E-07	730	7.30E-05
	2.00E-07	2190	2.19E-04

COMPONENT	DEMAND FAILURE RATE	FAILURE PROB.
CCF OF SENSORS	2.00E-05	2.00E-05
DC DIVISION POWER	2.00E-03	2.00E-03
LLS SOLENOID VALVE	1.31E-03	1.31E-03
TRANSMITTER	1.30E-05	1.30E-05

FAILURE PROB. OF LOW-LOW SET

TEST INTERVAL	FAILURE PROB.
1 MONTH	2.66E-05
3 MONTHS	6.58E-05
CHANGE FROM 1 TO 3 MONTHS	3.92E-05

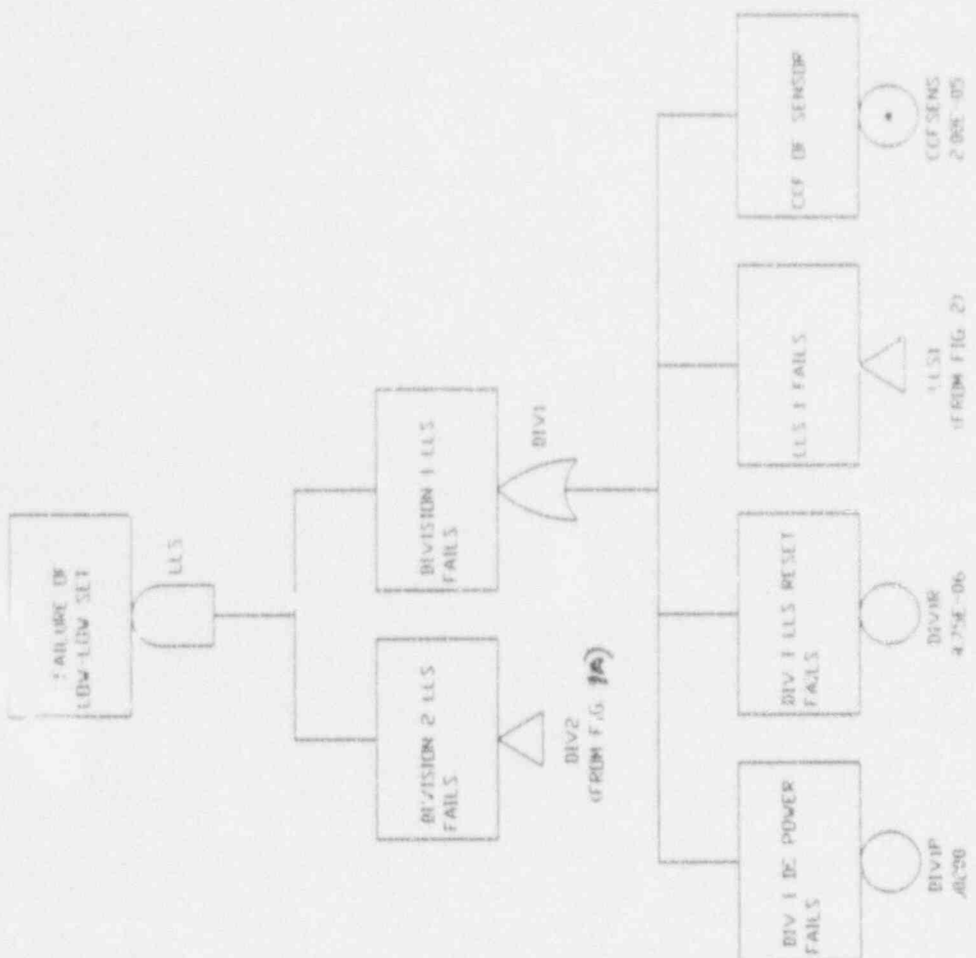


Figure 1
LOW-LOW SET FAULT TREE FOR HATCH

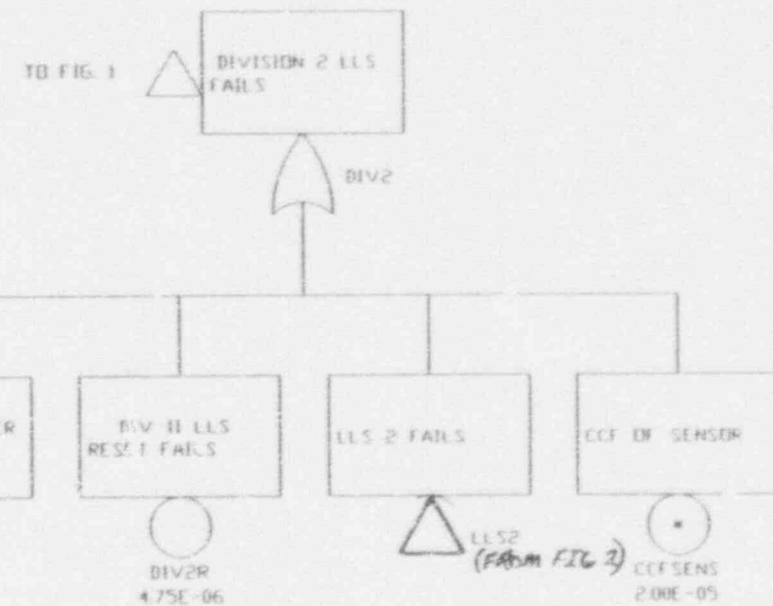


Figure 1A
LOW-LOW SET FAULT TREE FOR HATCH 2

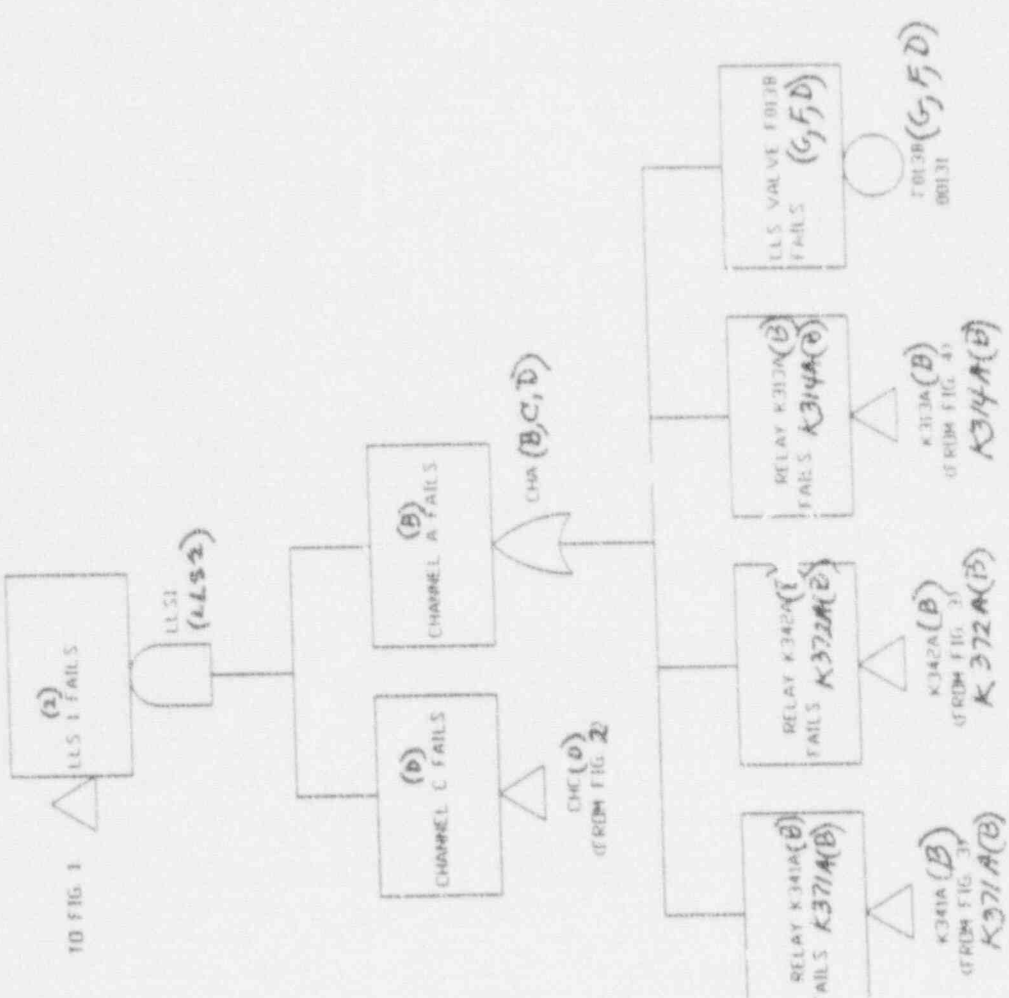


