Tennessee Valley Authority, Post Office Box 2000, Decatur, Alabama 35609-2000

August 31, 2010

10 CFR 50.73

ATTN: Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, D. C. 20555-0001

Browns Ferry Nuclear Plant, Units 1, 2, and 3
Facility Operating Licenses Nos. DPR-33, DPR-52, and DPR-68
NRC Docket Nos. 50-259, 50-260, and 50-296

Subject: **Licensee Event Report 50-259/2009-005 Revision 2**

On October 20, 2009, the Tennessee Valley Authority (TVA) submitted Revision 0 of Licensee Event Report (LER) 50-259/2009-005. At that time, TVA was completing engineering evaluations needed to determine the impact of the Reactor Vessel Water Level 1 initiation logic including the Common Accident Signal (CAS) Logic on the Appendix R Safe Shutdown Program at the Browns Ferry Nuclear Plant (BFN). TVA completed the engineering evaluation, determined the impact of the logic on the Appendix R Safe Shutdown Program, and on November 3, 2009 submitted Revision 1 of LER 50-259/2009-005. TVA subsequently performed additional sensitivity evaluations and identified the different results than previously reported; therefore, TVA is providing the enclosed LER revision.

TVA is submitting this report in accordance with 10 CFR 50.73(a)(2)(v)(A), as any event or condition that could have prevented the fulfillment of the safety function of structures or systems that are needed to shutdown the reactor and maintain it in a safe shutdown condition.

IE22
NRR

There are no new regulatory commitments contained in this letter. The revisions are identified by bars in the right-hand margin. Should you have any questions concerning this submittal, please contact James E. Emens, Site Licensing and Industry Affairs Manager, at (256) 729-2636.

Respectfully,

K. J. Polson
Vice President

Enclosure
cc (w/ Enclosure):

NRC Regional Administrator - Region II

NRC Senior Resident Inspector - Browns Ferry Nuclear Plant

NRC FORM 366 (9-2007) U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104 EXPIRES 08/31/2010
LICENSEE EVENT REPORT (LER) (See reverse for required number of	Estimated burden per response to comply with this mandatory collection request: 80 hours. Reported lessons learned are incorporated into the licensing process and fed back to industry. Send comments regarding burden estimate to the Records and FOIA/Privacy Service Branch (T-5 F52), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to infocollects@nrc.gov, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202, (3150-0104), Office of Management and Budget, Washington, DC 20503. If a means used to impose an information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection.

1. FACILITY NAME	2. DOCKET NUMBER	3. PAGE
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4. TITLE: Reactor Vessel Water Level 1 Initiation Logic Including the Common Accident Logic Not Evaluated for Appendix R Fire Event

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO.	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
									Browns Ferry Nuclear Plant Unit 2	05000260
08	21	2009	2009	- 005 -	02	08	31	2010	FACILITY NAME: Browns Ferry Nuclear Plant Unit 3	DOCKET NUMBER: 05000296

9. OPERATING MODE: 1

10. POWER LEVEL: 100

11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: *(Check all that apply)*

☐ 20.2201(b)	☐ 20.2203(a)(3)(i)	☐ 50.73(a)(2)(i)(C)	☐ 50.73(a)(2)(vii)
☐ 20.2201(d)	☐ 20.2203(a)(3)(ii)	☐ 50.73(a)(2)(ii)(A)	☐ 50.73(a)(2)(viii)(A)
☐ 20.2203(a)(1)	☐ 20.2203(a)(4)	☐ 50.73(a)(2)(ii)(B)	☐ 50.73(a)(2)(viii)(B)
☐ 20.2203(a)(2)(i)	☐ 50.36(c)(1)(i)(A)	☐ 50.73(a)(2)(iii)	☐ 50.73(a)(2)(ix)(A)
☐ 20.2203(a)(2)(ii)	☐ 50.36(c)(1)(ii)(A)	☐ 50.73(a)(2)(iv)(A)	☐ 50.73(a)(2)(x)
☐ 20.2203(a)(2)(iii)	☐ 50.36(c)(2)	☒ 50.73(a)(2)(v)(A)	☐ 73.71(a)(4)
☐ 20.2203(a)(2)(iv)	☐ 50.46(a)(3)(ii)	☐ 50.73(a)(2)(v)(B)	☐ 73.71(a)(5)
☐ 20.2203(a)(2)(v)	☐ 50.73(a)(2)(i)(A)	☐ 50.73(a)(2)(v)(C)	☐ OTHER
☐ 20.2203(a)(2)(vi)	☐ 50.73(a)(2)(i)(B)	☐ 50.73(a)(2)(v)(D)	Specify in Abstract below or in NRC Form 366A

