American Electric Power Service Corporation

## Nuclear Safety And Licensing Section
## Calculation Cover Sheet

NUC-15
8199(R10-90)AP25

| | |
|---|---|
| Calculation No. OA-92-18 | Plant Cook Nuclear Plant Unit 1+2 |
| Subject Qualitative Functional Diversity Assessment of UFSAR of Common Mode Failure of Digital Equipment Software | Company Indiana Michigan Co. |
| | Calculated By R.S. Sharma |
| | Verified/Checked By Vance VanderBurg 11Dec92 |
| Safety-Related System Yes X No ___ | Method Of Verification Detailed review |
| Supersedes Calc. No. NA | Approved By JEB K 12/14/92 |

Problem Description: The existing analog reactor protection equipment (H-Line equipment) is being replaced by digital Foxboro Spec 200/spec 200 Micro Electronics instrumentation. The NRC required AEPSC to assume a common mode failure (CMF) of the software of new digital equipment and perform a functional diversity assessment of each UFSAR event given a CMF of the software.

Design Basis Or References:

1. Updated FSAR
2. Technical Specifications
3. Emergency Operating Procedures
4. "Reduced Temp. and Pressure Operation For D.C. Cook Plant for Unit 1 and Unit 2", WCAP-11902
5. Vantage 5 Reload Transition Safety Report of D.C. Cook Plant, Unit 2

Revisions

| No. | Date | Change | Calc. By | Checked By | Date | Approved By | Date |
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Superseded By Calculation No. ______ Dated ______

Reason:

7223(9-83)
FORM GE-8(C)

ENGINEERING DEPT.
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SUBJECT Qualitative Functional Diversity Assessment

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SUBJECT. QUALITATIVE FUNCTIONAL DIVERSITY ASSESSMENT

A. Statement of Purpose and Executive Summary

See page 4/5

B. Assumptions

See Appendix A

C. Analysis

Qualitative Evaluation given in Appendices A and B

D. Verification

The evaluation was done based on U2 FSAR. The reviewer checked Unit 1 FSAR for consistency. WCAP 11902 and its supplement, RTP License Report, WCAP 12135, RTP Engineering Report, WCAP's 12078 and 12901 Input and Output Data, and Unit 2 cycle 8 RTSR were also used as a basis for reviewing the evaluations. Plant annunciator response procedures were used to review possible and probable alarms. Discussions with NED personnel especially I&C personnel resolved various issues such as which alarms were independent of the new digital equipment. Where the reviewer felt it was appropriate or necessary, changes to the evaluation were proposed and resolved with the evaluator.

E. Results

See Appendix A

F. Discussion of Results

See Appendix A

G. References

See Appendix A

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SUBJECT Qualitative Functional Diversity Assessment

STATEMENT OF PURPOSE AND EXECUTIVE SUMMARY

On April 21, 1992, AEPSC representatives had a meeting with the NRC on the replacement of existing analog reactor protection process instrumentation with digital Foxboro Spec 200/Spec 200 Micro Electronics instrumentation. During this meeting, AEPSC was asked to assume a common mode failure (CMF) of the software of the new digital equipment during an accident and then provide details as to whether operators could mitigate the consequences of the accident.

In response to this request, a functional diversity assessment of each updated FSAR (UFSAR) event assuming a common mode failure of the software has been performed. In this assessment, all the events for both Units 1 and 2 of the Cook Nuclear Plant given in the UFSAR were considered. A review was performed to divide events into potentially affected and not affected. Table-1 lists these events and indicates whether they would be potentially affected or not affected, if a CMF were to occur. The potentially affected transients were then individually evaluated qualitatively in light of the FSAR analysis as shown in the attached Appendix A. The transients which are not affected by the software failure are discussed in Appendix B.

The first column of the evaluations in the Appendix A contain the UFSAR transient number listed in Table-1. The second column includes the name of the transient. The third column depicts the trip/safeguard function for reactor trip. This information was obtained from the UFSAR. The fourth column includes the information on the impact of common mode failure on the reactor trip function. If the trip function is processed outside of the new digital reactor protection system, then the trip is available, e.g., trip on nuclear instrumentation system high flux. If the trip is processed by a function that is a part of the new digital equipment, then the trip/ESF function is assumed to be lost. However, for some functions, alternate indications and/or diverse alarms are available. The alarm/alternate indications that are available to the operator to mitigate the transient are given in the next column. The sixth column lists the pertinent diagram numbers. The seventh column summarizes the consequences of the unavailability of diverse alarm. The last column provides the evaluation of the event. In this column, we have discussed the consequences of the operator's response on reactor safety.

Based on this evaluation, we have concluded that the CMF of the new digital equipment has no significant adverse impact on the public safety. Some reactor trips are not affected by the installation of the new digital equipment. Among these trips are neutron high flux and high rate trips, undervoltage and underfrequency trips and reactor trip on turbine trip. However, for events protected by trips and actuations affected by CMF, should a CMF occur, the operator will be alerted to the event by an alarm from a diverse system. He will then provide the appropriate actuations manually and enter the emergency operating procedures. For some accidents, such as locked rotor, the consequences could be more severe than currently analyzed due to the longer response time for the required actuation. However, our evaluation indicates that the affected unit can be brought to a safe condition and the current LOCA offsite dose evaluation will remain bounding. From these results, it is believed that a CMF of the new digital system would have no adverse effect on the health and safety of the public.

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SUBJECT Qualitative Functional Diversity Assessment

Table-1

| UFSAR TRANSIENT # | TRANSIENT | POTENTIALLY AFFECTED (A)/ NOT AFFECTED(NA) |
|---|---|---|
| 14.1.1 | Uncontrolled RCCA Withdrawal from a Subcritical Condition | A |
| 14.1.2 | Uncontrolled RCCA Withdrawal at Power | A |
| 14.1.3 | Rod Cluster Control Assembly Misalignment | A |
| 14.1.4 | RCCA Drop | A |
| 14.1.5 | Chemical Volume and Control System Malfunction | A |
| 14.1.6 | Loss of Reactor Coolant Flow | A |
| 14.1.7 | Startup of an Inactive Reactor Coolant Loop | A |
| 14.1.8 | Loss of External Electrical Load | A |
| 14.1.9 | Loss of Normal Feedwater Flow | A |
| 14.1.10 | Excessive Heat Removal due to Feedwater System Malfunction | A |
| 14.1.11 | Excessive Load Increase Incident | A |
| 14.1.12 | Loss of All A.C. Power to the Plant Auxiliaries | A |
| 14.1.13 | Turbine-Generator Safety Analysis | A |
| 14.2.1 | Fuel Handling Accident | A |
| 14.2.2 | Accidental Release of Radioactive Liquids | A |
| 14.2.3 | Accidental Waste Gases Release | A |
| 14.2.4 | Steam Generator Tube Rupture | A |
| 14.2.5 | Rupture of a Steam Pipe | A |
| 14.2.6 | Rupture of a Control Rod Drive Mechanism Housing (RCCA Ejection) | A |
| 14.2.7 | Secondary System Accidents Dose Consequences | A |
| 14.2.8 | Major Rupture of a Main Feedwater Pipe | A |
| 14.3.1 | Large Break LOCA Analysis | A |
| 14.3.2 | Loss of Reactor Coolant from Small Ruptured Pipes or from Cracks in Large Pipes which Actuates the Emergency Core Cooling System | A |
| 14.3.3 | Core and Internals Integrity Analysis | NA |
| 14.3.4 | Containment Integrity Analysis | A |
| 14.3.5 | Environmental Consequences of a Loss of Coolant Accident | A |
| 14.3.6 | Hydrogen in the Containment After a Loss of Coolant Accident | A |
| 14.3.7 | Long Term Cooling | NA |
| 14.3.8 | Nitrogen Blanketing | NA |
| 14.4.2 | Postulated Pipe Failure Analysis Outside Containment | NA |
| 14.4.3 | Analysis of Emergency Conditions | NA |
| 14.4.4 | Stress Calculations | NA |
| 14.4.5 | Description of Pipe Whip Analysis | NA |
| 14.4.6 | Compartment Pressures and Temperatures | NA |
| 14.4.7 | Description of Jet Impingement Load Analysis | NA |
| 14.4.8 | Containment Integrity | NA |
| 14.4.9 | Plant Modifications | NA |
| 14.4.10 | Environment | NA |
| 14.4.11 | Electrical Equipment Environmental Qualification | A |

# APPENDIX A

| FSAR TRANSIENT # | TRANSIENT | TRIP/SAFEGUARD FUNCTION FOR RX TRIP (FSAR 14.1.1) | IMPACT OF COMMON MODE FAILURE (CMF) ON TRIP FUNCTION | ALARM/ALTERNATE INDICATION SYSTEM AVAILABLE | DIAGRAM # | CONSEQUENCES OF UNAVAILABILITY OF DIVERSE ALARM | EVALUATION OF EVENT |
|---|---|---|---|---|---|---|---|
| 14.1.1 | Uncontrolled RCCA Bank Withdrawal from a Subcritical Condition | 1. Source range neutron flux trip-actuated when either of 2 independent source range channels indicates a flux above a preselected, manually adjustable value.<br>2. Intermediate range neutron flux trip actuated when either of two independent intermediate range channels indicates a flux above a preselected, manually adjustable value.<br>3. Power range high neutron flux trip (low setting) - actuated when two out of 4 power channels indicate a flux above approximately 25% of full power flux.<br>4. Power range neutron flux level trip (high setting) - actuated when 2 out of 4 power range channels indicate a flux level above a preset setpoint. | Item Nos. 1-4 not affected (Memo dated Sept. 2, 1992 from W. G. Sotos to V. D. VanderBurg, T/S Table 3.3-1) | | | Not Affected | This transient is not affected by the replacement of H-line analog process protection system by Foxboro SPEC 200 and SPEC 200 MICRO, microprocess based modules. Trips 1 through 4, listed in Column 3, are not affected, since nuclear instrumentation for flux measurement is not replaced. For Rx trip from pressurizer high pressure, two diverse alarms are available. In addition, pressurizer high pressure trip is a backup trip. |
| | | 5. In addition, Rx trip from PZR high pressure serves as a backup to terminate the incident before an overpressure condition could occur. | LOST<br>(Memo dated Sept. 2, 1992 from W. G. Sotos to V. D. VanderBurg) | <u>Indications Available</u><br>Panel Indication<br>Panel Recorder<br>Plant Process Computer Indication<br><u>Diverse Alarms Available</u><br>Pressurizer High Pressure Deviation via. Control System. Four high pressure alarms via. control system.<br><u>Other Alarm/Indications</u><br>Audible indication of rod motion. | FD-2101 Sheet 1/6 | None. Two Diverse Alarms are available. | |

| FSAR TRANSIENT # | TRANSIENT | TRIP/SAFEGUARD FUNCTION FOR RX TRIP (FSAR 14.1.2) | IMPACT OF COMMON MODE FAILURE (CMF) ON TRIP FUNCTION | ALARM/ALTERNATE INDICATION SYSTEM AVAILABLE | DIAGRAM # | CONSEQUENCES OF UNAVAILABILITY OF DIVERSE ALARM | EVALUATION OF EVENT |
|---|---|---|---|---|---|---|---|
| 14.1.2 | Uncontrolled RCCA Bank Withdrawal at Power | 1. Nuclear Power range instrumentation actuates a reactor trip on high neutron flux if 2/4 channels exceed on overpower setpoint.<br>2. Rx trip on any two out of four ΔT channels exceed OTΔT setpoint. This setpoint is automatically varied with axial power distribution coolant average temperature and pressure to protect against DNB.<br>3. Rx trip on two out of four ΔT channels exceed OPΔT setpoint. This setpoint is automatically varied with coolant average temperature so that the allowable fuel power rating is not exceeded.<br>4. A high pressure reactor trip, actuated from any two out of four pressure channels, is set at a fixed point.<br>5. A high pressurizer water level, actuated from any 2/4 channels, is set at a fixed point. | Not Affected<br><br>OTΔT Rx Trip Lost (Memo dated 9/2/92 from W. G. Sotos to V. D. VanderBurg)<br><br>OPΔT Rx Trip Lost (Memo dated 9/2/92 from W. G. Sotos to V. D. VanderBurg)<br><br>Lost (Memo dated 9/2/92 from W. G. Sotos to V. D. VanderBurg)<br><br>Lost (Memo dated 9/2/92 from W. G. Sotos to V. D. VanderBurg) | NIS power range overpower rod stop at 103% alarm.<br><br>Wide range temperature recorders.<br><br>Wide range temperature recorders.<br><br>Indication Available<br>Panel indication<br>Panel recorder<br>Plant Process computer indication<br>Diverse Alarms Available<br>Pressurizer High Pressure Deviation via control system<br>Four High pressure alarms via control system<br><br>Indication Available<br>Panel indication<br>Panel recorder<br>computer indication<br>Diverse Alarms Available<br>Pressurizer High Level Deviation via control system<br>High level via control system<br>Other alarm/indications<br>Audible indication of rod motion | FD-2102 Sheet 3/4<br><br>FD-2102 Sheet 3/4<br><br>FD-2102 Sheet 1/6<br><br>FD-2101 Sheet 2/6 | Nuclear instrumentation system not changed.<br><br>Five diverse alarms available<br><br>Two diverse alarms available. Rx trip on high pressurizer water level actuates earlier than either the OTΔT or high neutron flux trip functions to demonstrate this protection during pressurizer filling scenarios (FSAR, page 14.1.2A-4) | 1. The Rx trip on NIS overpower setpoint is not affected by the replacement of H-line analog process protection system, since flux measurement instrumentation is not replaced.<br>2. The OTΔT Rx trip is lost by H-line replacement. The OTΔT trip ensures that DNB does not occur. The FSAR analysis of this event assumes that Rx trip on high pressurizer water level is assumed available. This trip actuates earlier than either the OTΔT or high neutron flux trip functions to demonstrate this protection during the slower pressurizer filling scenarios (FSAR, page 14.1.2A-4). The high pressurizer water level trip has two diverse high level alarms, therefore operator would get indications prior to OTΔT Rx trip for pressurizer fill events. Those scenario's that do not terminate on high NIS flux or high pressurizer water are terminated by OTΔT. They tend to be lower reactivity insertion scenarios or lower power scenarios. Although more time is available for response to these events, it cannot be stated with certainty that fuel clad damage will not occur. Westinghouse has reported in WCAP-8330 that minimum DNBR can be achieved for a rod withdrawal at power ATWAS although the particular case evaluated was a rapid reactivity insertion case which would have tripped on NIS high flux. Clad damage is an acceptable outcome bacuse the CMF is a multiple failure condition. However, as discussed below, rod withdrawal of power events are significantly mitigated by the full power base load operation of the Cook Units.<br><br>3. The replacement of H-line analog protection system causes a loss of OPΔT Rx trip. This could result in fuel rod cladding failure. However, the possibilities of this to occur is slim. First of all, this event would be terminated as soon as power is = 109% Rated Thermal Power (Trip Setpoint) by the NIS. This is always the limiting trip for minimum feedback, rapid reactivity insertion events. For maximum feedback, rapid reactivity insertion events, the pressure control system is not expected to keep up thereby also producing a high pressure deviation alarm. The slow reactivity insertion events are expected to fill the pressurizer and produce a level alarm. The escalation of power increases Tavg, and Wide Range RCS Temperature Recorder indications are |

| FSAR TRANSIENT # | TRANSIENT | TRIP/SAFEGUARD FUNCTION FOR RX TRIP (FSAR 14.1.2) | IMPACT OF COMMON MODE FAILURE (CMF) ON TRIP FUNCTION | ALARM/ALTERNATE INDICATION SYSTEM AVAILABLE | DIAGRAM # | CONSEQUENCES OF UNAVAILABILITY OF DIVERSE ALARM | EVALUATION OF EVENT |
|---|---|---|---|---|---|---|---|
| 14.1.2 (cont'd) | | | | | | | available to the poerator (Memo dated 9/2/92 from W.G. Sotos to V.D. VanderBurg). The high pressurizer Rx trip and high pressurizer water level Rx trip have Diverse Alarms available.<br><br>4. The Cook Units are base loaded so that they operate primarily at 100% RTP with rods essentially completely withdrawn. The lower power cases essentially address conditions which are transitory. During transition operation, operators will give close attention to indications as they manipulate the machine. Note that powers ≈70% are used occasionally to stretch a cycle. For these reasons this is a low probability event. |