Figure 2

LOW-LOW SET FAULT TREE FOR HATCH 2

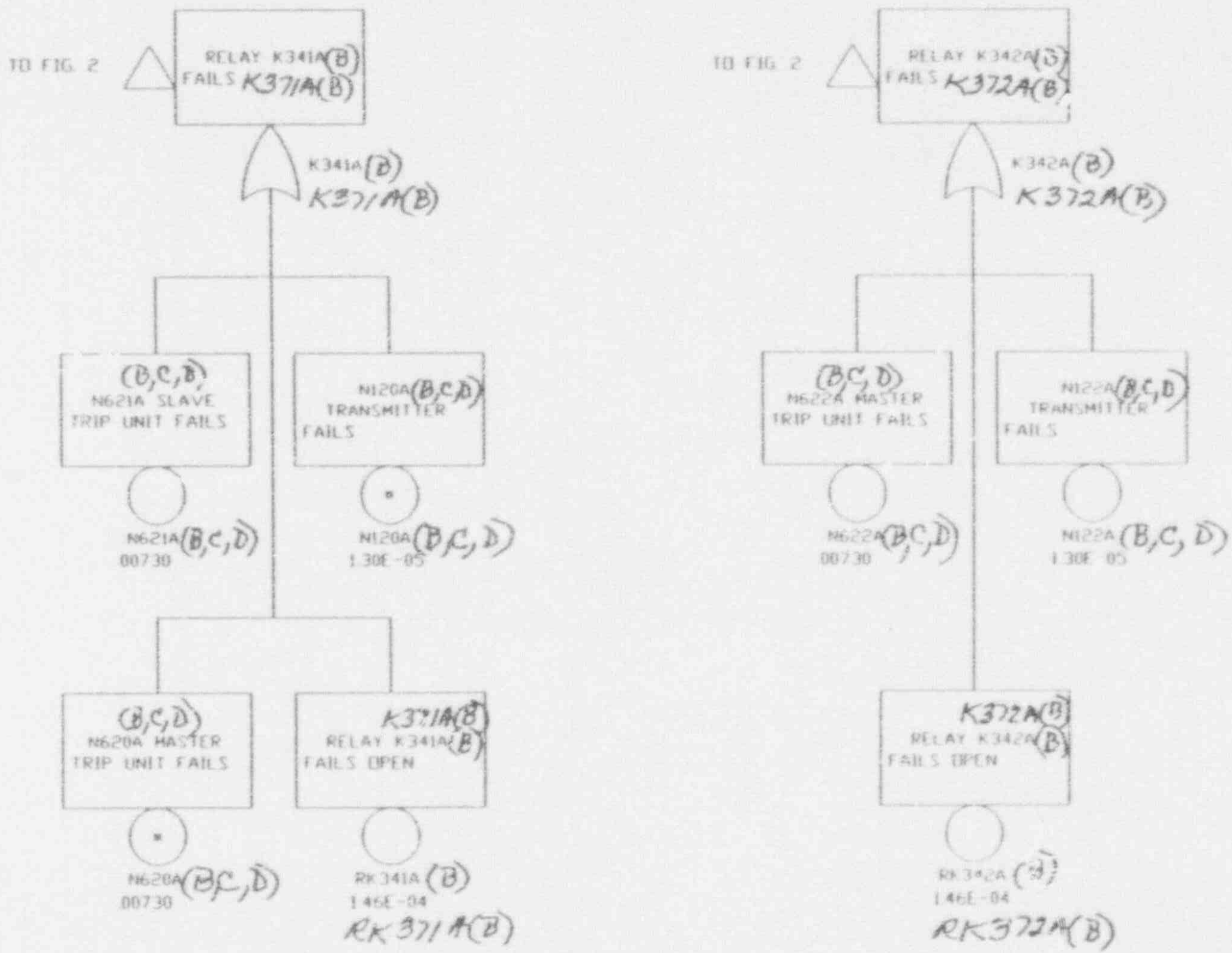


Figure 3

LOW-LOW SET FAULT TREE FOR HATCH 2

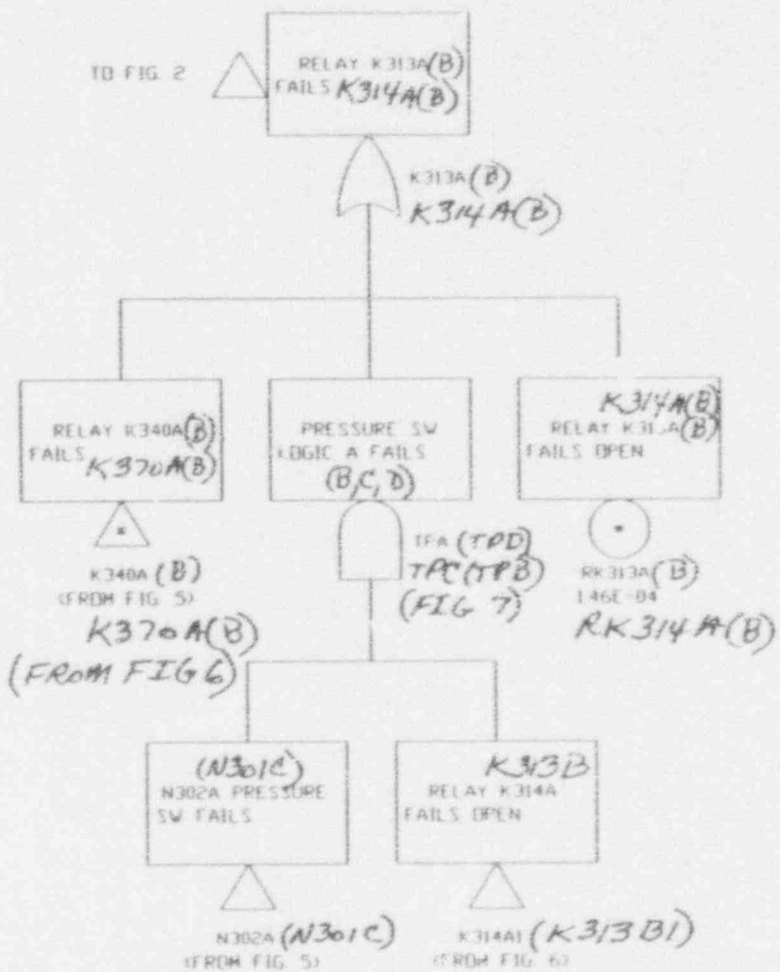


Figure 4
LOW-LOW SET FAULT TREE FOR HATCH 2

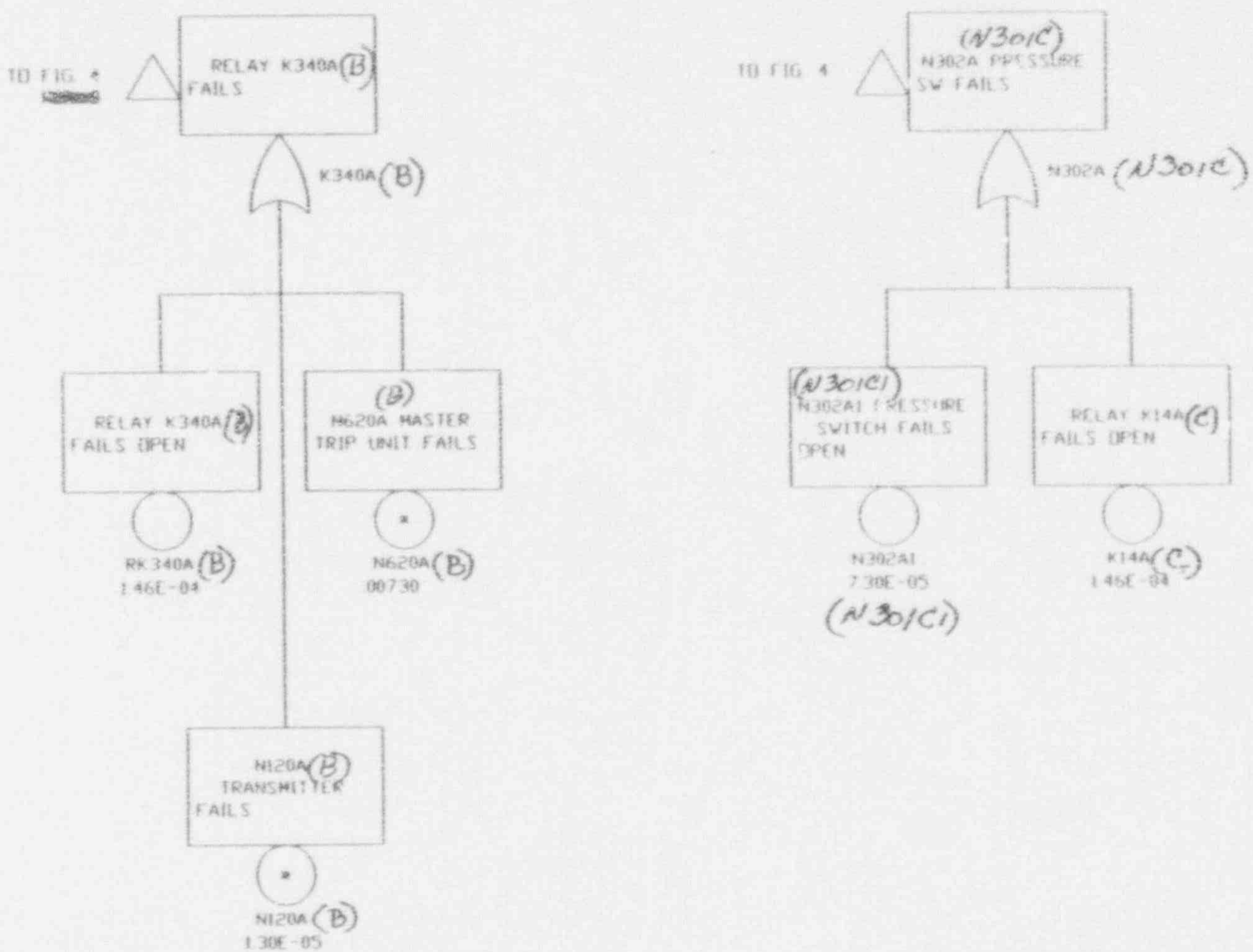


Figure 5

LOW-LOW SET FAULT TREE FOR HATCH 2

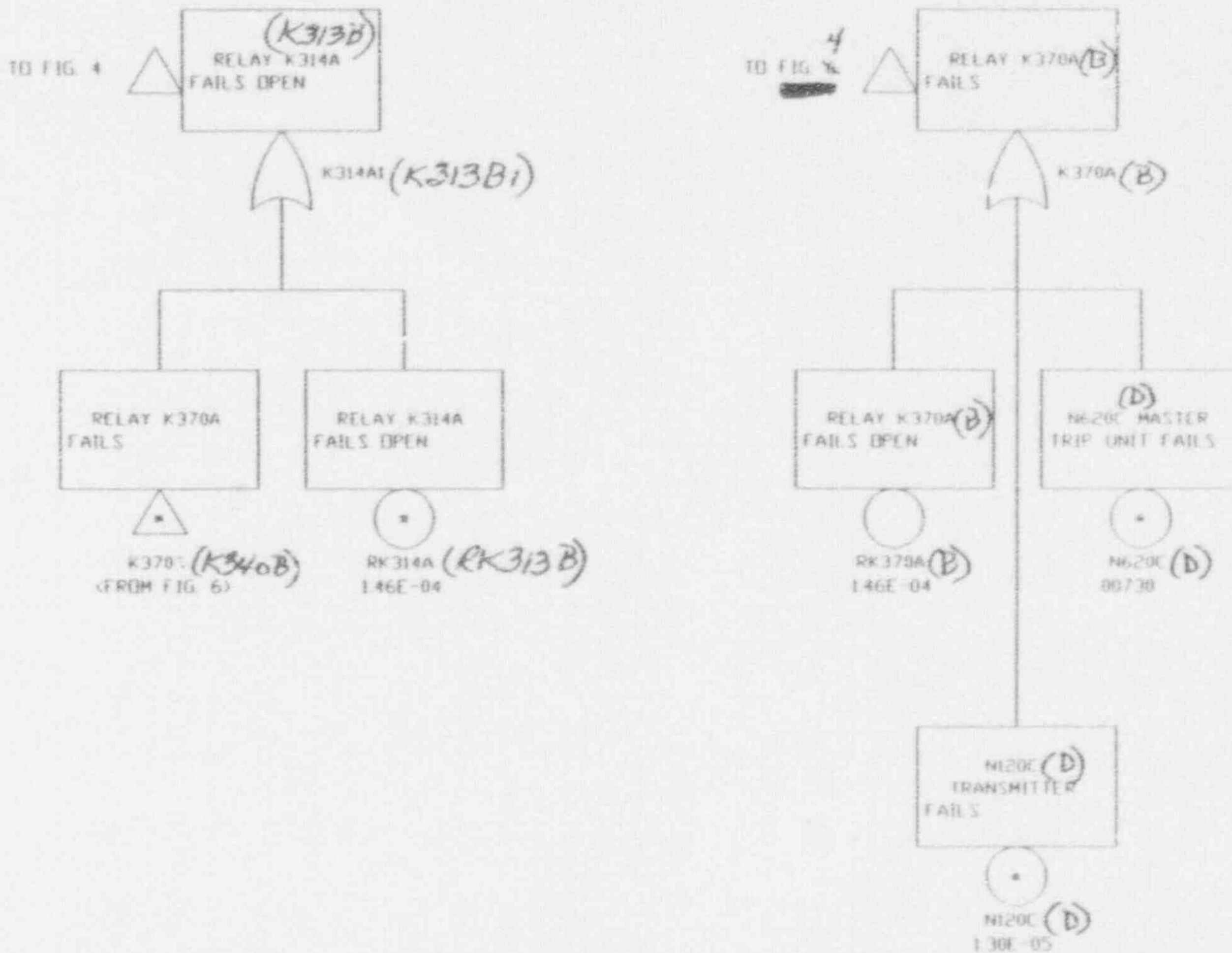


Figure 6

LOW-LOW SET FAULT TREE FOR HATCH 2

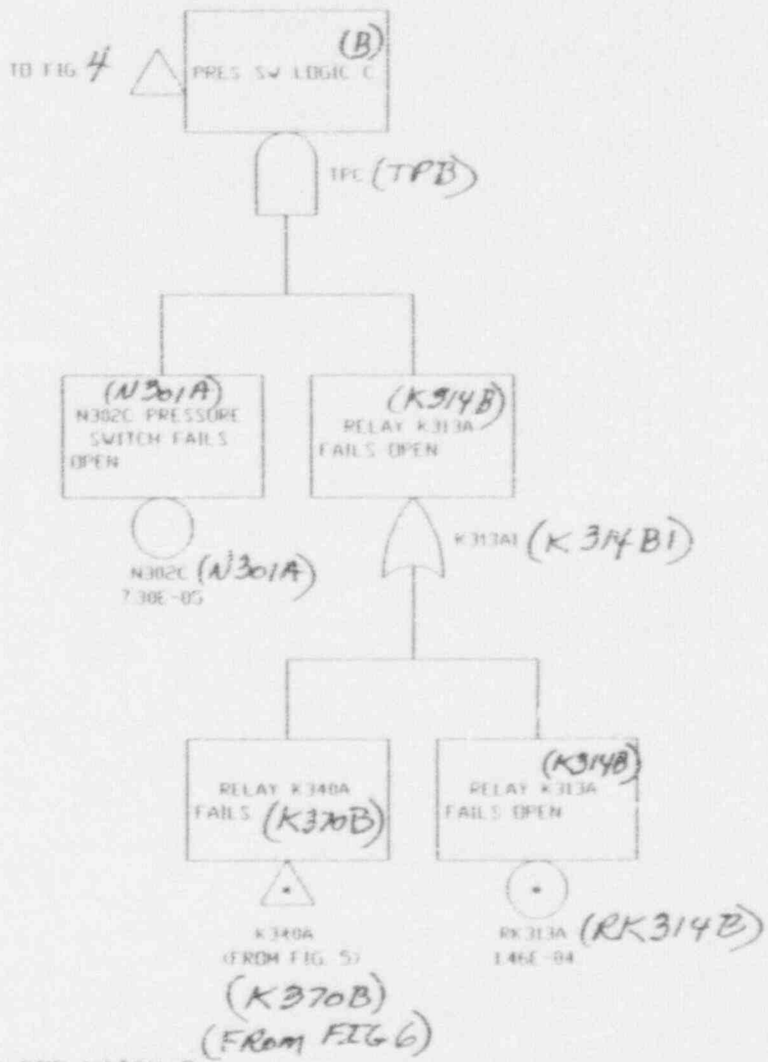


Figure 7
LOW-LOW SET FAULT TREE FOR HATCH 2

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Tree list:

LLS	AND	DIV1	DIV2		
DIV1	OR	DIV1P	DIV1R	LLS1	CCFSENS
LLS1	AND	CHA	CHC		
DIV2	OR	DIV2P	DIV2R	LLS2	CCFSENS
LLS2	AND	CHB	CHD		
CHA	OR	K341A	K342A	K313A	F013B
CHB	OR	K341B	K342B	K313B	F013G
CHC	OR	K371A	K372A	K314A	F013F
CHD	OR	K371B	K372B	K314B	F013D
K341A	OR	N621A	N120A	N620A	RK341A
K341B	OR	N621B	N120B	N620B	RK341B
K371A	OR	N621C	N120C	N620C	RK371A
K371B	OR	N621D	N120D	N620D	RK371B
K342A	OR	N622A	N122A	RK342A	
K342B	OR	N622B	N122B	RK342B	
K372A	OR	N622C	N122C	RK372A	
K372B	OR	N622D	N122D	RK372B	
K313A	OR	K340A	RK313A	TPA	
K313B	OR	K340B	RK313B	TPB	
K314A	OR	K370A	RK314A	TPC	
K314B	OR	K370B	RK314B	TPD	
K340A	OR	RK340A	N620A	N120A	
K340B	OR	RK340B	N620B	N120B	
K370A	OR	RK370A	N620C	N120C	
K370B	OR	RK370B	N620D	N120D	
TPA	AND	N302A	K314A1		
TPB	AND	N301A	K314B1		
TPC	AND	N302C	K313A1		
TPD	AND	N301C	K313B1		
N302A	OR	N302A1	K14A		
N301C	OR	N301C1	K14C		
K314A1	OR	K370A	RK314A		
K314B1	OR	K370B	RK314B		
K313A1	OR	K340A	RK313A		
K313B1	OR	K340B	RK313B		

Gate descriptions:

Gate Name	Description
CHA	CHANNEL A FAILS
CHB	CHANNEL B FAILS
CHC	CHANNEL C FAILS
CHD	CHANNEL D FAILS
DIV1	DIVISION 1 LOGIC FAILS
DIV2	DIVISION 2 LOGIC FAILS
K313A	RELAY K313A FAILS
K313A1	RELAY K313A FAILS OPEN
K313B	RELAY K313B FAILS
K313B1	RELAY K313B FAILS OPEN
K314A	RELAY K314A FAILS
K314A1	RELAY K314A FAILS OPEN
K314B	RELAY K314B FAILS
K314B1	RELAY K314B FAILS OPEN
K340A	RELAY K340A FAILS
K340B	RELAY K340B FAILS
K341A	RELAY K341A FAILS
K341B	RELAY K341B FAILS
K342A	RELAY K342A FAILS
K342B	RELAY K342B FAILS
K370A	RELAY K370A FAILS
K370B	RELAY K370B FAILS
K371A	RELAY K371A FAILS
K371B	RELAY K371B FAILS
K372A	RELAY K372A FAILS
K372B	RELAY K372B FAILS
LLS	FAILURE OF LOW-LOW SET LOGIC
LLS1	LLS LOGIC 1 FAILS
LLS2	LLS LOGIC 2 FAILS
N301C	N301C PRESSURE SW FAILS
N302A	N302A PRESSURE SW FAILS
TPA	PRESSURE SW LOGIC A FAILS
TPB	PRESSURE SW LOGIC B FAILS
TPC	PRES SW LOGIC C
TPD	PRES SW LOGIC D

Basic Event Descriptions:

Basic Event Name	Prob.	Description
CCFSENS	2.00E-05	CCF OF SENSOR
DIV1P	2.00E-03	DIV I DC POWER FAILS
DIV1R	4.75E-06	DIV I LLS RESET FAILS
DIV2P	2.00E-03	DIV II DC POWER FAILS
DIV2R	4.75E-06	DIV II LLS RESET FAILS
F013B	1.31E-03	LLS VALVE F013B FAILS
F013D	1.31E-03	LLS VALVE F013D FAILS
F013F	1.31E-03	LLS VALVE F013F FAILS
F013G	1.31E-03	LLS VALVE F013G FAILS
K14A	1.46E-04	RELAY K14A FAILS OPEN
K14C	1.46E-04	RELAY K14C FAILS OPEN
N120A	1.30E-05	N120A TRANSMITTER FAILS
N120B	1.30E-05	N120B TRANSMITTER FAILS
N120C	1.30E-05	N120C TRANSMITTER FAILS
N120D	1.30E-05	N120D TRANSMITTER FAILS
N122A	1.30E-05	N122A TRANSMITTER FAILS
N122B	1.30E-05	N122B TRANSMITTER FAILS
N122C	1.30E-05	N122C TRANSMITTER FAILS
N122D	1.30E-05	N122D TRANSMITTER FAILS
N301A	7.30E-05	N301A PRESSURE SWITCH FAILS OPEN
N301C1	7.30E-05	N301C1 PRESSURE SWITCH FAILS OPEN
N302A1	7.30E-05	N302A1 PRESSURE SWITCH FAILS OPEN
N302C	7.30E-05	N302C PPESSURE SWITCH FAILS OPEN
N620A	7.30E-03	N620A MASTER TRIP UNIT FAILS
N620B	7.30E-03	N620B MASTER TRIP UNIT FAILS
N620C	7.30E-03	N620C MASTER TRIP UNIT FAILS
N620D	7.30E-03	N620D MASTER TRIP UNIT FAILS
N621A	7.30E-03	N621A SLAVE TRIP UNIT FAILS
N621B	7.30E-03	N621B SLAVE TRIP UNIT FAILS
N621C	7.30E-03	N621C SLAVE TRIP UNIT FAILS
N621D	7.30E-03	N621D SLAVE TRIP UNIT FAILS
N622A	7.30E-03	N622A MASTER TRIP UNIT FAILS
N622B	7.30E-03	N622B MASTER TRIP UNIT FAILS
N622C	7.30E-03	N622C MASTER TRIP UNIT FAILS
N622D	7.30E-03	N622D MASTER TRIP UNIT FAILS
RK313A	1.46E-04	RELAY K313A FAILS OPEN
RK313B	1.46E-04	RELAY K313B FAILS OPEN
RK314A	1.46E-04	RELAY K314A FAILS OPEN
RK314B	1.46E-04	RELAY K314B FAILS OPEN
RK340A	1.46E-04	RELAY K340A FAILS OPEN
RK340B	1.46E-04	RELAY K340B FAILS OPEN
RK341A	1.46E-04	RELAY K341A FAILS OPEN
RK341B	1.46E-04	RELAY K341B FAILS OPEN
RK342A	1.46E-04	RELAY K342A FAILS OPEN
RK342B	1.46E-04	RELAY K342B FAILS OPEN

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Basic Event Descriptions:

Basic Event Name	Prob.	Description
RK370A	1.46E-04	RELAY K370A FAILS OPEN
RK370B	1.46E-04	RELAY K370B FAILS OPEN
RK371A	1.46E-04	RELAY K371A FAILS OPEN
RK371B	1.46E-04	RELAY K371B FAILS OPEN
RK372A	1.46E-04	RELAY K372A FAILS OPEN
RK372B	1.46E-04	RELAY K372B FAILS OPEN

QUESTION 5

GRAND GULF LLS/RELIEF LOGIC

- TOTAL OF 20 S/RVs
- 6 OF THE S/RVs PERFORM A LLS FUNCTION

LLS FUNCTION

LLS VALVES	LOGIC EACH DIVISION	SEAL-IN LOGIC DIV 1 TUs	SEAL-IN LOGIC DIV 2 TUs	LLS LOGIC DIV 1 TUs	LLS LOGIC DIV 2 TUs
F047D,G	1 OUT OF 3 TWICE	N668A	N668B	N618A	N618B
F051A,F	SEAL-IN LOGIC	N669A	N669B	N618E	N618F
		N670A	N670B		
	2 OUT OF 2				
	LLS LOGIC	N668E	N668F		
		N669E	N669F		
		N670E	N670F		
F051D	1 OUT OF 3 TWICE	N668A	N668B	N616E	N616F
	SEAL-IN LOGIC	N669A	N669B		
		N670A	N670B		
	1 OUT OF 1				
	LLS LOGIC	N668E	N668F		
		N669E	N669F		
		N670E	N670F		
F051B	1 OUT OF 3 TWICE	N668A	N668B	N617A	N617B
	SEAL-IN LOGIC	N669A	N669B		
		N670A	N670B		
	1 OUT OF 1				
	LLS LOGIC	N668E	N668F		
		N669E	N669F		
		N670E	N670F		

QUESTION 5

GRAND GULF BWR 6 LOW-LOW SET FUNCTION UNAVAILABILITY

	FAILURE RATE FAILURES/HR.
PRESSURE TRIP UNIT	2.00E-05
RELAY	4.00E-07

FAULT TREE INPUTS

	Failure Rate	STI (HOURS)	UNAVAIL.
LLS LOGIC CHANNEL (TRIP UNIT + RELAY)	2.04E-05	730	7.45E-03
		2190	2.23E-02
DIVISION POWER SUPPLY	2.00E-03	-----	2.00E-03
CCF OF SENSORS	2.00E-05	-----	2.00E-05

LLSL UNAVAILABILITY

	1 MONTH	3 MONTHS	CHANGE
1 LOGIC GROUP	2.40E-05	2.40E-05	4.48E-08
1 OUT OF 6 LLS VALVES (1 OUT OF 3 LOGIC GROUPS)	2.40E-05	2.40E-05	NEGLIGIBLE

QUESTION 6

GRAND GULF LLS/RELIEF LOGIC

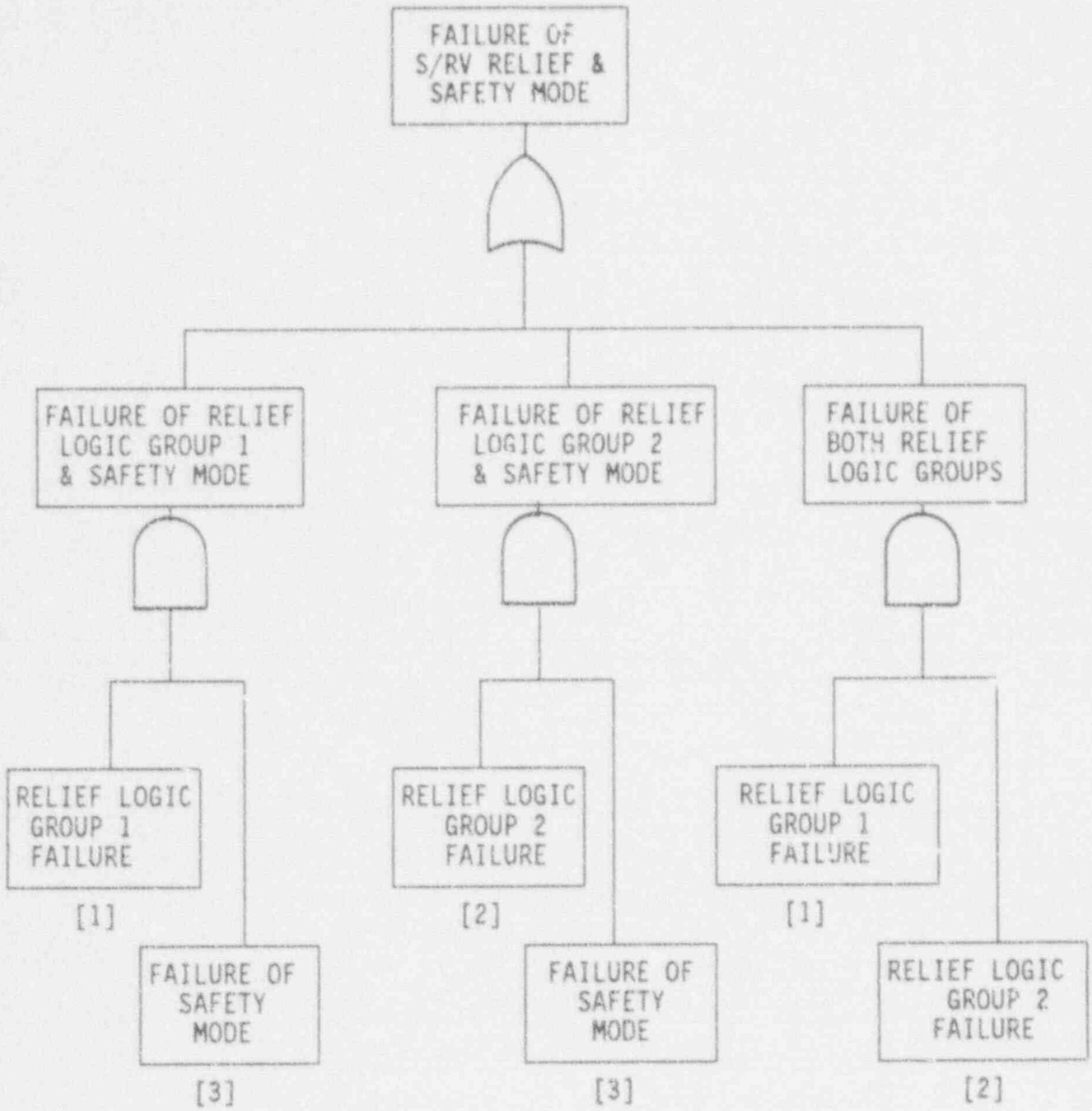
- TOTAL OF 20 S/RVs
- 6 S/RVs HAVE LLS FUNCTION