12. LICENSEE CONTACT FOR THIS LER

NAME	TELEPHONE NUMBER *(Include Area Code)*
Steve Austin, Licensing Engineer	256-729-2070

13. COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX

14. SUPPLEMENTAL REPORT EXPECTED	15. EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
☐ YES *(If yes, complete 15. EXPECTED SUBMISSION DATE)* ☒ NO		N/A	N/A	N/A

ABSTRACT *(Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines)*

On August 21, 2009, Browns Ferry Nuclear Plant (BFN) personnel concluded that a condition of non-compliance with the 10 CFR Part 50 Appendix R Safe Shutdown Program existed that could potentially prevent BFN from meeting one of the safe shutdown performance goals. That performance goal is to prevent fuel damage by limiting the fuel cladding temperature to less than 1500 degrees F. Engineering personnel evaluated the July 23, 2009, simulator response experienced during an operator training exercise for Fire Area 9, Safe Shutdown Instruction (SSI)-9, Unit 2 Reactor Building 4kV Electric Board Room 2A. During the depressurization phase of the scenario required by Safe Shutdown Instruction (SSI) 9, the reactor vessel water level momentarily dropped below the Reactor Vessel Water Level 1 setpoint (-122 inches). The reactor vessel water Level 1 setpoint initiates Common Accident Signal logic as well as the Primary Containment Isolation Signal, Core Spray system, and Residual Heat Removal system logic. The consequences of this logic initiation required the operators to reestablish some of the Appendix R system alignments specified in SSI-9. The actions necessary to reestablish the Appendix R alignments were not contained in SSI-9. The BFN Safe Shutdown Methodology, established in 1986, did not consider the impacts of the Level 1 logic actuations. Subsequent engineering evaluations of the plant response to these events resulted in a determination that the ability of the operators to safely shutdown the units and maintain them in a safe condition was not adversely affected.

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NARRATIVE *(If more space is required, use additional copies of NRC Form 366A) (17)*

I. PLANT CONDITION(S)

The condition was identified on August 21, 2009, by engineering evaluation of the July 23, 2009, response experienced on the Browns Ferry Simulator during the performance of Safe shutdown Instruction (SSI)-9, Unit 2 Reactor Building 4KV Electric Board Room 2A. Units 1, 2, and 3 were at 100 percent power and were not affected by the event.

II. DESCRIPTION OF EVENT

A. Event:

On August 21, 2009, Browns Ferry Nuclear Plant (BFN) personnel concluded that a condition of non-compliance with the 10 CFR Part 50 Appendix R Safe Shutdown Program existed that could potentially prevent BFN from meeting one of the safe shutdown performance goals. That performance goal is to prevent fuel damage by limiting the fuel cladding temperature to less than 1500 degrees F.

Engineering personnel evaluated the July 23, 2009, simulator response experienced during an operator training exercise for Fire Area 9. SSI-9 requires rapid depressurization of Unit 2 reactor coolant system to establish alternate decay heat removal. During the depressurization phase of the scenario required by SSI-9, the reactor vessel water level momentarily dropped below the Reactor Vessel Water Level 1 setpoint (-122 inches). The Level 1 setpoint initiates Common Accident Signal (CAS) [JC] logic as well as the Primary Containment Isolation Signal (PCIS) [JE], Core Spray (CS) [BM] system, and Residual Heat Removal (RHR) [BO] system logic. The consequences of this logic initiation required the operators to reestablish some of the Appendix R system alignments specified in SSI-9. The actions necessary to reestablish the Appendix R alignments were not contained in SSI-9.

For all of the SSIs that contain a requirement to depressurize the reactor vessel within 20 minutes, the reactor water level is predicted to decrease below the Level 1 setpoint. The consequences of this logic actuation may impact required system alignments as well as result in unanalyzed loading of the 4kV power supplied for credited equipment. The BFN Safe Shutdown Methodology, established in 1986, did not consider the impacts of the Level 1 Logic initiation.

The Tennessee Valley Authority is submitting this report in accordance with 10 CFR 50.73(a)(2)(v)(A), as any event or condition that could have prevented the fulfillment of the safety function of structures or systems that are needed to shutdown the reactor and maintain it in a safe shutdown condition.