| FSAR TRANSIENT # | TRANSIENT | TRIP/SAFEGUARD FUNCTION FOR RX TRIP (FSAR 14.1.3 and 14.1.4) | IMPACT OF COMMON MODE FAILURE (CMF) ON TRIP FUNCTION | ALARM/ALTERNATE INDICATION SYSTEM AVAILABLE | DIAGRAM # | CONSEQUENCES OF UNAVAILABILITY OF DIVERSE ALARM | EVALUATION OF EVENT |
|---|---|---|---|---|---|---|---|
| 14.1.3 | Rod Cluster Control Assembly (RCCA) Misalignment (14.1.3) | No reactor trip on RCCA misalignment (FSAR 14.1.3) | None | None | | None | For RCCA misalignment event (FSAR 14.1.3), there is no reactor trip. The analysis for RCCA drop rod(s) event does not take credit for any direct reactor trip due to dropped rods (WCAP-11394, page 1-2). Thus, the replacement of existing W-Line analog process protection system will not affect the FSAR results of these two events.<br><br>The following detection signals/alarms are available for the operator to respond to these transients (FSAR, Unit 2 pages 14.1.3-1 and 14.1.3-2):<br>(i) Sudden drop in core power level as seen by the NIS<br>(ii) Asymmetric power distribution as seen on out-of-core neutron detectors or core exit thermocouple,<br>(iii) Rod deviation alarm (Setpoint-individual rod position deviation ± 12 steps from demand counter, Procedure 2-OHP 4024.210 Drop 29),<br>(iv) Rod position indication.<br><br>In addition, for rod dropped event or dropped bank, the fully inserted assemblies are indicated by a rod at bottom signal, which actuates a control room annunciator (setpoint 20 steps from the bottom, Procedure 2-OHP 4024.210, Drop 22). |
| 14.1.4 | RCCA Assembly Drop (14.1.4) | For RCCA drop rod(s) event, the analysis does not take credit for any direct reactor trip due to dropped rods (WCAP-11394, page 1-2) | None | None | | None | |

| FSAR TRANSIENT # | TRANSIENT | TRIP/SAFEGUARD FUNCTION FOR RX TRIP (FSAR 14.1.5) | IMPACT OF COMMON MODE FAILURE (CMF) ON TRIP FUNCTION | ALARM/ALTERNATE INDICATION SYSTEM AVAILABLE | DIAGRAM # | CONSEQUENCES OF UNAVAILABILITY OF DIVERSE ALARM | EVALUATION OF EVENT |
|---|---|---|---|---|---|---|---|
| 14.1.5 | Uncontrolled Boron Dilution | 1) With reactor in manual control and no operator action taken to terminate the transient, the power and temperature will cause the reactor to reach the overtemperature $\Delta T$ (OT$\Delta$T) trip setpoint resulting in a reactor trip (FSAR, Page 14.1.5-5) | OT$\Delta$T reactor trip lost (memo dated 9/2/92 from W. G. Sotos to V. D. VanderBurg) | None<br><br>__Other alarms/indications__<br>NIS power range overpower rod stop at 103%<br>Primary water flow deviation alarm<br>Boric and flow deviation alarm with rods in automatic:<br>Rod bank D low alarm<br>Rod bank D low-low alarm<br>Audible indication of rod motion | FD-2102 Sheet 3/4 | | The FSAR section 14.1.5 has examined three phases of boron dilution accident, i.e. boron dilution during (i) refueling, (ii) startup, and (iii) power operation. For dilution during refueling, there are more than 33 minutes available for operator action from the time of initiation of the event to loss of shutdown margin (5% $\Delta k/k$) (FSAR, page 14.1.5-5). For refueling mode, the most likely source of dilution, CVCS, is tagged out. For other modes this source is not tagged out. For dilution during startup there are more than 35 minutes available for the operator action from the time of initiation of the event to loss of shutdown margin (1.3% $\Delta k/k$) (FSAR, page 14.1.5-5) for Unit 2 and 68 minutes for Unit 1. Startup is a transient operation. Operators will give close attention to indications as they manipulate the machine.<br><br>Dilution accident at power includes the reactor in automatic control or manual control. With the reactor in automatic control, the power and temperature increase from the boron dilution results in insertion of the control rods and a decrease in the available shutdown margin. There are more than 46 minutes from the time of alarm (low-low rod insertion limit) to loss of shutdown margin (1.3% $\Delta k/k$) (FSAR, page 14.1.5-5) for Unit 2 and 48 minutes for Unit 1. The Cook Units are operated with rods in automatic unless there is a compelling reason to operate in manual.<br><br>With reactor in manual control and no operator action taken to terminate the transient, the power and temperature would cause the reactor to reach OT$\Delta$T trip setpoint. This trip will be lost as a result of common mode failure of the new Foxboro digital system. The boron dilution transient in this case is essentially equivalent to an uncontrolled RCCA withdrawal at power (FSAR, page 14.1.5-1). There is no control room alarm from the $\Delta T$ system for this event. However, the increasing power and wide range temperature indications would indicate conditions to the operator. This event is a slow reactivity addition event, $\approx$1pcm/sec. |

| FSAR TRANSIENT # | TRANSIENT | TRIP/SAFEGUARD FUNCTION FOR RX TRIP (FSAR 14.1.5) | IMPACT OF COMMON MODE FAILURE (CMF) ON TRIP FUNCTION | ALARM/ALTERNATE INDICATION SYSTEM AVAILABLE | DIAGRAM # | CONSEQUENCES OF UNAVAILABILITY OF DIVERSE ALARM | EVALUATION OF EVENT |
|---|---|---|---|---|---|---|---|
| 14.1.5 (con't) | | | | | | | Following the discussion on uncontrolled RCCA bank withdrawal at power, the high pressurizer water level alarm is assumed available, which has two diverse alarms (memo dated 9/2/92 from W. G. Sotos to V. D. VanderBurg). This is a slow transient, and with the pressurizer level, wide range temperature indications, and other indications, the operator should be able to trip the reactor. |

| FSAR TRANSIENT # | TRANSIENT | TRIP/SAFEGUARD FUNCTION FOR RX TRIP (FSAR 14.1.6.1) | IMPACT OF COMMON MODE FAILURE (CMF) ON TRIP FUNCTION | ALARM/ALTERNATE INDICATION SYSTEM AVAILABLE | DIAGRAM # | CONSEQUENCES OF UNAVAILABILITY OF DIVERSE ALARM | EVALUATION OF EVENT |
|---|---|---|---|---|---|---|---|
| 14.1.6.1 | Loss of Forced Reactor Coolant Flow | 1. Rx trip on reactor coolant pump power supply undervoltage or underfrequency | Not Affected | Reactor Coolant Pump underfrequency and undervoltage alarm (Procedure 1, 2-OHP, 4024, 107, 207) | | None | The Rx trip on reactor coolant pump power supply undervoltage and underfrequency remains unaffected by a common mode failure (CMF) of the new digital instrumentation.<br>The reactor trip on loss of flow in a coolant loop is lost on CMF for each loop. These are no Diverse Alarms available; however, panel indication and computer indication are available for the low coolant loop flow.<br><br>Two cases of loss of flow are discussed in FSAR (14.1.6). The simultaneous loss of power to all 4 RCPs can occur due to either underfrequency or undervoltage, which is not impacted by CMF. This situation is highly unlikely, since each pump is connected to a separate bus, which is supplied by one of two transformers.<br><br>The consequences of the loss of flow include an increase in Tavg, pressurizer pressure, and pressurizer water level. Wide range RCS temperature recorders (memo dated 9/2/92 from W. G. Sotos to V. D. VanderBurg) are available to the operator to indicate an increase in Tavg. There is no Rx trip on high Tavg. The pressurizer pressure will continue to rise until the operator gets a high pressure deviation alarm at 2325 psia (2-OHP 4024.208 Drop 7) for Unit 2 and 2175 psia for Unit 1. The Rx trip on high pressure (setpoint ≤ 2400 psia) is lost due to CMF. However, diverse alarms (memo dated 9/2/92 from W. G. Sotos to V. D. VanderBurg) are available. It is evident that the high pressure deviation alarm will draw the operator's attention, and he will trip the Rx manually. The operator will also be likely to see the high level deviation alarm at 5% above program. The consequences of this manual Rx trip are discussed below.<br><br>Crude extrapolations of DNBR for these events suggest that MDNBR could be reached within ≈16 to ≈18 seconds for loss of flow in one loop. Similar extrapolations suggest that the high pressure deviation alarm would first be received ≈6 seconds into the transient although the operation of pressurizer sprays will increase this estimate. Allowing ≈60 seconds for operation response it is clear that DNB could |
| | | 2. Rx trip on low reactor coolant loop flow. | Low flow Rx trip lost (for all four loops) | <u>Indication Available</u><br>Panel indication<br>computer indication<br><u>Diverse Alarm Available</u><br>None<br><br><u>Other Indications</u><br>Pressurizer pressure panel indication<br>Pressurizer pressure recorder<br>Pressurizer pressure computer indication<br>Pressurizer level panel indication<br>Pressurizer level recorder<br>Pressurizer level computer indication<br>Wide range temperature records<br><br><u>Other Alarms</u><br>Pressurizer high pressure deviation via control system<br>Four high pressure alarms via control system<br>Pressurizer high level deviation via control system<br>High level via control system<br>Acoustic monitor flow detected | FD-2101 Sheet 3 and 4 | If the Rx is at power at the time of the accident, the immediate effect of a loss of coolant flow is a rapid increase in the coolant temperature which is magnified by a positive MTC. This increase could result in DNB with subsequent adverse effects to the fuel, if the Rx is not tripped promptly. (FSAR, page 14.1.6-1) | |

| FSAR TRANSIENT # | TRANSIENT | TRIP/SAFEGUARD FUNCTION FOR RX TRIP (FSAR 14.1.6.1) | IMPACT OF COMMON MODE FAILURE (CMF) ON TRIP FUNCTION | ALARM/ALTERNATE INDICATION SYSTEM AVAILABLE | DIAGRAM # | CONSEQUENCES OF UNAVAILABILITY OF DIVERSE ALARM | EVALUATION OF EVENT |
|---|---|---|---|---|---|---|---|
| 14.1.6.1 (cont'd) | | | | | | | occur resulting in clad damage. Since a massive multiple failure is assumed for this event, this is believed to be acceptable. With a loss of flow in one loop total core flow should remain ~80% removing the bulk of the heat from the core, limiting the deterioration of the core prior to manual reactor trip. The portion of the core that experiences DNB is expected to heat up until the Doppler coefficient shuts it down. Fuel is not expected to melt but clad burst and oxidation are anticipated. It should also be noted that this event was analyzed with a positive moderation coefficient (MTC) of +5 pcm/°F. This value is more limiting than the Technical Specification limit at 100% RTP. It is conservative and provides substantial margin throughout most of the life. This causes power to increase as the coolant temperature increases. A more realistic assumption for beginning of cycle is ~-4pcm/oF. A negative MTC will tend to shutdown the core as temperature increases mitigating the event. The MTC becomes substantially more negative as burnup progresses. The Cook Units are base loaded and operate with control rods in the all out position at full power. Therefore, the possibility that automatic rod control might withdraw rods will have no impact because rods are essentially fully withdrawn. After reactor trip, the emergency operating procedures provide for mitigation activities to bring the machine to a safe condition.<br><br>In the evaluation of the previous paragraph, an operator response time of ~60 seconds was assumed. Without a reactor trip, pressurizer pressure and level are expected to continue to increase after the first alarms are received. When pressure reaches 2250 psia, the PORV's will open resulting in an acoustic monitor flow detected alarm. Extrapolating the analysis curves, which do not model pressurizer spray, this could occur before MDNBR is reached. Therefore, it is likely that an accumulation of alarms will occur before 60 seconds have elapsed. Therefore, the operators response time may be less than 60 seconds for this event. |

| FSAR TRANSIENT # | TRANSIENT | TRIP/SAFEGUARD FUNCTION FOR RX TRIP (FSAR 14.1.6.1) | IMPACT OF COMMON MODE FAILURE (CMF) ON TRIP FUNCTION | ALARM/ALTERNATE INDICATION SYSTEM AVAILABLE | DIAGRAM # | CONSEQUENCES OF UNAVAILABILITY OF DIVERSE ALARM | EVALUATION OF EVENT |
|---|---|---|---|---|---|---|---|
| 14.1.6.1 (cont'd) | | | | | | | The most likely cause of an event of this type, is a failure of the reactor coolant pump (RCP) or its motor. The operator is provided with a significant number of alarms to give him information regarding the RCP's and motors. These alarms include RCP motor differential trip, RCP motor overload trip, and RCP motor overheated. Therefore, it is likely that the operator will have information available which will allow him to anticipate and, therefore, substantially mitigate the event. |

| FSAR TRANSIENT # | TRANSIENT | TRIP/SAFEGUARD FUNCTION FOR RX TRIP (FSAR 14.1.6.2) | IMPACT OF COMMON MODE FAILURE (CMF) ON TRIP FUNCTION | ALARM/ALTERNATE INDICATION SYSTEM AVAILABLE | DIAGRAM # | CONSEQUENCES OF UNAVAILABILITY OF DIVERSE ALARM | EVALUATION OF EVENT |
|---|---|---|---|---|---|---|---|
| 14.1.6.2 | Locked Rotor/Shaft Break Accident | Reactor trip on low flow signal | Low flow reactor trip lost (memo 9/2/92 memo from W. G. Sotos to V. D. VanderBurg) | Indications Available<br>Panel indication<br>Computer indication<br>Diverse Alarm Available<br>None<br><br>Other Indications<br>Pressurizer pressure panel indication<br>Pressurizer pressure recorder<br>Pressurizer pressure computer indication<br>Pressurizer level panel indication<br>Pressurizer level recorder<br>Pressurizer level computer indication<br>Wide range temperature records<br>Sound of pressurizer safety valves<br><br>Other Alarms<br>Pressurizer high pressure deviation via control system<br>Four high pressure alarms via control system<br>Pressurizer high level deviation via control system<br>High level via control system<br>Acoustic monitor flow detected | FD-2101 Sheet 3 and 4 | If the Rx is at power at the time of accident, the immediate effect of a loss of flow (seizure of a RCP rotor) is an increase in the coolant temperature. This increase could result in DNB with subsequent adverse effects to fuel, if the Rx is not tripped promptly (FSAR, Page 14.1.6-1) | The FSAR analysis for this event assumes an instantaneous seizure of a reactor coolant pump rotor. For this event, the reactor trips on low flow signal. The common mode failure (CMF) of the new digital instrumentation would result in a loss of low flow Rx trip signal.<br><br>The loss of flow will increase the coolant temperature and an increase in pressurizer pressure due to a reduction in heat removal. The wide range RCS temperature recorders (memo dated 9/2/92 from W. G. Sotos to V. D. VanderBurg) are available to the operator. The pressurizer pressure will continue to rise, and the operator will get a high pressurizer deviation alarm at 2325 psia (Procedure 2-OHP 4024.208 Drop 7) for Unit 2 and 2175 psia for Unit 1. The reactor trip on high pressure (≤2400 psia) is lost due to CMF. However, high pressure diverse alarms are available (memo dated 9/2/92 from W. G. Sotos to V. D. VanderBurg). Therefore, the high pressure deviation alarm will draw the operator's attention to trip the reactor manually.<br><br>This event is very much like the loss of forced reactor coolant flow in one loop. However, it is more severe in that total core flow is reduced more rapidly to a lower value. The total core flow is reduced to ~70% within ~2 seconds. As the coolant heats up, a significant increase in pressure occurs. The peak analyzed pressure for both units is ~2590 psia. This peak occurred at ~2 seconds after the reactor trip at 1 second. This pressure is less than 110% of the design pressure, i.e. 2750 psia. However, if reactor trip is delayed ~60 seconds, it cannot be stated with certainty that this pressure would not be exceeded. However, the analysis takes no credit for pressurizer spray or the pressurizer PORV's. It is also the case as with the loss of forced reactor coolant flow that the analysis was performed with a positive moderator temperature coefficient (MTC) of +5 pcm/°F. This value is more limiting than the Technical Specification limit at 100% RTP. It is conservative and provides substantial margin throughout the core life. |