S/RV LOGIC GROUP	NO. IN GROUP	Rx PRESSURE TRIP UNITS	LOGIC
F041 A,B,C,D,E,F,G,K F051 C	9	N670 A,B,E,F	2 OUT OF 2/DIV. 1 OUT OF 2 DIV.
F047 A,C,H,L F051 K	5	N669 A,B,E,F	2 OUT OF 2/DIV. 1 OUT OF 2 DIV.
F047 D,G F051 A,F	4	N669 A,B,E,F OR LLSL (SEE ENCL. 5)	2 OUT OF 2/DIV. 1 OUT OF 2 DIV. LLSL (SEE ENCL. 5)
F051 B	1	N669 A,B,E,F OR LLSL (SEE ENCL. 5)	2 OUT OF 2/DIV. 1 OUT OF 2 DIV. LLSL (SEE ENCL. 5)
F051 D	1	N668 A,B,E,F OR LLSL (SEE ENCL. 5)	2 OUT OF 2/DIV. 1 OUT OF 2 DIV. LLSL (SEE ENCL. 5)
	20		

NOTE: TRIP UNITS N668 A,B,E,F PROVIDE THE SEAL-IN FUNCTION FOR THE LLSL.
N669 A,B,E,F
N670 A,B,E,F

13 OUT OF 20 S/RVs ARE REQUIRED TO PREVENT REACTOR OVERPRESSURE. ASME CODE ALLOWS 1/2 TO OPEN ON RELIEF MODE (I.E., AT LEAST 7 S/RVs) AND 1/2 TO OPEN ON THE SAFETY MODE (I.E., AT LEAST 6 S/RV).

QUESTION 6
GRAND GULF LLS/RELIEF LOGIC



[1] RELIEF LOGIC GROUP 1 FAILURE (9 S/RVs AFFECTED)
FAILURE OF Rx PRESSURE TRIP UNIT CHANNELS N670AM <u>OR</u> N670E <u>AND</u> N670B <u>OR</u> N670F

[2] RELIEF LOGIC GROUP 2 FAILURE (10 S/RVs AFFECTED)
FAILURE OF Rx PRESSURE TRIP UNIT CHANNELS N669AM <u>OR</u> N669E <u>AND</u> N669B <u>OR</u> N669F

[3] FAILURE TO ACTUATE AT LEAST 6 S/RVs IN SAFETY MODE

QUESTION 6

GRAND GULF LLS/RELIEF LOGIC

	FAILURE RATE (FAILURES/HOUR) DESIGN ANALYSIS	CURRENT EXPERIENCE
PRESSURE TRIP UNIT	2.00E-05	5.00E-06
RELAY	4.00E-07	4.00E-07
	Failure Rate (FR) (TRIP UNIT + RELAY)	
PRESSURE TRIP CHANNEL	2.04E-05	5.40E-06

UNAVAILABILITY - S/RV RELIEF FUNCTION

	SURVEILLANCE INTERVAL 1 MONTH	3 MONTHS	CHANGE
RELIEF MODE (SINGLE LOGIC GROUP)			
DESIGN ANALYSIS FR	3.16E-04	2.68E-03	2.37E-03
CURRENT EXPR. FR	4.07E-05	2.06E-04	1.66E-04
RELIEF & SAFETY MODE			
DESIGN ANALYSIS FR	1.06E-07	7.24E-06	7.14E-06
CURRENT EXPR. FR	2.47E-09	4.68E-08	4.43E-08

WHERE:

RELIEF MODE UNAVAILABILITY (SINGLE LOGIC GROUP)=

RM(1) = RM(2) = (1/3)*(2*FR*T)*2 + CCF

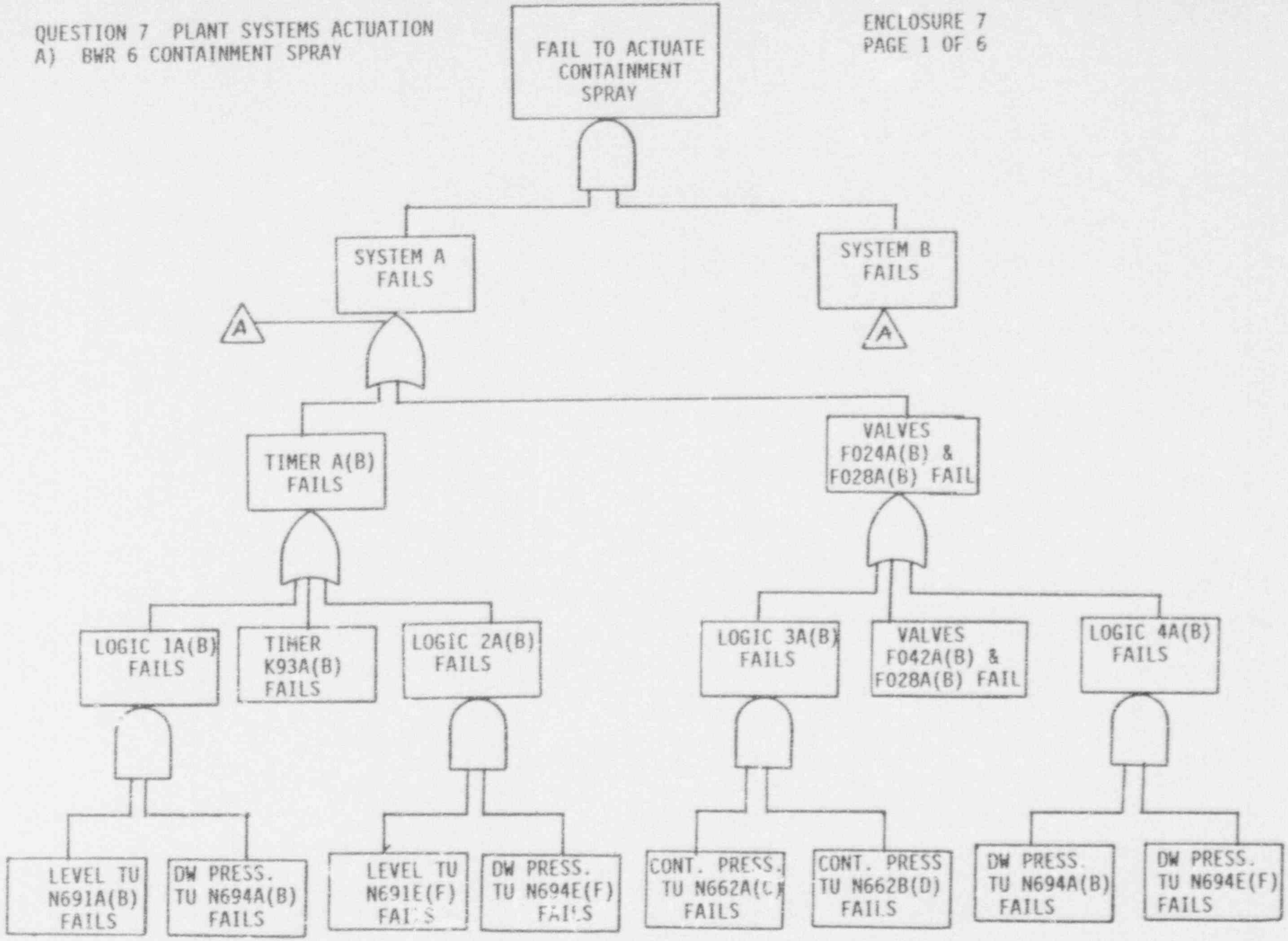
CCF = COMMON CAUSE MISCAL. OF TRIP UNITS
= 2.00E-05 /DEMAND

RELIEF & SAFETY MODE UNAVAILABILITY =

RM(1)*SM(1) + RM(2)*SM(2) + RM(1)*RM(2)

SM(1) = SM(2) = UNAVAILABILITY OF S/RV SAFETY MODE ACTUATION
= <1.00E-05/DEMAND

QUESTION 7 PLANT SYSTEMS ACTUATION
A) BWR 6 CONTAINMENT SPRAY



QUESTION 7 PLANT SYSTEMS ACTUATION
A) BWR 6 CONTAINMENT SPRAY

	FAILURE RATE (FAILURES/HOUR) DESIGN ANALYSIS	CURRENT EXPERIENCE
LEVEL/PRESSURE TRIP UNIT	2.00E-05	5.00E-06
RELAY	4.00E-07	4.00E-07
	Failure Rate (FR) (TRIP UNIT + RELAY)	
LEVEL/PRESSURE TRIP CHANNEL	2.04E-05	5.40E-06

UNAVAILABILITY - CONTAINMENT SPRAY AUTO INITIATION

	SURVEILLANCE INTERVAL 1 MONTH	3 MONTHS	CHANGE
SINGLE TRIP SYSTEM			
DESIGN ANALYSIS FR	6.29E-03	1.36E-02	7.31E-03
CURRENT EXPR. FR	6.02E-03	1.11E-02	5.11E-03
1 OUT OF 2 TRIP SYSTEMS			
DESIGN ANALYSIS FR	5.96E-05	2.05E-04	1.45E-04
CURRENT EXPR. FR	5.62E-05	1.44E-04	8.75E-05

WHERE:

SINGLE SYSTEM (TRAIN) UNAVAILABILITY =

SYS(1) = SYS(2) = 2*(2/3)*(FR*T)*2 + CCF +VLV + TIMER

VLV = PROB OF FAILURE TO OPEN ONE VALVE AND TO CLOSE ANOTHER
= 2*(1.6E-06/HR)*(2160 HRS)/2

TIMER = TIMER UNAVAIL. = (6.77E-06/HR)*(TEST INTERVAL)/2

FR = LEVEL OR PRESSURE TRIP CHANNEL FAILURE RATE (FAIL./HR)

T = SURVEILLANCE INTERVAL (HOURS)

CCF = COMMON CAUSE MISCAL. OF TRIP UNITS = 2.00E-05 /DEMAND

1 OUT OF 2 SYSTEMS UNAVAILABILITY = SYS(1)*SYS(2) + CCF

QUESTION 7 PLANT SYSTEMS ACTUATION

FEEDWATER/MAIN TURBINE TRIP

```
  LEVEL 8
 SWITCHES

  K624A -┐
         │
  K624B -┼-----   TWO OUT OF THREE TO TRIP
         │
  K624C -┘
```

QUESTION 7 PLANT SYSTEMS ACTUATION

FEEDWATER/MAIN TURBINE TRIP

	FAILURE RATE (FAILURES/HOUR) DESIGN ANALYSIS	CURRENT EXPERIENCE
LEVEL 8 TRIP UNIT	2.00E-05	5.00E-06
RELAY	4.00E-07	4.00E-07
	Failure Rate (FR) (TRIP UNIT + RELAY)	
LEVEL 8 TRIP CHANNEL	2.04E-05	5.40E-06

UNAVAILABILITY - FEEDWATER/MAIN TURBINE TRIP ON LEVEL 8

	SURVEILLANCE INTERVAL 1 MONTH	3 MONTHS	CHANGE
2 OUT OF 3			
DESIGN ANALYSIS FR	2.42E-04	2.02E-03	1.77E-03
CURRENT EXPR. FR	3.55E-05	1.60E-04	1.24E-04

WHERE:

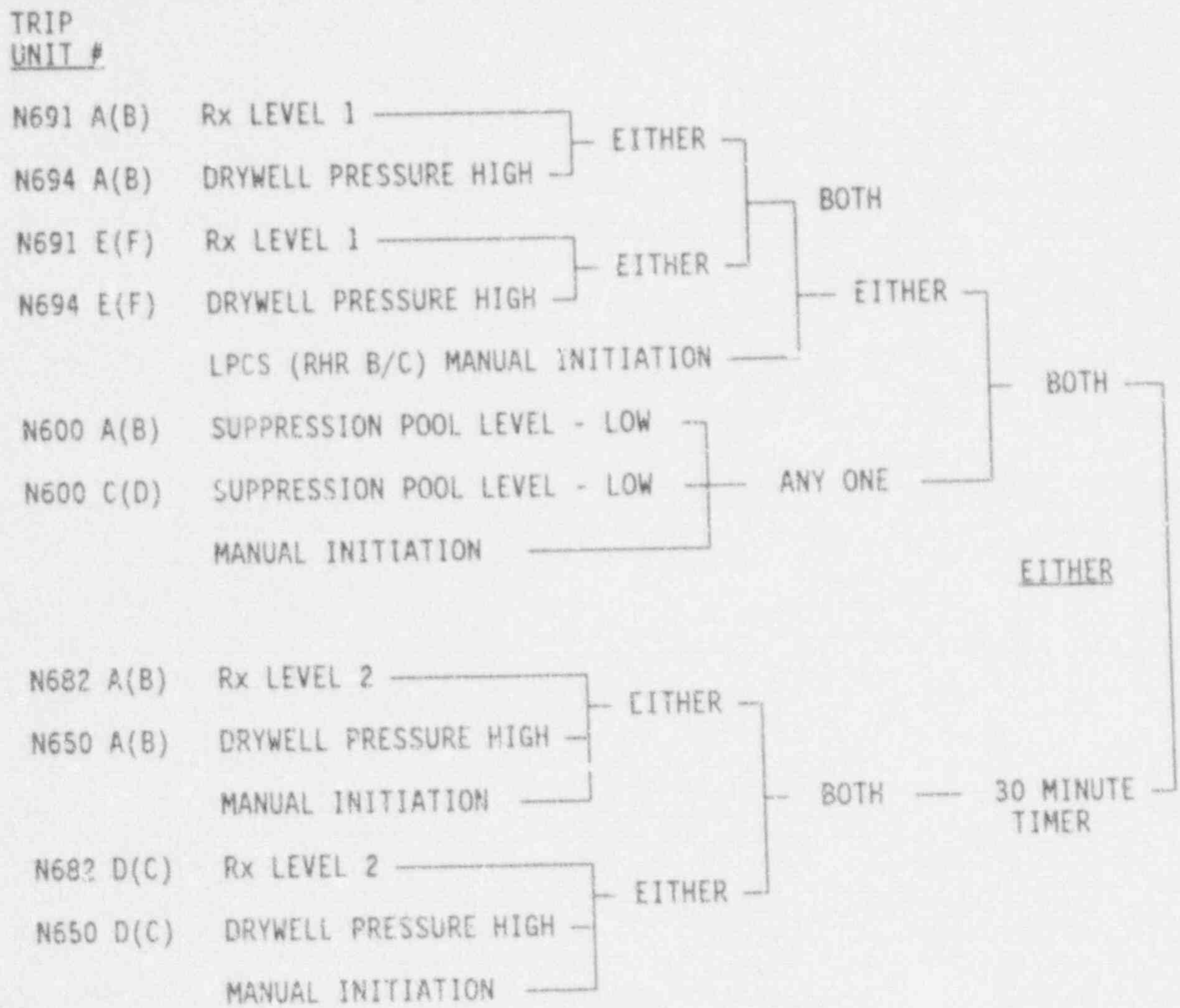
FEEDWATER/MAIN TURBINE LEVEL 8 TRIP UNAVAILABILITY =

(FR*T)*2 + CCF

FR = FAILURE RATE (FAILURES/HR)
T = SURVEILLANCE INTERVAL (HRS)
CCF = COMMON CAUSE MISCAL. OF TRIP UNITS = 2.00E-05 /DEMAND

QUESTION 7 PLANT SYSTEMS ACTUATION

SUPPRESSION POOL MAKEUP INITIATION



NOTES:

LETTERS IN PARENTHESES REPRESENT TRIP SYSTEM B

TRIP SYSTEM A OPENS VALVES F001A AND F002A

TRIP SYSTEM B OPENS VALVES F001B AND F002B

QUESTION 7 PLANT SYSTEMS ACTUATION

SUPPRESSION POOL MAKEUP INITIATION

UNAVAILABILITY - SUPPRESSION POOL MAKEUP AUTO INITIATION

	FAILURE RATE (FAILURES/HOUR) DESIGN ANALYSIS	CURRENT EXPERIENCE
LEVEL/PRESSURE TRIP UNIT	2.00E-05	5.00E-06
RELAY	4.00E-07	4.00E-07
	Failure Rate (FR) (TRIP UNIT + RELAY)	
LEVEL/PRESSURE TRIP CHANNEL	2.04E-05	5.40E-06

	SURVEILLANCE INTERVAL 1 MONTH	3 MONTHS	CHANGE
LOW SUPPR POOL LEVEL INITIATION - SYSTEM A(B)			
DESIGN ANALYSIS FR	3.75E-03	5.52E-03	1.77E-03
CURRENT EXPR. FR	3.54E-03	3.66E-03	1.24E-04
ANTICIPATORY TIME DELAY INITIATION - SYSTEM A(B)			
DESIGN ANALYSIS FR	6.14E-03	1.23E-02	6.12E-03
CURRENT EXPR. FR	6.01E-03	1.10E-02	5.02E-03
LOW SUPPR POOL OR ANTICIPATORY TIME DELAY INITIATION - SYSTEM A (B)			
DESIGN ANALYSIS FR	3.52E-03	3.[illegible]4E-03	1.69E-05
CURRENT EXPR. FR	3.52E-03	3.53E-03	1.01E-06

WHERE:

LOW SUPPR POOL LEVEL SYSTEM A(B) INITIATION UNAVAILABILITY = LSPL

2*(1/3)*(FR*T)*2 + (1/3)*(FR*T)*2 + CCF + VLV
  1.48E-04 + 7.39E-05 + 2.00E-05 + 3.50E-03 = 3.75E-03 (1 MONTH)
  1.33E-03 + 6.65E-04 + 2.00E-05 + 3.50E-03 = 5.52E-03 (3 MONTHS)

VLV = PROB. OF FAILURE TO OPEN TWO VALVES
    = 2*(1.6E-06/HR)*(2190 HRS)/2 = 3.50E-03

FR = LEVEL OR PRESSURE TRIP CHANNEL FAILURE RATE (FAILURES/HR)

T = SURVEILLANCE INTERVAL (HOURS)

CCF = COMMON CAUSE MISCAL. OF TRIP UNITS = 2.00E-05 /DEMAND

(CONTINUED ON NEXT PAGE)

QUESTION 7 PLANT SYSTEMS ACTUATION

SUPPRESSION POOL MAKEUP INITIATION (CONTINUED)

ANTICIPATORY TIME DELAY SYSTEM A(B) INITIATION = ATD

```
2*(1/3)*(FR*T)*2 + CCF + TIMER + VLV
  1.48E-04 +    2.00E-05 +    2.47E-03 +    3.50E-03 = 6.14E-03 (1 MONTH)
  1.33E-03 +    2.00E-05 +    7.41E-03 +    3.50E-03 = 1.23E-02 (3 MONTHS)
TIMER UNAVAIL. = (6.77E-06/HR)*(TEST INTERVAL)/2 = 2.47E-03 (1 MONTH)
                                                 = 7.41E-03 (3 MONTHS)
------------------------------------------------------------------------
LOW SUPPR POOL OR ANTICIPATORY TIME DELAY INITIATION - SYSTEM A (B)
(LSPL-VLV-CCF)*(ATD-VLV-CCF)+VLV+CCF =
(    3.75E-03 -    3.52E-03 )*(  6.14E-03 - 3.52E-03 )   +    3.52E-03
                                                   = 3.52E-03 (1 MONTH)

(    5.52E-03 -    3.52E-03 )*(  1.23E-02 - 3.52E-03 )   +    3.52E-03
                                                   = 3.54E-03 (3 MONTHS)
```

QUESTION 8

BWR 4 MCRECS INSTRUMENTATION

LOCA OR MCRECS ACTUATION

N691A	LEVEL 1	EITHER	BOTH	TRIPS MCRECS A
N694A	DRYWELL PRESSURE			
N691C	LEVEL 1	EITHER		
N694C	DRYWELL PRESSURE			
N691B	LEVEL 1	EITHER	BOTH	TRIPS MCRECS B
N694B	DRYWELL PRESSURE			
N691D	LEVEL 1	EITHER		
N694D	DRYWELL PRESSURE			

HIGH RADIATION IN STEAM LINE

K603A	HIGH MSL RADIATION	BOTH	TRIPS MCRECS A
K603C	HIGH MSL RADIATION		
K603B	HIGH MSL RADIATION	BOTH	TRIPS MCRECS B
K603D	HIGH MSL RADIATION		

MAIN STEAM LINE BREAK

N686A - N687A - N688A - N689A - STEAM LINE FLOW	BOTH	TRIPS MCRECS A
N686C - N687C - N688C - N689C - STEAM LINE FLOW		
N686B - N687B - N688B - N689B - STEAM LINE FLOW	BOTH	TRIPS MCRECS B
N686D - N687D - N688D - N689D - STEAM LINE FLOW		

QUESTION 8

BWR 4 MCRECS INSTRUMENTATION

HIGH RADIATION/CHLORINE LEVEL

K002A	REFUELING FLOOR AREA RAD.		
R615A	CONTROL ROOM AIR INLET RAD.	ANY ONE	TRIPS MCRECS A
N022A	CONTROL ROOM AIR INLET CHLORINE		
K002D	REFUELING FLOOR AREA RAD.		
R615B	CONTROL ROOM AIR INLET RAD.	ANY ONE	TRIPS MCRECS A
N022B	CONTROL ROOM AIR INLET CHLORINE		

QUESTION 9

SECONDARY CONTAINMENT ISOLATION ACTUATION

```
N650B   DRYWELL PRESSURE HIGH -┐
N682B   RPV LEVEL 2 -----------┼- ANY ONE --┐
        MANUAL INITIATION -----┘            │          TRIP
                                            ├- BOTH    SYSTEM
N650C   DRYWELL PRESSURE HIGH -┐            │            2
N682C   RPV LEVEL 2 -----------┼- ANY ONE --┘
        MANUAL INITIATION -----┘

N650A   DRYWELL PRESSURE HIGH -┐
N682A   RPV LEVEL 2 -----------┼- ANY ONE --┐
        MANUAL INITIATION -----┘            │          TRIP
                                            ├- BOTH    SYSTEM
N650D   DRYWELL PRESSURE HIGH -┐            │            1
N682D   RPV LEVEL 2 -----------┼- ANY ONE --┘
        MANUAL INITIATION -----┘
```

TRIPS PERFORMED BY EACH SYSTEM*

- o CLOSES CONTAINMENT COOLING SYSTEM DAMPERS
- o CLOSES AUX. BLDG. FUEL HANDLING AREA VENT. SYS. DAMPERS
- o CLOSES AUX. BLDG. VENT. SYS. DAMPERS
- o STARTS STANDBY GAS TREATMENT SYSTEM
- o CLOSES CONTROL ROOM VENT. SYS. DAMPERS
- o STARTS CONTROL ROOM EMERGENCY FILTRATION SYSTEM
- o CLOSES SECONDARY CONTAINMENT ISOLATION VALVES (SEVERAL)

* EACH TRIP SYSTEM ACTUATES EITHER AN INBOARD OR OUTBOARD ISOLATION VALVE/DAMPER OR EITHER ONE OF REDUNDANT SYSTEMS.

QUESTION 9

SECONDARY CONTAINMENT ISOLATION

```
K617B    FUEL HANDLING AREA VENT. --┐
         EXH. RAD. - HIGH           ├-- BOTH --┐
                                    │          │                TRIPS
K617C    FUEL HANDLING AREA VENT. --┘          │               SYSTEM
         EXH. RAD. - HIGH                      ├-- EITHER        2
                                               │
K618B    FUEL HANDLING AREA POOL  --┐          │
         SWEEP EXH. RAD. - HIGH     ├-- BOTH --┘
                                    │
K618C    FUEL HANDLING AREA POOL  --┘
         SWEEP EXH. RAD. - HIGH

K617A    FUEL HANDLING AREA VENT. --┐
         EXH. RAD. - HIGH           ├-- BOTH --┐
                                    │          │                TRIPS
K617D    FUEL HANDLING AREA VENT. --┘          │               SYSTEM
         EXH. RAD. - HIGH                      ├-- EITHER        1
                                               │
K618A    FUEL HANDLING AREA POOL  --┐          │
         SWEEP EXH. RAD. - HIGH     ├-- BOTH --┘
                                    │
K618D    FUEL HANDLING AREA POOL  --┘
         SWEEP EXH. RAD. - HIGH
```

CONTROL ROOM FRESH AIR INSTRUMENTATION

```
K621B    CONTROL ROOM VENT. RAD. - HIGH --┐            TRIPS
                                          ├-- BOTH    SYSTEM
K621C    CONTROL ROOM VENT. RAD. - HIGH --┘              2

K621A    CONTROL ROOM VENT. RAD. - HIGH --┐            TRIPS
                                          ├-- BOTH    SYSTEM
K621D    CONTROL ROOM VENT. RAD. - HIGH --┘              1
```

ENCLOSURE 2

PROPOSED MODIFICATIONS TO THE

BWR ACTUATION INSTRUMENTATION

STANDARD TECHNICAL SPECIFICATIONS

3/4.3.9 PLANT SYSTEMS ACTUATION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.9 The plant systems actuation instrumentation channels shown in Table 3.3.9-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.9-2.

APPLICABILITY: As shown in Table 3.3.9-1.