B. Inoperable Structures, Components, or Systems that Contributed to the Event:

None.

C. Dates and Approximate Times of Major Occurrences:

January 1986	BFN Safe Shutdown Analysis established the Safe Shutdown Methodology.
July 23, 2009	During a simulated performance of SSI-9, an unexpected breaker trip occurred on the electrical alignment supporting Unit 1 depressurization.
July 23, 2009	BFN personnel initiated a Problem Evaluation Report (PER) to investigate the condition and to validate the accuracy of the simulator modeling of the event.

August 21, 2009	The evaluation of the training scenario concluded the event presented during the training scenario associated with SSI-9 was valid and would affect time critical alignments in other SSIs and potentially affect safe shutdown during a postulated Appendix R event.
August 21, 2009 at 1502 hours CDT	BFN made an 8-hour non-emergency report per 10 CFR 50.72(b)(3)(v)(A).

D. Other Systems or Secondary Functions Affected

None.

E. Method of Discovery

Site engineering identified the BFN Appendix R Safe Shutdown Program noncompliance on BFN Units 1, 2, and 3, while performing an evaluation of the simulator response during a training exercise.

F. Operator Actions

None.

G. Safety System Responses

None.

III. CAUSE OF THE EVENT

A. Immediate Cause

Safe Shutdown Instructions did not contain the necessary steps to prevent the Level 1 Logic from interfering with equipment credited in the Safe Shutdown Analysis.

B. Root Cause

The basic approach for the Appendix R Safe Shutdown Analysis overlooked the automatic response of the plant control systems to predicted plant parameters. As a result, the analysis did not identify operator actions or plant modifications that would have been required to prevent the Level 1 Logic from interfering with safe shutdown during a postulated Appendix R Fire event. This was an error created in the original Safe Shutdown Analysis issued in 1986. The same approach was utilized in subsequent revisions, thus carrying the error forward.

C. Contributing Factors

None.

IV. ANALYSIS OF THE EVENT

Based on review of the Fire Area 9 training scenario, BFN determined that rapid depressurization for a postulated Appendix R Fire could result in an adverse affect on the 4kV Shutdown Board's EDG power source and/or sequencing of electrical loads for all Fire Areas except Fire Area 16, the Control Building [NA]. For Fire Area 16 all credited electrical alignments are isolated from the Level 1 logic.

TVA considered the potential for Level 1 logic initiation to affect both the 120 and 20 minute depressurization scenarios. A listing of the SSIs is provided on page 7 of this report. An evaluation of the potential to reach the Level 1 initiation setpoint is summarized below:

120 Minute Depressurization

For all of the SSIs that have high pressure injection system(s) available (High Pressure Coolant Injection (HPCI) [BJ] and/or Reactor Core Isolation Cooling (RCIC) [BN]) the operator is instructed to depressurize within 120 minutes of SSI entry. The capability of the high pressure injection systems and the decay heat removed from the reactor during operation of the high pressure injection system(s), prevent reaching the level 1 setpoint during depressurization and reflood phase of the SSIs. TVA has confirmed that under these conditions, the Level 1 initiation setpoint is not reached on the BFN simulator. Accordingly, safe shutdown would not be adversely affected.

20 Minute Depressurization

Although not credited for Appendix R Safe Shutdown, for all of the SSIs that require depressurization within 20 minutes, it is highly probable that the Condensate System [SD] would have been available for reactor water inventory makeup. The capacity and discharge pressure of the system would result in significant makeup water injection during and after the depressurization and would prevent the Level 1 setpoint from being reached for most of these SSIs that require depressurization within 20 minutes. Browns Ferry simulator results indicate that with only one of the three trains of Condensate System Pumps available, the reactor water level would remain above the Level 1 setpoint where 4 or more Main Steam Safety Relief Valves (MSRVs) are used by the SSI procedure to depressurize or depressurization occurs earlier than the 20 minute SSI requirement. The SSIs that have the potential to result in water level reaching -122 inches are those that either (1) are associated with Unit 3 (due to lower Condensate/Condensate Booster Pump discharge of the Unit 3 pumps) or (2) only require 3 MSRVs to be opened. These eleven SSIs are identified by an asterisk in the listing of SSIs provided on page 8 of this LER.