| FSAR TRANSIENT # | TRANSIENT | TRIP/SAFEGUARD FUNCTION FOR RX TRIP (FSAR 14.1.6.2) | IMPACT OF COMMON MODE FAILURE (CMF) ON TRIP FUNCTION | ALARM/ALTERNATE INDICATION SYSTEM AVAILABLE | DIAGRAM # | CONSEQUENCES OF UNAVAILABILITY OF DIVERSE ALARM | EVALUATION OF EVENT |
|---|---|---|---|---|---|---|---|
| 14.1.6.2 (con't) | | | | | | | Therefore, as Tavg is increased, power increases in the analysis. As indicated in the loss of forced reactor coolant flow, a more realistic beginning of cycle MTC, would be ≈-4pcm/°F. Throughout core life the MTC would decrease to the ≈-20pcm/°F. The feedback from the MTC would therefore tend to shut the reactor down rather than increase power in an actual event. The Cook units are base loaded and operate with control rods in the all out position at full power. The possibility that automatic rod control might withdraw rods will have no impact because rods are essentially fully withdrawn. These considerations lead us to conclude that it is unlikely that pressurizer pressure would exceed 2750 psia and virtually impossible to exceed 3200 psig, the ASME Boiler and Pressure Vessel Code Level C criterion, which was used for AMSAC design.<br><br>In the analysis, DNB is expected to occur. In the event of a delay of reactor trip by ≈60 seconds, this situation can only be exacerbated. The operation of pressurizer sprays and PORV's which were not modeled in the analysis will also result in an increase in fuel rods in DNB. However, it is believed that the available flow will prevent the core from degrading to condition where it cannot be cooled after trip. The portion of the core that experiences DNB is expected to heat up until the Doppler coefficient shuts it down. Fuel is not expected to melt but clad burst and oxidation are anticipated. Substantial core damage is acceptable for this event which is an ANS condition IV event with massive multiple failures.<br><br>In the evaluation of the previous two paragraphs, an operator response time of ≈60 seconds was assumed. However, this event is expected to be very dramatic. Several pressurizer alarms can be expected within seconds of the start of the event including the acoustic monitor flow detected alarm. The pressurizer safety valves can be expected to lift which creates an impressive sound in the control room. Therefore, the operators response may be less than 60 seconds for this event. |

| FSAR TRANSIENT # | TRANSIENT | TRIP/SAFEGUARD FUNCTION FOR RX TRIP (FSAR 14.1.6.2) | IMPACT OF COMMON MODE FAILURE (CMF) ON TRIP FUNCTION | ALARM/ALTERNATE INDICATION SYSTEM AVAILABLE | DIAGRAM # | CONSEQUENCES OF UNAVAILABILITY OF DIVERSE ALARM | EVALUATION OF EVENT |
|---|---|---|---|---|---|---|---|
| 14.1.6.2 (con't) | | | | | | | As in the case of loss of forced reactor coolant flow, the most likely cause of event of this type, is the failure of the reactor coolant pump (RCP) or motor. The operator is provided with a significant number of alarms to give him information regarding the RCP's and motors.<br><br>These alarms include bearing temp high, lower bearing seal water temperature high, lower bearing cooling water flow low, upper oil pot level high or low, and lower oil pot level high or low. Therefore, it is likely that the operator will have information available which will allow him to anticipate and therefore, substantially mitigate the event.<br><br>For Unit 2 an offsite dose calculation was performed as a part of the transition to Westinghouse Vantage 5 fuel. The site boundary doses were 3 rem, thyroid and 0.3 rem whole body. These are very small fractions of the 10CFR100 criteria. However, with a delay in reactor trip of ~60 seconds, it is anticipated that core damage will be increased significantly. Nevertheless, the 10CFR100 criteria are expected to be satisfied for this condition IV event. In section 14.3.5, an offsite dose analysis for LOCA which is identified as the maximum hypothetical accident is described. For this analysis, it is assumed that 50% of the core inventory of halogens and 100% of the core inventory of nobel gases are released to containment atmosphere. Table 14.3.5-10 of the Unit 2 UFSAR and Table 14.3.5-7 of the Unit 1 UFSAR display the doses for this analysis. They satisfy the criteria of 10CFR100. Since the RCS is anticipated to be intact after a locked rotor event, it is expected that the doses for the maximum hypothetical accident will substantially bound the locked rotor event doses. |

| FSAR TRANSIENT # | TRANSIENT | TRIP/SAFEGUARD FUNCTION FOR RX TRIP (FSAR 14.1.7) | IMPACT OF COMMON MODE FAILURE (CMF) ON TRIP FUNCTION | ALARM/ALTERNATE INDICATION SYSTEM AVAILABLE | DIAGRAM # | CONSEQUENCES OF UNAVAILABILITY OF DIVERSE ALARM | EVALUATION OF EVENT |
|---|---|---|---|---|---|---|---|
| 14.1.7 | Startup of an Inactive Reactor Coolant Loop | Unit 1 and Unit 2 operation during startup and power operation with less than four loops is not permitted (T/S 3/4.4.1) except for special testing as provided for in T/S 3/4.10.5 for Unit 1 and T/S 3.4.10.4 for Unit 2. License conditions for both Units prohibit operation above P-7 with less than four reactor coolant pumps in operation. However, the UFSAR contains analysis of this event for both Units. This information is provided for information and because it bounds the test conditions indicated above. These analyses result in reactor trips on nuclear instrumentation high flux. | None | | | None | In accordance with T/S 3/4.4.1, operation during startup and power operation with less than four loops is not permitted. As such, this accident was not analyzed for the VANTAGE-5 fuel transition (Unit 2 FSAR, Page 14.1.7-1) or for the Unit 1 reduced temperature and pressure program (Unit 1 UFSAR, Page 14.1.7-3). Therefore, the common mode failure (CMF) of the new Foxboro digital system would have no impact on this transient. |

| FSAR TRANSIENT # | TRANSIENT | TRIP/SAFEGUARD FUNCTION FOR RX TRIP (FSAR 14.1.8) | IMPACT OF COMMON MODE FAILURE (CMF) ON TRIP FUNCTION | ALARM/ALTERNATE INDICATION SYSTEM AVAILABLE | DIAGRAM # | CONSEQUENCES OF UNAVAILABILITY OF DIVERSE ALARM | EVALUATION OF EVENT |
|---|---|---|---|---|---|---|---|
| 14.1.8 | Loss of External Electric Load or Turbine Trip (Full Vantage-5 Core) | Reactor trips on following signals:<br><br>1. High pressurizer pressure signal | High pressure Rx trip lost | <u>Indication Available</u><br>•Panel indication<br>•Panel recorder<br>•computer indication<br><u>Diverse Alarms Available</u><br>•High Pressure deviation via. control system<br>•High pressure via control system (four alarms)<br>• Pressurizer PORV discharge temp high<br>• Pressurizer safety valve discharge temp hi (3 alarms)<br>• Pressurizer relief tank temp hi<br>• Pressurizer relief tank pressure high or low<br>• Pressurizer relief tank level high or low<br>• Acoustic monitor flow detected | FD-2101 Sheet 1/6 | | <u>Loss of Load on Turbine Trip</u><br>The most likely source of a complete loss of load in NSSS is a trip of the turbine-generator or a differential relay which results in a turbine trip. In this case, there is a direct reactor trip signal (unless power is below approximately 11% power, i.e., below P-7) derived from the turbine emergency trip fluid pressure and turbine stop valves (FSAR, page 14.1.88-1). Therefore, the common mode failure (CMF) of the new digital system has no impact on the reactor trip.<br><br><u>Loss of Load without Turbine Trip</u><br>Two initiating scenarios were considered for this event: Complete loss of electrical load, and loss of condenser vacuum.<br><br><u>Complete Loss of Electrical Load Without Reactor Trip</u><br>For this event, the reactor trips on four trip functions. For high pressurizer pressure trip function, three alternate indications and several diverse alarms are available. For high pressurizer water level trip, three alternate indications and two diverse alarm available. For low-low steam generator water level trip, three alternate indications and one diverse alarm are available. These indications, alarms, and other indications, especially the sound of safety valves should provide indications to the operator of abnormal situation and he would trip the reactor manually.<br><br>The impact of the common mode failure (CMF) of the digital system would result in a loss of OTΔT reactor trip function. The OTΔT reactor trip is the only function for which the alternate alarms/indications are not available. The loss of reactor trip would cause the RCS pressure and temperature to rise. This would result in an increase of pressurizer water level. Pressurizer pressure, pressurizer level and wide range temperature indications are available to the operator to trip the reactor (memo dated 9/2/92 from W. G. Sotos to V. D. VanderBurg). The high pressure deviation alarm activates at 2325 psia (Procedure 2-OHP 4024.208 |
| | | 2. High pressurizer water level | High pressurizer water level Rx trip lost | <u>Indications Available</u><br>•Panel indication<br>•Panel recorder<br>•Computer indication<br><u>Diverse Alarms Available</u><br>•Hi level deviation from setpoint via control system<br>• Pressurizer level high from control system | FD-2101 Sheet 2/6 | | |
| | | 3. Overtemperature ΔT(OTΔT) signal | OTΔT Rx trip lost | <u>Indications Available</u><br>Wide range RCS temperature recorders | FD-2101 Sheet 5 | | |

| FSAR TRANSIENT # | TRANSIENT | TRIP/SAFEGUARD FUNCTION FOR RX TRIP (FSAR 14.1.8) | IMPACT OF COMMON MODE FAILURE (CMF) ON TRIP FUNCTION | ALARM/ALTERNATE INDICATION SYSTEM AVAILABLE | DIAGRAM # | CONSEQUENCES OF UNAVAILABILITY OF DIVERSE ALARM | EVALUATION OF EVENT |
|---|---|---|---|---|---|---|---|
| 14.1.8 (con't) | | 4. Low-low steam generator water level | Lo-Lo water level reactor trip lost | <u>Indication Available</u><br>•Panel indication<br>•Panel recorder<br>•computer indication<br><u>Diverse Alarms Available</u><br>•Level deviation via control system<br><br><u>Other Indications/Alarms</u><br>• Power Range overpower Rod Stop<br>•Sound of steam generator and pressurizer safeties.<br>• Audible indication of control rod motion. | | | Drop 7) for Unit 2 and 2175 for Unit 1. This alarm would draw operators attention. Pressurizer sprays would begin to open at 2260 psig and would be full open at 2310 psig (FSAR, Table 4.1-2) for Unit 2 and from 2110 psig to 2160 for Unit 1. The PORV will be full open at 2335 psig, and safety valves open at 2485 psig (FSAR, Table 4.1-2).<br><br>Assuming the availability of this control equipment, the primary pressure should not exceed 2750 psia in the minimum reactivity feedback case. The MTC for this case is assumed to be +5pcm/°F and the Doppler coefficient is assumed to be =-6pcm/%. More realistic assumptions for beginning of cycle and HFP are MTC=-4pcm/% and Doppler =-8pcm/% . These values will increase the temperature feedback relative to the analysis tending to reduce power and consequently primary pressure.<br><br>In the maximum reactivity feedback, the reactor power and consequently primary pressure are reduced by thermal feedback. DNBR is not threatened in the maximum reactivity feedback case.<br><br>Additional control equipment may also operate to mitigate this event. The power mismatch channel for rod control can be expected to operate on a loss of load driving rods into the core. The time constant of first stage pressure is 40 sec. Therefore, rods can be expected to insert until the operator initiates protective action. If Tavg fails constant on a CMF or fails high, rods will continue to insert after the power mismatch signal has decayed. The steam dump to condenser would also operate with Tavg constant or high provided that condenser vacuum or offsite power are not lost.<br><br><u>Loss of Condenser Vacuum</u><br>The loss of condenser vacuum affects only the turbine and not the reactor protection system. Therefore, the turbine trip on condenser vacuum will result in a reactor trip since both remain unaffected by the common mode failure of the new digital system. |

| FSAR TRANSIENT # | TRANSIENT | TRIP/SAFEGUARD FUNCTION FOR RX TRIP (FSAR 14.1.9) | IMPACT OF COMMON MODE FAILURE (CMF) ON TRIP FUNCTION | ALARM/ALTERNATE INDICATION SYSTEM AVAILABLE | DIAGRAM # | CONSEQUENCES OF UNAVAILABILITY OF DIVERSE ALARM | EVALUATION OF EVENT |
|---|---|---|---|---|---|---|---|
| 14.1.9 | Loss of Normal Feedwater | 1. Reactor trip on low-low water level in any steam generator | Low-low level trip lost | Diverse Alarms Available<br>• steam generator level deviation via control system<br>Indications Available<br>•Panel indication<br>•Panel recorder<br>•computer indication | FD-2101 Sheet 5 | | The common mode failure (CMF) of the new digital equipment results in a loss of reactor trips on low-low water level, and on low feedwater flow signal (steam flow/feedflow mismatch in coincidence with low water level). Both the motor driven and turbine driven auxiliary feedwater systems are also lost except in situation described below.<br><br>The motor driven auxiliary feedwater pumps are not affected by CMF if the pumps started on 4 kv bus loss of voltage or loss of all main feedwater pumps (T/S Table 3.3-3, page 3/4 3-19). The turbine driven auxiliary feedwater pump is also not affected by CMF if the pump is started on reactor coolant pump bus undervoltage (T/S Table 3.3-3, page 3/4 3-20).<br><br>In case of the CMF of new digital equipment, steam generator level deviation alarm and AMSAC alarm are available to the operator. In addition, three alternate indications are also available.<br><br>For the loss of normal feedwater/ATWS transient, ATWS Mitigating System Actuation Circuitry (AMSAC) is available (memo dated 10/13/92 from W. G. Sotos to V. D. VanderBurg). The AMSAC automatically initiates a turbine trip and initiates AFW flow to maintain the RCS pressure below 3200 psig (ASME Boiler and Pressure Vessel Code Level C criterion). At 100% RTP these functions are initiated at 30 sec. of transient signal delay time. AMSAC is available to perform this function in the event the CMF of the new digital equipment occurs. An AMSAC annunciator is initiated after AMSAC is actuated (Procedure 2-OHP 4024.212 Drop 14). The turbine trip is not affected by the CMF of the new digital equipment (memo dated 9/2/92 from W. G. Sotos to V. D. VanderBurg). Therefore, the reactor would be tripped upon turbine trip.<br><br>At all powers the steam generator level deviation alarm, pressurizer level high level deviation and pressurizer level high are available to alert the operator to a loss of normal feedwater event. In addition, numerous alarms describing the status of the condensate and feedwater systems and pumps, such as condenser hotwell level, booster motor trip, |
| | | 2. Reactor trip on low feedwater flow signal in any steam generator (This signal is actually a steam flow-feedwater mismatch in coincidence with low water level) | Low feedwater flow trip lost | same as above (for steam generator low-low water level) | | | |
| | | 3. Two motor driven auxiliary feedwater pumps which are started on:<br>a. Low-low level in any steam generator<br>b. Trip of all main feedwater pumps<br>c. Any safety injection signal<br>b. 4 kv bus loss of voltage<br>e. Manual actuation | MDAFP starts (automatic initiation) on low-low steam generator level and safety injection from non-manual initiation are lost | same as above | | | |
| | | 4. Turbine driven auxiliary feedwater pump is started on:<br>a. Low-low level in any two steam generators<br>b. Reactor coolant pump bus undervoltage | TDAFP start (automatic initiation) on low-low steam generator level is lost | same as above<br><br>Other Alarms/Indications<br>• Pressurizer high level deviation<br>• Pressurizer level high | | | |