ACTION:

a. With a plant system actuation instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.9-2, declare the channel inoperable and either place the inoperable channel in the tripped condition until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value, or declare the associated system inoperable.

b. For the suppression pool (and drywell) spray system:

   1. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement for one trip system, place at least one inoperable channel in the tripped condition within one hour or declare the associated system inoperable.

   2. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement for both trip systems, declare the associated system inoperable.

c. For the feedwater system/main turbine trip system:

   1. With the number of OPERABLE channels one less than required by the Minimum OPERABLE Channels requirement, restore the inoperable channel to OPERABLE status within 7 days or be in at least STARTUP within the next 6 hours.

   2. With the number of OPERABLE channels two less than required by the Minimum OPERABLE Channels requirement, restore at least one of the inoperable channels to OPERABLE status within 72 hours or be in at least STARTUP within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.3.9.1 Each plant system actuation instrumentation channel shall be demonstrated OPERABLE* by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations for the OPERATIONAL CONDITIONS and at the frequencies shown in Table 4.3.9.1-1.

4.3.9.2 LOGIC SYSTEM FUNCTIONAL TESTS and simulated automatic operation of all channels shall be performed at least once per 18 months.

*A CHANNEL MAY BE PLACED IN AN INOPERABLE STATUS FOR UP TO 6 HOURS FOR REQUIRED SURVEILLANCE WITHOUT PLACING THE TRIP SYSTEM IN THE TRIPPED CONDITION.

TABLE 3.3.9-1

PLANT SYSTEMS ACTUATION INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM	APPLICABLE OPERATIONAL CONDITIONS
1. SUPPRESSION POOL (AND DRYWELL) SPRAY SYSTEM		
a. Drywell Pressure-High	1	1, 2, 3
b. Containment Pool Pressure-High	1	1, 2, 3
c. Reactor Vessel Water Level - Low Low Low, Level 1	1	1, 2, 3
d. Timers		
1) System A	1	1, 2, 3
2) System B	1	1, 2, 3
	MINIMUM OPERABLE CHANNELS	
2. FEEDWATER SYSTEM/MAIN TURBINE TRIP SYSTEM		
a. Reactor Vessel Water Level-High, Level (8)	2	1

TABLE 3.3.9-2

PLANT SYSTEMS ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. SUPPRESSION POOL (AND DRYWELL) SPRAY SYSTEM		
a. Drywell Pressure-High	$\leq$ (1.69) psig	$\leq$ (1.89) psig
b. Containment Pressure-High	$\leq$ (35) psig	$\leq$ ( ) psig
c. Reactor Vessel Water Level - Low Low Low, Level 1	$\geq$ -( ) psig	$\geq$ -( ) psig
d. Timers		
1) System A	$\leq$ (12) minutes	$\leq$ (13.2) minutes
2) System B	$\leq$ (14) minutes	$\leq$ (15.4) minutes
2. FEEDWATER SYSTEM/MAIN TURBINE TRIP SYSTEM		
a. Reactor Vessel Water Level-High, Level (8)	$\leq$ (54.5) inches*	$\leq$ (56.0) inches

*See Bases Figure B 3/4 3-1.

2

TABLE 4.3.9.1-1 (Continued)

PLANT SYSTEMS ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
1. SUPPRESSION POOL (AND DRYWELL) SPRAY SYSTEM				
a. Drywell Pressure-High	(NA)	~~(M)~~ Q	(Q)	1, 2, 3
b. Containment Pressure-High	(NA)	~~(M)~~ Q	(Q)	1, 2, 3
c. Reactor Vessel Water Level-Low Low Low, Level 1	(NA)	~~(M)~~ Q	(Q)	1, 2, 3
d. Timers	(NA)	~~(M)~~ Q	(R)	1, 2, 3
2. FEEDWATER SYSTEM/MAIN TURBINE TRIP SYSTEM				
a. Reactor Vessel Water Level-High, Level (8)	(NA)	~~(M)~~ Q	(R)	1

END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.4.2 The end-of-cycle recirculation pump trip (EOC-RPT) system instrumentation channels shown in Table 3.3.4.2-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.4.2-2 and with the END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME as shown in Table 3.3.4.2-3.

APPLICABILITY: OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to (30)% of RATED THERMAL POWER.

ACTION:

a. With an end-of-cycle recirculation pump trip system instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.4.2-2, declare the channel inoperable until the channel is restored to OPERABLE status with the channel setpoint adjusted consistent with the Trip Setpoint value.

b. With the number of OPERABLE channels one less than required by the Minimum OPERABLE Channels per Trip System requirement for one or both trip systems, place the inoperable channel(s) in the tripped condition within ~~one hour~~ 12 HOURS.

c. With the number of OPERABLE channels two or more less than required by the Minimum OPERABLE Channels per Trip System requirement for one trip system and:

   1. If the inoperable channels consist of one turbine control valve channel and one turbine stop valve channel, place both inoperable channels in the tripped condition within ~~one hour~~ 12 HOURS.

   2. If the inoperable channels include two turbine control valve channels or two turbine stop valve channels, declare the trip system inoperable.

d. With one trip system inoperable, restore the inoperable trip system to OPERABLE status within 72 hours or (take the ACTION required by Specification 3.2.3) (reduce THERMAL POWER to less than (30)% of RATED THERMAL POWER within the next 6 hours).

e. With both trip systems inoperable, restore at least one trip system to OPERABLE status within one hour or (take the ACTION required by Specification 3.2.3) (reduce THERMAL POWER to less than (30)% of RATED THERMAL POWER within the next 6 hours).

SURVEILLANCE REQUIREMENTS

4.3.4.2.1 Each end-of-cycle recirculation pump trip system instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3.4.2.1-1.

4.3.4.2.2. LOGIC SYSTEM FUNCTIONAL TESTS and simulated automatic operation of all channels shall be performed at least once per 18 months.

4.3.4.2.3 The END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME of each trip function shown in Table 3.3.4.2-3 shall be demonstrated to be within its limit at least once per 18 months. Each test shall include at least the logic of one type of channel input, turbine control valve fast closure or turbine stop valve closure, such that both types of channel inputs are tested at least once per 36 months. (The time allotted for breaker arc suppression, ( ) ms, shall be verified by test at least once per 60 months.)

TABLE 3.3.4.2-1

END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM[a]
1. Turbine Stop Valve - Closure	2[b]
2. Turbine Control Valve-Fast Closure	2[b]

[a] A trip system may be placed in an inoperable status for up to ~~2~~ 6 hours for required surveillance provided that the other trip system is OPERABLE.

[b] This function shall be automatically bypassed when turbine first stage pressure is less than or equal to (   ) psig, equivalent to THERMAL POWER less than (30)% of RATED THERMAL POWER.

TABLE 3.3.4.2-2

END-OF-CYCLE RECIRCULATION PUMP TRIP SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. Turbine Stop Valve-Closure	$\leq$ (5)% closed	$\leq$ (7)% closed
2. Turbine Control Valve-Fast Closure	$\geq$ (500) psig	$\geq$ (414) psig

TABLE 3.3.4.2-3

END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME

TRIP FUNCTION	RESPONSE TIME (Milleseconds)
1. Turbine Stop Valve-Closure	$\leq$ (100)
2. Turbine Control Valve-Fast Closure	$\leq$ (100)

TABLE 4.3.4.2.1-1

END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION
1. Turbine Stop Valve-Closure	~~M(*)~~ Q	R
2. Turbine Control Valve-Fast Closure	~~M(*)~~ Q	(R)

(*Including trip system logic testing.)

3/4.3.4 RECIRCULATION PUMP TRIP ACTUATION INSTRUMENTATION

ATWS RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.4.1 The anticipated transient without scram recirculation pump trip (ATWS-RPT) system instrumentation channels shown in Table 3.3.4.1-1 shall be OPERABLE with their trip setpoints set consistent with values shown in the Trip Setpoint column of Table 3.3.4.1-2.

APPLICABILITY: OPERATIONAL CONDITION 1.

ACTION:

a. With an ATWS-RPT system instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.4.1-2, declare the channel inoperable until the channel is restored to OPERABLE status with the channel setpoint adjusted consistent with the Trip Setpoint value.

b. With the number of OPERABLE channels one less than required by the Minimum OPERABLE Channels per Trip System requirement for one or both Trip System(s), restore the inoperable channel(s) to OPERABLE status within 14 days or be in at least STARTUP within the next 6 hours.

c. With the number of OPERABLE channels two or more less than required by the Minimum OPERABLE channels per Trip System requirement for ~~one~~ trip system and:

   1. If the inoperable channels consist of one reactor vessel water level channel and one reactor vessel pressure channel, place both inoperable channels in the tripped condition* within ~~one hour~~ 24 HOURS or declare the trip system inoperable.

   2. If the inoperable channels include two reactor vessel water level channels or two reactor vessel pressure channels, declare the trip system inoperable.

d. With one trip system inoperable, restore the inoperable trip system to OPERABLE status within 72 hours or be in at least STARTUP within the next 6 hours.

e. With both trip systems inoperable, restore at least one trip system to OPERABLE status within one hour or be in at least STARTUP within the next 6 hours.

*The inoperable channels need not be placed in the tripped condition where this would cause the Trip Function to occur.

SURVEILLANCE REQUIREMENTS

4.3.4.1.1 Each ATWS recirculation pump trip system instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST, and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3.4.1-1.

4.3.4.1.2 LOGIC SYSTEM FUNCTIONAL TESTS and simulated automatic operation of all channels shall be performed at least once per 18 months.

TABLE 3.3.4.1-1

ATWS RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM[a]
1. Reactor Vessel Water Level - Low Low, Level 2	2
2. Reactor Vessel Pressure - High	2

6 HOURS

(a) One channel may be placed in an inoperable status for up to ~~2 hours~~ for required surveillance provided the other channel is OPERABLE.

TABLE 3.3.4.1-2

ATWS RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. Reactor Vessel Water Level - Low Low, Level 2	$\geq$ -41.6 inches*	$\geq$ -43.8 inches
2. Reactor Vessel Pressure - High	$\leq$ 1095 psig	$\leq$ 1102 psig

*See Bases Figure B3/4 3-1.

TABLE 4.3.4.1-1

ATWS RECIRCULATION PUMP TRIP ACTUATION INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION
1. Reactor Vessel Water Level - Low Low, Level 2	S	~~M~~ Q	R*
2. Reactor Vessel Pressure - High	S	~~M~~ Q	R*

*Calibrate trip unit at least once per ~~31~~ 92 days.

3/4.4.2 SAFETY/RELIEF VALVES

SAFETY/RELIEF VALVES

LIMITING CONDITION FOR OPERATION

3.4.2.1 (At least (two) reactor coolant system code safety valves and) the safety valve function of at least (11) (of the following) reactor coolant system safety/relief valves shall be OPERABLE with the specified code safety valve function lift settings:*

(2) safety valves @ (1146) psig ±1%
(3) safety-relief valves @ (1175) psig ±1%
(3) safety-relief valves @ (1185) psig ±1%
(3) safety-relief valves @ (1195) psig ±1%
(2) safety-relief valves @ (1205) psig ±1%

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With (one or more of the above required reactor coolant system code safety valves or with) the safety valve function of one or more of the above required safety/relief valves inoperable, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

b. With one or more (code safety valves or) safety/relief valves stuck open, provided that suppression pool average water temperature is less than (95)°F, close the stuck open (code safety valves and/or) safety relief valve(s); if unable to close the stuck open valve(s) within 2 minutes or if suppression pool average water temperature is (95)°F or greater, place the reactor mode switch in the Shutdown position.

c. With one or more safety/relief valve (tail-pipe pressure switches) (acoustic monitors) inoperable, restore the inoperable (switch(es)) (monitor(s)) to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

*The lift setting pressure shall correspond to ambient conditions of the valves at nominal operating temperatures and pressures.

(4.4.2.1.1 (The code safety valve function of each of the above required safety/relief valves shall be demonstrated OPERABLE by verifying that the bellows on the safety/relief valves have integrity, by instrumentation indication, at least once per 24 hours.)

4.4.2.1.2 The (tail-pipe pressure switch) (acoustic monitor) for each safety/relief valve shall be demonstrated OPERABLE** with the setpoint verified to be ((20) ± (5) psig) by performance of a:

a. CHANNEL (FUNCTIONAL TEST) (CHECK) at least once per ~~31~~ 92 days, and a

b. CHANNEL CALIBRATION at least once per 18 months(*).

(*The provisions of Specification 4.0.4 are not applicable provided the Surveillance is performed within 12 hours after reactor steam pressure is adequate to perform the test.)

** A CHANNEL MAY BE PLACED IN AN INOPERABLE STATUS FOR UP TO 6 HOURS FOR REQUIRED SURVEILLANCE WITHOUT PLACING THE TRIP SYSTEM IN THE TRIPPED CONDITION.