The eleven SSI scenarios with the potential to result in water level reaching -122 inches during depressurization were further evaluated to determine the impact of the Level 1 logic initiation. Seven of the eleven SSI scenarios would result in a momentary loss of power to the credited RHR pump with the pump automatically restarting. Two of the eleven SSI scenarios have the potential to result in the opening of the breaker that cross ties power from a Unit 3 Emergency Diesel Generator (EDG) to a Unit 1 or 2 4kV Shutdown Board. The opening of the breaker in these two SSI scenarios does not impact the fire affected unit and allows approximately 100 minutes for the operators to repeat some of the SSI steps to re-establish the cross tie prior the required initiation of depressurization. Two other of the eleven SSI scenarios have the potential to affect the credited RHR pump on the fire affected unit. The limiting scenario out of these two scenarios is associated with 0-SSI-2-1 which requires an action to be completed remotely at the 4kV Shutdown Board to re-start the credited RHR pump on the fire affected unit. In these two scenarios, it is highly probable that the Condensate System will provide adequate make-up to flood up the reactor vessel. With reactor inventory restored by the Condensate System, this condition provides the operators with adequate time to re-perform the SSI steps necessary to re-start the credited RHR pump prior to water level reaching the top of active fuel.

The eleven scenarios were also evaluated for the potential of the logic initiation to result in starting unnecessary equipment that could result in the analyzed diesel generator loading being exceeded. Of the eleven scenarios, 0-SSI-2-3 is the only scenario that has the potential to result in the analyzed diesel generator loading capability being exceeded by starting a non-credited RHR pump during recovery from an initial CAS logic initiation. Alarms would be received in Unit 2, if not fire affected, that would alert operations personnel of the RHR pump start and separate alarms not likely to be fire affected would be received in Unit 3 that would alert operations personnel of the diesel generator overload condition. The alarm response procedures direct the operator to take

appropriate actions to secure the RHR pump and reduce diesel generator loading. In addition, procedure 0-SSI-001 precautions warn that spurious operations from fire induced damage may be encountered and frequent observation of the control boards is required to detect and take corrective action on malfunctioning equipment. If prompt actions were not taken by the operator, it is likely that the unnecessary pump would stall and trip from time-overcurrent protective relaying.

The Condensate System Pumps and associated power supplies are located in the Turbine Building [NM] outside the Fire Areas/Zones of concern. The Condensate System controls are located in both the Turbine and Control Building, also outside the Fire Areas/Zones of concern. The power supply protective relay settings provide adequate coordination. Therefore, there is reasonable assurance that a fire in the areas of concern will not disable the Condensate System during a postulated Appendix R in the Fire Areas/Zones of concern.

There are no Primary Containment Isolation System [JE] signals or procedural requirements that prevent reactor pressure vessel injection from the Condensate System Pumps. The SSIs or other plant operating procedures would not instruct the operators to secure the Condensate System Pumps. During a postulated Appendix R Fire event, up to three trains of Condensate Pumps and Condensate Booster Pumps would continue to operate drawing water from the main condenser hotwell which has a normal capacity of 90,000 gallons. When the operators manually depressurize the reactor vessel sufficiently below the Condensate Booster Pump discharge head pressure, Condensate System Pumps will inject into the reactor vessel without any required operator actions.

Therefore, it is reasonable to conclude the Condensate System would have either prevented the Level 1 logic initiation or mitigated the effects of the Level 1 logic initiation such that water level is maintained above the top of active fuel. Accordingly, safe shutdown would not be adversely affected.

V. ASSESSMENT OF SAFETY CONSEQUENCES

TVA concludes that for areas where high pressure injection (HPCI and /or RCIC) is available, the Level 1 setpoint is not reached during the depressurization phase and the RHR LPCI injection phase of the event is not impacted by the Level 1 initiation. For Fire Areas/Zones, which require depressurization within 20 minutes, the Condensate System, which is not credited in the Appendix R Safe Shutdown Analysis, would likely remain available for injection into the reactor without operator action. It is capable of injecting sufficient water to either prevent water level from reaching Level 1 and the resultant logic initiation or mitigate the effects of the potential Level 1 logic initiation such that water level is maintained above the top of active fuel.

In addition, a highly unlikely chain of events would have to occur in order to have a fire of the magnitude that damages enough control and power cables that it impacts the ability to attain safe shutdown. Current BFN design and operating practices aided by the BFN Fire Protection Program contribute to minimizing the likelihood of a postulated Appendix R Fire. With availability of the Condensate System Pumps, the Level 1 initiation logic will not impact the ability to attain safe shutdown conditions during a postulated Appendix R Fire event. Therefore, the failure to address the Level 1 logic in the BFN Appendix R Safe Shutdown Program, and associated SSIs, had no adverse impact on the operators' ability to safely shutdown the units and maintain them in a safe condition. Accordingly, there was no adverse impact on the health and safety of the public.