| FSAR TRANSIENT # | TRANSIENT | TRIP/SAFEGUARD FUNCTION FOR RX TRIP (FSAR 14.1.9) | IMPACT OF COMMON MODE FAILURE (CMF) ON TRIP FUNCTION | ALARM/ALTERNATE INDICATION SYSTEM AVAILABLE | DIAGRAM # | CONSEQUENCES OF UNAVAILABILITY OF DIVERSE ALARM | EVALUATION OF EVENT |
|---|---|---|---|---|---|---|---|
| 14.1.9 (con't) | | | | | | | main feedwater pump, etc. will activate. Below 40% rated thermal power, it is expected that these alarms would lead the operator to trip the reactor manually due to low steam generator level in accordance with 2-OHP 4023.E-0.<br><br>We also note that this event progresses relatively slowly so that the pressurizer fills in the order of minutes not seconds. The event as described in the UFSAR is analyzed using AFW flows based on flow retention. The operator will be able to open the flow retention valves to substantially increase feedwater flow. It is also not considered necessary to assume an AFW pump failure in addition to CMF. Assuming the availability of all three AFW pumps also substantially increases the flow of AFW. For all these reasons, we believe the outcome of this event will not be substantially different from the analyzed result. |

| FSAR TRANSIENT # | TRANSIENT | TRIP/SAFEGUARD FUNCTION FOR RX TRIP (FSAR 14.1.10) | IMPACT OF COMMON MODE FAILURE (CMF) ON TRIP FUNCTION | ALARM/ALTERNATE INDICATION SYSTEM AVAILABLE | DIAGRAM # | CONSEQUENCES OF UNAVAILABILITY OF DIVERSE ALARM | EVALUATION OF EVENT |
|---|---|---|---|---|---|---|---|
| 14.1.10.1 | Excessive Heat Removal due to Feedwater System Malfunctions | 1. High neutron flux trip | Not affected | NIS power range overpower rod stop at 103% alarm | | | The reactor trip on NIS overpower setpoint is not affected by the common mode failure (CMF) of the new digital equipment.<br><br>The OTΔT and OPΔT reactor trips are lost due to CMF of the new digital equipment. No alternate alarms are available for these trip functions. However, wide range hot and cold leg temperature indications are available. The cases of low pressure or high pressure feedwater heater bypass valve fully opening result in transients very similar to those for excessive increase in secondary steam flow. This transient is discussed in section 14.1.11. The Unit 2 feedwater events are bounded by the excessive load increase. The Unit 1 events are also expected to be bounded.<br><br>For an increase in feedwater flow in the absence of CMF, the turbine would trip on high-high steam generator water level, which would in turn trip the reactor. In case of CMF, this trip is lost (T/S Table 3.3-3).<br><br>At zero power, steam generator level is under manual control. Therefore, the operator would be expected to identify the event promptly and take corrective action. Below P-10, the NIS high flux setpoint at 25% RTP and the NIS intermediate range trips are also available. At 100% RTP, the steam generator deviation alarm (Procedure 2-OHP 4024.213 Drop 2) would activate at 5% above programmed level of 44%. Three steam generator level indications are available (memo dated 10/13/92 from W. G. Sotos to V. D. VanderBurg). In addition, power range overpower rod stop alarm (Procedure 2-OHP 4024.210 Drop 19) would actuate at 103% power, which would occur at about 20 sec. into the transient (WCAP-12901, Fig 10.61A). With the S. G. deviation alarm and level indications available, the operator should be able to trip the turbine, which in turn would trip the reactor.<br><br>Figure 10.16A of WCAP-12901 shows that, the power stabilizes at approximately 105% nominal (trip setpoint=109%). From Figure 10.29A of WCAP-12901, the steam generator deviation alarm would actuate at about 8 sec. into the transient. |
| | | 2. Overtemperature ΔT (OTΔT) trip | OTΔT reactor trip lost | Wide range temperature recorders | | | |
| 14.1.10.2 | Feedwater System Malfunctions causing and Increase in Feedwater Flow | 3. Overpower ΔT (OPΔT) trip | OPΔT reactor trip lost | Wide range temperature recorders | | | |
| | | 4. Steam generator water level high-high | Lost | <u>Indications Available</u><br>•Panel indication<br>•Panel recorder<br>•Computer indication<br><br><u>Diverse Alarms Available</u><br>•Level deviation via control system | | | |

| FSAR TRANSIENT # | TRANSIENT | TRIP/SAFEGUARD FUNCTION FOR RX TRIP (FSAR 14.1.10) | IMPACT OF COMMON MODE FAILURE (CMF) ON TRIP FUNCTION | ALARM/ALTERNATE INDICATION SYSTEM AVAILABLE | DIAGRAM # | CONSEQUENCES OF UNAVAILABILITY OF DIVERSE ALARM | EVALUATION OF EVENT |
|---|---|---|---|---|---|---|---|
| 14.1.10.2 (cont'd) | | | | | | | . Assuming the operator's response time to be 60 sec., the turbine would trip at approximately 68 sec. or the reactor trip time is approximately 70 sec. Figures 14.1.10A-2 and 14.1.10A-4 of the Unit 2 UFSAR show that the DNBR at this time is approximately 1.6. Figures 14.1.10-2 and 14.1.10-4 show DNBR at this time to be ~2.0. These values are well above the DNBR safety limits for both Units. Therefore, there would not be any fuel damage. |

| FSAR TRANSIENT # | TRANSIENT | TRIP/SAFEGUARD FUNCTION FOR RX TRIP (FSAR 14.1.11) | IMPACT OF COMMON MODE FAILURE (CMF) ON TRIP FUNCTION | ALARM/ALTERNATE INDICATION SYSTEM AVAILABLE | DIAGRAM # | CONSEQUENCES OF UNAVAILABILITY OF DIVERSE ALARM | EVALUATION OF EVENT |
|---|---|---|---|---|---|---|---|
| 14.1.11 | Excessive Load Increase Incident | 1. Overpower $\Delta T$ (OP$\Delta$T) trip<br><br>2. Overtemperature $\Delta T$ (OT$\Delta$T) trip<br><br>3. Power range high neutron flux<br><br>4. Low pressurizer pressure trip | OP$\Delta$T Rx Trip Lost (memo dated 10/13/92 from W. G. Sotos to V. VanderBurg)<br><br>OT$\Delta$T Rx Trip Lost (memo dated 10/13/92 from W. G. Sotos to V. VanderBurg)<br><br>Not Affected<br><br>Lost (memo dated 10/13/92 from W. G. Sotos to V. VanderBurg) | Wide range RCS temperature recorder<br><br>Wide range RCS temperature recorder<br><br>NIS power range overpower rod stop<br><br><u>Indications Available</u><br>•Panel indication<br>•Panel recorder<br>•Computer indication<br><u>Diverse Alarms Available</u><br>•Pressurizer low pressure deviation (turn on backup heaters) via control system<br><u>Other alarms/indications</u><br>Audible indication of rod motion below 103%.<br>Pressurizer low level deviation alarm<br>Pressurizer low level alarm | FD-2101 Sheet 1 | | The common mode failure (CMF) of the new digital equipment results in a loss of OP$\Delta$T trip, OT$\Delta$T trip and low pressurizer pressure trip. The reactor trip on power range high neutron flux is not affected by the CMF of the reactor process equipment.<br><br>The FSAR section 14.1.11 has considered four cases to analyze this event (i) Reactor control in manual with minimum moderator reactivity feedback; (ii) Reactor control in manual with maximum moderator reactivity feedback, (iii) Reactor control in automatic with minimum moderator reactivity feedback; and (iv) Reactor control in automatic with maximum moderator reactivity feedback.<br><br>The reactor trip and/or engineered safeguard actuation signal was not generated for this event (FSAR, page 14.1.11A-3). The FSAR analysis assumes that normal operating procedures would be followed to lower power. In the event that this event occurs concurrently with a CMF of the new digital reactor process equipment, the operator would be expected to bring the reactor to hot shutdown consistent with T.S. 3.0.3 |

| FSAR TRANSIENT # | TRANSIENT | TRIP/SAFEGUARD FUNCTION FOR RX TRIP (FSAR 14.1.12) | IMPACT OF COMMON MODE FAILURE (CMF) ON TRIP FUNCTION | ALARM/ALTERNATE INDICATION SYSTEM AVAILABLE | DIAGRAM # | CONSEQUENCES OF UNAVAILABILITY OF DIVERSE ALARM | EVALUATION OF EVENT |
|---|---|---|---|---|---|---|---|
| 14.1.12 | Loss of Offsite Power (LOOP) to the Station Auxiliaries | 1. Reactor trip on turbine trip | Not Affected | | | | The complete loss of all (non-emergency) AC power (e.g. offsite power) will result in the loss of all power to the plant auxiliaries, i.e., RCPs, condensate pump, etc. The loss of power may be caused by a complete loss of the offsite grid accompanied by a turbine generator trip, or by a loss of the onsite AC distribution system.<br><br>The FSAR analysis for this event assumes that the steam generator low-low level trips the reactor (FSAR, page 14.1.12-3). The common mode software failure (CMF) of the new digital equipment results in a loss of trip on SG low-low level. However, there are other trip functions, independent of new digital RPS, which are available. Following a loss of offsite power, the turbine generator will trip as a result of losing auxiliaries such as circulating water, stator cooling water, control fluid pressure, or feedwater. This will in turn trip the reactor. The loss of offsite power results in a coastdown of the control rod drive motor generator set resulting in the insertion of the control rods in the core. In addition, the reactor coolant pump (RCP) bus undervoltage or underfrequency trips are not affected by CMF of the new digital equipment. These trips have Unit 1 and Unit 2 delay times of 1.2 and 1.5 seconds, respectively for undervoltage and 0.6 seconds for the underfrequency. Since the FSAR analysis conservatively models reactor trip on low-low steam generator level, it is expected that the actual trip will occur earlier than modeled due to RCP bus undervoltage or underfrequency.<br><br>This transient is analyzed to show the adequacy of the heat removal capability of the auxiliary feedwater system. The auxiliary feedwater pump (AFP) initiating signal on low-low steam generator level is assumed to be lost on CMF of the new digital equipment. However, start signals on loss of voltage for the motor driven AFP's and on RCP bus undervoltage for the turbine driven AFP's are unaffected. Motor driven AFP starts on safety injection and loss of main feedwater pumps are also unaffected. Since the FSAR analysis conservatively models AFP start on low-low steam generator level, it is expected that the actual AFP start will occur |
| | | 2. Reactor trip on Loss of Power to the control rod drive mechanism as a loss of AC Power (e.g., Offsite Power) | Not Affected | | | | |
| | | 3. Reactor trip on Low-Low steam generator water level | Lost | Indications Available<br>•Panel indication<br>•Panel recorder<br>•Computer indication<br>Diverse Alarms Available<br>•Steam generator level deviation alarm | | | |
| | | 4. Reactor trip on low reactor coolant flow | Lost | Indications Available<br>•Panel indication<br>•computer indication | | | |
| | | 5. Reactor trip on high pressurizer pressure | Lost | Indications Available<br>•Panel indication<br>•Panel recorder<br>•computer recorder<br>Diverse Alarms Available<br>•High pressure deviation<br>•High pressure (2325 psia)<br>•Three high pressure alarms at 2350 psia | | | |
| | | 6. Reactor trip on RCP undervoltage/under-frequency | Not affected | | | | |

| FSAR TRANSIENT # | TRANSIENT | TRIP/SAFEGUARD FUNCTION FOR RX TRIP (FSAR 14.1.12) | IMPACT OF COMMON MODE FAILURE (CMF) ON TRIP FUNCTION | ALARM/ALTERNATE INDICATION SYSTEM AVAILABLE | DIAGRAM # | CONSEQUENCES OF UNAVAILABILITY OF DIVERSE ALARM | EVALUATION OF EVENT |
|---|---|---|---|---|---|---|---|
| 14.1.12 (con't) | | | | | | | earlier than modeled due to loss of voltage and RCP bus undervoltage.<br><br>There are also several alternate alarms available to the operator. The steam generator level deviation alarm is available for low-low steam generator water level. High pressurizer pressure deviation and high pressure alarms are also available.<br><br>Therefore, there is no adverse impact of the CMF of the RPS on this event. |

| FSAR TRANSIENT # | TRANSIENT | TRIP/SAFEGUARD FUNCTION FOR RX TRIP (FSAR 14.1.13) | IMPACT OF COMMON MODE FAILURE (CMF) ON TRIP FUNCTION | ALARM/ALTERNATE INDICATION SYSTEM AVAILABLE | DIAGRAM # | CONSEQUENCES OF UNAVAILABILITY OF DIVERSE ALARM | EVALUATION OF EVENT |
|---|---|---|---|---|---|---|---|
| 14.1.13 | Turbine-generator Safety Analysis | None | None | None | | | This event is related to mechanical failure of the main turbine-generators. There is no reactor trip associated with this analysis. If there were to be a failure, one or more turbine trips, would be expected. A reactor trip, unaffected by CMF, would result from the turbine trip. Therefore, the common mode failure of the software of the new digital system has no impact on this event. |

| FSAR TRANSIENT # | TRANSIENT | TRIP/SAFEGUARD FUNCTION FOR RX TRIP (FSAR 14.2.1) | IMPACT OF COMMON MODE FAILURE (CMF) ON TRIP FUNCTION | ALARM/ALTERNATE INDICATION SYSTEM AVAILABLE | DIAGRAM # | CONSEQUENCES OF UNAVAILABILITY OF DIVERSE ALARM | EVALUATION OF EVENT |
|---|---|---|---|---|---|---|---|
| 14.2.1 | Radiological consequences of Fuel Handling Accident | None | None | None | | | Bounding fuel conditions are selected for the analysis of a hypothetical dropped fuel assembly for both Unit 1 and Unit 2. They are described in FSAR Sections Unit 1, 14.2.1 and Unit 2, 14.3.5-3. These analyses also assume that the accident occurs 100 hours after shutdown. Since the accident occurs when the reactor is already tripped, the common mode failure of the new digital equipment has no effect on this event. |

| FSAR TRANSIENT # | TRANSIENT | TRIP/SAFEGUARD FUNCTION FOR RX TRIP (FSAR 14.2.2) | IMPACT OF COMMON MODE FAILURE (CMF) ON TRIP FUNCTION | ALARM/ALTERNATE INDICATION SYSTEM AVAILABLE | DIAGRAM # | CONSEQUENCES OF UNAVAILABILITY OF DIVERSE ALARM | EVALUATION OF EVENT |
|---|---|---|---|---|---|---|---|
| 14.2.2 | Postulated Radioactive Releases due to Liquid-Containing Tank Failures | None | None | None | | | This event is not affected by a reactor trip or safeguards actuation. Therefore, the common mode failure of the software of the new digital equipment will not impact the results of this event. |

| FSAR TRANSIENT # | TRANSIENT | TRIP/SAFEGUARD FUNCTION FOR RX TRIP (FSAR 14.2.3) | IMPACT OF COMMON MODE FAILURE (CMF) ON TRIP FUNCTION | ALARM/ALTERNATE INDICATION SYSTEM AVAILABLE | DIAGRAM # | CONSEQUENCES OF UNAVAILABILITY OF DIVERSE ALARM | EVALUATION OF EVENT |
|---|---|---|---|---|---|---|---|
| 14.2.3 | Accidental Waste Gas Release | None | None | None | | | This event is not affected by a reactor trip or safeguards actuation. Therefore, the common mode failure of the software of the new digital reactor protection system will not impact the results of this event.<br><br>In the event of a volume control tank (VCT) rupture, VCT low level and VCT low-low level alarms would be anticipated. Various radiation alarms would also be anticipated including the unit vent alarm. A VCT low-low level will result in a refueling water sequence which will start the shutdown of the reactor. This combination of alarms and automatic actions would lead the operator to isolate letdown and proceed with an orderly shutdown. This scenario is unaffected by CMF of the new equipment. |