SAFETY/RELIEF VALVES LOW-LOW SET FUNCTION

LIMITING CONDITION FOR OPERATION

3.4.2.2 The relief valve function and the low-low set function of the following reactor coolant system safety/relief valves shall be OPERABLE with the following settings:

Valve No.	Low-Low Set Function Setpoint* (psig) ± 1% Open	Low-Low Set Function Setpoint* (psig) ± 1% Close	Relief Function Setpoint* (psig) ± 1% Open	Relief Function Setpoint* (psig) ± 1% Close
______	(1033)	(926)	______	______
______	(1073)	(936)	______	______
______	(1113)	(946)	______	______
______	(1113)	(946)	______	______
______	(1113)	(946)	______	______

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With the relief valve function and/or the low-low set function of one of the above required reactor coolant system safety/relief valves inoperable, restore the inoperable relief valve function and low-low set function to OPERABLE status within 14 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With the relief valve function and/or the low-low set function of more than one of the above required reactor coolant system safety/relief valves inoperable, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

SURVEILLANCE REQUIREMENTS

4.4.2.2.1 The relief valve function and the low-low set function pressure actuation instrumentation shall be demonstrated OPERABLE** by performance of a:

a. CHANNEL FUNCTIONAL TEST, including calibration of the trip unit, at least once per ~~31~~ 92 days.

b. CHANNEL CALIBRATION, LOGIC SYSTEM FUNCTIONAL TEST and simulated automatic operation of the entire system at least once per 18 months.

** A CHANNEL MAY BE PLACED IN AN INOPERABLE STATUS FOR UP TO 6 HOURS FOR REQUIRED SURVEILLANCE WITHOUT PLACING THE TRIP SYSTEM IN THE TRIPPED CONDITION.

*The lift setting pressure shall correspond to ambient conditions of the valves at nominal operating temperatures and pressures.

REACTOR COOLANT SYSTEM

3/4.4.2 SAFETY VALVES

SAFETY/RELIEF VALVES

LIMITING CONDITION FOR OPERATION

3.4.2.1 Of the following safety/relief valves, the safety valve function of at least (6) valves and the relief valve function of at least (5) valves other than those satisfying the safety valve function requirement shall be OPERABLE with the specified lift settings:

Number of Valves	Function	Setpoint* (psig) ± 1%
(7)	(Safety)	(1165)
(5)	(Safety)	(1180)
(4)	(Safety)	(1190)
(1)	(Relief)	(1103)
(8)	(Relief)	(1113)
(7)	(Relief)	(1123)

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With the safety and/or relief valve function of one or more of the above required safety/relief valves inoperable, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.
b. With one or more safety/relief valves stuck open, provided that suppression pool average water temperature is less than (105)°F, close the stuck open safety/relief valve(s); if unable to close the open valve(s) within 2 minutes or if suppression pool average water temperature is (105)°F or greater, place the reactor mode switch in the Shutdown position.
c. With one or more safety/relief valve (tail-pipe pressure switches) (acoustic monitors) inoperable, restore the inoperable (switch(es)) (monitor(s)) to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.4.2.1.1 The (tail-pipe pressure switch) (acoustic monitor) for each safety/relief valve shall be demonstrated OPERABLE *** with the setpoint verified to be ((20) ± (5) psig) (   ) by performance of a:

a. CHANNEL (FUNCTIONAL TEST) (CHECK) at least once per ~~(**) 31~~ 92 days, and a
b. CHANNEL CALIBRATION at least once per 18 months.

4.4.2.1.2 The relief valve function pressure actuation instrumentation shall be demonstrated OPERABLE *** by performance of a:

a. CHANNEL FUNCTIONAL TEST, including calibration of the trip unit, at least once per ~~31~~ 92 days.
b. CHANNEL CALIBRATION, LOGIC SYSTEM FUNCTIONAL TEST and simulated automatic operation of the entire system at least once per 18 months.

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*The lift setting pressure shall correspond to ambient conditions of the valves at nominal operating temperatures and pressures.

(**The provisions of Specification 4.0.4 are not applicable provided the surveillance is performed within 12 hours after reactor steam pressure is adequate to perform the test.)

*** A CHANNEL MAY BE PLACED IN AN INOPERABLE STATUS FOR UP TO 6 HOURS FOR REQUIRED SURVEILLANCE WITHOUT PLACING THE TRIP SYSTEM IN THE TRIPPED CONDITION.

SAFETY/RELIEF VALVES LOW-LOW SET FUNCTION

LIMITING CONDITION FOR OPERATION

3.4.2.2 The relief valve function and the low-low set function of the following reactor coolant system safety/relief valves shall be OPERABLE with the following settings:

Valve No.	Low-Low Set Function Setpoint* (psig) ± 1%		Relief Function Setpoint* (psig) ± 1%	
	Open	Close	Open	Close
______	(1033)	(926)	______	______
______	(1073)	(936)	______	______
______	(1113)	(946)	______	______
______	(1113)	(946)	______	______
______	(1113)	(946)	______	______

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With the relief valve function and/or the low-low set function of one of the above required reactor coolant system safety/relief valves inoperable, restore the inoperable relief valve function and the low-low set function to OPERABLE status within 14 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With the relief valve function and/or the low-low set function of more than one of the above required reactor coolant system safety/relief valves inoperable, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

SURVEILLANCE REQUIREMENTS

4.4.2.2.1 The relief valve function and the low-low set function pressure actuation instrumentation shall be demonstrated OPERABLE** by performance of a:

a. CHANNEL FUNCTIONAL TEST, including calibration of the trip unit, at least once per ~~31~~ 92 days.

b. CHANNEL CALIBRATION, LOGIC SYSTEM FUNCTIONAL TEST and simulated automatic operation of the entire system at least onc per 18 months.

*The lift setting pressure shall correspond to ambient conditions of the valves at nominal operating temperatures and pressures.

** A CHANNEL MAY BE PLACED IN AN INOPERABLE STATUS FOR UP TO 6 HOURS FOR REQUIRED SURVEILLANCE WITHOUT PLACING THE TRIP SYSTEM IN THE TRIPPED CONDITION

INSTRUMENTATION

3/4.3.8 PLANT SYSTEMS ACTUATION INSTRUMENTATION

INSTRUMENTATION

3/4.3.8 PLANT SYSTEMS ACTUATION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.8 The plant systems actuation instrumentation channels shown in Table 3.3.8-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.8-2.

APPLICABILITY: As shown in Table 3.3.8-1.

ACTION:

a. With a plant system actuation instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.8-2, declare the channel inoperable and take the ACTION required by Table 3.3.8-1.

b. With one or more plant systems actuation instrument channels inoperable, take the ACTION required by Table 3.3.8-1.

SURVEILLANCE REQUIREMENTS

4.3.8.1 Each plant system actuation instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations for the OPERATIONAL CONDITIONS and at the frequencies shown in Table 4.3.8.1-1.

4.3.8.2 LOGIC SYSTEM FUNCTIONAL TESTS and simulated automatic operation of all channels shall be performed at least once per 18 months.

TABLE 3.3.8-1

PLANT SYSTEMS ACTUATION INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM[(a)]	APPLICABLE OPERATIONAL CONDITIONS	ACTION
1. CONTAINMENT SPRAY SYSTEM			
a. Drywell Pressure-High	2	1, 2, 3	130
b. Containment Pressure-High	1	1, 2, 3	131
c. Reactor Vessel Water Level-Low Low Low, Level 1	2	1, 2, 3	130
d. Timers			
1) System A	1	1, 2, 3	131
2) System B	1	1, 2, 3	131
2. FEEDWATER SYSTEM/MAIN TURBINE TRIP SYSTEM			
a. Reactor Vessel Water Level-High, Level 8	3	1	132
3. SUPPRESSION POOL MAKEUP SYSTEM			
a. Drywell Pressure - High (ECCS)	2	1, 2, 3	135
b. Drywell Pressure - High (RPS)	2	1, 2, 3	135
c. Reactor Vessel Water Level - Low Low Low, Level 1	2	1, 2, 3	135
d. Reactor Vessel Water Level - Low Low, Level 2	2	1, 2, 3	135
e. Suppression Pool Water Level - Low Low	1	1, 2, 3	133
f. Suppression Pool Makeup Timer	1	1, 2, 3	133
g. SPMU Manual Initiation	2	1, 2, 3	134

(a)A channel may be placed in an inoperable status for up to ~~2 hours~~ 6 HOURS during periods of required surveillance provided at least one other OPERABLE channel in the same trip system is monitoring that parameter.

TABLE 3.3.8-1 (Continued)

PLANT SYSTEMS ACTUATION INSTRUMENTATION

ACTION

ACTION 130 - a. With the number of OPERABLE channels one less than required by the Minimum OPERABLE Channels per Trip System requirement, place the inoperable channel in the tripped condition within 24 HOURS ~~one hour~~; otherwise, declare the associated containment spray system inoperable and take the action required by Technical Specification 3.6.3.2.

b. With the number of OPERABLE channels two less than required by the Minimum OPERABLE channels per Trip System requirement, declare the associated containment spray system inoperable and take the action required by Technical Specification 3.6.3.2.

ACTION 131 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement, restore the inoperable channels to OPERABLE status within one hour; otherwise, declare the associated containment spray system inoperable and take the action required by Technical Specification 3.6.3.2.

ACTION 132 - For the feedwater system/main turbine trip system:

a. With the number of OPERABLE channels one less than required by the Minimum OPERABLE Channels requirement, restore the inoperable channel to OPERABLE status within 7 days or be in at least STARTUP within the next 6 hours.

b. With the number of OPERABLE channels two less than required by the Minimum OPERABLE Channels per Trip System requirement, restore at least one of the inoperable channels to OPERABLE status within 72 hours or be in at least STARTUP within the next 6 hours.

ACTION 133 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement, declare the associated suppression pool makeup system inoperable and take the action required by Specification 3.6.3.4.

ACTION 134 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement, restore the inoperable channels to OPERABLE status within ~~6~~ 24 hours; otherwise, declare the associated suppression pool makeup system inoperable and take the action required by Specification 3.6.3.4.

ACTION 135 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement:

a. With one channel inoperable, place the inoperable channel in the tripped condition within ~~one hour~~ 24 HOURS or ~~declare the~~ associated system(s) inoperable.

b. With more than one channel inoperable, declare the associated system(s) inoperable.

TABLE 3.3.8-2

PLANT SYSTEMS ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. CONTAINMENT SPRAY SYSTEM		
a. Drywell Pressure-High	≤ 1.39 psig	≤ 1.44 psig
b. Containment Pressure-High	≤ 7.84 psig	≤ 8.34 psig
c. Reactor Vessel Water Level-Low Low Low, Level 1	≥ - 150.3 inches	≥ - 152.5 inches
d. Timers		
1) System A	10.85 ± 0.10 minutes	10.26 - 0.00, + 1.18 minutes
2) System B	10.85 ± 0.10 minutes**	10.26 - 0.00, + 1.18 minutes
2. FEEDWATER SYSTEM/MAIN TURBINE TRIP SYSTEM		
a. Reactor Vessel Water Level-High, Level 8	≤ 53.5 inches*	≤ 54.1 inches
3. SUPPRESSION POOL MAKEUP SYSTEM		
a. Drywell Pressure - High (ECCS)	≤ 1.39 psig	≤ 1.44 psig
b. Drywell Pressure - High (RPS)	≤ 1.23 psig	≤ 1.43 psig
c. Reactor Vessel Water Level - Low Low Low, Level 1	≥ -150.3 inches*	≥ -152.5 inches
d. Reactor Vessel Water Level - Low Low, Level 2	≥ -41.6 inches*	≥ -43.8 inches
e. Suppression Pool Water Level - Low Low	≥ 17 ft 5 inches	≥ 17 ft 2 inches
f. Suppression Pool Makeup Timer	≤ 29.0 minutes	≤ 29.5 minutes
g. SPMU Manual Initiation	NA	NA

*See Bases Figure B 3/4 3-1.
**Setpoint for System B is the sum of E12-K093B plus E12-K116. E12-K116 is not to exceed 10.00 seconds.