VI. CORRECTIVE ACTIONS

A. Immediate Corrective Actions

TVA established compensatory actions to inhibit accident logic after SSI entry in the form of operator manual actions. These include:

- Place CS Auto-Initiation inhibit switches in INHIBIT for all Fire Areas except Fire Area 16 (SSI-16) within 20 minutes of entering the SSIs.
- Place the RHR LPCI Auto-Initiation Inhibit Switches in INHIBIT for all Fire Areas except Fire Area 16 (SSI-16) within 20 minutes of entering the SSIs.
- For 4kV Shutdown Board (SDBD)-C in Fire Area 5 (SSI-5), within 20 minutes perform the following,
  - Locally isolate EDG-C output breaker to mitigate closure,
  - Locally isolate Inter-tie breaker between SDBD-A and EDG 3A thru SDBD-3A to mitigate a trip.
- For 4kV SDBD-A perform the following within 20 minutes.
  - Locally isolate EDG-A output breaker to mitigate closure,
  - Locally isolate the inter-tie breaker between SDBD-3A and EDG-A thru SDBD-A to mitigate a trip.

Common accident signal logic is inhibited by inhibiting CS logic. The CS and RHR LPCI inhibit switches are located in the Auxiliary Instrument Room within the Control Building. The operators can quickly access these switches and successfully complete the actions with minimal impact on the existing SSI timelines. Compensatory lights for compensatory action implementation in the Auxiliary Instrument Room have been added to support performance of these actions. Communicating confirmation of completed compensatory actions to the Main Control Room is accomplished in the Control Building hallway just outside the Auxiliary Instrument Room.

The EDG output breakers for SDBDs-A and -C are located on the SDBDs. Currently there are other SSI manual actions on the SDBDs. The operator can quickly access these switches while in the area and successfully complete these actions with minimal impact on the existing SSI timelines. Installed emergency lighting is available in the SDBD areas and communication is accomplished in the current Appendix R Program.

One additional Auxiliary Unit Operator has been added to the minimum shift manning requirements for the plant to support the additional operator manual actions, put in place as compensatory actions.

To increase the level of defense in depth until a permanent solution for this issue can be implemented, compensatory action fire watches have also been established.

B. Corrective Actions to Prevent Recurrence - The corrective actions to prevent recurrence are being managed by BFN's corrective action program.

Because the root cause describes an old design issue that was carried forward, no additional corrective actions are necessary to address the root cause of this event.

VII. ADDITIONAL INFORMATION

A. Failed Components

None.

NARRATIVE *(If more space is required, use additional copies of NRC Form 366A) (17)*

B. **PREVIOUS LERS ON SIMILAR EVENTS**

None.

C. **Additional Information**

Corrective action document for this report is Problem Evaluation Report 177130.

D. **Safety System Functional Failure Consideration:**

This event is a safety system functional failure according to NEI 99-02.

E. **Scram With Complications Consideration:**

This event was not a complicated scram according to NEI 99-02.

VIII. COMMITMENTS

None.

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NARRATIVE *(If more space is required, use additional copies of NRC Form 366A) (17)*