| FSAR TRANSIENT # | TRANSIENT | TRIP/SAFEGUARD FUNCTION FOR RX TRIP (FSAR 14.2.4) | IMPACT OF COMMON MODE FAILURE (CMF) ON TRIP FUNCTION | ALARM/ALTERNATE INDICATION SYSTEM AVAILABLE | DIAGRAM # | CONSEQUENCES OF UNAVAILABILITY OF DIVERSE ALARM | EVALUATION OF EVENT |
|---|---|---|---|---|---|---|---|
| 14.2.4 | Steam Generator Tube Rupture | 1. Reactor trip on low pressurizer pressure signal<br><br>2. Safety injection on pressurizer pressure-low | Reactor trip lost (memo dated 10/13/92 from W.G. Sotos to V.D. VanderBurg)<br><br>Safety injection lost (T/S Table 3.3-3) | <u>Indication Available</u><br>Panel indication<br>Panel recorder<br>computer indication<br><u>Diverse Alarms Available</u><br>Low pressure deviation (turn on backup heaters) via control system<br><br><u>Other:</u><br>High radiation alarm in:<br>Steam generator blowdown liquid<br>Steam jet air ejector vent effluent radiation monitor<br>Steam generator high level deviation (in affected S.G.)<br><br>Pressurizer low level deviation via control system<br>Pressurizer low level (block pressurizer heaters) via control system | FD-2101 | | The reactor trip assumed for calculating the mass transfer from the reactor coolant system through the broken tube in this event occurs on low pressurizer pressure signal. This trip is lost because of common mode failure (CMF) of the new digital equipment. The safety injection is also lost if CMF of the new digital equipment occurred.<br><br>The steam generator tube rupture event would result in a decrease in the pressurizer pressure and level. The pressurizer pressure low deviation alarm at 25 psig below controller setpoint (normal controller setpoint is 2085 psig for Unit 1 and 2235 psig for Unit 2)(Procedures 1,2 - OHP 4024.108, .208 Drop 8) and the pressurizer level deviation alarm at 5% below level programs (Procedures 1,2 - OHP 4024.108, .208 Drop 4) would actuate. This accident can be identified by the operator by either a condenser air ejector radiation alarm or a steam generator blowdown radiation alarm (FSAR, page 14.2.4-5 and SD.DCC-NE 101). The steam generator high level deviation alarm for the faulted steam generator is also available. Following these alarms, the operator actions are specified by plant procedure 01-OHP 4023.E-3. This emergency procedure will guide the operator through mitigation of the event.<br><br>It is anticipated that the incremental time for the operator to respond to the alarms produced by this event, evaluate the appropriate indications, and actuate protection and safeguards functions will result in a relatively small increase in the transfer of fluid from the primary to the secondary system. The ERG Background Document for E-3, SGTR indicates on p 26 that although the level in the affected steam generator may reach the top of the narrow range span, significant volume still exists before the steam generator fills with water.<br><br>Procedure 12 THP 6020 LAB.122 provides the guidelines for actions taken based on steam generator primary to secondary leak. |

| FSAR TRANSIENT # | TRANSIENT | TRIP/SAFEGUARD FUNCTION FOR RX TRIP (FSAR 14.2.5) | IMPACT OF COMMON MODE FAILURE (CMF) ON TRIP FUNCTION | ALARM/ALTERNATE INDICATION SYSTEM AVAILABLE | DIAGRAM # | CONSEQUENCES OF UNAVAILABILITY OF DIVERSE ALARM | EVALUATION OF EVENT |
|---|---|---|---|---|---|---|---|
| 14.2.5 | Rupture of a steamline (Steamline Break) | 1. Safety injection on following signals:<br>(i) Two out of three low pressurizer pressure signals | Signal lost | <u>Indication Available</u><br>Panel indication<br>Panel recorder<br>Computer indication<br><u>Diverse Alarms Available</u><br>Lo pressure deviation (turn on backup heaters) via control system | FD 2101, Rev.0, Sheet 1 | | For this transient, the reactor is assumed to be tripped with the most reactive control rod stuck in its fully withdrawn position at the start of the transient. All of the engineered safeguard functions are lost due to common mode failure (CMF) of the new digital equipment. The reactor trip on NIS overpower setpoint is not affected by CMF of new equipment. Also, there is audible power range overpower rod stop alarm at 103% of full power (Procedure 2-OHP 4024.210 Drop 19) which would alert the operator to a power increase for at power events.<br><br>Because of the failure of the software, automatic safety injection would not be available. For core response, zero power is limiting. However at high powers, the reactor power would escalate and the operator would enter the emergency operating procedure (EOP) due to high neutron flux trips (processed outside the new equipment)(setpoint≤ 109% power).<br><br>At zero power or just subcritical, reactor trips from the source range at 10E5 cps and intermediate range and power range low setpoint both at 25% RTP are also available.<br><br>Alternatively, the operator, when responding to the various diverse alarms available would observe from available indications that a Rx trip and safeguards actuation were required and provide manual trips and actuations.<br><br>The pressurizer pressure would decrease. There are three sources of pressure indication and one diverse alarm, i.e., Lo pressure deviation (turn on backup heaters) via control system, which are available to the operator. In addition, the low pressurizer level deviation and low level alarms would be expected . Level swell would be anticipated in the affected steam generator resulting in a steam generator level deviation alarm. A large break inside containment would produce an upper containment pressure high or low alarm. One alarm indicates either high or low containment pressure. Following reactor trip and safety injection the operator would verify feedwater isolation on safety injection |
| | | (ii) Two out of three differential pressure signals between a steam line and the remaining steamlines | Signal lost | <u>Indication Available</u><br>Panel indication<br>Computer indication | | | |
| | | (iii) Low steam pressure in two of four lines (Unit 2)<br>High steam flow in two lines coincident with low-low Tavg in two loops or steam pressure low in two loops (Unit 1) | Signal lost | <u>Indication Available</u><br>Same as for differential pressure signal (Unit 1 and Unit 2)<br>Steam flow and Tavg indications frozen on CMF (Unit 1)<br>Wide range temperature recorders available | | | |
| | | (iv) Two out of three high containment pressure signals | Signal lost | <u>Indication Available</u><br>Panel indication<br>Computer indication<br><u>Diverse Alarms Available</u><br>Upper containment pressure high or low (two alarms) | | | |
| | | 2. Reactor trip<br>(i) Overpower reactor trips (neutron flux) | Not affected | <u>Diverse Alarms Available</u><br>Power range over power rod stop | | | |
| | | (ii) OP ΔT reactor trip | Lost | | | | |
| | | 3. Reactor trip in conjunction with receipt of the safety injection (SI) signal | Not affected (However, all automatic SI actuations are lost. Therefore, this signal is functional on manual SI initiation only) | | | | |
| | | 4. Feedwater isolation on any safety injection signal | Not affected (However, all automatic SI actuations are lost. Therefore, this signal is functional on manual SI initiation only) | | FD-2103 Sheet 2 | | |
| | | 5. Steamline isolation:<br>(i) High-high containment pressure | Lost | <u>Indications Available</u><br>Panel indication<br>Computer indication<br><u>Diverse Alarms Available</u><br>Upper containment pressure high or low (two alarms) | | | |

| FSAR TRANSIENT # | TRANSIENT | TRIP/SAFEGUARD FUNCTION FOR RX TRIP (FSAR 14.2.5) | IMPACT OF COMMON MODE FAILURE (CMF) ON TRIP FUNCTION | ALARM/ALTERNATE INDICATION SYSTEM AVAILABLE | DIAGRAM # | CONSEQUENCES OF UNAVAILABILITY OF DIVERSE ALARM | EVALUATION OF EVENT |
|---|---|---|---|---|---|---|---|
| 14.2.5 (cont'd) | | (ii) High steam flow coincident with Lo-Lo Tavg<br><br>(iii) Low steam pressure in two loops (Unit 2)<br>High steam flow coincident with low steam pressure (Unit 1) | Lost<br><br>Lost | Indications Available<br>Wide range temperature recorders<br><br>Indication Available<br>Panel indication<br>Computer indication<br>Steam flow indication frozen on CMF (Unit 1)<br>Other Alarms/Indications<br>Low pressurizer level deviation<br>Low pressurizer level<br>Steam generator high level deviation containment dewpoint monitor (checked at least once per eight hours)<br>Ice condenser inlet doors open | | | or take manual action to trip them. The Emergency Operating procedures based on Emergency Response Guideline E-0 (HP-Rev.1B) provide recovery guidelines to the operator.<br><br>Simple extrapolations suggest that, with added delays for operator response, the return to power could be significantly higher than calculated for the FSAR. This could result in fuel clad damage. However, it is not believed that this will prevent the operator from bringing the unit to a safe condition using the Emergency Operating Procedures. The environmental impact of fuel clad damage is discussed in Section 14.2.7. |

| FSAR TRANSIENT # | TRANSIENT | TRIP/SAFEGUARD FUNCTION FOR RX TRIP (FSAR 14.2.6) | IMPACT OF COMMON MODE FAILURE (CMF) ON TRIP FUNCTION | ALARM/ALTERNATE INDICATION SYSTEM AVAILABLE | DIAGRAM # | CONSEQUENCES OF UNAVAILABILITY OF DIVERSE ALARM | EVALUATION OF EVENT |
|---|---|---|---|---|---|---|---|
| 14.2.6 | Rupture of Control Rod Drive Mechanism (CRDM) Housing (RCCA Ejection) | 1. Reactor trip on high neutron flux (high and low setting)<br>2. Reactor trip on high rate of neutron flux increase | Not affected<br><br>Not affected | | | | For this event, the two reactor trips occur on NIS overpower setpoint and the high rate of neutron flux increase setpoint. These two trip functions are not processed by the new digital equipment. Therefore, the FSAR results of this event are not affected by the common mode failure of the new digital reactor protection system.<br><br>No radiological dose assessment was performed, but the dose received at site boundary and a low population zone would be minimal (Unit 2 FSAR, page 14.3.5-5). The assessment previously performed by Advanced Nuclear Fuels, which is included in Tables 14.3.5-6 through 14.3.5-9, shows that the doses for this accident are well below 10CFR 100 guidelines. |

| FSAR TRANSIENT # | TRANSIENT | TRIP/SAFEGUARD FUNCTION FOR RX TRIP (FSAR 14.2.7) | IMPACT OF COMMON MODE FAILURE (CMF) ON TRIP FUNCTION | ALARM/ALTERNATE INDICATION SYSTEM AVAILABLE | DIAGRAM # | CONSEQUENCES OF UNAVAILABILITY OF DIVERSE ALARM | EVALUATION OF EVENT |
|---|---|---|---|---|---|---|---|
| 14.2.7 | Secondary Systems Accident Environmental Consequences (This Section of Unit 2 FSAR refers to Section 14.3.5 of Unit 2 FSAR) | Table I lists all events with dose consequences and indicates where the protection/safeguards functions are found.<br><br>TABLE I<br><br>EVENT — DISCUSSION OF EVENT<br>Loss of External Electric Load — 14.1.8<br>Loss of Normal Feedwater — 14.1.9<br>Loss of all AC Power to Plant Auxiliaries — 14.1.12<br>Fuel Handling Accident — 14.2.1<br>Locked Rotor — 14.1.6.2<br>Steam Generator Tube Rupture — 14.2.4<br>Rupture of a Steam Pipe — 14.2.5<br>Rupture of a Control Rod Drive Mechanism Assembly — 14.2.6<br>Single RCCA Assembly Withdrawal Incident — NA<br>LOCA — 14.3.1 and 14.3.2 | See TABLE I | See TABLE I | | | This section includes the discussion of the environmental consequences of a common mode failure (CMF) of the digital Foxboro equipment on several events. Table II lists all events for which dose consequences will be found.<br><br>TABLE II<br><br>EVENT — RADIOLOGICAL DISCUSSION OF EVENT<br>Loss of External Electric Load — 14.2.7 (this section)<br>Loss of Normal Feedwater — 14.2.7 (this section)<br>Loss of All AC Power to Plant Auxiliaries. — 14.2.7 (this section)<br>Fuel Handling Accident — 14.2.1<br>Locked Rotor — 14.1.6.2<br>Steam Generator Tube Rupture — 14.2.7 (this section)<br>Rupture of a steam Pipe — 14.2.7 (this section)<br>Rupture of a Control Rod Drive Mechanism Housing — 14.2.6<br>Single RCCA Assembly Withdrawal Incident — 14.3.5<br>LOCA — 14.3.5<br><br>The evaluations of the Loss of External Electrical Load (14.1.8), Loss of Normal Feedwater Flow (14.1.9), and Loss of all AC Power to the Plant Auxiliaries (14.1.12) did not indicate that the outcomes of these events would compromise any of this fission product barriers. These evaluations assumed alarms from control systems or other indications to alert the operator to the need for action. It was then assumed that he would take prompt action in accordance with his emergency operating procedures to manually actuate protection and safeguards functions as appropriate. Since no compromise of the fission product barriers resulted from the evaluations, the incident off site doses described in Section 14.2.7.2 remain valid.<br><br>For the steam break event, the evaluation of section 14.2.5 suggests a potential higher return to power when additional time is allocated for operator response to manually |

| FSAR TRANSIENT # | TRANSIENT | TRIP/SAFEGUARD FUNCTION FOR RX TRIP (FSAR 14.2.7) | IMPACT OF COMMON MODE FAILURE (CMF) ON TRIP FUNCTION | ALARM/ALTERNATE INDICATION SYSTEM AVAILABLE | DIAGRAM # | CONSEQUENCES OF UNAVAILABILITY OF DIVERSE ALARM | EVALUATION OF EVENT |
|---|---|---|---|---|---|---|---|
| 14.2.7 (cont'd) | | | | | | | initiate safety injection. If this leads to clad failure, the inventory of radioisotopes in the reactor coolant after the event will be larger than assumed in the 14.2.7 analysis. However, the analysis for 1% failed fuel and 10 gpm primary to secondary leak rate shows a 0-8 hr site boundary thyroid dose of 4 rem and a 0.3 rem site boundary whole body dose. These values are two orders of magnitude below the 10 CFR 100 acceptance criteria of 300 rem and 25 rem for thyroid and whole body doses respectively. Since these values are a very small fraction of the 10 CFR 100 criteria, it appears that clad failure will not cause these criteria to be exceeded.<br><br>An analysis to support alternate steam generator tube plugging criteria for Unit 1 has been submitted to the NRC. The analysis is described in WCAP-13187. It includes a methodology to ensure that the offsite dose is limited to 30 rem thyroid at the site boundary. This analysis assumes a 1% fuel defects and a 120 gpm leak during a steam break. At each outage when the steam generators are examined for degraded tubes, a conservative evaluation will be performed to ensure that, in the event of a steamline break, the 120 gpm leak rate is not exceeded. If a potential return to power should result in additional clad damage above that assumed in this evaluation, the 30 rem criterion could be exceeded. However, 30 rem is small compared to 10 CFR 100 limits.<br><br>We further observe that, in assuming multiple failures in safeguards actuation, it is not also necessary to assume other failures as well. If it is assumed that all rods insert, the very large $F_Q$ associated with the analyzed return to power will not be present. These $F_Q$'s can be ~10. It is the portion of the core associated with this power peak that is expected to suffer clad damage. Furthermore, when rods are inserted, the SDM will be doubled or more assuming a stuck rod worth greater than or ~800 pcm and excess SDM >400 pcm. It should also be noted, as discussed in Section 14.2.5, that at zero power or low powers, nuclear instrumentation trips from the source range and intermediate range detectors and the power range high range low setpoint are expected to protect against power excursion. |

| FSAR TRANSIENT # | TRANSIENT | TRIP/SAFEGUARD FUNCTION FOR RX TRIP (FSAR 14.2.7) | IMPACT OF COMMON MODE FAILURE (CMF) ON TRIP FUNCTION | ALARM/ALTERNATE INDICATION SYSTEM AVAILABLE | DIAGRAM # | CONSEQUENCES OF UNAVAILABILITY OF DIVERSE ALARM | EVALUATION OF EVENT |
|---|---|---|---|---|---|---|---|
| 14.2.7 (cont'd) | | | | | | | Finally, we believe that in the case of a large sudden steam break, there will be a major audible indication which would prompt the operator to early action. If the break were to develop gradually, the various alarms available will allow the operator to take action in a time frame that will prevent any clad damage.<br><br>Therefore, we conclude that a CMF in combination with other failures could result in releases larger than currently calculated but not in excess of 10 CFR 100 limits. In a more likely scenario in which large core peaking factors are avoided, the current calculations are expected to be unaffected because little or no clad damage would result.<br><br>Should CMF of the new digital equipment occur for the steam generator tube rupture event, the operator has to trip the reactor manually and isolate the broken steam generator following the guidelines given in emergency operating procedures. It has been assumed in our evaluation that the operator's response time is ~60 seconds. This one minute time is on addition to the 30 minutes alloted for operator action after the accident, within which time the pressure between the defective steam generator and the primary system is equalized, and the defective steam generator is isolated. Assuming a 1 gpm primary-to-secondary leak rate (maximum leak rate allowed by T.S) prior to the tube rupture, the 0-2 hour doses at site boundary are: thyroid=1.7 rem; whole body=0.02 rem. These doses are much lower than 10 CFR 100 guidelines of 300 rem thyroid and 25 rem whole body, respectively (Unit 1 FSAR page 14.2.7-6). The doses at the end of 31 minute of time would be minimally impacted by the delay in safeguards actuation hypothesized for a CMF. The release for SGTR are expected to remain much less than 10 CFR 100 guidelines even when a CMF is assumed. |