TABLE 4.3.8.1-1

PLANT SYSTEMS ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS IN WHICH SURVEILLANCE REQUIRED
1. CONTAINMENT SPRAY SYSTEM				
a. Drywell Pressure-High	S	~~M~~ Q	[illegible]	1, 2, 3
b. Containment Pressure-High	S	~~M~~ Q	$R^{(a)}$	1, 2, 3
c. Reactor Vessel Water Level - Low Low Low, Level 1	S	~~M~~ Q	$R^{(a)}$	1, 2, 3
d. Timers	NA	~~M~~ Q	Q	1, 2, 3
2. FEEDWATER SYSTEM/MAIN TURBINE TRIP SYSTEM				
a. Reactor Vessel Water Level-High, Level 8	S	~~M~~ Q	R	1
3. SUPPRESSION POOL MAKEUP SYSTEM				
a. Drywell Pressure - High (ECCS)	S	~~M~~ Q	$R^{(a)}$	1, 2, 3
b. Drywell Pressure - High (RPS)	S	~~M~~ Q	$R^{(a)}$	1, 2, 3
c. Reactor Vessel Water Level - Low Low Low, Level 1	S	~~M~~ Q	$R^{(a)}$	1, 2, 3
d. Reactor Vessel Water Level - Low Low, Level 2	S	~~M~~ Q	$R^{(a)}$	1, 2, 3
e. Suppression Pool Water Level - Low Low	S	~~M~~ Q	$R^{(a)}$	1, 2, 3
f. Suppression Pool Makeup Timer	NA	~~M~~ Q	Q	1, 2, 3
g. SPMU Manual Initiation	NA	R	NA	1, 2, 3

(a)Calibrate trip unit at least once per ~~31~~ 92 days.

MAIN CONTROL ROOM ENVIRONMENTAL CONTROL SYSTEM (MCRECS) ACTUATION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.6.7 The MCRECS actuation instrumentation channels shown in Table 3.3.6.7-1 shall be OPERABLE, with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.6.7-2.

APPLICABILITY: As shown in Table 3.3.6.7-1.

ACTION: As shown in Table 3.3.6.7-1.

SURVEILLANCE REQUIREMENTS

4.3.6.7 Each MCRECS actuation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST, and CHANNEL CALIBRATION operations during the OPERATIONAL CONDITION and at the frequencies shown in Table 4.3.6.7-1.

TABLE 3.3.6.7-1 (SHEET 1 OF 2)

MCRECS ACTUATION INSTRUMENTATION

TRIP FUNCTION	MINIMUM NUMBER OPERABLE CHANNELS PER TRIP SYSTEM(a)(b)	APPLICABLE OPERATIONAL CONDITION	ACTION
1. Reactor Vessel Water Level - Low Low Low (Level 1) (c) 2B21-N691 A, B, C, D	2	1, 2, 3	52
2. Drywell Pressure - High (c) 2E11-N694 A, B, C, D	2	1, 2, 3	52
3. Main Steam Line Radiation - High (c) 2D11-K603 A, B, C, D	2	1, 2, 3,(e)	53
4. Main Steam Line Flow - High (c) 2B21-N686 A, B, C, D 2B21-N687 A, B, C, D 2B21-N688 A, B, C, D 2B21-N689 A, B, C, D	2/line	1, 2, 3	53
5. Refueling Floor Area Radiation - High (c) 2D21-K002 A, B	1	1, 2, 3, 5, *	54
6. Control Room Air Inlet Radiation - High (c) 1Z41-R615 A, B	1	1, 2, 3, 5, *	54

MCRECS ACTUATION INSTRUMENTATION

ACTION

ACTION 52 - Take the ACTION required by Specification 3.3.3.

ACTION 53 - Take the ACTION required by Specification 3.[illegible].2.

ACTION 54 -

a. With one of the required radiation monitors inoperable, restore the monitor to OPERABLE status within 7 days or, within the next 6 hours, initiate and maintain operation of the MCRECS in the pressurization mode of operation.

b. With no radiation monitors OPERABLE, within 1 hour initiate and maintain operation of the MCRECS in the pressurization mode of operation.

c. The provisions of Specification 3.0.4 are not applicable.

NOTES

* When handling irradiated fuel in secondary containment.

6 HOURS

a. A channel may be placed in an inoperable status for up to ~~2 hours~~ for required surveillance without placing the trip system in the tripped condition, provided at least one other OPERABLE channel in the same trip system is monitoring that parameter.

b. With a design providing only one channel per trip system, an inoperable channel need not be placed in the tripped condition where this would cause the Trip Function to occur. In these cases, the inoperable channel shall be restored to OPERABLE status within ~~2 hours~~ or the ACTION required by Table 3.3.6.7-1 for that Trip Function shall be taken.

24 HOURS

c. Actuates the MCRECS in the control room pressurization mode.

d. (Deleted)

e. Within 24 hours prior to the planned start of the hydrogen injection test with the reactor power at greater than 20-percent rated power, the normal full-power radiation background level and associated trip setpoints may be changed based on a calculated value of the radiation level expected during the test. The background radiation level and associated trip setpoints may be adjusted during the test based on either calculations or measurements of actual radiation levels resulting from hydrogen injection. The background radiation level shall be determined and associated trip setpoints shall be set within 24 hours of re-establishing normal radiation levels after completion of hydrogen injection and prior to establishing reactor power levels below 20-percent rated power.

TABLE 3.3.6.7-2

MCRECS ACTUATION INSTRUMENTATION SETPOINTS

	TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1.	Reactor Vessel Water Level - Low Low Low (Level 1)	≥ -113 inches	≥ - 113 inches
2.	Drywell Pressure - High	≤ 1.92 psig	≤ 1.92 psig
3.	Main Steam Line Radiation - High	≤ 3 × full-power background*	≤ 3 × full-power background*
4.	Main Steam Line Flow - High	≤ 138% rated flow	≤ 138% rated flow
5.	Refueling Floor Area Radiation - High	≤ 20 mr/hour	≤ 20 mr/hour
6.	Control Room Air Inlet Radiation - High	≤ 1 mr/hour	≤ 1 mr/hour

*Within 24 hours prior to the planned start of the hydrogen injection test with the reactor power at greater than 20-percent rated power, the normal full power radiation background level and associated trip setpoints may be changed based on a calculated value of the radiation level expected during the test. The background radiation level and associated trip setpoints may be adjusted during the test based on either calculations or measurements of actual radiation levels resulting from hydrogen injection. The background radiation level shall be determined and associated trip setpoints shall be set within 24 hours of re-establishing normal radiation levels after completion of hydrogen injection and prior to establishing reactor power levels below 20-percent rated power.

TABLE 4.3.6.7-1

MCRECS ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

	TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS IN WHICH SURVEILLANCE REQUIRED
1.	Reactor Vessel Water Level - Low Low Low (Level 1)	S	~~M~~ Q	R	1, 2, 3
2.	Drywell Pressure - High	S	~~M~~ Q	R	1, 2, 3
3.	Main Steam Line Radiation - High	D	W(a)	R	1, 2, 3
4.	Main Steam Line Flow - High	S	~~M~~ Q	R	1, 2, 3
5.	Refueling Floor Area Radiation - High	D	~~W(a)~~ Q	Q	1, 2, 3, 5 *
6.	Control Room Air Inlet Radiation - High	NA	~~W(a)~~ Q	R	1, 2, 3, 5, *

* When handling irradiated fuel in the secondary containment.

a. Instrument alignment using a standard current source.

INSTRUMENTATION

3/4.3.6 CONTROL ROD BLOCK INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.6. The control rod block instrumentation channels shown in Table 3.3.6-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.6-2.

APPLICABILITY: As shown in Table 3.3.6-1.

ACTION:

a. With a control rod block instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.6-2, declare the channel inoperable until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

b. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, take the ACTION required by Table 3.3.6-1.

SURVEILLANCE REQUIREMENTS

4.3.6 Each of the above required control rod block trip systems and instrumentation channels shall be demonstrated OPERABLE* by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations for the OPERATIONAL CONDITIONS and at the frequencies shown in Table 4.3.6-1.

*A CHANNEL MAY BE PLACED IN AN INOPERABLE STATUS FOR UP TO 6 HOURS FOR REQUIRED SURVEILLANCE WITHOUT PLACING THE TRIP SYSTEM IN THE TRIPPED CONDITION, PROVIDED AT LEAST ONE OTHER OPERABLE CHANNEL IN THE SAME TRIP SYSTEM IS MONITORING THAT PARAMETER.

GE-STS (BWR/4)

TABLE 3.3.6-1

CONTROL ROD BLOCK INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP FUNCTION	APPLICABLE OPERATIONAL CONDITIONS	ACTION
1. ROD BLOCK MONITOR(a)			
a. Upscale	2	1*	60
b. Inoperative	2	1*	60
c. Downscale	2	1*	60
2. APRM			
a. Flow Biased Neutron Flux - Upscale	4	1	61
b. Inoperative	4	1, 2, 5	61
c. Downscale	4	1	61
d. Neutron Flux - Upscale, Startup	4	2, 5	61
3. SOURCE RANGE MONITORS			
a. Detector not full in(b)	3	2	61
	2	5	61
b. Upscale(c)	3	2	61
	2	5	61
c. Inoperative(c)	3	2	61
	2	5	61
d. Downscale(d)	3	2	61
	2	5	61
4. INTERMEDIATE RANGE MONITORS			
a. Detector not full in ((e))	6	2, 5	61
b. Upscale	6	2, 5	61
c. Inoperative	6	2, 5	61
d. Downscale(e)	6	2, 5	61
5. SCRAM DISCHARGE VOLUME			
a. Water Level-High	(2)	1, 2, 5**	62
b. Scram Trip Bypass	(2)	(1, 2,) 5**	62
6. REACTOR COOLANT SYSTEM RECIRCULATION FLOW			
a. Upscale	2	1	62
b. Inoperative	2	1	62
c. (Comparator) (Downscale)	2	1	62

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TABLE 3.3.6-1 (Continued)

CONTROL ROD BLOCK INSTRUMENTATION

ACTION

ACTION 60 - Declare the RBM inoperable and take the ACTION required by Specification 3.1.4.3.

ACTION 61 - With the number of OPERABLE Channels:

a. One less than required by the Minimum OPERABLE Channels per Trip Function requirement, restore the inoperable channelto OPERABLE status within 7 days or place the inoperable channel in the tripped condition within the next hour.

b. Two or more less than required by the Minimum OPERABLE Channels per Trip Function requirement, place at least one inoperable channel in the tripped condition within one hour.

ACTION 62 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, place the inoperable channel in the tripped condition within ~~one hour~~ 12 HOURS.

NOTES

* With THERMAL POWER ≥ (30)% of RATED THERMAL POWER.

** With more than one control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

a. The RBM shall be automatically bypassed when a peripheral control rod is selected (or the reference APRM channel indicates less than (30)% of RATED THERMAL POWER).

b. This function shall be automatically bypassed if detector count rate is > 100 cps or the IRM channels are on range (3) or higher.

c. This function shall be automatically bypassed when the associated IRM channels are on range 8 or higher.

d. This function shall be automatically bypassed when the IRM channels are on range 3 or higher.

e. This function shall be automatically bypassed when the IRM channels are on range 1.

TABLE 3.3.6-2

CONTROL ROD BLOCK INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. ROD BLOCK MONITOR		
a. Upscale	≤ 0.66 W + (40)%	≤ 0.66 W + (43)%
b. Inoperative	NA	NA
c. Downscale	≥ (5)% of RATED THERMAL POWER	≥ (3)% of RATED THERMAL POWER
2. APRM		
a. Flow Biased Neutron Flux - Upscale	≤ 0.66 W + (42)%*	≤ 0.66 W + (45)%*
b. Inoperative	NA	NA
c. Downscale	≥ (5)% of RATED THERMAL POWER	≥ (3)% of RATED THERMAL POWER
d. Neutron Flux - Upscale, Startup	≤ (12)% of RATED THERMAL POWER	≤ (14)% of RATED THERMAL POWER
3. SOURCE RANGE MONITORS		
a. Detector not full in	NA	NA
b. Upscale	≤ (2 x 10^5) cps	≤ (5 x 10^5) cps
c. Inoperative	NA	NA
d. Downscale	≥ (3) cps	≥ (2) cps
4. INTERMEDIATE RANGE MONITORS		
a. Detector not full in	NA	NA
b. Upscale	≤ (108/125) divisions of full scale	≤ (110/125) divisions of full scale
c. Inoperative	NA	NA
d. Downscale	≥ (5/125) divisions of full scale	≥ (3/125) divisions of full scale
5. SCRAM DISCHARGE VOLUME		
a. Water Level-High	≤ ( ) inches	≤ ( ) inches
b. Scram Trip Bypass	NA	NA
6. REACTOR COOLANT SYSTEM RECIRCULATION FLOW		
a. Upscale	≤ (108/125) divisions of full scale	≤ (111/125) divisions of full scale
b. Inoperative	NA	NA
c. (Comparator) (Downscale)	≤ (10)% flow deviation	≤ (11)% flow deviation

*The Average Power Range Monitor rod block function is varied as a function of recirculation loop flow (W). The trip setting of this function must be maintained in accordance with Specification 3.2.2.

TABLE 4.3.6-1 (Continued)

CONTROL ROD BLOCK INSTRUMENTATION SURVEILLANCE REQUIREMENTS

NOTES:

a. Neutron detectors may be excluded from CHANNEL CALIBRATION.

b. Within 24 hours prior to startup, if not performed within the previous 7 days.

c. Includes reactor manual control multiplexing system input.

* With THERMAL POWER $\geq$ (30)% of RATED THERMAL POWER.

** With more than one control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.