Safe Shutdown Instructions	Unit 1 Depressurization Time	Unit 2 Depressurization Time	Unit 3 Depressurization Time
SSI-1-1: * Unit 1, Reactor Building Fire EL 519 through 565 West of Column Line R4	**20 minutes**	120 minutes	120 minutes
SSI-1-2: Unit 1, Reactor Building Fire EL 519 through 565 East of Column Line R4	**20 minutes**	120 minutes	120 minutes
SSI-1-3: Unit 1, Reactor Building Fire EL 593 North of Column Line R	**20 minutes**	120 minutes	120 minutes
SSI-1-4: Unit 1, Reactor Building Fire EL 593 South of Column Line R and RHR Heat Exchanger Rooms from EL 565 through 593	**20 minutes**	120 minutes	120 minutes
SSI-1-5: Unit 1, Reactor Building Fire EL 621 and 639 North of Column Line R	**20 minutes**	120 minutes	120 minutes
SSI-1-6: Unit 1, Reactor Building Fire EL 639 South of Column Line R	120 minutes	120 minutes	120 minutes
SSI-2-1: * Unit 2, Reactor Building Fire EL 519 through 565 West of Column Line R11	120 minutes	**20 minutes**	120 minutes
SSI-2-2: * Unit 2, Reactor Building Fire EL 519 through 565 East of Column Line R11	120 minutes	**20 minutes**	120 minutes
SSI-2-3: * Unit 2, Reactor Building Fire EL 593 North of Column Line R	120 minutes	**20 minutes**	120 minutes
SSI-2-4: * Unit 2, Reactor Building Fire El 593 South of Column Line R and RHR Heat Exchanger Rooms from EL 565 Through 593	120 minutes	**20 minutes**	120 minutes
SSI-2-5: Unit 2, Reactor Building Fire EL 621 and 639 North of Column Line R	120 minutes	**20 minutes**	120 minutes
SSI-2-6: Unit 2, Reactor Building Fire EL 639 South of Column Line R	120 minutes	**20 minutes**	120 minutes
SSI-3-1: * Unit 3, Reactor Building Fire El 519 thru 565 West of R18. Equipment Hatch Between Col R15 and R17, T and U Line Ad El 593 and 621, EL 639 South of R Line.	120 minutes	120 minutes	**20 minutes**
SSI-3-2: * Unit 3, Reactor Building Fire EL 519 thru 565 East of R18	120 minutes	120 minutes	**20 minutes**
SSI-3-3: * Unit 3, Reactor Building Fire El 593 and RHR Heat Exchanger Rooms	120 minutes	120 minutes	**20 minutes**
SSI-3-4: Unit 3, Reactor Building Fire EL 621 and EL 639 North of R Line	120 minutes	120 minutes	120 minutes
SSI-4: Unit 1, 4kV Electric Board Room 1B	**20 minutes**	120 minutes	120 minutes
SSI-5: Unit 1, 4kV Electric Board Room 1A	**20 minutes**	120 minutes	120 minutes
SSI-6: Unit 1, 480V Shutdown Board Room 1A	**20 minutes**	120 minutes	120 minutes
SSI-7: Unit 1, 480V Shutdown Board 1B Room	120 minutes	120 minutes	120 minutes
SSI-8: Unit 2, 4kV Electric Board Room 2B	120 minutes	120 minutes	120 minutes
SSI-9: * Unit 2, Reactor Building Fire 4kV Electric Board Room 2A	120 minutes	**20 minutes**	120 minutes
SSI-10: Unit 2, 480V Shutdown Board 2A Room	120 minutes	120 minutes	120 minutes
SSI-11: Unit 2, 480V Shutdown Board 2B Room	120 minutes	120 minutes	120 minutes
SSI-12: * Unit 3, 480V Reactor MOV Board Room 3B	120 minutes	120 minutes	**20 minutes**
SSI-13: * Unit 3, 480V RMOV Board Room 3A	120 minutes	120 minutes	**20 minutes**
SSI-14: Unit 3, 480V Shutdown Board 3A Room	120 minutes	120 minutes	120 minutes
SSI-15: nit 3, 480V Shutdown Board 3B Room	120 minutes	120 minutes	120 minutes
SSI-16: Control Building Fire EL 593 through EL 617	**20 minutes**	**20 minutes**	120 minutes
SSI-17: Unit 1, Battery and Battery Board Room	**20 minutes**	120 minutes	120 minutes
SSI-18: Unit 2, Battery and Battery Board Room	120 minutes	**20 minutes**	120 minutes
SSI-19: Unit 3, Battery and Battery Board Room	120 minutes	120 minutes	120 minutes
SSI-20: Unit 1, and 2 Diesel Generator Building	120 minutes	120 minutes	120 minutes
SSI-21: Unit 3, Diesel Generator Building	120 minutes	120 minutes	120 minutes
SSI-22: Unit 3, 4160V Shutdown Board 3EA and 3EB Room	120 minutes	120 minutes	120 minutes
SSI-23: Unit 3, 4160V Shutdown Board 3EC and 3ED Room	120 minutes	120 minutes	120 minutes
SSI-24: Unit 3, 4160V Bus Tie Board Room	120 minutes	120 minutes	120 minutes
SSI-25I: Turbine Building, Cable Tunnel, Intake Pumping Station and Radwaste Building	120 minutes	120 minutes	120 minutes
SSI-25II: Turbine Building, Cable Tunnel, Intake pumping Station and Radwaste Building	120 minutes	120 minutes	120 minutes

*** Denotes SSIs that have a potential to initiate Level 1 accident logic due to vessel level reaching -122"**