| FSAR TRANSIENT # | TRANSIENT | TRIP/SAFEGUARD FUNCTION FOR RX TRIP (FSAR 14.2.8) | IMPACT OF COMMON MODE FAILURE (CMF) ON TRIP FUNCTION | ALARM/ALTERNATE INDICATION SYSTEM AVAILABLE | DIAGRAM # | CONSEQUENCES OF UNAVAILABILITY OF DIVERSE ALARM | EVALUATION OF EVENT |
|---|---|---|---|---|---|---|---|
| 14.2.8 | Major Rupture of Main Feedwater Pipe (Feedline Break) | a) A reactor trip on any of the following conditions:<br><br>1. High pressurizer pressure | Trip lost | Indications Available<br>•Panel indication<br>•Panel recorder<br>•Computer indication<br><br>Diverse Alarms Available<br>•Hi pressure deviation via control system<br>•Hi pressure (2325 psia) via control system<br>•Three high pressure alarms at 2350 psia (memo dated 10/13/92 from W.A. Sotos to V.D. VanderBurg) | FD -2101 Sheet 1 | | This event was only evaluated for Unit 2. It is not in the Unit 1 license basis. A Unit 1 analysis is provided in the Unit 1 UFSAR for information only.<br><br>The FSAR analysis for this event has been performed at full power with and without loss of offsite power. This analysis assumes that a reactor trip is initiated when the low-low steam generator level trip setpoint in the ruptured steam generator is reached. The low-low steam generator water level trip is lost, if a common mode failure (CMF) of the new digital equipment occurs.<br><br>All the reactor trips and safety injection signals will be lost (Column 4) when CMF of new equipment occurs. Both the motor driven and turbine driven auxiliary feedwater systems are also lost except in situation described below.<br><br>The motor driven auxiliary feedwater pumps are not affected by CMF if the pumps started on 4KV bus loss of voltage or loss of all main feedwater pumps (T/S Table 3.3-3, page 3/4 3-19). The turbine driven auxiliary feedwater pump is also not affected by CMF if the pump started on reactor coolant pump bus undervoltage (T/S Table 3.3-3, page 3/4 3-20).<br><br>In case of CMF of the digital equipment, steam generator level deviation alarm, pressurizer pressure low deviation alarm, pressurizer low level deviation alarm, and pressurizer low level alarm are available to the operator. In addition, three alternate indications of the steam generator water level, pressurizer pressure, and pressurizer level are available to the operator.<br>These alarms and indications are expected to cause the operator to initiate protective and safeguards action relatively early in the event. Using the emergency operating procedures, the operator would very likely supply auxiliary feedwater to the intact steam generators earlier than the 10 minutes after the initiation assumed in the analysis. In addition, we do not believe it is necessary to assume an AFW pump failure in addition to CMF. In view of this and the fact that a conservatively small feedwater flow of |
| | | 2. Overtemperature ΔT | Trip lost | Wide range RCS temp recorders | FD-2102 Sheet 3 | | |
| | | 3. Low-low steam generator water level in any steam generator | Trip lost | Indications Available<br>•Panel indication<br>•Panel recorder<br>•Computer indication<br><br>Diverse Alarms Available<br>•Level deviation via. control system (memo dated 10/13/92 from W.G. Sotos to V.D. VanderBurg) | FD-2101 Sheet 5 | | |
| | | 4. Safety injection signals from any of the following:<br><br>(i) Two out of three differential pressure signals between a steam line and the remaining steamlines | Signal lost | Indication Available<br>Panel indication<br>Computer indication | | | |
| | | (ii) Low steam pressure in two of four lines | Signal lost | Indication Available<br>Same as for differential pressure signal | | | |
| | | (iii) Two out of three high containment pressure signals | Signal lost | Indication Available<br>Panel indication<br>Computer indication<br><br>Diverse Alarms Available<br>Upper containment pressure high or low (two alarms) | | | |

| FSAR TRANSIENT # | TRANSIENT | TRIP/SAFEGUARD FUNCTION FOR RX TRIP (FSAR 14.2.9) | IMPACT OF COMMON MODE FAILURE (CMF) ON TRIP FUNCTION | ALARM/ALTERNATE INDICATION SYSTEM AVAILABLE | DIAGRAM # | CONSEQUENCES OF UNAVAILABILITY OF DIVERSE ALARM | EVALUATION OF EVENT |
|---|---|---|---|---|---|---|---|
| 14.2.8 (cont'd) | | b) Auxiliary feedwater<br>(i) Two motor driven auxiliary feedwater pumps which are started on:<br>a. Low-low level in any steam generator<br>b. Trip of all main feedwater pumps<br>c. Any safety injection signal<br>d. 4 kv bus loss of voltage<br>e. Manual actuation<br><br>(ii) Turbine driven auxiliary feedwater pump is started on:<br>a. Low-low level in any two steam generators<br>b. Reactor coolant pump bus undervoltage | MDAFP starts (automatic initiation) or low-low steam generator level and safety injection from non-manual initiation are lost<br><br>TDAFP start (automatic initiation) on low-low steam generator level is lost | Other Alarms/Indications<br>•Pressurizer pressure low deviation<br>•Pressurizer level low deviation<br>•Pressurizer low level<br>•Pressurizer high level deviation<br>•Pressurizer high level | | | 600 gpm was assumed to be supplied to the intact steam generators, a substantially larger auxiliary feedwater flow can be expected to be supplied to the intact steam generators. On this basis, it is likely that the event not only would not be worse than the analyzed case, but could likely be less severe.<br><br>At all powers, the steam generator level deviation alarm is available. In addition, numerous alarms describing the status of the condensate and feedwater system pumps and pressures, such as condensate hotwell level, booster motor trip, main feedwater pump, etc. will activate. When at least two channels of feedwater are lost above 40%, the AMSAC timer will also initiate. If the timer is allowed to time out, a turbine trip and auxiliary feedwater pump start will be initiated. The turbine trip will result in a reactor trip which is unaffected by CMF. |

| FSAR TRANSIENT # | TRANSIENT | TRIP/SAFEGUARD FUNCTION FOR RX TRIP (FSAR 14.3.1) | IMPACT OF COMMON MODE FAILURE (CMF) ON TRIP FUNCTION | ALARM/ALTERNATE INDICATION SYSTEM AVAILABLE | DIAGRAM # | CONSEQUENCES OF UNAVAILABILITY OF DIVERSE ALARM | EVALUATION OF EVENT |
|---|---|---|---|---|---|---|---|
| 14.3.1 | Large Break Loss of Coolant Accident | 1. Reactor trip on low pressurizer pressure<br><br>2. Safety Injection (SI) on low pressurizer pressure<br><br>3. Containment spray on hi-hi pressure | Reactor trip lost<br><br>Safety injection signal lost<br><br>Hi-hi pressure spray actuation and ESF trip lost. | <u>Indications Available</u><br>- Panel indication<br>- Panel recorder<br>- Computer indication<br><u>Diverse Alarms Available</u><br>Pressurizer pressure low deviation (turn on backup heaters) via control system (memo dated 10/13/92 from W.G. Sotos to V.D. Vanderburg)<br><br><u>Indication Available</u><br>Panel Indication<br>Computer Indication<br><u>Diverse Alarms Available</u><br>Upper containment hi/lo pressure alarms available via. control system (memo date 10/13/92 from W.G. Sotos to V.D. Vanderburg).<br><br><u>Other Alarms/Indications</u><br>Lower containment radiation monitors (isolated on phaseB).<br>Upper containment area radiation monitors.<br>Post accident high range containment area monitors.<br>Pressurizer level low deviation alarm.<br>Pressurizer low level alarm.<br>Lower containment sump level high.<br>Containment air temperature high.<br>Accumulator level high or low (one alarm per accumulator).<br>Accumulator pressure high or low (one alarm per accumulator).<br>RCS hot leg pressure low<br>RCP Seal 1 diff pressure low (one alarm per RCP). | FD-2101 Sheet 1 | Diverse alarm for Lo pressure (turn on backup heaters) via control system is available. Consequences of unavailability of SI system is decreasing RCS inventory resulting in an increase of peak clad temperature.<br><br>The only protective function prior to operator action will be accumulator injection. The operator will be inundated by alarms for this event as indicated under the other Alarms/Indications heading. Nevertheless, we assume ≈60 seconds for the operator response time. Since the outcome of this event depends on prompt safeguards actuation, as modeled under Appendix K rules, elevated PCT and extensive fuel damage would be expected to be calculated by an Appendix K model. | The FSAR analysis of this event shows that a large break LOCA with discharge coefficient (Cd) of 0.6 is the most limiting case for Unit 2 with the RHR cross-ties open. For Unit 1, max SI case is limiting. The FSAR analysis assumes a reactor trip on low pressurizer pressure and subsequent initiation of safety injection, and accumulator injection at 600 psia. The low pressurizer pressure reactor trip and low pressure safety injection signals are lost, if a common mode failure (CMF) of the new digital instrumentation system occurs.<br><br>The large break LOCA results in a rapid depressurization of the reactor coolant system (RCS). The low pressurizer pressure deviation alarm will actuate at 25 psig below controller setpoint of 2235 psig (Procedure 2-OHP 4024.208 Drop 8). Figure 14.3.1-3a of Unit 2 FSAR shows that this alarm would actuate in less than one second of transient. Three alternate indications are available for the low pressurizer pressure. The upper containment high pressure alarm will actuate at +0.2 psig (Procedure 2-OHP 4024.105 Drop 31). These and other alarms as indicated under Other Alarms/Indication effectively warn the operator that a major accident is occurring.<br><br>Assuming that the operator's response time to mitigate the event is 60 sec., the reactor would be tripped at about 61 seconds of transient and subsequently initiate the safety injection and accumulator injection. In our evaluation, we assumed that the results given in FSAR are delayed by about 60 seconds. From Figure 14.3.1-15a, the peak clad temperature (PCT) of 2140°F occurs at about 260 second of transient.<br><br>LBLOCA is a very complicated event to model. Therefore, extrapolations of PCT are very uncertain. Attempting to extrapolate Figures 14.3.1-15a for Unit 2 and 14.3.1-13f for Unit 1 by inserting a delay of 60 seconds for operator response time suggests PCT's as high as the 3000°F range. However, the real situation is in all likelihood much less severe. Best estimate models are known to result in substantially lower PCT's. However, even if the Appendix K |

| FSAR TRANSIENT # | TRANSIENT | TRIP/SAFEGUARD FUNCTION FOR RX TRIP (FSAR 14.3.1) | IMPACT OF COMMON MODE FAILURE (CMF) ON TRIP FUNCTION | ALARM/ALTERNATE INDICATION SYSTEM AVAILABLE | DIAGRAM # | CONSEQUENCES OF UNAVAILABILITY OF DIVERSE ALARM | EVALUATION OF EVENT |
|---|---|---|---|---|---|---|---|
| 14.3.1 (cont'd) | | | | RCP Seal 1 leak off low (one alarm per RCP).<br>Loop RCP trip or low flow (one alarm per RCP).<br>Ice condenser inlet doors open.<br>Containment dewpoint monitor (checked at least once per eight hours) . | | | model is conservative by as much as 600°F, the acceptance criteria for 10CFR50.46 could still possibly be exceeded.<br><br>Although these estimates of the impact of a CMF on LBLOCA is of concern, it is unlikely that such an event will occur and even more unlikely that such an event will occur in coincidence with CMF. As indicated in Section 14.3.3 of the Unit 2 UFSAR, p 14.3.3-4, pipe whip restraints and other protective measures against the dynamic effects of a break in the main coolant piping are not required because "leak before break" can be assumed to allow for shutdown of the Cook Units before an event as catastrophic as a LBLOCA occurs. This argument also gives reasonable assurance that such an event in conjunction with a CMF is extremely unlikely. |

| FSAR TRANSIENT # | TRANSIENT | TRIP/SAFEGUARD FUNCTION FOR RX TRIP (FSAR 14.3.2) | IMPACT OF COMMON MODE FAILURE (CMF) ON TRIP FUNCTION | ALARM/ALTERNATE INDICATION SYSTEM AVAILABLE | DIAGRAM # | CONSEQUENCES OF UNAVAILABILITY OF DIVERSE ALARM | EVALUATION OF EVENT |
|---|---|---|---|---|---|---|---|
| 14.3.2 | Loss of Reactor coolant from small ruptured pipes or from cracks in large pipes which actuate the Emergency Core Cooling System (Break size <1.0ft$^2$) | 1. Reactor trip on low RCS pressure<br>2. Safety Injection (SI) on low RCS pressure (auto initiation) | 1. Lo-pressure Rx trip lost<br>2. SI (auto initiation) lost (memo 9/2/92 from W. G. Sotos to V. D. VanderBurg) | <u>Indication Available</u><br>1. Panel Indication<br>2. Panel Recorder<br>3. Computer Indication<br><u>Diverse Alarm Available</u><br>1. Pressurizer pressure low deviation via Control System (memo 9/2/92 from W. G. Sotos to V. D. VanderBurg)<br><u>Other Alarms/Indications</u><br>Lower containment radiation monitors (Isolated on phaseB)<br>Upper Containment area radiation monitors.<br>Pressurizer Level low deviation alarm<br>Pressurizer low level alarm<br>Containment dewpoint monitor (checked at least once per eight hours) | FD-2101 Rev. 00 Sheet 1 | Diverse Alarm for Lo Pressure via Control System is available. Consequence of unavailability of SI system is decreasing RCS inventory resulting in an increase of peak clad temperature. The period of core uncovery could be extended, if SI system is not actuated in a timely manner. (FSAR 14.3.2) | The small break loss of coolant accident results in depressurization of the reactor coolant system. The limiting break (as determined by the highest calculated peak fuel rod clad temperature) for the high head safety injection cross-tie valves opened is 4 inches in diameter for Unit 2 and 3 inches in diameter for Unit 1. A cold leg break was initiated at RCS pressure of 2100 psia and Tavg of 581.3°F for Unit 2. The Unit 1 initial Tavg was 547°F. For the Unit 2 case, the Rx trip was actuated at 1860 psia (FSAR, page 14.3.2-9). In the Unit 2 analysis, the safety injection (SI) signal actuated at 1715 psia with a 27 second time delay to account for diesel generator startup and emergency power bus loading in case of offsite power coincident with an accident. The maximum fuel cladding temperature attained during the transient was 1426°F (Units 2 UFSAR, page 14.3.2-12).<br><br>The common mode failure (CMF) results in loss of both Lo pressure Rx trip and automatic SI. However, for Lo pressurizer pressure, three alternate indications, and low pressure deviation via control system Diverse Alarm are available for the operator to trip the reactor manually. The alarm, PZR Pressure Low Deviation Backup Heaters On, will activate at 2210 psig (2-OHP 4024.208 Drop 8). The corresponding setpoint is 2060 psig for Unit 1.<br><br>SBLOCA is a very complicated event to model. Therefore, extrapolations of PCT are very uncertain. Attempts to extrapolate figures 14.3.2-4 for Unit 2 and 14.3.2-5 for Unit 1 by inserting an additional 60 seconds of heat up time to account for operator response time in lieu of automatic actuation led to incremental increase in PCT's of ~450°F and 200°F respectively. For Unit 2 there is a margin to accomodate a 500°F PCT increase for the cross-tie open case. The incremental PCT would lead to only ~1900°F PCT. For Unit 1 such margin appears not to exist. However, the Unit 1 SBLOCA analysis was performed at 3588 MWT for 15x15 fuel with the intent of bounding both Units. If one assumes the rule of thumb, 45°F for each 1% of power, there is 450°F of PCT margin due to this conservatism. Unit 1 |

| FSAR TRANSIENT # | TRANSIENT | TRIP/SAFEGUARD FUNCTION FOR RX TRIP (FSAR 14.3.2) | IMPACT OF COMMON MODE FAILURE (CMF) ON TRIP FUNCTION | ALARM/ALTERNATE INDICATION SYSTEM AVAILABLE | DIAGRAM # | CONSEQUENCES OF UNAVAILABILITY OF DIVERSE ALARM | EVALUATION OF EVENT |
|---|---|---|---|---|---|---|---|
| 14.3.2 (con't) | | | | | | | operates at 3250 MWT and there is no intent to increase this power. Thus there appears to be substantial PCT margin in the Appendix K SBLOCA model for Unit 1 also.<br><br>We further note that, as in the case of LBLOCA, the Appendix K model is substantially conservative. Furthermore, the analyzed events assumed the loss of a train of SI pumps. Such an assumption, in addition to the multiple failures of CMF, is also a substantial conservatism. Therefore, it is concluded that, even with additional operator response times relative to automatic actuation, 10CFR 50.46 acceptance criteria would likely be met for SBLOCA.<br><br>The high head safety injection cross-ties closed cases were not considered because the Cook Units are operated with these cross-ties open except for short periods of surveillance testing and maintenance. |

| FSAR TRANSIENT # | TRANSIENT | TRIP/SAFEGUARD FUNCTION FOR RX TRIP (FSAR 14.3.4) | IMPACT OF COMMON MODE FAILURE (CMF) ON TRIP FUNCTION | ALARM/ALTERNATE INDICATION SYSTEM AVAILABLE | DIAGRAM # | CONSEQUENCES OF UNAVAILABILITY OF DIVERSE ALARM | EVALUATION OF EVENT |
|---|---|---|---|---|---|---|---|
| 14.3.4 | Long Term Containment Integrity Analysis (Section 14.3.4 of Unit 2 refers to Unit 1 UFSAR Section 14.3.4) | 1. Containment spray on high-high pressure signal | Lost | <u>Indications Available</u><br>Panel indication<br>Computer indication<br><br><u>Diverse Alarms Available</u><br>Upper containment hi/lo pressure alarms available via control system (memo dated 10/13/92 from W.G. Sotos to V.D. VanderBurg)<br><br><u>Other Alarms/Indications</u><br>Pressurizer pressure low deviation (turn on backup heaters) via control system.<br>Lower containment radiation monitors (isolated on phaseB).<br>Upper containment area radiation monitors.<br>Post accident high range containment area monitors.<br>Pressurizer level low deviation alarm.<br>Pressurizer low level alarm.<br>Lower containment sump level high.<br>Containment air temperature high.<br>Accumulator level high or low (one alarm per accumulator).<br>Accumulator pressure high or low (one alarm per accumulator).<br>RCS hot leg pressure low<br>RCP Seal 1 diff pressure low (one alarm per RCP).<br>RCP Seal 1 leak off low (one alarm per RCP).<br>Loop RCP trip or low flow (one alarm per RCP).<br>Ice condenser inlet doors open.<br><br>Containment dewpoint monitor (checked at least once per eight hours). | FD-2103 Sheet 4 | | Only the long term containment pressure analysis is considered in this evaluation. The short term pressure analyses typically have peaks prior to the actuation of any protective or safeguards functions and are therefore not applicable to this evaluation. The mass and energy release rates for steamline breaks are considerably less than the RCS double-ended pump suction pipe breaks (Unit 1, FSAR, p. 14.3.4-18) and are, therefore, bounded. The containment temperature effects of steamline breaks are considered in Section 14.3.4/14.3.11, Electrical Equipment Environmental Qualification (Mass and Energy Release Inside Containment and Outside Containment).<br><br>The FSAR analysis of this event shows that pressure peaks about 2 hours into the event when the ice bed melts out. Therefore, as long as additional energy is not added to the containment as a result of common mode failure (CMF) of the new digital instrumentation, the peak pressure should not change. In large break LOCA, the reactor is promptly shut down by voids. The long term LOCA cooling analysis assures that it does not become critical again. If actuation of safeguards is delayed, PCT will be expected to rise above the analyzed value until the core is quenched at a delayed time and, therefore, addition fuel damage may occur. However, the net energy delivered to the containment is not impacted by a relatively small change of a minute or two in the removal of thermal energy from the core and delivery to the containment in the early minutes of the event. It is concluded that a delay of a few minutes in the actuation of safeguards will have no impact on the analysis of record.<br><br>Furthermore, since it is not necessary to assume that one train of safeguards fails in addition to CMF, it is reasonable to believe that the operator can manually activate two full trains of safeguards early in the event. On this basis, it is likely that the event not only would not be worse than the analyzed case, but would likely be less severe. |

| FSAR TRANSIENT # | TRANSIENT | TRIP/SAFEGUARD FUNCTION FOR RX TRIP (FSAR 14.3.4) | IMPACT OF COMMON MODE FAILURE (CMF) ON TRIP FUNCTION | ALARM/ALTERNATE INDICATION SYSTEM AVAILABLE | DIAGRAM # | CONSEQUENCES OF UNAVAILABILITY OF DIVERSE ALARM | EVALUATION OF EVENT |
|---|---|---|---|---|---|---|---|
| 14.3.4 (cont'd) | | | | | | | Although the impact of CMF on the containment pressure analysis does not seem to be significant, the pressure analysis is based on LBLOCA. It is unlikely that such an event will occur and even more unlikely that such an event will occur in coincidence with CMF. As indicated in Section 14.3.3 of the Unit 2 UFSAR, p 14.3.3-4, of pipe whip restraints and other protective measures against the dynamic effects of a break in the main coolant piping are not required because "leak before break" can be assumed to allow for shutdown of the Cook Units before an event as catastrophic as a LBLOCA occurs. This argument also gives reasonable assurance that such an event in conjunction with a CMF is extremely unlikely. |

| FSAR TRANSIENT # | TRANSIENT | TRIP/SAFEGUARD FUNCTION FOR RX TRIP (FSAR 14.3.5) | IMPACT OF COMMON MODE FAILURE (CMF) ON TRIP FUNCTION | ALARM/ALTERNATE INDICATION SYSTEM AVAILABLE | DIAGRAM # | CONSEQUENCES OF UNAVAILABILITY OF DIVERSE ALARM | EVALUATION OF EVENT |
|---|---|---|---|---|---|---|---|
| 14.3.5 | Radiological Consequences of a Loss of Coolant Accident and other Events Consideration in Safety Analysis. | Reactor trip/safeguard functions are included in the evaluation of FSAR Event 14.3.1. | Impact of CMF is discussed in the evaluation of event 14.3.1 | Discussed in the evaluation of event 14.3.1 | | | The Unit 2 UFSAR analysis of Radiological consequences of a LOCA includes analyses of several events for radiological consequences which were performed by Advanced Nuclear Fuels Corporation. These events are reviewed for the impact of common mode failure (CMF) in other sections of this evaluation. Table I lists all events for which dose consequences have been analyzed for Cook Units 1 and 2 and indicates in which section of this review a discussion of the impact of a CMF on the radiological consequences will be found. Section 14.3.5 of the Unit 1 UFSAR addresses only the Environmental consequences of a LOCA.<br><br>TABLE I<br><br>EVENT — DISCUSSION OF EVENT<br>Loss of External Electric Load — 14.2.7<br>Loss of Normal Feedwater — 14.2.7<br>Loss of All AC Power to Plant Auxiliaries — 14.2.7<br>Fuel Handling Accident — 14.2.1<br>Locked Rotor — 14.1.6.2<br>Steam Generator Tube Rupture — 14.2.7<br>Rupture of a steam Pipe — 14.2.7<br>Rupture of a Control Rod Drive Mechanism Housing — 14.2.6<br>Single RCCA Assembly Withdrawal Incident — 14.3.5 (this section)<br>LOCA — 14.3.5 (this section)<br><br>The single RCCA withdrawal event was analyzed for Unit 2 for cycle 6 operation. As a part of the transition to Westinghouse fuel in cycle 8, AEP argued and the NRC concurred that this event was not in the license basis for Donald C. Cook Nuclear Plant, Unit 2. NRC concurrence is documented in a letter from Joseph G. Giitter of the NRC staff to M.P. Alexich, dated August 3, 1989 and in the cycle 8 SER, dated August 27, 1990. Therefore, no new analysis of this event has been performed.<br><br>For the Cook Units, single RCCA withdrawal is anticipated to be an event with minor consequences. The units are generally operated at full power and base loaded. In this mode of operation, the RCCA's are nearly fully |

| FSAR TRANSIENT # | TRANSIENT | TRIP/SAFEGUARD FUNCTION FOR RX TRIP (FSAR 14.3.5) | IMPACT OF COMMON MODE FAILURE (CMF) ON TRIP FUNCTION | ALARM/ALTERNATE INDICATION SYSTEM AVAILABLE | DIAGRAM # | CONSEQUENCES OF UNAVAILABILITY OF DIVERSE ALARM | EVALUATION OF EVENT |
|---|---|---|---|---|---|---|---|
| 14.3.5 (cont'd) | | | | | | | withdrawn. Therefore, withdrawal of one RCCA a few steps has no impact. If a unit should be operating at a reduced power, an increase in DNBR margin is available. The Units are operated using the constant axial offset control method so that the controlling bank is seldom deeply inserted. In addition, the rod deviation alarm, which is unaffected by CMF, would be expected to alert the operator to take appropriate action.<br><br>The evaluations of small break LOCA (Event 14.3.2) and large break LOCA (Event 14.3.1) show that the large break LOCA event is bounding, as there would be significant clad failure, if common mode failure (CMF) of new digital instrumentation occurred, simultaneously with a LBLOCA.<br><br>Evaluation of the large break LOCA event (14.3.1) shows that the CMF of the new digital equipment could result in a peak clad temperature of approximately 3000°F on an Appendix K basis for both units. This temperature exceeds the acceptance criterion of 2200°F, thus resulting in significant clad failure and release of fission products.<br><br>The UFSAR analysis of the radiological effects of LOCA for both Units includes two cases. In the first case, identified as the design basis accident, it is assumed that the entire inventory of volatile fission products <u>contained in the pellet-cladding gap</u> of all the fuel rods is released during the time the core is being flooded by the ECCS. Of the gap inventory, 50% of the halogens and 100% of the noble gases are considered to be released to the containment atmosphere. In the second case, identified as the maximum hypothetical accident, it is assumed that 50% of the <u>core inventory</u> of halogens and 100% of the <u>core inventory</u> of noble gases are released to the containment atmosphere. Table 14.3.5-10 of the Unit 2 UFSAR and Table 14.3.5-7 of the Unit 1 UFSAR display the doses for both the design basis accident and the maximum hypothetical accident. As discussed in section 14.3.1, the delays related to substituting operator response time for electronic response time could result in substantially increased |

| FSAR TRANSIENT # | TRANSIENT | TRIP/SAFEGUARD FUNCTION FOR RX TRIP (FSAR 14.3.5) | IMPACT OF COMMON MODE FAILURE (CMF) ON TRIP FUNCTION | ALARM/ALTERNATE INDICATION SYSTEM AVAILABLE | DIAGRAM # | CONSEQUENCES OF UNAVAILABILITY OF DIVERSE ALARM | EVALUATION OF EVENT |
|---|---|---|---|---|---|---|---|
| 14.3.5 (cont'd) | | | | | | | fuel damage on an Appendix K basis. However, since the consequences of the maximum hypothetical accident are based on core inventory and since they meet the acceptance criteria of 10CFR100, we conclude that the analysis of this section is unaffected by CMF.<br><br>We further note that the analysis of section 14.3.5, p.p. 14.3.5-3, 4 and 13 of the Unit 1 UFSAR, assumes only one train of safeguards including only one CEQ fan operating. Although not explicitly stated, it is clear that containment pressure is maximized by degradation of safeguards including containment spray. See figure 14.3.5-3 of the Unit 1 UFSAR. These failure assumptions in addition to CMF are excessive.<br><br>As indicated in the evaluation of Section 14.3.1, there is susbstantial real margin in the use of an Appendix K model to estimate PCT. It is also unlikely that a large break LOCA will occur and it is even more unlikely that such event will occur in coincidence with CMF. As indicated in Section 14.3.3 of the Unit 1 UFSAR, p. 14.3.3-4, pipe whip restraints and other protective measures against the dynamic effects of a break in the main coolant piping are not required because "leak before break" can be assumed to allow for shutdown of the Cook Units before an event as catastrophic as a LBLOCA occurs. This argument also gives reasonable assurance that such an event in conjunction with a CMF is extremely unlikely. |

| FSAR TRANSIENT # | TRANSIENT | TRIP/SAFEGUARD FUNCTION FOR RX TRIP (FSAR 14.3.6) | IMPACT OF COMMON MODE FAILURE (CMF) ON TRIP FUNCTION | ALARM/ALTERNATE INDICATION SYSTEM AVAILABLE | DIAGRAM # | CONSEQUENCES OF UNAVAILABILITY OF DIVERSE ALARM | EVALUATION OF EVENT |
|---|---|---|---|---|---|---|---|
| 14.3.6 | Hydrogen in the Containment After a Loss-of-Coolant Accident | Reactor trip/safeguard functions are included in the evaluation of event 14.3.1. | Impact of CMF is discussed in the evaluation of event 14.3.1. | Discussed in the evaluation of event 14.3.1. | | | There are two hydrogen analyses for the Cook Plant (memo dated 11/16/92 from R.B. Bennett to R.S. Sharma). The first analysis, which is a part of original design basis, is given in FSAR 14.3.6. The second analysis, which does not appear in the FSAR is a response to the Three Mile Island accident (see above referenced memo). In this analysis, a very large amount of hydrogen is assumed to be generated by a severely damaged core, equivalent to 75% zirconium - water reaction. The hydrogen igniters were installed to ensure the structural integrity of the containment building and survivability of equipment and instruments needed to stop the progression of the accident. The NRC review of this analysis is not yet complete. If the reactor safeguards initiation system were to fail for large break LOCA, the evaluation of Section 14.3.1 suggests high PCT's. High PCT's would be expected to increase the hydrogen production. However, the hydrogen igniters are expected to be turned on manually for large break LOCA conditions through the Status Trees. The Emergency Operating Procedures FR-Z.1 and FR-C-1 would be used by the operator in response to high-high containment pressure and inadequate core cooling, respectively, to ensure that the ignitors would be available.<br><br>Thus, sufficient instrumentation and procedural guidance is available to the operator to prevent any adverse consequences of hydrogen combustion in the event of CMF of the new digital equipment. In Section 14.3.1, it was concluded that, although the impact of a CMF on LBLOCA is of concern, it is unlikely that such an event will occur and even more unlikely that such an event will occur in coincidence with CMF. As indicated in Section 14.3.3 of the Unit 2 UFSAR, p 14.3.3-4, pipe whip restraints and other protective measures against the dynamic effects of a break in the main coolant piping are not required because "leak before break" can be assumed to allow for shutdown of the Cook Units before an event as catastrophic as a LBLOCA occurs. This argument also gives reasonable assurance that such an event in conjunction with CMF is extremely unlikely. |

| FSAR TRANSIENT # | TRANSIENT | TRIP/SAFEGUARD FUNCTION FOR RX TRIP (FSAR 14.3.4, 14.4.11) | IMPACT OF COMMON MODE FAILURE (CMF) ON TRIP FUNCTION | ALARM/ALTERNATE INDICATION SYSTEM AVAILABLE | DIAGRAM # | CONSEQUENCES OF UNAVAILABILITY OF DIVERSE ALARM | EVALUATION OF EVENT |
|---|---|---|---|---|---|---|---|
| 14.3.4 and 14.4.11 | Electrical Equipment Environmental Qualification (Mass and Energy Releases Inside Containment and Outside Containment) | 1. Safety injection on following signals:<br>(i) Two out of three low pressurizer pressure signals | Signal lost | <u>Indication Available</u><br>Panel indication<br>Panel recorder<br>Computer indication<br><u>Diverse Alarms Available</u><br>Lo pressure deviation (turn on backup heaters) via control system | | | This event is divided into two parts, Mass and Energy (M&E) Release Inside containment and M&E Release Outside Containment.<br><br>The Containment Integrity analysis for the double ended pump suction RCS break case bounds the main steamline break containment pressure response. (WCAP 11902, Supplement 1, p S-3.4-2). Review of the pressure curves in WCAP 11902 Supp. 1 suggests that there is sufficient margin so that this will remain the case even if safeguards actuations are delayed by 1 to 2 minutes. If this judgement should be optimistic and one of the steamline M&E Release events should cause the containment pressure to exceed 12 psig, it is noted that the NRC in a letter from Steven A. Varga of the NRC staff to Mr. John Dolan of Indiana and Michigan Electric Company accepted 36 psig as the containment ultimate strength. Therefore, this issue will not be considered further.<br><br>The temperature profiles in WCAP 11902 Supp 1 for the Main Steamline Break (MSLB) Containment Integrity were reviewed for this evaluation. Two limiting transients are discussed. They are 4.6 sqft double ended rupture (DER) at 102% RTP and a 0.86 ft split break at 102% RTP. Both of these include single failures, main steam isolation failure for the DER and auxiliary feedwater pump runout protection failure for the split. It is not necessary to assume these failures in addition to the common mode failure (CMF) of the new digital instrumentation.<br><br>The temperature and pressure peaks of the DER occur at 6.4 seconds and 14.01 seconds respectively. This is well before the first safeguards of steamline isolation at 18.5 seconds but near and after reactor trip at ~6 seconds. Therefore, it is estimated that the impact of the CMF would be relatively modest.<br><br>The temperature and pressure peaks of the split occur later at 50.72 seconds. The temperature and pressure trajectories are on the rise at the time of the peaks. The rise is terminated by containment spray (CTS) actuation. It appears that the temperature could exceed the 330°F to |
| | | (ii) Two out of three differential pressure signals between a steam line and the remaining steamlines | Signal lost | <u>Indication Available</u><br>Panel indication<br>Computer indication | | | |
| | | (iii) High steam flow in two lines coincident with low-low Tavg in two loops or steam pressure low in two loops (One analysis bounds both Units) | Signal lost | <u>Indication Available</u><br>Same as for differential pressure signal<br>Steam flow indication frozen on CMF | | | |
| | | (iv) Two out of three high containment pressure signals | Signal lost | <u>Indication Available</u><br>Panel indication<br>Computer indication<br><u>Diverse Alarms Available</u><br>Upper containment pressure high or low (two alarms) | | | |
| | | 2. Reactor trip<br>(i) Overpower reactor trips (neutron flux) | Not affected | <u>Diverse Alarms Available</u><br>Power range over power rod stop | | | |
| | | (ii) OP ΔT reactor trip | Lost | <u>Indications Available</u><br>Wide range RCS temperature recorders | | | |
| | | 3. Reactor trip in conjunction with receipt of the safety injection (SI) signal | Not affected (However, all automatic SI actuations are lost. Therefore, this signal is functional on manual SI initiation only) | | | | |
| | | 4. Feedwater isolation on any safety injection signal | Not affected (However, all automatic SI actuations are lost. Therefore, this signal is functional on manual SI initiation only) | | | | |
| | | 5. Steamline isolation:<br>(i) High-high containment pressure | Lost | <u>Indications Available</u><br>Panel indication<br>Computer indication<br><u>Diverse Alarms Available</u><br>Upper containment pressure | | | |

| FSAR TRANSIENT # | TRANSIENT | TRIP/SAFEGUARD FUNCTION FOR RX TRIP (FSAR 14.3.4 + 14.4.11) | IMPACT OF COMMON MODE FAILURE (CMF) ON TRIP FUNCTION | ALARM/ALTERNATE INDICATION SYSTEM AVAILABLE | DIAGRAM # | CONSEQUENCES OF UNAVAILABILITY OF DIVERSE ALARM | EVALUATION OF EVENT |
|---|---|---|---|---|---|---|---|
| 14.3.4 and 14.4.11 (cont'd) | | (ii) High steam flow coincident with Lo-Lo Tavg<br><br>(iii) High steam flow coincident with low steam pressure (One analysis bounds both Units) | Lost | <u>Indications Available</u><br>Wide range RCS temperature recorders<br><u>Indication Available</u><br>Panel indication<br>Computer indication<br>Steam flow indication frozen on CMF<br><u>Other Alarms/Indications</u><br>Low pressurizer level deviation<br>Low pressurizer level<br>Steam generator high level deviation<br>Ice condenser inlet doors open<br>Containment dewpoint monitor (checked at least once per eight hours). | | | which containment equipment is qualified if the actuation of CIS were delayed by ~1 to 2 minutes. However, transmitters are tested to ~400°F and are encased in thick cast iron cases. It is expected that the thermal lag of these cases can accommodate one or two minutes of delay. CIS actuation is step 13 of Emergency Operating Procedure E-0 and is expected soon after entry into the procedure. When CIS is actuated, it is expected that both trains would be available and that the spray would rapidly condense the steam and cool the environment to temperatures well below that calculated in the analysis of record which assumes only one train of CIS. This is expected with approximately one minute delay relative to the analysis of record.<br><br>The ability of the operator to respond to available alarms and indications and enter the emergency operating procedures is discussed in Section 14.2.5. It is expected that the delay in actuation of safeguards and protective functions would be ~1 minute. Based on this and the discussion above, it is concluded that a M&E release of the magnitude of the limiting cases with a CMF would result in acceptable consequences.<br><br>The M&E release outside of containment is analyzed to ensure survivability of instruments and equipment in the main steam enclosures. The following evaluation is based on a memo dated 11-20-92 from R.B. Bennett to R.S. Sharma, "Cook Nuclear Plant, Failure of Reactor Protection System Impact of steamline Break Inside and Outside of Containment". In this event, a large steam flow eventually uncovers the steam generator tubes, allowing the exiting steam to become superheated in passing across the tubes. Superheat is the primary concern for this event. Pressure effects are over in a few seconds, so the reactor protection and safeguards actuation system does not come into play for pressure effects. The analysis performed shows that, for the limiting breaks (1.0-1.2 $ft^2$), the reactor trip occurred at 108 seconds or greater based on |

| FSAR TRANSIENT # | TRANSIENT | TRIP/SAFEGUARD FUNCTION FOR RX TRIP (FSAR 14.3.4 + 14.4.11) | IMPACT OF COMMON MODE FAILURE (CMF) ON TRIP FUNCTION | ALARM/ALTERNATE INDICATION SYSTEM AVAILABLE | DIAGRAM # | CONSEQUENCES OF UNAVAILABILITY OF DIVERSE ALARM | EVALUATION OF EVENT |
|---|---|---|---|---|---|---|---|
| 14.3.4 and 14.4.11 (cont'd) | | | | | | | low steam generator level. Significant levels of superheat occurred minutes later. Since the steam generator level alarms would be reached much earlier than the conservatively calculated steam generator level setpoint, the effects of main steamline break on equipment would be within the analyzed bounds.<br><br>The only plausible fast acting break is 1.4 $ft^2$, which predicts a reactor trip at 8 seconds on either low steamline pressure (Unit 2) or low steamline pressure coincident with high steam flow (Unit 1). The reactor trip at 60 seconds delay (operators response time) for 1.4 $ft^2$ (68 seconds) should still be bounded by the analyzed 1.2 $ft^2$ break with trip at 108 seconds.<br><br>For the most recent mass and energy release outside containment analysis, a calculation of the heat up of the cast iron cases was performed. Therefore, part of the margin represented in the thermal lag due to the cast iron cases has been used. However, the fact that the transmitters have been tested to ~400°F does apply to these transmitters and provides assurance that the instruments are likely to function even if the temperature briefly exceeded the qualification temperature. In addition, in the very worst scenario, only the instrumentation associated with ruptured steam line and one other steam line would be damaged. This is the case because the steam enclosures for steam lines one and four exit containment on one side and the steam enclosures for lines two and three exit 180° away on the opposite side of the containment. Therefore two steam lines with functioning instrumentation are available to control the system until it can be placed on RHR in this worst case scenario. Based on this and the discussion above, it is concluded that a M&E release of the magnitude of the limiting cases with a CMF would result in acceptable consequences. |

## APPENDIX B

## NOT APPLICABLE EVENTS

FSAR Section 14.3.3

This section addresses the mechanical forces from LOCA, Design Basis Earthquake (DBE), and combined LOCA/DBE.

The Unit 2 FSAR documents the applicability of leak before break to Cook.

The most recent analyses of this type are described in WCAP 11902 and the Unit 2, Cycle 8 RTSR.

These events consider approximately the first second of the transient and are not impacted by protection or safeguards actuation.

FSAR Section 14.3.7

This section addresses the overpressurization of the vessel after cooldown. The UFSAR material from 1982 appears not to address the ERG based EOP's.

The current material is the ERG background material. The ERG material is symptom based.Actions required of the operator are based on the results of an analysis based on a step temperature change in the cold leg. The initial temperature was chosen to be a conservatively high 550°F. The actions are then based on the observed temperature during the course of the implementation of the EOP's. The temperature and pressure are monitored continuously throughout the application of the EOP's by status tree F-0.4, Integrity. (If one exceeds curve A of the status tree criterion, a soak time is required). See p.p. 4, 8 of F-0.4 background and p. 5 of FR-P.1 background. Based on the nature of the ERG analysis, this event is not believed to be impacted by a common mode failure of the new digital equipment.

This opinion was discussed with Satyan-Sharma on Nov. 13, 1992. He concurred.

FSAR Section 14.3.8

This section describes an analysis to show that the RCS will not depressurize below the $N_2$ injection point from the accumulators prior to the time when S.G. cooling is no longer needed for SBLOCA. Cases with and without operator action are considered.

This material is superseded, or at least modified, in view of the ERG based EOP's. Operator action is provided as required for any event to ensure isolation of the accumulators prior to the injection of nitrogen into the reactor coolant system. At least the following events were addressed. (The step numbers are ERG numbers not EOP numbers).

| | | | |
|---|---|---|---|
| LBLOCA | E-1 | Loss of Rx or Secondary Coolant | Step 15 |
| SBLOCA | ES-1.2 | Post LOCA Cooldown and Depressurization | Step 23 |
| Loss of Sump Recirculation | ECA-1.1 | Loss of Emergency Coolant Recirculation | Steps 23, 31 |
| Steam Break/4 Loop | ECA-2.1 | Uncontrolled Depressurization of all S.G.'s | Steps 10, 38 |
| SGTR | ECA-3.1 | Recovery Modes | Step 28 |
| | ECA-3.2 | | Step 23 |
| Inadequate Core Cooling | FR-C.1 | Response to ICC | Step 12 |
| Degraded Core Cooling | FR-C.2 | Response to DCC | Step 12 |

It should be noticed that the issue is more broadly addressed in the ERG's than in the UFSAR.

The UFSAR cases with no operator response are irrelevant to this evaluation because operator response must be achieved on the loss of nearly all protection and safeguards actuations to achieve a satisfactory outcome. The operator action cases are superseded by the ERG analyses.

The ERG decision to isolate the accumulators is based on observable parameters and is not impacted by an additional delay of ≈1 minute. The ERG analyses in support of SBLOCA's (1" break) show that the accumulators will be isolated on subcooling not on low primary pressure.

For larger breaks, those for which primary pressure stabilizes at or below approximately 300 psig, the accumulators are isolated after the accumulators have injected. See response not obtained for step 15 of E-1.

In conclusion, the ERG's address the issue in Section 14.3.8 more currently than the FSAR. The ERG's are symptom based and address a wide range of contingencies. They are not directly affected by an additional delay of ≈1 minute in obtaining a protection or safeguards action. They are designed in sufficient depth to provide assurance that a unit can be brought to a safe and stable condition following any accident.

FSAR Section 14.4.2

This section is a general description of the analysis of high energy line breaks outside of containment. The material in this section is further elaborated in sections 14.4.3 through 14.4.11. A high energy line is a line with normal service temperature above 200°F, a normal operating pressure above 275 psig, and a nominal diameter greater than 1 inch. Five systems were determined to include high energy lines. They are:

1) Main Steam
2) Feedwater
3) CVCS
4) S.G. Blowdown
5) Steam to TDAFP

Breaks in high energy lines were examined for:

1) Pipe Whip
2) Jet Impingement
3) Jet Erosion of Concrete
4) Compartment Pressure - Loading Stress
5) Structural Resistance to Loading
6) Equipment E.Q.

Item 3 was determined not to be a problem in general. Breaks were analyzed for criteria 1, 2, 4, 5, and 6. Cracks were analyzed for 1, 2, and 6.

An ESW flood incident is also included in this section.

No impact of the postulated "freeze" of the Foxboro digital software on these analyses or those of Sections 14.4.3 through 14.4.11 was identified except as indicated in the following comments.

FSAR Section 14.4.3

This section addresses, in a general way, the ability to bring the reactor to a safe condition following the events evaluated for high energy line breaks. As indicated on p 14.4.3-1 of the Unit 1 UFSAR, they are general because "it is deemed appropriate to allow for assessment of the incident prior to ultimately bringing the reactor to cold shutdown".

Main steamline breaks (MSLB) are discussed in section 14.2.5 from the point of view of core response and in section 14.2.7 from the point of view of offsite dose effects. MSLB outside of containment from the point of view of equipment qualification (EQ) is addressed in UFSAR sections 14.4.6, 14.4.10, and 14.4.11. The evaluation of the impact of common mode failure (CMF) of the new digital equipment on MSLB EQ has been placed in section 14.4.11.

Feed water line break was analyzed from the core response point of view in section 14.2.8. The M&E release from a feedline break is believed to be similar with or without CMF. Unit 2 UFSAR Figure 14.2.8-4 suggests that the affected S.G. blowdown for a feedwater line break takes ≈200 sec. By this time, it is believed that the operator will be well into his immediate actions. Steamline isolation is step 12 of E-0. The operator will certainly be well into immediate actions, if there is a turbine trip. If there is no turbine trip, the turbine is a significant competitor for steam from the intact steam generators. Failure of a steam generator stop valve would also not be assumed in addition to the multiple failures of the CMF. Therefore, blowdown of the mainsteam lines would not occur after manual initiation of mainsteamline isolation.

CVCS line break assumes operator action. The alarms assumed continue to be available from the control system, and therefore, are not affected. This description is not affected.

Both the turbine driven auxiliary feedwater pump and steam generator blowdown line rupture are considered to be small steamline ruptures according to the UFSAR. Therefore, their effects would be expected to be bounded by MSLB and feedwater line break.

No impact of the postulated "freeze" of the Foxboro digital software on events other than MSLB was identified. Since MSLB will be discussed under section 14.4.11, the section is classified as NA.

<u>FSAR Section 14.4.4</u>

This section provides the quantitative results of stress calculations for high energy line breaks. See the discussion of Section 14.4.2 above.

<u>FSAR Section 14.4.5</u>

This section provides some further elaboration on the pipe whip analysis. See the discussion of Section 14.4.2. Note that this analysis uses the maximum operating pressure for conservatism.

<u>FSAR Section 14.4.6</u>

This section provides further details on the pressure analysis outside containment due to a high energy line break. The pressure peaks appear in the first second or two and cannot be impacted by an increase in time until reactor trip. Therefore, the pressure peak aspect of this section is classified as not applicable.

Temperature peaks are ≈5 minutes into the event presumably due to heat sinks. The impact of steam generator superheat from a MSLB outside containment on equipment qualification is addressed in this section. Without automatic safeguards functions, the environmental conditions could potentially be worse.

The equipment qualification aspect of this section is combined with Section 14.4.11 where events which impact environmental conditions and which are mitigated by protection and safeguards actuations are discussed. These events are mass and energy release inside and outside containment.

FSAR Section 14.4.7

This section provides some further elaboration on the jet impingement analysis. It also uses the maximum operating pressure. See the discussion of Section 14.4.2.

FSAR Section 14.4.8

This section describes the impact of high energy line breaks on the containment exterior. See the discussion of Section 14.4.2.

FSAR Section 14.4.9

This section describes the modifications required by the high energy line analysis. It will not be affected by the Foxboro "freeze".

FSAR Section 14.4.10

This section describes the steps taken to ensure that the adverse environmental conditions that result from HELB do not inhibit the ability to bring the reactor to cold shutdown. Without automatic safeguard functions, the environmental conditions could potentially be worse. This section is combined with Section 14.4.11 where events which impact environmental conditions and which are mitigated by protection and safeguards actuations are discussed. These events are mass and energy release inside and outside containment.